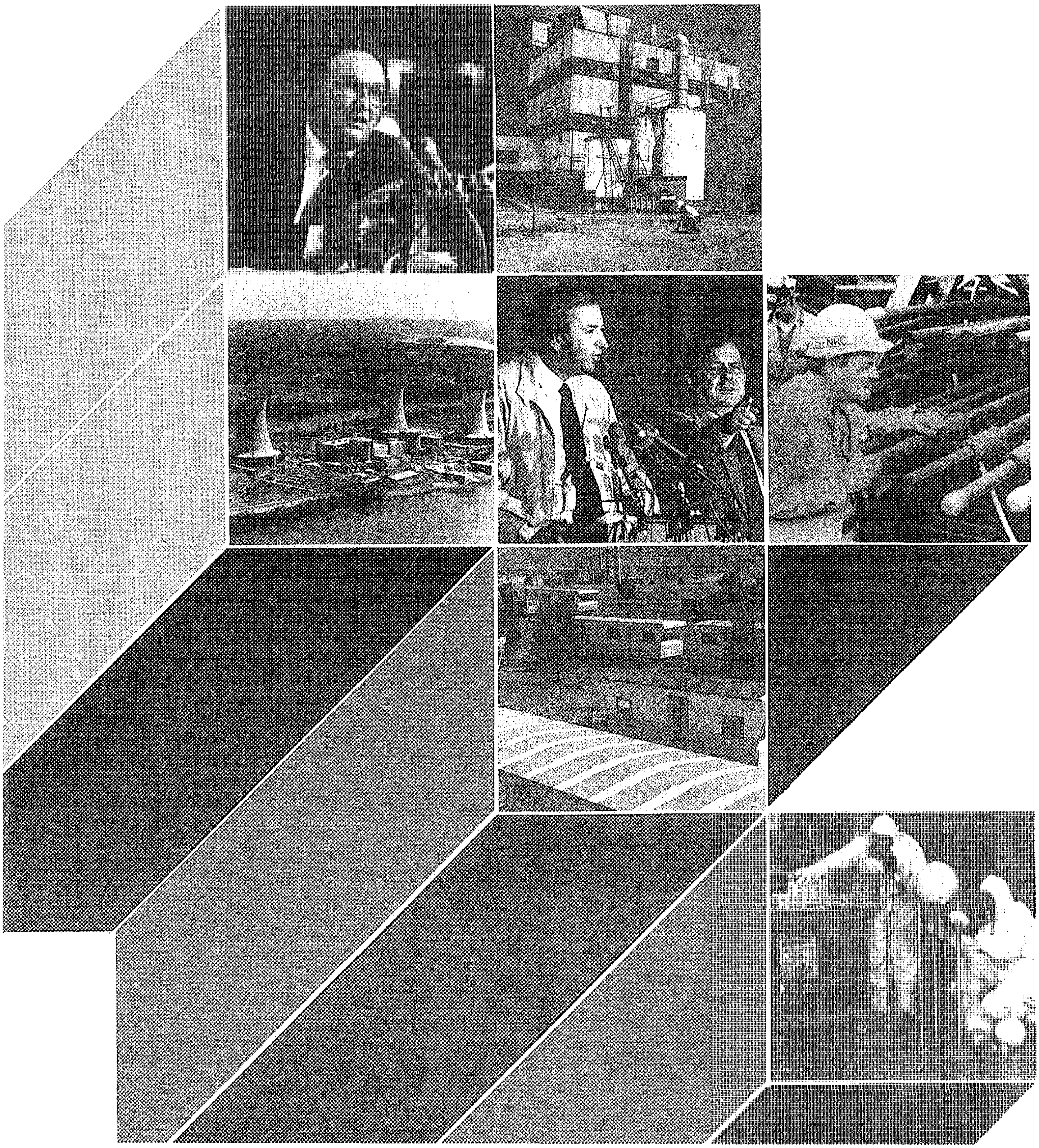
U.S. NUCLEAR
REGULATORY COMMISSION

# 1979 Annual Report



The President
The White House
Washington, D.C. 20500

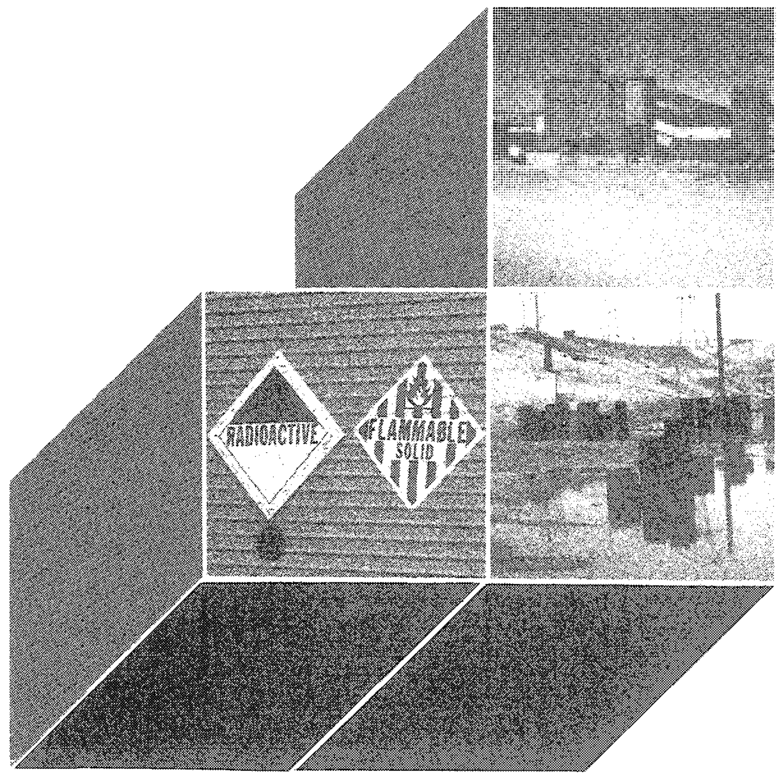
Dear Mr. President:

Enclosed is the fifth Annual Report of the United States Nuclear Regulatory Commission for your transmittal to the Congress, as required by Section 307(c) of the Energy Reorganization Act of 1974. This report covers the major activities of the NRC from October 1, 1978 through September 30, 1979 and briefly describes some additional actions through 1979 into 1980.

Respectfully,

John F. Ahearne
Chairman

# 1979 Annual Report




U.S. NUCLEAR
REGULATORY COMMISSION

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# Statutory Reporting Requirements Addressed

### Energy Reorganization Act of 1974, as Amended

*Section 307(c)* directs the Commission to include in its Annual Report statements and descriptions concerning:

"... the short-range and long-range goals, priorities, and plans of the Commission as they relate to the benefits, costs, and risks of nuclear power." (See Chapter 1 for overall statement. Specific goals concerning nuclear power reactors are also discussed in Chapters 2 and 3; fuel cycle in Chapter 4; safeguards, Chapter 5; wastes, Chapter 6; inspection and enforcement, Chapter 7; emergency response planning, Chapter 8; nuclear nonproliferation, Chapter 9; standards, Chapter 10; and research and risk assessment, Chapter 11.)

"... the Commission's activities and findings in the following areas—

"(1) insuring the safe design of nuclear power plants and other licensed facilities . . ." (For reactors, see Chapters 2, 3, 10 and 11; materials facilities, devices and transportation packages, Chapters 4, 10 and 11; waste facilities, Chapters 6 and 10.)

"(2) investigating abnormal occurrences and defects in nuclear power plants and other licensed facilities . . ." (See Chapters 2, 3, 7 and 8.)

"(3) safeguarding special nuclear materials at all stages of the nuclear fuel cycle . . ." (See Chapters 5, 10 and 11.)

"(4) investigating suspected, attempted, or actual thefts of special nuclear materials in the licensed sector and developing contingency plans for dealing with such incidents . . ." (Chapters 5, 7 and 10.)

"(5) insuring the safe, permanent disposal of high-level radioactive wastes through the licensing of nuclear activities and facilities . . ." (See Chapter 6.)

"(6) protecting the public against the hazards of low-level radioactive emissions from licensed nuclear activities and facilities . . ." (See Chapters 2, 3, 4 and 10.)

*Section 205* requires development of "a long-term plan for projects for the development of new or improved safety systems for nuclear power plants" and an annual updating of the plan. (See Chapter 11.)

*Section 209* requires the Commission to include in each Annual Report a chapter describing the status of NRC's domestic safeguards program. (See Chapter 5.)

*Section 210* directs the Commission to submit "a plan providing for the specification and analysis of unresolved safety issues relating to nuclear reactors," and to include progress reports in the Annual Report thereafter concerning corrective actions. (See Chapter 3.)

### Nuclear Nonproliferation Act of 1978

*Section 602* requires annual reports by the Commission and the Department of Energy to "include views and recommendations regarding the policies and actions of the United States to prevent proliferation which are the statutory responsibility of those agencies . . ." (See Chapter 9.)

### Atomic Energy Act of 1954, as Amended

*Section 170 i* directs the Commission to report annually on indemnity operations implementing the Price-Anderson Act which provides a system to pay public liability claims in the event of a nuclear incident. (See Chapter 3.)

# 1 Overview

John F. Ahearne, who was named NRC Chairman in December 1979, testifies at a congressional hearing.

This is the fifth Annual Report of the U.S. Nuclear Regulatory Commission. It is submitted to the President for transmittal to the Congress as required by Section 307(c) of the Energy Reorganization Act of 1974.

This report highlights major NRC activities in fiscal year 1979 under headings which correspond with the various facets of the agency's statutory responsibility. This introductory chapter presents a brief overview of these activities and provides updating on significant events and actions extending into early 1980.

The accident at Three Mile Island had a profound effect on the public, the utilities, the nuclear industry, and the Nuclear Regulatory Commission. The signs of change in the regulatory area are evident throughout this report as external and internal examinations of NRC have resulted in new policy directives. The chapters dealing with reactor regulation, inspection and enforcement and safety research discuss these changes in some detail. While the primary goals of nuclear regulation—protecting the public health and safety, safeguarding nuclear materials and facilities, and preserving environmental values—remain the same, the means needed to achieve these goals are changing to reflect the lessons of Three Mile Island.

If there is a common thread revealed in the ongoing reassessments, it is a complacency that has served to undercut the many conscientious efforts to assure nuclear safety. Current reappraisals must be seen as opportunities to orient nuclear regulation away from that complacency and toward an outlook befitting a technology that combines remote possibilities of fatal accidents or catastrophes with substantial day-to-day benefits. It is the Commission's philosophy that nuclear regulation must reflect a continuing commitment to come to grips with the realities of nuclear technology and its relationship to those who control it, to those who work with it, to those who live near it, and to the public at large.

As part of this commitment, the Commission is in the process of reappraising its priorities. It has decided to give explicit guidance to the staff for use in preparing plans, budgets, and programs over the next few years. As this Annual Report was being prepared, the Commission—for the first time—was developing a Policy, Planning, and Program Guidance document. The document includes the Commission's direction as to which regulatory areas need greater emphasis in planning for future agency activities such as:

- To define more clearly the level of protection of the public health and safety that the Commission believes is adequate based on statutes, public input, and NRC's subjective and quantitative evaluations.
- To increase efforts to describe to the public the risks of nuclear activities and the uncertainties in the judgments of risk.
- To regulate nuclear activities in a manner to achieve and maintain adequate protection of public health and safety. Licensees who cannot do this will not be permitted to operate.
- To give priority in reactor regulation, in terms of resources and schedules, to those activities that are expected to have the greatest effect on reduction of risks to the public health and safety. First priority will be assigned to operating facilities. Priorities of NRC activities involving those resources not engaged in assuring adequate levels of protection for operating facilities will be assigned according to risk reduction potentials.
- To organize and plan a waste management program to achieve in a timely fashion the ultimate objective set forth in the President's Policy Statement of February 12, 1980 on waste management. The NRC waste management program is critical to the success of this urgent national task.

NRC Commissioners testify at one of numerous Congressional hearings conducted on the TMI accident. From left: Commissioners John F. Ahearne (named Chairman in December) and Richard T. Kennedy; Chairman Joseph M. Hendrie; and Commissioners Victor Gilinsky and Peter A. Bradford.

- To emphasize prompt and vigorous enforcement in dealing with licensees who are unable or unwilling to comply with NRC requirements.
- To require, in any consideration of regulatory costs to licensees and their customers, that cost factors be set forth explicitly and applied with public health and safety being the paramount consideration.
- To consider, in determining the adequacy of public protection, the health and safety implications of not operating a facility as well as the potential radiological or other hazards of its operation.
- To increase emphasis on minimizing the consequences of possible accidents, theft or diversion of nuclear materials, and sabotage or other illegal acts.
- To license or permit continued operation of a nuclear facility only when the NRC is confident that, after termination of the license, there will be adequate protection of the public health and safety from potential hazards of the decommissioned facility itself and from wastes associated with it.
- To continue a research program whose objectives are (1) to assist in determining adequate levels of protection of the public health and safety and (2) to explore ways to achieve improved levels of protection. The research program should not include items that should be supported exclusively by the private sector. The research program must be focused on identifiable needs, and its relevance to the agency's regulatory mission must be the paramount basis for the program.

These policy and planning guidance statements form the basis for more detailed policy and budget guidance on each of the important NRC programs.

## HIGHLIGHTS AND UPDATES

### Accident at Three Mile Island

The accident that occurred on March 28, 1979 at Three Mile Island (TMI) Unit 2 was a traumatic event for the American public—especially for the public living near the facility—as well as for the licensee and other utilities; nuclear plant designers, manufacturers and suppliers; local, State, and Federal authorities responding to the emergency; and the Nuclear Regulatory Commission. The extent of the accident's impact on all of these and on the future of commercial nuclear activity may not be assessable for a long time, but it is certain that it is and will be a profound and lasting one. As serious as the event was, major investigations agreed that releases of radioactive material from the facility were low throughout the course of the accident.

Chapter 2 of this annual report, which is devoted entirely to the TMI accident, includes a narrative of the events of March 28-April 1; actions taken and investigations made by NRC in the aftermath of the accident up to the end of 1979, with conclusions and recommendations; and a full account of the findings and recommendations of the President's Commission on the accident and the NRC's response to each of them.

The President's Commission on the TMI accident was established on April 11, 1979. President Carter charged the Commission with investigating the accident and reporting to him within six months with recommendations based on its findings.

The NRC also chartered its own inquiry into the accident, under independent directorship, the results of which were published at the end of January 1980. In general, the conclusions and recommendations of this NRC Special Inquiry Group were consistent with, but more detailed than, those of the President's Commission. The Special Inquiry Group's report was still under review by the Commission in early 1980.

## NRC Organization and Management

The reports of the President's Commission on the TMI accident, NRC's Special Inquiry Group, and the recent five-year evaluation of the NRC by the General Accounting Office ("The Nuclear Regulatory Commission: More Aggressive Leadership Needed") stressed the need to improve the overall managerial functions of the Commission as a means of improving reactor safety. The Commission did not agree with the recommendation that the NRC be made an Executive Branch agency headed by a single administrator.

In addition to the formulation of explicit policy, planning, and program guidance mentioned earlier, the Commission has taken or is taking the following steps to provide more effective agency management:

- The Commission continues to pursue the consolidation of all NRC offices at a single location as a means of increasing effectiveness, as recommended by Congressional committees and various investigatory bodies.
- The Commission is moving to correct the significant organization and management weaknesses that were revealed by the TMI accident and subsequent investigations. These actions include clarifying the role of the Executive Director for Operations, making the Chairman solely responsible for emergency response, giving increased attention to human factors in nuclear regulation, developing mechanisms to assure more effective use of advice offered by the Advisory Committee on Reactor Safeguards, providing a staff dedicated solely to the evaluation of operating experience, and providing for a more effective role of the public in reactor licensing.

**An audience of NRC staff members and the public listens intently during a briefing at Commission offices concerning developments at the Three Mile Island site seven days after the beginning of the accident.**

- The Commission is considering appropriate delegations of authority to the NRC staff that would permit increased concentration by the Commissioners on matters of overriding significance.

As this report was in final preparation, the President sent to the Congress an NRC reorganization plan designed to improve agency management by, among other actions, strengthening the role of the Chairman.

## Reactor Safety

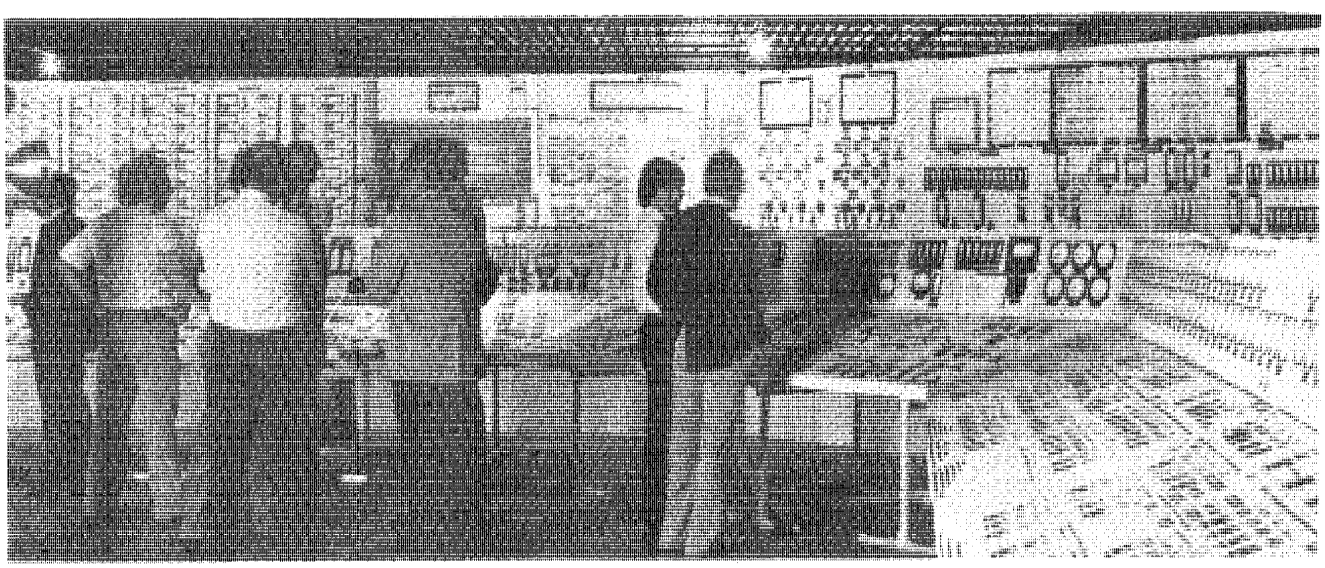
In July 1979, an NRC task force—brought together to develop a systematic NRC response to the several inquiries and investigations of the TMI accident—recommended a number of short-term actions to improve power reactor safety (see Chapter 2, "TMI-2 Lessons Learned Task Force"). New requirements were issued to all operating reactor licensees with the objective of completing the changes by January 1, 1980. Some licensees had difficulty meeting the deadline because of delays in obtaining necessary equipment. Thus, while most licensees had made significant progress by the end of 1979, the NRC took further action by issuing orders making continued operation of the reactor(s) in question contingent upon all changes being implemented by January 31, 1980. Extensions of the deadline were permitted only when a licensee could show that the needed equipment could not be obtained within the time frame or that a reasonable delay would alleviate severe power supply problems. In no case, however, will plants be allowed to operate beyond June 1, 1980 without completion of the changes.

By the end of 1979, the findings of the various investigations and studies of the TMI accident were used by the staff to draft a proposed program of work, "NRC Action Plans Developed as a Result of the TMI-2 Accident" (NUREG-0660). This draft "TMI Action Plan" contains schedules and resource requirements for more than 100 tasks to provide a higher level of protection of the public health and safety. Although many of the proposed tasks are to be completed in 1980, a significant number are multi-year projects that may extend into the mid-1980's.

No operating license was issued for a nuclear power plant during fiscal year 1979, and the need to devote licensing staff resources to TMI-related issues applicable to operating reactors, and subsequently to apply the lessons learned to new plants not yet in operation, resulted in a licensing hiatus that extended into early 1980. Further, because the TMI accident raised significant questions concerning the effectiveness of NRC regulations and practices in assuring adequate protection to the public, the Commissioners announced in October that new construction permits, limited work authorizations, or operating licenses would be issued only after careful review by the Commission itself.*

Certain NRC actions also are intended to improve the technical and managerial competence of reactor licensees and the quality of the human element in achieving the safe operation of nuclear reactors. Essentially, these actions are concerned with the interaction between man and machine—the "human

* On February 28, 1980, the Commissioners approved issuance of a license to the Tennessee Valley Authority authorizing the loading of fuel in its Sequoyah Nuclear Power Plant Unit 1, and performance of low-power testing under specified conditions. Several other similar actions were under consideration.



**This mock-up of the TMI-2 control room was constructed of precisely measured photographic enlargements of the real control room by a behavioral science research firm in Arlington, Va. The model was used in probing aspects of "human factors engineering" in the investigation by the NRC's independent Special Inquiry Group.**

factors" whose critical importance was revealed in the TMI accident. Proposed projects range from establishing new requirements for staffing and manning control rooms and for training and qualification of reactor operators and their supervisors to developing and procuring training and engineering simulators. In cooperation with the nuclear industry, the staff also proposed to study and identify means for improving control room design.

## Emergency Preparedness

The accident demonstrated that emergency preparedness both on-site and off-site should be considered comparable in importance to other elements of the "defense-in-depth" approach to nuclear safety, and that substantial work must be done in emergency preparedness, particularly with respect to off-site preparedness to deal with the aftermath of an accident.

Soon after the accident, Chairman Joseph Hendrie wrote to the Governors of applicable States urging the development and testing of emergency plans around the sites of nuclear power reactors in operation or under construction. Most of the State and local authorities have begun to move ahead with development or refinement of their plans.

In December 1979, the Commission proposed new rules that would require upgraded emergency plans in areas near nuclear power plants and concurrence by the NRC and the Federal Emergency Management Agency (FEMA) in State and local plans as a condition of continued operation of existing plants and issuance of new operating licenses. Both NRC and FEMA are working with State and local authorities to test and evaluate off-site emergency plans.

From April through July of 1979, a continuous watch was established in each region and at the NRC Incident Response Center in Bethesda, Md., to provide for immediate response to any incident or accident. In August, a communications system directly tied into the Operations Center (where 24-hour coverage is maintained) became operational, thus relieving the 24-hour duty offices in the Regions.

Other NRC priorities for improving emergency capabilities include:

- Developing NRC, licensee, and State/Local emergency procedures for all appropriate facilities.
- Instituting for emergency planning purposes a zone concept that would establish two concentric zones around each nuclear power plant—the first for plume exposure pathway and the second for the ingestive exposure pathway (milk and agricultural products).
- Requiring additional instrumentation that would: (1) provide more precise information on the status of key equipment during an accident, and (2) expand the means for measuring radioactive releases around major nuclear facilities.
- Upgrading NRC, licensee, and local communications facilities for prompt NRC response to emergencies.


**Several hundred State and local employees assigned to radiological emergency response teams have received training in the Radiological Emergency Response Operations course conducted at DOE's Nevada Test Site.**

## Analysis of Operating Data

The TMI accident revealed a need for NRC to put greater effort into systematically analyzing operational data from nuclear power plants to detect trends that would better enable all concerned to identify safety problems and take action before they cause accidents.

In July 1979, the NRC established an Office for Analysis and Evaluation of Operational Data to conduct systematic and rigorous assessments of licensee operating experience. The new office will analyze and evaluate operational data associated with all NRC licensed activities; develop formal NRC guidance on the collection, evaluation, and feedback of operational data; and take cognizance of the similar efforts of NRC program offices, industry organizations, and foreign countries. NRC reactor licensees will be required to conduct analyses of their operating experience and disseminate the results.

Within the industry, two new organizations, the Nuclear Safety Analysis Center and the Institute of Nuclear Power Operations have been created to systematically review plant operating experience.

These organizations will develop and implement programs designed to ensure a high quality of operation in nuclear power plants. The NRC's exchange arrangements with other countries and international nuclear organizations provide it with data on operating experience of overseas reactors, many of which are of U.S. design.

## Inspection and Enforcement

For some time, NRC had realized that greater NRC presence is needed at major licensed facilities. The resident inspector program provides this increased presence at nuclear power plants and other selected facilities. The resident inspectors conduct frequent direct observation of licensee activities, thereby relying less on the records and reports which were the principal sources of information in the past.

The resident program, approved by the Commission in 1977, has been expanded to entail the placing of at least two resident inspectors at each operating nuclear facility site in fiscal year 1980. By December 31, 1979, a total of 60 inspectors had been deployed at 45 nuclear power stations and three fuel facilities. By June 1980, each site with a reactor in operation or about to go into operation (as well as a substantial number of reactor construction sites) will have at least one resident inspector.

NRC is currently examining its enforcement policy and practices. The Commission is awaiting Congressional action on a proposal to increase the civil penalty authority from a maximum of $5,000 to $100,000 per violation; the higher authority is more in line with that available to other agencies with public health and safety missions. The Commission is also preparing a restatement of enforcement policy that would implement the new authority and provide clear guidance to the staff.

**An NRC reactor inspector (right) at work.**

## Radiation Protection

Significant steps are being taken in improving our understanding of the potential health effects from exposure to low-level radiation. Holding radiation exposures as low as reasonably achievable under normal conditions is a fundamental objective of NRC's radiation protection activities. NRC is taking the following steps to achieve this goal:

- Participating in the President's Radiation Policy Council to improve the coordination of Federal radiation protection activities.
- Working closely with the Environmental Protection Agency and other Federal agencies to develop improved standards for controlling occupational exposures. Cooperation with EPA includes joint hearings on occupational exposure standards, coordination of research programs, and a study on the health effects of low-level radiation.
- Improving NRC radiation protection criteria for the adequacy of licensee health physics programs, and conducting in-depth radiological safety evaluations at every operating reactor.
- Working with the National Institute of Occupational Safety and Health to establish a TMI worker registry that could be used for follow-up health studies.

## Research

As a direct result of the TMI accident, NRC's research emphasis is being shifted from large-break, loss-of-coolant accidents to small-break LOCA's and related transients. Research effort is increasing in the areas of verification of computer codes, fuel behavior under accident conditions, monitoring of radioactive releases, emergency response planning, and risk assessment. Research has also been initiated in areas not adequately considered in the past—fuel damage and its effects, core melt, and containment integrity. Also, a new study has begun into simulators, control rooms, and human factors in nuclear power plant operations.

The accident has also underscored the need to apply the fault-tree/event-tree techniques used in the Reactor Safety Study (WASH-1400) to each operating plant, in an effort to identify the likelihood and consequences of the accident sequences which are the largest contributors to risk. An "Interim Reliability Evaluation Program" to review all operating reactors over the next few years is already under way.

In areas unrelated to TMI, NRC research on seismic and structural engineering and code verification was increased during the reporting period following the shutdown of five reactors in early March 1979 because of errors in the seismic analyses of the plants. Research into the development and application of risk assessment techniques has allowed the Commission to identify and concentrate on the resolution of those generic issues that involve the highest risk to the public health and safety. Research has also increased substantially in waste management to provide technical data and methods needed to implement regulation being developed in that area.

In general, future research planning at NRC will be based on (1) assisting in the determination of adequate levels of public health and safety protection and (2) exploring ways to achieve improved protection levels. NRC research must be balanced among confirmation of existing practices, exploration of areas where new concerns may exist or where existing regulatory approaches may be inadequate, and examination of concepts for improving safety. The program must also be capable of some effort in areas with potential long-range benefits as well as work addressing more immediate goals and needs arising from current NRC activities.

## Waste Management

The NRC waste management program is critical to the successful resolution of the urgent national nuclear waste problem. NRC is organizing and planning its program to be consistent with the requirements set forth in the President's Policy Statement of February 12, 1980, on nuclear waste management.

The NRC has intensified its efforts over the last year in preparing or modifying regulations covering all types of nuclear wastes including high- and low-level wastes and uranium mill tailings.

**High-Level Waste.** The accumulation of spent fuel continues to lead utilities to seek expanded storage capacity of pools at reactor sites and occasionally to ship irradiated fuel from site to site to utilize unused capacity.

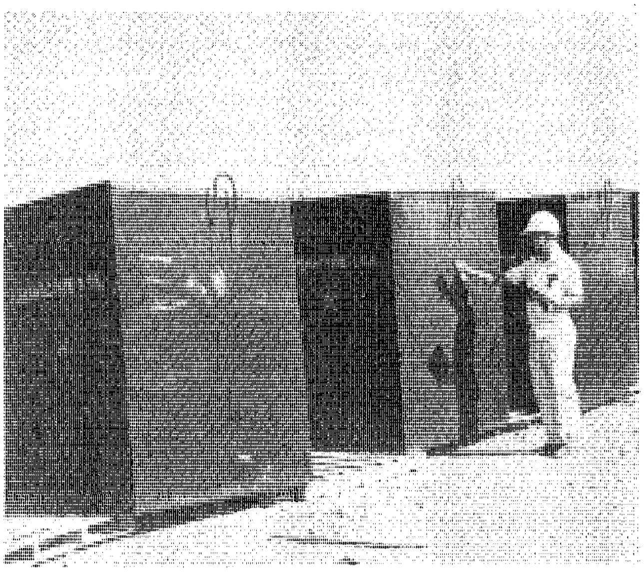
It is estimated that these pools will be filled by 1983. Although sites away from reactors will be used to store excess spent fuel, the President, NRC and DOE consider both these sites and the reactor pools only interim measures before the ultimate solution is provided by permanent geologic disposal. NRC is developing a general regulation (10 CFR Part 60) on the disposal of high-level radioactive wastes in geologic repositories which is expected to be published in two parts: the procedural requirements and the technical requirements. The procedural portion was published as a proposed rule for public comment in December 1979, and the technical part will be published in 1980 as an advance notice of proposed rulemaking. Operation of the first repository could begin in the mid-1990's.

**Low-Level Waste.** Major emphasis is being placed on developing a regulation for the disposal of low-level waste in land facilities.

The importance of assuring adequate regulation of the disposal of low-level radioactive wastes was dramatized during the past year by the temporary closure or restriction on operations of each of the country's three existing commercial waste disposal facilities. The States of Nevada, Washington, and South Carolina, which regulate these facilities under agreements with the NRC, closed or restricted operation of the three sites because of deficiencies in the packaging, transport, or disposal of the wastes being received. Extended closures could have resulted in curtailment of nuclear medicine services. All of these sites have subsequently been reopened.

**Governors of the three States with low-level radioactive waste facilities met with the Commission in November to discuss deficiences in the packaging, transport and disposal of the wastes being received. Shown, from left, are Governors Dixy Lee Ray, Washington, and Richard W. Riley, South Carolina; NRC Chairman Joseph M. Hendrie (standing) and Governor Robert List, Nevada.**

**Uranium Mill Tailings.** The NRC is proposing to require the disposal of uranium mill tailings underground instead of continuing to permit them to be piled on the surface with virtually no controls. A draft generic environmental impact statement on uranium milling, emphasizing tailings management, was published for public comment in April 1979, followed by a proposed regulation in August. The regulation is expected to be in place in late 1980. Meanwhile, all current licenses under NRC jurisdiction have been upgraded appropriately, and NRC is working with the Agreement States to upgrade their programs accordingly. The Uranium Mill Tailings Radiation Control Act of 1978 requires that, by late 1981, the minimum standards in Agreement States be equivalent to those of NRC regulations. The failure of the tailings dam at Church Rock, New Mexico, in July 1979 illustrates the need for more stringent regulatory control over tailings.



NRC inspector examines low-level radioactive waste containers before disposal at Beatty, Nevada burial facility. Inspections were increased at receiving areas at behest of the Governors of Nevada, South Carolina and Washington.

## Transportation

Substantial public concern over the transport of nuclear products (especially spent fuel) has prompted some Federal, State and local authorities to enact or consider restrictions affecting the highway transportation of radioactive materials. In April 1978, DOT published an opinion in the *Federal Register* concluding that it had the authority under the Hazardous Materials Transportation Act (HMTA) to preempt State and local routing requirements that are inconsistent with DOT regulations.

The NRC has amended its rules to impose DOT regulations on NRC's licensees. This action is expected to enhance the NRC's inspection and enforcement effort.

In 1979, the Commission also took two other actions which contribute to the regulatory framework for transportation of radioactive waste: it issued guidance on methods of packaging and it issued an interim rule requiring safeguards measures for spent fuel shipments and NRC approval of routes for shipping spent fuel. A supplemental draft generic environmental impact statement on transportation should be published in 1980.

## Domestic Safeguards

Under the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974, the NRC is responsible for the regulation of safeguards provided by certain of its licensees. NRC safeguards regulatory programs share the common goal of assuring that licensed activities do not pose undue risk to the public health and safety and are not inimical to the common defense and security. The NRC safeguards objective is to require the implementation of measures designed to prevent, deter, detect and respond to (1) the unauthorized possession, theft, diversion or use of special nuclear material (SNM) and (2) the sabotage of nuclear facilities and transportation activities.

NRC currently exercises safeguards regulatory control over 19 fuel cycle facilities that are authorized to possess formula quantities of highly enriched uranium or plutonium, transportation activities involving spent fuel or formula quantities of highly enriched uranium or plutonium (about 20 shipments per month), 70 power reactors and 71 non-power reactors. NRC also has safeguards responsibilities for other facilities which possess significant quantities of low enriched uranium as well as numerous small facilities that possess and ship SNM.

In November 1979, a physical security upgrade rule was published in final form. It became effective in March 1980. The rule indicates performance standards to be met by licensees in protecting formula quantities of SNM and presents specific statements about the kinds of threats, from insiders and outsiders, that their safeguards should be able to withstand. Certain non-power reactors are temporarily excepted from the new upgrade rule, but will be subject to interim requirements. All non-power reactors are now subject to special protection requirements.

As a result of an excessive inventory difference in August 1979, a fuel cycle facility operated by Nuclear Fuel Services at Erwin, Tenn., was ordered to be shut down for investigation and reinventory. While the inventory difference was not fully reconciled, the facility was allowed to resume operations in January 1980 subject to the implementation of additional physical protection and material control and accounting measures, with a study of potential alternative measures for improving accountability to be completed by DOE within one year. In addition, the license will be required to conduct a special reinventory in the event that an inventory difference is found to exceed 1.0 percent of throughput. Further, a plant shutdown would be required if the special reinventory does not reduce the inventory difference to below 1.5 percent of throughput. These new limits, which are less restrictive than the former limits, are considered to be representative of the level reasonably achievable for the process.

In 1979, NRC consolidated responsibility for integrating and coordinating the overall NRC safeguards program in the Office of Nuclear Material Safety and Safeguards (NMSS). NMSS coordinates those safeguards activities pertinent to reactors with the Office of Nuclear Reactor Regulation.

A detailed report on the status of domestic safeguards during fiscal year 1978 was sent to Congress on February 1, 1979, as required by Public Law 95-601, amending Section 209 of the Energy

Reorganization Act. The follow-up report for fiscal year 1979 is presented in Chapter 5 of this Annual Report.

## Exports and International Safeguards

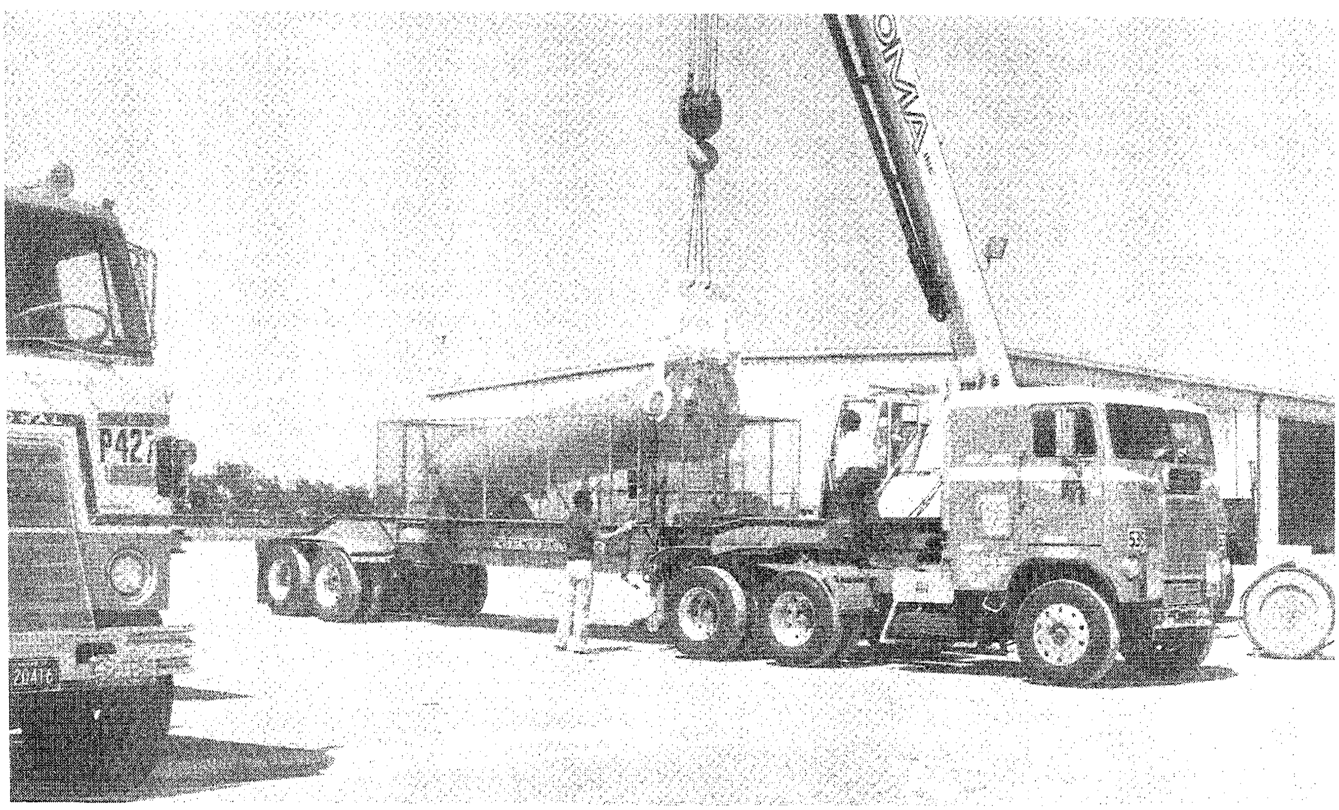
Under provisions of the Atomic Energy Act of 1954 and the Nuclear Non-Proliferation Act of 1978, the NRC ensures, through licensing, that effective U.S. controls are applied to the export and import of nuclear materials, equipment and facilities. It is also NRC policy to support the reliability of the U.S. in meeting its supply commitments to nations which adhere to effective non-proliferation policies by implementing procedures that facilitate the timely processing of export licenses. In addition to exercising its direct licensing authority, the NRC consults with the Departments of Commerce and Energy on nuclear export-related functions under their authority.

During fiscal year 1979, the NRC issued 678 nuclear export licenses and amendments, of which 154 were major licenses; provided views to DOE on 13 requests for approval of retransfers of U.S.-origin spent fuel to other countries for reprocessing; and consulted with DOE on several cases involving the export of technology associated with the production of special nuclear material outside the United States.

The Commission recently confronted the question of its authority and responsibility to consider the effects of reactor exports on the health, safety and environment in recipient countries. The question arose in the context of the Commission's consideration of a controversial license application to export a reactor to the Philippines. On January 29, 1980, the Commission decided to limit its review in the Philippines and other reactor export cases to health, safety and environmental factors affecting the global commons or the territory of the United States, and the relationship of these effects to the common defense and security of the U.S. Consideration of local impacts, including effects on any U.S. citizens located there, would continue to be the sole responsibility of foreign recipient governments.

## THE NRC COMMITMENT

Despite the problems and issues to be resolved, nuclear electric power is still acknowledged to be an important element in the nation's energy strategy for some time to come. The President has stated that "nuclear power has a future in the United States—it is an option that we must keep open. I call on the utilities and their suppliers, the NRC, the Executive Departments and agencies, and the State and local governments to assure that the future is a safe one."

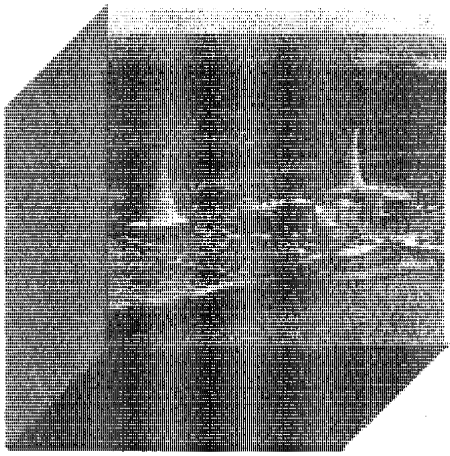


Spent fuel cask designed for highway transport is shown being placed on a protected truck flatbed. The cask, 18 feet long and four feet in diameter, weighs 50,000 pounds when fully loaded with spent fuel assemblies.

It is clear that the next few years will see many significant changes in the NRC's efforts to assure that the public health and safety is adequately protected from the potential risks of nuclear power plants. The changes will be widespread—in the NRC organization and its relations with Federal, State, and local authorities; the regulations; the design of nuclear plants; the utilities' mode of operation; and the regulatory process itself.

Neither the President's Commission nor the NRC's own Special Inquiry Group was able to define an explicit standard of nuclear safety—or, more simply, to answer the question, "How safe is safe enough?" It is quite clear that society itself must ultimately provide the answer as to what is acceptable. All that has happened during the past year has served to confirm the proposition that the part nuclear energy will play in the U.S. energy strategy is directly and inevitably linked to the public's perception of nuclear safety.

As the agency responsible for nuclear regulation, the NRC must play the fundamental role leading to the proper determination of what is an adequate level of protection. The NRC must bring its management and technical expertise to bear in assuring that the regulated industry achieves and maintains that protection. The NRC is fully committed to meeting this challenge.



# 2
# Accident at Three Mile Island

**The accident at Three Mile Island in March was the focus of nearly all NRC activity in 1979.**

Despite the fact that no one was killed and no physical injuries were sustained among the general public because of it, the accident at the Three Mile Island Nuclear (TMI) Station Unit 2 is unquestionably the most serious in the history of commercial nuclear power. It is also the most intensively studied and extensively reported incident in that history. This chapter can only attempt to cover the major investigative efforts devoted to the accident, only those whose results were available before the end of 1979 (the NRC's own Special Inquiry Group report was pending, as were the results of several Congressional studies), and only the most salient findings and recommendations or actions issuing from them. Other chapters of this report cover many aspects and effects of the TMI accident in connection with the particular NRC activities under discussion. These references are cited in the Index under "Three Mile Island accident."

The full reports of the various NRC investigations and other documents cited in this chapter are available from the GPO Sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and from the National Technical Information Service, Springfield, Va., 22161. The titles and catalogue numbers are listed in the box below. The report of the President's Commission on the Accident at Three Mile Island, which is discussed at length, is available from the U.S. Government Printing Office.

## WHAT HAPPENED

Located in Dauphin County, Pa., about 10 miles southeast of Harrisburg on an island in the Susquehanna River, the Three Mile Island Nuclear Station (TMI) consists of two pressurized water reactors and associated equipment, each one with two large steam generators and each employing two 370-ft. cooling towers—part of the system which condenses the steam after it has passed through the turbines to generate electricity. The utility licensed to operate the facility is the Metropolitan Edison Company, a subsidiary of General Public Utilities, Inc., of New Jersey. Unit 1 at TMI was licensed for operation in 1974, at a net capacity of 819 MWe; Unit 2 was licensed in February 1978 and went into commercial operation in December 1978. Each unit has its own reactor containment building, control room and auxiliary building. Each containment building houses a reactor, a pressurizer, and two steam generators; the turbine and electric generator are outside the containment.

**NRC REPORTS ON TMI CITED IN THIS CHAPTER**

| | |
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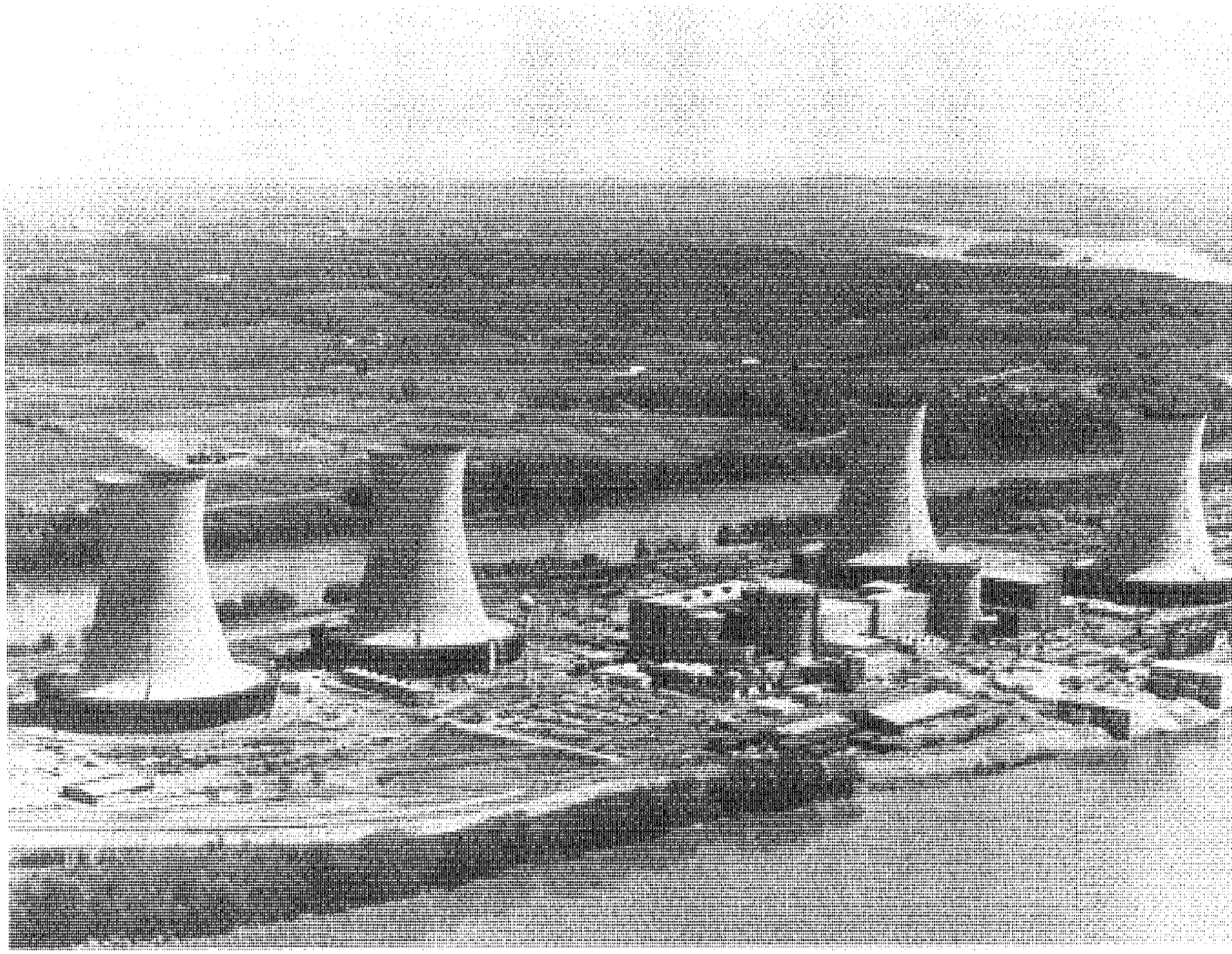
| NUREG-0558: | "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station" |
| NUREG-0560: | "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company" |
| NUREG-0578: | "TMI-2 Lessons Learned Task Force: Status Report and Short-Term Recommendations" |
| NUREG-0585: | "TMI-2 Lessons Learned Task Force: Final Recommendations" |
| NUREG-0596: | "The Non-Radiological Consequences to the Aquatic Biota and Fisheries of the Susquehanna River from the 1979 Accident at Three Mile Island Nuclear Station" |
| NUREG-0600: | "Investigation into the March 28, 1979 Three Mile Island Accident by [NRC] Office of Inspection and Enforcement" |



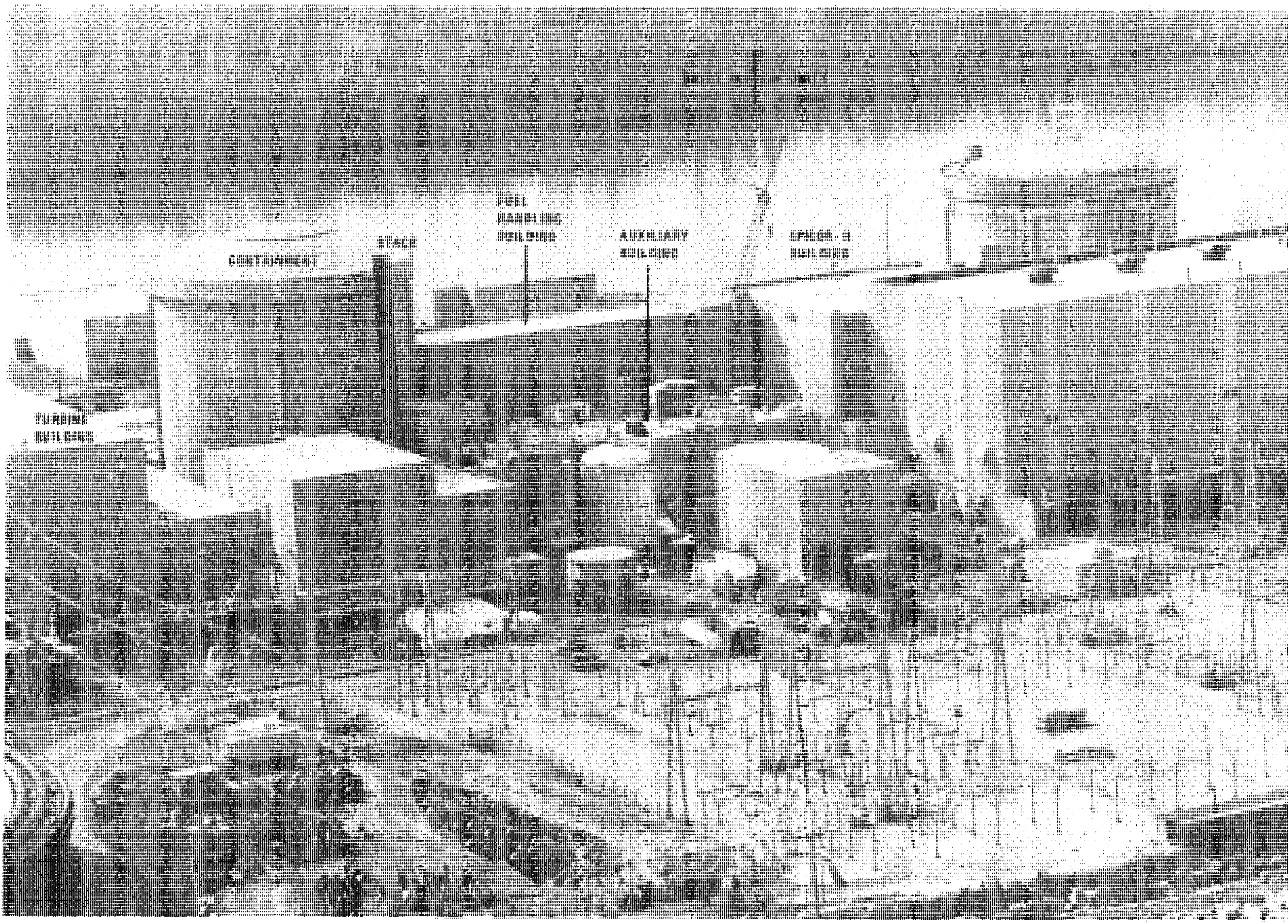
The Three Mile Island Nuclear Station. The four large towers cool the steam generator water used in the production of electricity by the two reactor units at the station, located in the cylindrical domed structure shown at right center in the photo. The Unit 2 reactor, scene of the accident, is housed in the cylindrical containment building farthest to the right.

## Wednesday—March 28

At about half a minute past 4:00 a.m., on Wednesday, March 28, 1979, a "condensate" pump and the main "feedwater" pumps connected with one of the Unit 2 steam generators shut down, causing an almost simultaneous and automatic shutdown of the Unit 2 turbine (Unit 1 was shut down at the time for refueling.) The initiating cause of the shutdown is not definitely known but may have been an alteration in the pressure in the feedwater system brought about by a maintenance procedure taking place at the time. An unexpected pump shutdown is not unusual or, in itself, serious. With the feedwater flow stopped, the steam generators stopped removing heat from the primary system, i.e., from the closed system of pressurized water which passes through the reactor, carries heat to the secondary system, and returns to the reactor. The buildup of heat in the primary system caused the pressure of the water to rise and a "pressurizer relief valve" to open. The reactor automatically shut down in response to the increase in primary coolant pressure. This reactor "scram" took place eight seconds after the condensate pump shut down on the secondary side of the system. Instantly the output of heat from nuclear fission in the reactor core was stopped, but a substantial amount of "decay heat" continued. The production of decay heat, like the momentum of a large ship at sea, cannot be ended by turning off the power source, and it is essential that sufficient primary coolant and pressure be maintained even after the reactor has shut down.

Through the first seconds of the accident, the performance of the equipment went according to design and the sequence of responses to the unexpected interruption of heat transfer from primary to secondary systems was "normal." After the reactor scrammed and the relief valve lifted, the primary coolant pressure dropped back to the point where the pressurizer relief valve was supposed to close, restoring a closed, fully pressurized primary system with coolant flowing through the reactor core and removing its decay heat (about 7 percent of its normal operating heat production). The relief valve did not close. At this same time, several pumps came on automatically on the secondary side to restore feedwater flow and remove heat through the steam generators. This action was thwarted by closed valves, a condition which was not corrected until eight minutes into the accident.

Because the pressurizer relief valve was stuck open, the pressure in the primary system did not level off at the proper point but continued to decrease. As the pressure of the coolant goes down so does its boiling point, and the danger arises that it may begin to turn into steam. Since steam cannot carry off decay heat effectively, the primary system could heat up to dangerous levels. When the pressure had decreased to about 75 percent of normal, an emergency core cooling system (ECCS) automatically came on, injecting cold water under high pressure into the reactor. Believing that the pressurizer relief valve was closed and seeing the level of coolant in the pressurizer rise with the injection of ECCS water into the reactor, the operators in the control room feared that the pressurizer would fill up with coolant and the system would lose the pressurizing bubble of steam that is normally maintained at the top of the pressurizer. Consequently, they shut off one ECCS pump and throttled back the ECCS flow from the other pump into the reactor. Ordinarily the level of coolant in the pressurizer is an accurate indicator of the volume of coolant in the entire primary system, so the operators were confident that the system was full, the reactor core was covered, and the heavy injection of ECCS coolant was unnecessary and was, in fact, making the system too full. As the four licensee personnel then present in the control room later testified, they were not aware that the level of coolant in the pressurizer is not necessarily an index to the amount and level of coolant throughout the system. As it happened, the drop in pressure following the failure of the relief valve to close and the failure of the auxiliary feedwater




**The major buildings making up the Three Mile Island Nuclear Station Unit 2 are shown, including the Epicor-II building at right which houses the system used to decontaminate the radioactive water held in the auxiliary building tanks. The containment building at the left houses the Unit 2 reactor, pressurizer and steam generators. The Unit 1 building is at the far right.**

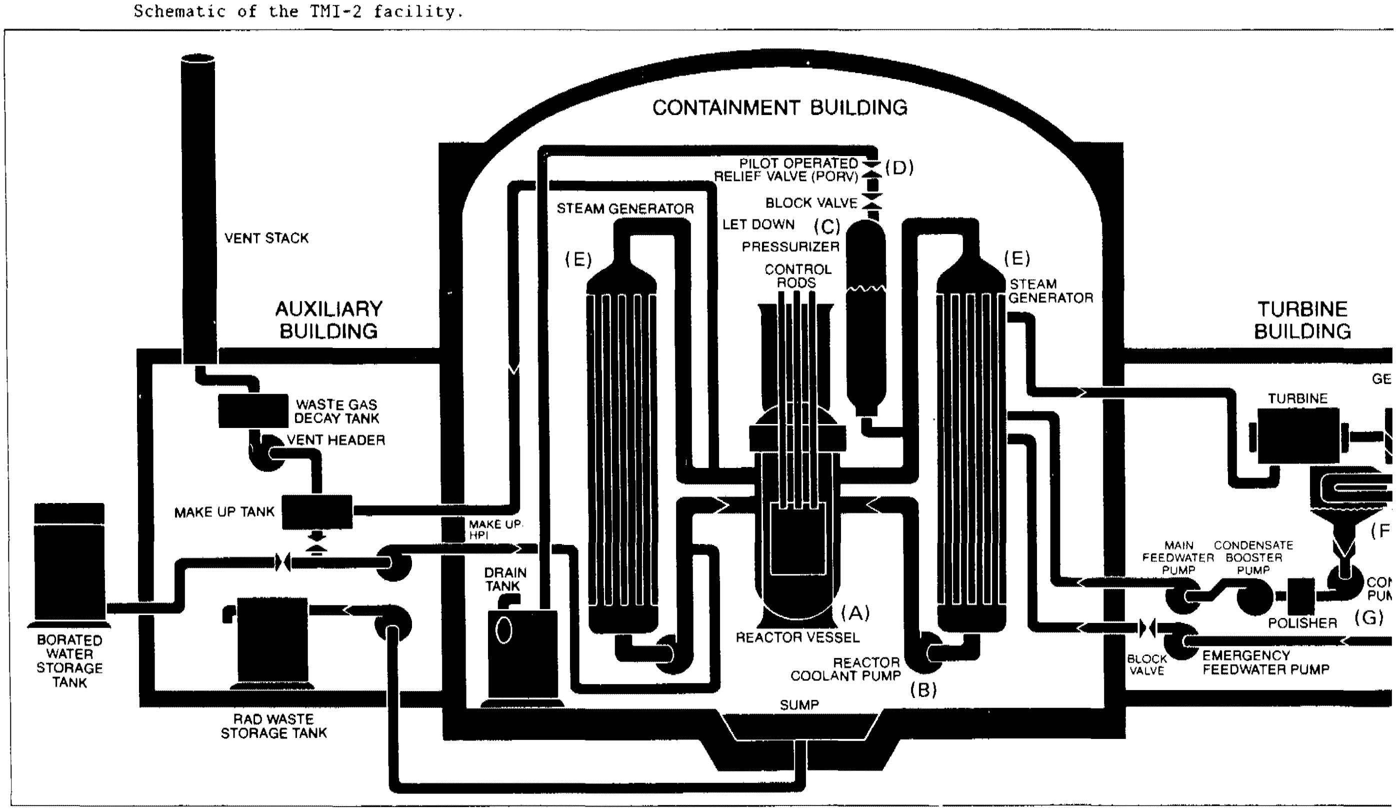
allowed the coolant going out of the core to boil, and steam voids or bubbles had formed within the primary system between the reactor core and the coolant in the pressurizer. Under such conditions, the level of coolant in the pressurizer would not disclose the amount of coolant in the primary system as a whole.

The pressurizer relief valve remained open for about two hours and 20 minutes, permitting the escape from the primary coolant system of more than 30,000 gallons of slightly radioactive water. Early in the accident, the operators were also letting coolant out through a "letdown" system, in the belief that the system was close to filling up. In fact, more coolant was leaving the primary system than coming into it, and this led eventually to "uncovering" of the upper portion of the reactor core resulting in sharp increases in temperature, damage to the fuel rods and releases of radioactive fission products. Just how extensive the damage was to the core and fuel cannot be determined until technicians are able to open the reactor vessel. Estimates of the damage are based on analyses of samplings taken from the atmosphere inside and from coolant standing on the floor of the containment, and they tend to indicate extensive damage to the fuel.

At 4:08 a.m., a sump pump came on automatically and began moving the slightly radioactive coolant—which had come down from the drain pipe for the relief valve and from the letdown system—into sump tanks located in the Unit 2 auxiliary building. It was at this point that radioactive material first left the containment; some of it was eventually vented to the outside air (though the more serious releases came later). At 4:11 a.m., the reactor building sump overflowed. Some minutes later the control room crew was apprised of this and, at 4:39, turned off the sump pumps in the containment. By that time, something over 8,000 gallons of water had been pumped to tanks in the auxiliary building, which was not sealed off from the outside air as the containment building was. At 4:50 a.m., the superintendent of technical support for Unit 2 arrived, but he too found a situation he had never experienced: a high level of primary coolant in the pressurizer but low pressure in the coolant system.

At 5:14 a.m., reacting to vibrations in the four pumps circulating coolant through the reactor (caused by steam in the coolant), the operators shut down two of them. Twenty-seven minutes later, for the same reason, they shut down the other two, cutting off all flow of coolant to the reactor core. The expectation at this stage was that the primary system could now work by "natural circulation" with the coolant heated by decay heat expanding and moving upward to the steam generators (whose feedwater was now restored and would carry off heat from the primary system) and with the cooler water flowing down and back to the reactor. The operators did not succeed, however, in establishing natural circulation.

Schematic of the TMI-2 facility.

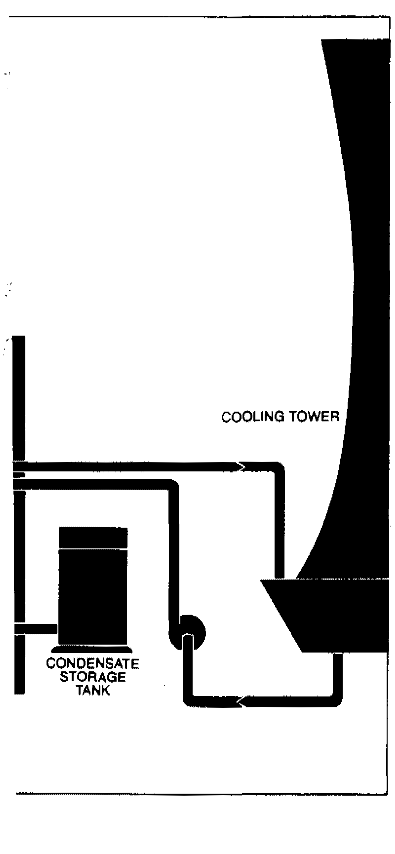


By 6:00 a.m., there was evidence, from radiation alarms, of radioactive gas in the containment. Primary coolant continued to escape through the relief valve, now containing non-condensible radioactive gas and hydrogen generated by a reaction between the zirconium cladding on the overheated sections of the fuel rods and the steam in the system. Finally the relief valve was sealed when a block valve on the pressurizer was closed at 6:20 a.m. That action ended the loss of coolant from the primary system, but the flow of coolant was not resumed until 6:45, when a reactor coolant pump was reactivated; vibrations again caused the operators to turn off the pump.

A conference telephone call took place beginning about 6:00 a.m., involving officials of the licensee company and a representative of the reactor manufacturer. About the same time, radiation readings at various points on the island began to show abnormal increases and instruments in the reactor core registered abnormally high temperatures. At 6:50 utility officials publicly declared a "site emergency," a procedure prescribed in the facility's emergency plans whenever an event posed the possibility of an "uncontrolled release" of radiation to the immediate environment. Local and State authorities were notified of the potential impact on public safety, beginning with the 7:02 a.m. notification of the Pennsylvania Emergency Management Administration (PEMA). The licensee tried to contact the NRC Region I office near Philadelphia starting at 7:10, but the switchboard there did not open until 7:45. The TMI station manager arrived on the scene shortly after 7:00 and at 7:24, he declared a "general emergency," signifying a situation with the potential for "serious radiological consequences" for public health and safety.

At 7:45 a.m., the NRC regional office was made aware of the situation at TMI and established an open line with the Unit 2 control room within a few minutes. By 8:00, the NRC headquarters was alerted and the Operations Center in Bethesda, Md., was activated. The regional office dispatched a first team of inspectors to the site about this time, and other agencies mobilized in response to communiques from NRC and State authorities.

Radiation monitoring on and near the island had begun before 8:00 a.m. and was to broaden and intensify throughout this and subsequent days of the accident. A helicopter engaged by the utility was taking samples above the plant by midday and another aircraft detailed from the Department of Energy (DOE) was in action by mid-afternoon of the first day. From the beginning, the level of radioactivity around the TMI site was in the range of one or two millirem-per-hour (thousandths of a rem) on the ground, though readings above the island and at some points on the plant grounds or just across the river were much higher and inside the containment ran up to thousands of rem-per-hour. The radioactive coolant which had overflowed the sump tanks in the containment building was automatically pumped over to the aux-




At left is a schematic drawing of the Three Mile Island Unit 2 facility. Some of the major components are labeled as follows:

(A) REACTOR VESSEL: A cylindrical vessel made of steel—40 feet high and 8½ inches thick—which contains the reactor (core and control rods) and through which the reactor coolant flows, carrying heat away from the core to the steam generators. The TMI-2 reactor contains 177 fuel assemblies with 208 fuel rods in each assembly.

(B) REACTOR COOLANT PUMP: One of four pumps which move the reactor coolant through the core to the steam generators and back to the core in a closed system (the primary system) of what is normally only slightly radioactive water. About one hour into the TMI accident, the operators shut down two of these pumps because they were vibrating severely, the result of the steam in the primary system. Half an hour later they shut down the other two pumps for the same reason. At that point, damage to the fuel in the core began, causing releases of radioactive material into the coolant.

(C) PRESSURIZER: A large vessel connected to the primary system between the reactor and the steam generators which is normally a little more than half full of water, with a steam bubble in the upper portion of the vessel. It is designed to keep the pressure in the reactor coolant relatively constant.

(D) PILOT OPERATED RELIEF VALVE: The pressurizer relief valve located at the top of the pressurizer and designed to open automatically when primary system pressure rises to a preset level and it becomes desirable to let off steam. When pressure is back to normal, the relief valve is supposed to close by itself. At TMI-2 it failed to do so, and reactor coolant flowed through the relief valve and down to a drain tank on the floor of the containment building. This valve remained open for more than two hours.

(E) STEAM GENERATOR: The large vessel in which the transfer of heat from the reactor coolant to the feedwater takes place. The transfer results in the conversion of the feedwater into steam, as it flows around tubes carrying the pressurized, core-heated coolant from the reactor. This steam is conveyed to the turbine which powers the electrical generator.

(F) CONDENSER: The vessel in which the steam which has passed through the turbine is condensed to a liquid state again. The heat is removed by pipes carrying condenser water which flows to the cooling towers and back to the condenser.

(G) CONDENSATE PUMP: The pump which moves the feedwater (the condensate) from the condenser to the polisher or demineralizer which cleanses the water before it flows back to the steam generator. The TMI accident began at this point in the feedwater system when plant personnel were trying to clear a line associated with the polisher and the condensate pump automatically shut down, followed by a similar "tripping" of the feedwater pump and subsequently of the turbine and the reactor.

Above is a photo of the conference room at the NRC Operations Center in Bethesda, Md., taken during the course of the TMI accident. Other rooms of the center are equipped and staffed to gather and analyze data and maintain secure communications with NRC regional offices and the accident site. Numerous technical experts from the NRC were at the center to inform and advise senior NRC officials on the Executive Management Team. Personnel from other Federal agencies involved in or assisting with management of the accident were officed in areas adjoining the Operations Center. In the foreground, at left, is Lee V. Gossick, NRC Executive Director for Operations.

iliary building tanks where it again overflowed. Since the auxiliary building is not isolated from the outside environment, some radioactive gases carried over in the coolant were vented to the outside. The reactor containment building was not sealed off from the auxiliary building until about 9:00 a.m., after more than eight thousand gallons of coolant had been transferred.

This transfer of coolant was not, however, the main cause or source of the release of radioactivity to the environment during the TMI accident. The transfer actually took place prior to any major fuel damage in the reactor. It was between one and two hours following the turbine trip, when the operators turned off the reactor coolant pumps to save them from vibration damage, that damage to the nuclear fuel began. For the next several hours, there was a large temperature difference between the coolant entering and exiting the nuclear core, indicating inadequate flow of coolant through the core. As a result of fuel damage, the concentration of radioactivity in the reactor coolant increased by several orders of magnitude. A flow of this highly contaminated reactor coolant was maintained from the primary coolant system through the letdown system and returned to the primary system via the makeup system. This flow, maintained for several days following the accident, was necessary to ensure adequate cooling of the reactor coolant pump bearings. Normally the gases evolving from the reactor coolant in the letdown and makeup systems are of little radiological significance. During this period, however, these gases caused very high radiation levels inside the auxiliary and fuel-handling buildings and resulted in much higher than normal environmental releases via the ventilation exhausts from these buildings. This flow was the principal pathway by which radioactivity passed from the damaged reactor core to the auxiliary building, fuel-handling building, and to the environment.

At about 8:00 a.m., the station superintendent and other officials on the site decided to try again to activate the reactor coolant pumps. After some difficulty, two of the four pumps (one in each loop) were restarted. By 8:30, there was new coolant entering the primary system from the ECCS.

At 9:15, the White House was notified of the accident by the NRC. The team dispatched by NRC Region I arrived at the site by 10:15. It was shortly afterwards that the radiation level in the Unit 2 control room required that personnel there don respiratory masks. These proved to be a hindrance to clear communication. At 11:00 a.m., all non-essential personnel were ordered off the island. It was about this time that the NRC and State radiation protection officials asked the Department of Energy (DOE) to send a team from the Brookhaven National Laboratory to help with radiation monitoring.

About 11:30 there began an attempt to depressurize the reactor coolant system so as to be able to activate the low-pressure decay heat removal system. The pressure, however, remained too high for this purpose because of the volume of hydrogen gas and steam in the primary coolant system. Hence, the decay heat removal sysyem could not be initiated, and the attempt at repressurization was terminated about 3 p.m. Repressurization began at about 5:30 and was completed at about 6:45.

Sometime around noon, three licensee employees entered the Unit 2 auxiliary building and found radiation levels of from 50 to 1,000 rem-per-hour; each of the three incurred radiation doses of 800 millirem. At 1:50 p.m., a hydrogen explosion or "burn" took place in the Unit 2 containment building. Personnel on hand later remembered hearing a thud about this time and the computer chart showed a sudden pressure surge in the containment up to 28 pounds-per-square-inch, but the meaning of the spike on the chart was not immediately recognized.

By evening of the 28th, NRC had 11 people on the TMI site and a mobile laboratory van for analysis of the radiation content of environmental samples. A team from the Brookhaven National Laboratory had been assisting with the radiation monitoring since mid-afternoon, as had the aerial survey aircraft from DOE. About 8:00 p.m., a reactor coolant pump was activated and coolant flow was established, carrying heat out of the reactor through one of the steam generators to the condenser, bypassing the turbine. The primary system remained essentially in this mode for a month, until natural circulation was finally achieved on April 27.

## Thursday—March 29

On Thursday morning, a team of seven specialists from NRC headquarters arrived at the site. At that time the radiation readings at and near the plant were not negligible but also were not alarming. No significant iodine releases were detected. These would be considered especially hazardous because radioactive iodine, should it enter the human food chain, tends to accumulate in the thyroid and can cause cancer of that gland. The Congress evinced immediate and urgent interest in events at the plant: Chairman Hendrie was called to explain the situation before the House Subcommittee on Energy and the Environment, and Senators Heinz and Schweiker and Congressmen Ertel and Goodling—all of Pennsylvania—were briefed by the utility and the NRC. During the afternoon, some waste water from the plant was discharged by the licensee into the Susquehanna River. Because it contained only slightly radioactive material, the release did not constitute a violation of NRC regulations, but, with all the uncertainties still surrounding the scene at TMI, the NRC Chairman ordered the discharges stopped. Late in the day, analyses of coolant samples confirmed the presence and showed something of the extent of the core damage that took place during the periods that the core was uncovered on Wednesday. (It was later determined that there had been three periods when a significant portion of the core was being cooled by steam rather than fluid coolant.) First concerns about the presence of a hydrogen bubble in the reactor vessel arose on Thursday, and the fact that there had been a hydrogen explosion outside the vessel in the containment building early Wednesday afternoon was brought to light.

## Friday—March 30

Friday was the day when it became clear to all concerned that the event was far from over; that radiation releases from the auxiliary building were not under control and were increasing; that there was a large gaseous bubble in the reactor vessel which could conceivably expand, forcing the level of coolant below the top of the core, uncovering it again; that, according to some analyses and expert judgments, the bubble might become flammable as oxygen evolving from the decomposition of water by radiation made its way into the upper part of the vessel; that radiation was emanating from the facility in a manner neither planned nor controlled.

Early in the day, reports of a 1200 mr/hr reading above TMI-2 precipitated serious discussion at the NRC Operations Center in Bethesda of the possibly urgent need to evacuate the residents of Goldsboro, Middletown and other communities and areas around the plant, even out as far as Harrisburg. The fact that there was a consensus favoring such a recommendation at the Operations Center was relayed to State officials in Pennsylvania, occasioning considerable anxiety and confusion, since the judgment was not shared by people at the plant site. The NRC position was clarified when Chairman Hendrie spoke with Governor Thornburgh about 10:00 a.m., and counseled against full-scale evacuation of the population, suggesting instead that the Governor recommend that people stay indoors for awhile until the true situation could be better defined. The Governor did so. About 40 minutes later, President Carter contacted Chairman Hendrie and directed that a senior NRC official be dispatched to the TMI site as his personal representative; the President also assured that the White House staff would see to it that an adequate and dependable communications system would be set up as soon as possible between the site, the White House and the NRC. Prior to this, communications between the plant and the NRC had been unreliable and had even been lost for a time. The Director of NRC's Office of Nuclear Reactor Regulation, Harold Denton, left NRC headquarters for the TMI site with a support staff of 12 to serve as the President's representative and as the primary NRC official on the scene. Shortly after noon, Chairman Hendrie indicated to Governor Thornburgh by telephone that a recommendation by the Governor that pregnant women and pre-school aged children within five miles of the plant leave the area temporarily was advisable. The Governor made this recommendation soon afterwards.

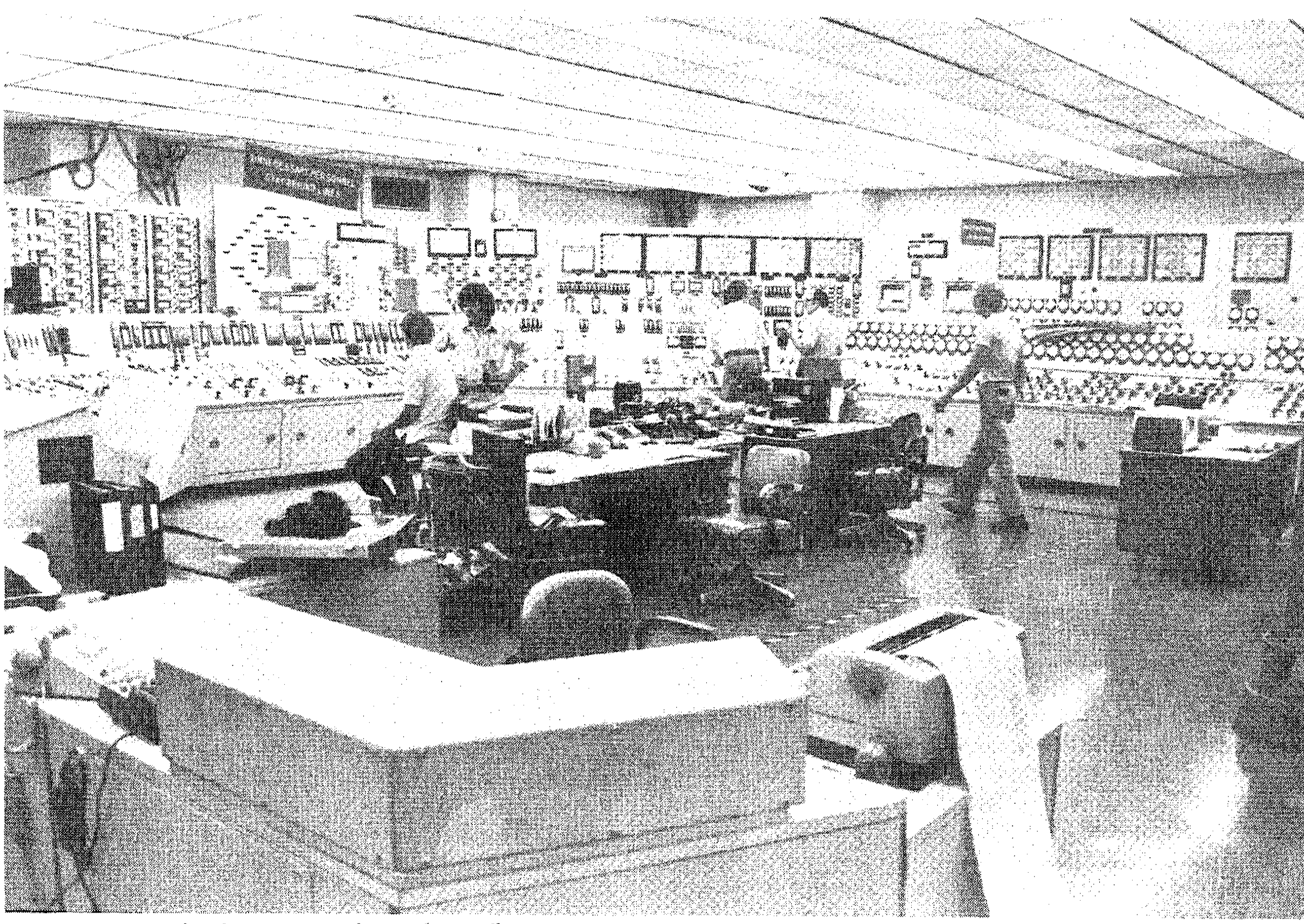
Discussions and assessments of the possible need for total evacuation of the population near TMI continued throughout the day among NRC, other Federal and State officials. About an hour after the former's arrival at TMI and a first assessment of conditions in and around the plant, NRR Director Denton and Chairman Hendrie reviewed various possible courses the accident might take—or that licensee personnel might take in their effort to gain control of events—and the implications of each for a judgement on whether and when to move people out of the area. Within an hour of their conversation, Chairman Hendrie was in contact with Governor Thornburgh, at which time he advised the Governor that, though the bubble in the reactor vessel could cause trouble later in keeping the core cooled, there was no appreciable amount of oxygen in it and the chance of a hydrogen explosion such as took place in the containment on Wednesday was "close to zero." The Chairman also appraised the chance of a core meltdown as being extremely low, but the possibility of a significant radiation release as being somewhat higher.

Additional contingents of NRC personnel were sent to TMI during the day and by 4:00 p.m. there were 83 NRC staff people at the site. Other Federal agencies—DOE, EPA, FDAA and others—and State officials responsible for emergency management and radiation protection were also present in force. In a press release issued around 6:00 p.m., the NRC Chairman declared that there was "no imminent danger of a meltdown" of the reactor core. By 8:30 on Friday evening, Governor Thornburgh decided, having consulted with NRC officials on the site, to lift the advisory that people within five miles of the plant should stay indoors but, with NRC concurrence, continued to recommend that pregnant women and young children leave and/or stay out of the area.

By day's end, there was deep uncertainty among all concerned as to the potential hazard represented by the hydrogen bubble in the reactor vessel. National laboratories and industrial experts, as well as NRC research personnel, were at work calculating how long it might be until the amount of oxygen finding its way into the hydrogen bubble would produce a flammable mixture in the upper portion of the vessel. Preliminary estimates of that time-frame varied. Later on it was realized that there was no appreciable build-up taking place because the oxygen resulting from the radiolytic decomposition of water was combining with free hydrogen in the reactor coolant.

## Saturday—March 31

On Saturday the focus of concern had shifted from periodic uncontrolled radiation releases to potential explosion of the hydrogen in the reactor vessel. Radiation readings were very low everywhere but inside the containment. The NRC and other Federal presence at the site was expanding. The NRC Commissioners meeting in Washington, D.C., continued discussions of what changes in the situation might warrant a recommendation that people leave the TMI area, or whether such a recommendation should be made immediately, as a precaution. The conditions at TMI-2 were improved in virtually every respect, except for the hydrogen problem, and the Commissioners were



The control room for TMI Unit 2 is shown above. The instruments and controls are deployed in a U-shaped pattern in a design intended to permit one operator to supervise operations under normal, stable conditions. During abnormal situations, it is expected that additional operators would be available to give any needed assistance.

conscious of the hardships an evacuation would impose upon the population. There was also the matter of range to consider, whether to evacuate out to five miles or 10 miles or more, and of how much time would be available, if core conditions began to deteriorate, before the defensive barriers of the plant would be breached.

Around noon the NRC Chairman and NRR Director at the site discussed the situation at length, considering both the kinds of events that would signal a need to begin moving people out and also various means by which to reduce the hydrogen hazard. Soon afterwards, estimates were received from various research teams that the conditions necessary for hydrogen combustion or explosion in the reactor vessel were perhaps days away, and it appeared that there would be amply sufficient time to vent the vessel into the containment or otherwise defuse the danger. In mid-afternoon, Chairman Hendrie held a press conference at the NRC Operations Center in Bethesda, Md., at which he affirmed that a precautionary evacuation of the TMI area was still a possibility, especially if it were decided to try to force the hydrogen bubble out of the reactor vessel. Soon after, the Chairman and the Governor conferred by phone. Responding to the latter's query, the Chairman advised that, since some low-level releases of radiation were still coming from the auxiliary building, it would be prudent to continue the recommendation on pregnant women and pre-school aged children and to keep emergency planning personnel and resources in readiness.

## Sunday—April 1

Following a brief meeting with the staff at Bethesda, Chairman Hendrie left Washington to go to the TMI site. President Carter was to arrive there in the early afternoon for a tour of the scene and briefings on the status of the reactor. During the morning, the NRC personnel at the site had augmented the radiation monitoring equipment by placing 37 thermoluminescent dosimeters within a 12-mile radius of the plant. By mid-afternoon the bubble in the reactor vessel seemed to be dissipating and the system stabilizing, though intense discussion of the evacuation question continued among Commissioners and staff in Washington. Chairman Hendrie communicated the favorable change in the situation to the group in Washington and characterized the next phase in management of the accident as a choice between moving at once to activate decay heat removal from the reactor or moving slowly and letting the reactor cool at its own rate.

Reactor cooling was maintained by the action of one of the main coolant pumps providing the flow through the reactor core, and heat removal through one of the steam generators to the condenser, until about 2:00 p.m. on April 27 when the reactor coolant pump was intentionally shut down and core cooling by natural circulation was achieved.

Governor Thornburgh and President Carter are escorted into the TMI Unit 2 control room by a Metropolitan Edison employee on Sunday, April 1, 1979.

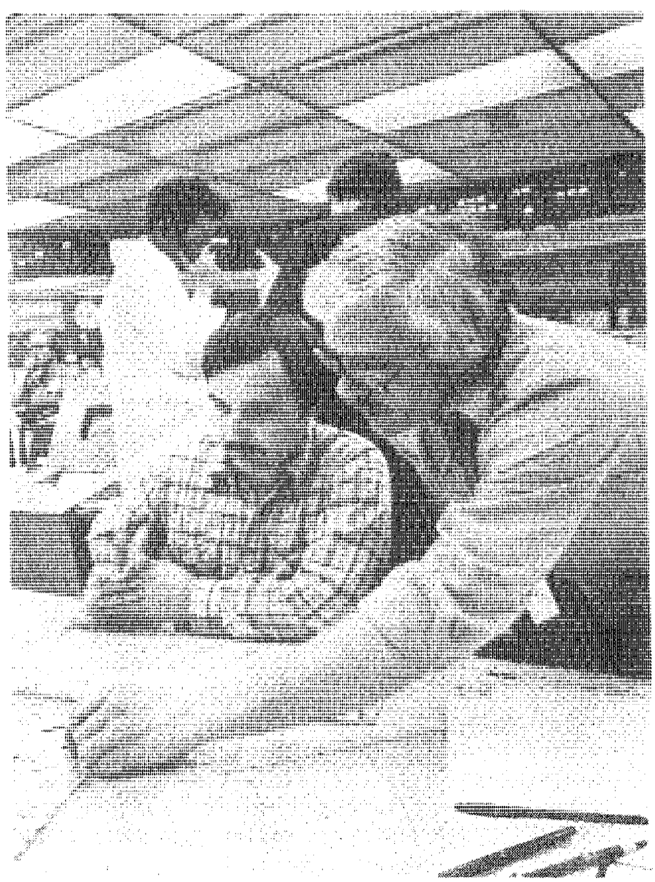
A bulletin was transmitted Sunday afternoon to all NRC licensees operating reactors of the B&W design to make an immediate review of plant conditions and to implement a number of precautionary measures derived from the TMI experience. NRC inspectors were also sent out to confirm that the prescribed actions were taken. The bulletin was the first in a series issued by NRC to licensees as analyses of the TMI accident revealed both necessary and prudential actions to be taken to prevent recurrence of the event (see "Bulletins and Orders Task Force," below).

Later in April, licensees for the other nuclear power plants employing B&W nuclear steam supply systems indicated that they would voluntarily shut down until prescribed alterations in design and procedures were completed. Confirmatory orders to that effect were subsequently issued by NRC for several of these units.

By the end of May, "dedicated" telephone lines had been established between the NRC Operations Center in Bethesda and 68 of the 70 licensed nuclear power plants and 14 licensed fuel cycle facilities. The lines make it possible for operations personnel in these facilities to communicate immediately and directly with members of the NRC's technical staff any time of the day or night on any day of the year. The system also provides for instant communication with any one of the five NRC regional offices.

The accident at TMI-2 generated investigations, reports, findings and recommendations literally too

Water from the industrial waste treatment systems of the plant (TMI), designed to be non-radioactive, is checked to insure that is has not been contaminated.



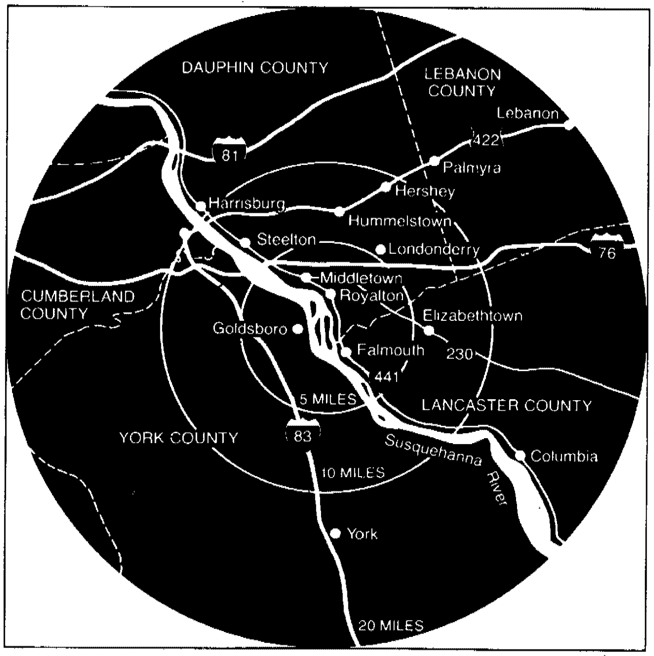
Plotting wind direction to assist in monitoring operations.

numerous to mention. The balance of this chapter attempts only to describe the major NRC undertakings in the matter and to cover the findings and recommendations of the commission appointed by President Carter to conduct an independent investigation of the accident and its implications, together with NRC's responses to those recommendations. At the time this report was prepared, the work of the NRC Special Inquiry Group—an investigatory body set up by the NRC under independent directorship—was not yet complete, nor had the various Congressional reviewers reported their results.

## RADIOLOGICAL CONSEQUENCES TO PERSONS AND THE ENVIRONMENT

**Individual and Collective Doses.** NRC staff members participated in an interagency study to determine the individual and population doses associated with the TMI accident. The results of the study are presented and discussed in the NRC report, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station" (NUREG-0558). Based on environmental measurements performed during the accident, it was estimated that the maximum individual off-site whole body dose was about 83 millirem, which is approximately one-sixth the NRC's allowable maximum whole body dose of 500 millirem-per-year. The population within 50 miles of the TMI site received an estimated integrated dose of 3,300 person-rem. This population dose is expected to result in less than one additional fatal cancer among the exposed population, in which 325,000 fatal cancers can be expected to occur as a result of other causes.

Radiation doses to licensee employees have also been estimated. Occupational whole-body doses accumulated from the date of the accident through May 31, 1979, totaled 225 person-rem. These doses were received by employees in performing recovery operations after the accident, such as changing filters in the cleanup systems for air leaving the auxiliary building and fuel-handling building, sampling of air and primary coolant, decontamination and radioactive waste processing operations, and routine inspection and maintenance activites. In the days immediately following the accident, four persons received exposures exceeding NRC regulatory standards. Two persons involved in taking a primary-coolant sample received doses substantially in excess of the standards. One person received a total body dose of 4.1 rem (the regulatory limit is 3.0 rem), an extremity (finger) dose of 147 rem (the limit is 18.75 rem) and skin dose to the top of the head of 13 rem (the limit is 7.5 rem). The second person received extremity doses of 54 rem. Two other persons received whole body exposures of 3.2 rem and 3.1 rem, which are slightly higher than the NRC limit of 3.0 rem.




**Map of the TMI area showing 5-, 10- and 20-mile evacuation zones.**

**Environmental Protection at Three Mile Island.** During the accident at Three Mile Island there was concern that a core meltdown might occur. This could have led to the contamination of the groundwater of the island and ultimately of the Susquehanna River and beyond. The staff developed contingency plans to mitigate the effects of groundwater contamination by isolating the immediate plant area from the regional water supplies. The plans provide for blocking groundwater movement, for withdrawing the potentially contaminated water, and for monitoring and temporarily storing the contaminated water. Working with the U.S. Army Corps of Engineers, the NRC staff formulated a plan to construct a bentonite-cement cut-off wall, dewatering wells, and a pumping system. Availability of equipment needed to carry out the plan was verified. Plans for monitoring and on-site storage were also completed. It did not prove necessary to implement planned isolation of the area.

Another problem encountered in the accident was the need for the staff to produce estimates of the transport or diffusion of gaseous releases, in order to plan for possible evacuation of the population and for assessment of the consequences thereof. These estimates were made by staff meteorologists assigned to the NRC Operations Center. Well into the accident, the staff ascertained that meteorological data were available from the TMI meteorological tower by remote access and made use of this information. In addition, the staff arranged for National Weather Service (NWS) to provide supplementary meteorological instrumentation at the site. The staff established communications with and utilized the forecasting services of the NWS Harrisburg River Forecast Center and NWS Philadelphia Area Weather Forecast Center. The staff's estimates of the transport and diffusion of TMI releases were used in estimating doses for the locations of dose-rate instrumentation both on the TMI site and off. Because the magnitude of the release was unknown during the early stages of the accident, data from environmental monitors and meteorological estimates were used to calculate releases. Atmospheric transport estimates were used to advise evacuation planners.

In further protection of the environment, the possible non-radiological consequences to the aquatic biota and recreational fisheries of the Susquehanna River from the accident at Three Mile Island Nuclear Station in late March of 1979 were investigated up through the post-accident period (through June). Data used in the investigation included site-specific biological and water quality information collected by the license under the Environmental Technical Specifications and National Pollutant Discharge Elimination System monitoring programs, and also information from State and Federal agencies, knowledgeable persons, and

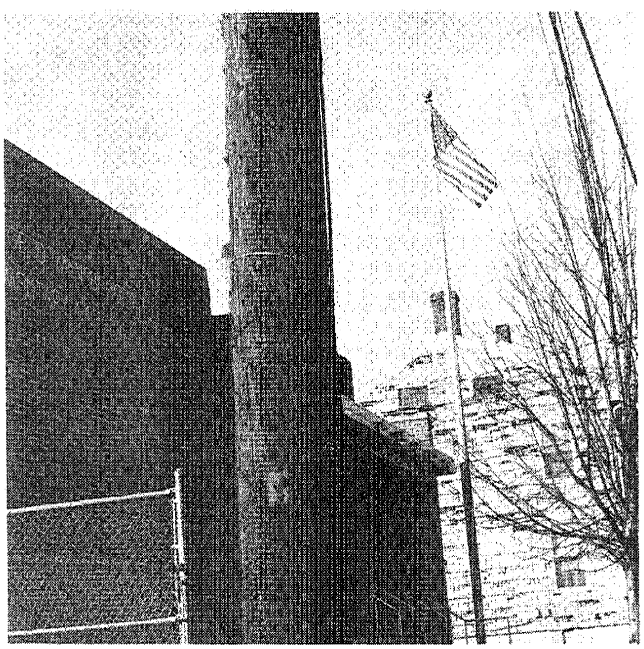
**Samples of grass from the area surrounding the plant and the Susquehanna River are shipped out by the EPA to be checked for radioactive contamination.**

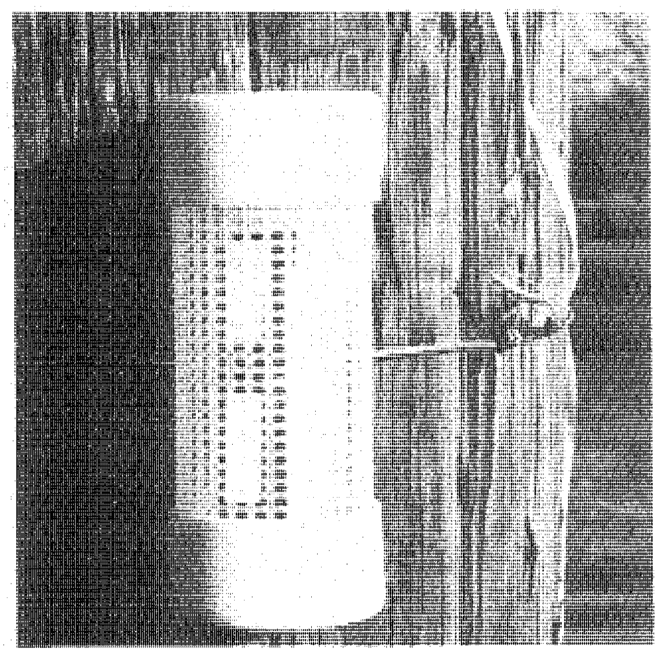
studies conducted in other upstream and downstream areas of the river. Thermal and chemical discharges during and following the accident did not exceed the effluent limitation established to protect the aquatic environment. Although several million gallons of treated industrial waste effluents were released into the river, these discharges were not of unusual volume compared with normal operation and were a very small portion of the seasonally high spring river flows. The extent and relative location of the effluent plume were defined and the fish species known to have been under its immediate influence were identified—including rough, forage, and predator/sport fishery species. Impacts to benthic invertebrates or fishes were not detected. No unusual conditions of fish disease or mortality were noted in the river following the accident. The normal spring increases in abundance and species-composition of riverine fauna occurred, as did the onset of the fish spawning season in April with peaks of ichthyoplankton abundance in May and June.

Nevertheless, post-accident recreational fishing in the Three Mile Island vicinity underwent significant departures from historical trends. Fishing activity appeared to shift away from the Susquehanna River waters near the nuclear station to other areas, especially downstream. Anglers returned greater proportions of their catches than during any comparable period within the previous five years. This was most notable during April when anglers fishing near the plant returned an unprecedented 100 percent of their catches. Thus, in the waters receiving station effluents during the month following the accident, the liquid radiological pathway leading to man via fin fish consumption could have been absent entirely. With the passage of time following the accident, the normal pattern of recreational fishing was approached. The investigation defined several generic aspects of the accident and lessons applicable to other facilities: the time of the accident with respect to the biological season, and to the ability to detect an impact; data availability and data needs for adequate monitoring; and the application of the non-radiological findings for radiological assessment. This investigation is described in an NRC report: "The Non-Radiological Consequences to the Aquatic Biota and Fisheries of the Susquehanna River from the 1979 Accident at Three Mile Island Nuclear Station" (NUREG-0596).

## TMI RECOVERY OPERATIONS

Following the accident of March 28, a substantial effort was mounted to provide technical assistance, regulatory guidance and review of the licensee's operations procedures and system addition and modification activities. A team began to form with the arrival of the Office of Inspection and Enforcement Region I inspectors shortly after the accident and continued to expand with the arrival of the first contingent from the Office





**Thermoluminescent dosimeter (enlarged at right), used by the NRC to measure the amount of airborne radiation delivered to a specific place, is shown mounted on a utility pole near a school in Middletown, Pa. Similar devices were installed both on the TMI site and at various locations around the plant up to 15 miles away by NRC, the Environmental Protection Agency, Metropolitan Edison Co., and an independent contractor. In addition to making independent evaulations, each group sent the data collected to both the NRC and EPA for analysis.**

of Nuclear Reactor Regulation (NRR) on March 29 and additional inspectors from all five regional offices. On March 30, the Director of NRR and additional NRR staff arrived on the site to assist in the recovery operation. A Public Affairs Office was also established in Middletown, PA, and staffed on a 24-hour-per-day basis to handle the flow of information to the public and the media.

NRR staff analysts in many of the major disciplines were brought to TMI to provide needed technical resources. The specific activities engaged in by the staff can be broken down into four major areas:

(1) A review was initiated of the system modifications and system additions (proposed by the licensee, the industry review group, or the NRC) as contingency measures to mitigate the consequences of the accident and to provide assurance for continued safe shutdown and long-term safe shutdown.

(2) Substantial effort was given to the review of all procedures, both emergency and normal operation and maintenance, which were necessary to post-accident activities. In many cases, because of changes in the use of normal systems and the addition of new ones, new operating procedures were necessary. Further, the facility license and technical specifications, which defined the limits for operating parameters and surveillance requirements, were no longer fully applicable to the post-accident facility, though existing facility procedures provided a mechanism for establishing specific operability limits and surveillance requirements. It was necessary, from a regulatory point of view, to have NRC's review and approval of any new procedures that might be in conflict with the pre-accident license.

(3) NRR provided close and continuous monitoring of the operations in progress to assure that system parameters stayed within expected limits and to provide prediction of future system performance and the capability of plant systems to maintain safe conditions.

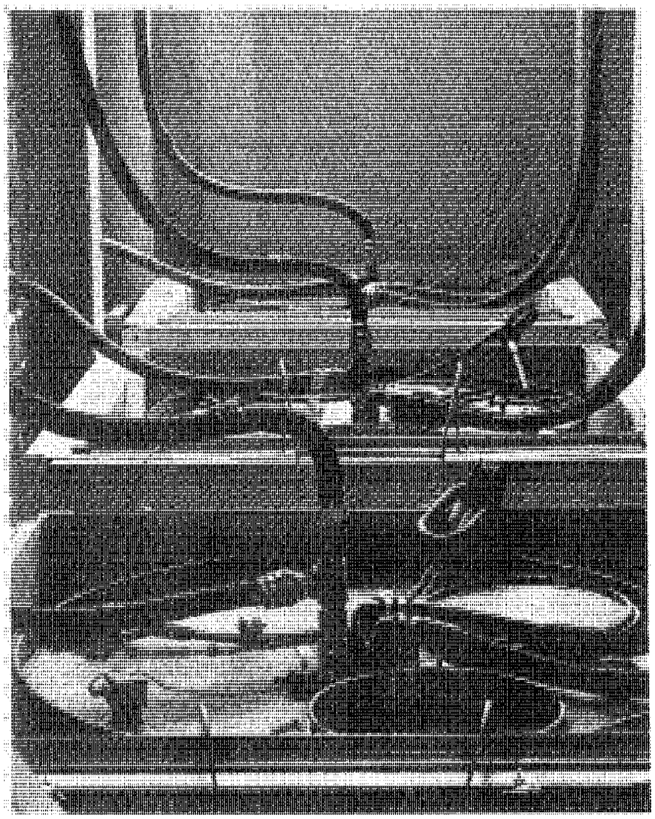
(4) Lastly, substantial NRR effort was committed to providing consultation, review and analysis of the ongoing radwaste, cleanup, and health physics activities. The accident generated a significant amount of contaminated water which, in turn, contaminated substantial portions of the facility and its systems. This made it difficult to have normal access to systems important to safety and also constituted a threat of further fission product release and occupational exposure. In addition, the radiological makeup of the contamination was different from that normally encountered in operating reactors, in terms of its airborne intensity as well as its ratio of beta and gamma activity. It was therefore an important concern—particularly in view of the intensive work activity needed to continue safe operation—that operator exposures be maintained within acceptable limits and the environment protected from undue radiological effluents.

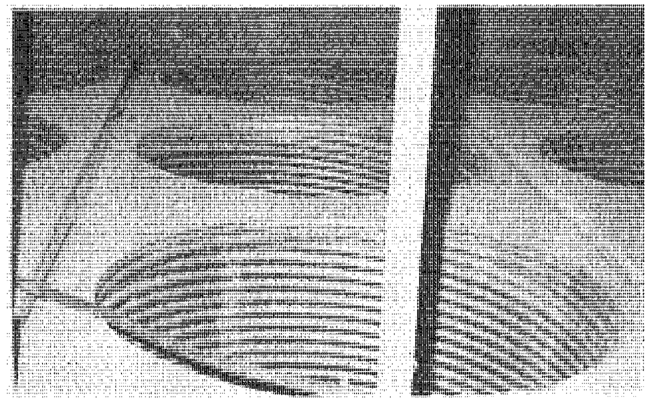
Examples of the system review activity undertaken by the NRR on-site staff were design reviews and evaluations of the following sytems:

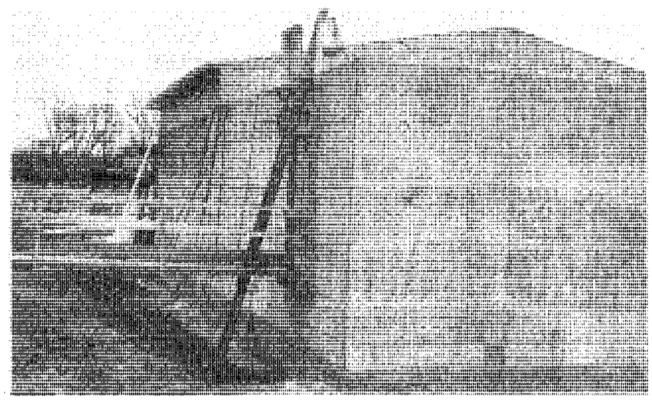
(a) Supplementary diesel generators
(b) Supplementary filtration systems
(c) Long-term cooling systems
(d) Alternative decay heat removal system
(e) Pressure volume control system
(f) Tank farm for storage of radioactive liquids
(g) EPICOR-II system for processing of contaminated liquids
(h) Many monitor modifications in existing systems which allowed operability in the post-accident environment.

Besides the systems reviews, approximately 250 procedures were reviewed and approved by the on-site staff. This activity was particularly important in the first two months following the accident because a serious shortage of personnel familiar with the facility existed; the NRC review constituted not only a regulatory approval of the intended operation, but also served as a quality assurance check on adequacy and operability. The review of procedures is continuing as the licensee rewrites emergency and operating procedures to reflect the changing status of the facility. It is anticipated that such procedure review will be necessary until a new set of facility technical specifications, which reflect the post-accident facility configuration, is implemented.

A substantial amount of staff effort was expended on the review and approval of the EPICOR-II operation, intended for use in decontaminating the 380,000 gallons of intermediate-level contaminated water held in the auxiliary building tanks and in the tank farm constructed following the accident. EPICOR-II was designed and constructed following the accident because it was clear that storage of water would be a significant problem and could not be accommodated with existing facility equipment. EPICOR-II is a three-stage demineralization system, constructed in an existing on-site building. EPICOR-II was provided with sufficient shielding and remote-handling capability to accommodate the processing of radioactive water up to a level of about 100-microcuries-per-milliliter. When facility operation was near, court actions were initiated to prevent operation of EPICOR-II or disposal of the decontaminated water. In response to these actions, the Commission directed that an environmental assessment for the use of







The "EPICOR-II" system being used to decontaminate some 380,000 gallons of intermediate-level radioactive water held in the auxiliary building tanks at the TMI-2 site is shown above. It consists of three process vessels (steel liners) shielded by four-inch lead enclosures located in the chemical cleaning building. Each vessel contains ion-exchange resin. The vessel at the top of the photo at the left is the system prefilter/demineralizer, the center vessel is a cation ion-exchanger, and the third vessel is a mixed-bed polishing ion-exchanger. Each is fitted with three quick-disconnect hoses: a liquid waste influent line, a processed waste effluent line, and a vent line with attached overflow hose. Vented air from each vessel passes through a special filter and charcoal adsorber. "Spent" ion-exchange resin liners containing radioactive material removed from the water are transferred by crane to cells (shown at top right) which are housed in modular concrete storage structures (above). The cells are concrete-shielded, galvanized corrugated steel cylinders seven feet in diameter and 13 feet high. The storage module shown under construction has 4-foot thick walls and will be 57 feet wide and 91 feet long. The modules, each holding about 60 storage cells, will be built on an as-needed basis. Shipment of the radioactive liners away from the site will depend on approval of a disposal facility and availability of shipping casks.

EPICOR-II be prepared, followed by the environmental assessment for the alternatives of disposal of decontaminated water. Both of these environmental assessments would be provided to the public for comment before any actions would be initiated. Environmental assessment for the use of EPICOR-II in the decontamination of the intermediate level of contaminated water in the auxiliary building was prepared and sent out for public comment on August 14, 1979. The assessment evaluated various alternatives to the proposed cleanup and concluded that the use of the already constructed system was the best alternative, and that the processing of water would constitute a negligible environmental impact.

Based on these evaluations, the Commission, on October 16, 1979, issued a Memorandum and Order directing the use of EPICOR-II.

## BULLETINS AND ORDERS TASK FORCE

The accident at TMI-2 involved a feedwater transient coupled with a small break in the reactor system (the open relief valve). Because of the severity of the ensuing events and the potential generic implications of the accident for other operating reactors, the NRC staff initiated prompt action to: (1) assure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2 type events, and (2) start comprehensive investigations into the potential generic implications of this accident on other operating reactors.

The Bulletins & Orders Task Force was established within the Office of Nuclear Reactor Regulation (NRR) in early May 1979. This task force was responsible for reviewing and directing the TMI-2 related staff activities regarding loss-of-feedwater transients and small break loss-of-coolant accidents for all operating reactors. The task force concentrated its efforts in the areas of: assessments of auxiliary feedwater system reliability; review of the analytical predictions of plant performance for both feedwater and small LOCA-induced transients; evaluations of generic operating guidelines; the review of emergency plant operating procedures; and the review of operator training.

The task force worked with operating plant licenses, and, for the review of generic items, with owners' groups for plants of each nuclear steam supply vendor (Babcock and Wilcox, Westinghouse, Combustion Engineering, and General Electric) and with the indilvidual vendors. Initial priority was placed on plants of the Babcock and Wilcox (B&W) design, but as short-term actions on these plants were completed, priority was shifted to other pressurized water reactor (PWR) plants, i.e., those manufactured by Westinghouse and Combustion Engineering. Activities related to boiling water reactors, a significantly different light water reactor type manufactured by the General Electric Company, were pursued as a third priority.

The task force, which was composed of approximately 30 technical professionals of widely varying disciplines and areas of expertise, evaluated licensees' responses to NRC Bulletins; the issuance and subsequent lifting of Orders to the B&W operating reactors; system reliability and predicted plant performance for each of the reactor vendors, with regard to feedwater transients and small break loss-of-coolant accidents; and related follow-on activities.

## Bulletins

The preliminary review of the accident chronology identified several events that occurred during the accident and contributed significantly to its severity. As a result, all holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors. These instructions were specified in a series of bulletins issued by the NRC's Office of Inspection and Enforcement (IE).

The initial bulletins defined actions by operating plants using the B&W reactor system, but as staff evaluations determined that additional actions were necessary, these bulletins were expanded, clarified, and issued to all operating plants for action. For example, as a result of staff evaluations, holders of operating licenses for B&W designed reactors were instructed by IE Bulletins to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer pilot-operated relief valve settings. A chronology of bulletins issued by IE is shown below.

The task force directed the evaluations of each licensee's response to the IE Bulletins. This process involved an inter-office review group, which included representatives from IE and from the NRR Division of Operating Reactors. When it was concluded that a licensee had understood and had provided an acceptable response to the bulletins, the bulletin review was completed and the evaluation issued as a staff report.

The prompt action taken by licensees in response to the IE Bulletins was considered an important contributor to the assurance of continued safe plant operation. The bulletins and related evaluations also provided substantial input to other staff activities, such as those associated with the generic study efforts and the Lessons Learned Task Force (see below). Thus, many of the subjects addressed by the bulletins were studied in greater depth through other staff activities and studies. Further, the bulletins and the associated responses were used as a basis for IE inspection activities and for auditing of reactor operator training.

## Orders on Babcock and Wilcox Plants

Soon after the TMI-2 accident, the NRC staff began a reevaluation of the design features of B&W reactors to determine whether additional safety corrections or improvements were necessary. This evaluation involved numerous meetings with the vendor and the affected licensees.

The conclusion of these preliminary staff studies was documented in an April 25, 1979 status report to the Commission. It was found that B&W designed reactors appeared to be unusually sensitive to certain transient conditions originating in the secondary system. The features of the B&W plants that contributed to this sensitivity were: (1) design of the steam generators which operate with relatively small liquid volumes in the secondary side; (2) lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system; (3) reliance on an integrated control system (ICS) to automatically regulate feedwater flow; (4) actuation before reactor trip of a pilot-operated relief valve on the primary system pressurizer (which, if the valve sticks open, can aggravate the event); and (5) a low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation (except for the Davis-Besse plant).

Because of these features, B&W design relies more than other PWR designs on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and the emergency core cooling system (ECCS) performance to recover from certain anticipated transients, such as loss of off-site power and loss of normal feedwater. This, in turn, can require greater operator knowledge and skill to safely manage the plant controls during such anticipated transients. As a result of the work supporting the April 25, 1979 report, the NRC staff concluded that certain other short-term design and procedural changes at operating B&W facilities were necessary in order to assure adequate protection to public health and safety.

After a series of discussions between the NRC staff and licensees of operating B&W plants, the licensees agreed to shut down these plants and keep them shut down until the actions identified to the Commission in the April 25, 1979 report could be completed. This agreement was confirmed by a Commission Order to each licensee (see "Actions Directed by Orders," below). Authorization to resume operation was issued in the period late May through early July, as individual plants satisfactorily completed the short-term actions and the NRC staff completed an on-site verification of the plant's readiness to resume operation. In addition to the modifications to be implemented promptly, each licensee also proposed to carry out certain additional long-term modifications to further enhance the capability and reliability of the plant systems to respond to transient events (see "Longer Term Actions," below).

Since some of the long-term modifications involve the design, procurement, and qualification of safety-grade hardware, not all of the actions of the long-term portion of the Orders were completed in 1979. Staff involvement will continue to assure that licensees complete each long-term action of the Order "as promptly as practicable" and that the Orders are closed out by a prompt staff acceptance review.

## Specific Plant and Generic Studies

For B&W operating reactors, an initial staff study has been completed and published in a staff report (NUREG-0560). This study considered the particular design features and operational history of B&W operating plants in light of the TMI-2 accident and related current licensing requirements. As a result of this study, a number of findings and recommendations resulted which are now being pursued.

Generally, the activities involving the B&W reactors are reflected in the actions specified in the Orders. Consequently, as noted earlier, a number of specific actions have been specified in the areas of transient and small break analyses, upgrading of auxiliary feedwater reliability and performance, procedures for operator action, and operator training.

Similar studies are now well underway for the Westinghouse and Combustion Engineering operating plants. These studies focus specifically on the predicted plant performance under different accident scenarios involving small break loss-of-coolant event and feedwater transients. Based upon analytically predicted system behavior, recommended guidelines for emergency operating procedures were developed and reviewed in the study. In addition, these studies include engineering assessments of the reliability of individual plant auxiliary feedwater systems and identification of dominant failure contributors and recommendations for corrective action. A similar study of the operating boiling water reactors is also in progress, but is at an earlier stage.

As the above studies developed firm conclusions and recommendations, implementing action was initiated. For example, the results of the Westinghouse and Combustion Engineering auxiliary feedwater system reliability assessments concluded that certain improvements were necessary. Individual plant licensees were then requested by letter to initiate corrective action or to propose design solutions for NRC staff review. Additional instructions were to be issued to licensees upon completion of other aspects of these reports.

## Follow-On and Interfacing Activities

It was planned that the task force would terminate its activities in late 1979, and therefore some of its activities were transferred prior to completion. Consequently, the task force concentrated on lead plants and established review guidelines and acceptance criteria that could be implemented by other NRR organizational elements.

As a result of the work performed in modeling small break and feedwater transients, longer range efforts were identified dealing with the procedures and systems available for core cooling under certain accident conditions, and with confirming analytical models through experiment or research programs. For example, plans are being implemented to conduct some small break loss-of-coolant tests at the Semiscale and LOFT facilities to obtain a better understanding of small break phenomena and to use the results to verify calculational techniques (see Chapter 11). Other recommendations in this regard are expected to result from the task force activities.

As noted previously, the task force concentrated on the immediate and near term actions necessary to assure the safe operation of operating plants. However, based on actions already completed, a number of items have been identified which warrant careful additional study. These actions have been and are continuing to be, documented for detailed assessment within the NRR organization.

IE BULLETINS ISSUED: APRIL—JULY 1979

| *Bulletin* | *Date Issued* | *Issued to* |
|---|---|---|
| 79-05 | April 1, 1979 | B&W plants |
| 79-05A | April 5, 1979 | B&W plants |
| 79-06 | April 11, 1979 | W and CE plants |
| 79-06A | April 14, 1979 | W and CE plants |
| 79-06B | April 14, 1979 | CE plants |
| 79-08 | April 14, 1979 | BWR plants |
| 79-06A (Rev. 1) | April 18, 1979 | W plants |
| 79-05B | April 21, 1979 | B&W plants |
| 79-05C | July 26, 1979 | B&W plants |
| 79-06C | July 26, 1979 | W and CE plants |

### Actions Directed by NRC Orders
(for immediate implementation)

(1) Reviewing and upgrading, as appropriate, auxiliary feedwater reliability and performance.

(2) Implement operating procedures for initiating and controlling feedwater independent of ICS.

(3) Implement hard-wired control grade reactor trip on loss of main feedwater and/or turbine trip.

(4) Complete analyses for potential small breaks and implement appropriate instructions for operator action.

(5) Provide at least one senior reactor operator, having TMI-2 training on B&W simulator, in control room.

### Longer-Term Actions Required by Orders

(1) Continue to review and upgrade performance of auxiliary feedwater system.

(2) Submit a failure mode and effects analysis of the integrated control system to the NRC.

(3) Improve the quality of the reactor trip following loss of main feedwater and/or turbine trip by upgrading to safety-grade design.

(4) Give continued attention to transient analysis and procedures for management of small breaks.

## TMI-2 LESSONS LEARNED TASK FORCE

In May 1979 an interdisciplinary team of engineers from the NRC Offices of Nuclear Reactor Regulation, Nuclear Regulatory Research, Inspection and Enforcement, and Standards Development began work on the identification and evaluation of safety concerns originating from the TMI-2 accident that required licensing actions. This team, the TMI-2 Lessons Learned Task Force, concentrated on issues separate from those specified in IE Bulletins and Commission Orders issued to operating plants early after the accident. The areas of interest to the Lessons Learned Task Force were applicable not only to operating plants but also to pending operating license (OL) and construction permit (CP) applications.

The task force was charged to review and evaluate investigative information, Commissioners' recommendations, ACRS recommendations, staff recommendations from NUREG-0560 ("Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company"), and recommendations from outside the NRC. In addition, the task force was charged to identify, analyze and recommend changes to licensing requirements and the licensing process for nuclear power plants based on the lessons learned. The scope of the task force assignment covered the following general technical areas:

(1) Reactor operations, including operator training and licensing.

(2) Licensee technical qualifications.

(3) Reactor transient and accident analysis.

(4) Licensing requirements for safety and process equipment, instrumentation, and controls.

(5) On-site emergency preparations and procedures.

(6) NRR accident response role, capability and management.

(7) Feedback, evaluation, and utilization of reactor operating experience.

### Two Phases of Work

The work of the task force proceeded in two phases. The first was the development of recommendations for short-term actions which, when combined with the requirements associated with implementation of the IE Bulletins on TMI-2—including the generic status reports issued by the task force and certain other changes in emergency preparations by licensees and operator training and licensing requirements—would ensure the safety of plants already licensed to operate and those to be licensed for operation in the near future. The first phase culminated with issuance in July 1979 of a report entitled "TMI-2 Lessons Learned Task Force: Status Report and Short-Term Recommendations" (NUREG-0578). The implementation of 23 short-term licensing requirements was directed for operating reactors by the Director of NRR in September 1979 based on a favorable ACRS review received in August.

**Metropolitan Edison staff members work in a room adjacent to the TMI control room to coordinate communication between the plant and local officials such as State police and fire departments.**

In the second phase of its work, the task force considered more fundamental questions in the design and operation of nuclear power plants and in the licensing process. The issues were grouped in four general categories: general safety criteria, system design requirements, nuclear power plant operations, and nuclear power plant licensing. A report entitled "TMI-2 Lessons Learned Task Force: Final Recommendations" (NUREG-0585) was issued in October 1979 to complete this phase.

The completion of these reports terminated the formal activities of the Lessons Learned Task Force, and its members returned to other duties. Two small groups among them, however, remained intact to make up the nuclei of interdisciplinary review teams which will see to the implementation of task force recommendations for new operating licenses and for operating reactors.

## Short-Term Recommendations

The decisionmaking process followed by the task force in determining which safety issues required short-term licensing action versus those that could be deferred for further evaluation by the task force or others was based on engineering evaluation and qualitative professional judgment of the safety significance of the various issues. In this regard, the task force selected items for "short-term action" if their implementation would provide substantial, additional protection required for the public health and safety. The task force recommendations presented in NUREG-0578 comprised 23 specific requirements. Each of these is discussed in detail in NUREG-0578, and a proposed two-stage implementation schedule is included as an appendix to that report. The 23 recommendations are briefly stated below.

(1) *Emergency Power.* For PWRs (pressurized water reactors), provide emergency power for the minimum number of pressurizer heaters required to maintain natural circulation conditions in the event of loss of off-site power, for power-operated relief valves and associated block valves, and for pressurizer level instrument channels.

(2) *Valve Tests.* For BWR (boiling water reactors) and PWR relief and safety valves, perform full-scale performance verification tests.

(3) *Valve Position Indication.* Provide direct position indication for PWR and BWR power-operated relief valves and safety valves.

(4) *Instrumentation for Inadequate Core Cooling.* Perform analyses and implement procedures and training for prompt recognition of low reactor coolant level and inadequate core cooling using existing or modified instrumentation; analyze and describe instrumentation for detection of low reactor vessel water level.

(5) *Containment Isolation Signals.* Provide containment isolation on diverse signals, review isolation provisions for non-essential systems and revise as necessary, and modify containment isolation designs as necessary to eliminate the potential for inadvertent reopening upon reset of the isolation signals.

(6) *Recombiner and Purge Penetrations.* For plants that have external hydrogen recombiners or purge systems, provide dedicated penetrations and isolation systems that meet the redundancy and single failure requirements of the Commission regulations.

(7) *Inerting BWR Containments.* Provide inerting for all Mark I and Mark II BWR containments. (Rulemaking required.)

(8) *Hydrogen Recombiner Capability.* Provide the capability to add, within a few days after an accident, a hydrogen recombiner system for post-accident hydrogen. (Minority view; rulemaking required.)

(9) *Systems Leakage.* Perform leakage rate tests on systems outside containment that process primary coolant and could contain high level radioactive materials. Develop and implement periodic testing and preventive maintenance programs.

(10) *Shielding Review.* Perform a design review of the shielding of systems processing primary coolant outside containment and assure that access to vital areas will not be unduly impaired due to radiation from these systems.

(11) *Automatic Initiation of the Auxiliary Feedwater System.* Provide means for automatic initiation

of all auxiliary feedwater systems; manual capability to initiate the auxiliary feedwater system from the control room must be retained.

(12) *Auxiliary Feedwater Flow Indication.* Provide indication in the control room of auxiliary feedwater flow for each steam generator.

(13) *Post-Accident Sampling.* Review and upgrade the capability to obtain and analyze samples from the reactor coolant system and containment atmosphere under high radioactivity conditions.

(14) *High-Level Radiation Monitors.* Provide high-range radiation monitors for noble gases in plant effluent lines and a high-range radiation monitor in the containment. Provide instrumentation capable of measuring and identifying radioiodine and particulate radioactive effluents in effluent lines under accident conditions.

(15) *Improved In-Plant Iodine Instrumentation.* Provide instrumentation for accurately determining in-plant airborne radioactive concentrations to minimize the need for unnecessary use of respiratory protection equipment.

(16) *Analysis of Transients and Accidents.* Provide the analysis, emergency procedures, and training to improve operator performance during a small break loss-of-coolant accident, to assure that the reactor operator can recognize and respond to conditions of inadequate core cooling, and to improve operator performance during transients and accidents, including events that are caused or worsened by inappropriate operator actions.

(17) *Shift Supervisor Responsibilities.* Promptly issue an operations policy directive that emphasizes the duties, responsibilities, and authority and lines of command of the control room operators, the shift supervisor, and the person responsible for reactor operations command in the control room.

(18) *Shift Technical Advisor.* Provide a shift technical advisor at each nuclear power plant who has a bachelor's degree or equivalent in a science or engineering discipline and with specific training in the plant response to off-normal events and in accident analysis of the plant.

(19) *Shift and Relief Turnover Procedures.* Review and revise plant procedures as necessary to assure that a shift turnover checklist is provided and required to be completed and signed by the oncoming and offgoing individuals responsible for command of operations in the control room.

(20) *Control Room Access.* Revise emergency procedures as necessary to assure that access to the control room under normal and accident conditions is limited to those persons necessary to the safe command and control of operations.

(21) *On-site Technical Support Center.* Provide an on-site technical support center, separate from the control room, for use by plant management, and technical and engineering support personnel in an emergency. This center shall be used for assessment of plant status, support of the control room command and control function, and in conjunction with implementation of on-site and off-site emergency plans. Communications links shall be established and the center shall be equipped as necessary for emergency engineering support activities.

(22) *On-site Operational Support Center.* Establish and maintain an on-site operational support center to which shift support personnel (e.g., auxiliary operators and technicians) other than those required and allowed in the control room report for further orders and assignment during an emergency.

(23) *Loss of Safety Function.* Require that a reactor be shut down if human errors lead to a complete loss of safety function (e.g., loss of emergency feedwater, high pressure ECCS, low pressure

**Portable communication units provided by the U.S. Forest Service were used to communicate between the TMI control room and various staff activities at the site. One such unit was manned on a 24-hour basis while periodic checks were made with the control room to record the status of the reactor.**

ECCS, containment, emergency power or other prescribed safety function), and allow the facility to return to power only after a public meeting and NRC approval of the remedial changes proposed by the licensee. (Rulemaking required.)

After considering comments on NUREG-0578 by various NRC offices, the Advisory Committee on Reactor Safeguards (ACRS), the industry and the public, the Director of Nuclear Reactor Regulation, with the approval of the Commission, added four requirements as follows:

(1) *Containment Pressure Indication (ACRS).* Provide wide-range continuous indication of containment pressure in the control room.

(2) *Containment Hydrogen Indication (ACRS).* Provide continuous indication in the control room of hydrogen concentration in the containment atmosphere.

(3) *Containment Water Level Indication (ACRS).* Provide continuous indication in the control room of containment water level.

(4) *Reactor Coolant System Vents.* To provide means for removing noncondensible gases, install reactor coolant system and reactor vessel head high point vents remotely operated from the control room.

## Implementing Short-Term Recommendations

The Commission directed that the staff proceed as soon as possible with implementation of all of the short-term recommendations, except those which were modified as set forth below, on the two-stage, 16-month schedule recommended by the task force.

In view of ACRS comments, the Director of Nuclear Reactor Regulation decided to delay any rulemaking action concerning inerting of BWR Mark I and II containments and provisions of hydrogen recombiner capability at operating plants until the final report of the task force had been issued. Final resolution of these matters is discussed in the section below covering the long-term recommendations of the Lessons Learned Task Force.

With respect to the recommendation to add a Shift Technical Advisor at each plant, the ACRS endorsed the concept but suggested that flexibility be maintained in implementation so that the objective could be reached through innovative approaches by individual licensees. For guidance, the task force prepared a statement of functional characteristics for the Shift Technical Advisor to be used in evaluating alternatives proposed by licensees.

The recommendation to review limiting conditions of operation to incorporate mandatory shutdown if human error causes loss of a safety function stimulated much interest inside and outside the staff. The Office of Standards Development has prepared a paper proposing such a new rule, but setting forth alternatives for achieving the same objectives as the task force recommendation.

On September 13, 1979, letters were sent to all operating nuclear power plants advising them that they should proceed with implementation of the recommendations of the Lessons Learned Task Force and the additional items resulting from ACRS comments and review by the Director of Nuclear Reactor Regulation. During the week of September 24, 1979, regional briefings were held to apprise reactor owners of these requirements. These meetings were followed by a 3-day series of meetings at NRC headquarters in Bethesda, Md. on some of the specific short-term requirements. Letters were also sent to applicants for construction permits and operating licenses instructing them to implement the short-term lessons learned.

All of the short-term "Category A" requirements deriving from conclusions of the Lessons Learned Task Force were conveyed to licensees of operating reactors by the end of 1979. It was expected that about two-thirds of these licensees would have met the Category A requirements by the end of January 1980, and the rest by May at the latest.

The approach adopted by NRC staff in seeking swift implementation of the short-term requirements allowed licensees to fulfill those requirements prior to NRC staff review. The approach necessitated a careful clarification of each requirement, and this was provided by means of regional as well as topical meetings and numerous discussions among NRC staff, the vendor-oriented owners' groups, and licensees.

The small number of action items that were not completed by the deadlines prescribed by NRC mainly involved problems of equipment availability. Some slippage is also permitted where it can be demonstrated that a severe impact on regional power supply would otherwise result.

## Long Term Recommendations

In contrast to the short-term recommendations in NUREG-0578, which were of a more narrow, specific, and urgent nature, the final report of the task force (NUREG-0585) dealt with safety questions of a more fundamental policy nature regarding nuclear plant operations and design and the regulatory process.

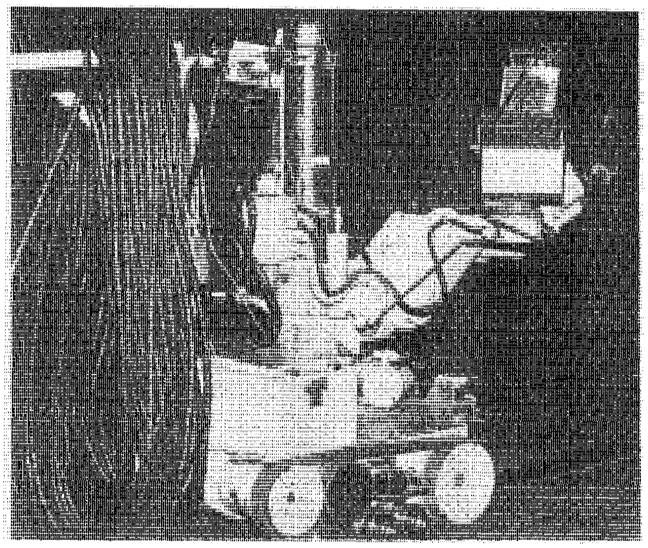
To stimulate discussion and speed the deliberative process, the task force developed a number of specific, final recommendations toward accomplishing the policy objectives and safety goals described in the report. The task force considered the modifications it outlined to be of fundamental importance to nuclear safety, and urged that immediate steps be taken to complete the deliberative process and initiate implementation of the recommendations.

Although the accident at Three Mile Island stemmed from many sources, the most important lessons learned fall in a general area the task force chose to call operational safety. This general area included the topics of human factors engineering; qualification and training of operations personnel; integration of the human element in the design, operation, and regulation of system safety; and quality assurance of operations. Specifically, the primary deficiency in reactor safety technology identified by the task force's review of the accident was the inadequate attention that had been paid by all levels and all segments of the technology to the human element and its fundamental role in both the prevention of accidents and the response to accidents. Thus, the policy recommendations and specific ideas in NUREG-0585 for stimulating and accomplishing change concentrated heavily on operations reliability and the associated design and licensing review measures that support or augment operations reliability.

The task force also devoted considerable attention to the basic mission of reactor regulation after Three Mile Island. It was not alone in these efforts; many people called for a clearer articulation of NRC's role and mission after March 28, 1979. The task force found that prescriptive and narrow licensing requirements only add to the quiltwork of regulatory practice and do little to directly address the nation's heightened concern for the safety of nuclear power plants. The task force called for the development of an articulate and widely noticed national nuclear safety policy with which to bind together the narrow and highly technical licensing requirements. Although the NRC and the President's Commission alluded to a more definitive safety policy by taking actions that in effect say, "no more Three Mile Islands," the task force urged that the feasibility and the adequacy of such a policy be critically examined and an opportunity provided for thorough and widespread public input.

More than 30 recommendations in 13 different areas were made by the TMI-2 Lessons Learned Task Force. In its review of these recommendations, the ACRS supported them in all 13 areas, offered advice on details of implementation and criteria employed in some of those areas, and added comments and recommendations on areas not addressed in the task force reports. Final recommendations of the task force and of the ACRS were being factored into the development of the NRC Action Plan for TMI-2 matters, which was in preparation at the end of 1979.

**Activity in the trailer office of NRC's Office of Inspection and Enforcement shortly after the accident, where personnel kept track of the environmental monitoring of radioactivity. At far right is Richard H. Vollmer of NRC's Office of Nuclear Reactor Regulation, who was designated in June to direct NRC's support activities related to recovery and cleanup operations at the TMI site.**



"Herman," a mobile manipulator borrowed from the Oak Ridge National Laboratory, proved too awkward for use at TMI-2. It was hoped that the robot could retrieve samples of radioactive water in the No. 1 Auxiliary Building, thereby reducing exposure to workers. The idea was abandoned when testing showed Herman's lack of pressure sensitivity presented the risk of flasks of contaminated water being dropped or crushed.

## Inspection and Enforcement Lessons

The NRC Office of Inspection and Enforcement (IE) also undertook an intensive investigation of the TMI accident but limited the scope of its inquiry to two sharply defined aspects of it: (1) the operational activities of the licensee from before the initiating event, about 4 a.m. on March 28 up to about 8 p.m. that evening, when primary coolant flow was reestablished by the starting of the reactor coolant pump; and (2) steps taken by the licensee to control the release of radioactive material to off-site environs and to implement its emergency plan, from the initiating event until midnight on March 30. These periods were selected for scrutiny because, in the judgment of IE, they encompassed those licensee actions which most significantly affected the course of the accident and its consequences.

In its report on this investigation, issued August 3, 1979, IE confirmed that the collective radiation dose to the general public resulting from the TMI accident constituted—as reported by the Ad Hoc Dose Assessment Group (made up of various Federal agencies) in its May 10, 1979 report—minimal risk to the health of the off-site population. At the same time, IE reported several inadequacies in the licensee's radiation protection activities inside the plant, as well as in the measuring of off-site radiation levels. These flaws, however, were not such as would cast doubt on or call for alterations in the conclusions of the ad hoc group.

The IE investigation also substantiated earlier conclusions regarding the underlying causes of the TMI accident and the factors that contributed to its severity. The six distinct areas of deficiency earlier identified as causative or complicating elements and confirmed by IE comprised equipment performance; licensee analysis of past transients and accidents; operator training and performance; equipment and systems design; the transmission of information (especially in the early phase of the accident); and the implementation of emergency planning. But what the IE report called "perhaps the most disturbing result" of the investigation was "confirmation of earlier conclusions that the Three Mile Island Unit 2 accident could have been prevented, in spite of the inadequacies" cited. The design, equipment, analyses, and procedures in place and in effect at TMI were, IE concluded, sufficient "to have prevented the serious consequences of the accident" if they had been allowed to function or had been adhered to as intended. For example, had the TMI operators permitted the ECCS to have its full effect, the damage to the core would most likely have been prevented (other examples were adduced in the report where a right action taken or a wrong one avoided could have significantly mitigated the consequences of the accident).

On the other hand, the IE report concedes, had certain equipment been designed differently it too could have prevented or diminished the effects of the accident. The investigation made it "difficult to fault only the actions of the operating staff." An undue preoccupation with the hazards of overfilling the reactor coolant system (that it was to be avoided "at almost any cost") was also evident in the decisions and actions of the operators, leading them to ignore prescribed procedures and to fail to respond to indications that the core was not properly cooled. Retraining of all licensed operators has now been required by NRC as well as an upgrading of procedures.

**Causes and Contributing Factors.** Soon after the shift came on at TMI-2 at midnight of March 27, 1979, the shift foreman and two auxiliary operators were engaged in transferring resin from a "condensate polisher" tank to a "resin regeneration" tank, on the secondary side of the plant. The chore was a carryover from the previous shift and was one with which plant personnel had encountered some difficulty. The staff thought the problem was a resin blockage in the transfer line and the foreman and auxiliary operators were trying to clear it. The IE report concluded that, "probably as a result of their efforts to clear the line," the plant underwent a total loss of feedwater flow, initiated by a loss of condensate flow and bringing about an almost simultaneous shutdown of the main turbine, at 37 seconds after 4 a.m., on March 28.

Ensuing events were found to be as described earlier in this chapter with certain noteworthy additions and conclusions. Among these was the finding that, about six minutes after the start of the accident, the pressurizer was completely filled with water and the reactor coolant system was, in fact, "solid," the condition which the control room crew strived to avoid throughout the crucial early hours of the accident by actions which delayed cooling of the core and compounded the consequences of the event. The IE report also indicated that "substantial fractions of the core were uncovered" by about 6:30 a.m. on March 28, although the fact went unrecognized by the operators and officials on the scene, and the high temperature readings in the core and the loops were considered too high to be realistic. The report also found that the operators interpreted the failure of the core flood tanks to inject a substantial portion of their volume into the reactor coolant system to be an indication that the core was covered, even though these tanks cannot be used for that purpose and are designed to supply water in the event of a large loss-of-coolant accident, which was not happening. With respect to the hydrogen explosion in the containment, the report observed that the release of this noncondensible gas from the reactor coolant system may have contributed to the later apparent success of the staff in collapsing the voids in the "A" loop of the reactor. That appearance of success in establishing natural circulation, despite the continued high temperatures in portions of the system, led the operators to believe that they had attained a reasonably stable condition by early afternoon of March 28.

Specific actions cited by the IE report as bringing about the extensive core damage that took place included: throttling the high pressure injection (ECCS) to a minimum during the first three and one-half hours of the accident; operating the reactor coolant pumps at pressures below procedural requirements (which led to greater loss of coolant through the stuck-open pressurizer relief valve); failure to isolate the relief valve after pressure continued to fall in the reactor coolant system, the drain tank disc had blown, and the sump pump operation all indicated that a large discharge of water from the system and the building was taking place; and failure to establish the conditions necessary for natural circulation in the system.

The report made note of other licensee actions which, while they did not directly affect the course of the accident as it actually unfolded, could have severely impaired the response of safety-related equipment if that course had taken another direction. Specifically, the disabling of the automatic startup features of the emergency diesel generators and the isolating of the core flood tanks early in the event constituted these kinds of lapses. The report was also critical of the communications provided during the event by the licensee, pointing out that persons assigned to furnish information off-site had concurrent duties related to management of the emergency. At the root of this and other problems, the report concluded, lay the misconception that even major accidents would be short-term events and that plans for mobilizing and communicating with off-site technical support over time, as an accident progresses, was not warranted as part of the emergency planning.

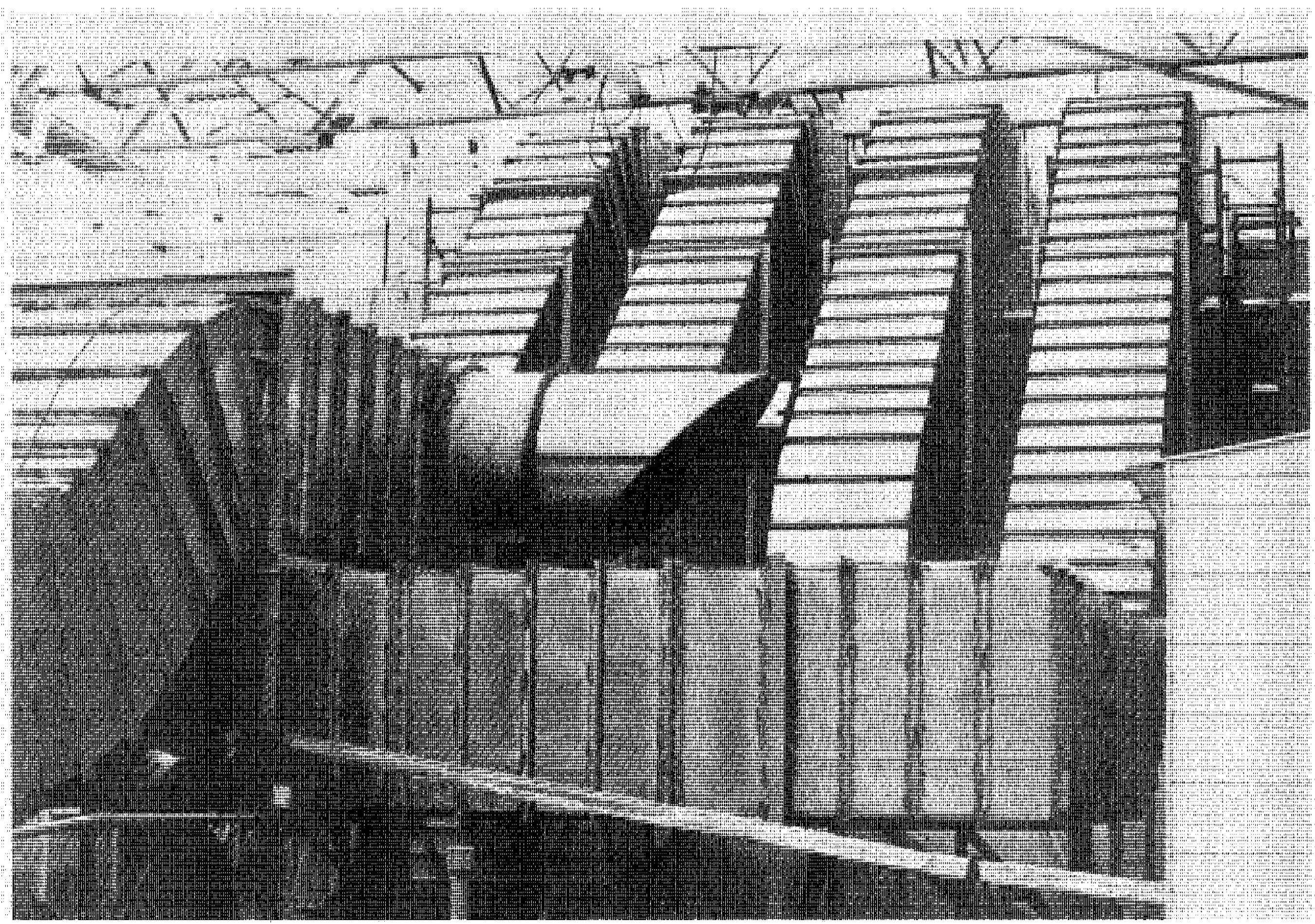
**Enforcement Action Proposed.** On October 25, 1979, the Director of Inspection and Enforcement notified the licensee for TMI that the IE investigation had revealed "numerous items of noncompliance" with NRC regulations on the part of the licensee. Six "violations"—the most serious breach of regulatory requirements—were alleged by IE, including serious weaknesses in the licensee's health physics program; control of maintenance activities; development and review of procedures; adherence to prescribed procedures; and audit activities. The licensee was cited for failure to operate the facility in accordance with the technical specifications approved and adopted for that particular plant, and for authorizing a surveillance procedure which placed certain valves in a status which rendered emergency feedwater unavailable on three separate occasions—including the last on March 28, when it was needed. Personnel training was also found insufficient, and record maintenance and in-house inspections as well.

The licensee was called upon to correct each of these deficiencies and departures from requirements and was notified that civil penalties were being proposed in the amount of $155,000, the legal maximum (although an assessment of $725,000 was justified for all violations identified).

**Task Force Urges Statutory Mandate on Lead Role.** The IE task force on leassons learned from TMI urged that IE be assigned, by statutory mandate, the lead role in NRC's emergency response in the future. Such a role flows from IE's *de facto* activity as the "principal contact with operating licensees," it was argued. It was also recommended that intra-office training be expanded and tightened surveillance of licensees be adopted. In the lead role for NRC emergency response, IE could give assistance to licensees in its response to an incident, as well as coordination to all NRC acitivities. It would also undertake training of other NRC offices regarding emergency preparedness and the respective responsibilities of those offices.

The task force also recommended that NRC create a new office to oversee training programs to upgrade the quality of inspectors and operations personnel, especially in the area of emergency response.

**ACRS Comment on IE Findings.** In a letter to Chairman Hendrie dated November 14, 1979, the Advisory Committee on Reactor Safeguards (ACRS) registered its view of the IE investigation and conclusions based on that investigation. Taking note of the limitation in scope of the IE study, the ACRS felt that the emphasis put by IE on the licensee's departure from technical specifications prior to the accident and from approved procedures during it resulted in too lit-



A new ventilation filtration system was installed on top of the auxiliary building of Unit 2 when the NRC determined that the existing system was not functioning satisfactorily after the accident. This system filters out airborne radioactive iodine and particulates before air is released to the environment.

tle consideration of other relevant factors. Examples of such factors taken from other investigations by NRC and others might be the peculiarities of a nuclear steam supply system that tended to inhibit recovery from interruption of normal operation or to confuse the operators by producing conditions and instrument readings not anticipated in the written procedures and, in general, by failing to convey clear, complete information to those in the control room. The ACRS concluded that the limited scope of the IE report tended to lead to a catalogue of violations and expressed its concern that the rationale behind the IE report would be perceived to be that a licensee's failure to follow accident procedures is automatically a violation. The ACRS noted that the procedures are prepared by the licensee and are not approved by NRC (although the licensee is required by NRC to follow them) and affirmed that such procedures cannot be so detailed as to allow for every accident situation. On the contrary, the ACRS declared, a deviation from conditions assumed in the framing of procedures may make it necessary to depart from those procedures. There is a question as to whether an operator who, using his best judgment, consciously takes an action at variance with procedures which in themselves may contain confusing or incorrect guidance is guilty of a violation. If this is the case, the ACRS affirmed its belief that it is "the wrong approach to protecting the public health and safety" in an emergency and that an operator, guided by written procedures, should be allowed to use his best judgment to deal with a problem. That judgment would be subject to *post-factum* appraisal by responsible parties, but it should not necessarily be deemed an error or a violation of regulations.

The ACRS found the IE report "less than satisfactory" for these reasons and recommended issuance of a consolidated report on the findings of the several NRC task forces investigating the TMI accident.

# The President's Commission

On April 11, 1979, President Carter issued an executive order (#12130) creating the President's Commission on the Accident at Three Mile Island and charging its members to "conduct a comprehensive study and investigation of the recent accident involving the nuclear power facility on Three Mile Island in Pennsylvania" and to include in their study the following elements:

- A technical assessment of the events and their causes.
- An analysis of the role of the managing utility.
- An assessment of the emergency preparedness and response of the NRC and other Federal, State and local authorities.
- An evaluation of the NRC's licensing, inspection, operation and enforcement procedures as applied to this facility.
- An assessment of how the public's right to information concerning the events at Three Mile Island was served and of the steps which should be taken during similar emergencies to provide the public with accurate, comprehensible and timely information.
- Appropriate recommendations based upon the Commission's findings.

The President appointed John G. Kemeny, the President of Dartmouth College and former chairman of the Mathematics Department at that institution, to the chairmanship of the Commission. Eleven other members were appointed, including a State Governor, a resident of Middletown, Pa., a labor union president, an industrialist, the president of a national society, an attorney, and five unversity professors. A full-time staff was engaged which eventually numbered over 60 persons; more than 30 separate staff reports were prepared and many of them published along with the report of the Commission, which was issued on October 30, 1979. In the course of its investigation, the Commission conducted 12 days of public hearings, and its staff compiled more than 150 separate depositions.

The report of the President's Commission was divided into three major sections: an overview, together with the principal specific findings of the Commission with respect to the causes of the accident; recommendations flowing from the findings and addressed to (1) the NRC, (2) the utility and nuclear industry, (3) the training of nuclear plant personnel, (4) certain technical considerations, and (5) the health and safety of plant workers and the general public; and a chronology of the accident with some further attribution of causality. Highlights of each section are provided below, together with the NRC's response to the Commission's recommendations and the President's statement about them.

## Findings and Judgments

The Commission affirmed at the outset of its report its basic conclusion that to prevent accidents as serious as TMI in the future it will be necessary to effect "fundamental changes" in the organization, procedures and practices, and, "above all, in the attitudes of the Nuclear Regulatory Commission and, to the extent that the institutions we investigated are typical, of the nuclear industry." The need for a change in attitude in NRC and in the industry is emphasized throughout the Commission's report. The Commission also declared at the start that its findings do not, "standing alone," require a conclusion that nuclear power plants are inherently too dangerous to continue in operation or that new ones should not be built, but neither would the Commission propose that the nation "move forward aggressively" in expanding commercial nuclear power uses.

In its discussion of causality, the Commission identified the root problems as being "people-related," rather than related to deficiencies in plant design or equipment (though these too were present and involved in the accident). The weaknesses identified were not only the "shortcomings of individual human beings," but problems of structure and communication "among key individuals and groups." The Commission asserted outright that the equipment involved at TMI was good enough that, "except for human failures, the major accident . . . would have been a minor incident." There was, the Commission found, a preoccupation with regulations as such, rather than with the safety they are supposed to promote, and that regulations as voluminous and complex as those in current effect were actually a negative factor with respect to safety. A particular distortion cited by the Commission was the concentration on large-scale or "worst case" hazards to the neglect of less consequential but more probable scenarios. Thus "the break of a huge pipe . . . [is] studied extensively and diligently," reflecting the attitude that if the worst accidents can be controlled there is little to fear from lesser events. The Commission pointed out that TMI was the result of a combination of minor equipment failures which is "likely to occur much more often than the huge accidents," and that successful handling of minor failures is usually going to depend more on quick and appropriate human reaction, in contrast to the necessarily automatic and extremely fast response of equipment to sudden, large-scale accidents. The Commission urged on the NRC and industry a newfound recognition that "human beings who manage and operate the plants constitute an important safety system."

On the subject of operator error at TMI, the Commission noted that the training of TMI operators (and that of reactor operators in general) was "greatly deficient" in that it did not prepare them for dealing with the extraordinary, with "something as confusing" as the conditions created by multiple equipment failures. Moreover, the TMI- 2 control room design was lacking in many ways, "the control panel is huge, with hundreds of alarms, and . . . some key indicators placed . . . where the operators cannot see them." (More than 100 alarms were in fact activated in the early stages of the accident, and, while the pressure and temperature in the reactor coolant registered in the control room, there was no indication to the operators that the combination of the two meant steam was forming.) Altogether the design of the room and its gauges and equipment gave little attention to "the interaction between human beings and machines" and "ignored the needs of operators during a slowly developing small break accident." Some members of the Commission favored a complete modernization of the control rooms of a TMI design, and all of them agreed that "a relatively few and not very expensive improvements in the control room could have significantly facilitated management of the accident." Thus the Commission found that, while inappropriate operator action was a major factor in the TMI accident, a number of deficiencies on the part of the utility, its suppliers, and the NRC—in training, in procedures, in control room design—and the failure to recognize these deficiencies and to learn from previous experience were among major contributing causes. Despite its findings as to the proximate and contributing causes of the TMI accident, and its judgment that the potential for such lapses could and should have been anticipated by various principals involved, the Commission expressed its conviction that, given all the deficiencies cited, "an accident like Three Mile Island was eventually inevitable."

Regarding the severity of the accident's impact on public health, the President's Commission determined that actual releases of radiation at TMI "will have a negligible effect on the physical health of individuals," and that the major health effect of the accident was mental stress. As to the possibility of an eventual TMI-radiation-induced cancer occurring among the exposed population, it found that there will be "either no case of cancer or the number of cases will be so small that it will never be possible to detect them." The mental stress experienced by people near the facility was "quite severe," however. The Commission ascribed this to several factors, especially the extensive speculation by public offícals during the first week of the accident on how serious it could become and whether evacuation of the population should or would take place. Concerning the effect of news media coverage during this time—its speculations, selections of items to cover, and general tone—the Commission decided that there was "overall, a larger proportion of reassuring than alarming statements in the coverage," and the news media "did not present only 'alarming' views, but rather views on both sides," although a "few newspapers . . . did present a more frightening and misleading impression of the accident." The severe stress was short-lived, the Commission concluded, and was worst among people living within five miles of the plant and in families with young children.

The damage to the facility itself was very extensive and, in the words of the report, the "ongoing cleanup operation at TMI demonstrates that the plant was inadequately designed to cope with the cleanup of a damaged plant. The direct financial cost of the accident is enormous. Our best estimate puts it in a range of $1 to $2 billion, even if TMI- 2 can be put back into operation. (The largest portion of this is for replacement power estimated for the next few years.) And since it may not be possible to put it back into operation, the cost could be much larger."

The Commission felt it an important part of their task to ascertain not only how bad the TMI accident was but how bad it might have been. It posed the question to itself, "What if one more thing had gone wrong?" Among the possibilities considered was whether a hydrogen or steam explosion could have breached the reactor vessel and also the containment. (That a nuclear explosion might have done so was not considered because, with the slightly enriched fuel used in a reactor, such an explosion is not a possibility.) Several scenarios potentially leading to the rupture of containment and release of massive amounts of radiation from the plant were studied. Of particular concern was the potential release of radioactive iodine which might enter the food chain. (There was only a trace off-site release of iodine from the actual TMI accident.) Some scenarios led to a better outcome than the actuality, and two or three would have resulted in more severe core damage than occurred and even a melting of the core. However, the Commission reported that—within the limits of current engineering knowledge of the interaction of molten reactor fuel with concrete, steel and water—its calculations show "that even if a meltdown occurred, there is a high probability that the containment building and the hard rock on which the TMI- 2 containment building is built would have been able to prevent the escape of a large amount of radioactivity." Being less than absolutely sure of this conclusion, the Commission urged more research into this vital but murky area of severe core damage and its worst plausible effects. The Commission averred that, whether or not TMI came close to becoming catastrophic, "accidents as serious as TMI should not be allowed to occur in the future," although "we must not assume that an accident of this or greater seriousness cannot happen again, even if the changes we recommend are made." The latter fact argues strongly for the need to be prepared to deal with the aftermath of such accidents.

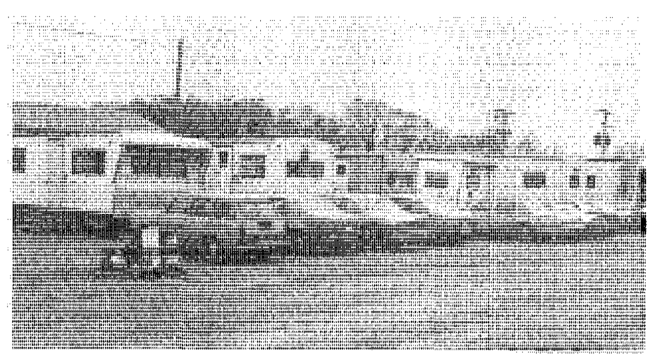
The next focus of Commission scrutiny, closely related to its last cited observation, was the matter of emergency preparedness among the various governmental elements involved at TMI. The Commission judged that the plans made by these agencies were not adequate and that their responses to the emergency were not satisfactory. It found problems associated with having multiple jurisdictions respond to a radiation emergency and an "almost total lack of detailed plans" in the local communities around TMI. The report noted that when "prompt defensive steps are necessary within a matter of hours, insufficient advance planning could prove extremely dangerous." The Commission advocated centralization of emergency planning and response in a single Federal agency which would maintain close coordination with State and local authorities and draw upon Federal and other expertise as the need arose. The report also criticized the NRC siting policy with respect to nuclear facilities and its requirement that reactors be located in a "low population zone," or LPZ, where protective action could be taken in the event of an accident. The Commission found "this concept implemented in a strange, unnatural and roundabout manner," with dimensions predicated on only a very serious hypothetical accident accompanied by very large doses to the population. (The NRC discontinued use of the LPZ in its siting requirements prior to publication of the Commission report.) The Commission proposed that a variety of possible accidents be considered in site evaluation—particularly the smaller scale accident with the higher probability of occurring—and protective action appropriate to each sector of the affected public be built into emergency plans for a facility. Also, State and local agencies must be ready to respond "once information is available on the nature of an accident and its likely levels of releases."

At TMI the emergency response "was dominated by an atmosphere of almost total confusion," the report stated, with "lack of communication at all levels."

On the subject of public and worker health and safety, the Commission noted that, in setting standards for worker exposure to radiation in licensed facilities, in its plant siting and other health-related decisions, the NRC "is not required to, and does not regularly seek" advice or review from other Federal agencies, such as HEW or EPA, concerned with health and radiation. Emergency plans did assign responsibilities to these agencies, as well as to DOE and NRC, in their response to the TMI accident, but, the Commission indicated, the plans were so poor that ad hoc arrangements had to be made and coordination improvised. In addition, the Commission found that the State agencies with responsibilities for public health did not have adequate resources "for dealing with radiation health programs related to the operation of TMI." Its recommendations on these matters appear later in this chapter.

On the issue of whether the public's right to information during the accident was well served, the Commission concluded that it was not. It found "serious problems with the sources of information, with how this information was conveyed to the press, and also with the way the press reported what it heard." Early in the accident the utility tended to minimize the hazards, according to the report, while later on the NRC "was the source of exaggerated stories." In particular, the Commission noted, "official sources would make statements about radiation already released... that were not justified by the facts—at least not if the facts had been correctly understood. And NRC was slow in confirming good news about the hydrogen bubble. On the other hand, the estimated extent of the damage to the core was not fully revealed to the public."

A separate problem concerned the way facts were presented to the press. It seemed that those who briefed the press either lacked the technical knowledge to explain the events transpiring or, when they did understand what was happening, they tended to speak in a technical jargon the press could not understand. Moreover, the report stated, "The press was further disturbed by the fact that, in order to cut down on the amount of confusion, a number of potential sources of information were instructed not to give out information. While this cut down on the amount of confusion, it flew in the face of the long tradition of the press checking facts with multiple sources." As mentioned, the Commission concluded that, with a few notable exceptions, the media "generally attempted to give a balanced presentation which would not contribute to an escalation of panic." (The Commissioner who was residing in Middletown, Pa., during the accident did not concur in that judgment; see "Supplementary Views," below.) A serious impediment in the conveying of accurate and complete information to the public was that "even personnel representing the major national news media often did not have sufficient scientific or engineering background to understand thoroughly what they heard, and did not have avail-



**NRC trailers at Three Mile Island used by the investigation team of the Office of Inspection and Enforcement and by the TMI support staff of the Office of Nuclear Reactor Regulation. Staff from these and many other NRC offices were on duty from time to time—many on a voluntary basis—during the first few weeks after the accident. Later, plans were made for office space in Middletown, Pa., for a long-term stay of some NRC staff.**

able to them people to explain the information." This applied particularly to the reporting of radiation releases, when numbers told the public nothing of the severity—or insignificance—of the releases. "Many of the stories were so garbled as to make them useless as a source of information."

Turning to an assessment of the NRC, the Commission took note that "when NRC was split off from the old Atomic Energy Commission, the purpose... was to separate the regulators from those who were promoting the peaceful uses of atomic energy." But the Commission found "evidence that some of the old promotional philosophy" persists in the regulatory practices of the NRC, and "evidence... that the NRC has sometimes erred on the side of the industry's convenience rather than carrying out its primary mission of assuring safety." In both the NRC's licensing and its inspection and enforcement activities, the Commission found "serious inadequacies."

The NRC licensing criteria and general approach were found exceptionable in several key respects:

- The application of a "single failure" criterion in the licensing process and the failure to analyze the consequences of a breakdown in two systems occurring independently (as happened at TMI).
- The inappropriately sharp distinction between "safety-related" components and "nonsafety-related," and the exemption of the latter from the stringent requirements applied to the former. (The report proposes instead "a system of priorities as to how significant various components . . . are for the overall safety of the plant.")
- The apparent assumption that plants can be made "people proof," and insufficient attention to operator training and operating procedures.
- The licensing of plants when relevant safety issues remain unresolved.
- Insufficient attention to the "ongoing process of assuring nuclear safety," as exemplified by NRC's categorization of a safety issue as a "generic problem," thereby relieving the licensee of responsibility for resolving the issue before licensing. (The report suggests there is evidence that "the labeling of a problem as generic may provide a convenient way of postponing decision on a difficult question.")
- A reluctance to apply new safety standards to previously licensed plants. (The report cites this as an instance of the "old AEC attitude" influencing NRC judgments and finds no evidence that "the need for improvement of older plants was systematically considered prior to Three Mile Island.")
- The tendency of a detailed and voluminous body of regulations "to focus industry attention narrowly on the meeting of regulations rather than on a systematic concern for safety." (The Commission felt that, in some instances, certain regulations may—because of the way rate bases are decided—have served to deter utilities and suppliers from initiating safety improvements.)
- The voluminous NRC inspection and enforcement manual, so extensive that "many inspectors do not understand precisely what they are supposed to do." The Commission also found that sometimes inspectors have had difficulty getting their superiors "to concentrate on serious safety issues," and also that incidents reported by licensees "tended to concentrate on equipment malfunction" while "serious operator errors have not been focused on."
- The lack of a systematic method for evaluating industry experience and to look for patterns that could warn of the presence of a basic problem, and a failure to use monetary fines to full effect.
- A heavy preoccupation in NRC with the safe operation of equipment to the neglect of "people-oriented" concerns, resulting in lack of attention to the operating procedures and "an almost total lack of attention" to the interaction between human beings and machines.

With respect to the NRC's response to the TMI accident, the Commission stated that it was "extremely critical of the role the organization played," citing the "serious lack of communication among the commissioners, those who were attempting to make the decisions about the accident in Bethesda, the field offices, and those actually onsite." The Commission questioned the suitability of the collegial structure of NRC, with five equipollent Commissioners, to manage an emergency and it found the "precise role" of the Commissioners unclear. In addition, the President's Commission observed that the "huge bureaucracy [NRC]... is highly compartmentalized with insufficient communication among the major offices," and it saw no "effective managerial guidance from the top," but rather "some of the old AEC promotional philosophy in key officers below the top." The Commission also cited the unnecessarily strict procedural rules within NRC which inhibited free communication among the NRC Commissioners and between them and the staff.

In conclusion, the President's Commission determined that, despite in-depth studies and critiques from within and outside the agency, there is "no well thought out, integrated system for the assurance of nuclear safety within the current NRC." For all of the reasons discussed, the Commission recommended a "total restructuring of the NRC," making the agency part of the executive branch, headed by a single administrator chosen from outside the NRC, with the freedom to "reorganize and bring new blood into the . . . staff. This new blood could result in the change of attitude that is vital for the solution of the problems of the nuclear industry." This and other Commission recommendations are treated below (see "Recommendations and Responses"), together with the NRC's response to each, as well as the President's statement on the Commission report.

With regard to the utility, the President's Commission felt that the necessary "management qualifications and attitudes" for conducting nuclear power operations were not given sufficient attention by the parent corporation whose subsidiary ran TMI. The Commission found "a divided system of decision-making within [the parent company, General Public Utilities Corporation] and its subsidiaries. While the utility has legal responsibility for a wide range of fundamental decisions, from plant design to operator training, some utilities have to rely heavily on the expertise of their suppliers and on the Nuclear Regulatory Commission. Our report contains a number of examples where this divided responsibility, . . . [as] in the case of TMI, may have led to less than optimal design and operating practices." The report notes that the design of the TMI control room "seems to have been a compromise among the utility, its parent company, the architect-engineer, and the nuclear steam system supplier (with very little attention from the NRC)." Operator training afforded the best example of the effects of divided responsibility, however. The utility has the legal responsibility for training operators and supervisors, but the TMI licensee did not have the expertise to conduct training by itself, so it contracted with the supplier of the nuclear steam system to do some portions of the training. The latter company had no responsibility for the quality of the total program, and coordination between it and the licensee was "extremely loose." The simulator employed in the program given by the reactor supplier differed "in certain significant ways" from the actual control console at TMI and, in any case, it was not programmed to reproduce the conditions faced by the TMI operators on March 28. The Commission believed that "the role that the NRC plays in monitoring operator training contributes little and may actually aggravate the problem." The NRC's "fairly routine licensing examinations" and limited spot-checking of requalifications exams (administered by the utility) "may be perpetuating a level of mediocrity," since the utility tends to equate the passing of the NRC exam with satisfactory operator training. The report was again very critical of operating procedures at TMI and the corresponding deficiencies they produce in the operators' training. Commission analysis of TMI management "raises the serious question of whether all electric utilities automatically have the necessary technical expertise and managerial capabilities for administering such a dangerous high-technology plant." Concluding that they do not, the Commission recommended higher standards of organization and management that a company must meet before receiving an operating license.

Recognizing that "recommendations as sweeping as ours will take a significant amount of time to implement," the Commission unanimously voted that "the NRC or its successor should, on a case-by-case basis, before issuing a new construction permit or operating license: (a) assess the need to introduce new safety improvements recommended in this report, and in NRC and industry studies; (b) review, considering the recommendations set forth in this report, the competency of the prospective operating licensee to manage the plant and the adequacy of its training program for operating personnel; and (c) condition licensing upon review and approval of the State and local emergency plans."

Expressing its "overwhelming concern about some of the reports" from other TMI investigations, and warning that proposed improvements carried out in a "business as usual" atmosphere will not suffice, the President's Commission concluded the Overview, stating:

"We believe that we have conscientiously carried out the mandate of the President of the United States, within our limits as human beings and within the limitations of the time allowed us. We have not found a magic formula that would guarantee that there will be no serious future nuclear accidents. Nor have we come up with a detailed blueprint for nuclear safety. And our recommendations will require great efforts by others to translate them into effective plans."

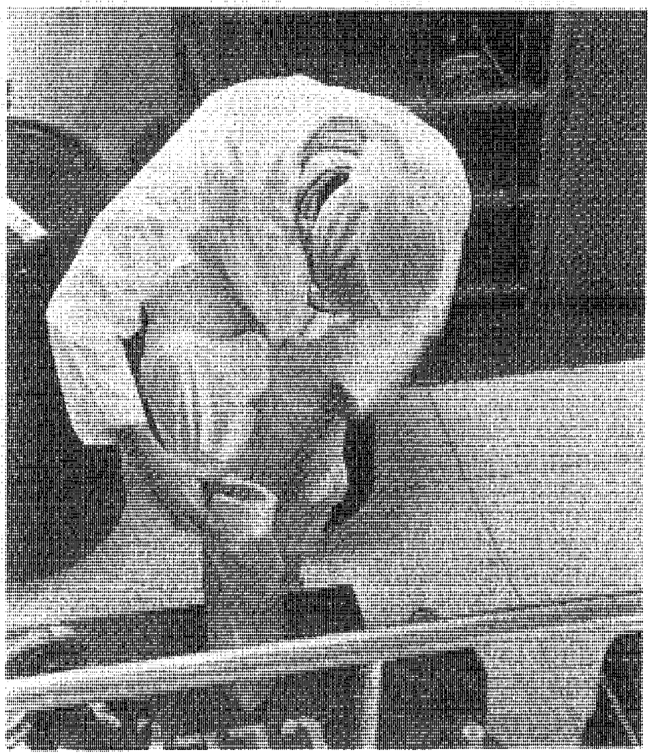
The Commission reaffirmed the need for fundamental change, charging that "unless portions of the industry and its regulatory agency undergo fundamental changes, they will over time destroy public confidence and hence, *they* [emphasis theirs] will be responsible for the elimination of nuclear power as a viable source of energy."

## Supplemental Views

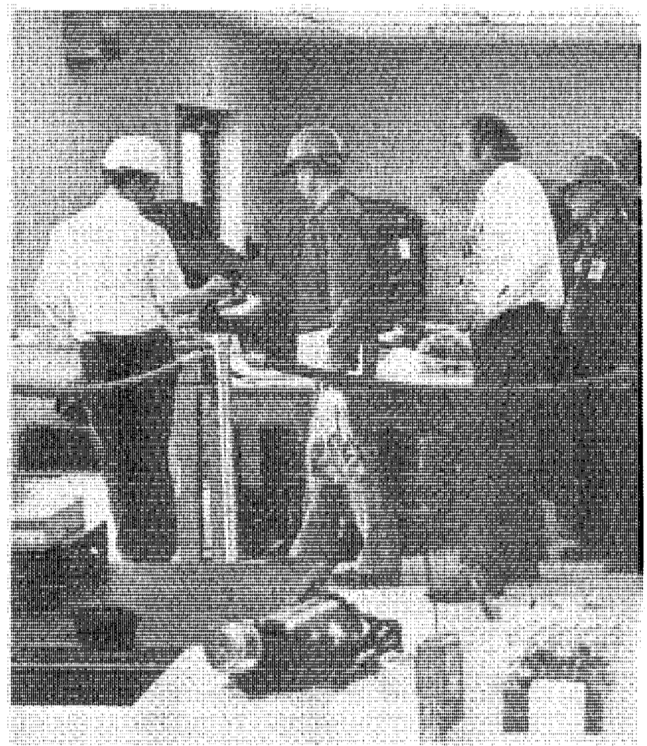
A number of Commissioners published comments of their own as supplements to the overall report of the President's Commission.

The Chairman and five other Commissioners cosigned a statement taking note of the fact that they had supported a recommendation, which failed of adoption by the full Commission, that "no new work authorization permits or constructions permits should be issued until such time as the NRC or its successor had adopted siting guidelines" consistent with the recommendation, which was adopted (unanimously), calling upon NRC to review its siting criteria (see "Recommendations and Responses," recommendation number 6, below).

Governor Bruce Babbitt of Arizona took up the matter of utility capability to operate nuclear power plants and gave his view that, while the "Commission



Here a workman prepares to enter a contaminated area by donning a suit of protective clothing. He is careful to tape his ankles to seal the area where the pant leg of the suit joins the overshoes.



Workmen about to leave the plant are carefully monitored with a pancake-type Geiger counter to insure that they have not been contaminated.

has clearly addressed the institutional shortcomings of the Nuclear Regulatory Commission, it has not addressed the institutional problems of the industry." The Governor expressed his belief that "this is one area where fewer entities with more depth and expertise might be justified for sake of public health and safety." The Commissioner also mentioned the possibility that certain facts known to TMI management on the first day of the accident had not been conveyed in timely fashion to the NRC and State officials, an issue which merits further investigation.

Commissioner Russell W. Peterson, President of the National Audubon Society, reaffirmed his endorsement of the recommendation noted above as having the support of six Commissioners, namely, that no new limited work authorizations or construction permits should be issued until the NRC siting requirements were changed. The Commissioner also felt that the President and Congress should "involve highly technically qualified critics of nuclear energy safety" in the continuing appraisal of nuclear safety called for by the Commission. He also urged serious study of nuclear waste disposal. Commissioner Peterson finally stated his conviction that a "much more serious accident" than TMI was going to occur somewhere at some time, because of the complexity of the technology and human limitations, and therefore he called for the development by the government of a "strategy which does not require nuclear fission energy."

Commissioner Thomas H. Pigford, Chairman of the Department of Nuclear Engineering at the University of California (Berkeley), issued a lengthy supplement to the report, setting forth, among others, the following observations:

- The report's stress on the need for more emphasis on people and less on equipment has obscured the "very important fact" that, despite the crucial errors of people, the safety equipment did indeed function to achieve its purpose; and despite the failures of equipment—the stuck valve and the leaks in the gas vent system—the overall system was good enough that, absent the effects of human error, the accident would have been a minor incident. Staff analyses show that even if all the fuel cladding had oxidized and even if fuel melting or meltdown had occurred, the containment would have stood up and the public would have been protected.
- Systems of equipment at TMI performed better than expected; earlier assumptions would have led to far greater core damage and radiation release to the containment than what actually occurred.
- TMI has revealed to all a number of remedies and improvements to be made, but there "seems to be some unwillingness to recognize that many of these remedies are already being implemented.

The NRC and the nuclear industry have taken and are taking steps . . . The problem with 'attitudes' emphasized in the Commission's report must refer largely to pre-TMI attitudes."

- More emphasis is needed on analysis of and planning for small break accidents, but "the possibility of an accident of this type was known and had been analyzed and predicted prior to the TMI-2 accident." Thus the facts of the present investigation provide no basis for concluding that reactors are unsafe.
- Since the attitudes of various persons and groups were not directly examined prior to the TMI-2 accident, valid conclusions can only be drawn on "actions taken, i.e., problems addressed and not addressed, regulations issued and complied with, and the occurrence of events that reflect upon the adequacy of these processes." It is "more constructive to assume that attitudes are symptomatic of . . . forces at work in the system, and it is those forces which must be addressed." It is the apparent failure of the system to assimilate lessons from plant experience and to incorporate up-to-date technology—in control room design, for example—that constitute "a more manageable and appropriate focus for the overall conclusion of this Commission. I believe that such technology is . . . or will be used by the industry and that changes . . . will be effected, not merely to satisfy critics or to demonstrate attitudinal penitence, but on the basis of sound judgment resting on sound data."
- The NRC must deal with the question of how much cost and delay is justifiable to realize a given increment in safety, and efforts to balance costs and benefits should not be considered evidence per se of a promotional philosophy. Both overreaction and inaction in this area carry social costs which must be weighed.
- While it is "confusingly" referred to as a "single failure" criterion, the NRC licensing process applies a criterion which assumes at least three failures: any credible component failure (1) in which all internal or all external power supply is lost, with (2) the additional failure of a single active component which (3) is the component whose failure causes the most serious aggravation of the accident.
- In the analysis of postulated accidents, there is no assumption that an active "nonsafety-related" item will not fail; it was not a preoccupation with a safety-related item list that proved inadequate in the analysis of TMI, but a failure to take into account lack of operator training and deficient operating procedures and practices.
- The finding that there is no systematic backfitting review of older plants "appears to take too little account" of NRC's Systematic Evaluation Program (SEP), initiated more than three years ago; progress in some areas, such as upgrading emergency plans, does appear to have been somewhat slow.
- The Commission's appraisal of NRC inspection and auditing of licensee compliance "calls for NRC to do more of what it already does and to do it better." Resident inspectors have been at some plants for more than a year, and unannounced on-site inspections "appear to be so frequent as to be commonplace." It is "clearly impractical" for the NRC to undertake substantial independent testing of construction work and cease to rely on testing done by the utility.
- A lack of quantified safety goals is a major problem in the NRC regulatory rationale, and its failure to set priorities leads to a disproportionate commitment of resources and efforts to sometimes marginal concerns. A large portion of the NRC management and staff are lacking practical experience in designing and operating the equipment they regulate, and too many requirements are unsupported by valid technical backup and value-impact analysis (an "overwhelming emphasis on conservative models and assumptions"). There is an insufficient exchange of information between NRC and industry because of the "adversary approach" existing between them, and NRC does not carry out the kind of systematic analysis of operating data that would disclose significant trends and patterns.
- There was not sufficient time allowed for a careful review of the President's Commission staff reports on which Commission findings were based (some were still incomplete when the final report was issued), and there were "several parts of some key staff reports with which I cannot agree, particulary the staff report on the NRC." There was unqualified acceptance in that report of testimony which was unconfirmed and uncorroborated, "an indicator of insufficient balance" in the staff investigation of the NRC. The staff report also "relies to a considerable extent upon excerpts from a book," without establishing the author's qualifications or taking his testimony. The Commissioner stated, "In my view, the . . . book does not express a comprehensive, accurate and balanced knowledge of the NRC and of the nuclear industry."
- Criticism of the NRC "should not obscure the central issue that primary responsibility for nuclear safety lies with the utility, shared to a large extent with the equipment suppliers and the architect-engineers. This also reflects my view of the responsibilities for the TMI-2 accident."

Commissioner Anne D. Trunk, a resident of Middletown, Pa., located about three miles from the TMI station, took exception to the Commission finding regarding the news media's treatment of the accident and its effect on the mental state of the people living near the facility. (Mental stress was identified by the Commission as the "major health effect" of the accident.) Commissioner Trunk, affirming that she spoke for herself "and a majority of her circle of citizens who lived through the TMI accident," stated:

"The report concluded that the errors and sensationalism reported by the news media merely reflected the confusion and ignorance of the facts by the official sources of information. It further concluded that the press did a creditable ('more reassuring than alarming') job of news coverage.

"In fact, these conclusions are not generally supported by the staff reports. There were reliable news sources available. Too much emphasis was placed on the 'what if' rather than the 'what is.' As a result, the public was pulled into a state of terror, of psychological stress."

The Commissioner called for a self-evaluation on the part of the news media. She also noted that she could not support a moratorium on the issuance of new construction permits because "it was not shown how this could result in a safer plant at TMI nor attain higher standards of safety and performance by the industry." Instead, the Commissioner recommended a defined period within which the parties concerned would be charged to act upon the Commission's recommendations, and a separate probationary operating period for the licensee at TMI.

## Recommendations and Responses

Starting below and in the pages that follow, the specific recommendations of the President's Commission—concerning the NRC, the utility and its suppliers, the training of operating personnel, a technical assessment, and both worker and public health and safety—are set forth in the left-hand column, with the response of the NRC to each recommendation set forth in the right-hand column.

Stacks of lead ingots were sent to the TMI accident site from industry groups and national laboratories responding to a general request from the NRC. The lead was used in various parts of the plant for radiation shielding during observation and measurement taking. In the weeks following the accident, however, it was determined that site radiation levels did not require all of the lead and much of it was returned to the donors.

In forwarding the NRC comments to Dr. Frank Press, Director of the Office of Science and Technology Policy, Executive Office of the President, Chairman Joseph Hendrie expressed a number of general comments on behalf of the Nuclear Regulatory Commission. (Two NRC Commissioners added supplementary remarks, cited at the close of this section.) The Chairman stated that, from NRC's own reviews of the accident, "we have generally found that the actions recommended by the President's Commission in the areas of human factors, operational safety, emergency planning, nuclear power plant design and siting, health effects, and public information are necessary and feasible." He affirmed that changes taken and intended by the NRC are in conformity with the recommendations of the President's Commission, and that some changes under consideration would go beyond those recommendations. Of particular importance, the Chairman noted, was the need for "prompt and positive assurance that the technical and management competence of all licensees is sufficient to operate nuclear power plants safely and to respond effectively to emergencies." Expeditious action would be taken in this area. Reporting that four of the five NRC Commissioners felt that effective reform could and should be accomplished within the existing agency, the Chairman also conveyed disagreement "with the overall thrust of the President's Commission recommendations to lessen the role of NRC in responding to emergencies and providing emergency information to the public." Estimating that it would take several months to develop the new or improved safety objectives and detailed criteria for implementing them, the Chairman disclosed that "we have decided that new plants will not be licensed until we have developed the required criteria."

## PRESIDENT'S COMMISSION (PC) RECOMMENDATIONS ON THE NRC

| PC RECOMMENDATIONS | NRC RESPONSES |
| --- | --- |
| (1) NRC SHOULD BE RESTRUCTURED AS A NEW INDEPENDENT AGENCY IN THE EXECUTIVE BRANCH. The present five-member Commission should be abolished, and a single administrator appointed by the President, with advice and consent of the Senate, to serve at the pleasure of the President. The administrator should be from outside NRC and should be given substantial discretionary authority in managing the agency. | (1) Four of the five Commissioners felt that the objectives of the President's Commission could be accomplished by reforms effected within the existing structure. It is desirable to have the statutory authority to delegate management responsibilities to a single Commissioner in event of an emergency. Clarifications in the law could remove ambiguity of the Chairman's authority, as well as that of the Executive Director for Operations. NRC has adopted a new "policy planning program guide" mechanism and is studying new modes of Commission involvement in developing key safety policy. |
| * * * | * * * |
| (2) AN OVERSIGHT COMMITTEE ON NUCLEAR REACTOR SAFETY SHOULD BE ESTABLISHED. Its purpose would be to examine, on a continuing basis, the performance of the agency and the industry in resolving important public safety issues related to nuclear power plants and in exploring the overall risks of nuclear power. Membership—up to 15 in number—would be drawn from the fields of public health, environmental protection, emergency planning, energy technology and policy, nuclear power generation, and nuclear safety; one or more State governors and members of the general public would serve on the committee, which would report to the President and Congress annually. | (2) Although this call for an oversight committee is tied to the recommendation for a new executive branch agency, this proposal should be examined on its own merits. Such an oversight or public advisory committee might contribute to the interaction among the Federal Government, States, utilities, public interest groups. and the general public on the controversial issues related to nuclear power. |
| * * * | * * * |

**PC RECOMMENDATIONS**

(3) THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) SHOULD BE RETAINED, IN A STRENGTHENED ROLE, TO CONTINUE PROVIDING AN INDEPENDENT TECHNICAL CHECK ON SAFETY MATTERS. The staff of the ACRS should be augmented, and its public health expertise especially improved. The ACRS would choose which licensee applications to review, and it would have a statutory right to intervene in hearings as a party. It should be authorized to raise any safety issue in a proceeding, give reasons and arguments for its views, and require formal response by the agency to its submissions. Any ACRS member would be exempt from subpoena in any proceeding in which he has not previously appeared voluntarily or made an individual written submission. ACRS should have similar rights in rulemaking proceedings and power to initiate such a proceeding to resolve any generic safety issue it wishes.

* * *

(4) INCLUDED IN THE AGENCY'S GENERAL SUBSTANTIVE CHARGE SHOULD BE THE REQUIREMENT TO ESTABLISH AND EXPLAIN SAFETY-COST TRADE-OFFS. Where additional safety improvements are not clearly outweighed by cost considerations there should be a presumption in favor of the safety change. The agency should be relieved of "any unnecessary responsibilities that are not germane to safety." In particular, operator and supervisor licensing should be upgraded, and accreditation of training centers required; a definition of "safety matters" should be formulated which is broader than the present inventory of "safety-related items"; an emphasis on examination of overall plant design and performance, from a systems engineer's standpoint, is needed, with attention to multiple failure potential, control room design, instrumentation; research with a broad scope that includes public health and which exploits all scientific knowledge available should be coordinated with the regulatory process.

* * *

**NRC RESPONSES**

(3) NRC endorses a strengthened role for the ACRS and the recently initiated ACRS Fellows Program should reinforce its analytic resources. But the strength and value of the independent ACRS reviews derives from the collegial interaction of its members; adding staff beyond reasonable needs will not contribute much to that strength. NRC has supported legislation which would enable the ACRS to choose applications for review. The proposed right to intervene may not be appropriate for a part-time advisory body; it would require a new ACRS legal staff and active involvement in hearings could severely compromise the independence and collegial nature of the committee. The ACRS can now recommend rulemaking to the NRC, but whether it should be able to mandate a proceeding needs and will be given further consideration. In general, NRC agrees that ACRS views warrant prompt response by the NRC staff. Comments on the matter have been requested from the ACRS.

* * *

(4) NRC has not, in the past, clearly articulated its policy on the effect of costs on safety decisions. Some safety-cost tradeoffs are presently authorized, e.g., value-impact analyses performed for proposed regulatory requirements or in research planning. A better articulation of NRC policy is needed. It is believed that benefits and detriments can be sufficiently quantified to aid in decision-making, and it is agreed that, in general, some sort of safety-cost tradeoffs are at least implicit in a regulatory system that concedes that a goal of zero risk is impossible of attainment. The reality should be made explicit. But NRC is in complete accord that in all comparative judgments of this kind there should be a presumption in favor of safety. NRC will seek views of the Congress, other agencies and the public in developing an explicit policy statement. Legislation may eventually be desirable for the definitive policy expression. Legislation would be required to divest NRC of its non-safety responsibilities, and the prospect raises problems in the area of nuclear exports. The Commissioners are not in agreement now on the best course of action. As to operator and supervisor training, a study is under way as to the options for NRC involvement and operator licensing requirements are being upgraded. The broadening of the definition of safety-related matters is a priority, including both equipment and human factors, and the interaction of safety- and non-safety grade equipment is under study. Control room design, overall plant design, and safety research are all undergoing reevaluation, and flexibility in assuring maximum application of scientific knowledge will be pursued.

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**PC RECOMMENDATIONS** / **NRC RESPONSES**

(5) RESPONSIBILITY AND ACCOUNTABILITY FOR SAFE POWER PLANT OPERATIONS, INCLUDING THE MANAGEMENT OF A PLANT DURING AN ACCIDENT SHOULD BE PLACED ON THE LICENSEE IN ALL CIRCUMSTANCES. Thus the competence of licensees to meet this obligation must be assured, and the agency should impose higher standards or organizational and managerial capability, especially confirming the "integration of decisionmaking" in the company licensed to construct or operate a plant; the necessary range of expertise; financial capability; quality assurance; operator and supervisor performance; surveillance and maintenance practices; and thorough analysis and reporting of unusual events.

(5) NRC fully agrees and has begun actions to upgrade standards and requirements to assure technical competence of licensees. The objective will be "to minimize accident occurrence and maximize proper response to accidents." Licensee performance will be subject to more frequent periodic reviews, involving licensee's top management. More immediate and decisive action is being contemplated (see response to recommendation 2 under "Commission Recommendations on the Utility," below).

* * *

(6) THE AGENCY SHOULD BE REQUIRED, TO THE MAXIMUM EXTENT FEASIBLE, TO LOCATE NEW POWER PLANTS IN AREAS REMOTE FROM CONCENTRATIONS OF POPULATION. Siting determinations should be based on technical assessments of various classes of accidents that can take place, including those involving releases of low doses of radiation.

(6) The NRC Siting Policy Task Force report under current review by the Commissioners recommends similar changes and goes beyond those proposed. Radiation releases from small accidents will be considered in appraising these recommendations. For the past five years, the Standard Review Plan has excluded sites with high population densities, but operating plants built before then may call for added design features, power reduction, or shutdown.

* * *

(7) THE AGENCY SHOULD BE DIRECTED TO INCLUDE IN ITS LICENSING REQUIREMENTS PLANS FOR THE MITIGATION OF THE CONSEQUENCES OF ACCIDENTS, including the cleanup and recovery of the contaminated plant. The agency should be directed to review existing licenses and to set deadlines for accomplishing any necessary modifications.

(7) The NRC Lessons Learned Task Force recommends similar action but goes beyond that proposed. The staff has already implemented new requirements for system leakage and shielding and has recommended operator training in core-melt accident mitigation, as well as NRC rulemaking on required design features to provide such mitigation.

* * *

(8) BEFORE ISSUING A NEW CONSTRUCTION PERMIT OR OPERATING LICENSE, THE NRC SHOULD DO THE FOLLOWING ON A CASE-BY-CASE BASIS: assess the need to introduce the safety measures recommended by the President's Commission and in NRC and industry studies; review the competence of the prospective licensee to manage the plant and the adequacy of operating personnel training; and make licensing contingent upon review and approval of State and local emergency plans.

(8) NRC has decided that new plants will not be licensed until the required criteria have been developed. The NRC will: (a) review and correlate recommendations of the President's Commission, the ACRS, the Congress, its own inquiries and others; (b) draw up safety objectives corresponding with those recommendations; (c) develop plans by which to realize those objectives by action affecting NRC structure and procedure or by requirements placed on licensees; (d) impose such requirements on operating plants; and (e) impose such requirements on plants under construction. Deadlines will be associated with the last two steps. Operator training will, as noted, be upgraded, and a rule requiring approval of State and local emergency plans prior to plant operation is being considered.

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**PC RECOMMENDATIONS**

(9) THE AGENCY'S AUTHORIZATION TO MAKE GENERAL RULES AFFECTING SAFETY SHOULD INCLUDE THE FOLLOWING REQUIREMENTS: that a public agenda be developed according to which rules will be formulated; that the agency set deadlines for resolving generic safety issues; that existing rules be reevaluated periodically and systematically; that rulemaking procedures be adopted which give interested persons a meaningful opportunity to participate, which ensure careful consideration and explanation of proposed rules, and which provide for the application of new rules to existing plants. In particular, proposed rules should be accompanied by analyses of the issues involved and identification of relevant technical material. Interested parties and organizations should have sufficient opportunity to assess and refute technical evidence and findings, and final rules should be fully explained, with responses for principal comments received. If needed, interim safeguards for operating plants affected by generic safety rulemaking should be imposed, and the possible need for retroactive application of new safety requirements to operating plants should be examined.

**NRC RESPONSES**

(9) NRC publishes an agenda of rulemaking petitions, a report of regulations under development, advance notices of proposed rulemaking in major actions, and proposed rules for comments. Analyses and discussions of these are made public, and public meetings or hearing are held in cases of special importance. The means for the public to petition NRC to issue, revise or withdraw a rule are provided, and proposed and final rules sent to NRC Commissioners for consideration are accompanied by a staff paper dealing with the relevant concerns, alternatives, benefits and detriments, and comments received and their resolution. The process is being reevaluated for clarity, sufficiency of public attention, effectiveness in resolving safety issues. In practice all new rules call for a judgment on backfitting to existing plants, but NRC is now considering including the practice in the regulations. Deadlines for the resolution of unresolved safety issues were set more than a year ago, and these issues are, by definition, the most significant of the generic issues. Other such issues will be addressed by priority based on safety significance. The review of NRC regulations usually has followed some specific event, such as a research result, a petition for rulemaking or new technology, with some exceptions in the area of transportation and safeguards. This will now change, with plans for an initial review of regulations by June 1980, completion of relevant rule changes by 1982, and completion of a systematic review of all safety regulations by 1984. The review cycle will be repeated thereafter every five-to-seven years.

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**PC RECOMMENDATIONS**

(10) LICENSING PROCEDURES SHOULD FOSTER EARLY RESOLUTION OF SAFETY ISSUES BEFORE MAJOR FINANCIAL COMMITMENTS IN CONSTRUCTION CAN OCCUR. The Commission recommends: (a) the reduction of duplicative consideration of issues in the several stages of a plant's licensing history by assignment of particular issues (such as need for power) to some single stage of the proceedings; (b) resolution of issues that recur in many licensing actions by rulemaking; (c) combining construction permit and operating license hearings whenever plans can be made sufficiently complete at the construction permit stage; (d) an initial adjudication of a license application and appeal to a board whose decision would be final, with no provision for subsequent appeal within the agency. Both adjudicators and appeal boards would have a clear mandate to pursue any safety issue it wished to; (e) the creation of an "Office of Hearing Counsel" in the agency to participate in formal hearings as "an objective party, seeking to assure that vital safety issues are addressed and resolved," and empowered to appeal

**NRC RESPONSES**

(10) The objective underlying this recommendation is shared by NRC, but it cannot make specific comment on it at present. A report is pending from a special advisory committee on its study of an NRC rule which permits plant construction during adjudication. The report may also have a bearing on the NRC practice of permitting discrete, specific issues to remain open up to the operating license stage and even beyond. (It can happen that a safety issue cannot be settled without additional information, but that such information can be obtained by research, even as construction proceeds.) On November 2, 1979, the NRC suspended its rule by which reactor licenses become immediately effective following a favorable initial decision by a licensing board. No license will become effective until the Commission itself has had the opportunity to determine the relevance of TMI-related issues to the case. The assignment of single issues to specific stages of the process, and possibly combining construction permit and operating license hearings, are matters in which NRC's authority is unclear (the latter step would require new

**PC RECOMMENDATIONS**

"any adverse licensing board determination to the appeal board;" and (f) a deadline on the resolution of any specific safety issue left open in a licensing proceeding.

**NRC RESPONSES**

statutory power). Even though it may be possible to combine the two kinds of hearings, there must still be a vehicle for verifying the design details, and that must necessarily be done late in construction when engineering of the design is complete. Also, new information affecting the early construction permit decision can arise at any time. It is current NRC practice to segregate recurrent issues for generic resolution whenever possible. The recommendation that appeal board decisions be made final NRC dispositions of applications for licenses would have the effect of removing the Commissioners (or Administrator) entirely from a major dimension of nuclear regulation. As to the mandate to pursue safety issues, the boards already have independent authority to pursue "serious matters" and the exercise of the right is no longer qualified by "sparingly" or "in extraordinary circumstances." The proposal that a new Office of Hearing Counsel be created has a purpose which is not entirely clear, but it might serve as an alternative to other devices for broadening public participation, such as intervenor funding, and merits consideration. Plant-specific safety issues left open at the time of licensing are now carried forward with clear deadlines as conditions on the operating license; NRC will consider whether it should also be conditioned with deadlines for resolution of relevant unresolved safety issues.

* * *

(11) THE AGENCY'S INSPECTION AND ENFORCEMENT FUNCTIONS MUST RECEIVE INCREASED EMPHASIS AND IMPROVED MANAGEMENT. The Commission recommends: (a) an improved program for the systematic safety evaluation of plants to assess compliance with requirements, to determine whether new requirements should apply, and to identify new safety issues; (b) systematic assessment of reactor operating experience to reveal any pattern of abnormal activilty and provide a measure of overall rises or declines in safety and a base for specific improvements; (c) substantial penalties be levied on licensees who fail to report new safety-related information or violate rules proscribing unsafe practices; (d) improved inspection and auditing of licensee compliance with regulations and unannounced onsite inspection; (e) periodic intensive and open review of each licensee's performance in meeting license requirements and regulations; and (f) agency adoption of criteria for revocation of licenses, for sanctions short of revocation (e.g., probation), and for requiring immediate plant shutdown or other operational safeguards.

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(11) In 1977, NRC set up a Systematic Evaluation Program (SEP) whose first phase called for review of conditions at 11 older plants. Extension of this program to all operating plants is being considered. The Interim Reliability Evaluation Program (see recommendation 4 under "Technical Assessment," below) is also under consideration. In July 1979, NRC created the Office for Analysis and Evaluation of Operational Data to give broad coordination to major program offices' assessment of operating experience; licensees have also been required to establish operating experience evaluation groups and to assess experience of other facilities than their own. The industry has created similar groups. The inspection and enforcement staff is being augmented with plant systems analysts to conduct independent technical evaluations and followup of licensee events, transients, and inspection findings. Potential generic problems and operating experiences will be conveyed promptly to licensees through Bulletins, Circulars, and Information Notices. Legislation to increase civil penalties imposed by NRC is pending before Congress, and the possible use of probation status is under review within NRC. The resident inspector program begun in 1977 has been expanded; at least two resident inspectors will be assigned to each site in fiscal year 1981. Licensee performance evaluations combined with

PC RECOMMENDATIONS

NRC RESPONSES

assessments of licensee management control systems by the Performance Appraisal Team will identify marginal utility operations and provide prompt correction. Unannounced inspections are carried out by NRC, but the need for these in light of the expanded resident inspector program is problematic.

## PRESIDENT'S COMMISSION (PC) RECOMMENDATIONS ON THE UTILITY AND ITS SUPPLIERS

PC RECOMMENDATIONS

(1) THE NUCLEAR INDUSTRY MUST DRAMATICALLY CHANGE ITS ATTITUDES TOWARD SAFETY AND REGULATIONS; IT MUST SET AND POLICE ITS OWN STANDARDS OF EXCELLENCE, to ensure the effective management and safe operation of nuclear power plants. It should develop standards for management, quality assurance, and operating procedures and practices, and it should conduct independent evaluations (perhaps through the Institute of Nuclear Power Operations). It should gather and analyze all power plant operating experience systematically, communicate information speedily to affected parties, and make needed changes on realistic deadlines.

NRC RESPONSES

(1) NRC agrees that improvements and maintenance of operational safety is a fundamental responsibility of licensees. The NRC role should be to provide acceptance criteria, detailed guidance where necessary, and any incentives needed to attain and sustain operational safety. NRC agrees with the other parts of recommendation 1 as well and feels the Institute of Nuclear Power Operations may well be the right vehicle for independent evaluation, especially with regard to important human factors. A statement of understanding between the Institute and the NRC should be executed within six months. In addition to creating the Office of Operational Data Analysis and Evaluation, the NRC has required each licensee to establish an engineering staff capability to assess and feed back pertinent operating experience. The intent is that programs of NRC, industry, and vendors will be complemented by and integrated with each licensee's program to assure that intelligible analyses of operating experience reach all reactor operators and plant technical support staff. A proposed rulemaking by NRC would require plant shutdown by a licensee upon discovery of human or operational errors that cause important safety systems to be inoperative.

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(2) ALTHOUGH RESPONSIBILITY FOR SAFETY LIES WITH THE TOTAL ORGANIZATION OF THE PLANT, EACH NUCLEAR POWER PLANT COMPANY SHOULD HAVE A SEPARATE SAFETY GROUP THAT REPORTS TO HIGH-LEVEL MANAGEMENT. The group's assignment would be to evaluate procedures and general operations regularly from a safety perspective, to assess quality assurance programs, and to develop continuing safety programs.

(2) Although NRC has taken action to augment on-site technical support capability with shift technical advisors and operations evaluators at each plant, it is considering a requirement that would expand the staff for on-site safety surveillance by all licensees. A group of technical specialists would be assembled with no direct operating responsibilities to distract them from day-to-day attention to safety; it would report to senior management independently of the power production staff. NRC is also considering a requirement for licensees to improve their systems for independent verification of operational safety by means of automatic system status monitoring and personal verification as well.

| PC RECOMMENDATIONS | NRC RESPONSES |
|---|---|
| (3) INTEGRATION OF MANAGEMENT RESPONSIBILITY AT ALL LEVELS MUST BE ACHIEVED CONSISTENTLY THROUGHOUT THIS INDUSTRY. There must be a single accountable organization with the requisite expertise to take responsibility for the integrated management of the design, construction, operation, and emergency response functions of nuclear power plant operation. Without such demonstrated competence, a company should not qualify for an operating license. At the design stage, the utility can either contract for a "turn-key" plant, a fully operational plant delivered by the vendor or architect-engineer, or the company can assemble expertise capable of integrating the design process. In either case, it is critical that knowledge gained during design and construction of the plant be transferred effectively to those responsible for operating the plant. Clear procedures, responsibilities, and communication serve to ensure accountability and are especially important in the event of a crisis. | (3) NRC has recently surveyed and is studying the technical resources available to each power reactor licensee. It is developing new criteria by which to judge the competence of licensees to operate nuclear facilities and expects to promulgate them by April 1980. NRC agrees emphatically that there is a need for clear definition of roles and responsibilities and has required that licensees for operating plants provide these kinds of well-defined procedures, for both normal and emergency conditions, by January 1, 1980. NRC needs to develop new criteria for determining acceptable technical qualifications to design and construct nuclear power plants. |
| * * * | * * * |
| (4) IT IS IMPORTANT TO ATTRACT HIGHLY QUALIFIED CANDIDATES FOR THE POSITIONS OF SENIOR OPERATOR AND OPERATOR SUPERVISOR. Pay scales should be high enough to attract such candidates. | (4) NRC has taken actions and will do more to substantially increase the qualifications of operating plant personnel (see next heading). NRC agrees it will be necessary for utilities to increase their pay scales. |
| * * * | * * * |
| (5) SUBSTANTIALLY MORE ATTENTION AND CARE MUST BE DEVOTED TO THE WRITING, REVIEWING, AND MONITORING OF PLANT PROCEDURES. Clearer wording, sound and practical content, clear diagnostic instructions for identifying abnormal occurrences, and insistence on the part of utility and vendor management on the early cure of safety questions (with deadlines, sanctions for delays, dissemination of results) are all recommended. | (5) NRC believes that licensees must evaluate and incorporate operating experience into their procedures, has ordered detailed analyses of small break loss-of-coolant accidents for all B&W operating reactors, and has ordered new analyses and procedures by all operating reactor licensees for responding to off-normal events which can be aggravated by operator action. Procedures which assist the operator in responding to inadequate core cooling have also been prescribed. Studies of the effects of stress on operator actions are underway and human factors will be afforded a prominence equal to that given equipment in NRC systems safety evaluations. |
| * * * | * * * |
| (6) STATE RATE-MAKING AGENCIES SHOULD GIVE EXPLICIT ATTENTION TO THE SAFETY IMPLICATIONS OF RATE-MAKING WHEN THEY CONSIDER COSTS ON "SAFETY-RELATED" CHANGES. | (6) NRC agrees and will consider further its role in the resolution of the problem and examine whether other financial considerations, such as deadlines for rate-making purposes or tax exemptions, affect the safety of a nuclear power plant. |
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# PRESIDENT'S COMMISSION (PC) RECOMMENDATIONS ON THE TRAINING OF OPERATING PERSONNEL

**PC RECOMMENDATIONS**

(1) AGENCY-ACCREDITED TRAINING INSTITUTIONS FOR OPERATORS AND SUPERVISORS SHOULD BE ESTABLISHED. Highly qualified instructors, high standards, and an emphasis on fundamentals of nuclear power plants and possible health effects thereof are recommended, and the training of operators to respond to emergencies. The institutions could be national, regional, or specific to nuclear steam systems; reactor operators should be required to graduate from one of them, with exemptions only when there is documented evidence that the candidate has equivalent training; the institutions should be subject to periodic reaccreditation by NRC; candidates must meet entrance requirements.

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(2) INDIVIDUAL UTILITIES SHOULD BE RESPONSIBLE FOR TRAINING OPERATORS WHO ARE GRADUATES OF ACCREDITED INSTITUTIONS IN THE SPECIFICS OF A PARTICULAR PLANT. The operators should be examined and licensed by the NRC both at initial licensing and at relicensing; operators must pass every portion of the examination, and supervisors of operators should have, at a minimum, the same training as operators.

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(3) COMPREHENSIVE ONGOING TRAINING MUST BE GIVEN TO MAINTAIN OPERATORS' LEVEL OF KNOWLEDGE. The training must be continuously integrated with operating experience, with emphasis on diagnosing and controlling complex transients, and on fundamental understanding of reactor safety. Each utility should have ready access to a control room simulator, and operators and supervisors should be required to train regularly on it. Retention of operator licenses should be made contingent upon simulator performance.

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**NRC RESPONSES**

(1) Although it agrees with the objective underlying the recommendation, NRC is not convinced that accreditation by NRC is the best way to proceed (although it does not object, in the long term, to having operators trained in a few, high-quality, accredited institutions closely controlled by NRC). But NRC's approach to date has aimed at upgrading the training requirements while leaving the choice of where to conduct training to the utility. The Institute for Nuclear Plant Operations established by the industry intends to give training to utility management and to instructors involved in operator training, and if the Institute can become the accrediting authority for reactor operator training, it might be preferable, although NRC will certainly be more deeply involved in auditing and monitoring training than ever before.

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(2) Utilities are now responsible for training operators in the specifics of a particular plant. Operators are initially examined and licensed by NRC, but licenses are renewed every 2 years afterward without NRC examination. NRC is taking action to reexamine operators for license renewal, to increase the overall passing grade and require it for each portion of the test (effective now), and will continue to require supervisors to have at least the same training as operators and be licensed as senior operators, as before. Managers at certain levels may also be required to be licensed as senior operators.

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(3) NRC requires ongoing training and requalification of operators with annual examinations conducted by the utility. Requalification programs are being revised to give more emphasis to diagnosing and controlling complex transients, improving the fundamental grasp of reactor safety, and taking account of operating experience. In the future, NRC will administer requalification exams. The use of simulators will be required in operator training and retraining and for recertification. NRC is considering a requirement that utilities upgrade training for all plant personnel, over and above the recommendation cited.

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**PC RECOMMENDATIONS**

(4) RESEARCH AND DEVELOPMENT SHOULD BE CARRIED OUT ON IMPROVING SIMULATION AND SIMULATION SYSTEMS, to bring a higher level of realism to operator training, including simulated transients, and to improve diagnostics and general knowledge of nuclear plant systems.

**NRC RESPONSES**

(4) NRC believes that different types of simulators are needed to upgrade training, on the one hand, and refine diagnostic techniques, on the other. Explicit requirements are being readied for the simulator exercises to be included in operator training, covering normal and abnormal situations and response to multiple and concurrent failures. NRC will undertake extensive research in this area.

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## PRESIDENT'S COMMISSION (PC) RECOMMENDATIONS BASED ON ITS TECHNICAL ASSESSMENT

**PC RECOMMENDATIONS**

(1) EQUIPMENT SHOULD BE EVALUATED ACCORDING TO THE EXTENT TO WHICH IT INFORMS AND ASSISTS OPERATORS TO HELP THEM PREVENT ACCIDENTS AND DEAL WITH THOSE THAT DO OCCUR. Instruments should give both monitory and precursory information, e.g., indications of the full range of temperatures in the reactor under normal or abnormal conditions, and indication of the actual position of valves. Computer technology should be used to furnish clear displays to operators and supervisors of measurements relevant to accident conditions and advance warnings of developing conditions. In the interim, for TMI and similar plants, grouping of key measurements should be considered, with distinct warning signals on a single panel available to a specific operator and a duplicate panel to the supervisor.

**NRC RESPONSES**

(1) NRC agrees with all PC recommendations on improved control room designs and believes that the need for improved design is one of the most important of TMI lessons. Actions have been taken to improve the ability of operators to prevent or cope with accidents by improving the information available to them. Revised procedures and operator training in recognizing inadequate core cooling are required to be completed by the end of 1979 at all operating reactors. Instrumentation to monitor water level in the reactor and pressure, water level, radiation and hydrogen in the containment will be required by the end of 1980, as will other safety items designed to inform the operators clearly and fully. The most important new requirement is the year-long review of control rooms employing experts in human factors and person-equipment interaction. In the long term, NRC is encouraging completion of an industry standard on control room design and will carry out research in this entire area.

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(2) EQUIPMENT DESIGN AND MAINTENANCE INADEQUACIES AT TMI SHOULD BE STUDIED WITH A VIEW TO MITIGATING THE CONSEQUENCES OF ANY SIMILAR FUTURE OCCURRENCE. Iodine filters, the hydrogen recombiner, the vent gas system, containmnet isolation, reporting of water and radiation levels in containment, and the fast analysis of containment samples all merit review and correction.

(2) The NRC staff has required all licensees to fix six of the seven types of components cited by January 1, 1981. Iodine filtration is the subject of ongoing study and criteria development which includes other post-accident radiation control and treatment matters. Requirements for design changes redressing other equipment and maintenance deficiencies have also been imposed.

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(3) MONITORING INSTRUMENTS AND RECORDING EQUIPMENT SHOULD BE PROVIDED TO RECORD CONTINUOUSLY ALL CRITICAL PLANT MEASUREMENTS AND CONDITIONS.

(3) NRC is in complete accord. General criteria for such a requirement were developed by the Lessons Learned Task Force in the form of instrument readings which characterize the plant's safety status. NRC has required that recording equipment and instrumentation be present in the new on-site technical support centers by January 1, 1981.

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**PC RECOMMENDATIONS**

(4) CONTINUING IN-DEPTH STUDIES SHOULD BE INITIATED ON THE PROBABILITIES AND CONSEQUENCES OF NUCLEAR POWER PLANT ACCIDENTS, including the consequences of meltdown. The studies should cover both onsite and offsite effects and encompass a variety of small break loss-of-coolant and multiple failure accidents, with particular attention to human failures. Such studies should be useful in planning for recovery and cleanup after a major accident and in modifying plant design to help prevent or mitigate accidents (e.g., venting hydrogen from the reactor coolant system); they could be carried out by industry or other organizations under NRC or other Federal sponsorship.

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(5) STUDY SHOULD BE MADE OF THE CHEMICAL BEHAVIOR AND THE RETENTION OF RADIOACTIVE IODINE IN WATER, which resulted in the very low release of radioiodine to the atmosphere in the TMI accident. The information should be taken into account in the studies of the consequences of other small break accidents.

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(6) BECAUSE OF HEALTH HAZARDS ASSOCIATED WITH THE CLEANUP AND DISPOSAL PROCESS, CLOSE MONITORING OF THE CLEANUP PROCESS AT TMI AND OF THE TRANSPORTATION AND DISPOSAL OF THE RADIOACTIVE MATERIAL THERE IS RECOMMENDED. As much data as possible should be preserved and recorded about the conditions within the containment building for future safety analyses.

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(7) AS PART OF THE NORMAL SAFETY ASSURANCE PROGRAM, EVERY ACCIDENT OR NEW ABNORMAL EVENT SHOULD BE SCREENED TO ASSESS ITS IMPLICATIONS for the existing system design, computer models of the system, equipment design and quality, operations, operator training, training simulators, plant procedures, safety systems, emergency measures, management and regulatory requirements.

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**NRC RESPONSES**

(4) NRC agrees and has increased or redirected its current program, requiring licensees to analyze small break loss-of-coolant accidents assuming multiple equipment failures. These are complete and revisions of procedures and training have been effected. Crystal River Unit 3, a B&W operating plant, is included in the Integrated Reliability Evaluation Program, as well other operating plants and possibly new operating plant licensees. NRC is also redirecting its research program to take in more probable transients and small break accidents, and is investigating core melt phenomena, including data from TMI relevant to recovery and cleanup after a major accident. Some specific deficiencies revealed at TMI and present elsewhere will be, as recommended, corrected before the end of 1980, but NRC believes that, since the deficiencies existed because this kind of TMI accident had not been considered in design and evaluation of the plant, mitigatory design features addressed to core damage and core melting may be required.

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(5) NRC agrees that more information is needed on the realistic behavior of iodine, other radioisotopes and chemicals in the primary coolant systems of severely damaged reactors, and will conduct the necessary research.

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(6) NRC agrees and has had a continuing presence at the site to monitor, audit and review the cleanup underway. As much important data as possible will be preserved and recorded for future use. NRC has also decided to prepare a programmatic environmental impact statement on the decontamination and disposal of wastes from the TMI accident.

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(7) NRC agrees on the need for thorough investigation of accidents and abnormal events and believes that the initiatives on operating experience evaluation, in close coordination with inspection and enforcement activities for the especially significant events, will meet the intent of this recommendation.

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# PRESIDENT'S COMMISSION (PC) RECOMMENDATIONS ON WORKER AND PUBLIC HEALTH AND SAFETY

| PC RECOMMENDATIONS | NRC RESPONSES |
|---|---|
| (1) EXPANDED AND BETTER COORDINATED RESEARCH INTO HEALTH-RELATED RADIATION EFFECTS SHOULD BE ESTABLISHED, and should include, among others, study of the biological effects of low levels of ionizing radiation; acceptable levels of ionizing radiation to which the general public and workers may be exposed; means for mitigating the adverse health effects of exposure to ionizing radiation; and the genetic or environmental factors which predispose individuals to incurring adverse effects. The research should be coordinated with the National Institutes of Health and other Federal agencies. | (1) NRC agrees with the recommendation. During 1978–79, the NRC staff worked in an interagency project chaired by the Department of Health, Education and Welfare, which also concluded that there was need for this kind of research. Thus, the interagency committee on radiation research, chaired by the National Institutes of Health, was established in early 1979, with NRC as a member. Topics cited by the PC will be introduced by NRC as agenda items for action by the committee. |
| * * * | * * * |
| (2) NRC POLICY STATEMENTS OR REGULATIONS CONCERNING RADIATION-RELATED HEALTH EFFECTS, INCLUDING REACTOR SITING ISSUES, SHOULD BE SUBJECT TO REVIEW AND COMMENT BY THE SECRETARY OF THE DEPARTMENT OF HEALTH AND HUMAN SERVICES. A time limit should be placed on such review to assure expeditious treatment. | (2) NRC agrees with the value of Federal oversight of NRC activities that affect public health. But NRC believes that a more effective and balanced result would be achieved through the role envisioned for the Federal Radiation Policy Council that the President has decided to establish. |
| * * * | * * * |
| (3) AN INCREASED PROGRAM, AS A STATE AND LOCAL RESPONSIBILITY, FOR EDUCATING HEALTH PROFESSIONALS AND EMERGENCY RESPONSE PERSONNEL IN THE VICINITY OF NUCLEAR POWER PLANTS SHOULD BE CREATED. | (3) NRC agrees with this recommendation and, although the suggestion is for a State and local program, NRC intends to give guidance and help in meeting their needs. In particular, NRC will supplement NRC/EPA guidance already available to States on the preparation of emergency response plans to provide more detailed guidance on the education and training of personnel who will respond to emergencies at nuclear power plants. In addition, NRC has offered and will continue to offer technical assistance to the States in the preparation or upgrading of emergency response plans. |
| * * * | * * * |
| (4) UTILITIES MUST MAKE SUFFICIENT ADVANCE PREPARATION FOR THE MITIGATION OF EMERGENCIES, by having radiation monitors available for normal or off-normal conditions; by having the emergency control center for health physics operations and analytic laboratory in a well-shielded area with its own air supply; by having enough instru- | (4) The recommendation of the NRC Task Force on Emergency Preparedness to expand coverage and improve offsite monitoring capability for accidents is being implemented by all operating plant licensees, and NRC has increased its capability in this area. Requirements for onsite monitoring for accident diagnostics and health physics purposes recommended by |

PC RECOMMENDATIONS

ments, respirators, and other equipment for normal or off-normal conditions; and by performing adequate maintenance on all such health-related equipment.

NRC RESPONSES

the Lessons Learned Task Force are also being implemented. Requirements for emergency health physics control centers and health physics equipment are being upgraded. These actions should substantially improve utility capability.

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(5) AN ADEQUATE SUPPLY OF POTASSIUM IODIDE FOR PROTECTION AGAINST RADIATION EFFECTS ON THE THYROID SHOULD BE AVAILABLE REGIONALLY FOR DISTRIBUTION TO THE GENERAL POPULATION AND WORKERS AFFECTED BY A RADIOLOGICAL EMERGENCY.

(5) NRC agrees and will require licensees to have adequate supplies of this agent available for nuclear power plant workers. For the general population, NRC expects to make its availability a necessary part of an acceptable State emergency response plan. Plans are not complete as to how and how much of the agent should be stockpiled and distributed; studies are underway.

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## PRESIDENT'S COMMISSION (PC) RECOMMENDATIONS ON EMERGENCY PLANNING AND RESPONSE

PC RECOMMENDATIONS

(1) EMERGENCY PLANS MUST DETAIL CLEARLY AND CONSISTENTLY THE ACTIONS PUBLIC OFFICIALS AND UTILITIES SHOULD TAKE WHEN OFFSITE RADIATION DOSES OCCUR. The State within which a prospective nuclear power plant will be sited should have an emergency response plan reviewed and approved by the Federal Emergency Management Agency (FEMA) before an operating license is granted. FEMA should have the Federal responsibility for radiological emergency planning and should consult with other agencies, including the NRC and health and environmental agencies. The State should coordinate its planning with the utility and local officials, and States with plants now operating should upgrade, without delay, their plans to conform with FEMA requirements.

NRC RESPONSES

(1) NRC agrees with the substance of the recommendation and has moved to upgrade plans in States with operating plants. Rulemaking has been initiated to raise emergency preparedness standards and an extensive review of all aspects of response capability is underway. A joint letter has been issued by FEMA and NRC confirming the former's lead role in Federal emergency planning and declaring joint responsibility for concurring in State emergency response plans prior to NRC's issuance of an operating license. NRC is considering a rule that would make such issuance contingent upon approval of State plans within a fixed time frame.

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(2) PLANS FOR PROTECTING THE PUBLIC FROM OFFSITE RADIATION RELEASES SHOULD BE BASED ON TECHNICAL ASSESSMENT OF VARIOUS CLASSES OF ACCIDENTS THAT CAN TAKE PLACE AT A GIVEN PLANT. No single plan based on fixed distances and responses can suffice; planning should involve the identification of several different kinds of accidents with different radi-

(2) The basis for emergency response planning has been under examination at NRC for some time. An NRC/EPA task force published the results of an extensive study in December 1978 and its conclusions were consistent with this recommendation. In October 1979, the NRC Commissioners endorsed the concept of a flexible planning base, including emergency planning over much larger areas than before. The base re-

PC RECOMMENDATIONS

ation effects. For each kind there should be clear criteria for the appropriate response at various distances, such as instructing people to remain indoors for a time, providing special medication, or ordering an evacuation. Response plans should be keyed to various possible scenarios and activated when the nature of the potential hazard is clear. Plans should exist for protecting the public from radiation levels lower than those in current NRC-prescribed plans. And all local communities should have funds and technical support adequate for preparing the plans recommended.

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(3) RESEARCH SHOULD BE EXPANDED ON MEDICAL MEANS FOR PROTECTING THE PUBLIC AGAINST VARIOUS LEVELS AND TYPES OF RADIATION. This research should include exploration of appropriate medications that can protect against or counteract radiation.

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(4) IF EMERGENCY PLANNING AND RESPONSE TO A RADIATION-RELATED EMERGENCY IS TO BE EFFECTIVE, THE PUBLIC MUST BE BETTER INFORMED. A program is needed to educate the public on how nuclear power plants operate, on radiation and its health effects, and on protective actions required in an emergency.

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(5) COMMISSION STUDIES SUGGEST THAT DECISION-MAKERS MAY HAVE OVERESTIMATED THE HUMAN COSTS, IN INJURY AND LOSS OF LIFE, IN MANY MASS EVACUATION SITUATIONS. Further study is needed into the human costs of mass evacuation and into the question of whether radiation-related evacuations differ from those occasioned by other events. Such studies should take into account the effects of improved emergency planning, public awareness of the planning, and costs.

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(6) PLANS FOR PROVIDING FEDERAL TECHNICAL SUPPORT, SUCH AS RADIOLOGICAL MONITORING, SHOULD CLEARLY SPECIFY THE RESPONSIBILITIES OF THE VARIOUS SUPPORT AGENCIES AND THE PROCEDURES BY WHICH THEY PROVIDE ASSISTANCE. Existing plans, expecially those of the Interagency Radiological Assistance Plan and the various memoranda of understanding among the agencies, should be reexamined and revised by Federal authorities in the light of TMI and better coordination and more efficient Federal support provided for.

NRC RESPONSES

quires that specific scenarios be used to test the adequacy of plans and that the activation of emergency response be keyed to various plant conditions according to revised emergency action guidelines published in September 1979. NRC currently uses the EPA protective action guides, but will give greater emphasis in the new action level guidance on the potential for exposure as distinct from the actual exposure levels. An NRC staff study on funding problems of State and local governments was recently published and is under consideration by NRC; it discusses the need for and possible sources of such funding.

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(3) NRC agrees that such research is needed and will encourage the Department of Health and Human Services to take steps in this area.

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(4) NRC agrees but believes that a broad public information program would be more appropriately handled by other agencies. Better information on radiation risks is among the subjects to be addressed by the planned Federal Radiation Policy Council. NRC will require, however, that licensees keep the public informed on a continuing basis of the nature of hazards in a radiation emergency and of actions that might have to be taken. Periodic response drills on the part of local and State organizations should contribute to this awareness.

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(5) NRC agrees that further study should be done on this and other protective actions.

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(6) NRC agrees that improvements are needed and has efforts underway to reexamine and revise Federal interagency agreements on emergency assistance.

# PRESIDENT'S COMMISSION (PC) RECOMMENDATIONS ON THE PUBLIC'S RIGHT TO INFORMATION

**PC RECOMMENDATIONS**

(1) FEDERAL AND STATE AGENCIES, AS WELL AS THE UTILITY, SHOULD MAKE ADEQUATE PREPARATION FOR A SYSTEMATIC PUBLIC INFORMATION PROGRAM, so that when a radiation emergency occurs, they can provide timely and accurate information to the news media and the public in a form that is understandable. Assignments of briefing responsibility and availability of informed sources are necessary to reduce confusion and inaccuracy. The utility has primary responsibility for providing information on the status of the plant to the news media and the public, as it has for the management of the accident. The NRC should also be available to provide background information and technical briefings. A designated State agency should convey all information related to State decisions on protective actions (including evacuations) and to offsite radiation releases. This agency should set up the means to keep local officials informed and to coordinate briefings to discuss Federal involvement in any evacuation measures.

* * *

**NRC RESPONSES**

(1) The procedure used before TMI was that NRC public affairs staff would be sent to an accident site to support NRC personnel in communicating with the media, but not to take charge of information activities. At TMI, the NRC in fact took over public information responsibilities on March 31. Although this recommendation prescribes a background role for NRC, it seems more realistic that the Federal regulator be in a position to talk about an emergency situaation, since NRC would expect the State and the public to look to NRC for authoritative information on the situation. NRC believes it would be more effective to have Federal, State, and utility personnel operate out of a single press center and, whenever possible, give a unified view of the situation.

* * *

(2) BECAUSE THE OFFICIAL SOURCES OF INFORMATION MUST MEET THE NEEDS OF THE MEDIA FOR INFORMATION WITHOUT COMPROMISING THE EFFORT OF OPERATIONAL PERSONNEL TO MANAGE THE ACCIDENT, it is recommended that those who brief the news media have direct access to informed sources of information, that technical liaison people be designated as contacts for the briefers and the media, and that primary official news sources have plans for promptly setting up press centers fairly close to the site, properly equipped and staffed.

* * *

(2) NRC agrees with the recommendations and will consider requirements to assure that licensee plans will achieve them. Licensees are now required to identify offsite emergency control centers where the utility, Federal, State, and local officials can gather. A press center would be established either at the off-site emergency control center or nearby, which will facilitate State activities set forth in the preceding recommendation.

* * *

(3) SPECIAL RESPONSIBILITIES ON THE NEWS MEDIA TO PROVIDE ACCURATE AND TIMELY INFORMATION REQUIRE THAT all major media hire and train specialists familiar with reactors and radiological language, and all other media in the area of nuclear power plants should have plans for securing such services in an emergency; reporters try to place complex information in an understandable context and allow the public to decide the hazard to their health and safety; reporters try to avoid raising "what if" questions needlessly and try to understand expressions of uncertainty and probability from the sources of information.

* * *

(3) NRC agrees and will urge the professional societies, such as the American Nuclear Society or the Health Physics Society, to sponsor seminars for the news media where reporters can learn how nuclear power plants operate and about radiation effects. NRC will consider in ongoing rulemaking whether the training program required to be developed by the licensee for local officials could be extended to include local news media personnel.

* * *

| PC RECOMMENDATIONS | NRC RESPONSES |
|---|---|
| (4) STATE EMERGENCY PLANS SHOULD INCLUDE PROVISION FOR CREATION OF LOCAL BROADCAST MEDIA NETWORKS FOR EMERGENCIES THAT WILL SUPPLY TIMELY AND ACCURATE INFORMATION. Arrangements should be made to have knowledgeable people available to go on the air and clear up rumors and explain conditions. Communications between State officials, the utility, and the network should be prearranged to handle the possibility of an evacuation announcement. | (4) NRC agrees the proposal has merit and will incorporate recommendations accordingly in guidance to the States. It will also consider in the ongoing rulemaking on emergency preparedness whether there is a need to include requirements for licensee planning and coordination to disseminate information to the public on these local broadcast networks and to provide information to such networks in the event of an accident. |
| * * * | * * * |
| (5) THE PUBLIC IN THE VICINITY OF A NUCLEAR POWER PLANT SHOULD BE ROUTINELY INFORMED OF LOCAL RADIATION MEASUREMENTS THAT DEPART APPRECIABLY FROM NORMAL BACKGROUND RADIATION, whether from normal or abnormal operation of the nuclear power plant, from a radioactivity cleanup operation such as that at TMI, or from other sources. | (5) NRC agrees with this recommendation, which is consistent with its current practice, in which public announcements are made on any releases to the environment from licensed facilities that appreciably exceed NRC limits (which are small in comparison with normal background, but are in addition to normal background). Most licensees also issue such announcements. |

In addition to providing the Executive Office of the President with responses to each of the President's Commission's recommendations, the NRC cited several examples of considerations and actions it had taken as a result of TMI which were outside the scope of the PC recommendations. Seven such examples were given.

(1) **Generic Requirements for Design Features for Core Melt Consequence Mitigation.** Severe core damage did occur at TMI, but significant exposure of the public was prevented because radiation releases were, for the most part, successfully kept in the containment building. There is substantial evidence that the residual risks of core melt accidents can be significantly reduced if some of the potential modes of containment failure can be prevented or controlled. The NRC Lessons Learned Task Force has recommended that this issue—whether to require additional design features and training for core melt accidents—be revised through the rulemaking process.

(2) **Expanded Reactor Safety Goals, Including Quantification of Reliability.** The President's Commission endorsed the conservative use of safety-cost tradeoffs, but did not confront the fundamental question as to just what level of safety is desired and acceptable. The Advisory Committee on Reactor Safeguards and the NRC Lessons Learned Task Force have recommended that policy guidance be developed within NRC on what is an acceptable safety goal of reactor regulation, reflecting a synthesis of views and priorities and setting forth an objective sufficiently clear for the staff to employ in day-to-day decisionmaking. This regulatory safety goal should comprise both evaluative and quantified reliability criteria, applicable to the development of any new regulatory requirements and to a decision on backfitting requirements to existing plants.

(3) **NRC Emergency Response Capabilities.** Events at TMI demonstrate that NRC has an important role in auditing and monitoring the licensee's actions, and NRC is strengthening the crisis management and technical capabilities of its emergency management staff. The emergency response teams of the NRC Office of Inspection and Enforcement are being tested and actually dispatched to various sites. NRC is also specifying the content and transmission requirements for a nuclear data link from all operating plants to its Operations Center.

(4) **Compensating Features for Plants with High Population Density Sites.** NRC is considering the need for additional protective action—such as shutdown, reduced power or additional design features—for currently operating plants located in densely populated areas.

(5) **Licensing of Operations Personnel in Addition to Reactor Operators and Their Supervisors.** NRC is considering making it a requirement that certain nuclear power plant personnel other than reactor operators and supervisors be licensed. TMI indicated in various ways that plant safety can be affected by persons in many positions, including managers, engineers, auxiliary operators, maintenance personnel, and technicians. The Institute of Nuclear Power

Operations, recently established by the nuclear industry, may have a role to play in this area.

(6) **Plant Security During an Emergency.** A need for clear instructions for plant security during an emergency was brought home by TMI, particularly to ensure that access control measures remain effective but do not hamper recovery operations.

(7) **Worker Protection.** Significant deficiencies in the worker protection program at TMI have been disclosed and, concerned that the problems may be widespread, NRC is developing new generic requirements in this area.

Two of the five NRC Commissioners made separate supplemental responses to the President's Commission findings and recommendations. These are summarized below.

**Commissioner Bradford's Views.** Commissioner Peter A. Bradford expressed his judgment that, while the PC report was helpful and insightful in a number of areas (including recommendations on the NRC, on operating personnel training, technical assessment, and emergency planning), it was "a flawed document" in three respects. First, the major recommendation for a restructuring of the nuclear regulatory process "does not make good sense." Second, there are a number of areas to which the report could have spoken but did not. Third, there is "no clear relationship between the narrative, the findings, and the recommendations, with the result that some important findings do not result in recommendations while some of the recommendations find little support elsewhere in the report."

On the first flaw, the Commissioner felt that the concept of an independent agency headed by a "single administrator appointed by the President . . . to serve at the pleasure of the President" presented a "contradiction in terms," since an agency cannot be independent if its head is removable at the pleasure of the President. Further, the "more this point is corrected by the granting of true independence to the agency the more undesirable it will be to vest what will become quite sweeping powers in a single individual."

The problems within NRC to which the recommendation is addressed are of two kinds: an "attitudinal" problem, which shows up in the agency's failure to pursue the questions which would have led it to discover the vulnerabilities now revealed by TMI; and the diversity of views among the NRC Commissioners which may make it difficult for the agency to correct itself. While the Commissioner agreed that the second problem was curable by setting up a single administrator, as recommended by the President's Commission, "it is also curable through changes within the current Commission structure" which would constitute a "potentially faster and certainly wiser" course of action. The Commissioner pointed out that the only real benefit of the single-administrator proposal (or proposals to reinforce the authority of the Chairman or the Executive Director for Operations) is "that it provides a shortcut away from the perceived stalemate at the current Commission." He felt that these proposals "ignore the fact that collegial agencies are perfectly capable of moving rapidly and innovatively in new directions *as long as they have a coherent and predictable majority that includes the Chairman and that supports the chief operational officers.*"

A number of items were cited by the Commissioner on which he believed the President's Commission "could usefully have taken a position had time permitted."

- On the question of whether and when evacuation was warranted at TMI, he notes that the PC report "said nothing about the validity of the actual recommendation that was made. This seems to me to be an oversight of some magnitude, for such decisions are often likely to involve the allocation of unquantifiable uncertainties. It would be very useful to know whether these twelve citizens . . . feel that a greater or lesser set of evacuation advisories were in order at different times during the accident."
- The report does not discuss "the pros and cons of intervenor funding . . . an essential tool to enable the proposed Public Counsel to guarantee effective outside skeptical participation in the licensing process."
- The PC report is "blurred as to what the fundamental standard for the safety of nuclear power should be. . . . [T]he considered view of twelve laymen on this subject would have been extremely valuable. Instead, one finds statements to the effect that 'accidents as serious as TMI should not be allowed to occur in the future.' . . . [S]ome statement as to how this group regarded the acceptability of risks from nuclear power plants in the context of other technologically imposed risks would have been a helpful guidance." The NRC is going to have to "fill the void with a rulemaking."
- There is no acknowledgment in the PC report of "the strides already achieved since Three Mile Island by the Nuclear Regulatory Commission. . . . This oversight would be easier to understand if it were explicitly acknowledged and explained. It would also be easier to understand if the TMI Commission had not gone out of its way to pat the nuclear industry on the back for having recently created the Institute of Nuclear Power Operations."
- The report speaks repeatedly of examples of AEC promotional attitudes and practices within the NRC but gives no specifics. The statements "tend to tar everyone with the same brush, and they are

not helpful in setting a clear course of corrective action."

- While the report criticizes the NRC's "single failure criterion," it makes no specific recommendation on the subject. If the criterion is to be abandoned, the implications for the nuclear licensing process "are considerable and would almost certainly result in extensive redesigning and backfitting to plants already under construction or in operation." If this is the recommendation of the report, it should have been made explicit.
- The PC report "lays a gentle and indecipherable hand on the state ratesetting process." In the relation between financing and safety, there are "at least two areas of much greater significance . . . the timing of state decisions that create an incentive to rush a plant into service (this allegation was specifically made in regard to TMI) and the Internal Revenue Service's practice of assuming for tax purposes that the plant was in service for the full calendar year if it is in commercial operation by midnight on December 31." Both questions are under study by NRC and "it might be well to ask the Internal Revenue Service and the National Association of Regulatory Utility Commissioners to have a look at them as well."
- On the subject of worker and public health and safety, the report "contains nothing on the vital subject of making sure that workers are adequately informed and trained with regard to radiation and its hazards. It also says nothing about the need to assure that workers who raise safety- or radiation-related concerns are adequately protected against reprisals by their management."
- The report fails to note that the Atomic Energy Act "currently preempts the States from setting most radiological health and safety standards involving nuclear power plants. . . . [I]f the states had a role in this area, they would no longer find themselves excluded from nuclear power plant radiation regulatory matters until the moment at which something really goes wrong and they are expected to step in and cope effectively with the offsite consequences."
- The report "says nothing about the effect of the attitudes of the Congressional Oversight Committees on the quality of the nuclear regulatory process." The approach of the former Atomic Energy Commission cited so often and so critically by the President's Commission "was shaped by the demands that were laid on the AEC by the Joint Committee on Atomic Energy. Anyone trying to understand where nuclear regulation went astray must realize that the AEC was responding not solely to its own or to Executive Branch notions of desirable Atomic Energy policy, but also to the continuing pressure for results from the one congressional committee to which it was answerable. The relationship as I understand it was a mutually reenforcing one, but the continuing role of the Congress setting the tone for nuclear regulation should not be overlooked."

**Commissioner Gilinsky's Views.** Commissioner Victor Gilinsky also put on record certain personal views on the report of the President's Commission. On the basic finding of a need for fundamental change, the Commissioner was in agreement, noting that publication of the report and the attention it received, especially from the President, strengthens the hand of "those concerned with improving nuclear safety and further shifts the burden of proof to those who would do less rather than more." The Commissioner expressed agreement with "almost all" of the findings and recommendations of the report, but stated, "I feel compelled to add that when we get below the general level, down to the nitty-gritty of reactor regulation, the report is less helpful."

The inventory of items that need fixing—operator training, emergency planning, improved use of operating information, etc.—are "almost all . . . the subjects of major NRC actions which were initiated before the report's publication." The more difficult questions "in each case are: What precisely needs to be done? Are NRC actions sufficient?" The President's Commission decided that the present NRC is unable to fulfill its responsibility for providing an acceptable level of safety, but the PC report "is silent on what an acceptable level is." It is up to the NRC, the Commissioner concludes, to "get more specific about overall standards for nuclear safety—on what is safe enough."

The section of the PC report dealing with utility management deficiencies carries "no attempt to judge whether these deficiencies are characteristic of the industry. Without such a determination, it is impossible to judge the overall system for public protection."

The report also fails to deal with the adequacy of the TMI licensee's communication to government authorities of plant conditions—high core temperatures and the containment hydrogen explosion—on the first day of the accident. "I regard this as a vital question," the Commissioner declared. "Given the dangers inherent in nuclear plants we have to be confident that the utilities will report promptly any conditions that require public protection."

The report "never comes to grips with the question of whether an evacuation should or should not have been ordered," a decision which "is critical to forming a judgment on the [Nuclear Regulatory] Commission's responses and to planning further response."

On the subject of NRC Commissioners' isolation from the licensing process, the Commissioner suggests that the single administrator called for in the PC report "would be even more removed from the licensing proceedings" because, as the report proposes it, the

President and Mrs. Carter toured the TMI site on Sunday, April 1, 1979, and are seen above in the TMI-2 control room. At left is NRC's Director of Reactor Regulation, Harold Denton, who was designated the President's personal representative at the site for the duration of the accident.

appeal board decisions would not be reviewable by the administrator. The Commissioner indicates that the experience of NRC is that "leaving all appeals to the Appeal Board leads to loss of policy control over the licensing process." He urges that "[t]he Commissioners need to be more involved in the adjudicatory reviews rather than less."

The PC report recommends, "after seemingly streamlining the NRC for emergencies by shifting to a single administrator," that the NRC "stay out of dealing with emergencies altogether" and leave emergency planning to FEMA and the handling of any accident—and public information related thereto—to the utility. The Commissioner does not think it "wise or realistic to downplay the NRC role to this extent."

The Commissioner also observes that the report, by emphasizing the human failures and "thereby vindicating the equipment," does not stress enough that the equipment "could have been designed to avoid this kind of trouble."

## The President's Response

On October 30, 1979, the President's Commission on the Accident at Three Mile Island presented its final report to the President. Following a period of study by a panel appointed from his staff, the President issued his response to the recommendations of the PC report on December 7, 1979. (The President's statement is reprinted on page 62 in its entirety.)

Among the salient points of the statement were the announcements that:

- A reorganization plan for the NRC would be sent to Congress in the next session which will strengthen the role of the Chairman to lead the Commission in the development of a unified and more reliable nuclear safety regulatory program.
- The President would appoint a new Chairman of the NRC from outside the agency.
- A five-member expert advisory committee would be established to monitor the progress of the

NRC, other Federal agencies, the States, and the utilities in improving the safety of nuclear reactors and in implementing recommendations of the President's Commission. The committee would report periodically to the President and the public.

- The President was asking the NRC and other agencies to accelerate placement of a resident Federal inspector at every reactor site and was asking the NRC to evaluate the need for a Federal presence in the control room of operating reactors.

- The President was directing that the Federal Emergency Management Agency (FEMA) assume responsibility for all offsite nuclear emergency planning and response. A supplemental appropriation of $8.9 million would be submitted to Congress to enable FEMA to complete the review of State emergency plans in all States with operating licenses by June 1980.

- The President was urging the industry to build on the progress it had made since the TMI accident to provide enhanced analysis and evaluation for safety of the design, construction, and operation of plants and a greatly strengthened training, retraining, and evaluation program for operators and supervisors. He asked the NRC to appraise and reinforce these efforts.

- To assure that the lessons of TMI were expeditiously absorbed and applied, the President was submitting a supplemental appropriation to Congress of $49.2 million for the NRC and $7 million for the DOE. These funds would allow the collection and evaluation of data and speed the implementation of reforms.

Affirming that he "agrees fully with the spirit and intent" of all recommendations of the PC report, the President chose to strengthen the NRC organization through enhanced executive powers for the Chairman, rather than by creation of a new agency. Since the collegial Commission, representing diverse and complementary views, would be retained, the President chose not to create a 15-member oversight committee. He did, however, announce his intention of establishing a smaller advisory committee to report to him on the progress of the NRC and others, as noted above. The President urged the NRC to complete its work of defining and effecting the reforms dictated by analyses of TMI as quickly as possible and, in any event, no later than May 1980. In doing so, the President observed that "we must resume the licensing process promptly so that the new plants which we need to reduce our dependence on foreign oil can be built and operated." He concluded by stating that "nuclear power has a future in the United States—it is an option that we must keep open. I call on the utilities and their suppliers, the NRC, the Executive Departments and agencies, and the State and local governments to assure that the future is a safe one."

# Statement by President Carter on the Kemeny Commission Report

I have reviewed the report of the Commission I established to investigate the accident at Three Mile Island nuclear power plant. The Commission, chaired by Dr. John Kemeny, found very serious shortcomings in the way that both the government and the utility industry regulate and manage nuclear power.

The steps I am taking today will help ensure that nuclear power plants are operated safely. Safety has always been, and will remain, my top priority.

As I have stated before, in this country, nuclear power is an energy source of last resort. By this I meant that as we reach our goals for conservation, direct use of coal, development of solar power and synthetic fuels and enhanced production of American oil and natural gas, we can minimize our reliance on nuclear power.

Many of our foreign allies must place greater reliance than do we on nuclear power, because thay do not have the vast natural resources that give us many alternatives. We must get on with the job of developing alternative energy sources—by passing the legislation I proposed to the Congress, and by making an effort at every level of society to conserve energy.

We cannot shut the door on nuclear energy.

The recent events in Iran have shown us the clear, stark dangers that excessive dependence on imported oil holds for our Nation. We must make every effort to lead this country to energy security.

Every domestic energy source, including nuclear power, is critical if we are to free our country from its overdependence on unstable sources of high-priced foreign oil. We do not have the luxury of abandoning nuclear power or imposing a lengthy moratorium on its further use. A nuclear plant can displace up to 35,000 barrels per day.

We must take every possible step to increase the safety of nuclear power production. I agree fully with the spirit and intent of the Kemeny Commission's recommendations, some of which are within my power to implement, others of which rely on the Nuclear Regulatory Commission or the utility industry itself.

To get the government's own house in order I will take several steps. First, I will send to Congress a reorganization plan to strengthen the role of the Chairman of the NRC and provide this person with the power to act on a daily basis as the chief executive officer, with authority to put needed safety requirements and procedures in place. The Chairman must be able to select key personnel, and act on behalf of the commission during an emergency.

Second, I will appoint a new Chairman of the NRC—someone from outside that agency, in the spirit of the Kemeny Commission's recommendation. In the meantime, I have asked Commissioner Ahearne, now on the NRC, to serve as Chairman. Dr. Ahearne will stress safety and the prompt implementation of the needed reforms. In addition, I will establish an independent advisory committee to help keep me informed of the progress the NRC and the industry are achieving in making nuclear energy safer.

Third, I am directing the Federal Emergency Management Agency to head up all off-site emergency activities, and complete a thorough review of emergency plans in all states with operating reactors by June.

Fourth, I have directed NRC and other agencies to accelerate our program to place a resident federal inspector at every reactor site.

Fifth, I am asking all relevant government agencies to implement virtually all of the other recommendations of the Kemeny Commission.

A detailed fact sheet is being issued to the public, and a more extended briefing will be given to the press.

With clear leadership and improved organization, the Executive branch and the NRC will be better able to act quickly on the critical issues of improved training and standards, safety procedures, and the other Kemeny Commission recommendations.

But responsibility to make nuclear power safer does not stop with the federal government. In fact, the primary day-to-day responsibility for safety rests with utility company management and suppliers of nuclear equipment. There is no substitute for technically qualified and committed people working on the construction, operation and inspection of nuclear power plants. Personal reponsibility must be charged both at the corporate level and at the plant site. The industry owes it to the American people to strengthen its commitment to safety.

I call on the utilities to implement the following changes:

First, building on the steps already taken, the industry must organize itself to develop enhanced standards for safe design, operation, and construction of plants.

Second, the nuclear industry must work together to develop and to maintain in operation a comprehensive training, examination and evaluation program for operators and supervisors. This training program must pass muster with the NRC through accreditation of training programs.

Third, control rooms must be modernized, standardized and simplified as much as possible to permit better informed decision-making during an emergency.

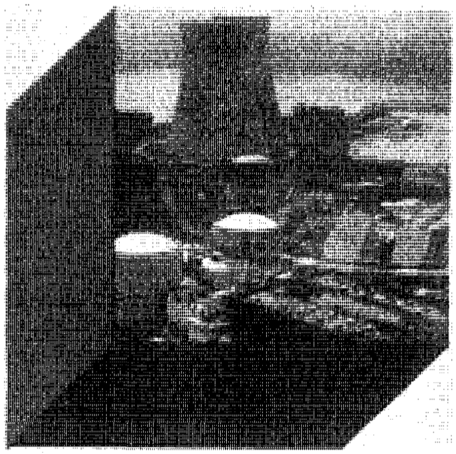
I challenge our utility companies to bend every effort to improve the safety of nuclear power.

Finally, I would like to discuss how we manage the transition period during which the Kemeny recommendations are being implemented. There are a number of new nuclear plants now awaiting operating licenses or construction permits.

Licensing decisions rest with the NRC and, as the Kemeny Commission noted, it has the authority to proceed with licensing these plants on a case-by-case basis, which may be used as circumstances surrounding a plant dictate. The NRC has indicated, however, that it will pause in issuing new licenses and construction permits in order to devote its full attention to putting its house in order. I endorse the approach the NRC has adopted, but I urge the NRC to complete its work as quickly as possible, and in any event no latter than six months from today.

Once we have instituted the necessary reforms to assure safety, we must resume the licensing process promptly so that the new plants which we need to reduce our dependence on foreign oil can be built and operated.

The steps I am announcing today will help assure our country of the safety of nuclear plants. Nuclear power has a future in the United States—it is an option that we must keep open. I call on the utilities and their suppliers, the NRC, the executive Departments and agencies, and the State and local governments to assure that the future is a safe one.



# 3
# Reactor Regulation

**In February 1980 NRC authorized fuel loading and low-power operation of the TVA's Sequoyah Plant.**

The goals of the NRC in licensing and regulating nuclear reactors in the United States are to protect the public health and safety, to protect the quality of the environment, and to assure compliance with the antitrust laws in civilian nuclear activities. The reactor licensing process is centered in the NRC Office of Nuclear Reactor Regulation (NRR), where each proposed nuclear power plant is reviewed by a staff drawn from a broad spectrum of professional disciplines. (See Appendix 1 for a description of the NRR organization.)

The Three Mile Island accident revealed the need for a number of changes in NRC's conception of and approach to nuclear safety, requiring action in the areas of human factors, operational safety, emergency planning, nuclear power plant design and siting, health effects, and public information. Much of the NRR staff was involved in these efforts, participating in the task forces on TMI Support, Bulletins and Orders, and Lessons Learned, as discussed in Chapter 2. This chapter deals with other matters related to the licensing and regulation of nuclear power plants: the licensing of reactor operators, unresolved safety issues, other technical issues, advanced nuclear power plants, protection of the environment, progress in standardization, antitrust and indemnity activities, and other subjects.

## Status of Nuclear Power Generation

As of September 30, 1979, there were 192 nuclear power units either in operation, being built or being planned, representing a total capacity of 187,000 net megawatts electric (MWe). This total is 20 units fewer than the total at the end of fiscal year 1978. Of the 192 units, 186 had entered the NRC licensing process, as follows:

- 70 licensed to operate, with a total capacity of 51,000 MWe.
- 91 with construction permits representing 100,000 MWe.
- 25 under review for construction permits, representing 29,000 MWe. (Initial construction work was proceeding on four of these under limited work authorizations.)

Of the remaining six units—those which had not entered the NRC licensing process—four had been ordered and two publicly announced.

Shortly after the close of fiscal year 1979, the Commission issued an "Interim Statement of Policy and Procedure," dated October 4, 1979, taking note of the various investigations—within and outside of NRC—of the TMI accident still underway, and the implications of those efforts for pending licensing actions. The statement affirmed that "new construction permits, limited work authorizations, or operating licenses for any nuclear power reactors shall be issued only after action of the Commission itself. . . . In these circumstances no full adjudicatory decision which authorizes issuance of such a permit, authorization or license shall be issued by an Atomic Safety and Licensing Board except after further order of the Commission itself."

On November 9, following publication of the report of the President's Commission (see Chapter 2), Chairman Hendrie notified the Director of the Office of Science and Technology Policy that new nuclear power plants would not be licensed until NRC had developed new or improved safety objectives and the criteria by which to implement them.

## Assistance From Other Agencies

Because of the necessity of reallocating a major portion of the staff of the Office of Nuclear Reactor Regulation to the conduct and support of the numerous investigations and associated activities

# THE LICENSING PROCESS

Obtaining an NRC construction permit—or a limited work authorization, pending a decision on issuance of a construction permit—is the first objective of a utility or other company seeking to operate a nuclear power reactor or other nuclear facility under NRC license. The process is set in motion with the filing and acceptance of the application, generally comprising ten or more large volumes of material covering both safety and environmental factors, in accordance with NRC requirements and guidance. The second phase consists of safety and environmental factors, in accordance with NRC requirements and guidance. The second phase consists of safety, environmental, safeguards and antitrust reviews undertaken by the NRC staff. Third, a safety review is conducted by the independent Advisory Committee on Reactor Safeguards (ACRS); this review is required by law. Fourth, a mandatory public hearing is conducted by a three-member Atomic Safety and Licensing Board (ASLB), which then makes an initial decision as to whether the permit should be granted. This decision is subject to appeal to an Atomic Safety and Licensing Appeal Board (ASLAB) and could ultimately go to the Commissioners for final NRC decision. The law provides for appeal beyond the Commission in the Federal courts.

As soon as an initial application is accepted, or "docketed," by the NRC, a notice of that fact is published in the *Federal Register,* and copies of the application are furnished to appropriate State and local authorities and to a local public document room (LPDR) established in the vicinity of the proposed site, as well as to the NRC-PDR in Washington, D.C. At the same time, a notice of a public hearing is published in the *Federal Register* and local newspapers) which provides 30 days for members of the public to petition to intervene in the proceeding. Such petitions are entertained and adjudicated by the ASLB appointed to the case, with rights of appeal by the petitioner to the ASLAB.

The NRC staff's safety, safeguards, environmental and antitrust reviews proceed in parallel. With the guidance of the Standard Format (Regulatory Guide 1.70), the applicant for a construction permit lays out the proposed nuclear plant design in a Preliminary Safety Analysis Report (PSAR). If and when this report has been made sufficiently complete to warrant review, the application is docketed and NRC staff evaluations begin. Even prior to submission of the report, NRC staff conducts a substantive review and inspection of the applicant's quality assurance program covering design and procurement. The safety review is performed by NRC staff in accordance with the Standard Review Plan for Light-Water-Cooled Reactors, initially published in September 1975 and updated periodically. This plan states the acceptance criteria used in evaluating the various systems, components and structures important to safety and in assessing the proposed site, and it describes the procedures used in performing the safety review.

The NRC staff examines the applicant's PSAR to determine whether the plant design is safe and consistent with NRC rules and regulations; whether valid methods of calculation were employed and accurately carried out; whether the applicant has conducted his analysis and evaluation in sufficient depth and breadth to support staff approval with respect to safety. When the staff is satisfied that the acceptance criteria of the Standard Review Plan have been met by the applicant's preliminary report, a Safety Evaluation Report is prepared by the staff summarizing the results of their review regarding the anticipated effects of the proposed facility on the public health and safety.

Following publication of the staff Safety Evaluation Report, the ACRS completes its review and meets with staff and applicant. The ACRS then prepares a letter report to the Chairman of the NRC presenting the results of its independent evaluation and recommending whether or not a construction permit should be issued. The staff issues a supplement to the Safety Evaluation Report incorporating any changes or actions adopted as a result of ACRS recommendations. A public hearing can then be held, generally in a community near the proposed site, on safety aspects of the licensing decision.

In appropriate cases, NRC may grant a Limited Work Authorization to an applicant in advance of the final decision on the construction permit in order to allow certain work to begin at the site, saving as much as seven months time. The authorization will not be given, however, until NRC staff has completed environmental impact and site suitability reviews and the appointed ASLB has conducted a public hearing on environmental impact and site suitability with a favorable finding. To realize the desired saving of time, the applicant must submit the environmental portion of the application early.

The environmental review begins with a review of the applicant's Environmental Report (ER) for acceptability. Assuming the ER is sufficiently complete to warrant review, it is docketed and an analysis of the consequences to the environment of the construction and operation of the proposed facility at the proposed site is begun. Upon completion of this analysis, a Draft Environmental Statement is published and distributed with specific requests for review and comment by Federal, State and local agencies, other interested parties and members of the public. All of their comments are then taken into account in the preparation of a Final Environmental Statement. Both the draft and the final statements are made available to the public at the time of respective publication. During this same time period NRC is conducting an analysis and preparing a report on site suitability aspects of the proposed licensing action. Upon completion of these activities, a public hearing, with the appointed ASLB presiding, may be conducted on environmental and site suitability aspects of the propsed licensing action (or a single hearing on both safety and environmental matters may be held, if that is indicated).

The antitrust reviews of license applications are carried out by the NRC and the Attorney General in advance of, or concurrently with, other licensing reviews. If an antitrust hearing is required, it is held separately from those on safety and environmental aspects.

About two or three years before construction of the plant is scheduled to be complete, the applicant files an application for an operating license. A process similar to that for the construction permit is followed. The application is filed, NRC staff and the ACRS review it, a Safety Evaluation Report and an updated Environmental Statement are issued. A public hearing is not mandatory at this stage, but one may be held if requested by affected members of the public or at the initiative of the Commission. Each license for operation of a nuclear reactor contains technical specifications which set forth the particular safety and environmental protection measures to be imposed upon the facility and the conditions that must be met for the facility to operate.

Once licensed, a nuclear facility remains under NRC surveillance and undergoes periodic inspections throughout its operating life. In cases where the NRC finds that substantial, additional protection is necessary for the public health and safety or the common defense and security, the NRC may require "backfitting" of a licensed plant, that is, the addition, elimination or modification of structures, systems or components of the plant.

related to the accident at Three Mile Island, substantial delays were encountered in the review of applications for operating licenses for nuclear power plants. To help alleviate this situation, the NRC sought the help of technical experts in other Government agencies on a temporary basis, under interagency contracting arrangements.

During the latter part of 1979, the Department of Energy made available technical specialists from several national laboratories to assist in technical reviews of applications. Review teams have been established at the Argonne National Laboratory, the Idaho National Engineering Laboratory, the Savannah River Laboratory, the Oak Ridge National Laboratory, the Los Alamos Scientific Laboratory, the Pacific Northwest Laboratory, the Battelle Columbus Laboratory, the Lawrence Livermore Laboratory, and the Energy Technology Engineering Center.

Technical assistance has also been obtained from the U.S. Geological Survey, the U.S. Army Corps of Engineers, and the Naval Research Laboratory.

These resources will be used during fiscal year 1980 to supplement NRC staff resources for the review of license applications, pending recruitment and hiring of additional personnel authorized by Congress.

### Licensing Reactor Operators

The safety of a nuclear facility depends not only on its design but on the qualifications of the people who operate it. To assure that the people in charge of each nuclear power plant are capable of directing and performing the activities necessary to reactor operation, the NRC requires each individual who handles the controls of the reactor to be licensed. The requirements for issuance of operators' licenses are set forth in 10 CFR Part 55. Two types of licenses are issued by the NRC: one for "operators" and one for "senior operators." During fiscal year 1979, the NRC issued 212 new operator licenses, 256 renewals, and 26 amendments, bringing the number of operator licenses in effect on September 30, 1979 to 992. During the same period, 184 new licenses, 434 renewals, and 36 amendments were issued for senior operators, bringing the total to 1,437 in effect.

**TMI Related Activities.** Following the accident at Three Mile Island Unit 2 (TMI), members of the Operator Licensing Branch were assigned to the Lessons Learned Task Force, Bulletins and Orders Task Force, and TMI Support Task Force to determine the role of nuclear power plant operators in the TMI-2 accident, to assist in the development of recommendations for the upgrading of operator training requirements, and to review and recommend changes to normal, abnormal, and emergency procedures. A number of these proposals have been submitted to the Commission for review. The activities of these task forces and other groups dealing with the causes and consequences of the TMI accident, both within and outside of the NRC, are covered at length in the preceding chapter of this report.

## UNRESOLVED SAFETY ISSUES

Section 210 of the Energy Reorganization Act of 1974, as amended, reads as follows:

> "Unresolved Safety Issues Plan"
>
> "Section 210. The Commission shall develop a plan for providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report to the Commission thereafter."

In response to this reporting requirement, the NRC provided a report to the Congress, NUREG-0410, in January of 1978 describing the generic issues program of the Office of Nuclear Reactor Regulation (NRR) that had been implemented early in 1977. The NRR program described in NUREG-0410 provides for the identification of generic issues, the assignment of

**Table 1. Nuclear Power Plant Licensing Actions—Fiscal Year 1979**

CONSTRUCTION PERMITS

| *Applicant* | *Facility* | *Date Issued* | *Location* |
|---|---|---|---|
| Tennessee Valley Authority | Yellow Creek 1 & 2 | 11-29-78 | Tishomingo County, Miss. |
| Long Island Lighting Co. | Jamesport 1 & 2 | 1-4-79 | Suffolk County, N.Y. |

(No Limited Work Authorizations or Operating Licenses for nuclear power plants were issued during FY 1979.)

priorities, the development of detailed task action plans to resolve the issues, the projections of dollar and man-power costs, continuing high level management oversight of task progress, and public dissemination of information related to the tasks as they progress.

The 1978 NRC Annual Report described the NRC's progress towards resolving those issues addressed in the NRR program that had been identified as "Unresolved Safety Issues" (p. 19). Seventeen "Unresolved Safety Issues" were identified, to be addressed by 22 generic tasks. Three of these generic tasks have now been reported as complete.

## Evaluation Process

The definition of an "Unresolved Safety Issue" developed by the NRC for use in identifying issues that require reporting to the Congress (pursuant to Section 210) is as follows:

> "An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants affected."

The process used to determine which issues met the definition of an "Unresolved Safety Issue" was described briefly in the 1978 NRC Annual Report and in considerably more detail in NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants—A Report to Congress." (NUREG-0510 accompanied the 1978 Annual Report when it was transmitted to the Congress in January of 1979.) The review process included a systematic review of over 130 generic issues. As an aid to this review, an evaluation was made of the subject areas involved according to their relative importance from the standpoint of public risk. This risk-based characterization was used together with a substantial body of additional information (e.g., heavy weight was given to issues arising from events reported to the Congress as "Abnormal Occurrences") to determine which issues qualified for reporting to Congress.

Table 2 provides a listing of each of the "Unresolved Safety Issues" and related generic tasks identified in the 1978 NRC Annual Report. It also provides last year's projected dates for issuing NRC staff reports and the corresponding dates as of December 1, 1979.

As indicated in Table 2, three reports providing the staff's resolution of three "Unresolved Safety Issues" were issued for public comment as of January 1, 1980. Four more staff reports addressing four more "Unresolved Safety Issues" are expected to be issued by the end of February 1980. A number of the tasks have undergone schedule slips in 1979. These schedule slips can be attributed in large measure to the temporary staff reassignments occasioned by the TMI-2 accident. The technical scope of several of the tasks also required some further definition as a result of the accident. The impacts of the accident on those Unresolved Safety Issue tasks which were affected are discussed briefly in the individual task progress reports below.

In an effort to minimize the impact of the TMI-2 accident on these tasks, the Office of Nuclear Reactor Regulation established, in June of 1979, an interim organization specifically assigned to continue work on "Unresolved Safety Issues." This interim structure was still in place as of January 1, 1980.

## Identification of New Issues

Although a number of safety-related issues came to light in 1979 as a result of the TMI–2 accident and other events, the NRC staff has not been able to perform an in-depth review to identify and evaluate new issues. Therefore, no new "Unresolved Safety Issues" have been defined for reporting in 1979. As of January 1, 1980, NRC efforts were being concentrated on implementing new TMI-related requirements on operating plants and on identifying, defining and scoping additional TMI-related issues and tasks. Several broad program areas where issues and tasks are being scoped will likely result in designation of new Unresolved Safety Issues. These program areas include the following:

(1) Man-machine interface and control-room design.

(2) Qualification and training of operation, maintenance and supervisory personnel.

(3) Off-site emergency response, emergency planing, and action guidelines.

(4) Siting policy, including compensatory design and operating provisions for plants in areas where evacuation would be difficult.

(5) Systems reliability and interactions.

(6) Consideration in licensing requirements of accidents involving degraded or melted fuel.

The NRC staff performed a cursory review of a number of candidate issues from sources other than TMI accident investigations, including a review of events reported as Abnormal Occurrences in 1979. None of these issues was judged to be of such safety urgency and importance as to require reporting in advance of the staff's and the Commission's in-depth and systematic review of all candidate issues. Such a systematic and in-depth review will be performed in 1980, after the major recommendations of the major TMI investigations are available. A special report will be provided to the Congress by July of 1980, describing the review and the new issues to be designated Unresolved Safety Issues.

## Table 2. Schedules for Tasks Addressing Unresolved Safety Issues

| *Task No.* | *Unresolved Safety Issue* | *Schedule for Issuing Staff Report in 1978 NRC Annual Report* | *Schedule for Issuing Staff Report as of January 1, 1980* |
|---|---|---|---|
| A-1 | Water Hammer | 1980 | August 1981 |
| A-2 | Asymmetric Blowdown Loads | Early 1979 | January 1980 |
| A-3 | PWR Steam Generator Tube Integrity | Early 1980 | May 1980 |
| A-4 | " " " " " | | |
| A-5 | " " " " " | | |
| A-7 | BWR Mark I and Mark II Pressure Suppression Containments | A-7—October 1979 | February 1980 |
| A-8 | " " " " " " " " " | A-8—October 1980 | November 1980 |
| A-39 | " " " " " " " " " | A-39—October 1979 | March 1980 |
| A-9 | Anticipated Transients Without Scram | Early 1979 | April 1980 |
| A-10 | BWR Nozzle Cracking | Late 1979 | February 1980 |
| A-11 | Reactor Vessel Materials Toughness | July 1979 | December 1980 |
| A-12 | Steam Generator and Reactor Coolant Pump Supports | August 1979 | Issued November 1979 |
| A-17 | Systems Interactions | Phase I—September 1979<br>Phase II—September 1980 | April 1980<br>May 1981 |
| A-24 | Qualification of Class IE Safety-Related Equipment | 1979 | Issued December 1979 |
| A-36 | Control of Heavy Loads Near Spent Fuel | Early 1979 | January 1980 |
| A-40 | Seismic Design Criteria | Phase I—1979<br>Phase II—1981 | February 1980<br>March 1981 |
| A-42 | Pipe Cracks in Boiling Water Reactors | Not Scheduled | Issued October 1979 |
| A-43 | Containment Emergency Sump | Not Scheduled | Not Scheduled* |
| A-44 | Station Blackout | Not Scheduled | Not Scheduled* |

*Task Action Plan under development when TMI accident occurred; it is anticipated that the task can be completed in 1982.

### Progress Reports

Background information and progress reports for each of the Unresolved Safety Issues listed in Table 2 are provided below. Progress reports on the staff's efforts on tasks reported as unresolved in last year's annual report are also provided. As indicated above, NRC staff reports have been issued for three Unresolved Safety Issue tasks as of January 1, 1980. These tasks are: Task A-12, "Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports"; A-24, "Qualification of Class 1E Safety-Related Equipment"; and Task A-42, "Pipe Cracks in Boiling Water Reactors." The reports describe the technical studies conducted by the NRC staff or its contractors and the safety conclusions that provide the NRC staff's resolution of each of these safety issues. Broad public and industry comment is being solicited on these three reports.

### Water Hammer

Water hammer events are intense pressure pulses in fluid systems (such as commonly experienced when rapidly closing a water faucet) and they often occur in nuclear power plant fluid systems. In the past few years, over 200 incidents involving water hammer in nuclear power reactors have been reported. These incidents have involved many types of fluid systems, including steam generator feed-rings, feedwater and steam supply piping, residual heat removal systems, emergency core cooling systems, containment spray systems, and service water systems. Water hammer can have various causes, such as the rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motions. Most of the damage has been relatively minor, though there have been several cases of failure or partial failure of system piping.

While no water hammer incident has resulted in the release of radioactivity outside of a plant, the concern is that water hammer could result in the failure of a pipe in the reactor coolant system or disable a system required to cool the plant after a reactor shutdown.

The means to prevent one particular type of water hammer caused by the rapid condensation of steam in the steam generator feed-rings of some pressurized water reactors are being instituted. In addition, applicants with new steam generator designs are being required to demonstrate through test or analysis that water hammer will not occur in these designs. Plants with steam generators—of the top feeding type that are subject to water hammer—are being required to modify the feed-rings and/or test the systems to assure water hammer will not occur. Other actions to correct the specific causes of water hammer in other nuclear power plant systems are also being required.

Under Generic Task A-1, the potential for water hammer in various systems is being evaluated and appropriate requirements and systematic review procedures are being developed to ensure that water hammer is given appropriate consideration in all areas of licensing reviews. A technical report, NUREG-0582, "Water Hammer In Nuclear Power Plants," providing the results of an NRC staff review of water hammer events in nuclear power plants and stating current staff licensing positions, was published in July 1979. Issuance of this report completes a major subtask of Generic Task A-1.

In addition, seven technical reports on water hammer have been issued by NRC contractors participating in Task A-1. Issuance of these reports completes four other major subtasks of Generic Task A-1. Collectively, these seven reports have provided: (1) a review and evaluation of actual and potential water hammer events in nuclear power plants, (2) analytical methods and calculational procedures to be used in the evaluation of water hammer incidents, and (3) current state-of-the-art information on water hammer. At the close of the report period, a report summarizing these NRC sponsored water hammer studies had been completed in draft form for review. Issuance of this report will complete another major subtask.

The need for additional work to evaluate the safety significance of various water hammer scenarios has been identified, and the scope of Task A-1 has been expanded to include such studies. This need, combined with the manpower impacts of the Three Mile Island Unit 2 accident, resulted in a schedule slip of about 7 months in the projected completion date for Task A-1 to August 1981.

## Asymetric Blowdown Loads On the Reactor Coolant System

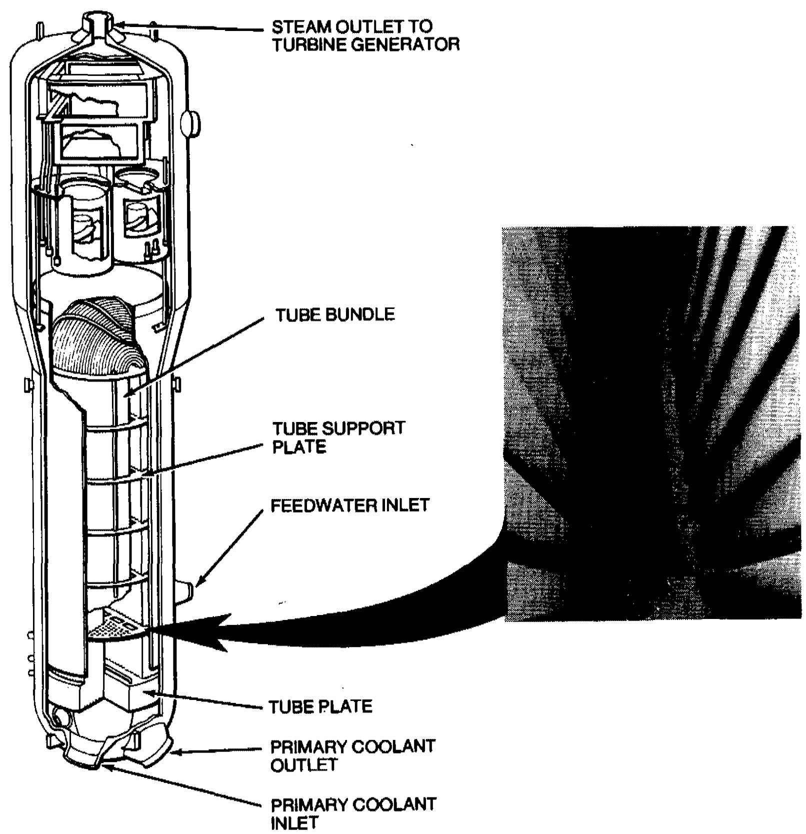
In the very unlikely event of a rupture of the primary coolant piping in light water reactors, large non-uniformly distributed loads would be imposed upon the reactor vessel, reactor vessel internals, and other components in the reactor coolant system. The potential for such asymmetric loads, which result from the rapid depressurization of the reactor coolant systems, was first identified in 1975 and thus was not considered in the original design of some facilities. Details on the safety significance of this issue and actions taken by NRC and industry prior to fiscal year 1979 in response to it may be found in the 1978 NRC Annual Report, pp. 21-24.

Plant modifications to assure that the postulated loads are accommodated have been implemented late in the construction stage of several plants and have been proposed and are under staff review for some operating plants. For plants still under operating license review, the NRC staff requires that plant-specific analyses and any necessary plant modifications be completed prior to issuance of an operating license. The staff reviewed and approved topical reports from the vendors of pressurized water reactors (PWRs) explaining their generic approaches to the calculation of the asymmetric loads in a loss-of-coolant accident (LOCA). Lead plant evaluations and sensitivity studies were also conducted by the staff. A report providing the NRC staff's resolution of this issue is expected to be issued by the end of January 1980. This report will provide acceptance criteria and guidelines for use in plant-specific analyses.

Plant-by-plant implementation of the results of Task A-2 actually began with a request for plant-specific analyses in January 1978. All licensees with operating PWRs are in the process of evaluating their plant's capacity to sustain asymmetric LOCA loads. The PWR owners' groups have met with the staff periodically to present interim results and progress. All of the requested evaluations are scheduled to be completed by early 1980 at which time NRC staff review and implementation of plant modifications will begin.

Asymmetric blowdown loads may also be important in boiling water reactor (BWR) plants, but they are expected to have lesser safety significance than in PWRs, because of the lower operating pressures in BWRs. A plan for resolving the matter for BWR plants will be developed by the NRC staff and pursued separately from the PWR issue.

PWR STEAM GENERATOR




**The buildup of corrosion deposits between the steam generator tubes and the tube support plates, in addition to constricting the tubes, exerts stresses on the tube support plates. The stresses cause hourglassing of the normally rectangular internal bypass flow holes located between the innermost tube rows.**

## PWR Steam Generator Tube Integrity

The heat produced in the reactor at a nuclear power plant is used to convert water into the steam which drives the turbine-generators. In plants employing pressurized water reactors, the primary coolant water, which extracts heat by circulating through the reactor core and is radioactive, is kept under pressure sufficient to prevent boiling. This high-pressure water passes through tubes around which a secondary coolant (also water, but not radioactive) is circulating under somewhat lower pressure. The water in the secondary system boils and produces steam to drive the turbine generators. The assembly in which the heat transfer takes place is the steam generator. The tubes within it are an integral part of the primary coolant boundary, keeping the radioactive primary coolant in a closed system and isolated from the environment. The primary concern is the maintenance of steam generator tube integrity during both normal operation and postulated accident conditions. Another concern is that the increased steam generator tube inspections and repairs have resulted in significant increases in occupational exposures to workers.

A discussion of the specific problems associated with steam generator tube integrity that were occurring at operating reactors was provided in the 1977 and 1978 NRC Annual Reports on pp. 95 and 22, respectively. A more detailed discussion of steam generator operating experience is provided in NUREG-0523, "Operating Experience with Recirculation Steam Generators," published in January 1979 and in NUREG-0571, "Operating Experience with Once Through Steam Generators," to be published in fiscal year 1980.

The significant developments in Westinghouse and Combustion Engineering steam generators, since August 1978, were the following:

- Steam generator replacement at Surry Unit 2 is essentially completed. Replacement is planned for Surry Unit 2, Turkey Point Units 3 and 4, and Palisades. In the interim, the units are operating under restrictions imposed by the NRC.
- Condenser retubing to reduce in-leakage of seawater and the installation of full-flow demineralizers in the secondary coolant system, to remove any chlorides which might leak, has retarded the rate of tube denting at Millstone Unit 2.
- Yankee Rowe performed a 100 percent inspection of Steam Generators 1 and 4 in November 1978.

Fifteen and 12 defective tubes were discovered in Steam Generators 1 and 4, respectively. The mode of degradation was secondary side wastage. A leak rate of approximately 125 gallons-per-day existed prior to the inspection. The Unit, which has stainless steel tubing, converted from phosphate to AVT secondary water treatment in 1968.

- Point Beach Unit 1 was required to shut down on September 20, 1978 and March 12 and August 5, 1979 because of steam generator tube leaks. The cause of the leaks was cracking of the tubes in the crevice between the tubes and the tubesheet. The cracking was a result of caustic stress corrosion. A 100 percent inspection of both steam generators was performed. The inspection revealed 52 defective tubes in Steam Generator A and 45 defective tubes in Steam Generator B. All the cracks were located within the tubesheet and are therefore not considered a significant safety concern.

Oconee Units 1, 2 and 3 and Crystal River Unit 3 are the only Babcock and Wilcox units which have had steam generator tube leaks. Tubes in one localized area of the Oconee Unit 1, 2 and 3 steam generators have failed because of cracks of unknown origin propagated circumferentially by flow-induced vibration.

The status in the B&W steam generators, since August 1978, is the following:

- The Oconee Units have not had a steam generator leak related to a fatigue crack since April 1978. However, a steam generator tube leak (not believed to be related to a fatigue crack) occurred on July 24, 1979.
- Crystal River Unit 3 was shut down for a steam generator tube leak on August 19, 1979. The leak is believed to be through a steam generator tube-to-tubesheet weld which was damaged when a burnable poison rod assembly broke up in March of 1978.
- A demonstration tube sleeving program was initiated by Duke Power Company at the Oconee Units. The tube sleeves will be installed to change the vibrational characteristics of the tubes and decrease the dynamic stress and the susceptibility of the tubes to fatigue cracking. They will not serve as part of the primary coolant boundary.
- An additional degradation mechanism, defined as an "erosion-corrosion" phenomenon and resulting in tube wall thinning, has been identified at Oconee and other B&W units.

Plant technical specifications require routine inservice inspection of steam generators to be performed every 12 to 24 months. The NRC has imposed license conditions on plants with severely degraded steam generators to increase the required frequency of inspection. The conditions also require that following inspection of steam generators and completion of any necessary repair programs by the licensees, the NRC must approve or concur in the restart of each severely affected facility. To date, the units severely affected by tube wall thinning have completed inspection and repair programs and received NRC approval for operation for limited time periods. Safe operation is assured by the imposition of strict conditions requiring the plugging of affected tubes and restricting allowable leak rates during operation. While the NRC continues to closely monitor and evaluate the acceptability of continued operation of plants experiencing steam generator tube problems, it is proceeding with three generic tasks in the NRC program for the resolution of generic issues (specifically, Generic Tasks A-3, A-4, and A-5, addressed to the problems of Westinghouse, Combustion Engineering, and Babcock and Wilcox steam generators, respectively.)

The Task Action Plans for these tasks have been combined in a single plan encompassing all three tasks. The approach taken in the Task Action Plan is to integrate technical studies in the three areas of system analyses, inservice inspection, and tube integrity in order to establish improved criteria by which to ensure safe and reliable steam generator operation. The purpose of the system analyses is to evaluate the consequences of failures involving different numbers of steam generator tubes during postulated accident conditions (LOCA and main-steam-line break, or MSLB)—considering predicted fuel behavior, emergency core cooling system (ECCS) performance, radiological consequences, and containment response. The results will be used to define the tolerable level of steam generator tube leakage during postulated accidents. The major emphasis in the inservice inspection portion of the tasks is to develop a statistically based inservice inspection program that will provide assurance that no more than the tolerable level of tube leakage, defined by the system analyses, would occur in an accident. The tube integrity portion of the tasks is primarily concerned with experimental verification of the tube behavior during postulated accidents, development of tube plugging criteria, and definition of operating procedures for minimizing tube degradation.

The statistical analyses of inservice inspection programs, which is being performed parametrically, was scheduled to be complete in February 1980. The system analyses and tube integrity evaluation were scheduled for completion in early 1980. The results of the Task Action Plan will be: (1) tube plugging criteria based on new experimental data, (2) statistically based inservice inspection methods, (3) recommendations for improved methods of operation, and (4) recommendations for design improvements for new plants. These will be described in an NRC staff report scheduled to be issued for comment in May 1980.

## BWR Mark I and Mark II Pressure Suppression Containments

Boiling water reactor pressure suppression containments designed by the General Electric Company utilize a large mass of water as the principal heat sink by which to condense the steam and absorb the energy that might be released from the reactor in the unlikely event that certain pipes in the primary system should fail. As postulated, the absorption of energy by the stored water reduces the potential buildup of pressure inside the containment, and that in turn reduces the driving force that might lead to a release to the environment of fission products that may have been released from the reactor core.

In the course of performing large scale testing of an advance design pressure-suppression containment (Mark III), and during in-plant testing of Mark I containments, new suppression pool hydrodynamic loads were identified which had not been explicitly included in the original Mark I or Mark II containment design basis. These additional loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA, and from the suppression pool response to various modes of safety relief valve (SRV) operation generally associated with operating conditions during an accident. Since these new hydrodynamic loads had not been explicitly considered in the original design of the Mark I and Mark II containments, the NRC staff determined that a detailed reevaluation of these containment system designs was required. The affected utilities formed Mark I and Mark II Owners' Groups and drew up both short-term and long-term programs for resolution of the pool dynamic problems for their respective containment designs. The programs include a number of comprehensive experimental and analytical programs to establish pool dynamic loads, load combinations, and design criteria.

The NRC staff has identified and initiated a number of generic tasks to review and evaluate the results of the industry programs and to develop criteria for licensing actions on individual plants using the Mark I and Mark II containment designs. These generic tasks are included in the NRC Program for Resolution of Generic Issues. Specifically, they are Task A-6, "Mark I Short-Term Program"; Task A-7, "Mark I Long-Term Program"; Task A-8, "Mark II Containment Program"; and Task A-39, "Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containment."

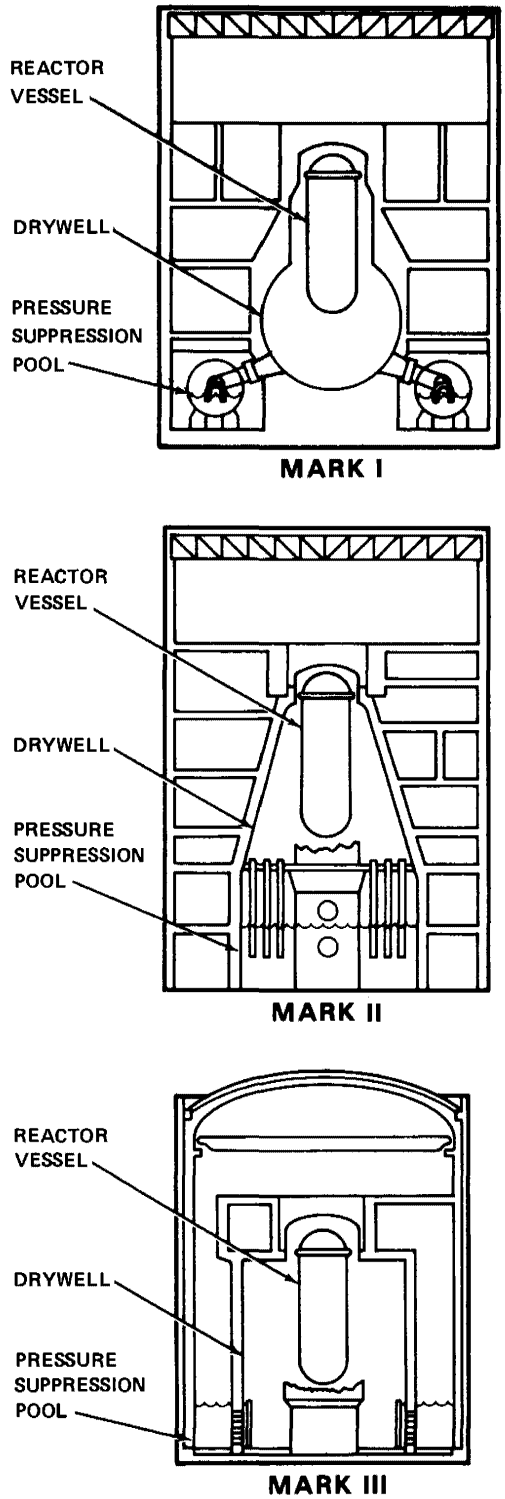
The objectives of the Mark I Short-Term Program were: (1) to examine the containment system of each BWR facility with a Mark I containment design to verify that it would maintain its integrity and functional capability when subjected to the most probable hydrodynamic loads induced by a postulated design basis loss-of-coolant accident; and (2) to verify that licensed Mark I BWR facilities may continue to operate safely, without undue risk to the health and safety of the public, while a methodical, comprehensive long-term program is conducted.

The NRC determined that, for the Short-Term Program, "maintenance of containment integrity and function" would be adequately assured if a safety factor to failure of at least two were demonstrated to exist for the weakest structural or mechanical component in the Mark I containment system (that is, if the calculated stresses in all components of the affected containment structure were shown to be less than one-half the stress which would cause the component to lose its structural integrity). The NRC concluded that the objectives of the Short-Term Program had been satisfied and documented the basis for this conclusion in the "Mark I Containment Short-Term Program Safety Evaluation Report," NUREG-0408, dated December 1977. (Thus Task A-6 was completed in December 1977.)

The objectives of the Mark I Long-Term Program are: (1) to establish design basis (conservative) loads that are appropriate for the anticipated life (40 years) of each Mark I BWR facility, and (2) to make whatever plant changes may be required to restore the original intended design safety margins for each Mark I containment system. The industry program includes experiments and calculations designed to provide a detailed basis for hydrodynamic load definition and structural assessments. The generic aspects of the program are described in a Plant Unique Analysis Applications Guide (General Electric Topical Report NEDO-24583), which was submitted by the Mark I Owners' Group in February 1979, and a Load Definition Report (General Electric Topical Report NEDO-21888), which was submitted, in two parts, in December 1978 and March 1979. These reports described the proposed load definition and assessment techniques for Mark I containments. They were reviewed by the NRC staff, who issued a set of acceptance criteria for the generic assessment techniques in September 1979. A Safety Evaluation Report describing the staff's review and the bases for the acceptance criteria was scheduled to be issued in February 1980, marking the completion of Generic Task A-7.

Subsequently, each utility will be required to perform a plant-unique analysis using approved load definition and structural analysis techniques to demonstrate conformance with the structural acceptance criteria. The scheduled completion date for the Mark I Long-Term Program—including the issuance of license amendments and the implementation of any plant modifications necessary to satisfy the Long-Term Program structural acceptance criteria—is December 1980. To maintain this schedule, a number of utilities

BWR PRESSURE SUPPRESSION CONTAINMENTS




**The design objective of BWR containment systems is to condense the steam released during certain postulated accidents to limit the release of fission products within the reactor building and to provide** water for emergency cooling systems. NRC's Generic Task A-39 entails the review and evaluation of an industry-based program to establish pool dynamic loads for BWR Mark I, II and III pressure suppression containment designs.

have undertaken plant modifications prior to the completion of their plant-unique analysis. This action has been considered necessary to minimize the potential for unduly long extensions of exemptions or extended plant outages later in the program. Similarly, modifications to components external to the containment (e.g., support structures) have and are being conducted during normal plant operation.

The Mark II Owners' Group developed a program consisting of a number of analytical and experimental tasks to support these pool dynamic loads application methods. They divided the overall program into two parts: A Lead Plant Program and a Long-Term Program. The objective of the Mark II Lead Plant Program was to establish design basis (conservative) loads appropriate for the anticipated life of each Mark II BWR facility. The Mark II owners' specification of the Lead Plant Program loads are described in Revision 2 of the Dynamic Forcing Function Report, NEDO-21061-P, and in several application memoranda. The tasks comprising the Lead Plant Program and Long-Term Program are listed in NUREG-0487.

The staff has reviewed and evaluated the pool dynamic loads associated with a postulated large loss-of-coolant accident proposed by the Mark II Owners' Group to determine their acceptability for use in plant-unique analyses for the lead plants. The Mark II Lead Plant Program was essentially completed in October 1978 with the publication of NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria." The lead Mark II plants proposed several exceptions to these criteria. The staff has reviewed these proposed exceptions and has found most of them acceptable. A safety evaluation of these exceptions and their bases was to be discussed by the staff in a supplement to NUREG-0487.

The purpose of the Mark II Long Term Program is to refine the load definitions used in the Lead Plant Program and to support by additional experimental and theoretical work, a reduction in some of these loads for use in the evaluation of the Mark II plants following the lead plants. The Mark II owners' technical program includes a total of 101 tasks. This program is about 70 percent complete. During the past year, the Mark II owners have completed a number of the Long-Term Program tasks and numerous technical reports have been provided for NRC review. Late in 1979, the Mark II owners modified the Long Term Program to include several new analytical and test programs. The addition of these new tasks could delay the Mark II Long Term Program review beyond the projected completion date of January 1981.

Under Generic Task A-39, the NRC staff will review and evaluate the results of industry experimental and analytical programs to establish and justify the safety relief valve-related pool dynamic loads for BWR Mark

I, II, and III containment designs. The results of Generic Task A-39 will be an integral part of the final acceptability of these designs. The portions of this generic task related to the Mark I and Mark II containments are currently scheduled to be completed in March 1980.

## Anticipated Transients Without Scram

Nuclear plants have safety and control systems to limit the consequences of abnormal operating conditions transients. Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. "Anticipated Operational Occurrences" or "Anticipated Transients" are defined (10 CFR Part 50, Appendix A) as "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all off-site power." In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe anticipated transient and the reactor shutdown system did not function as designed, then an "anticipated-transient-without-scram," or ATWS, would have occurred.

This issue has been discussed throughout the NRC and AEC and the nuclear industry for a number of years. Details on the safety significance of the issue and actions taken by NRC and industry prior to fiscal year 1979 in response to it may be found in the 1978 NRC Annual Report, pp. 27 and 28.

On the basis of discussions with senior NRC management, the Advisory Committee on Reactor Safeguards, and industry representatives, and the review of the Lewis Committee report on the Reactor Safety Study, the NRC staff in December 1978 proposed a combination of preventative and mitigative means of providing protection from ATWS events. In this supplement, the NRC staff proposed different types of plant modifications. The design alternatives which were proposed take into consideration the status of the plants—whether operating, under construction or nearly ready for operation—and questions of practicability, including the cost of such modifications.

In order to confirm that these alternatives provided the needed level of safety, the industry was required to provide the necessary confirmation analyses and the staff originally intended to make its recommendations to the Commission in the spring of 1979. The Three Mile Island Unit 2 accident affected these plans in several ways. First, both industry and NRC staff manpower were diverted from ATWS work; second, the Three Mile Island event scenario indicated that a number of aspects of the ATWS accident evaluation required reconsideration, especially for PWRs.

The shortage of available industry manpower delayed several of the required submittals of confirmation analyses. The result was a substantial slip in the projected completion date for the ATWS task. NRC staff manpower was partially restored in June 1979 and meetings were held with industry representatives in July and August of 1979 to discuss the impacts of the Three Mile Island Unit 2 accident on their respective ATWS evaluations. As of January 1, 1980, the NRC staff planned to propose an ATWS rule to the Commission by April 15, 1980, with a goal of issuing a final ATWS rule by December 1980.

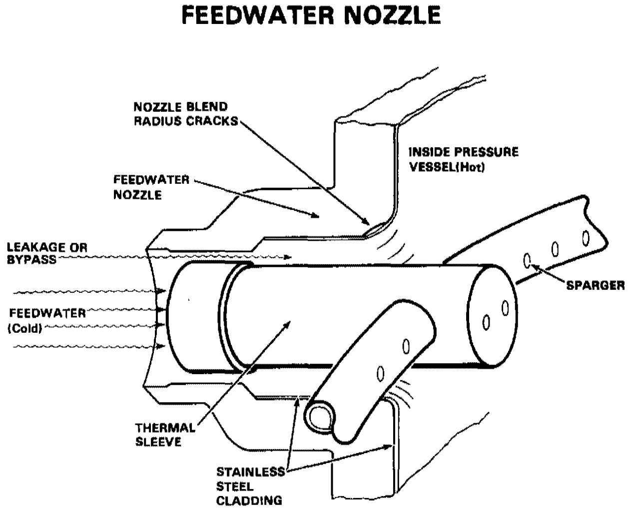
## BWR Nozzle Cracking

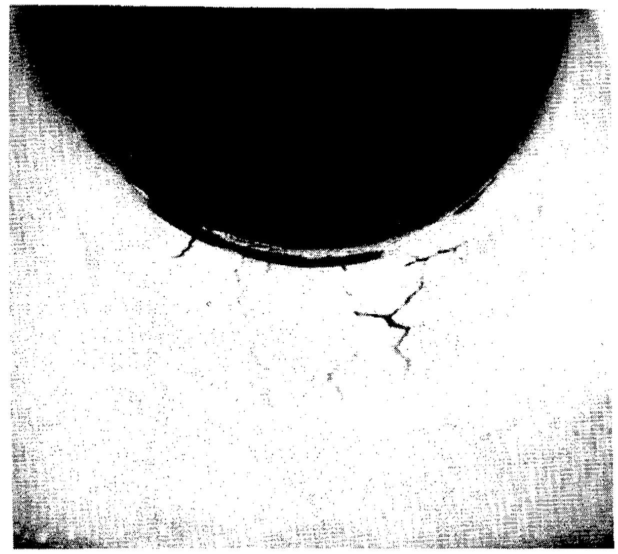
Over the last several years, inspections at 22 of the 31 boiling water reactor (BWR) plants licensed for operation in the U.S. have disclosed some degree of cracking in the feedwater nozzles of the reactor vessel at 18 facilities. One facility has not yet accumulated significant operating time and has, therefore, not yet been inspected.

The feedwater nozzles are an integral part of the primary pressure boundary of the reactor coolant system and the second barrier (after the fuel cladding) to the release of radioactive fission products. All of the repaired BWR feedwater nozzles met the ASME pressure vessel code limits, however, and no immediate action was necessary. Because only relatively small amounts of metal have been removed by repair operations, there has been no significant reduction in safety margins. Nevertheless, the cracking is potentially serious because:

- Excessive crack growth could lead to impairment of pressure vessel safety margins.
- The design safety margin could also be reduced by excessive removal of nozzle reinforcement while grinding out cracks, and repair by welding would be complicated.
- The exposure to radiation of the personnel performing inspection and repair tasks can be considerable.
- The repair of these kinds of cracks can result in considerable shutdown time at the plant affected.

The reactor vendor (the General Electric Company) and the NRC have concluded from their separate studies that the cracks are initiated by rapid fluctuations in water temperature on the inside surface of the nozzles during periods of low feedwater temperature when flow may also be unsteady and perhaps intermittent. The cracks then grow deeper as a result of operational startup and shutdown cycles or other operationally induced transients. The stainless steel cladding exhibited less resistance to crack initiation than the underlying low-allow steel.



Cracks in nozzles of the feedwater and control rod drive lines of BWR reactor pressure vessel have been studied by the vendor (General Electric) and the NRC staff for several years. Evidence indicates that abrupt and wide fluctuations in water temperatures (see diagram above) are the initial causes of cracking. Photo above shows such cracks.

The vendor has performed extensive analysis and testing to confirm the suspected cause of the cracking and to develop possible long-term solutions—a newly designed sleeve, removal of the stainless steel cladding, reduction of the temperature differential at the nozzle, or some combination of these. The licensees involved have increased the number and extent of in-service inspections of feedwater nozzles, with careful repair and reinspection where cracks were found. The vendor advised these licensees to closely monitor startup and shutdown procedures in an effort to substantially reduce the time during which cold feedwater is being injected into the hot pressure vessel.

In a closely related area, the NRC was informed in March 1977 by the General Electric Company that a crack had been found in the nozzle of the control rod drive (CRD) return line in a reactor vessel. The CRD return line nozzles are the openings in BWR pressure vessels through which the high pressure water in excess of that needed to operate and cool the CRDs is returned to the pressure vessel. The cracks resembled those found in the feedwater nozzles and seemed to be the result of the same kind of cyclic thermal stresses that were causing feedwater nozzle cracks. The maximum crack depth has been 0.87 inch.

The NRC staff efforts related to the resolution of these two similar issues regarding nozzle cracking in boiling water reactors were consolidated into a single staff effort, Generic Task A-10, in 1977. Under Generic Task A-10, the staff issued interim guidance to operating plants in a report entitled, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking," in 1977.

The staff has now completed its review of the General Electric studies on feedwater nozzle cracking and has concluded that the new sparger design—in conjunction with other remedial measures, such as clad removal and more appropriate operating procedures—is an effective means of greatly reducing the probability of crack initiation. The new sparger design includes flow discharge nozzles, a triple thermal sleeve, and two piston ring seals in the nozzle bore. The effectiveness of the new design in minimizing bypass leakage and other problems encountered in the older designs was confirmed by extensive testing and analyses by General Electric, including vibration, thermal-hydraulic, materials, and thermal fatigue evaluations. Other designs may also be acceptable.

Feedwater system changes, necessary on some low feedwater temperature plants to assure no cracking over the lifetime of the plant, are being evaluated on a plant-specific basis. An NRC staff report incorporating guidance for operating reactors and plants under licensing review is in preparation and is scheduled to be issued for comment in February 1980.

The resolution of questions regarding the future selection of improved inservice inspection techniques and frequency of inspection has been separated from the generic task while major industry investigations continue (including thermal cracking in a full-size nozzle mockup to be used in ultrasonic evaluation). A supplement to the NRC staff report cited above may be necessary upon completion of these studies. In the meantime, stringent inspection requirements, based mainly upon dye-penetrating testing, are still in force. All licensee efforts, such as system and operational changes, to lengthen the time to crack initiation and to slow crack growth are taken into account in the determination of inspection techniques and criteria.

The CRD nozzle issue will be resolved by a combination of actions which includes nozzle inspection and repairs and some CRD system notifications. Certain system modifications recommended by General Electric involved cutting and capping the nozzle and return line but that action would reduce the capability to direct high pressure water through the CRD system when the vessel is otherwise isolated. Although this system is not normally expected to perform this function in safety analyses, the capability played a major role in keeping the core covered during the incident at Browns Ferry Unit 1 on March 22, 1975. As a result of its review of these modifications, the NRC has concluded that only a limited number of plants will be allowed to modify the CRD system in accordance with the GE recommendations. Unless the licensees of the remaining plants demonstrate, by testing, that sufficient flow is available to the reactor vessel with the return line removed, they will be required to retain the return line, rerouted to the feedwater line or a similar suitable connection that doesn't have the potential for cracking in the reactor vessel nozzle. The staff's evaluation, conclusions, and guidance on the CRD return line nozzle issue will also be included in the February 1980 NRC staff report referred to above.

Plant-specific implementation of the generic licensing positions developed under this task (with the exception of future inservice inspection questions) has already begun.

## Reactor Vessel Material Toughness

Nuclear reactor pressure vessels are required to have an adequate margin of protection against fracture in the presence of relatively large postulated flaws. This requirement is imposed for the sake of conservatism, even though extensive, periodic inservice inspection programs serve to provide protection against the presence of such flaws. Fracture mechanics—the engineering method used to establish the failure margin—employs a quantitative material property called fracture toughness to calculate the conditions under which catastrophically rapid crack propagation will occur. Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in nuclear reactor pressure vessels, three facts are important. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, the technical specifications for power reactors set limits on the operating pressure during heatup and cooldown operations. These restrictions assure that the combination of pressure and temperature will remain well below that which might cause brittle fracture of the reactor vessel if a significant flaw were present in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these technical specifications over the life of the plant.

For the service time and operating conditions typical of current operating plants, reactor vessel fracture toughness provides adequate margins of safety against vessel failure. Further, for most plants the vessel material properties are such that adequate fracture toughness can be maintained over the life of the plants. However, results from a reactor vessel surveillance program indicate that up to 20 older operating pressurized water reactor pressure vessels were fabricated with materials that will have marginal toughness after comparatively short periods of operation. This issue has been incorporated in the

The protective insulation has been pulled aside following the testing of a weld-repair portion of a six-inch thick pressure vessel. A flaw more than five inches deep and 18 inches long was created in the area which was then subjected to pressure overloads more than double the design pressure, without disruptive failure.

NRC staff's program for the resolution of generic issues as Task A-11.

The fundamental goal of Task A-11 is to provide an improved engineering method by which to assess the safety margin in nuclear reactor pressure vessels and to develop appropriate criteria for the evaluation of normal, transient, or postulated accident conditions under the improved method. This method could then be used to provide such an assessment for those older reactor pressure vessels that will eventually have marginal toughness according to the current method. Because relatively large amounts of prefracture plastic deformation can be expected at high temperatures even in pressure vessel steels of low toughness, the new evaluation method will employ "elastic-plastic" fracture mechanics concepts. The basis for this improved methodology is described in NUREG-0311, "A Treatment of the Subject of Tearing Instability," developed under an NRC-sponsored program at Washington University. Additional Washington University work extending the methodology to reactor pressure vessels was funded by the Department of Energy. The engineering method developed will account for radiation-induced material degradation.

Task A-11 also includes or relies on programs sponsored by the NRC Office of Nuclear Regulatory Research to provide: (1) an improved evaluation of material degradation mechanisms resulting from neutron irradiation, and (2) the development of improved testing methods for use in determining the elastic-plastic properties of materials.

Since last year's report, the following has been accomplished:

- Although delayed, an elastic-plastic fracture test method for routine determination of fracture toughness was developed. Verification of the test method is underway.
- The elastic-plastic fracture mechanics methods of NUREG-0311 were confirmed by work supported by an Electric Power Research Institute program, "Methodology for Plastic Fracture."
- The methods developed in these programs were successfully used by NRC contractors to analyze two pressure vessel burst tests reported in the Heavy Section Steel Technology Program, sponsored by the NRC Office of Nuclear Regulatory Research.
- The potential for restoring by thermal annealing the pressure vessel toughness lost by neutron radiation was shown to be impractical.

Significant delays have developed over the past year as a result of difficulties encountered in extending the new engineering methodology to reactor pressure vessels. There is agreement among experts that the methodology can be extended, but it will require a significantly greater effort than that accomplished under the DOE contract referred to above. The staff will carry out this additional work by contract with the Oak Ridge National Laboratory. Because of this problem, the schedule for completing Task A-11 has slipped about one year, to December 1980.

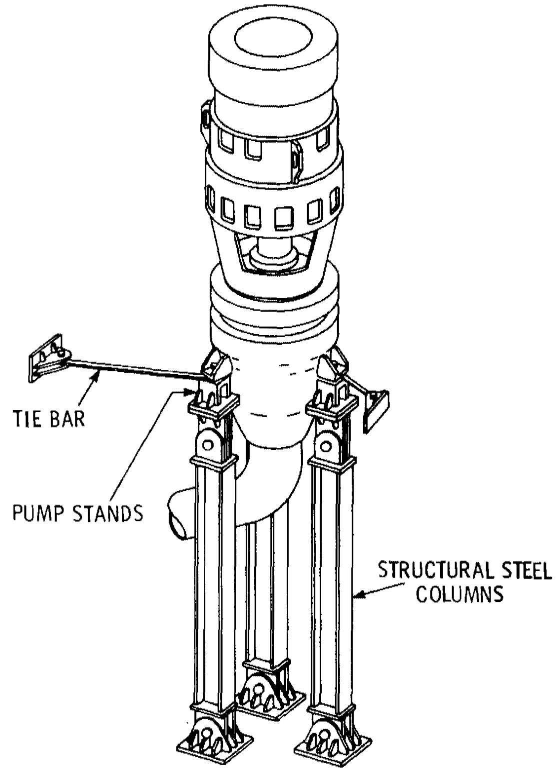
## Fracture Toughness and Potential for Lamellar Tearing of PWR Steam Generator And Reactor Coolant Pump Supports

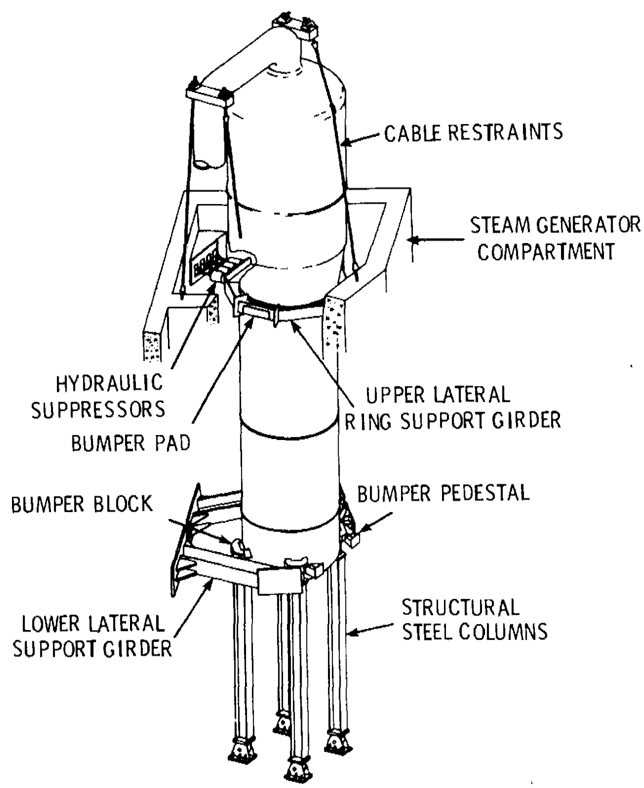
During the course of licensing review for a specific pressurized water reactor (PWR) a number of questions were raised as to (1) the adequacy of the fracture toughness properties of the material used to fabricate the reactor coolant pump supports and steam generator supports, and (2) the potential for failure due to lamellar tearing of these same supports. The safety concern is that, although these supports are designed for worst-case accident conditions, low fracture toughness or lamellar tearing could cause the support to fail during such accidents. Support failure could conceivably impair the effectiveness of systems designed to mitigate the consequences of the accident. An example of a postulated event sequence of potential concern would be a large pipe break in the reactor coolant system which would severely load the supports, followed by a support failure of sufficient magnitude that a major component such as a steam generator would be displaced resulting in failure of the emergency core cooling system piping needed to provide cooling water to the core.

Because materials and designs similar to those of the PWR originally reviewed have been used in other plants, review of this issue was included in the NRC Program for Resolution of Generic Issues as Generic Task A-12.

A consultant was engaged to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in the later stages of operation license review. This reassessment included review of the materials utilized in the support of 38 potentially affected PWRs. Based on the consultant's evaluation, it was determined that there are 21 plants whose supports are of questionable toughness and, accordingly, further detailed plant-specific review is required. This decision concluded the generic study of this subject under Task A-12. During the plant-specific reviews that will follow, either the structural integrity of the supports must be demonstrated, or measures to assure their structural integrity will be required.

A report describing the NRC staff's safety evaluation and conclusions and describing its plans for implementation (i.e., the more detailed plant-specific reviews referred to above) was issued for comment in November 1979. It is entitled, "Potential for Low Fracture Toughness for Lamellar Tearing in PWR




**NRC's Generic Safety Issues Task A-12 deals with the ability of certain PWR reactor coolant system component supports to withstand accidents. Shown here are typical reactor coolant pump and steam generator support structures at the Prairie Island Nuclear Generating Plant in Minnesota.**

Steam Generator and Reactor Coolant Pump Supports," dated September 1979 (NUREG-0577). Review guidelines and acceptance criteria for use in licensing reviews of new facilities are also being prepared to incorporate the results of Task A-12 into the staff's Standard Review Plan. A contractor has been chosen to undertake the plant-specific reviews of operating plants during the implementation phase.

Lamellar tearing is a cracking phenomenon which occurs beneath welds and is principally found in rolled steel plate fabrications. The tearing always lies within the parent plate, often outside the transformed (visible) heat-affected zone (HAZ) and is generally parallel to the weld fusion boundary. Lamellar tearing occurs at certain critical joints usually within large welded structures involving a high degree of stiffness and restraint. Restraint may be defined as a restriction of the movement of the various joining components that would normally occur as a result of expansion and contraction of weld and metal and adjacent regions during welding.

The results of an extensive survey by the staff's consultant revealed that, although lamellar tearing is a common occurrence in structural steel construction, virtually no inservice failures due to lamellar tearing are known. Nonetheless, additional NRC-sponsored research is being planned to provide a more definitive and complete evaluation of the importance of lamellar tearing to the structural integrity of nuclear power plant support systems. This research will be a follow-on effort to Generic Task A-12.

## Systems Interactions in Nuclear Power Plants

In November 1974, the Advisory Committee on Reactor Safeguards requested that the staff give attention to the evaluation of safety systems from a multidisciplinary point of view, in order to identify potentially undesirable interactions between plant systems. The concern arises because the design and analysis of systems is frequently assigned to teams with functional engineering specialties such as civil, electrical, mechanical, or nuclear. The question is whether the work of these functional specialists is sufficiently integrated in their design and analysis activities to enable them to identify adverse interactions between and among systems. Such adverse events might occur, for example, because designers did not assure that redundancy and independence of safety systems were provided under all conditions of operation required, which might happen if the functional teams were not adequately coordinated.

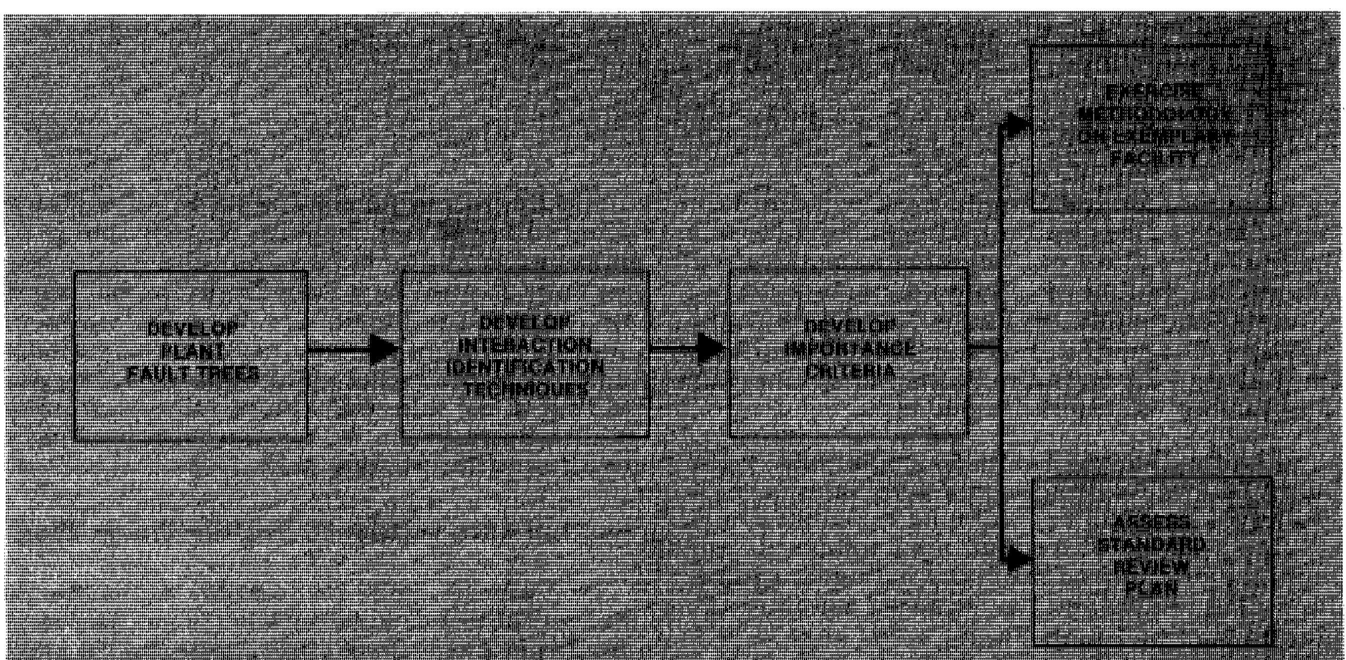
The NRC staff believes that its current review procedures and safety criteria provide reasonable assurance that an acceptable level of redundancy and independence is provided for systems that are required for safety. Nonetheless, in mid-1977, this task (Task A-17) was initiated to confirm that present procedures adequately take into account the potential for undesirable interactions between and among systems. Because systems interactions are potentially of great significance to plant safety, this issue has been identified as an "Unresolved Safety Issue."

The NRC staff's current review procedures assign primary responsibility for review of various technical areas and safety systems to specific organizational units and assign secondary responsibility to other units where there is a functional or interdisciplinary relationship. Designers follow somewhat similar procedures and provide for interdisciplinary reviews and analyses of systems. Task A-17 will provide an independent investigation of safety functions—and systems required to perform these functions—in order to assess the degree to which the current review procedures take potential systems interactions into account. This investigation is being conducted by Sandia Laboratories under contract assistance to the NRC.

The contractor effort, Phase I of the task, began in May 1978 and is scheduled to be completed in March 1980. The Phase I investigation is structured to identify areas where interactions are possible between and among systems *and* have the potential of negating or seriously degrading the performance of safety functions. The investigation will then identify how NRC review procedures account for these interactions.

The functional investigation of Phase I is being conducted by the method of "fault tree" analysis. As of the end of the report period, the detailed fault trees were completed. Analysis of the fault trees, and the comparison of the results against the NRC review procedures was scheduled to be completed by the contractor by December 31, 1979. A final contractor report was scheduled to be issued in March 1980. A report providing the NRC staff's conclusions based on the contractor's work was scheduled to be issued in April 1980.

The Three Mile Island Unit 2 accident caused the NRC staff to consider reorienting the Task A-17 Phase I effort so as to include improved treatment of such matters as operator actions, design errors, and maintenance procedures. It was decided not to disrupt




For the purpose of its Generic Issues Task A-17, "Systems Interactions," NRC has defined a system as "a set of components or subsystems working as a unit to execute a specific function," Systems interaction is defined as "a situation where the likelihood of an undesired event is increased due to the relationship between components." The methodology used to address Phase 1 of this task is reflected in this flow chart.

the Phase I effort, which is nearing completion, but rather to consider expanding the Phase II effort to include treatment of TMI-2 related issues. Scoping of the follow-on work of Phase II is expected to be completed in April 1980.

## Environmental Qualification of Safety-Related Electrical Equipment

Safety systems are installed at nuclear plants to mitigate the consequences of postulated accidents. Certain of these postulated accidents could create severe environmental conditions inside the containment. The most serious of these accidents would be a high energy pipe break in the reactor coolant system piping or in a main steam line. In either case, the release of hot pressurized water and steam to the containment would create a high temperature environment (250 to 400°F) at high humidity (including steam) and pressure (as high as 50 psig). For some applications, fission product removal chemicals are added to the containment sprays that are used to reduce the pressure in the containment. Additionally, some electrical equipment is predicted to be submerged following a large pipe break.

In order to assure that electrical equipment in safety systems will perform its function under accident conditions, the NRC requires that such equipment—principally equipment associated with the reactor trip system, the emergency core cooling system, and the containment isolation and cleanup systems—be qualified to perform in the environment associated with the accident. Although such requirements have been applied to varying degrees since the early days of commercial nuclear power, they have come to be defined in clearer detail over the years.

The process of clarifying the criteria has given rise to certain questions regarding the adequacy of qualification tests or analyses. Generic Task A-24, "Environmental Qualification of Safety-Related Electrical Equipment," was established to address this question for newer plants.

"IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations," (IEEE std. 323) and its ancillary standards have provided the focal point for the development of environmental qualification requirements in recent years. These standards set forth basic requirements for environmental qualification of electrical equipment and provide varying degrees of detail for implementation of these requirements.

The staff requires in part that, for newer plants (specifically those for which a construction permit (CP) safety evaluation report (SER) was issued after July 1, 1974), the methods and programs developed to qualify safety-related equipment should conform to the requirements of IEEE 323-1974 and that this standard be used as a guide in evaluating these qualification programs. For plants for which a construction permit SER was issued prior to July 1, 1974, the staff has required that the qualification programs be developed in conformance with the guidelines established in IEEE 323-1971: "IEEE Trial-Use Standard: General Guide for Qualifying Class IE Electrical Equipment for Nuclear Power Generating Stations." This requirement has been applied on a case-by-case basis to older plants that have been or are currently undergoing an operating license OL review. Several ongoing staff actions related to evaluating the adequacy of environmental qualification of safety-related electrical equipment at operating plants are discussed below under "Other Technical Issues."

Several aspects of equipment qualification are being pursued at this time by the NRC staff and the nuclear industry on a generic basis, in order to achieve a more uniform implementation of requirements established in IEEE 323-1974. One such activity is a continuing process of revising and upgrading industry standards by providing more detailed guidelines for implementing the basic requirements. Another such activity is the development of NRC staff positions which address selected areas of the qualification issue. These positions are applicable only to plants that are or will be in the CP or OL review process and that are required to satisfy the requirements set forth in either the 1971 or 1974 version of the IEEE-323 standard. This activity was a part of Generic Task A-24, and was scheduled to be completed with the publication of interim NRC positions in NUREG-0588, in December 1979. It is anticipated that a supplemental report will be issued reflecting any changes to these interim positions which might result from investigations of the Three Mile Island Unit 2 accident, the staff's review of the responses to Bulletin 79-01 on operating plants, and the resolution of several issues that are currently being pursued by the NRC and the nuclear industry such as aging effects, sequential vs. simultaneous testing, etc. (see Chapter 2).

Further efforts under Generic Task A-24 were originally planned to involve the review of the environmental qualification methods actually used for qualifying safety-related electrical equipment, pursuant to the requirements of IEEE-Standard 323-1974. The staff had planned to perform these reviews on a generic basis, rather than on case-by-case licensing reviews, since this was likely to save time and resources for the NRC and the industry. However, the staff's initial attempt at performing these generic reviews indicated that it would be unproductive to review the methods without, at the same time, reviewing the way these methods are used when qualifying specific pieces of equipment. Therefore, these methods will be reviewed in conjunction with the review of the qualification programs submitted as part of the operating license review of the first plant applications (lead plants) required to comply with IEEE-323-1974. Since there will be a considerable gap in time between the issuance of the interim positions and the initiation of the first lead plant review, the scope of Task A-24 has been redefined to eliminate the reviews of specific qualification methods.

## Reactor Vessel Pressure Transient

For a number of years, incidents known as "pressure transients" occurred at various PWR facilities. A pressure transient occurs when the pressure-temperature limits included in the facility technical specifications for the purpose of protecting the reactor vessel from brittle fracture have been exceeded. There have been over 30 such events. Half of them occurred before the plant achieved initial criticality (i.e., before initiation operation of the reactor); the majority occurred during startup or shutdown operations. In all of these incidents, fracture mechanics and fatigue calculations indicated that the reactor vessels were not damaged and continued operation of the vessels was acceptable. Nevertheless, the staff concluded that appropriate regulatory actions were necessary (1) to reduce the frequency of pressure transient events, and (2) to provide equipment which would restrict future transients to acceptable pressures. This action was necessary because reactor vessel safety margins would be reduced over the lifetime of the vessel by neutron irradiation, which reduces material toughness. This matter is discussed in more detail above under "Reactor Vessel Material Toughness."

Upgraded procedural controls were implemented in early 1977 at operating PWR facilities to reduce the likelihood of reactor coolant system pressure transients. In addition, system design changes, such as added pressure relief capability during low temperature conditions, were also being implemented in the last two years. No pressure transient events occurred at operating PWRs during the report period.

Task A-26 involved the development of acceptable criteria for system design changes at operating plants and for use in the review of construction permit and operating license applications. All operating PWR licensees have completed an evaluation of their plant's reactor coolant system response to potential pressure transients and, where necessary, have submitted a description of proposed design changes.

NRC staff review and approval of the proposed design modifications continues. As of the end of the report period, 14 facilities have received staff approval. The remaining reviews are expected to be completed by mid-1980.

## Residual Heat Removal Shutdown Requirements

The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident (LOCA) has been typically interpreted as achieving a "hot-standby" condition (i.e., the reactor is shut down, but system temperature and pressure are still at or near their normal operating values). Considerable emphasis has been placed on the hot-standby condition of a power plant in the event of an accident or abnormal occurrence. A similar emphasis has been placed on long-term cooling, which is typically achieved by the residual heat removal (RHR) system. The RHR system can operate only when the reactor coolant pressure and temperatures have been reduced to substantially lower than their hot-standby condition values.

Even though it may generally be considered safe to maintain a reactor in a hot standby condition for a long time, experience shows events sometimes require eventual cooldown and long-term cooling until the reactor coolant system is cold enough to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for both PWRs and BWRs. It is essential that a power plant be able to go from hot standby to cold shutdown conditions (when this is determined to be the safest course of action) under any normal or accident conditions.

This issue was included in the NRC Program for Resolution of Generic Issues as Generic Task A-31, "RHR Shutdown Requirements." In accordance with the Task Action Plan for this task, the staff's views on requirements for residual heat removal systems were translated into proposed changes to Standard Review Plan Section 5.4.7. These proposals were considered by the NRC's Regulatory Requirements Review Committee (RRRC) during its 71st meeting on January 31, 1978. The RRRC recommended approval of the proposed changes and further recommended that: (1) the changes be applied on a case-by-case basis to all operating reactors and all other plants (custom or standard) for which the issuance of the operating license is expected before January 1, 1979, and (2) the changes be backfitted to all plants (custom or standard) for which construction permit or preliminary design approval applications were docketed before January 1, 1978, and for which the operating license issuance is expected after January 1, 1979. These recommendations were approved by the Director of NRR and are being implemented. Accordingly, Generic Task A-31 has been completed.

Standard Review Plan Section 5.4.7 has been modified to incorporate the approved changes which are presently being implemented through CP and OL reviews.

The staff positions on design requirements for residual heat removal systems were incorporated into Regulatory Guide 1.139, "Guidance for Residual Heat Removal," which was issued for public comment in May 1978. Comments were received during the latter part of 1978. Work on revision of the guide has been delayed but it is expected that it will be issued in final form by late 1980.

In addition, the staff has been reviewing 15 operating nuclear power plants to determine how well they meet the recommendations of Regulatory Guide 1.139. Eleven of these plants are included in the Systematic Evaluation Program (SEP). The remaining four plants (one representative plant for each of the four reactor vendors) are of more recent design than those included in the SEP. The review for the 11 plants in the SEP has been completed. Because of possible design and operational changes based on actions taken as a result of the Three Mile Island Unit 2 accident, the review of the other four plants has been postponed. These reviews are expected to be resumed and completed during calendar year 1980. On the basis of this review of representative plants, recommendations for implementation of Regulatory Guide 1.139 for all operating plants (except those included in the SEP) will be presented to NRC management for approval in calendar year 1980. Implementation of SEP findings, including those related to Regulatory Guide 1.139, will begin after the end of the Systematic Evaluation Program, scheduled for May 1982 as of the end of the report period.

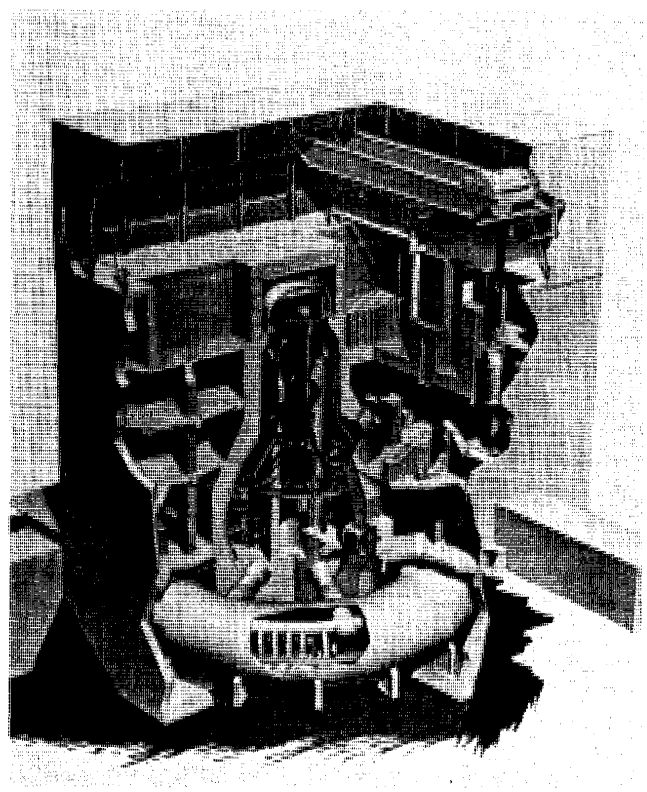
## Control of Heavy Loads Near Spent Fuel

Overhead cranes are used to lift heavy objects, sometimes in the vicinity of spent fuel, in both PWRs and BWRs. If a heavy object, such as a spent fuel shipping cask or shielding block, were to fall or tip onto spent fuel in the storage pool, or in the reactor core during refueling, and damage the fuel, there could be a release of radioactivity to the environment. If the dropped object were large and were assumed to drop on fuel containing a large amount of fission products with small decay time, calculated offsite doses could be high and could exceed the siting guideline values in 10 CFR Part 100.

The NRC staff's review of this safety issue has been incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-36. The objective of the task is to develop standard review criteria which will reduce the possibility that heavy loads might cause unacceptable damage to spent fuel in a storage pool or in the reactor core. The review included a detailed survey of design provisions and procedures currently used at operating plants and reevaluation of current NRC requirements.

Operating facilities use a variety of design and administrative measures to minimize the potential for dropping a heavy object over the reactor core or over the spent fuel pool. These design and administrative measures have been effective, since no heavy load handling accidents resulting in damaged fuel rods have occurred in over 400 reactor-years of U.S. operating experience. For facilities that have requested increases in spent fuel pool storage capacity, the NRC has imposed restrictions that prohibit the movement of any load over the fuel assemblies in the spent fuel pool that weighs more than the equivalent weight of one fuel assembly. Also, for those plants where the review of the cask drop or the crane handling system is not complete, movement of shielded casks over or near spent fuel has been prohibited. It is the NRC staff's view that continued operation during review of this generic issue, in compliance with the restrictions cited, presents no undue risk to the health and safety of the public.

As noted in the 1978 NRC Annual Report, licensees were requested to examine their current procedures



**This cutaway drawing illustrates a typical BWR reactor building layout, showing the overhead handling system used for movement of the spent fuel shipping cask and for removal and reinstallation of the reactor vessel head, moisture separators and steam dryers. Safe handling of such heavy loads is important to prevent damage to equipment or fuel. Task A-36 will result in upgraded NRC criteria to assure safe handling of heavy loads.**

for the movement of heavy loads over spent fuel to assure that the potential for a handling accident that could result in damage of spent fuel is minimized while the generic evaluation proceeds. In addition, the licensees were requested to provide information on load handling operations for use in the Task A-36 review. Responses were received from all licensees by December 1978.

The staff has completed its survey of load handling operations at operating plants, including design and procedural measures that prevent or mitigate the consequences of a heavy load handling accident and has prepared a draft report containing the NRC staff's resolution of this issue including revised criteria and other recommendations. This report is expected to be issued for public comment in January 1980. The report will provide the basis for revisions to the Standard Review Plan (SRP) and Regulatory Guides, if needed, that can be used in future reviews of new plants and will provide the basis for implementing additional requirements and procedures in operating plants.

Although Task A-36 will result in generic criteria, implementation of these criteria will be dependent on plant design characteristics and the specific procedures in effect at each particular plant, and will consequently require a plant-by-plant review.

## Seismic Design Criteria

NRC regulations require that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants is provided in the NRC regulations and in Regulatory Guides. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken (principally as part of the Commission's Systematic Evaluation Program) to assure that these plants do not present an undue risk to the public.

The NRC staff is conducting Generic Task A-40 as part of the NRC Program for Resolution of Generic Issues. Task A-40 is a compendium of short-term efforts to support the reevaluation of the seismic design of operating reactors, and to support licensing activity in general. The objective of the task is, in part, to investigate selected areas of the seismic design sequence to determine the conservatism for all types of sites, to investigate alternative approaches to part of the design sequence, and to estimate quantitatively the overall conservatism of the design sequence. In this manner the program will aid the NRC staff in performing its reviews of the seismic design of operating reactors.

The NRC Office of Nuclear Regulatory Research is also undertaking a related, but more comprehensive and long-term program to develop mathematical models to realistically predict the probability of radiactive releases from seismically induced events in nuclear power plants. This Seismic Safety Margin Research Program will utilize input from Task A-40 in a number of areas.

Generic Task A-40 is subdivided into two phases. Phase I includes a number of subtasks related to the response of structures, systems, and components to earthquakes. These subtasks include studies on: (1) quantifying conservatisms in seismic design, (2) electro-plastic seismic analysis methods, (3) site-specific response spectra, (4) nonlinear structural dynamic analysis procedures, and (5) soil structure interaction. These studies were performed under NRC-sponsored contracts and all were completed by October 1979. Review of the results of these studies is underway. The results will support the effort on seismic reevaluation of operating plants, particularly in the area of site-specific definition of seismic input. As of January 1, 1980, Phase I was scheduled to be completed in February 1980, with the issuance of recommendations for changes in the Standard Review Plan and Regulatory Guides in those seismic design areas related to response of structures, systems, and components to seismically induced events.

Phase II of Task A-40 includes several subtasks related to numerical modeling of earthquake motion at the source, analysis of near source ground motion, and attenuation of high-frequency ground motion. Studies under these subtasks being conducted by NRC contractors are scheduled for completion by the end of 1980. Review and implementation of the results of these studies in terms of recommended revisions to the Standard Review Plan and Regulatory Guides are scheduled for March 1981.

## Pipe Cracks at Boiling Water Reactors

Pipe cracking has occurred in the heat affected zones of welds in primary system piping in BWRs since the mid-1960s. These cracks have occurred mainly in Type 304 stainless steel that is being used in most operating BWRs. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components that have been made susceptible to this failure mode by being "sensitized," either by welding or by post-weld heat treatment. Although the likelihood is extremely low that IGSCC-induced cracks will propagate far enough to create a significant hazard to the public, the occurrence of such cracks is undesirable and measures to minimize IGSCC in BWR piping systems are indicated to improve overall plant reliability.

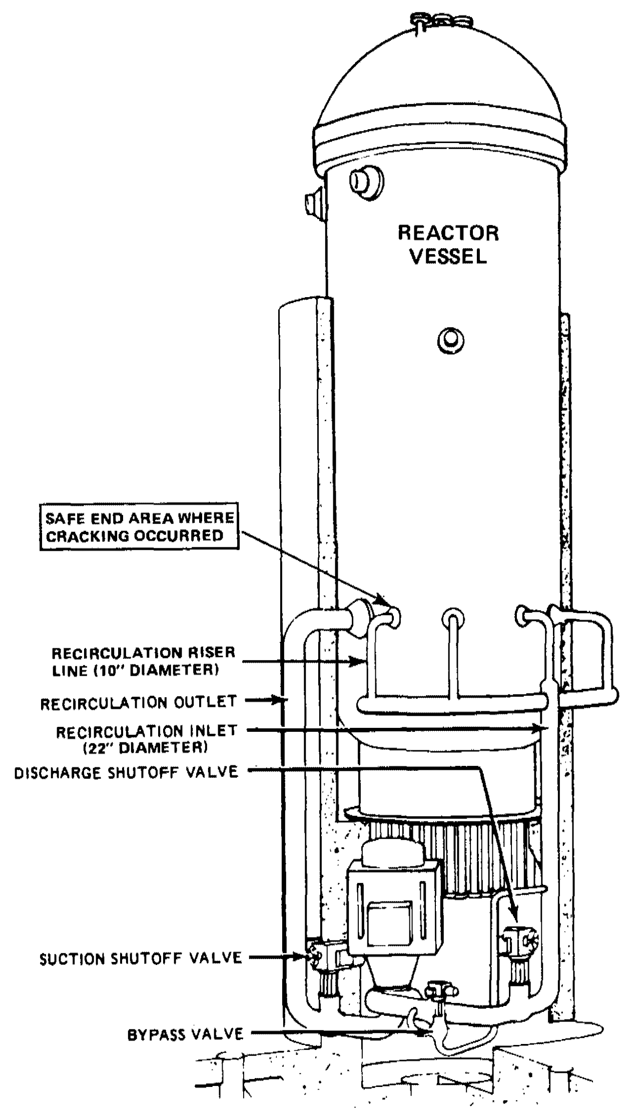
"Safe ends" (short transition pieces between vessel nozzles and the piping) that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication were found to be susceptible to IGSCC in the late 1960s. Because they were susceptible to cracking, the Atomic Energy Commission took the position in 1969 that furnace-sensitized safe ends in older plants should be removed or clad with a protective material, and there are only a few BWRs that still have furnace-sensitized safe ends in use. Most of these, moreover, are in small diameter lines and are subjected to augmented inservice inspection.

Earlier reported cracks (prior to 1975) occurred primarily in 4-inch diameter recirculation loop bypass lines and in 10-inch diameter core spray lines. More recently cracks were discovered in recirculation riser piping (12- to 14-inch) in all foreign plants. All these crack locations are part of the reactor primary system. Cracking is most often detected during inservice inspection using ultrasonic testing techniques. Some piping cracks have been discovered as a result of small primary coolant leaks.

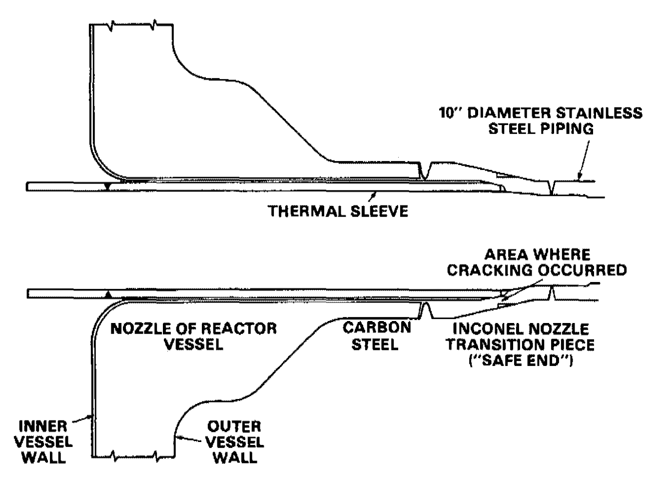
In response to these occurrences of BWR primary system cracking, a number of remedial actions were undertaken by the NRC. These actions included:

- Issuance of Regulatory Guide 1.44 on "Control of the Use of Sensitized Stainless Steel."
- Issuance of Regulatory Guide 1.45 on "Reactor Coolant Pressure Boundary Leakage Detection Systems."
- Closely following the incidence of cracking in BWRs, including foreign experience.
- Encouraging replacement of furnace sensitized safe ends.
- Requiring augmented inservice inspection of lines having less corrosion resistant stainless steel, especially those that have a high potential for cracking (service sensitive lines).
- Requiring upgrading of leakage detection systems.

More recently pipe cracking and furnace-sensitized safe end cracking have been reported in larger (24-inch diameter) lines in a GE-designed BWR in Germany with over 10 years of service. Because the safe ends in that facility had been furnace-sensitized during fabrication, IGSCC was suspected. As a result of concerns regarding these furnace-sensitized safe ends, a safe end was removed and subjected to destructive examination. During laboratory examination of the removed safe end, including a small section of attached pipe, cracks were discovered at various locations in the safe end and in the weld heat affected zone of the pipe. The cracks in the pipe weld area were very shallow with the maximum depth less than 5 mm (about 1/8-inch) in a wall thickness of about 1.5 inches. Cracking in the furnace-sensitized safe end, also having a wall thickness of about 1.5 inches, was




Cracks appeared at one facility in the so-called "safe ends" between pipes and vessel nozzles. This diagram shows the general location of the safe-end area (above), and detail of the cracking area (below).



somewhat deeper. The German experience was the first known occurrence of IGSCC in pipes as large as 24 inches in diameter.

In June 1978, a through-wall crack was discovered in an Inconel recirculation riser safe end (10-inch diameter) at the Duane Arnold facility. The crack has been attributed to IGSCC, although the material in this instance is different from the Type 304 stainless steel that has been historically found to be susceptible to IGSCC. Prior to safe end removal, ultrasonic examination showed several indications of possible cracks. Following their removal, cracking was discovered in all eight safe ends. The cracking appeared to have originated in a tight crevice between the inside wall of the safe end and the internal thermal sleeve attachment. Such crevices are know to enhance IGSCC. Differences in materials, geometry, stress levels, and crevices appear to make the problem at Duane Arnold unique to a particular type of recirculation riser safe end (Type I). As a result of this event, ultrasonic examination of the other Type I safe ends in U.S. BWRs (i.e., at the Brunswick 1 and 2 facility) was conducted. No significant indications of possible cracks were found in Unit 2 and one indication was identified at Unit 1. Although this latter indication was relatively minor and too small to be reportable pursuant to the NRC Regulations, periodic reevaluation of the Unit was deemed necessary. This ultrasonic indication at Brunswick Unit 1 was remeasured and reevaluated in the presence of NRC ultrasonic testing consultants at another plant shutdown in January 1979. It was concluded that: (1) there is no apparent change of this indication between inspections, and (2) although the existence of a very small localized area of cracking cannot positively be ruled out, the most likely cause of this indication is irregularities at the weld-to-base metal interface of the first bead weld at the thermal sleeve to safe end weld. This indication will be reexamined during the next refueling outage.

General Electric (the reactor vendor) has been asked to provide an in-depth report on the significance of recent events, including current inspection, repair, and replacement programs. They were also asked to address any new safety concerns related to the occurrence of cracking in large main recirculation piping. Based on information presented by General Electric to date and on extensive staff evaluation, the staff concluded that the recent occurrences do not constitute a basis for immediate concern about plant safety, nor require any new immediate actions by licensees.

Based on the earlier incidents of pipe cracking discussed above, the NRC formed a Pipe Crack Study Group to: (a) investigate the cause of cracks, (b) make interim recommendations for operating plants, and (c) recommend corrective actions to be taken for future plants. The Study Group published its report (NUREG-75/067) in October 1975, containing recommendations to reduce the incidence of IGSCC in sensitized stainless steel piping. Following staff review of the Study Group's recommendations, the staff issued an implementation document (NUREG-0313) which established staff positions consistent with the recommendations of the Study Group.

As a result of the more recent incidents, the NRC reestablished a second Pipe Crack Study Group on September 14, 1978. The new Study Group specifically addressed the following issues:

- The significance of the cracks discovered in large diameter pipes relative to the conclusions and recommendations set forth in the referenced report and its implementation document, NUREG-0313.
- Resolution of concerns raised over the ability of ultrasonic techniques to detect cracks in austenitic stainless steel.
- The significance of the cracks found in large diameter sensitized safe ends, and any recommendations regarding the current NRC program for dealing with this matter.
- The potential for stress corrosion cracking in PWRs.
- The significance of the safe end cracking at Duane Arnold relative to similar material and design aspects at other facilities.

The new Study Group completed its evaluation in February 1979 and issued a report, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants" (NUREG-0531). The new Study Group not only reaffirmed the conclusions and recommendations reached by the previous group in NUREG-75/067, but also presented some new ideas to reduce the potential for IGSCC and addressed the subject of IGSCC in safe ends. On March 13, 1979, NRC issued a Notice in the Federal Register soliciting public comments on NUREG-0531. After expiration of the public comment period and review of the Study Group's conclusions and recommendations, the staff initiated Task A-42. The work to be performed under Task A-42 was defined at that time as the development of an update to the implementation document, NUREG-0313, to incorporate the new Study Group's conclusions and recommendations and public comments received on NUREG-0531.

Revision 1 to NUREG-0313 was issued in October 1979, and public comments have been solicited on the report. Revision 1 sets forth the NRC staff's revised guidelines for reducing the IGSCC susceptibility of BWR piping. The guidelines describe a number of preventive and corrective measures acceptable to the NRC, including guidelines for: (1) corrosion resistant materials for installation in BWR piping, (2) methods of testing, (3) processing techniques, (4) augmented in-service inspection, and (5) leak detection. The report also included recommendations for developmental work to provide future improvements in limiting the extent of IGSCC or detecting it when it occurs.

## Containment Emergency Sump Reliability

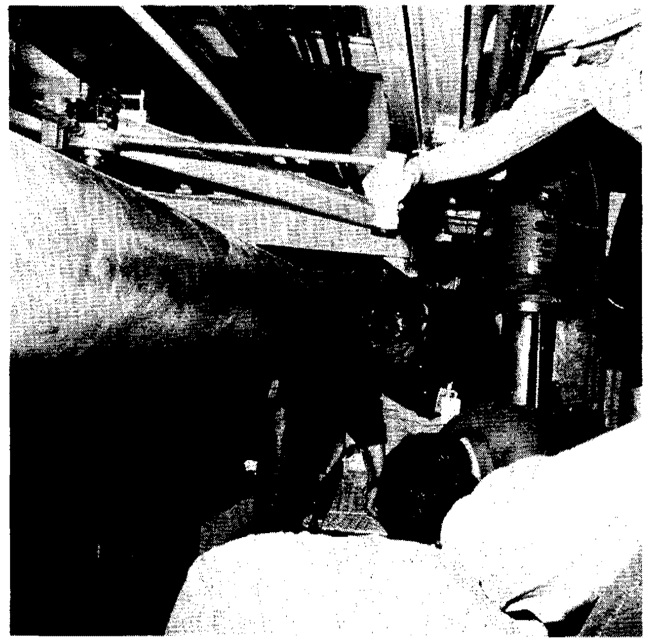
Following a postulated loss-of-coolant accident, i.e., a break in the reactor coolant system piping, the water flowing from the break would be collected in the emergency sump at the low point in the containment. This water would later be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. Loss of the ability to draw water from the emergency sump could therefore disable the emergency core cooling and containment spray systems. The consequences of the resulting inability to cool the reactor core or the containment atmosphere could be melting of the core and/or loss of integrity in the containment.

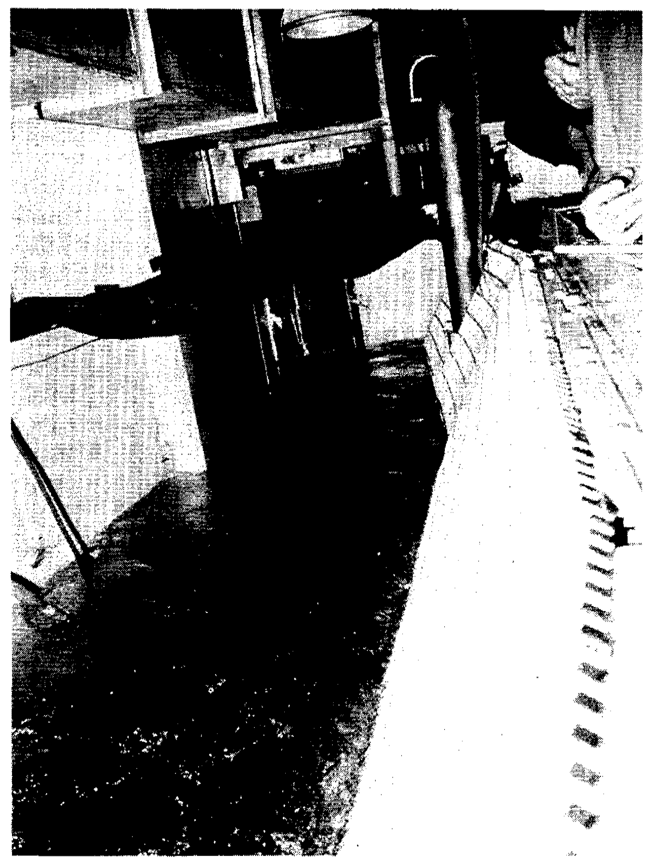
One potential way the ability to draw water from the emergency sump can be lost is from blockage by debris. A principal source of such debris could be the thermal insulation normally installed on the reactor coolant system piping. In the event of a piping break, the subsequent violent release of the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. The loose insulation material could then be swept into the sump and block it.

A Task Action Plan was under development in March 1979 when the Three Mile Island Unit 2 accident disrupted work on it. As of January 1, 1980, the Task Action Plan was nearing completion. Nonetheless, several technical studies related to sump reliability which were already underway will either be incorporated into Task A-43 or will provide input into Task A-43 efforts.

A study program investigating PWR vortex technology has been completed by the Iowa Institute of Hydraulic Research and a technical report issued. A summary report of NRC experience with containment sump testing is being prepared. This summary will be issued as a NUREG report in 1980. Based on the Iowa study program and the review of tests, NRR expects to draft interim positions on sump design guidelines and preoperational test requirements in early 1980. Criteria for the evaluation of operating containment sumps will be formulated at about the same time.

Finally, a program is being sponsored by the Department of Energy, in cooperation with NRC, to aid in resolving this issue as part of their Light Water Safety Research Program. This is an experimental pro-





NRC staff members traveled to the North Anna Power Station Unit 1 in Virginia to conduct evaluations of the emergency recirculation sump as part of its work on Generic Issues Task A-43. The photo at the top shows project personnel looking down into the recirculation sump area. Above, the reactor containment area was deliberately flooded to permit observation of flow patterns, blockage of pipes, types of debris, etc.

gram, begun in July 1979 at Alden Research Laboratory, Worcester Polytechnic Institute, to study the hydraulic aspects of containment sump operation. The program will continue through February 1981.

It is anticipated that this task can be completed in 1982.

### Station Blackout

Electrical power for safety systems at nuclear power plants is supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these electric power supply requirements. Each electrical division for safety systems includes an offsite alternating current (a.c.) power connection, a standby emergency a.c. power supply (usually one or more diesel generators), and direct current (d.c.) sources.

The issue of station blackout involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all a.c. power, i.e., a loss of offsite a.c. sources and all onsite emergency diesel generator sources. Loss of all a.c. for an extended period of time in pressurized water reactors, accompanied by loss of the auxiliary feedwater pumps (usually one of two redundant pumps is a steam turbine driven pump that is not dependent on a.c. power for actuation or operation), could result in an inability to cool the reactor core, with potentially serious consequences. If the auxiliary feedwater pumps are dependent on a.c. power to function, then a loss of all a.c. power for an extended period could of itself result in an inability to cool the reactor core. Although this is a low probability event sequence, it could be a significant contributor to risk.

Current NRC safety requirements require as a minimum that diverse power drives be provided for the redundant auxiliary feedwater pumps. As noted above, this is normally accomplished by utilizing one or more a.c. power electric motor driven pumps and one or more redundant steam turbine driven pumps. One concern is the design adequacy of plants licensed prior to adoption of the current requirements.

The degree of dependence of decay heat removal systems on a.c. power supplies and their reliability with a total loss of a.c. power has recently been reviewed for a large number of plants. For some plants, modification in design and/or operating procedures were recommended in the short term. This evaluation was carried out using simplified analytical techniques.

A Task Action Plan for Task A-44 was under development in March of 1979 when the Three Mile Island Unit 2 accident disrupted work on this task. As of January 1, 1980, the Task Action Plan was still under development. It is anticipated that the task can be completed in 1982.

Under Task A-44 a more detailed and comprehensive assessment will be performed for both PWRs and BWRs. Preliminary scoping work indicates that this should include consideration of: (1) the failure modes that can result in a station blackout, (2) the probability and frequency of occurence of a blackout including site variability and time dependence, (3) the potential consequences of a blackout, and (4) potential preventative and mitigating actions. The results of this effort will be used to determine if changes to licensing criteria are necessary and, if so, to develop criteria for use in the review of CP and OL applications and for evaluating operating plants.

## OTHER TECHNICAL ISSUES

### Design Errors in Control Building

In the spring of 1978, Portland General Electric Company (PGE), operator of the Trojan Nuclear Plant, reported design errors in the control building walls, i.e., conditions at variance with the design criteria set forth in the Final Safety Analysis Report for the facility and incorporated into its operating license.

A detailed NRC staff review of PGE's investigation and analysis of the design revealed the following errors:

- The steel reinforcement in the reinforced concrete core of the walls was permitted to be generally discontinuous and, therefore, the concrete core could not be relied upon to resist shear (in case of an earthquake) to the extent assumed in the approved design.
- The shear capacity of the reinforced concrete and grouted masonry block was not correctly computed.
- The steel reinforcement needed to resist shear beyond the capacity of the concrete and grouted masonry block was computed incorrectly, resulting in a lower level of conservatism than intended.

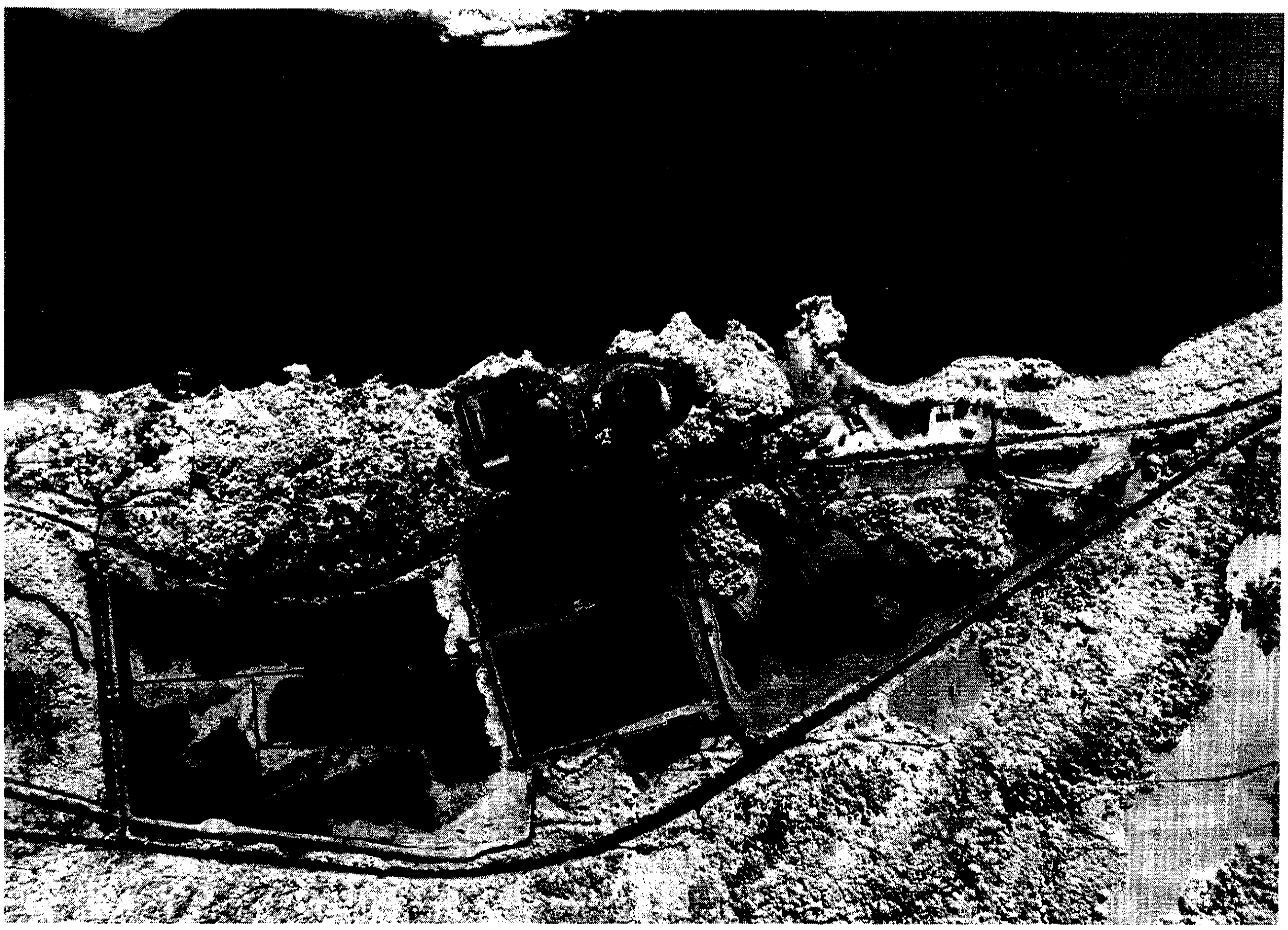
A detailed reevaluation of the control building in its existing configuration was performed by PGE to assess the capability of the structure to withstand the Operating Basis Earthquake and the Safe Shutdown Earthquake. The NRC staff determined that there had been a significant reduction in conservatism and design margins, with respect to the control building seismic capability, below the level intended and desired for the 33 years remaining in the expected plant life and that the margins should be appropriately restored by plant modification. PGE indicated its intent to make such modifications.

The NRC staff also determined that, despite the design errors, the affected structures remained qualified to withstand the licensed safe shutdown earthquake and that there was adequate assurance of safety to permit continued operation in the interim without endangering the health and safety of the public, provided that no modifications were made that would in any way reduce the strength of the existing shear walls. The NRC staff also concluded that, since the OBE capability had been reduced, actions that would otherwise be required for a 0.15g earthquake would have to be taken in the event of an 0.11g earthquake.

On May 26, 1978, the NRC issued an Order dealing with this matter. The Order called for:

- Design modifications to restore the seismic design margins originally intended to the control building.
- An implementation schedule, to be reviewed and approved by the NRC.
- Detailed design information for NRC staff review and approval, together with supporting analyses and application for license amendments as necessary to implement these modifications.
- Conditional license waiver of the areas of nonconformance noted above until the control building has been brought into substantial compliance in these areas. The conditions spelled out were that no modifications affecting the strength of the control building shear walls were to be made without NRC approval and the facility should be brought to cold shutdown in the event of a 0.11g earthquake at the site and that subsequent restart would require prior NRC approval.

Numerous requests for a hearing were received, and a hearing was ordered by an Atomic Safety and Licen-



**The Trojan Nuclear Plant, situated on the Oregon side of the Columbia River, has been the subject of an NRC staff review concerning the ability of its control building to withstand earthquakes. Based on an NRC order, Portland General Electric Company (PGE), Trojan owners, submitted plans to modify the building during 1979, and a hearing was set for April 1980. PGE had notified NRC of the deficiencies in the spring of 1978.**

sing Board (ASLB). Accordingly, a hearing on interim operations was held October 23 to November 3, and December 11 to 14, 1978. On December 21, the ASLB issued a Partial Initial Decision that authorized interim operation, with conditions, pending further hearings on the nature of modifications to the control building to bring it into substantial compliance with the requirements of the operating license. The conditions prohibited any modification that would reduce the strength of existing shear walls; required plant shutdown in the event an earthquake exceeding 0.08g should occur at the site; and required modifications to some pipe supports and restraints prior to resuming operation in order to ensure qualification of related piping systems to earthquake levels up to the Safe Shutdown Earthquake (0.25g).

A conforming amendment was issued on December 22, and plant operation resumed on December 30, 1978.

On January 17, 1979, PGE filed a report describing the proposed modifications to the control building. These changes consist of the addition of three shear walls in the existing railroad bay, and the strengthening of the west shear wall by the addition of three-inch thick steel plate. The design report submitted by the company proved to be preliminary in nature and incomplete in some respects. Because of this, extensive staff questions seeking design details, supporting analyses and justification for design assumptions were submitted to the licensee in March, May, July and August of 1979. Less than adequate responses to some of these questions caused a delay in issuance of the staff's Safety Evaluation Report on the modification (scheduled for issuance in September 1979) and postponement of hearings on the modification which had been scheduled for October. It also prompted additional staff questions to the licensee in September and October. The hearing on the acceptability of these modifications has been scheduled to commence on April 1, 1980.

While not directly connected with the original control building design deficiencies described above, another problem in masonry wall design was identified by PGE in their Licensee Event Report 79-15 of November 4, 1979. Some walls were found to have inadequate structural strength to sustain support reactions from attached piping. This led to concerns over the load-carrying capability not only of the wall supporting significant piping loads but also of other walls, and other elements supporting piping and equipment. A detailed investigation of all masonry walls and other structural elements was conducted, which resulted in modifications to 127 piping supports attached to block walls and several modifications to other structural elements. These corrective actions were completed on December 27, 1979, and NRC review of the matter was concluded on December 31, 1979.

## PWR Feedwater Line Cracks

On May 20, 1979, Indiana and Michigan Power Service informed the NRC of cracking in two feedwater lines at the D. C. Cook Unit 2. Leaking circumferential cracks were identified in the 16-inch lines in the immediate vicinity of the steam generator nozzles. Subsequent volumetric examination (by radiography) revealed crack indications at similar locations in all feedwater lines of both Units 1 and 2. As a result of a letter sent to all PWR licensees and the issuance of Bulletin 79-13, inspections have disclosed piping cracks, crack-like indications or fabrication defects requiring repair in the vicinity of the feedwater nozzles at 16 out of 25 Westinghouse PWR facilities. To date, eight Combustion Engineering (CE) and two Babcock & Wilcox facilities have been inspected. The two B&W facilities were found to be free of cracks in the feedwater piping to steam generator nozzle weld vicinity. Cracks were found on two of the eight CE facilities inspected.

The mode of failure at the majority of facilities where cracking has occurred has been identified as fatigue assisted by corrosion. The principal cracking at these facilities has been located in counter-bored areas of the piping at which stress risers, caused by machining to obtain fit-up for welding, are present. Very shallow cracks have also been identified in the nozzle at several units.

Repairs at the affected facilities have been made in accord with improved design and fabrication practices. In addition, the NRC staff has required that preservice and augmented inspections be performed.

The NRC staff concluded that these measures were adequate pending the outcome of test programs in progress. The staff has advised licensees that other remedial measures may be required at a later date, if results from test programs and/or from further evaluations by the staff show them to be warranted.

Reanalyses of normal piping system stresses and visual inspections of the feedwater lines have not, to date, uncovered any anomolies that would be expected to cause cracking. No significant deviations from proper feedwater chemistry control have been discovered. Thermal stresses, both high- and low-cycle, which could occur because of the mixing of hot and cold water in the nozzle region during hot functional testing, hot standby, start-up and shutdown are suspected as major factors contributing to the incidence of cracking. Test results from the instrumentation placed on the affected plants have shown that thermal stratification does exist at start-up, hot-standby and shutdown conditions. A Utilities Owner's Group has been formed to conduct test programs to identify the cause or causes of cracking and to find a long-term solution to the problem.

The NRC has instituted an Action Plan regarding the PWR Feedwater Cracking problem. The following items are included in the Action Plan:

(1) Consequence of Cracks—This item included identification of possible challenges to pipe integrity, and consideration of system effects in terms of the health and safety of the public.

(2) Failure Investigation: Cause of Cracking—This item included (a) independent metallurgical analyses to identify the mode of failure; and (b) review and independent stress analyses to verify the licensees' conclusions.

(3) Review PWR Designs and Operations—This item includes review of the piping layout, fabrication and inspection history, preservice and inservice performed, and evaluation of the operating histories.

(4) Evaluation of the Remedial and Corrective Actions—This item includes review of the repairs effected; the evaluation of the efficacy of the repairs in light of the results of the test results from the Westinghouse PWR facilities; and evaluation of design changes and operating procedures as a long term solution to the problem.

The NRC staff has considered the safety significance of the feedwater piping cracks and has concluded, based on the available information, that the most severely degraded piping found to date (D. C. Cook Unit 2) would be unlikely to rupture in the event of an earthquake, though it might leak. The staff considers it conceivable that a severely degraded feedwater line may rupture from a severe water hammer event. However, it is considered unlikely that a severe water hammer event would occur in two or more feedwater lines simultaneously. Thus, the worst consequence to be reasonably expected in a facility with degraded feedwater piping which experienced a severe water hammer would be the rupture of a single feedwater line. Because this event has been considered as a design basis accident, the facilities are designed to protect against the occurrence and to contain and control its consequences.

## Mechanical Operability of Containment Purge Valves

In November 1978, the NRC requested that all licensees of operating reactors respond to generic concern about containment purging or venting during normal plant operation. The generic concerns involved both electrical and mechanical aspects of containment purge valve operability. First, events had occurred in which licensees overrode or bypassed the saftey actuation isolation signals to the containment isolation valves. These events were determined to be abnormal occurrences and reported to Congress in January 1979. Second, recent licensing reviews have required a demonstration, by test or analyses, that containment purge or vent valves would shut without degrading containment integrity during the dynamic loading imposed by a postulated design accident, a loss of coolant accident (LOCA).

The November 1978 request emphasized the importance of the mechanical qualification of purge and vent valves. For quick closing capability most facilities use butterfly valves for containment isolation.

The concern centered around the capability of the containment isolation valves in the purge and vent systems—after being opened during hot standby, hot shutdown, startup or power operation modes—to have the capability to close against the fluid dynamic conditions of a LOCA upon receipt of an isolation signal. These fluid or aerodynamic forces originate from the pressure drop imposed across the closing valves by the ascending pressure in containment following a LOCA. Typically these valves will receive an isolation signal to close either from high radiation monitors, high pressure monitors, or low reactor water level for BWRs.

Potential failures affecting the purge and vent valves could lead to degradation in the containment integrity and, for PWR's, a degradation in ECCS performance.

From staff studies and discussions with manufacturers involved in supplying these valves, the following conclusions can be made:

(1) Most valves of this type will tend to close under the dynamic forces of a LOCA.

(2) Partial opening of the valves between 30° and 50° of full open will in most cases significantly reduce dynamic loads put on valve components.

(3) Demonstration of operability for most valves of this type can be obtained through analysis and previous testing data.

Guidelines for demonstration of purge and vent valve operability have been developed and issued to all licensees and will be used to assess the valves installed in operating plants. This effort is scheduled to be completed by mid-fiscal year 1981. In the interim, licensees have been asked to limit the opening of their valves to between 30° and 50° of full-open and to limit the use of these valves until such time as long term operability can be demonstrated. It is anticipated that some systems modification may be required in attaining these goals.

## Environmental Qualification of Safety-Related Electrical Equipment At Operating Plants

In the fourth quarter of calendar year 1978, the staff conducted inspections of the activities of licensees of all operating reactors, in response to a staff issued circular, dated May 31, 1978, concerning the qualification of safety-related electrical equipment. The inspections disclosed that the review and resolution of problem areas encountered by the licensees in this effort were not receiving sufficient attention. Also identified were certain components for which documentation was lacking as to which were found qualified and which were found unqualified for their intended service. In order that this subject be given the proper emphasis, the staff issued Bulletin 79-01, dated February 8, 1979, to licensees of all operating reactors facilities. The bulletin required written responses from the licensees, within 120 days, regarding the results of their reviews of environmental qualification of all safety-related electrical equipment. Further, it required that the licensees report to the staff, within 24 hours of discovery, any equipment determined to be unqualified for its service conditions. A supplement to this bulletin, dated June 6, 1979, was also issued providing feedback to the licensees regarding deficiencies in the environmental qualification of certain pilot solenoid valves.

In response to the bulletin and its supplement the staff has received several "24-hour reports" (23 separate reports from 29 different nuclear plants) involving seven different types of equipment—(1) limit switches mounted on safety-related valve stems to indicate stem position; (2) containment isolation valve motor operators; (3) instrument and control cable insulated terminal lugs; (4) aluminum limit switch housings on containment isolation valves; (5) ASCO pilot solenoid valves for miscellaneous valve air operators; (6) terminal blocks enclosed in boxes; and (7) interconnecting wires. In each instance where an item of equipment was determined to be unqualified, the staff immediately evaluated the impact on the health and safety of the public and the adequacy of the remedial steps proposed by the licensees. In some cases, the licensees elected to replace the unqualified equipment immediately; in others, a basis for continued operatión pending corrective action at a future date was provided. In those cases where the licensees proposed to continue to operate the plant for a period of time before shutting down and replacing the affected equipment, the following factors were considered in the staff evaluations of whether the plants could continue to be operated safely: (1) redundant/diverse components available to perform the required safety functions; (2) locking of the affected component in its safety position; (3) administrative actions and revised operating procedures; (4) additional operability tests and inspections; (5) post-accident mitigating actions available; and (6) fail safe design features. In all cases where continued operation was requested by the licensees, based on a plant-specific safety evaluation, the staff has concluded (contingent, in some cases, upon additional staff requirements being satisfied) that the plants could continue to be operated safely.

An NRC task group has been formed to review in detail the licensees' responses to Bulletin 79-01. The reviews will be conducted in accordance with guidelines being prepared specifically for evaluating the qualification of Class IE equipment in operating reactors. The guidelines will address all of the significant aspects of the most current industry standard for Class IE electrical equipment qualification, IEE Std. 323-1974.

The Office of Nuclear Reactor Regulation has prepared guidelines for the Office of Inspection and Enforcement to use in identifying safety-related electrical equipment installed in operating reactors whose documentation does not provide reasonable assurance of environmental qualification.

## Pipe Support Base Plate Problem

During the review of pipe support designs at the North Anna Units 1 and 2 and Shoreham Unit 1, a potential design deficiency was discovered in the modeling assumptions used to calculate pipe support anchor bolt loads. Inspections at Shoreham also revealed deficient installations of a large number of the anchor bolts. During June 1978, the NRC initiated audits of engineering firms involved in the design of pipe supports to determine the potential generic extent of the problem in operating plants. As a result of these audits and the review of operating experience related to pipe support failures, the NRC issued Bulletin 79-02 on March 8, 1979, to all operating plants and to all plants with construction permits. The bulletin requires all licensees to re-calculate anchor bolt loads using appropriate modeling assumptions and to inspect the anchor bolt installations to insure that anchorage systems have an adequate margin of safety for the imposed loads.

Preliminary review of licensee responses has been completed. Modifications to most plants should be completed by March 1980. Several operating plants are making substantial modifications to their existing anchor bolts to meet the requirements of the bulletin. The staff has reviewed the existing design margins and has determined that sufficient margin exists with respect to the functional capability of the affected safety systems during a seismic event to allow operation while these modifications are being made.

## Nonconformance of Actual Installation of Safety-Related Piping Systems To Design Documents

During seismic reanalyses of safety-related piping with acceptable computer codes (see "Shutdown and

Seismic Reanalysis of Five Operating Reactors" below), inspection of several piping systems by NRC and its licensees identified substantial differences between design documents and the as-built condition of the piping and supports. Some examples are:

- Surry 1—Mislocated supports, wrong support type, and different pipe geometry.
- Fitzpatrick—Supports installed which were not accounted for in the analyses, cracked welds on support components, and bent rod hangers.
- Brunswick 1 and 2—Undersized pipe supports.
- St. Lucie 1—Missing seismic supports and mislocated supports.

In order to obtain accurate and valid results, the data used as input to the required pipe stress analyses and associated support evaluations, e.g., pipe size, geometry, support location and type, must reflect, within construction tolerances, the "as-built" condition of the plant. Specific seismic reanalysis results for Surry 1, Fitzpatrick, and Beaver Valley 1 demonstrated that when the analyses reflected the as-built condition of the facilities, piping stresses, nozzle, penetration, and support loads exceeded their allowable values. Several support modifications, additions, and deletions were necessary to the facilities' piping and supports in order to show compliance with the originally intended design criteria.

The magnitude and type of discrepancies discussed above were discovered at 11 power reactors, indicating that the problem may be generic and could substantially lessen the ability of a large number of facilities to adequately withstand the effects of a seismic event. Therefore, on July 2, 1979, the NRC issued Bulletin 79-14, addressing seismic analyses for as-built safety related piping systems. Revision 1, which exempts from the requirements of this bulletin all two-inch and smaller piping not computer analyzed, was issued July 18, 1979. Additional guidance was provided by the issuance of supplements to the bulletin on August 15 and September 7, 1979.

Bulletin 79-14 requests that all licensees verify that the seismic analysis input information accurately reflects the facility as actually constructed. Licensees were requested to inspect piping geometry, piping support and restraint designs, pipe attachments, and valve and valve-operator locations and weights. The bulletin requests that licensees establish an ad hoc inspection program to be completed within 120 days. Further, the bulletin requires that licensees resolve specific nonconformances by either making changes to the system so that it conforms to design or by correcting the erroneous seismic analyses to demonstrate conformance of the as-built system to design criteria. It also requires that licensees take action to correct administrative problems which could allow this problem to recur.

Because of the extensive effort required—involving highly qualified stress analysts and related engineering disciplines—the time for completion of the requirements of the bulletin has been extended beyond 120 days. Most facilities are scheduled to complete the requirements by April 1980.

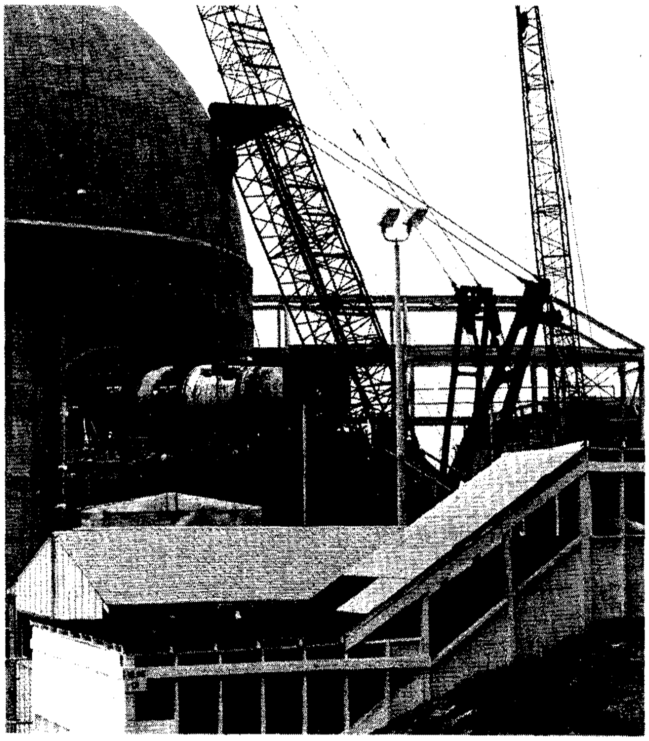
The NRC is assessing the results of the continuing inspections by licensees. Although not yet complete, the inspections conducted to date indicate that most facilities will be required to make some equipment changes. Most of the operating plants have already modified or added piping supports because of deviations found between the existing, as-built, equipment and design documents. Several plants, including Fort St. Vrain, Millstone Unit 2, D.C. Cook Unit 1 and Rancho Seco, have shut down for various lengths of time, in compliance with technical specification requirements, as a result of discrepancies discovered during the inspections.

Several architect-engineer firms and licensees have been audited to discuss and examine their procedures in the implementation of bulletin requirements, and more audits are planned for the future.

## Shutdown and Seismic Reanalysis of Five Operating Reactors

The Nuclear Regulatory Commission staff ordered five plants to shut down on March 13, 1979, until reanalysis and necessary modifications were made to safety-related piping systems, in order to bring them into conformance with requirements for withstanding earthquakes. The plants ordered shutdown were; Beaver Valley Unit 1, James A. FitzPatrick, Maine Yankee and Surry Units 1 and 2.

Stone and Webster Engineering, the architect-engineer for these five plants, and Duquesne Light Company, the licensee for the Beaver Valley facility, had earlier reported to the NRC, during a meeting on March 8, 1979, that they had discovered that an algebraic summation method has been used to combine seismic forces in the computer code called SHOCK II. The algebraic summation method can result in cancellation of seismic forces and thus may result in the prediction of stresses significantly lower than would be predicted by NRC-approved techniques. Following the meeting on March 8, members of the NRC staff met for three days with Stone & Webster Engineering officials in Boston. Additional analyses of piping systems for the Beaver Valley facility were performed. These analyses indicated significant overstress in the piping systems under postulated earthquake conditions when computer codes were used which did not combine seismic loads algebraically. Piping systems involving the integrity of the reactor coolant pressure boundary, emergency core cooling systems and safe shut down systems were involved. It



The Virginia Electric Power Co. began repairs on the steam generator of Surry Power Station, Unit 2, including the replacement of the lower sections of all three steam generators with new, improved sections. The effort will take about one year to complete. Shown in this photograph is one of the sections being removed through the equipment hatch of the reactor containment. NRC is arranging to obtain one of the steam generator sections for use in its research and development programs.

was also determined that the same computer code (SHOCK II) was used in the design of four other facilities. The NRC staff ordered all five plants shutdown because there was no assurance that a severe earthquake at any of these facilities would not cause an accident, damage emergency core cooling systems, and prevent safe shutdown of the plant.

The required reanalysis and necessary modifications were completed for Maine Yankee and Beaver Valley and orders were issued permitting resumption of operation, on May 24, 1979 and August 8, 1979, respectively. Sufficient reanalysis and modifications were completed for FitzPatrick and Surry Unit 1 to permit orders, issued on August 14, 1979, and August 24, 1979, respectively, allowing resumption of operation for 60 days while some remaining pipe support analyses were completed.

Surry Unit 2 was shut down for steam generator repair and replacement prior to the March 13, 1979, shutdown order. Because of the long shutdown for steam generator work, the seismic reanalysis required by the order was delayed by the licensee. It was not anticipated that the required seismic reanalysis would lengthen the plant shutdown.

Several actions have been taken by the NRC staff related to review, evaluation and approval of computer codes used for seismic analysis of saftey-related piping. Following issuance of the show cause orders, a computer code verification program was initiated by the staff, with three principal parts: (1) review of actual computer code listings, (2) solution of NRC benchmark problems to compare results to known values, and (3) independent check analyses of piping problems using NRC's own computer code. In addition, the NRC staff reviewed the development of the mathematical model which represents the piping system.

On April 13, 1979, the Florida Power and Light Company, licensee for Turkey Point Units 3 and 4, reported that algebraic summation techniques had been used by Westinghouse in the design of the main reactor coolant system piping. The NRC reviewed the results of Westinghouse's reanalysis, determined that piping design was acceptable, and permitted resumption of operation of both Turkey Point units. However, as a result of this information, an IE Bulletin was issued on April 14, 1979, requiring all licensees to review the computer codes used in the design of safety-related systems to determine if algebraic summation had been used. A total of 24 additional plants had used an algebraic summation technique. Four of these plants were still under construction and had not yet been issued operating licenses. The computer codes identified were:

SHOCK II—Stone & Webster Engineering

WESTDYN—Westinghouse

DAPS—General Electric

PIPDYN II—Franklin Institute

ADLPIPE—Arthur D. Little Company

The NRC staff has required reanalysis of all affected piping, with modification as necessary, and computer code verification for those codes used for reanalysis. The majority of the 25 operating reactors used algebraic summation methods on very few piping systems and had reanalyzed these systems prior to responding to the bulletin. In a few cases (Pilgrim 1, Brunswick 1 and 2, Indian Point 3 and Salem 1) the use of algebraic summation was more extensive. One unit, Salem 1, shut down since March 31, 1979, for refueling and other modifications, did not resume operation until November 13, following resolution of the algebraic summation issue. All other units have been resolved completely or, based upon NRC staff evaluation, have been permitted to continue operation during reanalysis. In each case where continued operation was permitted (Brunswick 1 and 2 and Indian Point 3), the analysis methods used and the margin allowed in the piping design were such that modification to piping systems was unlikely. The staff, however, required detailed reanalyses to confirm that the designs were acceptable.

## Petition for the Seismic Reanalysis Of Operating Reactors

On March 28, 1979, the Union of Concerned Scientists (UCS) filed a petition which proposed that the NRC require all plants with an operating license to perform a seismic reanalysis within a 120-day time period. The proposed reanalysis consisted of reevaluation of: (1) the magnitude of the safe shutdown earthquakes; (2) the free-field ground motion at the site; (3) the motions of the structures during a seismic event; (4) the plant equipment motions during a seismic event; (5) the seismic loads on structures, systems and components in appropriate combinations with other loads, and the corresponding allowable loadings; and (6) the conformance of the "as-built" plant to the design specifications. This petition was issued based upon concerns arising from issues surrounding the five-plant shutdown, including the lack of rigorous computer code verifications, and concerns regarding the evolution of seismic design criteria through the years.

Currently acceptable seismic design requirements for nuclear plants are delineated in Title 10 CFR, the NRC Standard Review Plan, and associated Regulatory Guides and engineering codes and standards.

There are many variations in the parameters used to define the ground motion imparted by an earthquake. In evaluating the seismic hazard to a plant for a given definition of the ground motion, a detailed engineering evaluation is conducted considering ground motion, foundation/structure interaction, piping system equipment, and component response. The uncertainties in the various steps of the overall analysis and design lead to conservative assumptions being made in each step.

The NRC has four programs underway to assess the seismic design adequacy of operating nuclear plants:

- Bulletins have been issued to each licensee requiring evaluations and, if needed, hardware modifications to affected plant systems. These items to be evaluated include: (1) verification of the desired safety margins in piping supports anchored by expansion bolts to concrete structures; (2) verification that the seismic analysis of safety-related piping systems was based on acceptable summation methods of seismic loading components; (3) verification that as-installed piping systems and supports are essentially the same as used in seismic analysis and design.
  As the reviews proceed, the NRC will take any necessary actions deemed appropriate. The review to date indicates that some installation and design deficiencies exist; these are being resolved in a timely manner.
- The Systematic Evaluation Program (see discussion later in this chapter) includes review of the seismic design of 11 older operating plants. No major deficiencies in the seismic design which would affect public safety have been identified. Several issues have been identified which will require more detailed studies, and possibly retrofitting, to verify the adequacy of the seismic design to meet the intent of current design criteria.
- The Seismic Design Criteria study (Task Action Plan A-40) is a short-term program to provide generic, quantitative estimates of the conservatisms in selected individual parts of and the overall seismic design when following current criteria.
- The Seismic Safety Margins Research Program, expected to continue for about six years, is oriented toward improvements in seismic design methodologies. These studies are being carried out concurrently with and will extend beyond the Seismic Design Criteria short-term effort.

The NRC response addressing in detail the concerns voiced in the UCS petition was in preparation at the close of the report period. (The UCS petition was denied by the Director of Nuclear Reactor Regulation in January 1980.)

## Humboldt Bay

(The background to licensing problems associated with this facility can be found in the 1977 NRC Annual Report, pp. 26–27, and the 1978 NRC Annual Report, pp. 49-50.)

The Humboldt Bay Power Plant (Cal.) has been shut down since July 2, 1976, because of unresolved geologic/seismic concerns regarding local geologic fault definition and the potential for surface faulting at the site. Since 1976, the Pacific Gas and Electric Company (PG&E) licensee for the facility, has conducted extensive geologic investigations and plant seismic modifications.

A request for hearing with respect to resumption of operation of the facility was submitted by representatives of individuals from the Humboldt Bay area. The NRC Atomic Safety and Licensing Board (ASLB) has directed that such a hearing be held. On May 7 and June 19, 1979, the ASLB issued separate but related orders to the licensee requiring that it provide the board and interested parties progress reports on the status of ongoing geologic investigations. The licensee is providing such information bi-monthly and estimates that the earliest practicable date for proceeding with a hearing is October 1, 1980. At the close of the report period, the ASLB was considering the licensee's motion to hold the proceeding in abeyance until October 1980.

## Possible Faulting Near Reactor

On October 24, 1977, the NRC staff ordered that the General Electric Test Reactor (GETR) at the Vallecitos Nuclear Center in California be shut down. This action was based on evidence, revealed by field investigations, of possible faulting (Verona Fault) at the GETR site. Since surface faulting was not a design basis for the GETR facility and the appropriate seismic design ground acceleration was in question, the licensee was directed to show cause why the suspension of operation should not be continued.

Following attempts to argue that the Verona Fault did not exist, the licensee conducted an extensive field-trenching investigation. This investigation revealed additional evidence of faulting at the site. In its report on these investigations, submitted February 28, 1979, the licensee concluded that the origin of observed offsets was probably large-scale landsliding but could also be the result of earthquake faulting. Furthermore, the licensee proposed a zero surface offset design basis for the reactor building and no more than three feet displacement on observed offsets.

By letter dated September 27, 1979, the NRC staff issued its evaluation of the seismology and geology of the GETR site. The NRC staff concluded that a surface offset of two and a half meters could occur beneath the GETR. This is in excess of the one meter surface offset to which the GETR facility has been analyzed by the licensee.

This issue will be subject to further reviews by the NRC Advisory Committee on Reactor Safeguards and the Atomic Safety and Licensing Board.

## Fire Protection

Following the fire at the Brown's Ferry Plant in March 1975, the NRC initiated a review of the fire protection programs for all operating plants and for plants not yet operational. Improved guidelines have been developed and are being implemented. Minimum requirements for specific aspects of fire protection for operating plants are being added to 10 CFR 50. The fire protection program reviews have been completed for the 70 licensed power plants and modifications to improve plant capabilities are being implemented. The modifications to most plants will be made by late calendar year 1980.

On November 4, 1977, the Union of Concerned Scientists (UCS) filed a Petition for Emergency and Remedial Action. Part of this petition dealt with fire protection concerns at plants under construction and at operating plants.

The Commission issued an order on April 13, 1978, denying the UCS petition on the basis that plants under construction or in operation are in compliance with General Design Criterion 3—Fire Protection.

On May 2, 1978, the UCS submitted a petition requesting that the Commission reconsider its April 13, 1978, decision on the earlier petition filed on November 4, 1977. The Commission took this petition under consideration, and was reviewing public and staff comments, developed as a result of the reconsideration, at the close of the report period.

## Control Rod Guide Tube Integrity

In December 1977, extensive wear and some holes were observed in the upper section of numerous control rod guide tubes at Northeast Nuclear Energy Company's Millstone Unit 2 facility (Connecticut). Subsequent inspections at other facilities with reactors designed by Combustion Engineering (CE) disclosed similar indications of guide-tube wear. The cause was found to be flow-induced vibration of fully withdrawn control rods. The rod tips, vibrating against the guide tubes, induced degrading wear, probably aided by corrosion.

The safety significance of the incidents is related to the functions of the guide tubes. They serve both as fuel assembly structural members and as channels for control rod movement. Thus, a guide-tube failure could adversely affect either the preservation of a coolable core geometry or the scram (shutdown) capability of the control rods, or both.

The observed wear of the guide tubes thus far has been confined to facilities designed by Combustion Engineering (CE). There are basic differences between the CE design of the control rod systems, which insert into the guide tubes of the fuel assemblies, and the other designs (Westinghouse and Babcock & Wilcox). These design differences appear to have reduced the severity of wear on the guide tubes in the latter vendors' facilities. However, such wear in Westinghouse and Babcock & Wilcox plants and in Exxon Nuclear fuel assemblies is under investigation by the NRC staff.

To overcome the susceptibility to wear of the guide-tube material (Zircaloy-4) and to recover the design margin lost by wear, CE designed stainless steel sleeves for use in the guide tubes. Prior to installation of stainless steel sleeves during a refueling outage, operators of CE reactors instituted the practice of inserting the control rods three inches further into the core than the normal fully withdrawn position. That action both reduced the local wear intensity and provided added assurance of scram capability. NRC approval was granted for this short term administrative procedure allowing continued operation.

The use of sleeved guide tubes was approved by the NRC as an interim repair to mitigate the guide-tube wear on a cycle-specific basis. In conjunction with the use of the stainless steel sleeves, the staff required that inspection programs be submitted for review and approval well in advance of refueling shutdowns.

Additional out-of-reactor hot loop testing by CE showed the important role of flow-induced vibration of the control rods in the guide-tube wear problem. The vibration and, hence, the wear, was reduced by decreasing some of the guide-tube coolant (water) flow. Two fuel assembly modifications were designed to reduce the coolant flow. One involved inserting a splined cylinder in the top of the guide tube. The second involved reducing the size and number of flow holes in the bottom of the guide tube. Both modifications, in limited number, are being tested in currently operating cores to confirm the loop test results.

The NRC has closely followed the analyses and experiments performed by CE and is in substantial agreement with the vendor that the results point to control rod flow-induced vibration as the principal factor in guide-tube wear. Therefore, design modifications intended to alter flow in the guide-tubes were judged appropriate. The NRC has approved the modified designs for limited operation on the basis that they will mitigate the wear problem. Approval of either design modification as a final solution to the problem will be contingent upon the results of further out-of-reactor experiments and examination of the modified assemblies which are currently subject to in-reactor operations.

The first opportunity to evaluate the performance of the sleeved guide tubes after reactor operations occurred during the Millstone Unit 2 refueling outage in the spring of 1979. Subsequent to the Millstone 2 refueling, the St. Lucie Unit No. 1 (Florida) and the Calvert Cliffs Unit No. 1 (Maryland) also provided evidence on the performance of the sleeved guide tubes. These inspections indicate that the sleeving modification has performed well as an interim solution to mitigate the guide-tube wear but that it does not eliminate the cause of the wear. (During the October-November 1979 refueling outage Calvert Cliffs Unit 2 was scheduled to undergo inspections of modifications made as interim solutions to guide tube wear.)

The NRC staff will continue to maintain close liaison with representatives of the licensees and vendors on this issue and any related problems. Approvals have been granted to allow operaton of the CE plants on a cycle-specific basis with the stainless steel inserts. All proposed programs have been reviewed prior to taking action at any facility, and the staff has required that all inspection programs continue to be submitted for review well in advance or refueling shutdowns.

## PROGRESS IN STANDARDIZATION

The NRC believes that standardization of the design of nuclear power plants is in the interest of public health and safety, and of effective and efficient regulation. Thus, the NRC is committed to the support and expanded use of standardization within the Commission's regulatory activities.

Four procedural options are available (see 1976 NRC Annual Report, p. 36, for details) to applicants for standardization of nuclear power plants: "Reference Systems" (approved design used repeatedly by reference), "Duplicate Plants" (approved design for several identical plants), "License to Manufacture"(approved design for manufacture of identical units at the central location), and "Replicate Plants" (reuse of recently approved custom design).

Since June 1973, when applications were first accepted which included a standardization option, the standardization program has realized substantial progress. Overall, approximately two-thirds of the applications received in the 1974-1978 time frame have employed one or more options of the standardization program. See Table 3 for a listing of the status of applications.

In August 1978, the Commission approved a number of changes to the program to encourage its expanded use, as well as to incorporate both industry and regulatory changes introduced since the program was first announced in 1972. The revised program adopted a good many such changes, some of which are as follows:

(1) The term of holders of all new preliminary design approvals (PDAs) for reference system designs was extended from three to five years. Holders of all issued PDAs were given the opportunity to extend them to a full 5-year term.

(2) Final design approvals (FDAs) for reference system designs were eligible for reference in applications for construction permits. Two types of FDAs were established. The first, denoted FDA-1, can be referenced from the time it is docketed to 3 years after expiration of the PDA on which it is based. The second, denoted FDA-2, can be referenced from the time it is docketed to 5 years after it is approved.

(3) A qualification review was devised to permit the duplicated plant concept to be used in a manner similar to the reference system concept. In this regard, five-year preliminary duplicate design approvals (PDDAs) and final duplicate design approvals (FDDAs) were established which can be used in new applications for construction permits in a manner similar to the use of PDAs and FDAs under the reference system concept.

(4) A qualification review was defined for replicate plants and the period for replication was established as 3 years after publication of the base plant Safety Evaluation Report.

(5) A 5-year period of design approval was established for manufacturing licenses and an uppper limit of 10 units was established.

## Table 3. Standardization Applications

(as of August 31, 1979)

| PROJECT | APPLICANT | DOCKET DATE | COMMENTS |
|---|---|---|---|
| **Reference Systems** | | | |
| *Nuclear Island* | | | |
| GESAR-238(NI) | General Electric | 7/30/73 | Nuclear Island, PDA-1 (Preliminary Design Approval) issued 12/22/75 |
| *Turbine Island* | | | |
| C F BRAUN SSAR | C.F. Braun | 12/21/74 | Turbine Island Matched TO GESSAR-238(NI). PDA-5 Issued 5/07/76 |
| **Nuclear Steam Supply System (NSSS)** | | | |
| BSAR-205 | Babcock & Wilcox | 3/01/76 | PDA-12 issued 5/31/78 |
| BSAR-241 | Babcock & Wilcox | 5/14/74 | (withdrawn) |
| CESSAR | Combustion Engineering | 12/19/73 | PDA-2 issued 12/31/75 |
| GASSAR | General Atomic | 2/05/75 | Review suspended at request of applicant. |
| GESSAR-238 | General Electric | 10/16/75 | PDA-10 issued 3/10/77 |
| GESSAR-251 | General Electric | 2/14/75 | PDA-9 issued 3/31/77 |
| RESAR-3S | Westinghouse | 7/31/75 | PDA-7 issued 12/30/76 |
| RESAR-41 | Wesinghouse | 3/11/74 | PDA-3 issued 12/31/75 |
| RESAR-414 | Westinghouse | 12/30/76 | PDA-13 issued 11/14/78 |
| **Balance of Plant (BOP)** | | | |
| BOPSSAR/BSAR-205 | Fluor Power | 10/31/77 | BOP matched to BSAR-205 |
| BOPSSAR/RESAR-41 | Fluor Power | 1/27/76 | PDA-11 issued 8/17/77 BOP matched to RESAR-41 |
| ESSAR/BSAR-205 | Ebasco | 5/19/78 | BOP matched to BSAR-205 |
| ESSAR/CESSAR | Ebasco | 2/02/78 | BOP matched to CESSAR |
| ESSAR/RESAR-414 | Ebasco | 11/23/77 | BOP matched to RESAR-414 |
| GAISSAR/BSAR-205 | Gilbert Commonwealth | 8/21/78 | BOP matched to BSAR-205 |
| GAISSAR/CESSAR | Gilbert Commonwealth | 8/21/78 | BOP matched to CESSAR |
| GAISSAR RESAR-414 | Gilbert Commonwealth | 8/21/78 | BOP matched to RESAR-414 |
| GIBBSSAR | Gibbs & Hill | 5/10/77 | BOP matched to RESAR-414 |
| SWESSAR/BSAR-205 | Stone & Webster | 12/22/75 | BOP matched to BSAR-205 |
| SWESSAR/CESSAR | Stone & Webster | 10/21/74 | BOP matched to CESSAR PDA-6 issued 8/16/76 |
| SWESSAR/RESAR-3S | Stone & Webster | 10/02/75 | BOP matched to RESAR-3S BPDA-8 issued 3/31/77 |
| SWESSAR/RESAR-41 | Stone & Webster | 6/28/74 | BOP matched to RESAR-41 PDA-4 issued 5/05/76 |

| PROJECT | APPLICANT | DOCKET DATE | COMMENTS |
|---|---|---|---|
| **Utility Applications Using Reference Systems** | | | |
| Cherokee 1,2&3 | Duke Power | 5/24/74 | References CESSAR. CP issued 12/30/77 |
| Perkins 1,2&3 | Duke Power | 5/24/74 | References CESSAR |
| South Texas 1&2 | Houston Light and Power Co. | 7/05/74 | References RESAR-41 CPs issued 12/22/75 |
| WPPSS 3&5 | Washington Public Power Supply System | 8/02/74 | References CESSAR CPs issued 4/11/78 |
| Palo Verde 1,2&3 | Arizona Public Service | 10/07/74 | References CESSAR. CPs issued 05/25/76 |
| Hartsville 1,2,3&4 | Tennessee Valley Authority | 11/22/74 | References GESSAR-238(NI) CPs issued 05/09/77 |
| Palo Verde 4&5 | Arizona Public Service | 03/31/78 | References CESSAR |
| Black Fox 1&2 | Public Service of Oklahoma | 12/23/75 | References GESSAR-238 (NSSS) |
| Phipps Bend 1&2 | Tennessee Valley Authority | 11/07/75 | References GESSAR-38 CPs issued 1/16/78 (NI) |
| Erie 1&2 | Ohio Edison Co. | 3/01/77 | References BSAR-205 |
| Yellow Creek 1&2 | Tennessee Valley Authority | 3/16/76 | References CESSAR |
| **Duplicate Plants** | | | |
| Byron 1&2 | Commonwealth Edison | 9/20/73 | Two units at each of two sites. CPs issued 12/31/75 |
| Braidwood 1&2 | | | |
| Cherokee 1,2&3 | Duke Power | 5/24/74 | Three units at each of two sites. Also references CESSAR. Cherokee CPs issued 12/30/77. |
| SNUPPS | | | Five units at four sites. |
| Wolf Creek | Kansas Gas & Electric Co. Kansas City Power & Light | 5/17/74 | CP issued 5/17/77 |
| Callaway 1&2 | Union Electric | 6/21/74 | CPs issued 4/14/76 |
| Tyrone 1 | Northern States Power | 6/21/74 | CPs issued 12/27/77 |
| Sterline | Rochester Gas & Electric | 6/21/74 | CP issued 9/01/77 |
| *WNP* Koshkonong 1&2 | Wisconsin Electric Power Madison Gas & Electric Wisconsin Power & Light Wisconsin Public Service | 8/09/74 | Initially submitted under duplicate plant option with intent for as many as six total units at three sites. Utility's change in plans led to removal from standardization program by staff. Review discontinued because of site problems |
| **License to Manufacture** | | | |
| Floating Nuclear Plant (FNP) 1-8 | Offshore Power Systems | 7/05/73 | Entire plant design |
| **Replication** | | | |
| Jamesport 1&2 | Long Island Lighting | 9/06/74 | Replicates Millstone 3 |
| Marble Hill 1&2 | Public Service of Indiana | 9/17/75 | Replicates Byron 1&2 |
| New England 1&2 | New England Power & Light | 9/09/76 | Replicates Seabrook 1&2 |
| Palo Verde 4&5 | Arizona Public Service | 3/31/78 | Replicates Palo Verde 1,2&3 |
| Haven 1 | Wisconsin Electric Power | 4/05/78 | Replicates Koshkonong 1&2 |

Staff studies (NUREG-0427) have shown that the NRC standardization program is about at the break-even point, that is, the staff resources spent on the review of standardization plants and design approval applications is about equivalent to the resources that would have been used if only custom plants had been involved. To the extent that utilities reference approved designs in the future, the balance will become more and more favorable for the standardization program. On the other hand, should the staff be requested to review additional PDA's and new applications that do not reference PDA's, FDA's, or ML's (Manufacturing Licenses), the use of standardization to reduce the use of staff resources would not be realized.

Staff studies also have revealed that use of the standardization options have not, to date, resulted in a reduction of schedules. These studies show that the potential exists for significant schedule reductions only when there is preapproval of the Nuclear Steam Supply System (NSSS), the Balance of Plant (BOP), and the site, the three review areas that separately can define the critical path. Thus, a strong incentive exists for pursuing site approvals via the Early Site Review Program, since approved PDAs now exist for the NSSS and BOP portions of the plant. Utility-related matters of the application, such as the quality assurance program or the financial qualifications, generally do not control the overall review schedule.

Program actions completed during fiscal year 1979 included: (a) extending Balance-of-Plant PDAs to a full 3-year term; (b) extending six PDAs to a full 5-year term based upon a completeness review; and (c) issuing a PDA for RESAR-414. Additional reviews and policy initiatives were temporarily suspended in April 1979 as a result of the TMI-2 accident. Staff resources were re-directed to high priority activities associated with the accident-related studies.

## ADVANCED NUCLEAR POWER PLANTS

On April 7, 1977, President Carter issued a statement on Nuclear Power Policy which restated the role that nuclear energy was to have in the total energy prospects of the country. The President's policy would defer indefinitely the commercial reprocessing and recycling of plutonium produced in nuclear power reactors, restructure the U.S. breeder reactor program to give high priority to alternative designs, and defer the time when breeder reactors are to be commercialized.

During this reporting period, the NRC has continued its participation in the review and assessment of a variety of reactor types and fuel cycles being considered by the Department of Energy (DOE) as part of the Nonprolifereation Alternative Systems Assessment Program (NASAP); it also continued performing reviews and providing comments on the studies and assessments being performed under the International Nuclear Fuel Cycle Evaluation (INFCE) program. In its reviews and comments, the staff focused on the potential licensability of these reactor types and associated fuel cycles, with respect to safety and safeguards concerns and environmental acceptability.

Based on advanced reactor licensing experience and preliminary safety documents supplied by DOE, the staff prepared its initial comments on alternative reactors and fuel cycles and forwarded them to DOE in June 1979. These initial findings are summarized in the first of a series of reports to Congress published in October 1979.

### Clinch River Breeder Reactor

The status of the staff review of the Clinch River Breeder Reactor remained inactive throughout the year and will remain so pending enactment of legislation clarifying the status of the facility.

### Fast Flux Test Facility

The Fast Flux Test Facility (FFTF) is a major LMFBR test facility which, with a power of 400 megawatts (thermal), will provide an intense field of fast neutrons for irradiating fuels and materials in connection with advanced reactor research and development. The facility, which is located about 10 miles north of Richland, Washington, is owned by the Department of Energy (DOE) and is not subject to licensing by the NRC. An NRC staff safety review was performed, however, under terms of an interagency agreement with DOE. The staff completed the major part of its review effort and, in August 1978, issued its Safety Evaluation Report (NUREG-0358). A supplement to the SER (NUREG-0358, Supplement No. 1) was issued in May 1979. Sodium filling of one secondary sodium loop took place in July 1978. Fuel loading was expected in October 1979. Prior to full power operation, now scheduled for early 1980, a series of tests was to be performed to determine whether natural circulation is a viable method of removing decay heat as predicted by analyses.

The Advisory Committee on Reactor Safeguards (ACRS) was extensively involved in the review of FFTF and meetings addressing that review were held in July, August, September and November 1978. The ACRS concluded that the startup and operation of the FFTF is acceptable, provided that due regard is given to NRC consequences of certain low probability accidents, and other specified matters. DOE is presently evaluating the NRC staff recommendations regarding containment adequacy for low probability accidents.

## Gas-Cooled Reactors

As a consequence of the withdrawal of the General Atomic Company from the commercial nuclear power market in late 1975, regulatory activities related to gas-cooled reactors have been confined primarily to the Fort St. Vrain reactor. Limited reviews of advanced high-temperature gas-cooled reactors and of a gas-cooled fast breeder reactor have been undertaken in conjunction with the NRC's participation in the NASAP study.

**Fort St. Vrain.** Fort St. Vrain, a 330-MWe high-temperature gas-cooled reactor (HTGR), was designed by the General Atomic Company and is operated by the Public Service Company of Colorado near Platteville, Colorado. Transfer of ownership to Public Service was made in June 1979. Power level is restricted to 70 percent of initially rated power pending resolution of the fluctuation problem described on page 40 of the 1978 NRC Annual Report.

**Advanced High-Temperature Gas-Cooled Reactors.** In early 1978, a group of utilities formed an organization, Gas Cooled Reactor Associates (GCRA), for the purpose of developing a commercially viable advanced HTGR. GCRA manages the DOE funds supporting the project and is responsible for carrying out initial phases of the licensing review. In early 1979, a decision was made to terminate work on a standardized 900 MWe steam cycle plant in favor of working toward the demonstration of the gas turbine cycle in the mid-1990's. NRC review of this concept is being performed under NASAP auspices.

**Gas-Cooled Fast Breeder Reactor.** In late 1976, an organization of utilities, Helium Breeder Associates (HBA), was formed to work with both the General Atomic Company and DOE (then the Energy Research and Development Administration) toward the development and demonstration of the Gas-Cooled Fast Breeder Reactor (GCFR). The GCFR demonstration unit would produce 330 MWe. Both DOE and HBA have accented General Atomic's revised reactor design that would permit emergency core cooling by means of natural convection. This concept is now being reviewed under NASAP auspices.

## Floating Nuclear Power Plants

Floating nuclear power plants (FNPs) are electrical generating stations of a standardized design which would be constructed at a shipyard facility using assembly line techniques. The proposed FNPs would utilize a conventional pressurized light water reactor system design mounted on floating platforms, similar to the hull of a barge, and can be sited at offshore or nearshore sites in the ocean or in estuaries and rivers. Offshore Power Systems (OPS), a subsidiary of Westinghouse Electric Corporation, filed an application with the NRC in 1973 for a license to manufacture up to eight identical floating nuclear power plants at Blount Island near Jacksonville, Fla.

An NRC staff Safety Evaluation Report (NUREG-75/100) was issued in September 1975; Supplement No. 1 (NUREG-0054) was issued in March 1976 and Supplement No. 2 in October 1976. It was anticipated that Supplement No. 3 will be issued in early 1980.

The staff has also prepared a three-part Final Environmental Impact Statement (FES) to assess the potential impacts from the construction, siting and operation of FNPs. Part I, issued in October 1975, relates to the construction and nonnuclear testing of the FNPs at the manufacturing site in Florida. Part I concluded that foreseeable adverse impacts from manufacturing the FNPs would be acceptable in consideration of the benefits expected from the plants and therefore recommended that a manufacturing license be issued, subject to certain license conditions. Part II, issued in September 1976, relates to the potential impacts associated with siting, constructing, and operating FNPs at generalized unspecified locations offshore in the Atlantic Ocean, Gulf of Mexico and in certain rivers and estuaries. At the request of the Council of Environmental Quality, the NRC issued an Addendum to Part II of the Final Environmental Statement, in June 1978, which elaborated upon the data and analyses presented in Part II with respect to the estuarine and riverine siting of FNPs. The staff concluded there was reasonable assurance that eight FNPs could be sited, constructed and operated with acceptable environmental impact at offshore sites along the Atlantic Ocean and Gulf of Mexico and at carefully selected shoreline locations, including estuarine waters. The U.S. Environmental Protection Agency, however, believes that it will be difficult to find environmentally acceptable sites in any of the estuarine or barrier island areas along the East and Gulf Coasts.

Part III of the FES, issued in December 1978, compared the total risk to the environment from accidental releases of radioactivity for both floating and land-based nuclear power plants. A wide spectrum of accidents were considered including, for the first time, low-probability, core-melt accidents (Class 9) in the liquid pathway. Part III also included an overall cost-benefit analysis for all elements of the environmental statement and concluded that a manufacturing license should be issued subject to several license conditions, including a specific license condition to mitigate the potential environmental impacts from a core-melt accident at an FNP. This involves the use of a material such as magnesium oxide beneath the reactor vessel in order to retard the penetration by the melting core through the bottom of the FNP hull.

Part III also listed NRC generalized requirements for compliance by a utility/operator of an FNP when an application is made to the NRC for locating an FNP at a specific site. These include modification of FNP sites in estuaries, rivers or near barrier islands so as to limit the release of radioactive materials into the surrounding water body following the unlikely event of a core-melt accident. A principal reference used in the preparation of the FES, Part III was the *Liquid Pathway Generic Study* (NUREG-0440), issued in February 1978.

Public hearings on safety and environmental issues were started in March 1975 and continued through 1979. Offshore Power Systems appealed the staff's precedent-setting decision to include Class 9 accidents in the comparative analyses of the FES. In December 1978, the Commissioners agreed to review whether Class 9 accidents were a proper subject for treatment in the environmental impact statement. On September 14, 1979, the Commission issued a *Memorandum and Order in the Matter of Offshore Power Systems* which stated the Commission's position that the staff's analysis of the Class 9 accident question is properly included in the environmental impact statement in this proceeding in order to meet NRC's NEPA responsibilities. Both the applicant and the staff have submitted partial proposed findings of fact to the licensing board and all safety and environmental contentions have been addressed during the hearing process. Additional hearings were held in late 1979 in order to discuss the licensing board's questions regarding the staff's Class 9 analysis.

The first application for a permit to construct and operate an offshore floating nuclear power station was filed in 1973 by the Public Service Electric and Gas Company (PSE&G) of New Jersey. The proposed Atlantic Generating Station (AGS) consisted of two floating units (1150 MWe each) located approximately three miles off the coast of New Jersey and about 11 miles northeast of Atlantic City. In December 1978, PSE&G cancelled its contract with OPS, citing among its reasons the lower than anticipated electricity growth rate in its generating area. The application has been withdrawn and the licensing proceeding dismissed.

## PROTECTING THE ENVIRONMENT

### Health Effects of the Coal and Nuclear Fuel Cycles

As noted in the 1978 Annual Report, the NRC is actively involved in developing estimates of potential effects of the coal and nuclear fuel cycles to aid in the analysis of alternative energy sources for generating electricity. Final revision of the draft report, "Health Effects Attributable to the Coal and Nuclear Fuel Cycles" (NUREG-0332), is being held in abeyance pending release of the latest National Academy of Sciences Report of the Committee on Nuclear and Alternative Energy Sources (CONAES). A contracted study with the Argonne National Laboratory on health effects models for the nuclear and coal fuel cycle alternatives is nearing completion.

In November 1979, the staff issued a report prepared under contract by Teknekron, Inc., entitled, "Activities, Effects and Impacts of the Coal Fuel Cycle" (NUREG-CR-1060). The report provides a current data base related to the health, ecological, economic and social impacts of the coal fuel cycle. The report considers the impacts resulting from the various phases of the coal fuel cycle: resource extraction, coal cleaning, transportation, storage, power production and waste disposal.

### Assessment of Radiological Consequences Of Radionuclide Releases

By means of *Federal Register* notice of January 13, 1977 (42 FR 2858), the Environmental Protection Agency officially issued 40 CFR Part 190, *Environmental Radiation Protection Standards for Nuclear Power Operations.* The standards require that operations covered by the subpart B, *Environmental Standards for the Uranium Fuel Cycle,* shall have no planned discharges that will result in an annual dose equivalent to any member of the public that will be in excess of 25 millirems to the whole body. Other requirements involve organ doses and releases of specific radioisotopes. The standards are effective as of December 1, 1979, except that for doses arising from operations associated with the milling of uranium ore, the effective date is December 1, 1980. For releases of iodine-129 and krypton-85, the standard will be effective January 1, 1983. The standards may be exceeded during a given year of operation only if the regulatory agency has granted an exemption based on a determination that a temporary and unusual operating condition exists and that continued operation is in the public interest.

The NRC has been developing provisions to be incorporated in license conditions requiring that NRC licensees meet the conditions of Part 190. Most nuclear power plants that meet the requirements on radioactive effluents promulgated by Appendix I to 10 CFR Part 50 have been shown generically to meet Part 190. To assure full compliance, the model Radiological Effluent Technical Specifications (RETS), contained in NUREGs-0482 and -0473, have been modified to include Part 190 requirements as a limiting condition for operation. Staff documents (NUREGs) further describing acceptable methods for demonstrating compliance with Part 190 are in progress. The Office of Nuclear Material Safety and Safeguards is developing license conditions for the uranium fuel cycle facilities under its cognizance. The Office of Standards Development is preparing modifications for Title 10 regulations and for regulatory guidance documents that will further identify the requirements that Part 190 places on NRC licenses.

## Control of Effluents

**Standard Technical Specifications.** As a result of the staff's continuing review and discussions with the Atomic Industrial Forum and other parties of interest, substantive revisions were made to the NRC draft reports on "Radiological Effluent Technical Specifications" (NUREG-0472 for PWRs, and NUREG-0473 for BWRs). The revised reports still incorporate the fundamental requirements and concepts contained in the original, but equations for dose calculations, setpoint determination, and meteorological dispersion factors have been eliminated. These equations, among other items, are now required to be included in an Offsite Dose Calculation Manual (ODCM) that is to be provided by each licensee to NRC for review and approval along with the proposed Technical Specifications. Regional seminars were held in late 1978 to provide guidance in the preparation of the Technical Specifications. All affected utilities were invited to send representatives to these seminars. Licensees are in the process of submitting their proposed Technical Specifications and review of these by the staff was in progress at the close of the report period. Licensee submissions and NRC reviews should be completed by mid-1980.

**In-Plant Measurements Program.** In order to assure that the best available data is employed in improving the calculational models used to appraise conformance of licensees with Appendix I to 10 CFR Part 50, the NRC contracted with Idaho National Engineering Laboratory to perform in-plant measurements on pressurized water reactors (PWRs). The measurements will provide a data base for radidoisotope inventories in plant systems, radioactive waste management system performance, and source term for both liquid and gaseous systems. As of the end of the report period, measurements had been completed at four plants (Zion, Fort Calhoun, Turkey Point, and Rancho Seco).

**Three Mile Island Accident Response.** The performances of the effluent treatment systems following the onset of the TMI accident were evaluated and the amounts of released radioactivity have been assessed. Recommendations were made for operating procedures involving the use of existing equipment and the installation of new equipment needed to assure that releases of radioactivity during this emergency period would be kept at levels as low as possible under the circumstances and to remain within established NRC effluent standards (see Chapter 2).

During the long-term recovery period for the TMI plant, the TMI-2 Support Task Force of the NRC will continue to review all matters related to maintenance of the reactor in a safe shutdown condition, decontamination of equipment and buildings, installation of new equipment and systems, the processing of liquids contaminated from the accident for removal of radioactivity, the storage and shipment of radioactive wastes, and releases of low levels of radioactivity to the environment. Safety evaluations and environmental assessments for the more significant recovery operations are being prepared. These activities are being coordinated with local, State, and other Federal officials.

## Site-Related Problems

**Rejection of Greene County Site Due to Esthetic Impacts.** It was during 1979 that the NRC for the first time rejected a proposed nuclear site primarily because of adverse socioeconomic impacts. The staff concluded that the proposed Greene County plant at the Cementon, New York site would result in unacceptable esthetic impacts on certain local, regional, and national historic, scenic, and cultural resources. The major reason for that judgment was the visual intrusion of plant facilities—primarily the natural-draft cooling tower and its plume—into the central view from Olana, the home of 19th Century painter Frederick Church, which had been designated as a National Historic Landmark. Other visually sensitive areas that could be adversely affected were also identified. The staff analyzed the esthetic impacts from alternative cooling systems but determined that even the least visually obtrusive alternative would still be likely to have undesirable effects. The staff also identified severe (although generally mitigable) socioeconomic impacts arising from the potential loss of a local industry because of the facility's land needs and from the need to substantially change the local transportation network to serve the facility. The staff concluded that there are several alternative sites in New York State that, on the basis of environmental considerations, are obviously superior to the proposed site.

**Mass Mortality of Biota.** NRC continues to monitor potential environmental problems arising from operating nuclear power plants. In the summers of 1978 and 1979, a sizeable number of weakfish (also known as sea trout) were drawn onto the intake screens at the Salem Nuclear Station. At the Oyster Creek Station in August 1979, unusually high temperatures resulted in the apparent loss of a small but still significant number of several species of fish. The staff is working closely with EPA in reviewing the facts associated with these occurrences to determine whether corrective action should be taken.

**Water Quality Monitoring Requirements.** In order to ascertain the environmental consequences of power plant licensing, NRC is placing increasing reliance on EPA's permit system, a result of the National Pollution Discharge Elimination System (NPDES). A major step to avoid the confusion and inequity resulting from regulation of the aquatic environment by two Federal agencies was taken with the closely coordinated review of TVA's Yellow Creek Nuclear Station Construction Permit Application. As a consequence of the Yellow Creek Proceeding, which suggested that this approach was not only desirable but legally necessary, the NRC staff is striving to obtain EPA or State agency resolution of questions pertaining to water quality that may arise during NRC's environmental review.

**Implementation of Executive Order 11988 on Floodplains.** By an Executive Order issued in May 1977, President Carter called upon Federal agencies to consider any action they undertake affecting the nation's floodplains as an opportunity to reduce the impact of floods on human safety, health and welfare, and to restore and preserve the natural and beneficial values served by floodplains. The NRC staff developed procedures for reviewing reactor sites in a manner consistent with the intent of the Executive Order and published these in the *Federal Register* on October 6, 1978. In addition, licensing procedures and Environmental Standard Review Plans were revised to address floodplain issues more explicitly. During 1979 the staff undertook the evaluation of several reactor sites with a view to improving floodplain management. Most sites for nuclear power plants require placement of some type of facilities in floodplains, such as auxiliary buildings, pipelines, and roadways associated with intake and discharge structures. Usually they are small in size, relative to the floodplain cross-sectional area, and do not interfere significantly with its flood-handling capability. If significant impacts are identified, it is generally required that structures be relocated or redesigned, or that other measures be taken to preserve the floodplain function.

The Executive Order on Floodplain Management also requires that floodplain considerations be addressed in NRC environmental impact statements and that guidance be afforded to applicants so that they can evaluate the effects of their proposals on floodplains prior to submitting their applications. The NRC's *Environmental Standard Review Plans*, published in May 1979 as NUREG-0555, satisfy these requirements for environmental concerns. Specific instructions are given for the NRC staff analysis of potential floodplain impacts and the discussion of these impacts in the Commission's Environmental Statements. A separate portion of each environmental standard review plan, describing data and information sources needed to conduct the environmental review, may be used by applicants as a guide for the treatment of floodplain concerns in their environmental reports.

**Evaluation of Breakwaters for Coastal Nuclear Plants.** Breakwaters made up of massive rocks or concrete armor units are often used to protect nuclear power plants from the effects of damaging waves. In 1979, the staff studied two special cases involving breakwaters and related to both safety and environmental issues. The breakwater at Pilgrim Nuclear Generating Station near Plymouth, Massachusetts had been damaged by the same storm which produced record snowfalls in the Boston area in February 1978. The breakwater was damaged again during the winter and early spring of 1979. The staff evaluated the probable causes of the damage, inspected the repair of the structure, and is currently evaluating the possible redesign of the breakwater to assure that the plant safety systems are not compromised.

Breakwater designs for floating nuclear plants were also studied with a view to assuring safety of the plant and to minimizing environmental impacts to oceans and estuaries. For estuarine siting, the staff identified a special problem which might occur in the event of an accidental release of radiation through an open breakwater. In estuaries, the mixing and dispersion of radioactive liquid from the accident would be slow and would produce long-term radioactivity levels with severe effects on the biota. The staff has recommended that estuary sites be such that the consequences of postulated accidents will not be worse than they would be at comparable land-based sites.

**Modification of Environmental Technical Specifications for Operating Reactors.** In accordance with provisions of the National Environmental Policy Act (NEPA), NRC has included in operating licenses a requirement for environmental monitoring programs. A number of fully licensed operating stations have compiled five years or more of data, and the staff has found, in reviewing the records of several such stations, that actual environmental impacts are generally within the range of expectations set out in the Environmental Impact Statement prepared prior to licensing. Review of other stations' results is expected to provide significant feedback to the license application review process.

**Adequacy of Analysis of Alternative Sites.** An important means of protecting the environment against undue adverse impacts is an appropriate screening of the area, as part of the site selection and evaluation process. Sufficient effort must be made by the utility applicant and the NRC review team to identify those alternative sites which are among the best which reasonably could have been found and to identify and properly assess the major potential environmental impacts. Such an effort is essential to a staff determination that no alternative site is obviously superior to the site proposed by the applicant. Without that reasonable assurance on the part of the staff, the site is to be rejected. In the Limited Early Site Review for the proposed Perryman nuclear plant site about 20 miles northeast of Baltimore, the NRC staff determined that the applicant's site selection and evaluation methods were inadequate and also concluded, on the basis of its own reconnaissance-level investigations, that there were obviously superior alternative sites. The staff conclusion that there are sites which are obviously superior to the Perryman site was based on considerations of population density, risks posed by the proximity of potentially hazardous activities and the overall project costs.

The review of the Perryman and other sites considered by the applicant failed to identify any environmental considerations that would suggest that Perryman offers significant advantages over alternative sites. The applicant did provide information to support his view that there were economic advantages in locating at Perryman, primarily resulting from lower transmission costs. In the staff's view, this cost avantage would be more than offset by the special design provisions which the staff expects will be required to protect against nearby external hazards. Among other reasons, the staff concluded that the applicant's alternative site analyses were inadequate because important siting parameters were omitted. In particular, no demographic or safety-related characteristics were used in the comparisons among sites. The process used to select, and the scoring scheme used to compare, the relative merits of the candidate sites have a number of deficiencies which render the results unreliable. Together, these factors cast doubt as to whether the submitted candidate sites represent the realistic siting resources available to the applicant. Following this review, the staff has made substantial progress during 1979 in formulating review policies to improve the adequacy of analysis of alternative sites.

Progress in the staff's reevaluation of considerations which are important to a determination of the adequacy of alternative site analyses was also stimulated by licensing actions involving Seabrook Units 1 and 2 and Pilgrim 2. During 1978, the NRC staff was ordered by the Atomic Safety and Licensing Appeal Board (ASLAB) to reexamine the question of alternative sites to the Seabrook Station located in Seabrook, N. H. The study was conducted under the assumption that the Environmental Protection Agency might order the construction of cooling towers for that station rather than permit once-through cooling; a valid alternative site analysis had already been done for the latter design and it had been determined that the Seabrook site was the environmentally superior site if cooling towers need not be used. The staff studied 22 candidate sites located throughout New England. Using a coarse screening process, the staff reduced this number of alternative sites to eight, and these were then compared in considerable detail to Seabrook. The analysis involved the contributions of eight different environmental disciplines. In addition, assistance was obtained from Argonne National Laboratory and Oak Ridge National Laboratory in computer modeling of cooling towers. Assistance from the Federal Energy Regulatory Commission was obtained for the analysis of transmission stability and reliability at each site, in comparison with Seabrook. After extensive comparative analyses, the staff concluded that five sites were environmentally equivalent to Seabrook and one had minor environmental advantages, while the remaining sites were environmentally disadvantageous. None of the alternative sites was found to be "obviously superior" to Seabrook under an assumption that Seabrook Station would be required to have cooling towers. The staff presented its testimony in a three-day hearing before the ASLAB in January 1979. Subsequently, the EPA reached a final decision that once-through cooling at the Seabrook Station was environmentally acceptable and the issue of alternative sites with cooling towers at Seabrook became moot.

As for Pilgrim 2, on December 1, 1977, the Atomic Safety and Licensing Board issued a partial initial decision regarding only the alternative site analysis section of the environmental review. The board's decision denied the Limited Work Authorization requested by the Boston Edison Company, citing as its reason the inadequacies of the NRC staff's review of alternative sites. This decision necessitated a reevaluation of alternative sites by the staff which, in turn, led to a request of the applicant to provide supplemental information on alternative siting. The staff conducted a detailed review of 13 sites located in Massachusetts, New Hampshire, and Connecticut. Each alternative site was evaluated against the Pilgrim site with respect to prospective impacts in the area of aquatic biology and water quality; terrestrial ecology and land use; demography; adjacency to industrial, transportation and military facilities; hydrology; socioeconomics; project economics; geology, seismology and geotechnical engineering; and meteorology. The staff analysis was presented in a Final Supplement to the Final Environmental Statement, issued in May 1979. As a result of this more detailed analysis, the staff concluded that none of the alternative sites was obviously superior to the proposed Pilgrim site and therefore recommended acceptance of the proposed site for the second unit.

Reasonably close proximity to an adequate water supply and the likely extent of water-related environmental impacts are generally regarded as among the more important factors in the identification of a superior site for large baseload electrical generating facilities. The cost and effectiveness of controls for mitigating such impacts is also examined. In both the Seabrook and Pilgrim 2 reviews, a number of water-related aspects were given detailed analyses for the proposed and alternative sites: local drainage considerations; erosion control; flood protection; pipeline location for coolant water; location of intake and discharge structures; water supply availability; and possible contamination of water supplies. In both cases, reviews of these considerations revealed that most were not critical to a demonstration of obvious superiority among the final group of candidate sites. This was so because site-screening criteria had already eliminated the most objectionable sites in terms of adverse water-related impacts, and because a number of the adverse water-related impacts for the final groups of candidate sites analyzed in detail could readily be mitigated at reasonable cost.

The major exception to this conclusion was that water availability remained a key siting issue. In particular, for inland sites on rivers or streams where flows may be seasonably very low, careful consideration will be required to assure that an adequate and dependable supply can be provided to meet the coolant water needs of the generating plant and of other users in the region.

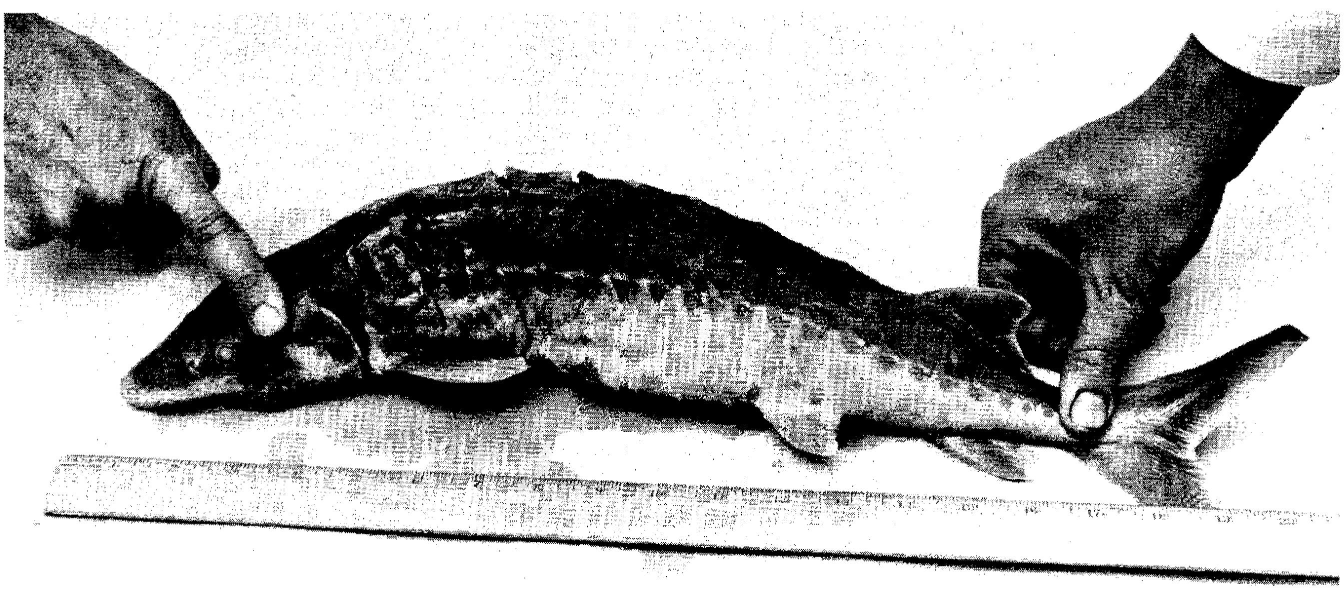
## INTERAGENCY COORDINATION

### Compliance with Regulations Of Other Agencies

**Cooperation with EPA and DOE on Occupational Radiation Dose Limits.** NRC staff members are participating in an Environmental Protection Agency (EPA) Interagency Committee on Federal Guidance for Occupational Exposures to Ionizing Radiation. The objective is to assist the EPA in developing guidance on occupational dose limits, responding in part to the recommendations in Publication No. 26 of the International Commission on Radiological Protection. In a related development, the NRC staff is working with EPA and the Department of Energy (DOE) in response to petitions by a private citizen and the Natural Resources Defense Council. Among other matters, these petitions request significant reductions in the occupational dose limits. EPA and NRC are considering joint hearings on these subjects to provide an opportunity for interested parties and members of the public to participate and to make their views known.

**Floodplain Management.** In accordance with the provisions of Executive Order 11988. "Floodplain Management," NRC has consulted with the Federal Interagency Panel on Floodplain management concerning procedures for floodplain management associated with power plant licensing applications (see discussion earlier in this chapter).

**Endangered Species Act (ESA).** This Act was amended in 1978 to require Federal agencies to consult



**NRC has asked for formal consultation with the National Marine Fisheries Service concerning the possible impact of nuclear power plants on populations of shortnose sturgeon in the Delaware River. The sturgeon shown here was recovered from the water intake system of the Salem Nuclear Generating Station, Unit 1, in New Jersey. Recent surveys indicated that the species, once threatened, may be making a comeback.**

with the Fish and Wildlife Service or the National Marine Fisheries Service (NMFS) to determine the possible existence of any endangered species in the locality of proposed projects prior to their initiation. If listed species are in the project location, biological analysis and further consultation are required. However, if none is found, no further interagency consultation is needed.

In 1978 two shortnose sturgeon had been found on the intake structure at Salem 1, and, in 1979, the NRC requested formal consultation with the NMFS to determine whether operation of the plant jeopardized the continued existence of the species in the Delaware River. Increasing numbers of reports of shortnose sturgeon in recent years indicate that the species may be making a comeback, not only in the Delaware, but in other Atlantic coast riverine systems as well. Formal consultation has also been requested on the shortnose sturgeon at Salem 2 and Hope Creek 1 and 2.

Shortnose sturgeon are also found at the Indian Point (New York) and Hatch (Georgia) plants. At Indian Point, the status of the shortnose sturgeon is under review by the NMFS. At Hatch, the NRC has initiated actions required by Section 7 of the ESA.

For other operating facilities, consultation under the ESA will be initiated when the staff becomes aware of the existence of threatened or endangered species in the plant vicinity.

**Fish and Wildlife Coordination Act.** During 1979, the staff reviewed proposed regulations of the Fish and Wildlife Service for implementation of the Fish and Wildlife Coordination Act. The regulations, if adopted, would require Federal agencies to mitigate significant impacts on fish and wildlife in all water-related projects and to compensate for significant wildlife losses or enhance wildlife values for all Federally licensed projects. Consultation between NRC and the Fish and Wildlife Service would be required by the regulations in all reactor licensing.

**Coastal Zone Management Act.** Under this legislation the siting of a nuclear plant in a region bordering the seacoast or the Great Lakes must be consistent with the Coastal Zone Management Plan developed by each State, commonwealth or territory in which the site is located. The State or commonwealth (e.g., Puerto Rico) makes this determination. As of the end of fiscal year 1979, over half of the 30 affected States have completed or made substantial progress in developing management plans for their coastal zones.

**Clean Water Act.** The environmental review of NRC licensing actions involves extensive coordination with other Federal and State agencies concerning provisions under the Clean Water Act. A principal area of coordination is with the EPA or delegated State agencies under the National Pollutant Discharge Elimination System Permit Program. NRC's relationship to EPA was modified in 1978 by the Yellow Creek decision of an NRC licensing Board which required that NRC coordinate its nuclear plant water quality monitoring needs through the EPA or "permitting" State, rather than impose water monitoring requirements of its own directly on applicants. EPA is revising its Effluent Limitation Guidelines in conjunction with quality standards, which will regulate the concentrations of non-radiological contaminants in nuclear power plant effluents.

**Toxic Substances Control Act.** Guidelines developed under the Toxic Substances Control Act may result in the imposition of limitations by EPA on certain substances found in power plant cooling water discharges. On the list of substances currently being evaluated by EPA are several which are produced by chlorination of natural surface waters used for power plant cooling.

## Interagency Topical Studies

**Interagency Committee on Ocean Pollution Research, Development, and Monitoring.** This committee was created by Public Law 95-273 to prepare for fiscal years 1979-1983 a Federal Plan for Ocean Pollution Research, Development and Monitoring. The final draft of the plan was completed in August 1979 and submitted to the Executive Office of the President by NOAA, the study coordinator. The study identified current Federal ocean pollution research activities and established a prioritized program for research and development. The study also initiated planning for coordination of future ocean pollution research and for dissemination of the information resulting from the research and monitoring programs. The NRC funded approximately $1.2 million of ocean pollution research in fiscal year 1979 to support its licensing actions. In addition, considerable monitoring of the ocean environment has been performed by applicants for nuclear power plant construction permits and licensees of operating nuclear plants. The next version of the Federal Plan on Ocean Pollution Research will include work being done by private and State organizations as well.

**Interagency Committee on Environmental Monitoring.** At the direction of the President, the Council on Environmental Quality established the Interagency Task Force on Environmental Data and Monitoring. Its function is to review environmental monitoring and data programs and to recommend improvements that would make these programs more effective. Specific activities of the Task Force in which NRC participates consist of assessing the manner in which the various Federal agencies accumulate and disseminate water and air data, developing a catalog of agency monitoring, and otherwise coordinating the use of this information among other Federal agencies. Recommendations were made in a report to CEQ that is to be forwarded to the President. The primary purpose of these recommendations is to minimize overlapping responsibilities among the Federal agencies and to provide for communication and coordination of data-collection efforts related to water and atmospheric properties and constituents.

**TVA/EPRI Workshop on Waste Heat Utilization.** During the report period, the NRC staff participated in this workshop sponsored by the Tennessee Valley Authority and the Electric Power Research Institute by chairing the session on nuclear and public health aspects of waste heat utilization, and by presenting an analysis of the specific factors that would be of concern to NRC in the use of effluents from nuclear power plants. The results of this workshop will be applicable during the Watts Bar licensing process because of the applicant's plan to provide for a large waste heat utilization facility associated with the power plant.

**Water Resources Council.** During the report period, the staff participated in activities of the Hydrology Committee of the Water Resources Council. The activities of the committee centered on coordinating the various participating agencies working to assess the state-of-the-art in various subject areas and to recommend standardization. The staff participated in several activities in this area, including an assessment of the state-of-the-art in hurricanes surge modeling (i.e., induced flooding), reassessment of groundwater study requirements, assessment of the state-of-the-art in low flow considerations, and in identifying standardization of flood frequency determinations for ungaged water sheds.

**Interagency Advisory Committee on Water Data.** The staff participated on the Interagency Advisory Committee on Water Data chaired by the U.S. Geological Survey. During fiscal year 1979, the staff identified NRC's water data uses and needs and cooperated with other Federal agencies under the provisions of OMB Circular 67 on water data.

**Interagency Committee on Dam Safety.** In April 1977, President Carter requested a review of procedures and criteria issued by Federal agencies involved in the design, construction, operation, and regulation of dams, and the preparation of guidelines for management procedures to ensure dam safety. The guidelines, published in June 1979, are based on an intensive review of agency practices conducted by three department groups: the departments and agencies themselves; an ad hoc interagency committee of the Federal Coordinating Council for Science, Engineering and Technology (FCCSET); and an Independent Review Panel of recognized experts from the academic and private sectors. NRC was requested to join in the FCCSET review and guideline development activity. The guidelines—which address organization, management, and the management of technical activities, including site investigation and design, construction, and operation and maintenance—await Presidential direction for implementation.

**Interagency Committee on Seismic Safety in Construction.** NRC staff participated principally on a subcommittee of the Interagency Committee dealing with the problems of tsunami (seismic sea wave) protection. This subcommittee's purpose is to identify methods and criteria for assessing tsunami and other seismically-induced flood wave threats for the protection of Federal facilities. State-of-the-art in assessing tsunami threats and design criteria were drafted in fiscal year 1979.

**The National Weather Service Study on Emergency Response.** Subsequent to the accident at Three Mile Island, the staff coordinated with its consultants, the National Weather Service of the National Oceanic and Atmospheric Administration (NOAA), and DOE in an attempt to determine the actions taken by NOAA and DOE during the TMI accident. The NRC staff response was independent of both NOAA and DOE, and it was only subsequent to the accident that the staff learned of the extent of both NOAA's and DOE's participation. These facts exposed the need to coordinate future agency responses to any accidents involving core melt with significant offsite radiation releases (Class 9). Moreover, the staff has requested NOAA's participation in a study of portable meteorological instrumentation and of an assessment of meteorological models for use during any such future accidents. The purpose of the instrumentation and models would be to supplement on-site and regional meteorological data and models to provide prompt and accurate estimates of the location and concentrations of radioactive releases.

**Interagency Committee on Electric Field Effects from High Voltage Transmission Lines.** NRC participates as a permanent member on the Interagency Advisory Committee on Electric Field Effects and aids in the identification, review, and coordination of Federal research programs investigating potential problems associated with the operation of electrical transmission systems. The Committee's primary emphasis has been on the study of potential short and long term health effects from the operation of high voltage transmission systems. These effects are routinely considered by NRC staff in its environmental reviews.

**National Ecological Assessment Workshops.** NRC participated in a workshop sponsored by the U.S. Department of Energy for the purpose of identifying the ecological concerns, methods and problems of performing non-site-specific ecological assessments of energy development. The workshop which was attended by prominent ecologists and government planners deliberated for three days on the possible ecological and human environmental impacts of energy development alternatives. Concerns about additions of carbon dioxide, sulfur dioxide, and nitrogen oxides to the atmosphere and widespread acid rainfall were among the serious problems identified and discussed. Although nuclear plants produce none of these atmospheric pollutants, the impacts of generating electricity from fossil or other organic fuels include such pollutants and fall within the scope of NRC reviews of the comparative impacts of alternative sources of energy for baseload generation of electricity.

**Third Conference on Water Chlorination—Environmental Impact and Health Effects.** The NRC has joined the Department of Energy, the Environmental Protection Agency, the National Cancer Institute, The Tennessee Valley Authority, and Oak Ridge National Laboratory in sponsoring the Third Conference on Water Chlorination. The intentional use of toxic substances to control biological growth within power plant condenser cooling systems is carefully examined during NRC's review of a power plant license application and the preparation of an environmental impact statement. The series of conferences on chlorination provide current information necessary for the better understanding of environmental effects and of alternative "bio-fouling" control practices.

## COOPERATION WITH STATES

### State Participation in NRC's Environmental Impact Statements

During the past year, NRC and the State of New York took steps which led to the signing of an agreement that allows State participation in the preparation of NRC's environmental impact statements. By terms of the agreement the technical staff of the New York State Public Service Commission will write specific sections of the Draft and Final Environmental Statements. The sections involved are mainly concerned with environmental description and impact, alternatives to the proposed action, and the need for the facility. In the past, these sections have been prepared through environmental assistance agreements with the Department of Energy's National Laboratories. In addition to the staff of the Public Service Commission, the technical staff of the New York State Department of Environmental Conservation has agreed to provide technical support to the Public Service Commission in the form of review and comments on the write-ups for the environmental statements. State participation of this kind in the preparation of NRC's Environmental Impact Statements will require close coordination with the relevant review functions of the NRC regulatory staff.

## IMPROVING THE LICENSING PROCESS

### Generic Rulemaking to Improve Licensing

In June 1977, an NRC study group seeking to identify ways to improve the effectiveness of NRC Nuclear power plant licensing procedures recommended, among other things, that rulemaking should be considered for the generic resolution of certain major issues that are presently litigated in individual licensing proceedings. The study groups recommendations are presented in a report, "Nuclear Power Plant Licensing: Opportunities for Improvement" (NUREG-0292). In response to a Commission directive, the staff prepared an interim statement of general policy and plans for rulemaking, which the Commission approved for publication in the *Federal Register* (December 14, 1978). This interim policy statement fully supports Executive Order 12044 of March 23, 1978, requesting improvement of existing Federal government regulations, so as to make them as simple and clear as possible and to avoid imposing unnecessary burdens on the economy, on individuals, on public and private organizations, or on State and local governments. The interim policy statement and supporting discussions are presented in an NRC report, "Preliminary Statement on General Policy for Rulemaking to Improve Nuclear Power Plant Licensing" (NUREG-0499).

Ten candidate issues were identified by the staff for generic rulemaking: (1) future availability and price of uranium, (2) alternative energy sources to the nuclear option, (3) need for adding baseload generating capacity, (4) methodological and information requirements in the analysis of alternative sites (5) criteria for the assessment of nuclear plant impacts and mitigative measures; (6) generic procedural criteria to define more concretely NRC responsibility in assessments and decisions regarding certain water-related impacts in relation to the statutory authorities of EPA and permitting States, (7) NEPA decision criteria for OL reviews, (8) occupational radiation exposure control, (9) generic radiological impact for normal light water reactor radionuclide releases, and (10) threshold limits for generic disposition of cooling tower effects. Criteria developed by the Steering Committee on Reactor Licensing Rulemaking to aid in identifying suitable candidate issues for rulemaking include the following: the issues must be generic; there must be a likelihood that a useful, definitive rule can be formulated and there must be a likelihood that a stable rule can be formulated. Value-impact criteria for appraising the desirability of, and the priorities associated with, specific proposals for generic rules include:

- Achievement of more effective public input and improved public understanding of NRC's analytical procedures and decision criteria in treating potential environmental and safety issues in the licensing process for nuclear power plants.
- Improvement of the stability and predictability of the licensing process, including the provision of orderly and clear procedures for State-Federal cooperation in treating generic licensing issues.
- Accomplishment of an overall savings of manpower and financial resources of the NRC, the public, the utility industry, and other local, State, and Federal agencies involved in the nuclear licensing process.
- The short-term increase in dollar costs of the various participants in the rulemaking action, including contractual support.
- The additional impacts (i.e., opportunity costs) of diverting manpower and other resources to the rulemaking process and away from other productive uses for a temporary period.

Public comment was invited on the merits of the candidate issues for generic rulemaking and related decisions criteria, and additional suggestions for candidate subjects for generic rulemaking were solicited. Fifty-eight comments were received but, except for Issue No. 4 on alternative siting methodology, further NRC activity on generic rulemaking was temporarily suspended because of the diversion of staff effort to studies related to the Three Mile Island nuclear accident and related remedial measures.

## Evaluation of Alternative Sites

As noted above, one of the ten issues identified for possible generic rulemaking was that of alternative site methodology and information requirements. In order to refine and clarify this issue, the staff, on December 14, 1978, issued for comment a report entitled, "General Considerations and Issues of Significance on the Evaluation of Alternative Sites for Nuclear Generating Stations under NEPA" (NUREG-0499, Supplement 1). In addition to receiving public comments on the report, the staff conducted a three-day public workshop in March 1979 to actively seek comments and ideas on rulemaking for alternative sites. Representatives from industry, State and Federal government, public interest groups and others participated. Utilizing public comments and the results of the workshop, the staff drafted proposed amendments to 10 CFR Part 51 which pertain to the evaluation of alternative sites. These amendments were submitted to the Commission in July 1979 for their consideration. The results of the staff deliberations on generic rulemaking and public comments received in response to the Federal Register notification of December 14, 1978 as well as the March workshop have already yielded benefits in staff review practices and the revision of environmental standard review plans to deal more effectively with alternative siting issues.

## Siting Policy Task Force

The essential elements of nuclear power plant siting policy are derived from the Atomic Energy Act of 1954 and are contained in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and in 10 CFR Part 100, "Reactor Site Criteria." These regulations were promulgated by the Atomic Energy Commission in 1962 and have remained essentially unchanged since that time. The authors of Part 100 recognized that experience with siting nuclear power plants was at that time limited and, in anticipation of subsequent changes as experience was gained, included in Paragraph 100.1 the statement that:

"(b) Insufficient experience has been accumulated to permit the writing of detailed standards that would provide a quantitative correlation of all factors significant to the question of acceptability of reactor sites. This part is intended as an interim guide to identify a number of factors considered by the Commission in the evaluation of reactor sites and the general criteria used at this time as guides in approving or disapproving proposed sites."

In the time since Part 100 was promulgated, the NRC has issued additional siting-related pronouncements in the form of siting decisions on specific cases, General Design Criteria, Regulatory Guides, Standard Review Plans, Licensing and Appeals Board decisions, and advice from the Advisory Committee on Reactor Safeguards (ACRS). All of these sources have contributed to the formulation of the Commission's current siting policy and practice. During this evolutionary period, the nuclear industry experienced a rapid expansion, the use of nuclear power plants became commonplace, and the size of such plants increased significantly. As a consequence of this expansion, some in staff practice and in the implementation of the siting regulations have evolved. In addition, the Commission's implementation of the National Environmental Policy Act of 1969 (NEPA) has added new dimensions to siting policy.

In August 1978, the Nuclear Regulatory Commission directed the staff to develop a general policy statement on nuclear power reactor siting and a Task Force was formed for the purpose. The *Report of the Siting Policy Task Force* (NUREG-0625) was issued in

August 1979. The report provides a review of current NRC policy and practice and recommends a number of changes to achieve the following goals:

- To strengthen siting as a factor in defense-in-depth by establishing requirements for site approval that are independent of plant design consideration. The present policy of permitting plant design features to compensate for unfavorable site characteristics has resulted in improved designs but has tended to deemphasize site isolation.
- To take into consideration in siting the risk associated with accidents beyond the design basis (Class 9) by establishing population density and distribution criteria. Plant design improvements have reduced the probability and consequences of design basis accidents, but there remains the residual risk from accidents not considered in the design basis. Although this risk cannot be completely reduced to zero, it can be reduced by selective siting.
- To require that sites selected will minimize the risk from energy generation. The selected sites should be among the best available in the region where new generating capacity is needed. Siting requirements should be stringent enough to limit the residual risk of reactor operation but not so stringent as to eliminate the nuclear option from large regions of the country. This is because energy generation from any source has its associated risk, with risks from some energy sources being greater than that of the nuclear option.

Nine changes were recommended by the Siting Policy Task Force for consideration by the Commission (NUREG-0625, pp. 46-63):

(1) Revise Part 100 to change the way protection is provided from accidents by incorporating fixed exclusion and protective action distances and population density and distribution criteria.
   (i) Specify a fixed minimum exclusion distance based on limiting the individual risk from design basis accidents. Furthermore, the regulations should clarify the required control by the utility over activities taking place in land and water portions of the exclusion area.
   (ii) Specify a fixed minimum emergency planning distance of 10 miles. The physical characteristics of the emergency planning zone should provide reasonable assurance that evacuation of persons, including transients, would be feasible if needed to mitigate the consequences of accidents.
   (iii) Incorporate specific population density and distribution limits outside the exclusion area that are dependent on the average population of the region.
   (iv) Remove the requirement to calculate radiation doses as a means of establishing minimum exclusion distances and low population zones.

(2) Revise Part 100 to require consideration of the potential hazards posed by man-made activities and natural characteristics of sites by establishing minimum standoff distances for:
   (i) Major or commercial airports
   (ii) LNG terminals
   (iii) Large propane pipelines
   (iv) Large natural gas pipelines
   (v) Large quantities of explosive or toxic materials
   (vi) Major dams
   (vii) Capable faults.

(3) Revise Part 100 by requiring a reasonable assurance that interdictive measures are possible to limit groundwater contamination resulting from Class 9 accidents within the immediate vicinity of the site.

(4) Revise Appendix A to 10 CFR 100 to better reflect the evolving technology in assessing seismic hazards.

(5) Revise Part 100 to include consideration of post-licensing changes in offsite activities:
   (i) The NRC staff shall inform local authorities (planning commission, county commissions, etc.) that control activities within the emergency planning zone (EPZ) are the basis for determining the acceptability of a site.
   (ii) The NRC staff shall notify those Federal agencies, as in Item (i) above, that may reasonably initiate a future Federal action that may influence the nuclear power plant.
   (iii) The NRC staff shall require applicants to monitor and report potentially adverse off-site developments.
   (iv) If, in spite of the actions described in Items (i) through (iii), there are off-site developments that have the potential for significantly increasing the risk to the public, the NRC staff will consider restrictions on a case-by-case basis.

(6) Continue the current approach toward site selection from a safety viewpoint, but select sites so that there are no unfavorable characteristics requiring unique or unusual design to compensate for site inadequecies.

(7) Revise Part 100 to specify that site approval be established at the earliest decision point in the

review and to provide criteria that would have to be satisfied for this approach to be subsequently reopened in the licensing process.

(8) Revise Part 51 to provide that a final decision disapproving a proposed site by a State agency whose approval is fundamental to the project would be a sufficient basis for NRC to terminate review. Such termination of a review would then be reviewed by the Commission.

(9) Develop common bases for comparing the risks for all external events.

## Early Site Reviews

Utilities are continuing to use the early site review process adopted by the NRC in 1977 to improve reactor licensing. Two additional applications have been tendered under these procedures—the Carroll County Station (Ill.) and the Fulton Station (Pa.). The Fulton application is an amendment to the utility's previous application for construction permits. In addition, the staff has completed site environmental and safety documents for the North Coast (Puerto Rico) application and submitted these documents to the Atomic Safety and Licensing Board and the Advisory Committee on Reactor Safeguards. The review is being delayed awaiting notification from the utility as to whether it wishes NRC to continue the review. Three other applications are in various stages under the early site procedures—Blue Hills, Texas; Douglas Point, Maryland; and Fort Calhoun, Nebraska.

## Environmental Standard Review Plans

Environmental standard review plans (ESRPs) constitute a series of instructions developed for the NRC staff's environmental review of applications for nuclear power plant construction permits. Their main purpose is to improve the quality of staff reviews of environmental issues. The plans also provide guidance to applicants regarding the information and criteria considered essential to the staff's environmental review process. The ESRPs, 93 in number, were issued throughout 1977 for draft review and public comment as NUREG-0158, Parts I, II and III. In May 1979 the revised *Environmental Standard Review Plans* were issued as NUREG—0555. As internal procedures and positions or Commission policies change, the ESRPs will be modified to keep them current with these changes.

## Environmental Impact Statements

During 1979, the staff had undertaken to revise the format used for the Commission's environmental impact statements for the construction and operation of nuclear power plants. This effort was performed within the framework of the requirement to revise the Commission's regulations covering licensing and regulatory policy and procedures. Such revision was undertaken in compliance with new regulations published by the Council on Environmental Quality (CEQ) in the *National Enviromental Policy Act: Implementation of Procedural Provisions (Federal Register*, Vol. 43, No. 230, November 29, 1978).

## Social and Economic Issues

**Forecasting Socioeconomic Impacts.** Hearing issues on socioeconomic impacts have led to a heightened appreciation of the importance of empirical studies of these impacts at regionally and environmentally diverse locations of nuclear power plants as an aid to improving the analytical basis for forecasting such impacts in new licensing actions. The first of these retrospective studies analyzed socioeconomic impacts on the communities surrounding the Pilgrim I Nuclear Station (Massachusetts) and the Millstone I and II Nuclear Station (Connecticut). Performed for the NRC under contract by the Oak Ridge National Laboratory (ORNL), it was issued in September 1977 under the title, "A Post Licensing Study of Community Effects At Two Operating Nuclear Power Plants" (ORNL/NUREG/TM-22). A continuation of this type of post-licensing study of socioeconomic impacts has led to two additional NRC-funded studies, the first focusing on the nuclear plant sites at Brunswick 1 and 2 (North Carolina) and Hatch 1 and 2 (Georgia), and the second at the Trojan plant (Oregon):

(1) "Socioeconomic Impacts of Nuclear Power Plants: A Paired Comparison of Operating Facilities" (NUREG/CR-0916), ORNL, July 1979.

(2) "Social and Economic Impacts of the Trojan Nuclear Power Plant: A Confirmatory Technology Assessment" (NUREG/CR-0973), University of Washington, Program in the Social Management of Technology, October 1979.

A socioeconomic study of specialized scope was performed for the NRC by the Pennsylvania State University on the " Effects of Nuclear Power Plants on Community Growth and Residential Property Values" (NUREG/CR-0454). Issued in April 1979, this study concluded that the four northeastern plants (Pilgrim, Millstone, Oyster Creek, and R.E. Ginna) demonstrated no significant influence on the price of housing and that growth rates for the years following plant construction were higher than the period prior to construction.

Contract research is continuing in the development of analytical tools to evaluate visual esthetic impacts of alternative closed-cycle cooling systems and to improve the forecasting of the number of incoming construction workers, their family characteristics and probable residential location, in order to assess the likely degree of stress on community services and housing.

A new contractual effort was initiated with Mountain West Research, Inc., in late 1978 to study the social and economic consequences of siting, constructing, and operating nuclear power stations in the United States. Fourteen stations at 13 sites were selected for study: Surry 1 and 2 (Va.); Three Mile Island 1 and 2 (Pa.); Peach Bottom 2 and 3 (Pa.); Zion 1 and 2 (Ill.); Cook 1 and 2 (Mich.); Oconee 1-3 (S.C.); Rancho Seco (Cal.); Fitzpatrick/Nine Mile Point (N.Y.); Calvert Cliffs 1 and 2 (Md.); Crystal River 3 (Fla.); St. Lucie 1 and 2 (Fla.); Arkansas 1 and 2 (Ark.); and Diablo Canyon 1 and 2 (Cal.). Selection criteria prescribed that plant sizes be in excess of 800 MWe, with an expected operating period of at least 12 months; regional diversity and an appropriate spectrum of variations in the rates of population growth in the host county were provided for in the selection, as were plant locations at varying distances from population centers of 50,000 or more. Specific socioeconomic effects at the local and regional level being studied include: employment, retail sales, public services, housing, public finance (especially tax benefits), community participation and conflict, and community perception of social well-being. Work on the methodology phase of the study was completed in June 1979, and detailed case study work was undertaken at four sites in July 1979, with completion of the study of all 13 sites expected by December 1980.

As would be expected, the accident at Three Mile Island (TMI) on March 28, 1979, substantially affected the study plan underway at that time. Not only was TMI one of the case study sites, but there was conjecture that TMI might affect the way in which other stations would be evaluated by local residents. The original design had to be modified, therefore, to include four analytic time periods: siting, construction, operation, and the post-accident period. For TMI there was yet a fifth period, the two-week period following the accident, that must be studied. In order to be able to document both the accident and the post-accident social and economic effects at TMI, it became clear that primary data would have to be collected from area residents. This data requirement led to the Three Mile Island Telephone Survey, which included 1500 households within 55 miles of the plant site. The scope of the survey included: evacuation behavior; the decision-making process regarding evacuation; the evaluation of the quality of information resources; general attitudes about nuclear power and the community's economic and social outlook following the accident; the direct and indirect social and economic costs of evacuation; and demographic descriptors. In October 1979, a preliminary report, *Three Mile Island Telephone Survey* (NUREG/CR-1093), was published which presented procedures and findings of the survey. The magnitude of community anxieties raised by the TMI accident is evident in the report's estimate that about 144,000 persons temporarily moved out of the zone within 15 miles of the plant site, travelling an average distance of 100 miles to a total of 21 States, mainly to stay with friends and relatives.

A separate contract study is in the planning stages within NRC that would seek to determine the effect of the TMI accident on property values as a function of distance from the plant and of time through 1982, not only for the TMI site but also for the four sites studied in NUREG/CR-0454.

**Independent Analysis of Need For Facility.** Progress has continued in 1979 in improving the analytical tools for independent assessment by the staff of need-for-facility. In October 1978, the NRC published the contract study by the Oak Ridge National Laboratory (ORNL) on "Regional Econometric Model for Forecasting Electricity Demand by Sector and by State" (NUREG/CR-0250). The sectors for which total electrical energy demand are forecasted by the model include residential, commercial, industrial, and other. The model provides flexibility for deriving separate forecasts for comparative purposes by making different scenario assumptions regarding such basic causal factors as population growth, per capita income and value added in manufacturing. Related contract studies underway by the ORNL are expected to be published in 1980 which will extend, update, and improve the model for staff reviews in dealing with a variety of controversial hearing issues associated with the need-for-power issue. The titles of these studies reflect the added dimensions of model improvement:

- Comparison and Projection of Electricity Cost by State.
- Econometric Model for the Disaggregation of State-Level Electricity Demand Forecasts to the Service Area.
- Peak Electricity Demand Predictions Using Hourly Variations.

An additional improvement not reflected in these titles is the disaggregation of the industrial demand through the use of 2-digit Standard Industrial Classification Codes (SICs).

**Economic Comparison of Coal and Nuclear Energy for Generating Electricity.** Controversial hearing issues in the licensing of nuclear power plants frequently involve the question as to whether coal or nuclear energy is the more economical method of generating electricity at a particular location. To improve the basis for the staff's independent analysis of the comparative economic evaluations provided by the applicant for a construction permit, the NRC issued a staff report in December 1978 on the subject, "Coal and Nuclear: A Comparison of the Cost of Generating Baseload Electricity by Region" (NUREG-0480).

The study compares the economics of a 2400 MWe nuclear and coal electric generating station in 10 different regions of the United States. Delivered coal costs are the primary cause of regional generating cost variations; therefore, the regions were based on the Department of Energy's (DOE) regions for delivered coal costs. The capital cost for coal-fired generating units includes the cost of sulfur removal. The economics are based on a station beginning operation about 1990 for an investor-owned utility.

The study is based on data inputs from numerous sources, and it avoids the pitfalls of cost analyses based on national averages by highlighting regional differences which—in addition to the transportation costs of coal affecting the delivered cost of coal to different regions—include variations in coal characteristics, and construction costs for labor and materials, as well as labor productivity.

A computer program called CONCEPT, developed by the Oak Ridge National Laboratory, was used to generate capital cost estimates by region.

The study results indicate that nuclear generating units are either more economical or about equal to coal generating costs for each of the 10 regions, even using an assumption of no recycle of plutonium or uranium in the nuclear option. However, the study results are regionally generic and applied to individual licensing cases by an evaluation of the site-specific and other causal factors, if any, which would cause the comparative cost estimates to depart significantly from the generic assumptions and data inputs. For example, air quality standards at one site may affect the cost of coal-fired generating units differently than nuclear; also seismic conditions at another site may affect nuclear generating costs differently than coal. Moreover, given that forecasting is an imprecise art, the methodology provides flexibility in facilitating the computation of other forecasts based on assumptions of cost-related inputs different from those used in the study. Among the latter are such assumptions as a plant capacity factor for both coal and nuclear plants of 65 percent, a 30-year operating life, a 10 percent discount rate, a 5 percent general inflation rate during the operating period, and fixed charge rates varying from 11 to 17 percent.

## Information Base for Licensing Decisions

**Demographic Data for Nuclear Sites.** A draft report was issued by the NRC in December 1977 on the subject, "Demographic Statistics Pertaining to Nuclear Power Reactor Sites" (NUREG-0348). The staff has prepared a revised and expanded version of this document, increasing from 104 to 145 the number of sites treated, and including additional tables of population and population center information. This improved data for siting policy and practice was published in October 1979 (NUREG-0348).

**Data Base for Aircraft Risk Assessment.** Recent licensing hearings have experienced an increasing scrutiny of reliability factors, such as crash densities used to estimate aircraft-impact likelihoods for nuclear stations which have a commercial airport within 5 miles of the site. In response to the need for reliable historical data for aircraft accidents, the NRC has issued a report in June 1979 on "Aircraft Impact Risk Assessmeent Data Base for Assessment of Fixed Wing Air Carrier Impact in the Vicinity of Airports" (NUREG-0533).

**Hudson River Fish Impact Study.** In support of testimony at EPA's hearing on Hudson River power plant cooling systems, NRC funded the development of an investigation resulting in a report on "Fish Protection at Steam-Electric Power Plants: Alternative Screening Devices," pubished by Oak Ridge National Laboratory as Report No. ORNL/TM-6472, July 1979. This report, discussing applications and limitations of alternative intake designs, has general applicability to power plant siting and design where potential loss of early life stages of fish is of concern.

**Predicting Fishery Impacts.** Because of the proliferation of complex mathematical models describing fish population dynamics, NRC has contracted with the University of Wahington College of Fisheries to undertake a comparison of existing models to provide the staff with guidance for use of models in evaluating the impact of power plants on fisheries. The first project report, "Comparison of Simulation Models Used in Assessing the Affects of Power-Plant-Induced Mortality on Fish Populations" (NUREG/CR-0474), was published in October 1978. This study concludes that existing models are limited in usefulness for making quantitative predictions of population impacts. Specific deficiencies and recommendations for future efforts of modelers are presented. A continuation of this study will result in a comparison of the cost-effectiveness of various modeling approaches. A related study at the Battelle Pacific Northwest Laboratory has resulted in the publication in March 1979 of a report, "The Application of Fisheries Management Techniques to Assessing Impacts: Task I Report" (NUREG/CR-0572). This study was undertaken to learn whether existing fisheries management techniques, which usually require significantly less data than do mathematical models, can be used to document the impact of a power plant on a fishery resource.

**Generic Study on Asbestos Fibers.** Because of national concern over the potential carcinogenicity of airborne asbestos fibers, NRC sponsored a study by the Argonne National Laboratory to determine more precisely the basis of concern over the use of asbestos fill material in power plant cooling towers. The final report, "Asbestos in Cooling-Tower Waters" (NUREG/CR-0770), was published in March 1979. The study concluded that the concentration of fibers found in a number of power plant effluents would not constitute a health hazard.

**Other Information on Ecological Impacts.** Other NRC studies under way which will improve the information base for assessing ecological impacts are:

- The relationship between shipworm abundance and distribution at Barnegat Bay in New Jersey and changes in temperature and salinity caused by the operation of the Oyster Creek Nuclear Station.
- The ecological significance of fish impingement on the intake screens of the Arkansas Unit One Nuclear Station.
- The toxicity and environmental importance of chlorine and heavy metal discharges in the effluents of nuclear power plants, the frequency and significance of pathogenic amoebae in cooling systems, and quantification of mortality by entrained organisms in once-through condenser cooling systems.
- The application of aerial remote sensing techniques to routine terrestrial monitoring, and the use of reconnaissance level information for evaluating potential impacts of alternative sites.

**Meteorological Measurement and Prediction.** During 1979, a survey study sponsored by the NRC was completed by the Brookhaven National Laboratory on the state-of-the-art in assessing atmospheric diffusion conditions in coastal regions. The study identified meteorological measurement programs, test conditions, and needs for additional research to avoid underestimating concentrations in the event of accidents at reactor sites in the coastal zone.

The staff also sponsored a state-of-the-art survey of the transport and diffusion of hazardous materials at the Los Alamos National Laboratory. The purpose of the study was to identify modeling requirements of either buoyant or sub-buoyant plumes resulting from releases, including explosions, of hazardous materials. The summary also indicated research needs.

The staff sponsored technical assistance by the Naval Surface Weapons Center on the assessment of the state-of-the-art regarding the potential for missiles to become airborne in tornadoes. The principal purpose of this study was to determine whether the types of missiles the staff routinely postulates for purposes of assessing reactor design are adequate. The study concluded that several missiles specified by the staff would be unlikely to fly in the event of a severe tornado. As a result of these studies, the staff is reconsidering its present criteria.

**Improved Interfacing with Utilities Regarding Meteorological Data.** The staff has standardized the format for reporting meteorological data collected at reactor sites for reactor licensing. In the past, summarized data were required for consideration in reactor licensing, but the format for such information was not specified. Improved data acquisition recording systems in the private sector, and the need for standardization in the NRC's consideration of meteorological data, prompted the specification of a standard format for reporting on-site meteorological data on magnetic tape. Subsequent to the specification of the standard format, receipt of magnetic tape from individual reactor sites has expedited evaluations by the staff and has reduced errors in data handling.

**Standardization of Meteorological Assessments for Accidental Releases and Routine Releases.** During 1979, the staff developed and promulgated computer codes for assessing meteorological conditions following an accident and for routine releases. The publication of these computer codes and reference to them in NRC standards is expected to facilitate both the industry and staff's efforts in future licensing situations.

**Improved Access to Agencies' Water Data.** During 1979, the staff established and implemented direct computer access to EPA's STORET and the USGS's WATSTORE computer information and retrieval systems. Both of these systems allow rapid access to significant water-related data collected at many locations around the country. The access to these systems by NRC has allowed more speedy and accurate evaluations of both safety-related and environmental subjects.

**Installation of Computer Information Retrieval System for Environmental Data.** During the past year, a computerized document control system (known as TERA) was installed in NRC. This system will allow the professional and administative staff to search for and retrieve NRC documents, including environmental data from the files more efficiently than before (see Chapter 14.)

## Quality Assurance

The application of disciplined engineering practices and thorough management and programmatic controls to the design, fabrication, construction, and operation of nuclear power plants is essential to the protection of public health and safety and of the environment. Quality Assurance (QA) provides this necessary discipline and control. Through a QA program that meets NRC requirements, all organizations performing work that is important to safety are required to conduct work in a preplanned and documented manner; to independently verify the adequacy of completed work; to provide records that will confirm the acceptability of work and manufactured items; and to assure that all individuals are properly trained and qualified to carry out their responsibilities.

Each NRC licensee is held responsible for assuring that its nuclear power plants are built and operated safety and in conformance with the NRC regulations. In addition, the NRC has several specific QA responsibilities. First, it has a responsibility for developing the criteria and guides for judging the acceptability of nuclear power plant QA programs. Second, it has a responsibility for reviewing the QA programs of each licensee and its principal contractors to assure that sufficient management and program control exist. Finally, NRC inspects selected activities to determine that the QA programs are being implemented effectively.

Where QA programs are found deficient, the NRC requires appropriate upgrading. In those cases where the QA program is not being properly implemented, the NRC uses enforcement authority as necessary to achieve proper implementation. If a generic QA problem develops, improvements in QA programs are made industry wide.

Through the NRC topical report program, the industry has widely adopted standardized QA programs which can be used on new projects without a new review. As of the end of the fiscal year, a total of 38 topical reports on quality assurance from manufacturers of nuclear steam supply systems, architect-engineering firms, constructors, and utilities have been found acceptable by the NRC and other reports are under review.

NRC is engaged in activities, also under the topical report program, that are intended to minimize or eliminate the need for redundant audits of suppliers without reducing the confidence that work is proceeding satisfactorily in accordance with regulations. NRC is in the process of reviewing a topical report describing the ASME certification and inspection program which, if found acceptable, could be endorsed as a "third party" audit program. Successful achievement of this objective should further reduce the need for pre-award audits and for yearly programmatic audits by purchasers.

In an effort to improve QA, the acceptance criteria contained in Section 17, "Quality Assurance," of the Standard Review Plan, NUREG-75/087, which serves as the basis for determining the acceptance of QA programs, were updated to provide additional QA controls to give further confidence in the acceptability of QA programs.

Since TMI and other incidents, the overall structure for determining and acceptable QA program, including the capabilities and qualifications of individuals performing quality-affecting activities, are undergoing a review and evaluation to identify areas where further improvements can be made.

## Systematic Evaluation of Operating Reactors

The Systematic Evaluation Program (SEP) staff is responsible for the review of 11 older licensed operating power reactors, applying current licensing criteria, and for documenting the results—including the need for any necessary plant changes. The major objectives of the SEP are set forth in the 1978 NRC Annual Report, pp. 59 and 62.

Phase I of the SEP, the development of a list of topics to be used in performing the systematic evaluations, has been completed. As a result, a comprehensive list of topics and definitions of staff saftey objectives, together with a review procedure that considers the effect of these topics on Design Basis Events, were developed. Phase II of the SEP, the actual evaluation of the eleven older facilities, was approved by the Commission in November 1977 and is now scheduled for completion by May 1982. The original completion date had been January 1981. The principal reasons for the slippage is the fact that the level of effort was underestimated and the other, higher priority efforts—such as response to the TMI-2 accident and equipment qualification reviews—have diverted significant manpower from the SEP effort. Steps have been taken to address these concerns by establishing an Assistant Director for SEP and by the dedication of additional manpower to the program.

Topics not applicable to a plant design or under generic review have been deleted from the plant topic lists. Of the remaining topics for each plant, more than 50 percent are in various stages of review. This effort has progressed to the point where facility Design Basis Event (DBE) reviews, which directly constitute another 25 percent of the topics, have been started concurrent with the review of the remaining plant-specific topics.

The DBE reviews will become the basis for determining the capability of a plant to properly respond to postulated accident/incident scenarios and the need for conformance to current licensing criteria. Most topics and all DBEs will be integrated into a final assessment for each facility to determine the overall requirements for facility upgrading.

One of the major topics in the SEP involves seismic design considerations. Seismic design criteria evolved significantly during the period 1956 to 1967, during which the 11 SEP facilities received their Construction Permits. Consequently, the seismic designs of these plants vary considerably.

The SEP facilities follow two groups based upon the degree to which seismic design was originally considered. The licensees of the earlier SEP facilities are embarking on seismic re-evaluation programs of their own to supplement the existing data base which is for the most part far less rigorously developed than would be expected today. These programs are being developed such that they are comprehensive enough to provide the staff with sufficient data to enable an overall assessment of the seismic safety of these facilities.

The NRC staff is currently reviewing the original seismic design documentation of the later facilities. In some cases, the existing information has been supplemented by NRC staff studies to verify staff judgments. All of these plants have been visited to date by specially staffed seismic teams to gain first hand knowledge of facility geometry and to visually identify any anomalies.

In September 1978, a team of expert seismic consultants was formed to assist the staff in reviewing the plant designs. The team had completed a review of Dresden 2 and the printed version of the evaluation report from the team on that review was pending at the close of the report period. No major deficiencies in the seismic design of this facility which would affect public health and safety have been identified, but several issues have been identified which will require more detailed studies, and possibly retrofitting, to verify the adequacy of the seismic design to meet the intent of current design criteria.

## ANTITRUST ACTIVITIES

As required by law since December 1970, the NRC has conducted prelicensing antitrust reviews of all applications for nuclear power plants and certain other commercial nuclear facilities. These reviews assure that the issuance of a particular license will neither create nor maintain a situation inconsistent with the antitrust laws. The NRC holds a hearing whenever one is recommended by the Attorney General and also considers whether antitrust issues raised by the NRC staff or intervenors should be subject to a hearing. Remedies to antitrust problems usually take the form of conditions attached to licenses. Such license conditions may result either from hearings or from non-hearing negotiated settlements.

Antitrust hearings are held separately from those on environment, health and radiological safety matters. So that antitrust reviews do not delay NRC licensing decisions, applicants are required to submit specified antitrust information to the NRC at least nine months, but not earlier than 36 months, before other parts of the construction permit applications are filed for acceptance review. Additionally, NRC performs antitrust reviews prior to issuing operating licenses to determine whether significant changes in applicants' activities have occurred since the construction permit antitrust reviews which would necessitate an antitrust hearing.

Since the inception of NRC's antitrust program, 90 initial construction permit antitrust reviews have been performed and one is pending. Based on these reviews, the Department of Justice recommended 17 for hearing, 24 for "no hearing" because applicants agreed to antitrust license conditions, and 49 for "no hearing," without need for conditions. In addition to these reviews, NRC has reviewed and sought advice from the Department of Justice in 34 cases in which additional applicants are seeking part ownership participation in nuclear plants for which the initial applications had been reviewed previously. No hearings have been recommended for these additional applicants.

The NRC has also sought the Attorney General's advice for two applications for operating licenses where the Commission determined that significant changes in the applicants' activities have occurred. The Attorney General recommended hearings in both cases. Additionally, the NRC staff has completed operating license reviews of twelve applications in which it found no significant changes to have occurred and is currently reviewing twelve others.

In its antitrust program, NRC has reviewed over 170 private, public and cooperative utilities, which accounted for approximately 84 percent of total kilowatt hour sales in the United States in 1977. The NRC has reviewed 72 of the top 100 utilities, ranked by kilowatt hour sales, in the United States.

Significant developments have occurred during fiscal year 1979 in several antitrust proceedings. These developments include the following:

- On September 6, 1979, an Atomic Safety and Licensing Appeal Board issued its decision on the antitrust hearing conducted for the application by five Ohio and Pennsylvania utilities to construct and operate the Davis-Besse Nuclear Power Station, Units 1-3, and the Perry Nuclear Power Plant, Units 1-2. The appeal board's decision essentially affirmed the initial decision of the licensing board and supported the position of the NRC staff that issuance of licenses to the applicants of these facilities would tend to maintain a situation inconsistent with the antitrust laws. The appeal board also affirmed, with some modifications, the license conditions believed necessary to remedy the situations inconsistent with antitrust laws found by the licensing board.
- In June 1978, the NRC issued a Notice of Violation to the Cleveland Electric Illuminating Company (CEI) regarding non-compliance with antitrust conditions imposed on the Davis-Besse Unit

1 license pertaining to transmission services for the City of Cleveland. Since that time, the NRC has issued an order modifying the license, the Cleveland Electric Illuminating Company has requested a hearing, and the Department of Justice has requested NRC to impose civil penalties on CEI. These matters are currently under consideration.

- On June 28, 1978, the Commission ordered an antitrust hearing with respect to Florida Power and Light Company's application to construct and operate the St. Lucie, Unit 2, Nuclear Power Plant. The Commission's decision was in response to a late petition to intervene and request a hearing filed by the Florida Municipal Utilities Association and several Florida cities. During 1979, discovery procedures were initiated but were subsequently delayed pending possible settlement. Settlement negotiations are continuing.
- In 1978, the Attorney General advised the Commission that "significant changes" had occurred since the construction permit antitrust reviews for both the South Texas and Comanche Peak facilities. Consequently, the Attorney General recommended hearings in both cases. During fiscal year 1979, the discovery phases of both antitrust proceedings have been consolidated because of overlapping issues and parties. Hearings are scheduled to begin in 1980.
- Discovery has been progressing in the antitrust proceeding for the Pacific Gas and Electric Company's application for its Stanislaus nuclear power plant.
- The Commission has put in effect and published in the *Federal Register* two changes in its antitrust review procedures. The first is an effective rule that reduces or eliminates the burden of reporting antitrust information by license applicants who own small amounts of generating capacity. The second is a revised procedure by which the Commission has delegated to the staff the authority to determine during the operating license antitrust review whether, since the construction permit antitrust review was completed, "significant changes" have occurred in an applicant's activities which would raise antitrust concerns.

## INDEMNITY AND FINANCIAL PROTECTION

### Increase in Levels Of Required Financial Protection

In January 1979, the two nuclear energy liability insurance pools, American Nuclear Insurers (ANI) and Mutual Atomic Energy Liability Underwriters (MAELU), informed the Commission that effective January 1, 1979, the combined maximum amount of primary liability insurance available from the pools would be increased from $140 million to $160 million.

Subsection 170b. of the Atomic Energy Act of 1954, as amended, requires licensees of commercial nuclear power plants having a rated capacity of 100,000 electrical kilowatts or more to provide proof to the Commission that they have financial protection in an amount equal to the maximum amount of liability insurance available at reasonable cost and on reasonable terms for private sources. In view of the increase, the Commission amended Part 140 of its regulations to increase that amount of primary financial protection required for certain reactor licensees effective May 1, 1979, to give these licensees adequate time to purchase this insurance. In addition, in compliance with 10 CFR Part 140, those persons licensed to possess plutonium in the amount of five kilograms or more and persons licensed to process plutonium in the amount of one kilogram or more for use in plutonium processing and fuel fabrication plants were also required to provide financial protection in the amount of $160 million.

Subsection 170c. of the Act provides that (a) the aggregate indemnity for all persons indemnified in connection with each nuclear incident shall not exceed $500 million, and (b) that this amount of indemnity shall be reduced by the amount that the financial protection required exceeds $60 million. The aggregate liability for each nuclear incident is limited to $560 million. Presently, the secondary financial protection layer is $335 million, (i.e., 67 licensed operating power reactors over 100 MW(e) x $5 million). As a result of the increase in the pools' underwriting capacity, Government indemnity will be $65 million for facilities required to maintain the maximum amount of financial protection, i.e, $560 million less financial protection of $495 million ($160 million plus $335 million). Government indemnity for large power reactors will be phased out when the sum of the first and second layers provides liability coverage of $560 million. Under the current level of primary financial protection required by the Commission, this will occur when 80 commercial reactors have been licensed. After that point, the limit of liability for a single nuclear incident would increase without limit in increments of $5 million for each new commercial reactor licensed.

### Financial Protection for Three Mile Island Units 1 and 2

On May 1, 1979, ANI and MAELU informed the Commission and Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company, the holders of licenses authorizing operation of the Three Mile Island (TMI) Nuclear Station, Units 1 and 2, that because of the March 28, 1979, accident at TMI, (see Chapter 2) the pools were unwilling at that time to make $160 million in nuclear liability insurance available for the TMI site, despite the licensee's request for such increased coverage. The pools' principal reason for not increasing the primary insurance available (from $140 million to $160 million) for TMI was their desire to limit clearly to $140 million their potential liability for claims and claims expenses arising out of the March 28 accident. The pools were opposed to increasing the primary insurance layer to $160 million because they could not be assured that the additional $20 million would not be used to satisfy public liability claims associated with the March 28 accident which arise either prior to or subsequent to May 1, 1979.

The Commission notified the licensees for TMI that it will be necessary for them to demonstrate that they are in compliance with NRC regulations by providing to the Commission evidence that $160 million in primary financial protection for both Units 1 and 2 is in place as of May 1, 1979, i.e., is effective as of that date. At present, the primary financial protection being provided for the Three Mile Island site is $140 million. The insurance pools have proposed an endorsement, which the Commission staff has reviewed and finds to be acceptable, that would provide $140 million in primary insurance to both Three Mile Island, Units 1 and 2, with an additional $20 million for Unit 1. The licensee is presently trying to obtain additional insurance coverage of $20 million apart from the present policy maintained by the licensee with the insurance pools.

On a related matter, the indemnity agreement executed by the licensee and the Commission requires that, in the event of payments made by the insurers under an insurance policy used as financial protection which reduces the aggregate limit of the policy, the licensee must apply to its insurers for reinstatement of the amount of these payments. The licensee has requested reinstatement of the approximately $1.3 million paid out for claims and claims expenses arising out of the March 28 accident. Insurance pools representatives have informed the Commission staff that they have decided not to reinstate these funds for Unit 2, although they will reinstate them for Unit 1 through a separate supplementary insurance policy. The practical effect of not reinstating that funds paid out for the March 28 accident is that, if there were another accident at Unit 2, there would not be the full amount of primary liability insurance to pay public liability claims resulting from such an accident.

If damages in a new accident exceed $140 million and the secondary financial protection layer is utilized, then other power reactor licensees will make up both the $20 million and $1.3 million claims expenses differences through the retrospective premium assessment, by contributing at an earlier point to their share of the damages than would be the case if the accident had occurred at some other site with $160 million in primary insurance. If the damages exceed both primary and secondary financial protection layers, then government indemnity would make up for the increment of $20 million and would be a maximum of $85 million instead of $65 million. The limitation of liability would remain at $560 million. Total protection for the public would be unchanged.

## Indemnification of Storage of Spent Fuel At Distant Reactor Locations

In November 1977, after public notice, the Commission issued amendments to the operating licenses of Carolina Power and Light Company's (CP&L) Brunswick Steam Electric Plants, Units 1 and 2, and H. B. Robinson Steam Electric Plant, Unit 2, to authorize CP&L to store irradiated fuel from the Robinson reactor in either of the spent fuel storage pools at the Brunswick facility and to have this storage indemnified. On January 8, 1979, the Commission published a notice in the *Federal Register* (44 FR 1751) requesting public comment on specific requests by two other utilities, Duke Power Company and Commonwealth Edison Company, to indemnify spent fuel stored at a reactor site different from the one where it was generated, as well as the generic issue of indemnification of spent fuel generated at one reactor but stored at another. Sixteen comments were received by the Commission and evaluated by the staff. The questions relating to the storage of spent fuel are presently under consideration by the staff.

## Claims Handling Procedures Following TMI

Representatives of American Nuclear Insurers (ANI) arrived at Harrisburg, Pa., on March 29, 1979, the day after the Three Mile Island accident and began to assess the desirability of establishing a claims office. Following the advisory by the Governor of Pennsylvania that pregnant women and pre-school aged children living within a five mile radius of the plant should leave the area, ANI established a claims office to pay claims for living expenses for these people, as well as others who had special medical problems.

On March 31, 1979, the first day of operation at the emergency claims center, ANI made payments of almost $12,000. The payments increased daily and reached a peak of $167,286 on April 9, 1979. As of the end of 1979, cumulative payments for evacuation expenses and lost wages made to approximately 12,000 individuals were $1,306,495.

A total of over 4,200 claims were received by ANI in 1979, including 116 economic consequence claims. Not included in the total payments were the expenses incurred by the insurance pools, totaling approximately $160,000 at year's end.

### Determination of an Extraordinary Nuclear Occurrence

On July 23, 1979, the Nuclear Regulatory Commission published a notice in the *Federal Register* (44 FR 43128) that the Commission was undertaking a determination as to whether the March 28, 1979, accident at the Three Mile Island Unit 2 reactor (TMI-2) constituted an "extraordinary nuclear occurrence" (ENO) as defined in the NRC regulations, 10 CFR Part 140, subsections 140.84 and 140.85. On August 17, 1979, the Commission directed that a panel composed of members of the principal staff be formed to evaluate public comments received in connection with our July 23 notice and other technical information assembled by the Commission from its own and other sources. In late December 1979, the panel completed its investigation, evaluation, and analysis and reported to the Commission its findings and recommendation. The panel recommended that the Commission determine that the Three Mile Island accident did not constitute an ENO as defined in the Commission's regulations. This recommendation is advisory only. The Commission will make the final determination as to whether the accident constitutes an ENO. If the Commission accepts the panel's recommendation, defendants in Three Mile Island lawsuits would not be required to waive certain traditional defenses available to them and claimants would have the same rights that they normally have under existing negligence law.

### Indemnity Operations

As of September 30, 1979, 134 indemnity agreements with NRC licensees were in effect. Indemnity fees collected by the NRC from October 1, 1978, through September 30, 1979, totaled $1,068,175. Total fees collected since the inception of the program are $20,103,254. Future collection of indemnity fees will continue to decrease as the indemnity program is phased out for commercial reactor licensees. No payments have been made under the NRC's indemnity agreements with licensees during the 22 years of the program's existence.

### Insurance Premium Refund

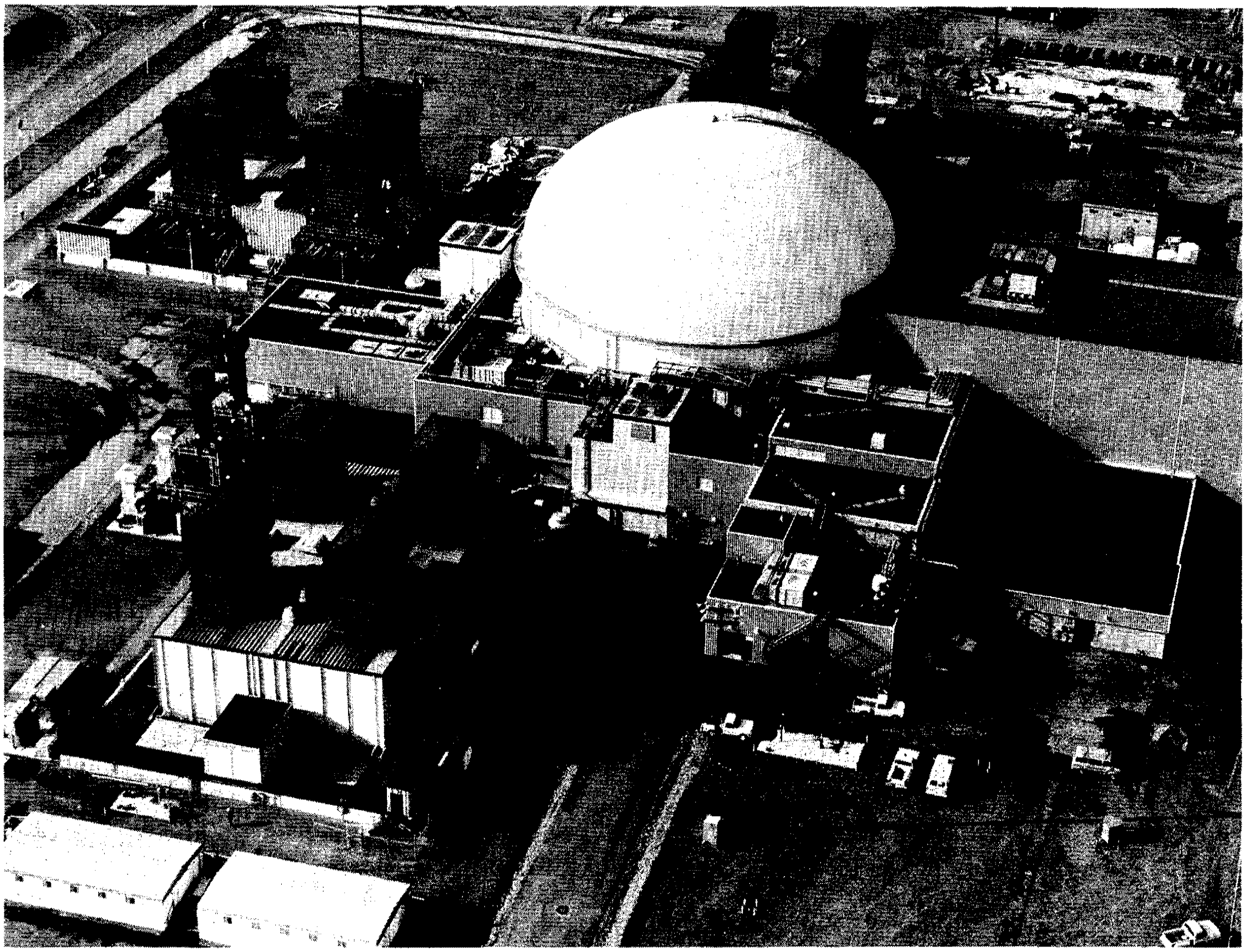
The two private nuclear energy liability insurance pools, American Nuclear Insurers and the Mutual Atomic Energy Liability Underwriters paid to policyholders the thirteenth annual refund of premium reserves under their Industry Credit Rating Plan. Under the plan, a portion of the annual premiums is set aside as a resrve for either payment of losses or ultimate return to policyholders. The amount of the reserve available for refund is determined on the basis of loss experience of all policy holders over the preceding 10-year period. Refunds paid in 1979 totaled, $2,054,989, which is approximately 60 percent of all premiums paid on the nuclear liability insurance policies issued in 1969 and cover the period 1969-1978. The refunds represent 84.9 percent of the premiums placed in reserve in 1979.

## ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards is an independent panel of advisors statutorily established to review construction permit and operating license applications for nuclear power reactors and other nuclear facilities and to report its findings to the Nuclear Regulatory Commission (NRC) which are made part of the public record. The Committee also provides advice to the Commission on a wide variety of safety-related issues such as the adequacy of proposed reactor safety standards, reactor safety research, specific technical issues of a topical nature, and the safety of operating power reactors. Topical reviews are performed by the Committee upon request by the NRC Commissioners or upon its own initiative. Upon request by the Department of Energy (DOE), the Committee reviews and provides reports with regard to the possible hazards of DOE nuclear activities and facilities. An expansion of the Committee's statutory responsibilities (Public Law 95-209) also requires Committee review of the NRC's Reactor Safety Research Program and submittal of an annual report to the Congress regarding its adequacy.

During fiscal year 1979, the Committee reviewed construction permit applications for two nuclear power units and operating license applications for four nuclear power units. The Committee also completed its review of the request for Preliminary Design Approval for a balance-of-plant standard safety analysis (BOPSSAR) for Fluor Pioneer Services Incorporated Balance-of-Plant design as applied to the Babcock-205 Standard Nuclear Steam Supply System (SNSSS). The Committee took note of and took no exception to the NRC Staff recommendations to license the Millstone Nuclear Power Station Unit 2 to operate at an increased power level of 2700 MWt.

The Committee completed its review of proposed operation of the DOE Fast Flux Test Facility (FFTF), a 400 MWt sodium cooled fast reactor located at DOE's Hanford Reservation in Benton County, Washington. The FFTF design and its use of sodium as a coolant make this reactor considerably advanced in



The DOE Fast Flux Test Facility (FFTF) at Hanford, Washington, will be an important test bed for determining the behavior of fuels, alloys and other materials under breeder-reactor operating conditions. Both the NRC staff and the ACRS were involved in safety reviews for this unique facility, under agreement with the DOE. The FFTF was completed late in the year and expected to go critical early in 1980.

comparison to light-water-cooled reactors (LWRs) and, consequently, require standards and operating limits which differ from those applied to LWRs.

At the request of the NRC, the Committee prepared and in July 1979 submitted to the Commissioners "Comments on the NRC Safety Research Program Budget," NUREG-0603. This report included comments on the budget levels and program plans for the supplemental request for fiscal year 1980 to support research related to the accident at Three Mile Island Nuclear Station Unit 2 (TMI-2) as well as for the fiscal year 1981 budget. Special attention was focused on both the short- and long-term implications of the TMI-2 accident.

As a result of the accident at Three Mile Island Unit 2 (TMI-2), the Committee met on several occasions to inquire into the event and the effects of this accident on other reactor systems. Eleven subcommittee meetings were held to investigate accident implications, assessments, and lessons learned, and a visit was made to the Babcock and Wilcox (B&W) simulator at Lynchburg, Va., to observe simulator training for reactor operators. Several members of the Committee visited the NRC Emergency Operations Center and TMI-2 site shortly after the accident to gain first-hand information regarding the status of plant conditions. As a result of this concentrated effort, the Committee prepared and submitted to the Commission five reports in which the Committee recommended a number of nuclear power plant studies and changes. The Committee also reviewed and reported to the Commission on NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." At the request of the NRC/TMI

Special Inquiry Group, the Committee has provided information regarding significant recommendations applicable to B&W and non-B&W reactors.

During the latter part of the year, the Committee began a discussion of the basic, underlying causes which may have contributed to the accident at TMI-2. This evaluation is to be continued during the forthcoming year.

In other activities, the Committee generated and provided reports to the NRC on the following topics:

- Transportation of Radioactive Materials
- Combination of Dynamic Loads (Interim Report)
- Status of Generic Items Relating to LWRs
- Quantitative Safety Goals
- Proposed Rules on the Shipment of Spent Fuel
- Summary Comparison of Stainless Steel and Zircaloy Fuel Rod Cladding
- Bailly Generating Station, Nuclear 1 Piling Modifications
- Pipe Cracking in Light Water Reactors
- Studies to Improve Reactor Safety
- Licensee Event Reports

During the fiscal year, the Committee prepared the following special reports to the Congress and Congressional Oversight Committees:

- The Committees Second Annual Report to the Congress, 1978 Review and Evaluation of the Nuclear Regulatory Commission's Safety Research Program (NUREG-0496). This report focused on the NRC Safety Research Program with particular attention directed to Loss-Of-Coolant Accident/Emergency Core Cooling Systems, Fuel Behavior, Primary System Integrity, Operational Safety, Advanced Reactor Safety, Extreme External Phenomena, Radiological Effects, Waste Management, Safeguards and Security, Risk Assessment, and Improved Reactor Safety.
- Report to Hon. Morris K. Udall, Chairman, Committee on Interior and Insular Affairs, House of Representatives, on the use of Licensee Event Reports to identify those events which have implications for improved reactor safety.

Advice to the NRC was provided on 18 proposed Regulatory Guides and Standards, which dealt with topics such as the following:

- Loose Parts Detection
- Medical Evaluation of Personnel Requiring Operator Licenses
- Quality Assurance Program Requirements
- Containment Isolation Provision for Fluid Systems
- Fuel Oil Systems for Standby Diesel-Generators
- Design Guidance for LWR Plant Radioactive Waste Management Systems
- Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units in LWRs
- Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems

The Committee reviewed proposed amendments to Appendices to 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, on the following:

- Fracture Toughness Requirements
- Reactor Vessel Material Surveillance Program Requirements
- Modification of Emergency Core Cooling Systems

In addition, the Committee reviewed plans to implement 40 CFR 190, "Environmental Radiation Protection for Nuclear Power Operation."

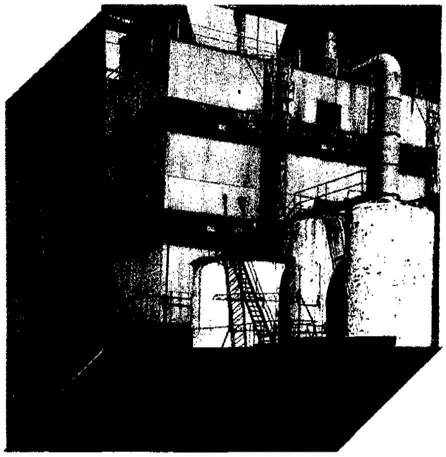
Many other safety-related matters concerning nuclear power facility operations were considered by the Committee. Some of these are as follows:

- Upgraded seismic design bases
- Anticipated transients without scram (ATWS)
- Seismic design for nuclear power plant piping
- Use of probabilistic assessment in the licensing process

The Committee also began activities during the fiscal year related to the probabilistic assessment of selected nuclear power plant incidents, the development of quantitative safety criteria for nuclear facilities, and development of failure rate data for nuclear plant systems and components. These activities will continue into the forthcoming year.

In performing the reviews and preparing the reports referenced above, the Committee met in 12 regular and one special full-session meetings. In addition, 73 subcommittee and working group meetings were held and a total of 7 site-facility visits were made. Comments were received from members of the public with respect to several matters evaluated by the Committee.

Members of the Committee met in Japan with representatives of Japanese nuclear safety agencies to discuss reactor safety policy and practice and the cooperative Emergency Core Cooling System program. Members of the Committee also participated in an NRC Staff meeting with the German Reactor Safety Commission (Gesellschaft Fur Reaktorsicherheit) to exchange experience and knowledge in the areas of waste management and interim spent fuel storage.



# 4 Materials Regulation

Planning for the decontamination of former materials sites included this Kerr-McGee plant in West Chicago, Ill.

The NRC's responsibilities for regulating the possession, use and disposition of nuclear materials, and the safeguarding of nuclear materials and facilities, are carried out by the Office of Nuclear Material Safety and Safeguards (NMSS) under three major programs: the fuel cycle and material safety program, discussed below; the safeguards program, discussed in Chapter 5; and the waste management program, discussed in Chapter 6. Qualified States in the NRC's State Agreements Program have assumed regulatory authority within their borders over byproduct and source material and small quantities of special nuclear material, and the NRC exercises oversight responsibilities concerning these programs (see Chapter 8).

The fuel cycle and material safety program includes licensing and other regulatory activities associated with (1) the purification and conversion of uranium ore concentrates (after mining and milling) to uranium hexafluoride; (2) the conversion of the uranium hexafluoride (after enrichment in Government-owned diffusion plants) to ceramic uranium dioxide pellets and their fabrication into fuel for light water nuclear reactors; (3) storage of "spent" reactor fuel; (4) transportation of nuclear materials; and (5) production and use of reactor-produced radioisotopes. Among actions in these areas during 1979, the NRC:

- Completed 14 major licensing actions concerning uranium fuel.
- Completed actions on more than 5,100 applications for new materials licenses, license amendments and license renewals.
- Completed reviews of terminated AEC licenses involving source and special nuclear material operations to assist in uncovering indications of possible residual contamination at former licensee sites.
- Implemented Environmental Protection Agency regulations limiting environmental radioactivity from reactor fuel production plants and exposure to the public therefrom.
- Issued the final "Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel" (NUREG-0575).
- Prohibited the use of seven spent fuel shipping casks which did not meet fabrication requirements until an assessment could be made of their safety performance.
- Ordered medical licensees to perform tests for excessive molybdenum-99 levels in technetium-99 produced in generators at medical facilities, and to refrain from administering technetium-99 containing excessive amounts of Mo-99.
- Took a number of steps, in conjunction with the Department of Transportation, to strengthen regulation and inspections of shipments of low-level radioactive wastes to the three commercial burial facilities in the States of South Carolina, Nevada and Washington.

## Fuel Cycle Activities

Analyses of all light water reactor fuel production plants have been completed in order to implement EPA regulations (40 CFR Part 190, effective December 1, 1979) requiring control of environmental radioactivity so that the maximum potential exposure of any member of the public will be limited to 25 millirems. Each license will be conditioned to limit radioactivity in effluents to meet the requirements.

NRC has completed environmental impact appraisals for all uranium fuel cycle production plants. Such appraisals are updated before any license is renewed,

*OPEN PIT MINING OF URANIUM.* **In this view of an open pit mine, the shovel and high-lift in the foreground remove overburden (usually crumbly sandstone) to expose the uranium ore. The pit in the background has reached the depth where the uranium ore can be identified by scanning with a Geiger Counter. It is then marked with small red flags for excavation and hauling.**

**At this mine-mill complex, the short hauling distance from mine to mill is an added advantage. Open pit mines sometimes reach depths of 400 feet.**

and they are also prepared before issuance of major amendments to existing licenses if any significant environmental impact can be anticipated.

## URANIUM FUEL CYCLE SURVEY

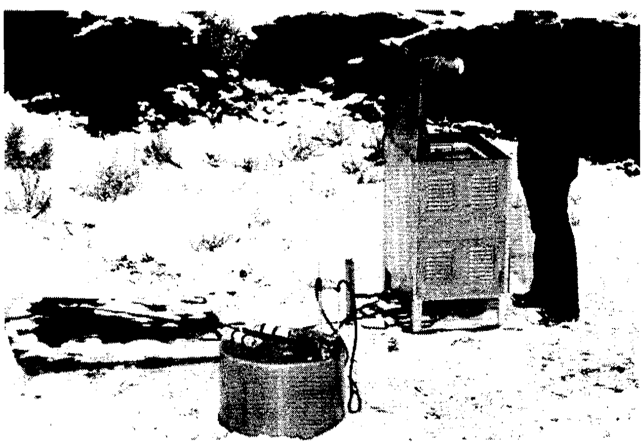
In an effort to avoid repetitive analyses of the fuel cycle in each light water power reactor license proceeding, the Commission in 1974 provided a summary of uranium fuel cycle environmental effects to be expected in support of a typical 1,000-MWe reactor. Estimates of the environmental impact values to be ascribed to an individual reactor were stated in the regulation 10 CFR 51.20, Table S-3. The Commission noted that the impact values would be reexamined from time to time to accommodate new technology and information. Several proceedings, NRC actions and litigation over the matter have taken place since that time. (See 1978 NRC Annual Report, pp. 76-78 for background; also see Chapter 13 of this Annual Report under "Commission Decisions.")

In July 1979 the Commission adopted a revised Table S-3 as a final rule to replace an interim Table S-3 that had been promulgated in March 1977. The final rule, which became effective on September 4, 1979, was the result of extensive rulemaking that included public hearings before a three-member board and oral presentations to the Commission. However, consideration was limited to the environmental effects of spent fuel reprocessing and waste management. Radon and technetium releases are not now included in the table, and the amount and significance of such releases are subject to litigation in individual reactor licensing proceedings.

The NRC staff is studying the impact of the entire fuel cycle in a program to revise and update fuel cycle impact values. A narrative explanation of the Table S-3 summary will include discussions of effects on public health, socioeconomic conditions in fuel cycle plant areas, and the potential risks from radioactivity as long as it endures.

**Improved Radon Estimates.** NRC has initiated several research projects on measurement of radon from uranium mines and mills. Results were published in 1979 by Battelle Pacific Northwest Laboratory, Argonne National Laboratory, Oak Ridge National Laboratory, and Ford, Bacon and Davis Utah, Inc. The data have been used in the draft Generic Environmental Impact Statement on Uranium Milling, and in developing a new environmental impact assessment of radon releases from uranium mining and milling. A technical report on radon impacts is being prepared as the basis for amending the radon value in Table S-3. It will be published in 1980 in advance of proceedings to amend the rule.

**Appeal Board Hearing on Radon.** Pending amendment of Table S-3, the NRC staff has presented testimony on the new data and revised estimates for radon in 17 individual nuclear power plant license proceedings. The different licensing boards involved all concluded that the environmental and human health effects of radon were not significant.



Uranium ore and traces of uranium in topsoils or overburdens associated with mining emit the naturally radioactive gas radon-222. Equipment such as this continuous radon emission monitor, designed and built as part of an NRC research program by the Argonne National Laboratory, is used to measure radon ground flux and air concentrations—data essential to the protection of miners.

There were 17 nuclear power plant proceedings before appeal boards in 1978 when the Commission deleted the radon-222 value from Table 3 and opened this issue for litigation. Intervenors who had been participating requested reopening of the radon issue in four cases. By mutual agreement among the parties, a single appeal board hearing on the radon issue is scheduled to be conducted in February 1980.

**Updating Fuel Cycle Rule.** In addition to amending specific portions of the fuel cycle rule, the NRC is working toward updating of the entire environmental survey. A draft report of the updated "Environmental Survey of the Uranium Fuel Cycle" was nearing completion at the end of 1979. It will examine in detail the environmental impacts of the fuel cycle from uranium mining through waste disposal and the decontamination and decommissioning of nuclear facilities. The study will concentrate on the "once-through" fuel cycle with no reprocessing of spent fuel and no recycle of plutonium. Alternatives will be considered, however, so that if reprocessing should be initiated, with attendant recycle of uranium and solidification and disposal of high-level reprocessing wastes, the environmental effects can be taken into account if they occur during the lifetime of reactors under licensing consideration.

## SPENT FUEL STORAGE ACTIONS

The need for storage of spent nuclear fuel continues to stimulate actions by nuclear power plant licensees to increase capacities of storage pools at reactor sites and to ship irradiated fuel from sites with filled pools to others where room is available. Interest also continues in proposals for off-site facilities dedicated to spent fuel storage.

During fiscal year 1979, NRC approved expansion of storage capacity for seven pools at reactor sites, and applications for expanding 12 others were pending at year-end.

### Final Environmental Statement

The final "Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel" (NUREG-0575), issued in August 1979, confirmed findings of the March 1978 draft statement that spent reactor fuel generated through the year 2000 can be stored in a safe and environmentally sound manner by modifying pools at reactor sites or by providing independent storage facilities either at the sites or away from them. The statement discussed and took into account extensive public comments received in response to the draft document, and supported issuance of a rule to specifically cover storage at independent spent fuel installations.

### Movements Between Reactors

An application by Duke Power Company for the transfer of spent fuel from its operating Oconee Nuclear Station, Seneca, S. C., to the utility's McGuire Nuclear Station in North Carolina, which has not yet received an operating license, is an example of methods sought to deal with the shortfall in storage space.

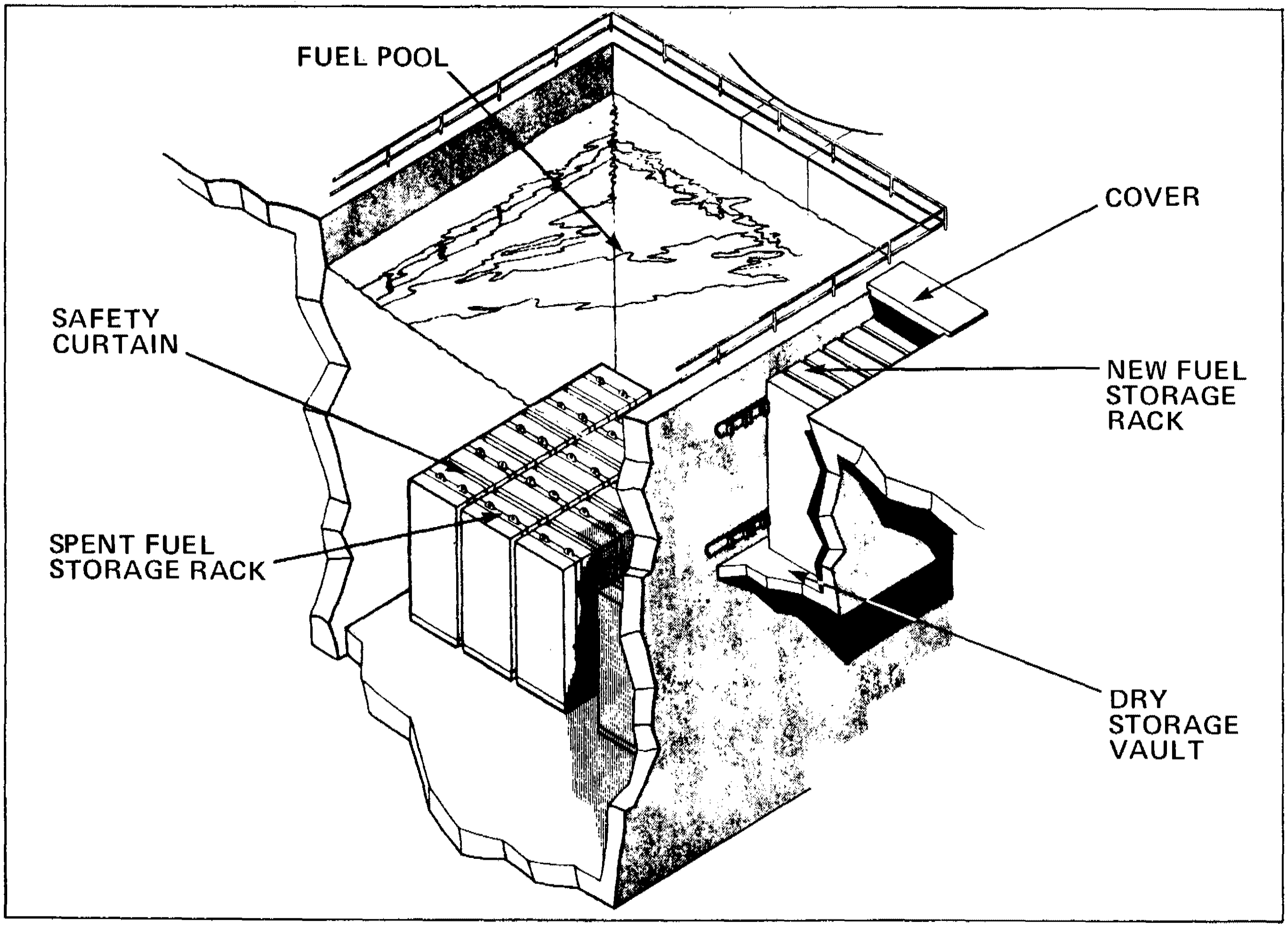
The NRC staff completed safety and environmental reviews of Duke's application during 1979. Approval would require amendment of the McGuire plant's special nuclear material license and authorize receipt and storage of spent fuel generated at the Oconee station. It would be the second licensing action of its kind, approval having been given in 1978 to Carolina Power & Light Company for the intersite transfer of spent fuel from its H. B. Robinson Plant Unit 2 to its Brunswick Station. (See 1978 NRC Annual Report, p. 75.)

The staff's Environmental Impact Appraisal and Safety Evaluation Report were issued in December 1978 and January 1979, respectively. The action has been contested by two intervenors—Carolina Environmental Study Group and Natural Resources Defense Council—and an evidentiary hearing was in progress before an Atomic Safety and Licensing Board at year-end.

### Away-from-reactor Storage

An Atomic Safety and Licensing Board has been appointed to conduct a hearing on the General Electric Company's application for renewal of its license for the receipt and storage of spent fuel at its Midwest Fuel Recovery Plant at Morris, Illinois. Petitions for leave to intervene were filed by the Illinois attorney general and a group of individuals.

The staff extended its review of a topical report by Stone & Webster Engineering Corporation, "Independent Spent Fuel Storage Facility," containing a design for a standard facility to be located on the site of a parent facility such as a nuclear power station. A letter of approval for the conceptual design had been issued in July 1978; a second letter of approval was issued in January 1979 permitting specific sections of the topical report to be referenced in any future site-specific application. The standard installation visualized could store up to 1,300 metric tons of uranium oxide as spent fuel, an amount equivalent to the volume of spent fuel that would be discharged during about 35 years of operation of a 1,000-MWe nuclear power station.




**This is typical spent fuel storage pool for a light water reactor power plant. Pools are designed to particular plant configurations, but generally are box-like containments about 100' long by 50' wide by 40' deep. They are constructed of reinforced concrete, lined with stainless steel. Diagram shown depicts a fuel storage pool at the Browns Ferry Nuclear Plant in Alabama.**

## ADVANCED FUEL ACTIVITIES

### Review of Plutonium Plants

The NRC staff made substantial progress during 1979 in evaluating the integrity and safety of six plutonium processing and fuel fabrication plants which are licensed to possess and process five kilograms or more of unencapsulated plutonium. The objective is to improve, to the extent practicable, the capabilities of these facilities to withstand the effects of adverse natural phenomena and to protect the health and safety of the public (see 1978 NRC Annual Report, p. 75).

Two of the six plants have been completely analyzed and summary documents issued which describe the effects of damage to the facilities from natural phenomena. (NUREG-0547, regarding the Babcock & Wilcox facility at Parks Township, Pa., and NUREG-0621, concerning Westinghouse's facility at Cheswick, Pa.) The analyses of plant capability include site characterization with regard to seismology/geology, surface hydrology, normal and severe meteorology, and the structural capacity to withstand severe seismic and meteorologic events. Analysis of risk to the public involves source term estimation, meteorological dispersion, demography, ecology, and radiological impact. Analyses of the remaining four plants were in varying stages of completion at year-end.

**Babcock & Wilcox License Renewal.** During the year, NRC renewed for a five-year period the license for continued operation by Babcock & Wilcox Company at its Parks Township, Pa., facility, where mixed oxide fuel pins are being manufactured for use in the Fast Flux Test Facility at Richland, Wash. The staff determined that adequate protection is provided for employees and the public after evaluation of administrative procedures, management commitments and engineered safety features, potential impact on the environment, and capability of the plant to withstand severe natural phenomena. As a result of the review, B&W improved certain ventilation and fire safety features. License conditions require decontamination of the facility at the end of its useful life, and the inclusion of a B&W financial commitment to carry that out.

**Decommissioning Activities.** During 1979, the Babcock & Wilcox Company removed all contaminated equipment from its high-enriched uranium fuel fabrication plant at Parks Township, Pa., and completed other decontamination requirements. Final decommissioning and release of the site for unrestricted use will be completed when B&W determines the future use of the facility.

The Kerr-McGee Nuclear Corporation has begun to decommission its plutonium fuel fabrication plant at Cimarron, Okla., which is in a shutdown, standby condition. Current work involves dismantlement of the solvent extraction equipment used in liquid scrap recovery; license amendments will be needed to permit further decommissioning efforts.

At year's end, Westinghouse announced plans to decontaminate and decommission its plutonium fuel fabrication facilities, and other firms were considering similar actions.

## OTHER FUEL CYCLE ACTIVITIES

### Evaluating Sites for Radioactivity

The NRC continues to be active in evaluating sites of former radioactive material operations in order that corrective action can be taken wherever required to protect the public.

**Formerly Licensed Sites.** In response to an earlier General Accounting Office inquiry concerning potential radiation safety problems at sites previously operated under AEC licenses, the NRC has been ex-

**An NRC Region III radiation specialist performs a radiation survey in connection with the decontamination and dismantling of the Kerr-McGee thorium plant at West Chicago, Illinois. Such radiation surveys are used in determining alternative methods of decommissioning.**

amining the files of licenses terminated before 1965 to ascertain that proper decontamination has been carried out.

The Oak Ridge National Laboratory has completed for NRC the evaluation of docket files for old source and special nuclear material licenses, and identified approximately 225 sites which require further evaluation. Further investigations of these sites will be carried out by either NRC or the Agreement States where involved. The next step in this program will be the evaluation of docket files for byproduct material licenses issued before 1965.

**Surveys of Burial Sites.** NRC is planning for radiological surveys at the Westlake landfill, St. Louis County, Missouri, and the Reed-Keppler Park, West Chicago, Ill., where radioactive materials were buried in the past. The surveys will define the location and quantities of materials present in order that corrective actions can be taken.

**Kerr-McGee Site, West Chicago.** The Kerr-McGee Corporation has submitted a plan for removal of factory buildings, decontamination of the site, and stabilization of the ore residue area at a thorium and rare-earth compound production plant formerly operated by Kerr-McGee and predecessor companies at West Chicago, Ill. Production operations ceased in 1973, and the facility is retained under a possession-only license. The licensee's plan is being reviewed by Federal, State and local authorities, and the NRC is preparing an environmental impact statement.

### West Valley, N. Y., Facility

The future of the West Valley, N. Y., site of the shutdown spent fuel reprocessing plant formerly operated by Nuclear Fuel Services, Inc., remained to be settled at the end of fiscal year 1979. A Congressionally mandated study by the Department of Energy of options for the future of the site and/or allocation of responsibility among the Federal and State governments and NFS has been issued to the public for comment and submitted to the Congress for consideration. Formal agreements and decisions on the disposition of the site have not been made, although some preliminary discussions between Federal and State officials have begun.

One of the key aspects of deliberations on what should be done involves the disposition of the high-level liquid wastes presently stored in underground tanks. The NRC staff has been assessing the safety of continued storage of these liquid wastes. As part of this assessment, a test of a level detection instrument showed that the pan under the tank containing the high-level, neutralized waste is defective. The same test confirmed that the tank itself is not leaking. The loss of the pan as a secondary collection point for any tank leakage that might occur is undesirable, and underscores the need for commencement of dedicated work to remove, solidify, and dispose of the high-level liquid wastes at West Valley. The staff, through its contractors, is continuing its program to assess the safety conditions associated with this storage.

The NRC staff has continued its assessment of the effects of severe natural phenomena on the dormant reprocessing facility. Analysis of the effects of a severe earthquake on the fuel receipt and storage pool and the carbon steel high-level liquid waste tanks has demonstrated that there is no undue risk to the health and safety of the public from the effects of an earthquake. Analysis of the effects of tornadoes on the separations plant is in progress.

# Transportation of Radioactive Materials

Transportation of radioactive materials is regulated at the Federal level mainly by the NRC and the Department of Transportation (DOT). NRC sets the standards for "Type B" packages (those whose content of radioactive materials requires that they be safely retained in their containers under both normal and accident conditions) and for packages containing fissile material. NRC also makes independent evaluations of package designs submitted by applicants and serves as a technical advisor to DOT. The roles of the DOT and the NRC were redefined in a memorandum of understanding signed by the two agencies in June 1979, and published in the *Federal Register* on July 2, 1979.

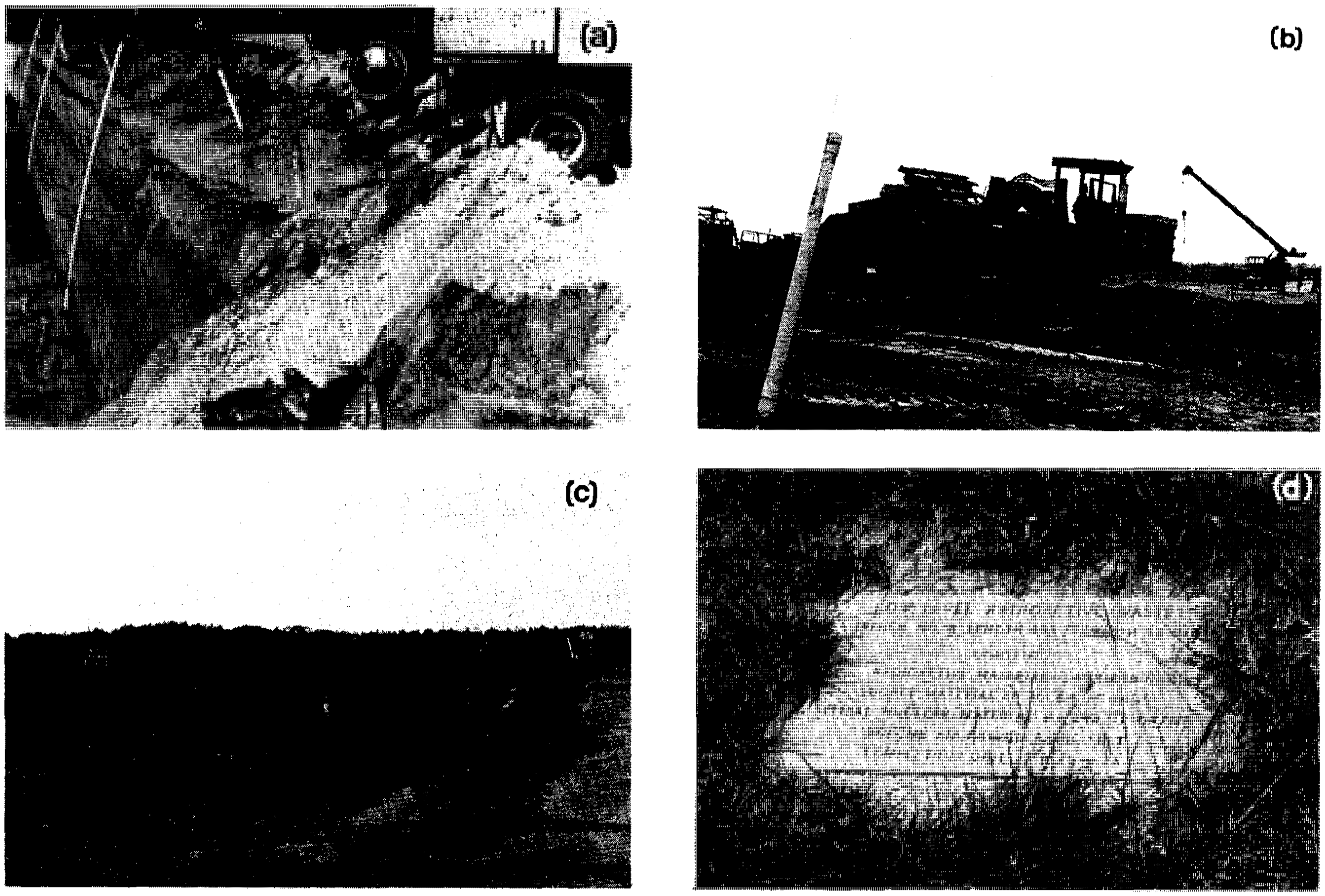
Package designs used by contractors for the Department of Energy (DOE) are reviewed and approved by that agency; however, the NRC has been reviewing such package designs on a continuing basis. These NRC reviews are not binding on the DOE, but an NRC approval permits commercial licensees to use these packages.

## Low-Level Waste Shipments

In the summer of 1979, attention was called to a number of instances in which packages of low-level waste were not in compliance with Federal requirements on arrival at one or more of the three commercial burial facilities in the country. Such items of noncompliance included a fire on a truck carrying combustible waste, leaking packages, truck contamination from improperly closed packages, free liquid in packages of supposedly dry solid material, inadequately labeled packages, and improperly documented shipments. While none of these items of noncompliance by itself represented a significant health concern, collectively they showed a lack of proper attention to Federal requirements for packaging and shipping of radioactive waste materials.

The Governors of the Agreement States of South Carolina, Washington, and Nevada, in which the commercial burial facilities are located, notified the NRC of the repeated disregard for the Federal transportation rules and at various times closed or limited these facilities to certain shippers. The NRC, in conjunction with the DOT, determined that the Federal Government should improve its assurance that Federal regulations governing such shipments are met and took several steps:

- The NRC changed its regulations to specifically subject its licensees to DOT requirements, and thus effectively increase the Federal inspection capability.
- The NRC issued bulletins to all licensees to (1) inform them of the transportation incidents that occurred, of requirements for transportation of low-level radioactive waste materials, and of burial site requirements, and (2) direct licensees to submit written management-approved procedures for the safe transfer, packaging, and transportation of these materials.
- The NRC increased its inspections at shipper and receiver sites.
- The NRC modified its enforcement criteria to increase penalties.
- The NRC and DOT are jointly investigating ways to improve the safety of low specific activity material packages.
- The NRC is acquiring support from the Society of Nuclear Medicine to improve medical waste packages and from the Atomic Industrial Forum to improve industrial waste packages.
- The NRC and DOT are making an effort to better inform shippers of requirements.
- The NRC is developing a primary draft regulation for burial of low-level wastes.




*Low Level Waste Burial*

(a) Low level waste containers are dumped into burial trench and quickly covered as portions of the trench are filled. Instrumented pipes are emplaced along deep edge of trench to measure the radioactivity of any underground seepage which might occur. Trench bottom is sloped toward instrument side. (b) Instrument pipe protrudes above covered trench as burial nears completion. (c) Temporary markers are used following burial and grassing of covered area (instrument pipes again are visible at right). (d) A permanent marker specifying dimensions of trench and its contents is emplaced when earth cover is settled and grassed over. This marker at a site in South Carolina describes a pit of 493 feet by 47 feet. Depth of such trenches depends on the medium of burial, depth of water table, etc., but on average, are between 20 and 30 feet deep.

The American National Standards Institute Subcommittee on Transportation of Radioactive Waste, with NRC participation, is preparing a standard for the packaging for transportation of liquid aqueous radioactive wastes from nuclear power plants. It will require that liquid wastes be solidified prior to shipment and that a high-integrity container be used.

In March 1979, the NRC published a draft report, NUREG-0535, "Review and Assessment of Package Requirements (Yellowcake) and Emergency Response to Transportation Accidents." The report was prepared by an NRC/DOT task force following a truck accident in September 1977 in which a shipment of uranium concentrate (yellowcake) was spilled onto a highway near Springfield, Colo. The draft report has been issued for public comment, which will be considered in preparing the final report. (See 1978 NRC Annual Report, p. 81.)

## Irradiated Fuel Packaging

Spent (irradiated) nuclear fuel is transported off-site in specially designed shipping casks that are capable of containing the radioactive fuel assembly materials during normal and postulated design accident transportation conditions.

On April 6, 1979, NRC issued an Order to Show Cause (immediately effective) prohibiting the use of the Model No. NFS-4 packaging until a determination is made that it meets specified requirements. During a meeting on March 29, 1979, and later by letter dated April 2, 1979, Nuclear Assurance Corporation informed the NRC staff that a cask had not been fabricated in accordance with its certificate of compliance. In view of the unknown safety implications for this and other packages fabricated to the same design, and in the interest of public health and safety, all seven packages were removed from use until an assessment could be made of their safety as fabricated. On Dec. 12, NRC permitted three casks to be returned to service. Eighteen owner/users were affected by the orders.

The staff is reviewing two applications for spent fuel casks designed for shipment by rail that are significantly different from design concepts presently used. Each of these designs uses a thick, solid, cylindrical carbon steel containment vessel wall as a gamma shield. Existing designs use lead or uranium in the gamma shield. One factor being considered in the review is fracture toughness of thick steel forgings. The two spent fuel casts are the Transnuclear Inc. Model No. TN-12 and the Nuclear Assurance Corporation Model No. NAC-3K.

## Safety of Transportation Workers

In June 1979, the NRC initiated a study entitled "Radiation Exposure of Transportation Workers Handling Large Numbers of Radioactive Material Packages." The objectives of the study are to:

(1) Identify carriers where employees may receive exposures exceeding regulatory limits.

(2) Determine actual exposures received either by direct measurement or by studying the carrier's records.

(3) Observe procedures in use at carrier's facilities and prepare suggestions for techniques to reduce exposure.

(4) Identify relationship between quantity of radioactive material handled and exposure.

The study began in July 1979 and should be completed by October 1980. The information will be used to prepare a recommendation to DOT on what further measures may be necessary to control radiation exposures in selected portions of the transportation industry.

During fiscal year 1979, the NRC-sponsored State Surveillance Program on Transportation of Radioactive Materials was continued. (See Chapter 8, "State Programs," for details of the program's activities during 1979.)

## Transportation in Urban Areas

During 1979, Sandia Laboratories, under contract to the NRC, continued its work to assess the environmental impacts resulting from the transportation of radioactive materials through urban areas. The study has been examining the impacts resulting from incident-free transport, vehicular accidents during transport, and from other abnormal situations. In performing this study, Sandia has developed computer


Canisters such as these (shown at left) are used in "cleaning" contaminated water at nuclear plants. The steel containers hold resins that create an ion-exchange process as radioactive liquids are flushed through. When the resins are no longer clean, or when a prescribed level of radioactivity is reached, the canisters are placed in sealed cylinders (on flatbed truck/trailer at right) for transport to waste burial sites.

models to account for the special features of the urban environment. The study will form the basis for a generic environmental impact statement, to be published by the NRC, on the transportation of radioactive material in urban areas.

Information resulting from the study suggests that the sabotage of spent fuel shipments has the potential for producing serious radiological consequences in areas of high population density. This information, in part, led the NRC in June 1979 to establish interim requirements for the protection of spent fuel in transit. (See Chapter 5, "Domestic Safeguards.")

## Emergency Response Planning

In October 1978, Indiana University, under contract to the NRC, began work on a study entitled "Survey of Current State Radiological Emergency Response Capabilities for Transportation Related Incidents." The objective is to assemble and condense available information on current state emergency response capabilities for transportation-related radiological incidents in order to assist NRC in its role regarding radiological incident planning, emergency response training, and other assistance activities within State and local governments.

The NRC will use the information obtained from this study to (1) identify response requirements for protection of the health and safety of the public with regard to transportation-related radiological incidents, (2) develop and promulgate guidance to State and local governments in coordination with other Federal agencies for the preparation of emergency response plans, and (3) determine whether additional Federal participation is required to ensure adequate protection of the health and safety of the public with regard to transportation-related radiological incidents. The study will be completed in fiscal year 1980.

## Packaging Standards

In March 1979, the NRC issued for comment Regulatory Guide 7.9, which identifies the information to be provided in an application for the approval of packaging for shipping Type B, large quantity, and fissile radioactive material and presents a uniform format for presenting the information. The guide assists the applicant in preparing an application and ensures the completeness of the information provided. Use of a uniform format assists the NRC staff and others in locating the information and aids in shortening the time needed for the review process.

In August 1979, the NRC issued for public comment a revision to 10 CFR Part 71, "Packaging of Radioactive Material for Transportation and Transportation of Radioactive Material Under Certain Conditions." The NRC is considering revising its regulations for the transportation of radioactive material to make them compatible with those of the International Atomic Energy Agency (IAEA) and thus with those of most major nuclear nations of the world. Although several substantive changes are proposed in order to provide a more uniform degree of safety for various types of shipments, the Commission's basic standards for radioactive material packaging would remain unchanged. The DOT is also proposing a corresponding rule change to its hazardous materials transport regulations.

The major changes to NRC's regulations that are being proposed are:

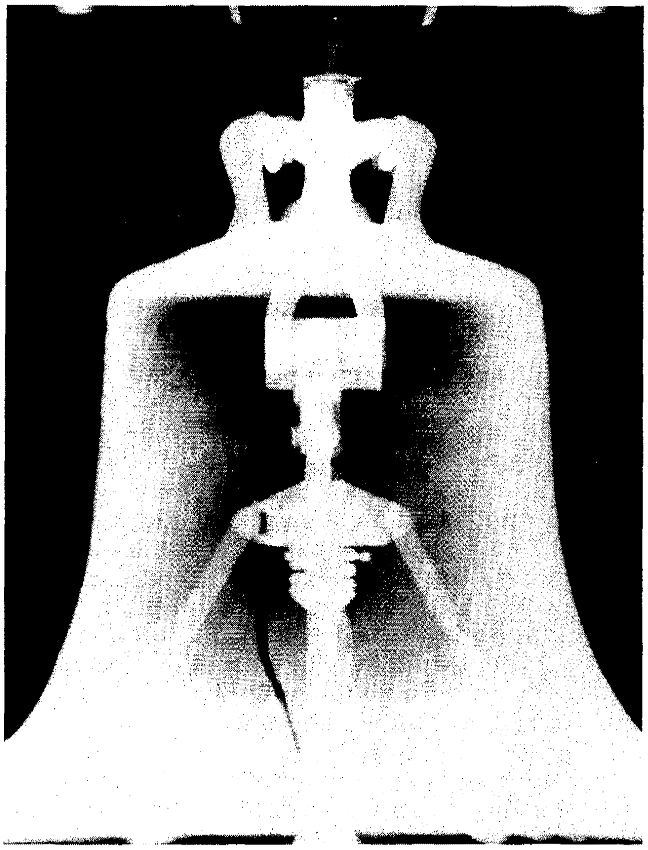
(1) Elimination of the system currently used to specify the quantities of radioactive materials permitted to be shipped in certain types of packages. Under the present system, all radioactive materials are divided into seven transport groups that are used as the basis for determining the amount of those materials that can be shipped in Type A packages and the amounts that must be shipped in the more stringently designed, accident-resistant Type B packages. This system has proved to be unduly restrictive because less hazardous radioactive materials included in one transport group are required to be packed in the same manner as other, more hazardous radioactive materials belonging to the same transport group. Under the proposed rules, the use of a Type A or Type B package would depend on the degree of radioactivity for each material being shipped.

(2) Establishment of two classifications of Type B packages. This change would facilitate foreign acceptance of U.S. export shipments by conforming package types to international standards.

## International Standards

In 1983, the IAEA will issue a revision to its Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Materials." In preparation for this revision, the IAEA has asked member countries to submit proposed changes to the regulations and to identify areas where revision should be considered. The NRC prepared recommended changes to the regulations which were submitted to the IAEA by the DOT, which serves as the U.S. competent authority on matters involving international shipments of radioactive material. In addition, the NRC assisted the DOT in its review of comments submitted by various private organizations and other government agencies.

The NRC also participated as an observer in a meeting of the IAEA Advisory Group on Radiation Protection and Safety Principles for Safe Transport of Radioactive Materials. The meeting was held in July 1979 at the IAEA Headquarters in Vienna, Austria. The purpose of the meeting was to review the principles upon which the IAEA's transport regulations are



This radiograph of the Liberty Bell is reduced from its original size of seven feet by 12⅓ feet, reportedly the largest single sheet of X-ray film ever exposed. The exposure to a 670-curie cobalt-60 source lasted 7½ hours. The famous crack in the bell is the dark irregular line at left center.

based and to provide recommendations to a revision panel that will meet in 1980 to begin the comprehensive review of the transport regulations.

# Radioactive Material Licensing

Radioactive materials have wide use in industrial applications, medical diagnosis and treatment, applied research and development, in the academic fields, and in products distributed to the public. Some 8,500 materials licenses are administered by the NRC which require the annual processing of 5,000 to 6,000 applications for new licenses, license amendments and license renewals. An additional 12,000 licenses are administered by 26 states which have assumed authority over certain materials under regulatory agreements with the NRC, as part of the Agreement States program, (see Chapter 8). The NRC licensing program is designed to provide reasonable assurance that the public health and safety is adequately protected and that applications for licenses are processed in an efficient and timely manner.

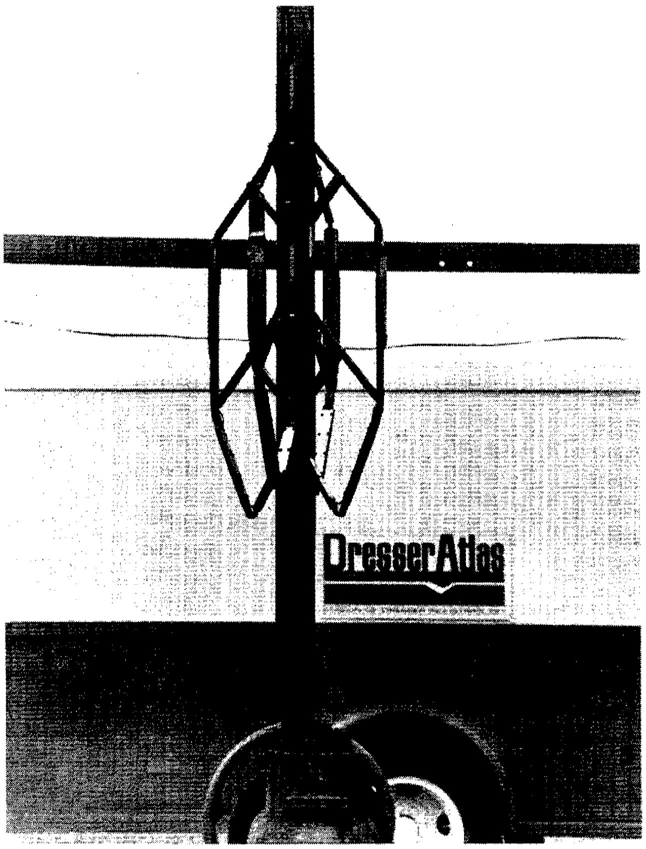
In 1979, the materials licensing function was reorganized. The former Radioisotopes Licensing Branch was dissolved and two new branches were created: the Material Licensing Branch and the Material Certification and Procedures Branch. The purpose of the reorganization was twofold. It permits the Material Licensing Branch to focus its efforts on the review of license applications and the Material Certification and Procedures Branch to devote its efforts to functions related to materials licensing, such as sealed source and device evaluations, preparation of guides for licensees and applicants, the review of regulatory practices, and other non-casework needs.

In March 1978, the NRC initiated a pilot regionalization licensing program to determine the feasibility of conducting licensing activities from NRC regional offices. The initial effort involved six of the eight States in NRC's Region III and reviews were conducted for medical institutions and industrial firms using gauging devices. As the program progressed, all eight of the States were included in the program, and academic institutions and industrial research and development licensees were included in those involved in the regional licensing program. The program is scheduled for completion in early 1980 and the feasibility of continuing or expanding the regional licensing program will be determined.

**Consumer Products.** Numerous products containing small amounts of radioactive materials are in daily use. These products are authorized for distribution only after careful evaluation by the NRC indicates that there is minimal risk to the general public. Among these products are smoke detector devices containing americium-241 and liquid crystal display timepieces containing tritium. A two-year study to determine the environmental impact of the use of radioactive materials in consumer products is expected to be completed in October 1980.

**Gauging Devices.** These devices have wide use for controlling density, levels, thickness, and weight of materials. NRC approves their use only after evaluation of the sealed sources and the devices to determine that the gauging devices may be used safely by individuals who have minimal training and experience in radiation safety. Due to the relatively low radiation levels, normal use of these devices presents minimal hazard to workers and the general public.

**Gas Chromatographs.** These devices typically contain radioactive materials in the form of foil or plated sources containing nickel-63 or tritium. Due to increased concern for the environment and the usefulness in measuring small amounts of materials, gas chromatograph usage has increased dramatically




This well-logging tool uses a radioactive source in taking measurements in boreholes. The neutron source, shown in the upper part of the logging tool, activates the different earth strata as it passes down the hole permitting identification of the composition of the strata.

for analyses of substances which could contain environmentally undesirable constituents.

**Well Logging.** The use of radioactive materials in well logging—in search of new energy sources and or utilization of gas and oil fields formerly thought underproductive—has increased markedly. Most of these activities are being performed by large service companies, although many small companies are also active in the use of well logging techniques.

## Nuclear Medicine

The NRC issues licenses to hospitals and physicians for the use of radioactive materials in the diagnosis and treatment of patients. These diagnostic procedures include both *in vitro* tests involving the addition of radioactive materials to laboratory samples taken from patients and *in vivo* tests, the direct administration of radioactive drugs to patients. *In vivo* tests are for the purpose of:

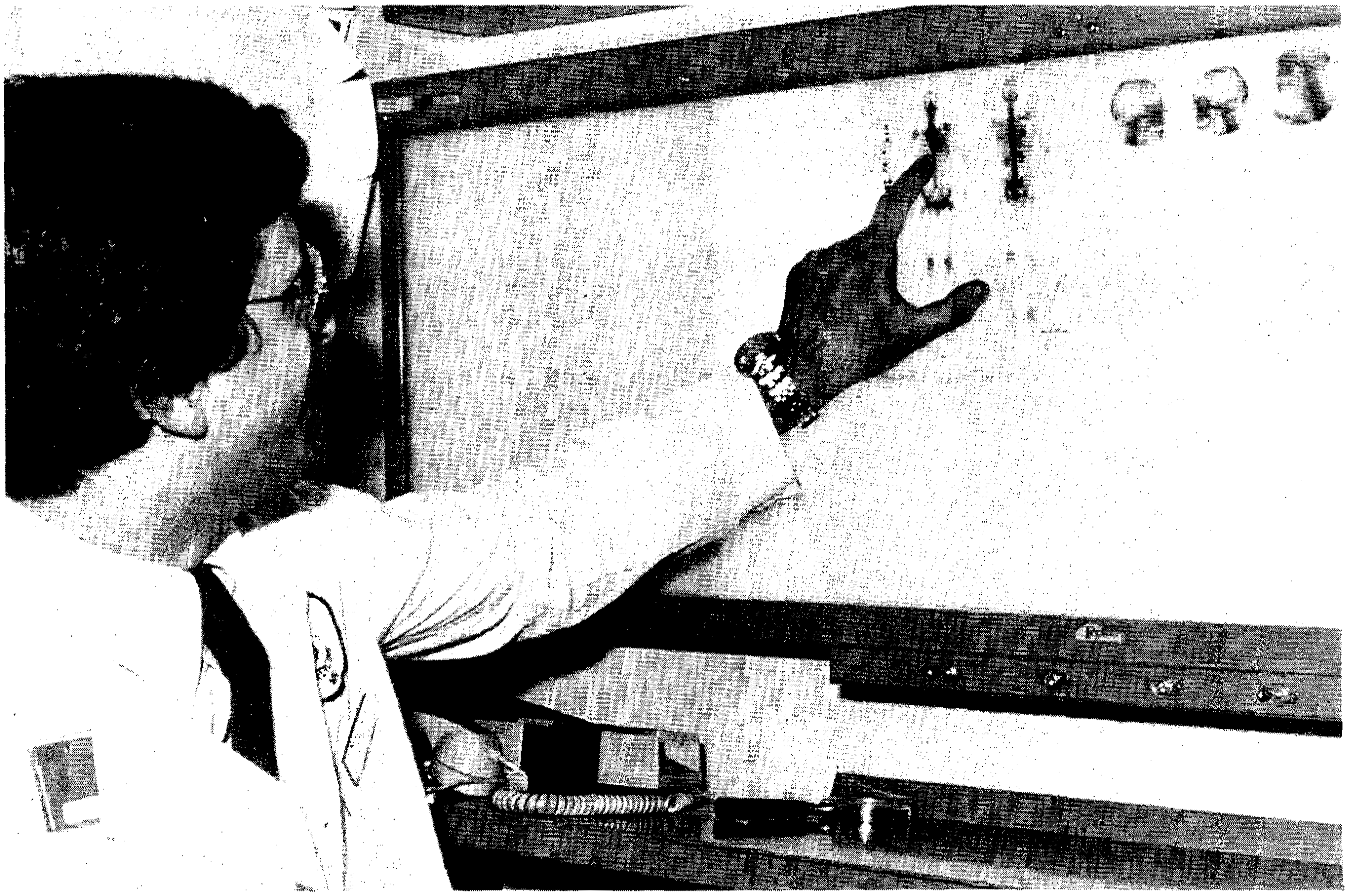
- Measuring the uptake, dilution or excretion of a radioactive drug within an organ system (blood volume determinations and kidney function studies are examples of these).
- Visualizing the distribution of a radioactive drug within an organ in order to locate tumors, blood clots, etc.

Therapeutic radiation treatment continues to be an important element licensed by the NRC. These procedures include the use of liquid radioactive drugs to treat certain medical conditions such as the use of iodine-131 for treatment of hyperthyroidism. In the radiation therapy mode called brachytherapy, encapsulated or sealed radiation sources are placed directly on or in the patient's body to treat cancer. Naturally occurring radium sources with their many problems are rapidly being replaced by NRC-licensed materials such as cesium-137 and iridium-192. Teletherapy is another area of radiation therapy licensed by NRC. In this method the patient is treated at a distance with radiation from a sealed radioactive source, usually cobalt-60.

Currently, the most rapidly growing area in nuclear medicine is nuclear cardiology. Using radioactive materials, physicians are able to identify specific areas in the heart that are not receiving an adequate blood supply, thus predicting potential heart attack victims. Nuclear cardiology studies also enable the physicians to locate infarcted areas and monitor healing and recovery processes after a heart attack occurs.

**Advisory Committee on Medical Uses of Isotopes.** NRC's Advisory Committee on Medical Uses of Isotopes (ACMUI) consists of physicians and medical physics specialists from the public sector and it provides NRC with advice on many medical questions. During a December 1978 meeting, the NRC staff recommended minimum criteria for training and experience of physicians to be licensed to perform nuclear cardiology studies. The committee also discussed NRC's cooperative efforts with the Society of Nuclear Medicine to develop a medical licensee model program designed to minimize radiation exposure to personnel and minimize releases of radioactive material to the environment. During the year, in accordance with NRC's policy of rotating membership on the ACMUI, NRC formally requested nominations from the public to replace several members. Four well-qualified physicians were selected from 47 persons nominated by professional societies and members of the public.

**Order on Use of Technetium-99m.** Many hospitals obtain technetium-99 (Tc-99m), a radioactive isotope widely used in nuclear medicine, from a molybdenum-99 (Mo-99) generator. The generator is a shielded device that contains Mo-99 absorbed onto an alumina column. When a sterile saline solution is fed through the column, Tc-99m, a daughter product, is removed and Mo-99 remains on the column. Very little, if any of the Mo-99 is normally removed with the



A nuclear medicine physician examines a technetium (Tc)-99m whole body bone image. This radiopharmaceutical, a Tc-99m-labeled phosphate compound, localizes in skeletal areas and allows physicians to visualize pathologic bone process.

Tc-99m. Mo-99 in Tc-99m serves no diagnostic purpose, and excessive levels of this isotope in the Tc-99m results in unnecessary radiation exposure to patients. Although it is good practice to check for the presence of Mo-99 as a contaminant, an NRC investigation revealed that these tests were not being routinely performed. Since several thousand generators are sold to medical facilities each week, and one generator can provide Tc-99m for up to 50 patients per day, there is a potential for exposure of large numbers of persons if excessive levels of Mo-99 occur.

On March 12, 1979, NRC issued an order requiring medical licensees to perform a test for Mo-99 and to refrain from administering Tc-99m contaminated with excessive levels of Mo-99. On June 6, 1979, NRC published a proposed rule change that contains the essentials of the order and requested comments from the public.

**Other Significant Actions.** In other 1979 actions having a significant impact on medical licensees, the NRC:

- Published for comment a revised medical licensing guide. Use of the revised application form with the revised guide is making it easier to apply for and obtain a license for the medical use of radioisotopes.
- Issued a final rule requiring teletherapy licensees to perform periodic full calibration and spot-check measurements to ensure accurate determination of the amount of radiation administered to patients.
- Issued a new rule allowing physicians greater latitude in using approved radioactive materials for diagnostic procedures.
- Published a medical policy statement setting forth NRC's future role in regulating the medical uses of radioactive materials.

# 5 Domestic Safeguards

**Fixed site physical security at nuclear facilities was upgraded in new NRC regulations published in November 1979.**

Section 209 of the Energy Reorganization Act of 1974, as amended, requires the Nuclear Regulatory Commission to include in each Annual Report to Congress a chapter describing the status of NRC's domestic safeguards program for the protection of certain nuclear materials and facilities. This chapter discusses safeguards provided for licensed facilities and activities during fiscal year 1979, covering the general areas of (1) scope of NRC safeguards efforts; (2) adequacy and effectiveness of safeguards; (3) safeguards policy issues, regulatory actions, and research and technical assistance; and (4) NRC safeguards organization and management.

## Scope of NRC Programs

The Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974 direct the NRC to regulate the safeguards provided by its licensees for certain nuclear facilities and activities. With the objective of assuring protection of the public health and safety and the national defense and security, the NRC designs and enforces measures to deter, prevent, and respond to (1) unauthorized possession, theft, diversion, or use of special nuclear material; and (2) sabotage of nuclear facilities.

Safeguards for fuel cycle facilities emphasize protection against theft or diversion of "formula quantities" of strategic special nuclear material (SSNM), while power reactor safeguards concentrate on protection against industrial sabotage.

NRC safeguards regulation during 1979 covered the following:

- Nineteen fuel cycle facilities.
- Selected transportation activities.
- Seventy power reactors licensed for commercial operation.
- Seventy-one non-power reactors (for research, testing, training or the production of radioisotopes).

The 19 fuel cycle facilities are authorized to possess formula quantities of SSNM, which includes uranium-235 (contained in uranium enriched to 20 percent or more in the U-235 isotope), uranium-233, or plutonium. A "formula quantity" is 5,000 grams or more of SSNM as computed by the formula: grams = (grams U-235) + 2.5 (grams U-233 + grams plutonium). Four additional fuel cycle facilities are authorized to possess more than one "effective kilogram" of low-enriched uranium and are, therefore, subject to NRC safeguards material control and accounting requirements. For uranium with an enrichment in the isotope U-235 of one percent and above, "effective kilograms" is computed as its weight in kilograms multiplied by the square of its enrichment expressed as a decimal weight fraction.

The selected transportation activities mentioned above involve shipments of spent fuel or formula quantities of licensed SSNM, currently amounting to about 20 per month.

## Assessment of Safeguards Adequacy Policy

The fiscal year 1978 report to Congress on domestic safeguards (NUREG-0524), which was forwarded as a separate document, noted that the several NRC offices with safeguards responsibilities had varying approaches to safeguards regulation. These differences were manifested in definitions of adequacy and also in methods of deciding whether safeguards were adequate for the regulated fuel cycle facilities, transportation activities, power reactors and non-power reactors.

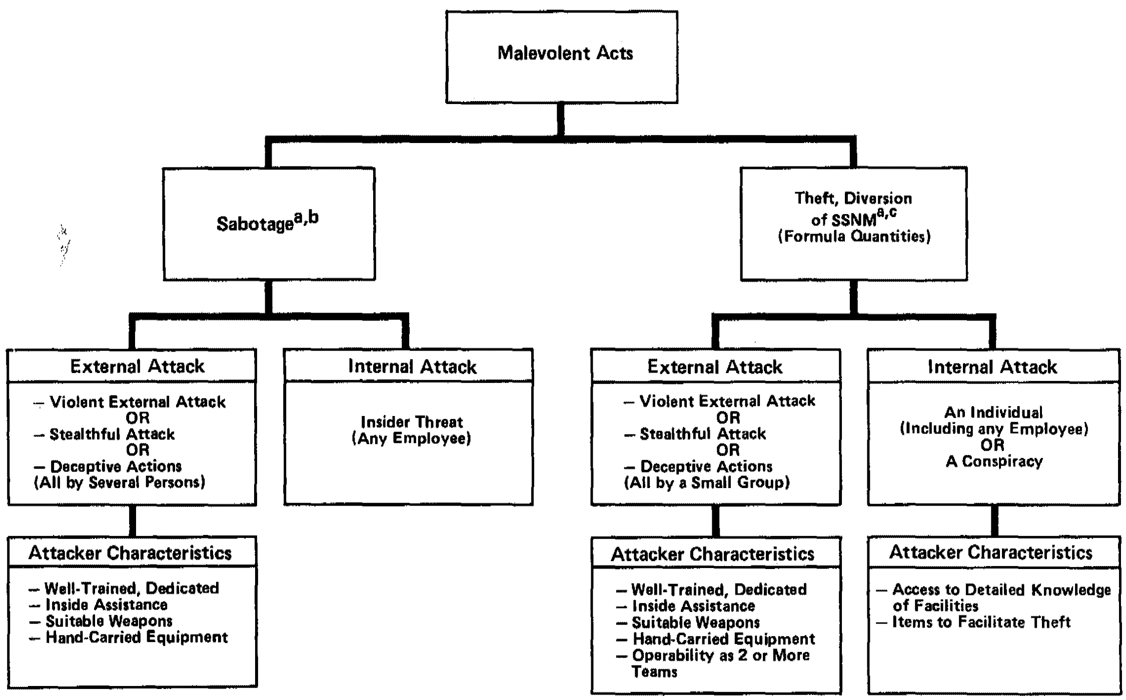
In January 1979, the NRC Executive Director for Operations established an internal "Task Force on Safeguards Policy" to resolve these differences and develop an integrated approach to safeguards regulation. In July 1979, the Commission directed the staff to revise its rules in 10 CFR Part 73 to incorporate the task force's recommendation that safeguards regulatory requirements be generally specified in terms of malevolent acts to be thwarted (i.e., theft and sabotage), rather than by types of facilities to be protected. To help a licensee design a safeguards system, such malevolent acts are characterized as shown in Chart 1.

The rule revisions also specified that the objective of physical protection systems for reactors and fuel cycle facilities is to provide high assurance protection against radiological sabotage, theft or diversion.

In August 1979, as a further step toward eliminating differences and conflicts within NRC safeguards regulatory programs, the Commission decided to consolidate all safeguards review responsibilities within the Office of Nuclear Material Safety and Safeguards. This consolidation became effective on October 1, 1979.

The teams then judged whether the particular facility provided high, good, fair, or poor assurance of protecting against a hypothesized threat. Licensee safeguards judged to provide high assurance of protection against the hypothesized threat are considered to possess the desired level of safeguards capability. Licensee safeguards judged to provide good assurance of protection are considered to be adequate, but not providing the extra measure of capability that NRC deems prudent. Licensee safeguards judged fair are considered adequate to permit continued operation only if the observed deficiencies do not pose an undue

## Characterization of Malevolent Acts




[a] In January 1979, the Commission directed the staff to conduct a study of the potential insider threat to the licensed nuclear industry. The Commission expects to receive the study report by early 1980.

[b] As defined in 10 CFR 73.1(a)(1).

[c] As defined in 10 CFR 73.1(a)(2).

## Fuel Cycle Facilities

In January 1979, the NRC staff completed an 18-month program of comprehensive evaluations of safeguards at the 11 licensed facilities which process formula quantities of nuclear material. Four separate field teams examined each facility with respect to:

- Vulnerability to external assault.
- Physical security.
- Possible inside diversion paths.
- Nuclear material control and accounting.

risk to the public health and safety or common defense and security during the short term required for their correction. Operations at facilities with safeguards capabilities judged good or fair are permitted to continue only where agreement has been reached to undertake an immediate remedial program to bring the licensee's safeguards program back to a state of high assurance. Continued operation of licensee facilities where safeguards are judged poor will not be permitted unless the observed deficiencies can be corrected immediately. To correct deficiencies, NRC modifies licenses as needed to upgrade the systems, and requires the applicable licensees to make the necessary changes. NRC then conducts special follow-up inspections to ensure compliance.

Eight formal reports on findings and corrective actions concerning safeguards were sent to Congress during fiscal year 1979. The other three are completed and are being sent to Congress early in fiscal year 1980.

Table 1 summarizes the teams' initial assessments of the 11 fuel facilities examined. As noted above, if a facility did not receive a high rating, it was required to take corrective actions designed to improve safeguards to the desired level of high assurance. Licensees were allowed varying amounts of time to correct deficiencies, depending on how long it would take for permanent corrective action; however, interim measures had to be put into effect immediately. Inspectors from NRC's Office of Inspection and Enforcement routinely inspect both the interim and the permanent corrective measures taken.

By the end of fiscal year 1979, all required corrective actions for the 11 facilities had been identified, and interim measures required by NRC had been put into effect in those cases where permanent improvements had not been completed. As a result of these improvements coming out of the comprehensive review program, all facilities were judged to provide high assurance of protecting against the hypothesized threat. However, NFS Erwin experienced an excessive inventory difference and was shut down in late September 1979, as discussed below.

On November 28, 1979, the Commission published the Physical Protection Upgrade Rule, requiring licensees to provide strengthened physical protection for formula quantities of special nuclear material. These new requirements will become effective March 25, 1980.

**Inspection and Enforcement at Fuel Cycle Facilities.** NRC inspection and enforcement activities at fuel cycle facilities during fiscal year 1979 included more than 8,000 hours of on-site safeguards inspections at 14 fuel cycle facilities (those authorized to possess formula quantities of unirradiated SSNM in an unsealed form). These inspections revealed 73 items of noncompliance with safeguards requirements. (See Table 2 for summary of inspections.) The NRC took particularly significant action (a shutdown order) against one major licensee, Nuclear Fuel Services, Inc., operator of a plant at Erwin, Tennessee, where the inventory difference for the bimonthly physical inventory exceeded the upper limit specified in the conditions of the license. The licensee had also exceeded the limit of error associated with the inventory difference. Based on inspection results, NRC has judged the safeguards performance of the remaining licensees as satisfactory.

## Transportation Activities

**Spent Fuel Shipments.** In July 1979, NRC imposed new safeguards requirements on licensed spent fuel shipments. A recent, government-sponsored study* suggested that high-explosive sabotage of a spent fuel shipment, should it occur in an area of high population density, *might* produce serious radiological consequences. Although NRC does not have information indicating that a sabotage threat against spent fuel shipments exists in the United States, additional safeguards were imposed as prudent, interim measures, until a current research project confirms or refutes the potential for such consequences.

The interim rule requires that licensed spent fuel shipments avoid, where possible, highly populated urbanized areas, and that the licensee obtain NRC approval of the proposed route before such shipments are

* Ducharme, A.R., *et al.*, "Transport of Radionuclides in Urban Environs: Working Draft Assessment," SAND-77-1927, Sandia Laboratories, May 1978.

## Table 1. Original Field Evaluation Ratings, Before Corrections, at 11 Facilities

| | RATINGS | | | | | | |
|---|---|---|---|---|---|---|---|
| CRITERIA | Poor | Poor-Fair | Fair | Fair-Good | Good | Good-High | High [b] |
| EXTERNAL ASSAULT | | 1 [a] | | 4 | 2 | 2 | 2 |
| INSIDER | | | 1 | 1 | 2 | 1 | 6 |

[a] The NRC staff issued an immediately effective order increasing the level of safeguards.
[b] Safeguards evaluation goal.

## Table 2. Safeguards Inspections During FY 1979,[a] at Fuel Cycle Facilities

| *Strategic Fuel Facilities* | *Number of Safeguards Inspections* | *Number of Manhours of Inspection Onsite* | *Number of Items of Noncompliance* | *Number of Unannounced Inspections* |
|---|---|---|---|---|
| 1. Babcock & Wilcox, Apollo, Pa. | 12 (1/11)[b] | 390 (13/377)[b] | 0 | 83 |
| 2. Babcock & Wilcox, Leechburg, Pa. | 13 (3/10) | 787 (108/679) | 3 (0/3) | 77 |
| 3. Babcock & Wilcox, Lynchburg Research Center, Lynchburg, Va.[c] | 5 (0/5) | 126 (0/126) | 4 ( /4) | 100 |
| 4. Babcock & Wilcox, Naval Nuclear Fuels Division, Lynchburg, Va. | 12 (4/8) | 468 (102/366) | 6 (1/5) | 83 |
| 5. Exxon Nuclear, Richland, Wash.[c] | 5 (2/3) | 388 (30/358) | 3 (2/1) | 100 |
| 6. General Atomic, San Diego, Cal. | 7 (4/3) | 539 (174/365) | 6 (2/4) | 100 |
| 7. General Electric, Vallecitos, Cal. | 5 (3/2) | 308 (45/263) | 3 (3/0) | 100 |
| 8. Kerr McGee Nuclear, Crescent, Okla.[c] | 2 (1/1) | 65 (50/15) | 4 (4/0) | 100 |
| 9. Nuclear Fuel Services, Erwin Tenn. | 25 (6/19) | 1,644 (117/1,527) | 18 (2/16) | 80 |
| 10. Rockwell International, Canoga Park, Cal. | 5 (3/2) | 289 (51/238) | 4 (4/0) | 100 |
| 11. Texas Instruments, North Attleboro, Mass. | 10 (3/7) | 444 (126/318) | 2 (2/0) | 100 |
| 12. United Nuclear, Montville, Conn. | 14 (3/11) | 1,266 (155/1,111) | 5 (0/5) | 86 |
| 13. United Nuclear, Wood River Junction, R.I. | 13 (1/12) | 1,110 (72/1,038) | 12 (0/12) | 46 |
| 14. Westinghouse, Plutonium Fuel Development Laboratory, Cheswick, Pa. | 12 (4/8) | 394 (117/277) | 3 (0/3) | 75 |
| TOTAL | 140 (38/102) | 8,218 (1,160/7,058) | 73 (20/53) | 83 |

[a] Based on information on file as of November 5, 1979.
[b] For numbers in parentheses, the first number refers to physical security inspection activities; the second refers to material control and accounting inspection activities.
[c] These facilities are either not operating or not holding formula quantities in unirradiated form.

made. Data used to identify the applicable urbanized areas are from the 1979 *Supplemental Report* of the U.S. Bureau of the Census.* For NRC route-approval purposes, the following criteria define which populated areas are to be avoided:

A. Any "urbanized area" with more than one million persons.

B. Other "urbanized area" containing a city with more than 500,000 persons.

C. The boundary of any city, other than that included in A and B (excluding the rural part of an extended city), with more than 100,000 persons.

In addition, NRC's interim rule requires the following measures to be taken to guard against an attempt to sabotage a spent fuel shipment:

- Arrangements to call upon law enforcement agencies along the route for emergency assistance, if needed.
- Mobile communications equipment accompanying each shipment in transit.
- Status report, every two hours, from a shipment in transit to a central location.
- Capability to immobilize the transports, should a hijack be attempted.
- At least two individuals with each shipment, highly trained in emergency response procedures.
- No intermediate halts, where practicable, except for necessary refueling and food stops, and continuous occupation of the vehicle during necessary halts.
- Procedures for coping with threats and emergencies during transit, including arrangements for calling local law enforcement assistance.
- Armed escort, or local law enforcement agency

* U.S. Bureau of the Census, *Supplemental Report, 1970 Census of Population*, "Population and Land Area of Urbanized Areas for the US: 1970 and 1960," Series PC (S1)-108, April 1979.

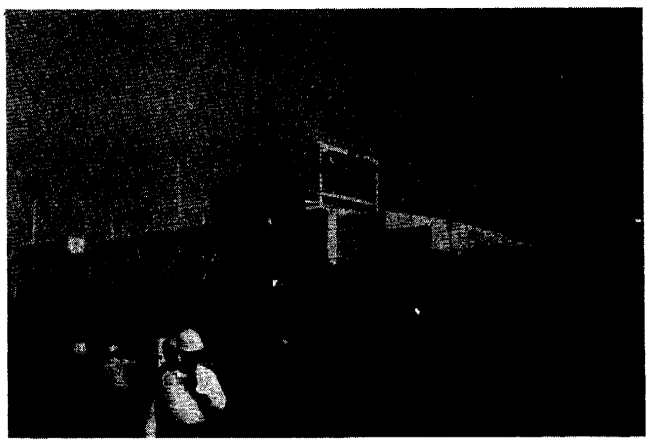
escort, if transport is necessary (and NRC-approved) through a highly-populated area.

**SSNM Shipments.** During 1979, NRC requirements for safeguarding shipments of formula quantities of highly enriched uranium or plutonium remained the same as those during 1978.

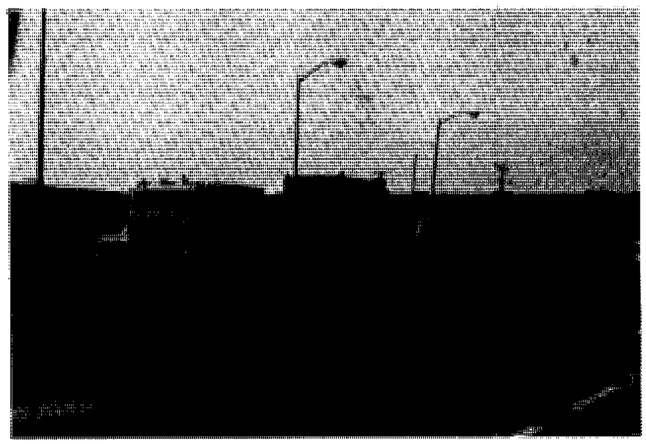
**Route Surveys.** NRC Safeguards teams conduct field surveys over transportation routes proposed for spent fuel or strategic special nuclear material shipments. These surveys obtain information for NRC contingency planning and route approval considerations. During such surveys, the teams coordinate with local law enforcement agencies along the way to increase their awareness and knowledge of the shipments and to identify local contacts that can be helpful, if needed.

During fiscal year 1979, the NRC teams surveyed five routes for shipments of formula quantities of special nuclear material and four routes for shipments of spent fuel. They collected data in the field, traveling approximately 6,400 road miles through 33 States, and meeting with approximately 270 local and State law enforcement agency representatives along the way. Licensees transporting nuclear materials received route profile reports which described appropriate law enforcement contacts and communications along the way.

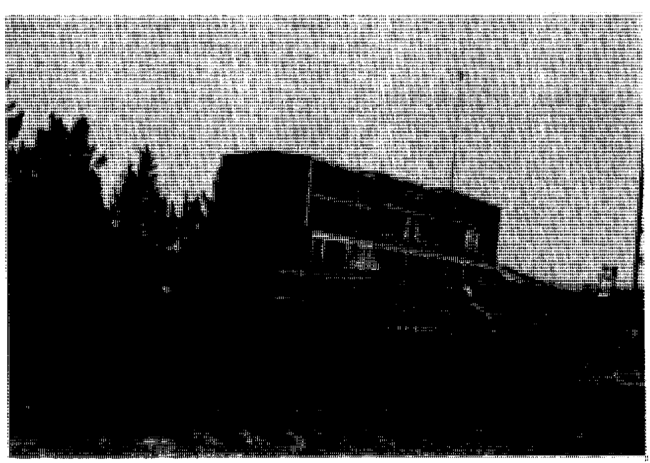
During the year, NRC determined the adequacy of safeguards by both the licensing process and the inspection of all licensed shipments involving special nuclear material. These inspections covered all domestic shipments and the domestic segments of the



A shipment of foreign irradiated fuel is transferred from ship to flatbed trailer for shipment to the DOE's Savannah River Plant near Barnwell, South Carolina. Under the Atoms for Peace agreement the U.S. sells nuclear fuel to foreign countries with the condition that spent fuel will be returned to the U.S. for storage.



A special steel cover is placed over the cask and bolted to the bed of the flatbed trailer. This reduces visibility of and accessibility to the cask while in transit to a railhead.



Rail cars can carry several truckloads of fuel elements. The picture shows several spent fuel casks with special covers now mounted on a rail flatbed car.

Regulations require the placement of buffer cars between any car carrying special nuclear material and other cars in the train both to provide safety space between the shipment and the other cars and to allow a 360° view of the shipment itself. This photo shows shipment as observed from the observation cupola of the caboose, looking forward towards engine. The shipment was observed from the rear by the conductor and from the engine by the train master and/or engineer throughout.

import and export shipments, including all storage and transfer points. Twenty-seven shipments were covered, involving 689 man-hours of inspection activity. No items of noncompliance were noted.

## Reactor Safeguards

The main factors in evaluating the adequacy of safeguards at power reactor facilities are:

- Licensing Reviews (including field reviews of equipment and operations) of licensee security plans prepared in response to the requirements to 10 CFR 73.55. ("Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors against Industrial Sabotage.")
- Inspection for licensee compliance with the approved security plan.

The NRC staff is also planning a future program to evaluate the practical effectiveness of safeguards as implemented at licensed operating power reactors throughout the U.S.

**NRC safeguards regulations require that all personnel entering critical areas of nuclear reactor facilities be thoroughly searched as a measure of assurance against possible installation sabotage. Here, NRC Resident Reactor Inspector Leif J. Norrholm is subjected to a pat-down search for weapons or possible sabotage devices at the Salem 1 Nuclear Generating Station in New Jersey.**

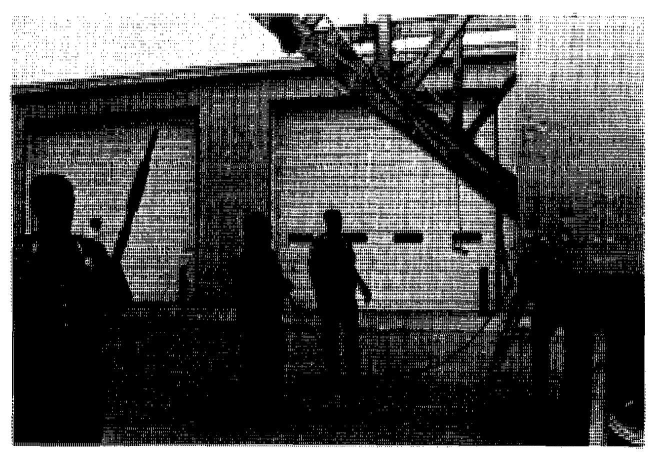
**Status of Safeguards at Power Reactors.** All operating power reactor licensees have put into effect approved physical security plans meeting the general and specific requirements of 10 CFR 73.55, with some aspects of measures employed against the inside threat being further refined. In this connection, the Commission has deferred the required implementation of certain defensive measures against potential sabotage by a licensee employee inside the facility pending further evaluation of the need and alternatives. These defensive measures include pat-down searches, "two-man rule" procedures in vital areas, and additional physical compartmentalization within such areas. (Two-man rule procedures are those where two employees observe each other's activities, in order to minimize the opportunity for malevolent acts by an insider.) The Commission also is considering various programs for determining trustworthiness of reactor facility employees authorized to enter vital areas.

There have been some construction and equipment delivery delays in the implementation of certain safeguards measures. Therefore, certain facilities are using approved interim measures, such as additional armed guards, until final systems components can be installed and their operation verified.

As of February 23, 1979, new security requirements for all power reactors went into effect. Since that time, 69 power reactors have been inspected to determine compliance with the new security requirements. With the exception of escalated enforcement actions, discussed in Chapter 7, the concerned licensees dealt promptly with all items of noncompliance.

**Status of Safeguards at Non-Power Reactors.** All licensed non-power reactors have operative security plans, required by 10 CFR 73.40, ("Physical Protection: General Requirements at Fixed Sites") for protection against sabotage. Specific safeguards measures for non-power reactors with less than formula quantities of SSNM are not defined in NRC regulations (although the new NRC regulations of July 1979 for these quantities of material apply to any facilities possessing such materials). However, those in effect at such reactor sites include:

- Identification of security areas.
- Security organization.
- Access controls.



**Shipments of special nuclear material are guarded during all phases of transit. Here, guards prepare to offload a truck shipment at a user facility. NRC monitors and inspects SNM shipments, surveys shipping routes, supervises guard and driver training, and, in cooperation with the Department of Transportation, develops standards for every aspect of nuclear material transportation.**

- Methods for detecting and reporting intrusions.

Also, non-power facilities with formula quantities of SSNM that are not self-protecting have activated anti-theft measures meeting the specific requirements of 10 CFR 73.50 ("Requirements for Physical Protection of Licensed Activities") and 10 CFR 73.60 ("Additional Requirements for the Physical Protection of Special Nuclear Material at Non-Power Reactors").

**Inspection and Enforcement at Reactors.** NRC inspection and enforcement activities at reactors also provided a measure for judging the effectiveness of safeguards. During fiscal year 1979, the NRC expended nearly 8,000 hours in on-site safeguards inspections at power reactors, and 2,300 hours at non-power reactors and research facilities. These inspections revealed 385 items of noncompliance with safeguards requirements (see Table 3). NRC has issued two Immediate Action Letters that identify measures which licensees must take to improve their safeguards systems. Three civil penalties, totaling more than $25,000, were issued against three licensees.

## Table 3. Reactor Safeguards Inspections During FY 1979 [a]

| *Facility* | *Number of Safeguards Inspections* | *Manhours of Onsite Inspection* | *Number of Items of Noncompliance* | *Percent of Unannounced Inspections* |
|---|---|---|---|---|
| Power Reactors | 190 (163/27)[b] | 7,778 (7,010/768)[b] | 340 (330/10)[b] | 96 |
| Non-power Reactors | 55 (41/14) | 926 (743/183) | 19 (17/2) | 96 |
| Research & Specialty Reactors | 53 (5/48) | 1,399 (152/1,247) | 26 (9/17) | 76 |

[a] Based on information on file as of November 5, 1979.
[b] For numbers in parentheses, the first number refers to physical security inspection activities, the second refers to material control and accounting inspection activities.

### Contingency Planning

During fiscal year 1979, the NRC staff reviewed and approved safeguards contingency plans developed by each of the 19 fuel cycle facility licensees authorized to have formula quantities of strategic nuclear material. During this time, the staff also reviewed 28 of 61 safeguards contingency plans developed by power reactor licensees and applicants. Each of the power reactor contingency plans evaluated required certain modifications or additions. None was given final approval during fiscal year 1979.

At the national level, NRC concluded a memorandum of understanding with the Federal Bureau of Investigation which formalized procedures for information exchange and coordinated response actions between the two agencies. NRC is now coordinating three similar interagency agreements with the Federal Aviation Administration, the National Security Agency, and the Bureau of Alcohol, Tobacco and Firearms.

## SAFEGUARDS EVENTS—FISCAL YEAR 1979

In December 1978, a licensee shipped four drums containing 4.5 kilograms of highly enriched uranium to Romania, with each drum sealed by the NRC. Upon arrival at the port of embarkation in New York, all four seals were found broken. The NRC inspector examining the drums and seals decided that the contents of the drum had not been disturbed. Consequently, he resealed the containers—without inspecting the contents—and permitted the export of the shipment. When the containers arrived in Romania, the seals that were affixed in New York were found to be intact. The IAEA, upon request, examined the shipment upon arrival, and confirmed that the nuclear material contents were as shipped by the licensee. As a result of this event, NRC has changed its procedures for accounting for the seals that allow inspectors at the destination to determine whether anyone has replaced or tampered with them.

In January 1979, the General Electric Company reported to the NRC that an alleged theft of low-enriched uranium oxide, and an attempted extortion had occurred at its fuel plant at Wilmington, North Carolina. An individual was arrested on criminal charges, and the material was recovered. The material was of no safeguards significance because of the amount and enrichment involved. It represented a minimal health hazard, being less hazardous than many industrial chemicals. As a result of the incident, the licensee reevaluated its security and accountability system for such materials. For some time, the NRC has been reexamining requirements for the protection of nuclear materials of this type. As a result, new rules are being proposed to prevent future incidents of this type from occurring. (See Chapter 7 for further details on this incident.)

In May 1979, 62 of 64 new fuel assemblies at the Surry Nuclear Power Station in Virginia were found to have been coated with sodium hydroxide, in an apparent attempt at sabotage. Subsequent investigations revealed that two licensee employees were responsible for the act. They were later convicted and sentenced to prison. (See Chapter 7 for further details on this incident.) The NRC has subsequently taken steps to modify its regulations to tighten access controls at such facilities. Requirements being proposed are contained in an Inspection and Enforcement Bulletin, IE Bulletin No. 7416, "Vital Area Access Controls."

As a result of a physical inventory taken in August 1979, Nuclear Fuel Services (NFS) reported that the inventory difference at the NFS plant at Erwin, Tennessee, was in excess of the upper limit specified in the license. (An inventory difference occurs when the total SNM listed in the account books from previous measurements does not agree with the total SNM measured in the most recent physical inventory.) The licensee was ordered to shut down the facility and begin an immediate investigation and re-inventory. NRC sent an investigation team to the site to observe the inventory and verify measurements being made. While the re-inventory results partially explained the inventory difference, the discrepancy was not reduced to a level normally expected as a result of uncertainties in the measurement of nuclear materials. At the end of fiscal year 1979, the facility remained shut down pending resolution of the problem.* The NRC investigation, which included FBI and Department of Energy participation, had not discovered any information (other than the presence of the inventory difference which is of itself indeterminate because of measurement uncertainties) to indicate that a theft of material had occurred. However, the investigation had not been able to rule out that possibility.

## SAFEGUARDS REGULATORY ACTIVITIES

Fiscal year 1979 was an active period in the development and adoption of NRC regulations designed to improve nuclear safeguards. Attempting to solve several major safeguards problems also constituted an important part of NRC's activities. Solutions were not always found and efforts to solve some problems must continue into 1980 and beyond, if necessary.

**Physical Security.** In November 1979, the Commission published the final version of a new regulation designed to upgrade the physical protection of formula quantities of special nuclear material. Any facility holding or transporting five formula kilograms of such material is subject to the new rules. The only temporary exception is non-power reactors, for the reasons indicated below. The regulation will become effective in March 1980. Licensees are expected to implement it by the fall of 1981. The rule indicates performance standards to be met by licensees and presents specific statements about the kinds of threats, from insiders and outsiders, that their safeguards should be able to withstand.

Most non-power reactors are research reactors operated by universities. Applying the new rules must take into account the unique characteristics of a university and its reactor. The universities are concerned that applying the strict new physical security requirements of the regulation will force them to shut down their reactors. They cite unacceptable costs and impact on their education programs. They also cite specific design and fuel features of their reactors. Some of these views appear to have merit, and the NRC is considering whether a balance can be struck between the specific new requirements and alternative ways to achieve the necessary safeguards performance. This problem was not resolved in 1979, and continues to be studied.

Another important problem to be resolved in 1980 relates to the regulation upgrading power reactor physical security safeguards. When first issued in 1977, the rule called for conducting either a physical or an instrument search for the detection of prohibited material. The Commission was petitioned to eliminate the possible interpretation of "physical search" as requiring a "pat-down" search. The issue is under study, and the staff is considering alternatives for the most effective search techniques to be employed. In the interim, instrument searches are routinely employed, supplemented by physical searches when circumstances dictate.

In July 1979, final regulations were published to provide new physical protection requirements for special nuclear materials in less than five formula kilogram quantities—materials of moderate- and low-strategic significance (Category II and Category III materials). These regulations make the U.S. rules consistent with International Atomic Energy- (IAEA) worldwide standards.

Another new rule issued in fiscal year 1979 placed carriers of five or more formula kilograms of special nuclear material under NRC regulation. Formerly, the shippers and receivers, but not the actual

* On January 21, 1980, the facility was permitted to return to production operations after implementation of significant improvements in the physical security, internal control, and material accounting systems.

transporters of material, had been under NRC license. A general license has been issued to the carriers, making them directly subject to NRC requirements and inspections.

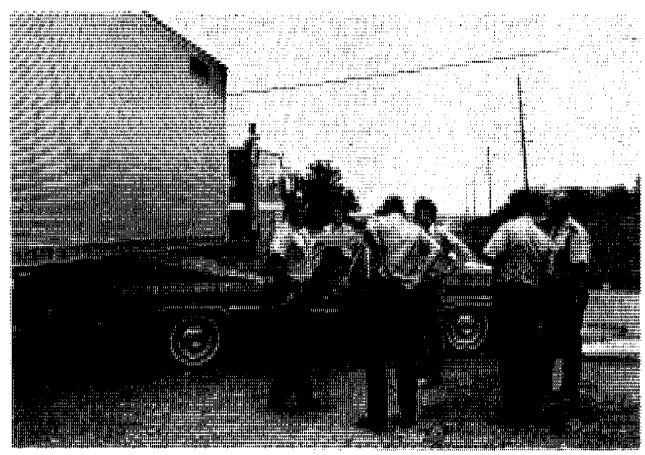
Requirements of the new interim rule for spent fuel shipments have been discussed in detail earlier in this chapter. The staff must now reassess the need for changes based on public comments, and as may be required by the result of recent research. Among the issues being addressed are routing restrictions, preemption of local ordinances, call-in procedures, and handling of vessels carrying spent fuel.

Transient shipments of formula quantities of special nuclear material are also a matter of concern. A transient shipment is one that temporarily uses U.S. facilities while moving from one foreign country to another. NRC is preparing a regulation that would require protection of such shipments, which usually are carried by an aircraft transiting a U.S. airport. Enforcing such a regulation may pose problems, because it would require that NRC obtain advance details about the shipment, sometimes from sources outside NRC's immediate control.

Transient shipments of spent fuel are a matter of more recent concern. The staff has taken note of growing public concern about the need to provide safeguards protection for such shipments. This concern was highlighted when Representative Heftel and the Governor of Hawaii urgently requested that the NRC adopt regulations to protect transient shipments of spent fuel. These requests came as a result of an unscheduled refueling stop, at Honolulu, by a freighter carrying spent fuel. The NRC plans to analyze the alternatives involved in providing such protection and possible regulatory changes to implement such protection.

Along with its concern about shipments of spent fuel, the NRC also is trying to estimate the potential hazards of sabotage (or theft, if that should occur) at high level nuclear waste storage sites. Conceivably, the radioactive dispersal hazards might be similar to those resulting from sabotage of spent fuel. The staff is also analyzing the alternatives involved in transporting Three Mile Island wastes to disposal sites. When the results of these analyses are known, the staff will address the issue of what safeguards measures, if any, should be required for nuclear waste activities.

The Kemeny Commission's report on the Three Mile Island accident indicated that human attitudes and practices were a principal contributory factor to the accident. In the area of nuclear security, people—particularly security managers and guards—play a vital role in ensuring that safeguards systems achieve their intended purpose at all licensed facilities. In response to these concerns, the NRC staff has begun to review how well people perform in the safeguards area. The staff will make recommendations to improve such performance, if appropriate.



The NRC role in safeguards training includes setting training requirements for transport security personnel, ensuring appropriate cooperation and communications between shipment security guards and local law enforcement agencies, and monitoring such activities through inspections and reports. Shown here are two examples of training inspected by NRC in 1979. The photos show [from top] a formal security-officer weapons training class, and scenes from a staged accident involving a "nuclear shipment" in which accident response measures were practiced.

**Material Control and Accounting.** While not always immediate or inexpensive, solutions to security problems at NRC licensed facilities seem to be more obvious and straightforward than solutions to accountability problems. An example of the intractability of these problems was provided during fiscal year 1979 by the occurrence of a large inventory difference at the NFS fuel plant at Erwin, Tenn., as described earlier in this chapter. (An inventory difference occurs when the total SNM listed in the account books from previous measurements does not agree with the total SNM measured in the most recent physical inventory.) Another example of material control and accounting problems is provided by NUREG-0627, an NRC report on the Nuclear Material and Equipment Corporation (NUMEC) situation in the mid-1960's. (The report was requested by Congressman Udall and prepared by NRC staff during fiscal year 1979.)

For years NRC (and previously the AEC) has used inventory differences to signal accountability problems or out-of-control processing situations. However, inventory differences are based on periodic plant-wide inventories and are not a timely indicator of a loss of material. Moreover, the causes of unusual or excessive inventory differences are not always clear, even after extensive investigation. As a result, the NRC staff is examining several alternatives to relying on inventory difference as a primary indicator of accounting problems.

Currently, the staff is formulating a major rule aimed at improving the level of safeguards assurance provided by material control and accounting systems. The goals are to provide more timely material control and accounting indicators which can be resolved more clearly, and which will better locate the accountability problems within a licensee's plant. A draft rule is planned by the end of fiscal year 1980.

**NRC/IAEA Interaction.** In 1979, NRC staff accelerated preparations for the application of IAEA safeguards to U.S. nuclear facilities other than those having direct national security significance. The US/IAEA Safeguards Agreement has been approved by the IAEA. The President has submitted it to the Senate for ratification as a treaty. The Agreement will enter into force when the U.S. notifies the IAEA that its constitutional and statutory requirements have been met. (See Chapter 9 for detailed discussion of international safeguards.)

## SAFEGUARDS RESEARCH AND TECHNICAL ASSISTANCE

The NRC safeguards contractual program includes both research (long-term, comprehensive effort) and technical assistance (short-term efforts supporting operational assignments). In fiscal year 1979, about $12 million was spent on safeguards research and technical assistance. Approximately $5 million of the total was spent on research projects and the remaining $7 million on technical assistance projects. The Commission reviewed and approved all safeguards contracts exceeding $20,000 in funding, as required by the Congress.

The safeguards research program has now developed to the point where the products are being tested in applications to assist (a) the formulation of regulations; (b) the determination of safeguards adequacy; and (c) the assessment of the effectiveness of licensee safeguards systems.

Fiscal year 1979 research projects that have helped or are expected to improve safeguards programs are the following:

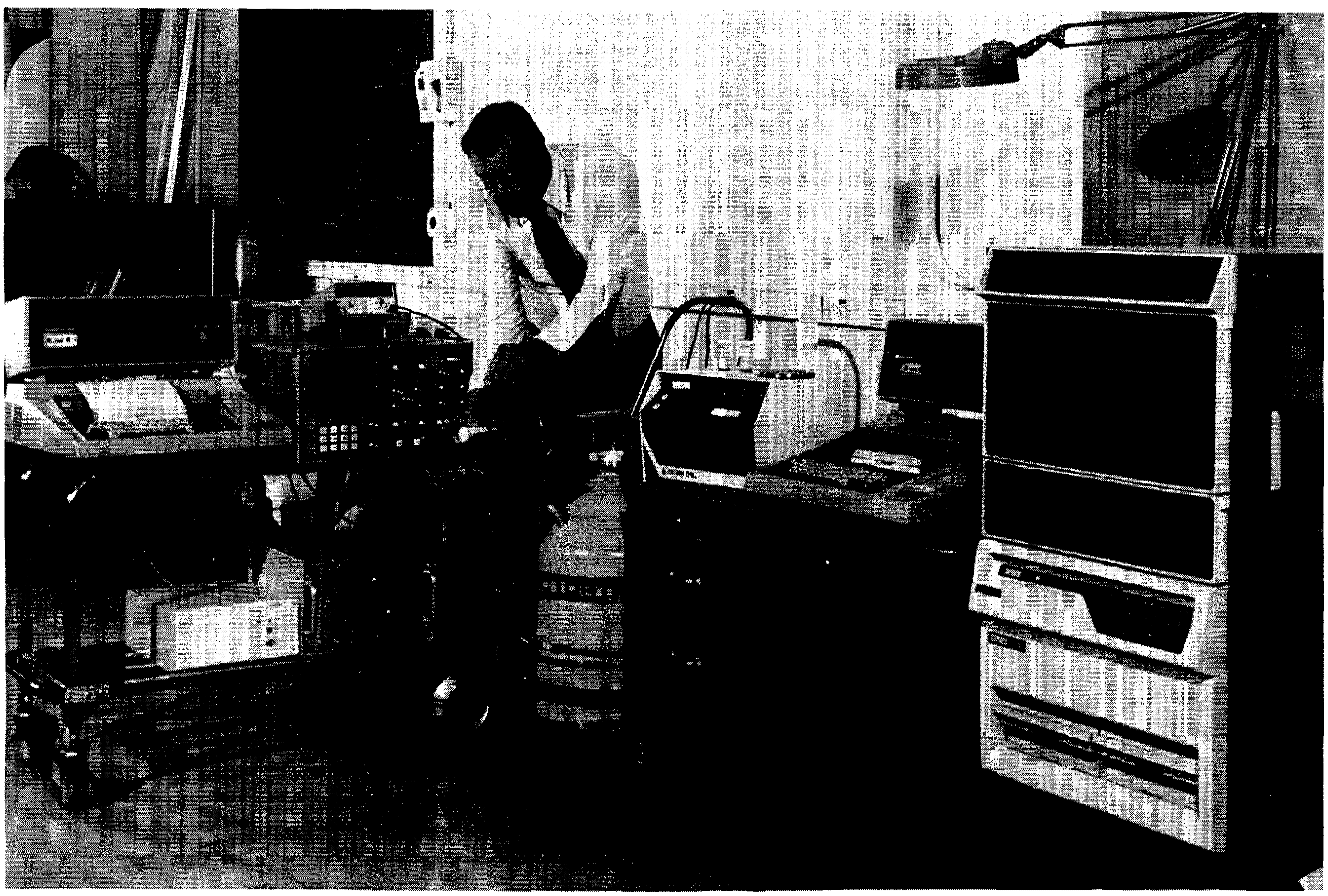
- *"Effectiveness Evaluation Methods for Physical Protection of SNM in Transit" Project.* Results from this program have helped evaluate guard levels required for an in-transit physical protection system established in the physical protection upgrade rule.
- *The "Effectiveness Evaluation Methods for Material Control and Accounting" research project,* which has provided technical inputs to development of the proposed material control and accounting upgrade rule, and associated guidance.
- *The "Insider Crime Analogous to the Potential Threat to Nuclear Programs" study,* which is expected to aid in the formulation of prudent standards and regulations related to the potential insider threat.
- The *"Spent Fuel Cask Vulnerability Program,"* results of which will help formulate policy on safeguarding shipments of spent fuel from light-water reactors. They will help confirm and/or modify, as necessary, safeguards regulations protecting spent fuel shipments.
- *The "Nuclear Power Plant Design Concepts for Sabotage Protection" project.* Design alternatives and damage control measures for nuclear power plants were studied in order to improve their inherent protection against sabotage. A recent typical plant design* was selected and characterized to provide a baseline against which the effectiveness and impact of proposed changes will be measured. A set of potentially useful design alterations, as well as methods to mitigate damage, were identified. These features have been reviewed and the most promising alter-

*The Standardized Nuclear Unit Power Plant System is a system identified by Bechtel Corporation. It takes advantage of standardized engineering and installation practices for the purpose of simplifying licensing and acceptance reviews.

natives will be selected for more comprehensive evaluation.

Fiscal year 1979 research projects directed toward improving safeguards adequacy and effectiveness include:

- *The "Effectiveness Evaluation Models for Fixed Site Physical Protection" project.* The method developed in this project identified vital areas within reactor facilities, using computerized, generic fault-tree techniques. It was applied to more than 27 pressurized water reactor and boiling water reactor sites, and has proved an effective aid in licensing and evaluating power reactors. As a result of last year's feasibility test of the Safeguards Automated Facility Evaluation method, some Safeguards Automated Facility Evaluation modules were further developed and applied, on a trial basis, to several reactor facilities. This method will be used extensively in fiscal year 1980, with the vital area analysis already discussed, to help evaluate the adequacy of safeguards at power reactor facilities. The Safeguards Network Analysis Procedure is an evaluation tool to help in physical security field evaluations of operating facilities and to evaluate proposed licensee safeguards. This procedure may also be useful for formulating insights into guard tactics and strategies planned as responses to possible terrorist attack.
- *The "Effectiveness Evaluation Methods for Material Control and Accounting" project.* The structured assessment analysis method was substantially developed in fiscal year 1979 for use in assessing the effectiveness of material control and accounting safeguards at fuel-cycle facilities. The structured assessment analysis assists in analyzing the vulnerability of a facility to both insider and



The improvement of safeguards techniques often means combining existing methods or equipment toward greater precision or speed in obtaining results. Dr. Warren McGonnagle of NRC's Region III Office of Chicago is shown here observing measurements during a non-destructive enrichment assay of nuclear fuel material. The material, physically located in a detector (under McGonnagle's hand), is automatically evaluated in the multi-channel analyzer on his right. Calculations then can either be printed by the attached recorder (left of the analyzer) or displayed on the computer at the right of the photo. The canister beneath the detector feeds nitrogen into the device to cool the test sample. Data from these independent NRC assays are compared with information supplied by commercial firms for verification of enrichment percentages or other properties.

outsider adversaries with authorized and unauthorized access and extensive capabilities. The method is scheduled for testing in fiscal year 1980.

- *The "Communicated Threat Credibility Project."* This project provides multidisciplinary tools for investigating the credibility of communicated threats and for providing advice to the Department of Energy, the NRC, the Federal Bureau of Investigation, and other appropriate agencies during an actual or perceived emergency from nuclear extortion threats.

Technical assistance projects were conducted by the major program offices to support their operational missions. These projects ranged from helping establish a technical basis for determining safeguards requirements for byproduct materials, to providing assistance in developing NRC's physical security upgrade rule. Technical assistance projects of greater than $20,000 were approved by the Commission, as required by Congress.

## SAFEGUARDS MANAGEMENT

Under the NRC's new lead office management concept, the Office of Nuclear Material Safety and Safeguards (NMSS) is the lead office for safeguards and is responsible for integrating and coordinating the overall NRC safeguards program.

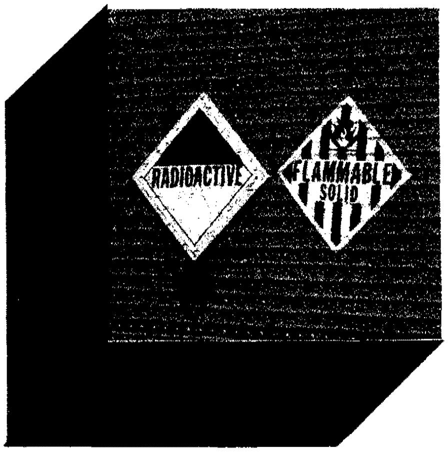
**Safeguards Consolidation.** In August 1979, NRC decided to transfer all reactor safeguards functions formerly in the Office of Nuclear Reactor Regulation to NMSS, which also has primary safeguards regulatory responsibility for fuel cycle facilities and transportation activities. The consolidation was effective on October 1, 1979. Staff members in NMSS coordinate their reactor safeguards licensing activities with the Office of Nuclear Reactor Regulation. This coordination will involve administrative maintenance of the safeguards portion of reactor licenses. It will also involve areas of concern common to both reactor security requirements and reactor safety matters.

Reactor safeguards functions transferred to NMSS include the following:

- Safeguards licensing reviews for power and non-power reactors.
- Generic physical security policy and guidance development for reactors.
- Work on the potential implementation of the United States/International Atomic Energy Agency Safeguards Agreement.
- Administration of reactor safeguards technical assistance contracts.

The Office of Inspection and Enforcement will continue safeguards inspection functions. The Office of Standards Development will continue to develop safeguards standards. The Office of Nuclear Regulatory Research will continue both its present and future safeguards research function.

**Integrated Program Plan and Safeguards Technical Assistance and Research Coordinating Group.** Each of the major program offices participates in the planning and implementing of NRC's domestic safeguards program. The Safeguards Technical Assistance and Research Coordinating Group provides inter-office coordination of NRC-contracted safeguards activities. An Integrated Safeguards Program Plan was developed and sent to the Commission in January 1979. This program plan, which will help coordinate safeguards activities in NRC, will be updated in fiscal year 1980 after the Commission issues its policy, planning, and programming guidance. NRC's Budget Review Group, the Executive Director for Operations, and the Commission also review and approve the entire NRC safeguards program during the annual NRC budget review process.




# 6 Waste Management

Waste shipments are clearly identified by NRC-approved placards.

The NRC waste management function was elevated to divisional status in 1979 under the NRC Office of Nuclear Material Safety and Safeguards (NMSS). The new Division of Waste Management consists of five branches which carry out a number of functions that were formerly among those of the Division of Fuel Cycle and Material Safety:

- *The High-Level Waste Technical Development Branch*—responsible for high-level waste regulatory development and development of the technical bases for high-level waste licensing and regulation.
- *The High Level Waste Licensing Management Branch*—responsible for licensing high-level waste commercial repositories.
- *The Low-Level Waste Licensing Branch*—responsible for low-level waste licensing and regulation.
- *The Uranium Recovery Licensing Branch*—responsible for licensing and regulation of uranium mills, heap-leach operations, commercial scale solution mining operations, and research and development (R&D) uranium recovery operations. These types of operations represent the first step of the nuclear fuel cycle. Since large amounts of waste are generated as a result of these operations, especially uranium milling, it was decided that these operations should come under the Division of Waste Management.
- *The Licensing Process and Integration Branch*—responsible for coordinating and integrating the entire NRC waste management program. In order to do this, the branch works with elements within NMSS, with other NRC offices, and with other governmental agencies having waste management responsibilities, to ensure that the entire NRC program is well focused and proceeding on established schedules.

## Overview of 1979 Activity

The main focus of NRC effort in 1979 for the high-level waste program was in regulatory development. The NRC is developing a comprehensive regulation for high-level waste repositories—to be Part 60 of the NRC regulatory code—in two parts, procedural and technical. The procedural part was published as a proposed rule for comment in December 1979. The technical part is expected to be published in early 1980 pursuant to an advance notice of proposed rulemaking.

The main focus of NRC work in 1979 for the low-level waste program has also been in regulatory development. The NRC is developing a comprehensive regulation for low-level waste disposal. This regulation will be Part 61 of the code. A preliminary draft of the regulation has been completed and sent to various organizations for review. The draft will be made available to the public in 1980.

A large part of NRC effort under the uranium recovery program has been concerned with the licensing of uranium recovery facilities, and a significant number of licenses were issued, renewed, and amended. In addition, a draft regulation for uranium mills (Amendment to Part 40) was issued for public comment in August 1979. The supporting generic environmental impact statement (GEIS) on uranium milling was issued for public comment in April 1979.

A number of notable events in nuclear waste disposal took place in 1979. There were only three low-level waste disposal sites in operation at the beginning of the year, all of them located in Agreement States. Two of the sites closed and then reopened, and a curtailment was placed on the amount of waste that could be received at the third site. These actions further demonstrated the large regional imbalance in low-level waste disposal locations and induced a

On July 16, 1979, a tailings dam near Grants, N.M., gave way, releasing nearly 100 million gallons of radioactive water and sediment into the Rio Puerco. Flow from the break reached into Arizona, some 75 miles down river. The break occurred as efforts were being made to reinforce the dam, and heavy equipment on site for that purpose enabled workers to stop the flow in a few hours.

number of States to seriously consider the desirability of regional burial sites. Also, on July 16, 1979, a tailings impoundment failure occurred at the United Nuclear Corporation uranium milling operation at Church Rock, N.M. (New Mexico is also an Agreement State.) A major effort was undertaken by the NRC to assist the State in correcting the situation. (See discussion under "Technical Assistance to Agreement States," later in this chapter.)

It is important to note three studies which have affected and will affect the course of the NRC waste management program. These studies are the Interagency Review Group (IRG) Study on Nuclear Waste Management, the Congressionally requested NRC Study on Regulation of Federal Radioactive Waste Activities; and the Congressionally requested NRC study on Means for Improving State Participation in the Siting, Licensing and Development of Federal Nuclear Waste Facilities. Also of potential importance to the NRC waste management program is the "confidence hearing" on radioactive waste disposal to be held by the Commission in 1980.

**Interagency Review Group.** As reported in the 1978 NRC Annual Report (pp. 93 and 94), the NRC staff participated in the IRG study on Nuclear Waste Management. (Because of NRC's status as an independent regulatory agency, the agency participated as a non-voting member.) The IRG draft report was issued in 1978 and the final report was issued in 1979. Many of its recommendations affect the NRC, which has reviewed the impact of these recommendations on its program in 1979 and will continue to do so in 1980.

**Federal Radioactive Waste Study.** The NRC's Authorization Act for fiscal year 1979 (P.L. 95-601) required the NRC to prepare a study on the regulation of Federal radioactive waste activities. The study was completed in 1979 and it was issued as NUREG- 0527, entitled "Regulation of Federal Radioactive Waste Activities." Two principal recommendations came out of the study. The first was that NRC licensing authority should be extended to cover all new Department of Energy (DOE) facilities for disposal of transuranic waste and non-defense low-level waste. This recommendation was consistent with one of the IRG recommendations. The second was that a pilot program should be established to test the feasibility of extending NRC regulatory authority on a consultative basis to DOE waste management activities not now covered by NRC's licensing authority, or to the new facilities cited in the first recommendation. The pilot program would focus on a few specific DOE waste management activities and would result in a report to Congress on the feasibility of an NRC consultative role in existing DOE waste disposal and storage activities. The decision on whether to extend NRC regulatory authority and to establish the pilot program and on what waste management activities the program should include was considered one for the Congress to make. If the Congress decides that the NRC should implement these recommendations, it will significantly affect NRC's current and future waste management programs. The exact impact cannot be assessed until specific legislation is proposed and implemented.

**Improving State Participation.** The NRC's Authorization Bill for fiscal year 1979 (P.L. 95-601) also required the NRC to prepare a study on means for improving the opportunities for State participation in the process for siting, licensing, and developing nuclear waste storage or disposal facilities. The study was completed in 1979 and it was issued as NUREG 0539, entitled "Means for Improving State Participation in the Siting, Licensing, and Development of Federal Nuclear Waste Facilities."

There were a number of recommendations as a result of the study. The Commission recommended the establishment of a planning council consisting of Federal and State representatives, to be supported by a small administrative staff and Federally financed. A review capability should be established under the direction of the planning council in order to enable the States to make technical evaluations of waste management technology and Federal waste management activities. The review capability should also be Federally funded. These recommendations were consistent with the IRG recommendations. In addition, the Commission recommended that measures be taken to involve the States throughout the process for planning, siting, developing, and licensing nuclear waste storage disposal facilities. It is also recommended that the Congress establish a grant program to allow the States to participate more fully in the Federal Waste Management program. Federal agencies should consider such transportation related issues as shipping routes, emergency planning, enroute liability, shipping containers, and the like, in their overall waste management activities and should develop institutional arrangements as appropriate for consulting with

**NRC continued to study ways to improve Federal/State cooperation in waste storage matters in 1979, as visits to Agreement-State activities were stepped up. Representatives of several NRC program offices are shown here during a briefing on low-level waste storage monitoring techniques by officials of the Barnwell, S.C., storage site and South Carolina State offices.**

the States in a timely manner. Lastly, the Commission recommended that legislation for improving State participation in the Federal Waste Management Program should provide recognition of the legitimate concerns of host States; considerations affecting a State concurrence or veto, if authorized by law, were identified.

If the Congress elects for the NRC and other Federal agencies to implement any or all of these recommendations, these actions will affect NRC's current and future waste management programs. The exact impact cannot be assessed until specific legislation is proposed and implemented.

**Confidence Hearing.** The NRC decided in 1979 to conduct a generic proceeding to reassess the Commission's degree of confidence that radioactive wastes produced by nuclear facilities will be safely disposed of, and to determine when any such disposal will be available, and whether such wastes can be safely stored until they are disposed of. Notice of the proceeding appeared in the Federal Register in October 1979, and the hearing will take place in 1980 and 1981. The proceeding has been initiated in response to the decision of the U.S. Court of Appeals for the District of Columbia Circuit in *State of Minnesota v. NRC*, 602 F.2d 412,but is also a continuation of proceedings previously conducted by the Commission in this area. The notice described the procedures the Commission will employ and how members of the public can participate. The results of the hearing and any rules issuing therefrom may have an effect on NRC's current and future waste management program. (See also "Commission Decisions," in Chapter 13.)

The three sections which follow describe the 1979 accomplishments of the NRC waste management programs dealing with high-level waste, low-level waste, and uranium recovery. Each section discusses near-term objectives of the program and activity during the report period in regulatory development, licensing, and associated matters.

## HIGH-LEVEL WASTE MANAGEMENT

### Regulatory Development

NRC continued its high-level waste regulatory development effort in 1979 with the objective of developing and publishing a draft regulation (10 CFR Part 60) and supporting environmental impact statement (EIS). The regulation as currently envisioned will be published in two parts: the procedural requirements and the technical requirements. The procedural portion would contain sub-parts covering general provisions, licenses, and participation by State governments. The technical portion would contain sub-parts covering performance objectives and technical criteria, physical protection, quality assurance, and emergency plans. Particular emphasis is being placed on waste form performance requirements and geologic site characterization issues. In December 1979, the procedural portion of the regulation was published as a proposed rule for public comment. The technical portion of the rule is expected to be published in early 1980 pursuant to an advanced notice of proposed rulemaking. Work is also continuing on a supporting environmental impact statement which would be published with the proposed technical rule in 1980.

Work began in 1979 in developing regulatory guides to support the regulation. These include format and content guides for the safety analysis report, the environmental report, and reports detailing DOE plans for site characterization work. These guides will be published for public comment in 1980.

Additional regulatory guidance will be provided to DOE in the form of technical directives. The technical directives that were under development in 1979 and which will be issued in 1980 will cover generic topics addressing site selection and characterization, repository design, and waste form. Work also continued in 1979 on identifying research needs.

In 1979, work was begun on outlining license review procedures both to aid the staff in establishing priorities for research and regulatory guides and to provide DOE with guidance on how NRC will conduct its review.

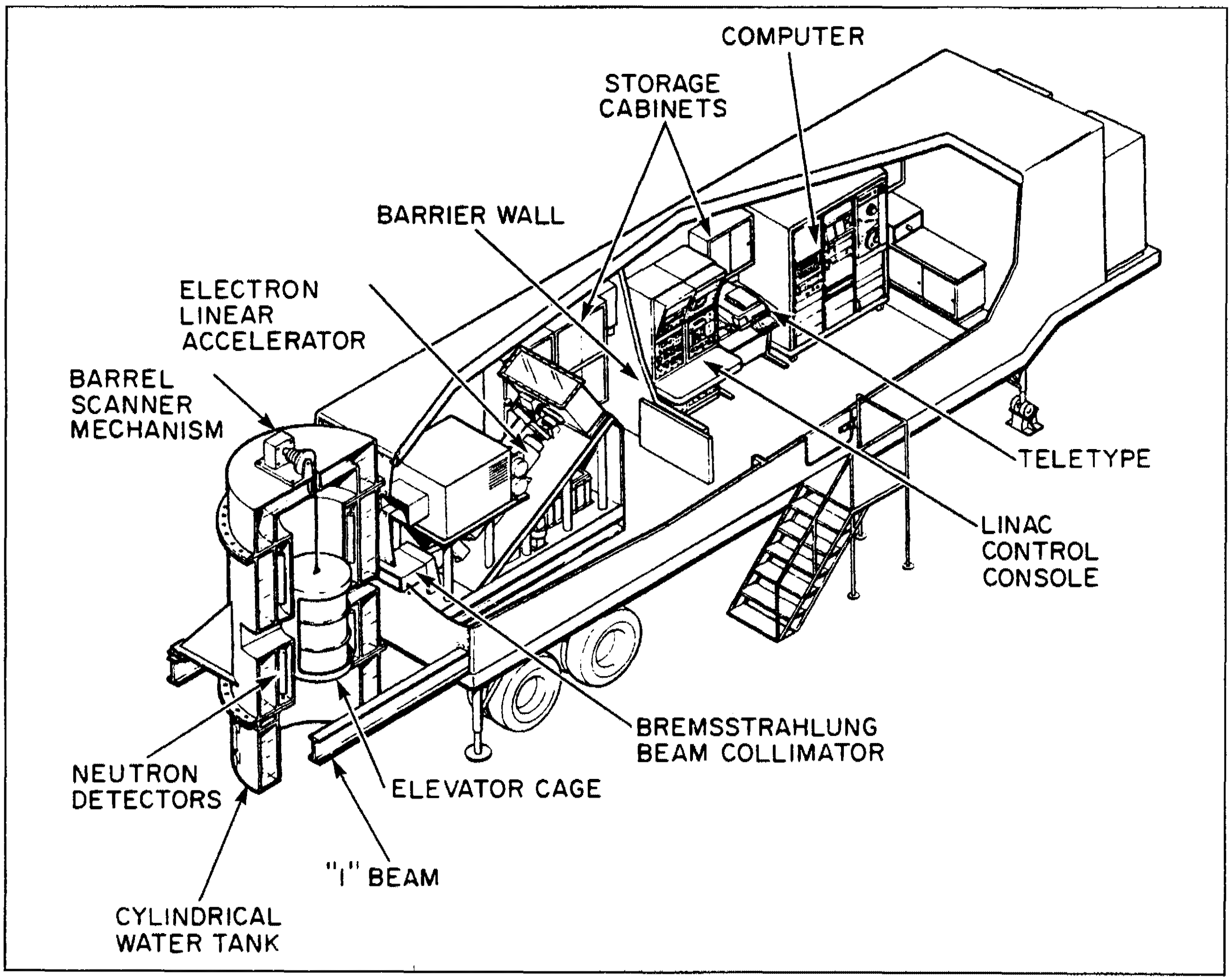
## Licensing

NRC continued its efforts in 1979 to develop a capability to review a license application for a high-level waste repository. The development of models for assessing radionuclide transport in bedded salt was continued and is expected to be completed in 1980. A model for assessing the safety and environmental risks of a repository after sealing was delivered to the NRC by the contractor so that NRC could test and evaluate the model.

## Assessment of DOE High-Level Waste Management Program

The NRC has continued its assessment of the DOE high-level waste management program in 1979. The NRC reviewed and provided comments to DOE on the draft EIS for the proposed Waste Isolation Pilot Plant (WIPP) and the draft GEIS on the management of commercially generated radioactive waste.

The NRC initiated in 1979 a program to critically assess the DOE high level waste management pro-




**One of NRC's continuing objectives is the improvement of nuclear material inventory and accounting techniques. This cutaway drawing shows the main features of a mobile measurement system used to identify, measure and record uranium and plutonium contents of metal waste drums. The barrel scanner at rear (left) of the mobil unit remotely places, lifts and "reads" the container, and transmits readings through analytical devices to the recording instruments in the manned area of the trailer.**

gram. DOE and its contractors have made formal presentations to NRC on various phases of the DOE program. On November 15-16, a meeting was held with the Office of Nuclear Waste Isolation and other DOE contractors at Columbus, Ohio, to formally inaugurate NRC's assessment program. Arrangements are being made to maintain an overview of all DOE activities in high-level waste management by systematically receiving and reviewing all documents generated by the DOE program. Task groups have been established to perform an initial, limited assessment of DOE activities in waste packaging, repository siting, and repository design. Comparisons will be made between needs identified in NRC's draft regulation and information expected to be generated by DOE programs. Finally, plans have been prepared for conducting a comprehensive critical assessment of the DOE repository siting and in-situ testing programs.

## LOW-LEVEL WASTE MANAGEMENT

### Regulatory Development

NRC continued its low-level waste regulatory development effort in 1979 with the objective of publishing a draft regulation (10 CFR 61) on low-level waste disposal. An environmental impact statement (EIS) will be prepared to support the rulemaking action. Work was also continued on supporting regulatory guides and staff positions.

The draft regulation as currently envisioned will consist of basic performance objectives applicable to the disposal of low-level waste on land by various methods. These objectives will be met by establishing appropriate requirements for siting a disposal facility and assuring adequate operations site closure and decommissioning and adequate institutional arrangements. Technical details specific to the individual disposal techniques of shallow-land burial and other alternative disposal methods will be contained in appendices to the regulation and in regulatory guides. A preliminary draft of the regulation was completed in 1979 and made available to a wide cross section of persons for informal review. The draft will be made available to the public in 1980.

The regulatory guides associated with the regulation are also under development and are currently envisioned to cover waste form and content; site design and operations; site monitoring and surveillance; site closure, stabilization, and post-operational care; standard contents for license application and environmental report; records and reports; and funding.

In addition to the above work, NRC has contracts with various organizations to develop a base of supporting technical information. Contractual studies are underway in such areas as systems analysis, waste classification, and volume reduction. The systems analysis contractor is developing models for analyzing radioactive waste disposal by shallow-land burial. The waste classification contractor is characterizing wastes, waste forms, and waste sources in addition to recommending requirements for safely disposing the waste. The volume reduction contractor is investigating various volume reduction techniques including compaction and incineration. The contractor is also performing economic analysis for the various techniques.

Other contractual efforts are planned to develop specific technical criteria for disposal of wastes in mined cavities and engineered structures, and to investigate in detail requirements for disposal of waste generated as a result of decontamination and decommissioning of nuclear facilities.

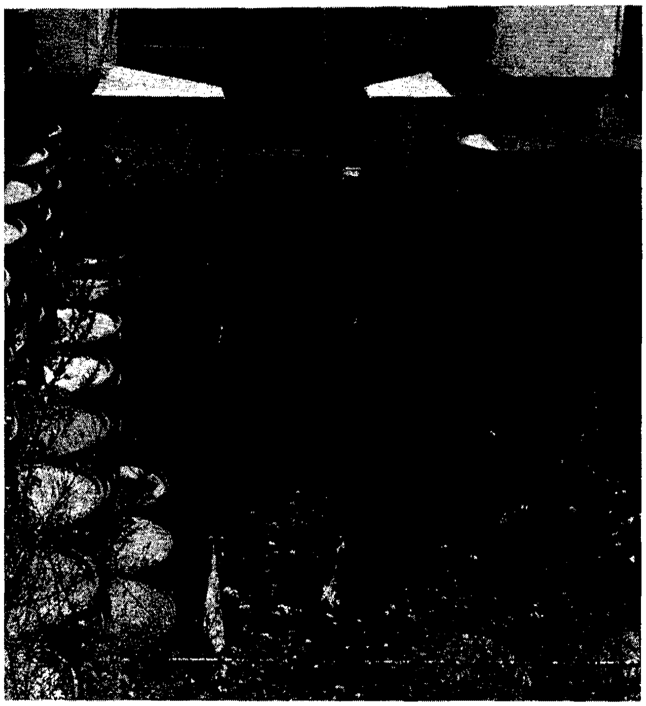
This "fish pole" radiation survey meter permits inspectors to accurately assess the radioactivity of low-level waste material in trenches prior to burial. Containers have just been delivered and dumped by trucks in background and will be covered by earth-moving equipment as soon as radiation levels and distribution have been recorded.

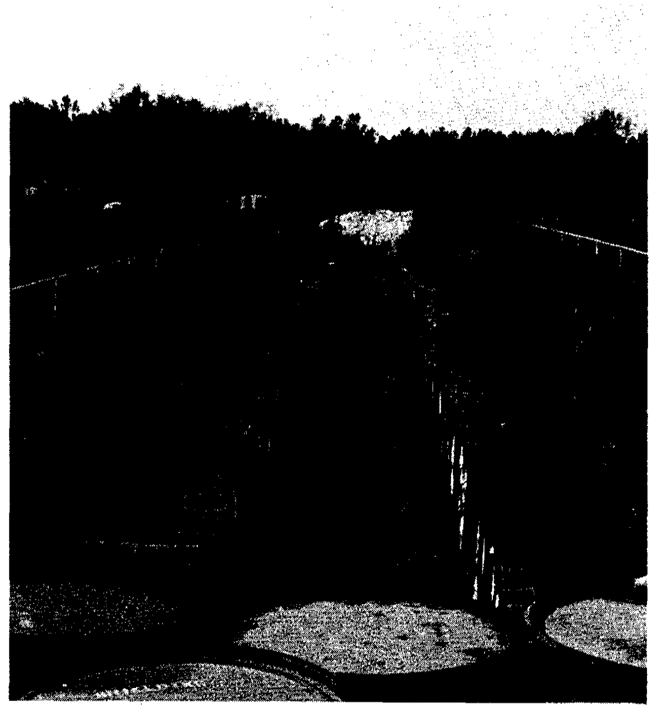
NRC's work in regulatory development in 1979 has been focused on development of requirements that can apply to a broad range of disposal alternatives. It has become increasingly clear to the NRC during 1979 that alternative disposal methods are critically needed and a regulatory base should be put in place in timely manner.

### Licensing

NRC continued its licensing activity in low-level waste management in 1979. The NRC license for disposal of special nuclear material (SNM) at Hanford, Wash., was renewed in November 1979. This license was closely coordinated with the State of Washington and contains many upgraded requirements for operations at the site.

An environmental assessment was continued in 1979 for decommissioning of the Sheffield, Ill., facility and should be completed in 1980. The licensee applied for an expansion and continued operation of the site.





Low-level waste containers that contain higher-activity materials (or that emit higher levels of radioactivity) than those dumped in standard low-level disposal trenches are deposited in trenches or containment holes which offer greater depths and heavier shielding. Two types of such containments are shown here: (left) a reinforced concrete lined pit, and (right) a narrow, deep trench shielded by the filled barrels along the top. Both such containments are in protected, posted areas at a supervised site.

However, the licensee subsequently petitioned to withdraw the renewal and expansion application. The Atomic Safety and Licensing Board approved the withdrawal of the expansion request but the renewal will be subject to hearings. The applicant's withdrawal of the operating/expansion application was based on recognized technical problems for which the solution proposed by the licensee was not acceptable to the NRC. In addition to the above, five license amendments were granted for existing sites.

Since two of the previous six commercial disposal operations have closed, (West Valley, N.Y. and Maxey Flats, Ky.) and the Sheffield, Ill., disposal operation is effectively closed, only three commercial operations currently exist (Barnwell, S.C.; Beatty, Nev.; and Richland, Wash.). Thus the present disposal capacity is primarily located in the West and Southeast and represents an undesirable regional imbalance. The waste from reactors and other waste generators located in the Northeast and Midwestern United States must be transported either to the Southeastern United States or to the West.

A number of significant events occurred in 1979 that affect low-level waste disposal operation. It became obvious that more attention should be paid to decontamination and decommissioning wastes, from the viewpoint of low-level waste disposal operations. Some of these activities pose unique problems, such as the TMI waste and the waste from the decontamination of the Dresden I reactor. It also became obvious that further work is required for liquid scintillation waste. The State of South Carolina decided in 1979 not to accept any more shipments of this type of waste, and the waste must presently be shipped to the disposal operations in the West. NRC is investigating various alternatives for the treatment and disposal of this type of waste.

Lastly, it became obvious that NRC must take a more active role in upgrading packaging requirements and waste form for certain types of waste and increase inspection and enforcement of existing regulations covering the shipment of waste. For example, a fire occurred on a truck containing waste packages at the Beatty, Nev., site and large volumes of free-standing liquids were found upon inspection of packages of solidified wastes received at various low-level waste disposal sites. As a result of such events, the governors of the three States having commercial low-level waste disposal operations sent a joint communique dated July 10, 1979, to NRC demanding action by NRC and the Department of Transportation to improve packaging requirements and increase inspection and enforcement of existing regulations. In response, NRC issued a bulletin to all licensees stressing the need to give careful attention to the packaging and transportation of waste and instituted action, with the cooperation of the States and the DOT, to inspect shipments on a more frequent basis and take more stringent enforcement actions. (See also Chapter 4.)

As mentioned above, a severe regional imbalance has emerged from the locations of today's low-level waste burial grounds. This imbalance was aggravated in 1979 when two of the sites closed and then reopened and a curtailment was placed on the amount of waste that could be received at the third site. As a result, NRC went on record to state its judgment that low-level waste disposal is the responsibility of the States, for the States receive the benefits of the operations which generate the waste. NRC has worked with a number of States in 1979 and will continue to do so in 1980, to help the States explore the possibility of establishing new sites. The NRC effort took the form of assistance in setting forth licensing and regulatory requirements; however, NRC cannot promote the opening of new sites. This is a responsibility of the States, with assistance available from the Department of Energy should the States request such assistance.

### Technical Assistance to Agreement States

NRC has provided technical assistance to Agreement States in the licensing and regulation of low-level waste disposal operations in their jurisdiction. NRC has provided in 1979, assistance to the State of Washington as part of their renewal action for the State disposal license at Richland. In addition, NRC has provided,and will provide in 1980, assistance to the State of Kansas in evaluating an application for a new disposal site license at Lyons. Technical assistance was also given to the State of Nevada in 1979, and NRC is expecting to provide further assistance to Nevada in 1980 regarding renewal of the State license for the Beatty site. The NRC technical assistance supplements the State's resources and assures that the technical criteria used to license and regulate a low-level waste disposal operation in an Agreement State are compatible with the criteria used to license and regulate a low-level waste disposal operation under NRC's jurisdiction. In 1979, NRC worked with the States of South Carolina, Nevada and Washington to develop and implement new requirements at existing sites to upgrade and define acceptable waste forms.

## URANIUM RECOVERY MANAGEMENT

### Regulatory Development

NRC continued its uranium recovery regulatory development effort in 1979 with the objective of upgrading its regulations for uranium milling in 1980. The NRC published a draft generic environmental impact statement (GEIS) in April 1979 covering the U.S. uranium milling industry to the year 2000, with particular emphasis on mill tailings. In addition, NRC published draft regulations in August 1979, deriving from the environmental statement, and conducted extensive public meetings on the proposed regulations. The final GEIS and the final regulations are expected to be published in 1980.

The proposed regulations cover radioactive airborne emissions during operation, mill tailings disposal, decommissioning of mill structures and sites, supplementary institutional and procedure requirements, implementation of proposed requirements at existing sites, and heap leaching and small processing sites.

### Licensing

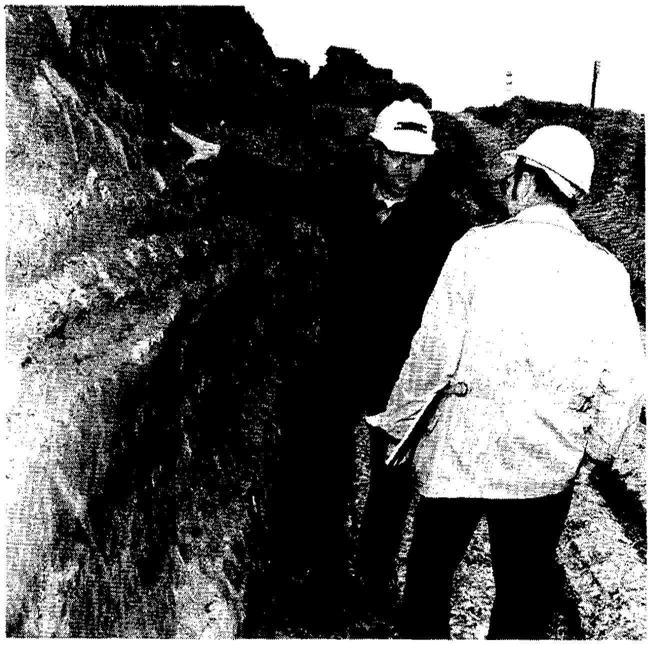
NRC continued its licensing effort in 1979. Twelve new uranium recovery facilities were licensed and one facility license was renewed. In addition, five major amendments were issued based upon licensee requests for facility modifications. There were 15 uranium mills, 5 heap leach/ore buying stations, 2 solution mining operations and 16 research and development (R&D) operations under NRC license in 1979.

Similar facilities exist in Agreement States. All these types of facilities are expected to grow numerically in the future. It is currently projected that in 1981 there will be 22 operating mills, 8 heap leach operations and ore buying stations, 6 commercial scale solution mining operations, and 23 R&D operations under NRC jurisdiction. A similar growth is expected in the number of these types of operations in Agreement States. Thus, the NRC and Agreement State workload in this area will experience a substantial growth in the next few years.

### Technical Assistance to Agreement States

During 1979, NRC provded technical assistance to the States of Washington, Oregon, Colorado, New Mexico, Arizona, California and Nevada in the licensing and regulation of uranium recovery operations under Agreement State jurisdiction. A total of six project reviews were completed. These reviews covered uranium mills, heap leach operations, solution mining operations, and R&D operations. The NRC assistance assures that the technical criteria used to license and regulate uranium recovery operations in Agreement States are compatible with those criteria used to license and regulate similar operations under NRC jurisdiction.

The Uranium Mill Tailings Radiation Control Act of 1978 was amended in 1979 to provide further clarification of the NRC/Agreement States interface with respect to the licensing and regulation of mill tailings. The Commission will continue to license tailings in non-Agreement States and the Agreement States will continue to license the mill tailings under State



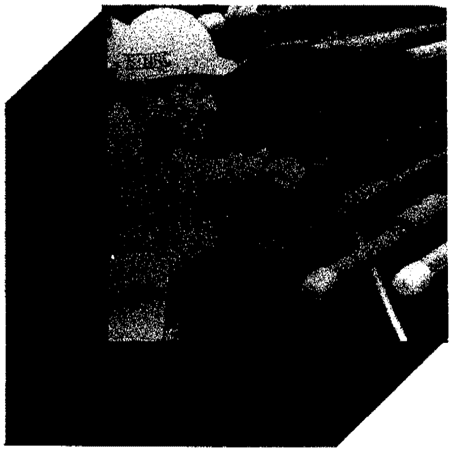
A State inspector and a State Agreements program reviewer examine a waste burial trench at Barnwell, S.C. Low-level radioactive wastes are deposited in such trenches and covered with backfill. Only three low-level waste burial sites are now operating in the United States. Barnwell is the only site in the eastern part of the country. The other two sites are at Hanford, Washington and Beatty, Nevada.

jurisdiction. NRC will provide technical assistance to the States in carrying out their responsibilities under the Act.

Technical assistance to the Agreement States by NRC will continue to cover non-routine safety and environmental assessment. For example, a tailings impoundment failure occurred at the United Nuclear Corporation uranium milling operation at Church Rock, N.M., on July 16, 1979. New Mexico is an Agreement State and the milling complex was licensed by the State in May 1977. Estimates of the amount of tailings released have varied, but it appears that about 100 million gallons of acidic tailings solutions and 1,100 tons of tailings solids escaped from the tailings impoundment area before the break in the dam could be closed. The State of New Mexico requested technical assistance from NRC and NRC personnel were dispatched to the site to aid the State. Extensive technical studies and analyses were also performed by NRC. Technical assistance to the State of New Mexico will continue to be provided by NRC in 1980.

## NRC Assessment of DOE Remedial Action Plans

NRC initiated in 1979 its evaluation of DOE remedial action plans for inactive sites. This will be a five year program which implements NRC's part of Title I of the Uranium Mill Tailings Radiation Control Act of 1978. DOE is responsible for remedial action at 21 inactive mill tailings sites and one other former ore processing site as specified in the Act. NRC is required to review DOE's proposed remedial actions and determine whether the remedial action plans are acceptable.



# 7 Inspection and Enforcement

**New emphasis was given in 1979 to direct NRC inspection of design, analytical and other technical activities of contractors.**

During 1979, the NRC continued to implement the plan calling for resident inspectors at each operating power reactor plant, at those plants in the later stages of construction, and at selected fuel cycle facilities. The accident at Three Mile Island (TMI) led to a decision to increase the number of resident inspectors to a level of one inspector for each unit at a multireactor site. Single unit sites will have two resident inspectors. By December 31, 1979, 60 inspectors were stationed as residents at 48 power reactor and fuel facility sites. Table 1 provides a listing of these sites. This additional effort has required an increase in the number of personnel from a staff ceiling of 715 in 1979 to 861 in 1980 for the NRC Office of Inspection and Enforcement. By the end of fiscal year 1980 there will be 157 resident inspectors on site compared to the original goal of 76. The reactor training provided for operations inspectors will be increased from a minimum of seven weeks to 10 weeks during 1980, with additional simulator and special plant observation training.

TMI impacted heavily on the planned inspection program. Special teams were sent to all operating pressurized water nuclear plants to review with licensee management the actions required as a result of the TMI accident. Review groups were formed to study the TMI accident and the lessons learned from it that would affect future inspection programs. An augmented 24-hour surveillance program was established at TMI that has required staffing support from all five NRC regional offices. From April through July of 1979, a 24-hour watch was established in each region and at the NRC Operations Center in Bethesda, Maryland, to provide the capability for responding immediately to any incidents or accidents. A direct "hotline" telephone system was installed in the Operations Center. This provides a direct line to each operating reactor power plant and all fuel processing facilities in the country. The system provides conference call capability between the NRC Operations Center, a plant, and the regional NRC office. In August, the 24-hour duty Officers in the regions were replaced by a communications system connected directly to the NRC Operations Center where 24-hour duty officer coverage is maintained. All calls to regional offices during non-duty hours are now diverted to the NRC Duty Officer at the Operations Center, who can promptly respond to the situation.

As a result of these actions associated with the TMI accident and related inspections, the number of routine inspections in 1979 was less than originally planned. Table 2 summarizes the inspections conducted during fiscal year 1979.

One or more noncompliance items were found in 33 percent of the more than 6,000 inspections and in 36 percent of the 121 investigations. The more severe sanctions imposed on licensees for failure to comply with NRC requirements included nine civil penalties and three orders to "cease and desist" operations, or for modifications, or suspensions of licenses (see Tables 4 and 5).

## THE INSPECTION PROGRAM

The inspection and enforcement program is directed by NRC's Office of Inspection and Enforcement (IE), with a headquarters staff located in Bethesda, Maryland, and a field staff deployed in NRC's five regional offices located in or near Philadelphia, Atlanta, Chicago, Dallas, and San Francisco. About 80 percent of the total office on-board staff of 730 is assigned to the regions.

The objectives of inspections are:

- To determine whether licensees are complying with NRC requirements.
- To identify conditions that may adversely affect public health and safety, the common defense

## Table 1. Sites Manned by Resident Inspectors During 1978 and 1979

| *Facility* | *Location* | *Licensee* |
|---|---|---|
| *Arkansas Nuclear Plant | Russelville, Ark. | Arkansas Power & Light Co. |
| Beaver Valley Power Station | Shippingport, Pa. | Duquesne Light Co. |
| Bellefonte Nuclear Plant | Scottsboro, Ala. | Tennessee Valley Authority |
| *Browns Ferry Nuclear Plant | Decatur, Ala. | Tennessee Valley Authority |
| Brunswick Steam Electric Plant | Southport, N.C. | Carolina Power & Light Co. |
| Callaway Plant | Fulton, Mo. | Union Electric Co. |
| Calvert Cliffs | Lusby, Md. | Baltimore Gas & Electric Co. |
| *Comanche Peak Steam Electric Station | Glen Rose, Tex. | Texas Power & Light, Dallas Power & Light, Texas Electric Service |
| Davis-Besse Nuclear Power Station | Oak Harbor, Ohio | Toledo Edison Co. |
| *Donald C. Cook Plant | Bridgman, Mich. | Indiana & Michigan Electric Co. |
| *Diablo Canyon Nuclear Power Plant | San Luis Obispo, Cal. | Pacific Gas & Electric Co. |
| *Dresden Nuclear Power Station | Morris, Ill. | Commonwealth Edison Co. |
| *Edwin I. Hatch Plant | Baxley, Ga. | Georgia Power Co. |
| Fort St. Vrain Nuclear Station | Platteville, Colo. | Public Service Co. of Colorado |
| Hartsville Nuclear Power Plant | Hartsville, Tenn. | Tennessee Valley Authority |
| *Indian Point Station | Indian Point, N.Y. | Consolidated Edison Co. |
| Joseph M. Farley Nuclear Plant | Dothan, Ala. | Alabama Power Co. |
| LaSalle County Nuclear Station | Seneca, Ill. | Commonwealth Edison Co. |
| Limerick Generating Station | Pottstown, Pa. | Philadelphia Electric Co. |
| Marble Hill Plant | Madison, Ind. | Public Service of Indiana |
| *Midland Nuclear Power Plant | Midland, Mich. | Consumers Power Co. |
| *Millstone Nuclear Power Station | Waterford, Conn. | Northeast Nuclear Energy Co. |
| North Anna Power Station | Mineral, Va. | Virginia Electric & Power Co. |
| *Oconee Nuclear Station | Seneca, S.C. | Duke Power Co. |
| Palisades Nuclear Power Station | South Haven, Mich. | Consumers Power Co. |
| Palo Verde Nuclear Station | Winterburg, Ariz. | Arizona Public Service Co. |
| *Peach Bottom Atomic Power Station | Peach Bottom, Pa. | Philadelphia Electric Co. |
| *Prairie Island Nuclear Plant | Red Wing, Minn. | Northern States Power Co. |
| Quad Cities Station | Cordova, Ill. | Commonwealth Edison Co. |
| Rancho Seco Nuclear Station | Sacramento, Cal. | Sacramento Municipal Utility District |
| *Salem Nuclear Generating Station | Salem, N.J. | Public Service Electric & Gas Co. |
| *San Onofre Nuclear Station | San Clemente, Cal. | Southern California Edison Co. & San Diego Gas & Electric Co. |
| Seabrook Nuclear Station | Seabrook, N.H. | Public Service Co. of N.H. |
| Sequoyah Nuclear Power Plant | Daisy, Tenn. | Tennessee Valley Authority |
| Shoreham Nuclear Power Station | Suffolk County, N.Y. | Long Island Lighting Co. |
| South Texas Nuclear Project | Bay City, Tex. | Houston Lighting & Power Co. |
| Summer Nuclear Station | Broad River, S.C. | South Carolina Electric & Gas Co. |
| *Surry Power Station | Gravel Neck, Va. | Virginia Electric & Power Co. |
| *Susquehanna Steam Electric Station | Berwick, Pa. | Pennsylvania Power & Light Co. |
| *Trojan Nuclear Plant | Prescott, Ore. | Portland General Electric Co. |
| Turkey Point Station | Florida City, Fla. | Florida Power & Light Co. |
| Washington Nuclear #2 | Richland, Wash. | Washington Public Power Supply System |
| *Watts Bar Nuclear Plant | Spring City, Tenn. | Tennessee Valley Authority |
| William H. Zimmer Nuclear Power Station | Moscow, Ohio | Cincinnati Gas & Electric Co. |
| Zion Nuclear Plant | Zion, Ill. | Commonwealth Edison Co. |
| *B&W-Apollo & Leechburg** (Fuel Facility) | Apollo, Pa. | Babcock & Wilcox Co. |
| *Westinghouse-Cheswick** (Fuel Facility) | Parks Township, Pa. | Westinghouse Electric Corp. |
| *Nuclear Fuel Services (Fuel Facility) | Erwin, Tenn. | Nuclear Fuel Services, Inc. |

*Assigned during calendar year 1978.
**Inspector stationed at Apollo, Pennsylvania, acts as inspector on a rotating basis at B&W's Apollo and Leechburg facilities and Westinghouse's Cheswick facility.

and security, the environment or the safeguarding of nuclear materials and facilities.

- To provide information to assist in developing a basis for issuance, denial, or amendment of an authorization, permit or license.
- To determine whether licensees and their contractors and suppliers have implemented adequate quality assurance programs.

When an inspection or investigation discloses events or conditions that present a potential or actual threat to public health and safety, the environment, or the safeguarding of nuclear materials and facilities, the NRC takes prompt action and routinely communicates with other parts of government, licensees and the public.

During fiscal year 1979, 174 new inspection procedures and/or instructions were issued and 123 were revised. In the area of construction inspection, for example, 22 extensively revised inspection procedures pertaining to welding were issued.

## Reporting Defects and Noncompliance

On June 6, 1977, the NRC published in the *Federal Register* a regulation (10 CFR Part 21) setting forth the requirements for implementing Section 206 of the Energy Reorganization Act of 1974. Individual directors or responsible officers of a firm involved in the nuclear industry are required to report noncompliance with NRC regulations or the existence of defects which could create a substantial health and safety hazard. Any such person who knowingly and consciously fails to provide the required reports to the NRC is subject to a civil penalty not to exceed $5,000 for each failure and a total amount not to exceed $25,000 within any 30-day period. The regulation became fully effective on January 6, 1978.

About 150 Part 21 Reports have been received by the NRC since the regulation became effective. The reports are reviewed to assess the reported deficiency, the adequacy of the proposed corrective action and the possibility of generic problems. IE inspectors ensure that appropriate followup actions are taken.

## Types of Inspections

NRC's inspections are of two basic types: routine and reactive. In routine inspections, NRC inspectors concentrate on determining the effectiveness of quality assurance systems by direct observation and verification of licensee activities, and by reviewing procedures, checking records, interviewing people, and, where appropriate, making direct measurements. Reactive inspections are conducted in response to information received by NRC regarding conditions or events affecting licensed facilities or material under NRC jurisdiction. Such information may come from routine NRC inspections; from an applicant, licensee, contractor or supplier; or from licensee employees or other members of the public.

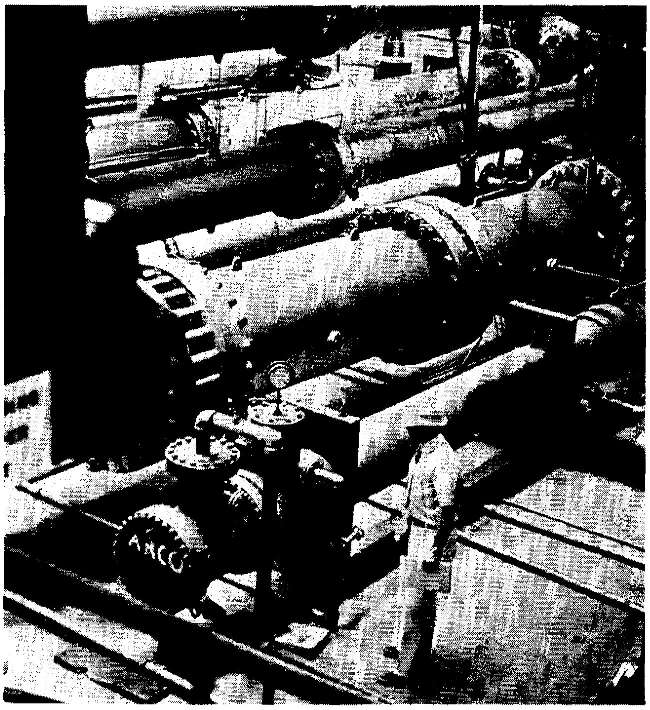
Inspections cover the entire range of NRC licensed activities. Reactor-related inspections cover all phases of nuclear power plants (preconstruction activities, construction, preoperational testing and startup, operation, shutdown and decommissioning) and similar phases of research and test reactors. In addition, NRC inspects the quality assurance programs of contractors and vendors who supply safety-related equipment, components and services to power reactors under construction or in operation.

## Licensee, Contractor and Vendor Inspection Program

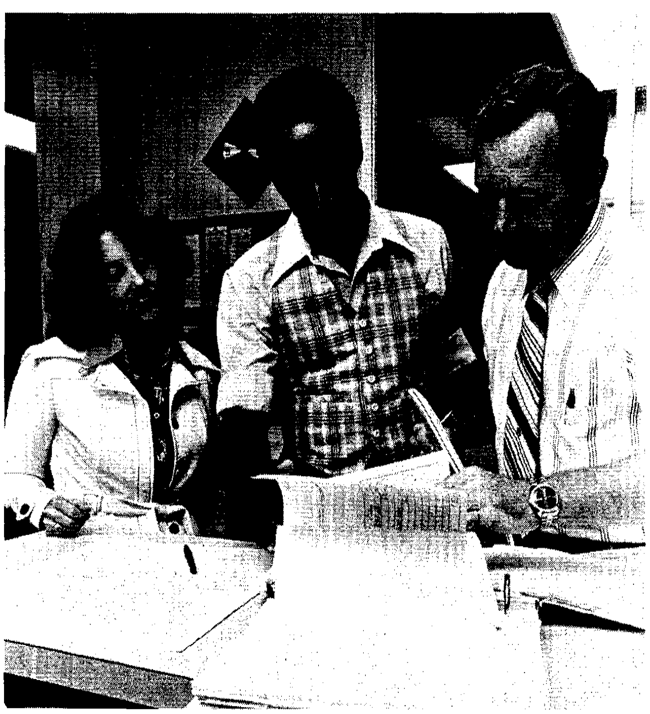
Approximately one-half the work associated with constructing a nuclear facility is accomplished off-site.

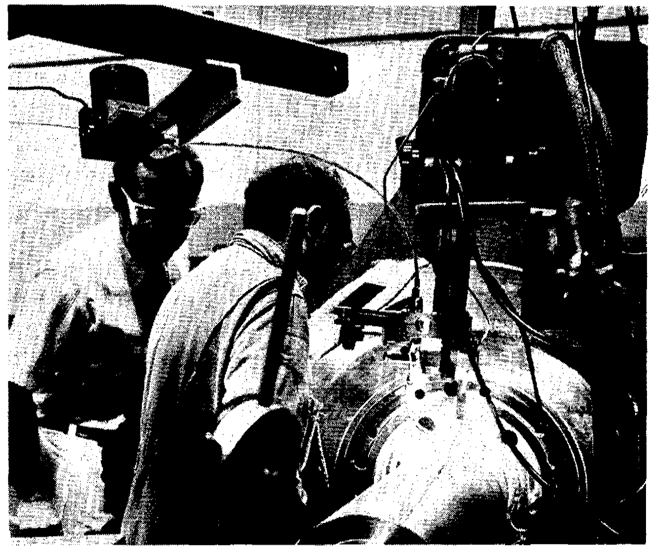
## Table 2. Inspections Conducted in Fiscal Year 1979

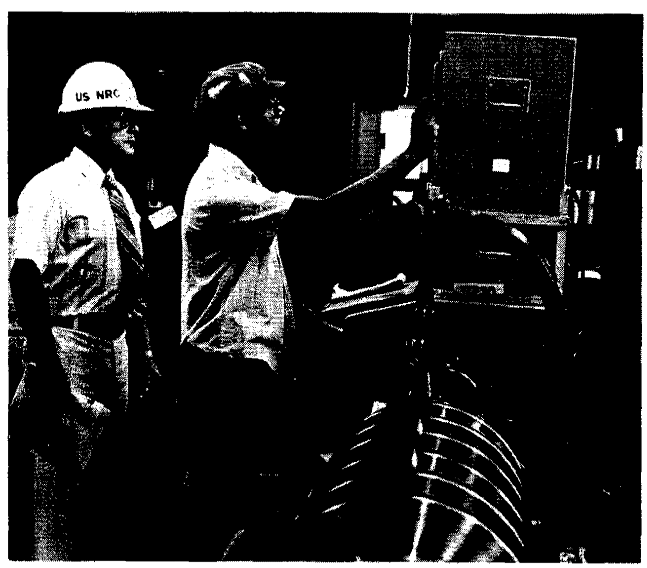
| *Program* | *Number of Licenses* | *Number of Inspections* |
|---|---|---|
| Power Reactor Construction | 114 | 1,787 |
| Operating Power Reactors | 70 | 1,761 |
| Other Reactors | 94 | 93 |
| Fuel Facilities | 39 | 203 |
| Materials | 8,586 | 1,976 |
| Vendors | 248 | 228 |
| Safeguards | 203 | 526 |



Starting above and reading clockwise, an inspector is shown during an inspection of pump vendor facilities, as he reads and records data during reactor coolant pump test, checks gas-metal arc welding being performed on a pump rotor, observes machining of pump rotor housing, watches vendor technician balance a pump rotor, and (below) reviews records of material identification and traceability.







This includes facility design and the fabrication of components of safety-related systems. Inspections of nuclear steam system suppliers, architect-engineers and vendors of safety-related components are performed by NRC's Licensee, Contractor and Vendor Inspection Branch (LCVIB) inspectors, located in the Region IV (Dallas) office. During fiscal year 1979, some 250 inspections were performed by the 21 LCVIB inspectors. Approximately 30 percent of these inspections were special reactive inspections involving component fabrication or design-related problems.

During the coming year, a modest shift in inspection emphasis is expected in the LCVIB. Activities experiencing change will include:

- Performing more reactive inspections.
- Redirecting emphasis toward the inspection of technical activities performed by contractors.
- Followup on Part 21 Reports, Bulletins and Circular issues.
- Inspecting and witnessing environmental qualification of electrical, instrumentation and control equipment.
- Inspecting design and analytical work performed by licensee contractors.

## Performance Appraisal Program

During fiscal year 1979, five licensee management appraisal inspections and one IE program appraisal inspection series (pertaining to surveillance testing) were completed. Nine management appraisal inspections and four IE program appraisal inspections are planned for fiscal year 1980. Objectives of the program are to:

- Evaluate performance of utility management.
- Analyze effectiveness of the NRC inspection program.
- Confirm objectivity of NRC inspectors.

Three Performance Appraisal Team (PAT) inspectors participated in the IE investigation of the TMI accident; PAT inspectors also participated in other investigations and special inspections.

## Independent Measurement/Verification Program

IE has increased its efforts associated with direct verification of licensee/ contractor activities during the construction phase. NRC periodically uses contractors to perform non-destructive testing activities, and, in August 1979, selected a contractor to perform destructive testing of selected materials used in safety-related structures and systems. Continued effort in these areas is planned for fiscal year 1980.

Inspections related to nuclear materials include inspection of the construction and operation of uranium mills; fuel fabrication, processing and reprocessing plants; waste disposal facilities; and the industrial, educational and medical uses of radioactive material. NRC inspections also include measures for safeguarding nuclear material from theft and sabotage, for physical protection of reactors and fuel cycle facilities, and for transportation of nuclear materials.

The number of inspections carried out during fiscal year 1979 (ending September 30) for each of these activities is shown in Table 2.

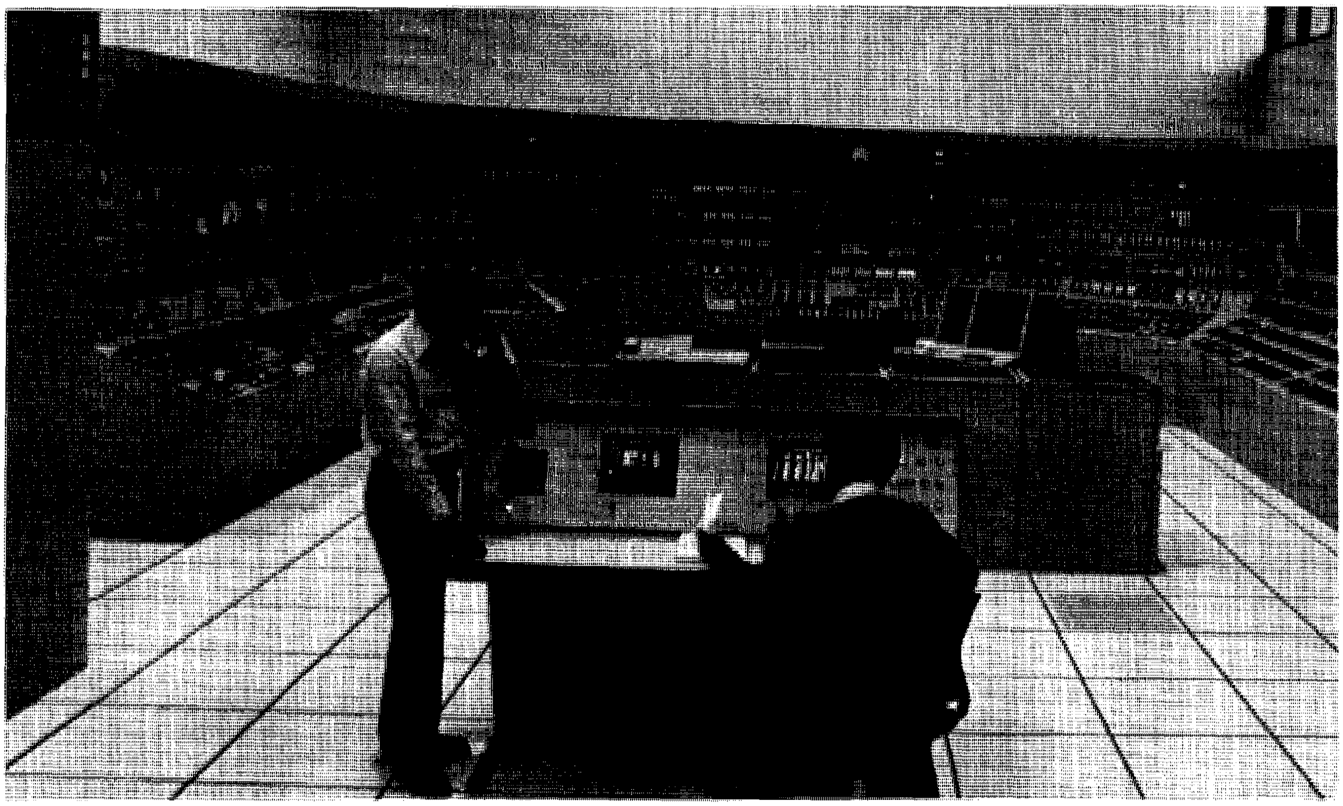
## Government-Industry Efforts

The NRC inspection program is based on the premise that the licensee is responsible for carrying out licensed activities safely and in compliance with NRC requirements. NRC determines whether the licensee has established the management control systems necessary to meet regulatory responsibilities. The inspection pattern for large, complex nuclear facilities is pyramidal, with each level of activity verified, inspected or audited by those above. The NRC inspection effort is essentially the apex of the pyramid, i.e., NRC performs the last in the series of inspections and audits conducted by many different groups. Since NRC inspection manpower is usually far less than that of licensees and contractors, NRC inspectors cannot inspect all components and activities; thus, they probe the "pyramid" to determine whether the licensees' and contractors' activities are properly performed. In addition, the IE inspection program provides for independent effort by NRC inspectors whenever the inspector determines such action is necessary.

## Inspection Activities Resulting from TMI

Shortly after the Three Mile Island (TMI) accident, a series of Inspection and Enforcement (IE) Bulletins were issued to all operating power reactor licensees addressing the early lessons learned. The IE Bulletins provided licensees with information about the series of events that occurred at TMI and directed each licensee to make changes to certain equipment and operating procedures consistent with the reactor design, and to conduct special operator training. In response to the Bulletins, licensees provided details for completion of immediate actions and plans for completion of longer term actions. Special follow-up inspections were conducted to verify that licensees had taken appropriate action.

During the period April 18–23, 1979, six specially trained NRC teams visited all operating pressurized water nuclear power plants, except those designed by Babcock and Wilcox, designer of the TMI plant. These



Pressurized water reactor control room simulators, such as this one at the Tennessee Valley Authority's Sequoyah Nuclear Power Plant, came into greatly increased use in 1979. The training of new inspectors to accommodate the growing NRC resident inspector program brought increased student loads to such facilities early in the fiscal year, and the training was intensified even more in the later months as deficiencies in operator training highlighted by the Three Mile Island accident became apparent.

teams reviewed and discussed with licensee operations personnel and station management the TMI accident chronology and licensee actions that had been specified in the IE Bulletins. For the Babcock and Wilcox designed facilities, the resident inspector, with assistance from regional-based inspectors, conducted this special briefing.

By April 2, following the TMI accident, resident inspectors had been assigned to all operating Babcock and Wilcox designed plants where resident inspectors had not previously been assigned. In addition to responsibilities normally assigned resident inspectors, inspectors at these sites performed additional inspections to assure plant safety in light of the events at TMI.

## Impact of TMI on Inspection Program

The impact on the routine inspection program for the first several months following the TMI accident was significant. Efforts expended by inspectors at the TMI site, the expedited assignment of inspectors to all Babcock and Wilcox power reactor facilities and the requirements imposed by the need for special inspections of all operating power reactor facilities caused a thinner coverage, and in many cases deferral or deletion of portions of the routine inspection program activities.

On a continuing basis, increased emphasis has been placed on identifying isolated plant problems and generic issues and managing their resolution. To accomplish this, the headquarters staff has been augmented with a group of highly specialized systems engineers whose responsibilities include more in-depth review and follow-up on plant events.

Long-term inspection program changes to reflect lessons learned from the TMI accident are still in various formative stages. Specific problems requiring program changes have generally been diagnosed. Program modification, implementation and attendant process evaluations have been done to the extent possible for changes that represent an expansion of current programs, such as resident inspection.

Studies to evaluate certain major changes in emphasis of the inspection program have been initiated to determine the effectiveness and efficiency of these changes as implemented. The results of the IE Special Review Group on lessons learned from Three Mile Island have provided a basis for the integration of lessons learned into the current inspection program.

## Resident Inspector Program

During 1979, the NRC made further progress in the program to station inspectors full time at the sites of nuclear power reactors and major fuel cycle facilities.

Experience with resident inspection results and licensee events and actions have led to plans for further expansion of the resident inspector program. The program is being accelerated in consonance with the President's message of December 7, 1979 on the Kemeny Commission Report. Steps to upgrade its effectiveness also are being taken in response to recommendations in a General Accounting Office report issued to Congress in November 1979. As noted above, approval has been given to assigning, in addition to the site resident inspector, resident inspectors to nuclear power reactor plant units (many sites have more than one unit). The total number of resident inspectors at any site will generally equal the number of units at that site, with a minimum of two inspectors per site. This augmented coverage will provide additional safety assurances through increasing NRC presence, including the number of independent observations of licensee safety-related activities and equipment.

By December 31, 1979, 60 inspectors were deployed to the sites of 45 nuclear power stations—including several power reactor plants under construction—and of three fuel facilities. By June 1980, each site with an operating or preoperational reactor should have at least one resident inspector. Each such site is expected to have its full complement of at least two inspectors by September 30, 1980, at which time some 130 resident inspectors will be deployed at 60 sites. Thereafter, resident inspectors will be assigned to reactors as they reach the pre-operational stage.

The NRC also is assigning resident inspectors to sites where nuclear plant construction is in the final stage. Further, resident inspectors will be assigned to sites where problems are evident in earlier stages of plant construction.

## Bulletins, Circulars and Information Notices

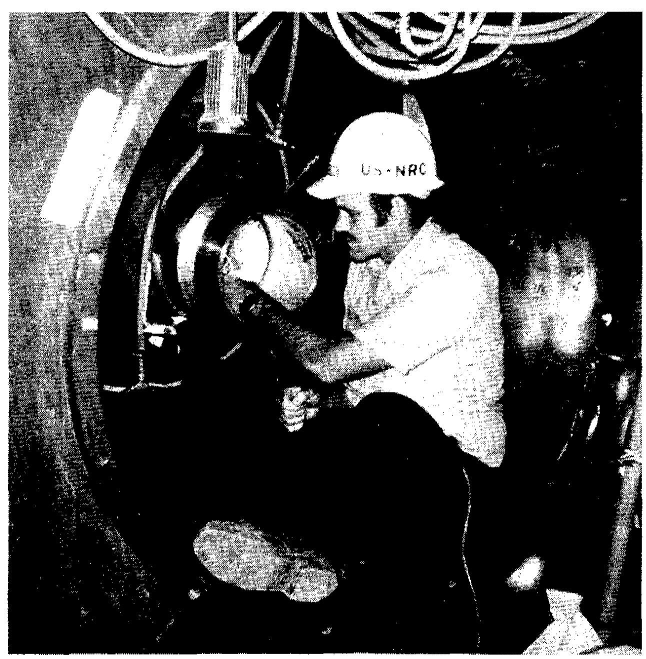
During 1979, the NRC's issuance of Bulletins, Circulars and Information Notices was increased both in number and significance. The NRC's Office of Inspection and Enforcement has issued Bulletins since 1971, Circulars since 1976 and Information Notices for the first time in 1979.

The IE Bulletin is used to notify licensees of specific actions to be taken. It usually requires that the licensees provide a report to the NRC describing the actions they take in response to the Bulletin. The Bulletin addresses matters of concern or events related to reactor safety, material safeguards, radiological safety or environmental protection.

Bulletins usually, although not always, require the action on a one-time only basis. However, Bulletins are not intended to substitute for new or revised license conditions or requirements. If a licensee refuses to perform an action set forth in the Bulletin, the requirement for the action may be imposed on the licensee by an Order.

Particular considerations which might require the issuance of a Bulletin include events in which the safety significance is of such a magnitude as to result in an immediate impact on all of a certain type of licensee. The Three Mile Island accident represents such an event, and it was addressed by multiple Bulletins. Other considerations include events having a potential generic problem impact and where the event requires action by a particular class of license or permit holder.

The IE Circular is used to notify licensees of actions which the NRC recommends be taken. These matters are generally of lesser significance than those address-




**NRC resident inspector checks a weld in a reactor vessel thermal sleeve at the Susquehanna Steam Electric Station at Berwick, PA.**

## Table 3. IE Bulletins, Circulars, and Information Notices Issued in 1979

### BULLETINS

| *Bulletin No.* | *Subject* | *Date Issued* | *Issued to* |
|---|---|---|---|
| 79-01 | Environmental Qualification of Class IE Equipment | 2/8/79 | All power reactor facilities with an OL or CP |
| 79-01A | Environmental Qualification of Class IE Equipment (Deficiencies in the Environmental Qualification of ASCO Solenoid Valves | 6/6/79 | All power reactor facilities with an OL or CP |
| 79-02 | Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts | 3/2/70 | All power reactor facilities with an OL or CP |
| 79-02 (Rev. 1) | Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts | 6/21/79 | All power reactor facilities with an OL or a CP |
| 79-02 (Rev. 1) | Pipe Support Base Plate Designs Using Concrete (Supplement 1) | 8/20/79 | All power reactor facilities with an OL or a CP |
| 79-03 | Longitudinal Welds Defects in ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Co. | 3/12/79 | All power reactor facilities with an OL or CP |
| 79-04 | Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation | 3/30/79 | All power reactor facilities with an OL or CP |
| 79-05 | Nuclear Incident at Three Mile Island | 4/2/79 | All power reactor facilities with an OL and CP |
| 79-05A | Nuclear Incident at Three Mile Island | 4/5/79 | All B&W power reactor facilities with an OL |
| 79-05B | Nuclear Incident at Three Mile Island | 4/21/79 | All B&W power reactor facilities with an OL |
| 79-05C&06C | Nuclear Incident at Three Mile Island - Supplement | 7/26/79 | To all PWR power reactor facilities with an OL |

## BULLETINS

| *Bulletin No.* | *Subject* | *Date Issued* | *Issued to* |
|---|---|---|---|
| 79-06 | Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident | 4/11/79 | All pressurized water power reactors with an OL except B&W facilities |
| 79-06A | Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident | 4/14/79 | All pressurized water power reactor facilities of Westinghouse design with an OL |
| 79-06A (Rev. 1) | Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident | 4/18/79 | All pressurized water power reactor facilities of Westinghouse design with an OL |
| 79-06B | Review of Operational Errors and System Misalignments Identified During the Three Mile Island Incident | 4/14/79 | All Combustion Engineering designed pressurized Water power reactor facilities with an OL |
| 79-07 | Seismic Stress Analysis of Safety-Related Piping | 4/14/79 | All power reactor Facilities with an OL or CP |
| 79-08 | Events Relevant to BWR Reactors Identified During Three Mile Island Incident | 4/14/79 | All BWR power reactor facilities with an OL |
| 79-09 | Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems | 4/17/79 | All power reactor facilities with an OL or CP |
| 79-10 | Requalification Training Program Statistics | 5/11/79 | All power reactor facilities with an OL |
| 79-11 | Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems | 5/22/79 | All power reactor facilities with an OL or a CP |
| 79-12 | Short Period Scrams at BWR Facilities | 5/31/79 | All GE BWR facilities with an OL |
| 78-12B | A Typical Weld Material in Reactor Pressure Vessel Welds | 3/19/79 | All power reactor facilities with an OL or CP |

## Table 3. IE Bulletins, Circulars, and Information Notices Issued in 1979—Continued

### BULLETINS—Continued

| *Bulletin No.* | *Subject* | *Date Issued* | *Issued to* |
|---|---|---|---|
| 79-13 | Cracking in Feedwater System Piping | 6/25/79 | All PWRs with an OL for action; all BWRs with a CP for information |
| 79-14 | Seismic Analyses for As-Built Safety-Related Piping System | 6/2/79 | All power reactor facilities with an OL or a CP |
| 79-15 | Deep Draft Pump Deficiencies | 7/11/79 | All power reactor licensees with a CP and/or OL |
| 79-16 | Vital Area Access Control | 7/26/79 | All holders of and applicants for OL |
| 79-17 | Pipe Cracks in Stagnant Borated Water Systems at PWR Plants | 7/26/79 | All PWR's OL |
| 79-18 | Audibility Problems Encountered on Evaluation | 8/7/79 | All power reactor facilities with an OL |
| 79-19 | Packaging Low-Level Radioactive Waste for Transport and Burial | 8/10/79 | All power and research reactors with OLs, fuel facilities except uranium mills, and certain materials licensees. |
| 79-20 | Packaging Low-Level Radioactive Waste for Transport and Burial | 8/10/79 | All materials licensees who did not receive Bulletin No. 79-19 |
| 79-21 | Temperature Effects on Level Measurements | 8/13/79 | All PWRs with an OL |
| 79-22 | Possible Leakage of Tubes of Tritium Gas in Timepieces for Luminosity | 9/5/79 | To each licensee who receives tubes of tritium gas used in timepieces for luminosity |
| 79-23 | Potential Failure of Emergency Diesel Generator Field Exciter Transformer | 9/12/79 | All power reactor facilities with an OL or a CP |
| 79-24 | Frozen Lines | 9/27/79 | All power reactor facilities which have either OLs or CPs and are in the late stage of construction |

### CIRCULARS

| *Circular No.* | *Subject* | *Date Issued* | *Issued to* |
|---|---|---|---|
| 79-01 | Administration of Unauthorized Byproduct Material To Humans | 1/12/79 | All holders of licenses except teletherapy medical Licenses and Each Radiopharmaceutical Supplier |

| *Circular No.* | *Subject* | *Date Issued* | *Issued to* |
|---|---|---|---|
| 79-02 | Failure of 120 Volt Vital AC Power Supplies | 2/16/79 | All holders of reactor OLs and CPs |
| 79-03 | Inadequate Guard Training Qualification and Falsified Training Records | 2/23/79 | All holders of and applicants for special nuclear material licenses in safeguards Group |
| 79-04 | Loose Locking Nut on Limitorque Valve Operators | 3/16/79 | All holders of reactor OLs or CPs |
| 79-05 | Moisture Leakage in Stranded Wire Conductors | 3/20/79 | All holders of reactor OLs or CPs |
| 79-06 | Failure to Use Syringe and Bottle Shields In Nuclear Medicine | 4/19/79 | All holders of medical licenses except teletherapy licenseesIssued to |
| 79-07 | Unexpected Speed Increase of Reactor Recirculation MG Set Resulted in Reactor Power Increase | 5/2/79 | All holders of BWR OLs or CPs |
| 79-08 | Attempted Extortion Low Enriched Uranium | 5/18/79 | All fuel facilities licensed by NRC |
| 79-09 | Occurrences of Split or Punctured Regulatory Diaphrams in Certain Self Contained Breathing Apparatus | 6/22/79 | All materials priority I, fuel Cycle and Operating reactor licenses |
| 79-10 | Pipefittings Manufactured from Unacceptable Material | 6/26/79 | All power reactor licensees with a CP and/or OL |
| 79-11 | Design/Construction Interface Problem | 6/27/79 | All applicants for, and holders of Power Reactors CPs |
| 79-12 | Potential Diesel Generator Turbocharger Problem | 6/28/79 | All power reactor operation facilities and all utilities having a CP |
| 79-13 | Replacement of Diesel Fire Pump Starting Contactors | 7/10/79 | All power reactor Operations facilities and all utilities having a CP |
| 79-14 | Unauthorized Procurement and Distribution of XE-133 | 7/13/79 | All medical licensees except teletheraphy medical licensees and to all radiopharmaceu tical suppliers |

## Table 3. IE Bulletins, Circulars, and Information Notices Issued in 1979—Continued

### CIRCULARS

| Circular No. | Subject | Date Issued | Issued to |
|---|---|---|---|
| 79-15 | Bursting of High Pressure Hose and Malfunction of Relief Valve "O" Ring in Certain Self-Contained Breathing Apparatus | 8/8/79 | All materials Priority I, fuel cucle and operating power reactor licensees |
| 79-16 | Excessive Radiation Exposures To Members of the general Public and a Radiographer | 8/16/79 | All radiography licensees |
| 79-17 | Contact Problem in SB-12 Switches on General Electric Company Metal Clad Circuit Breakers | 8/14/79 | All power reactor licensees with a CP and/ or OL |
| 79-18 | Proper Installation of Target Rock Safety-Relief | 9/10/79 | All holders of power reactors OLs and CPs |
| 79-19 | Loose Locking Devices on Ingersoll-Rand Pumps | 9/13/79 | All holders of power reactors OLs and CPs |
| 79-20 | Failure of GTE Sylvania Relay, Type PM Bulletin 7305, Catalog 5&12-11-AC | 9/24/79 | All holders of power reactors OLs and CPs |

### INFORMATION NOTICES

| Information Notice No. | Subject | Date Issued | Issued to |
|---|---|---|---|
| 79-01 | Bergen-Paterson Hydraulic Shock and Sway Arrestor | 2/2/79 | All power reactor facilities with an OL and or CP |
| 79-02 | Attempted Extortion of Low Enriched Uranium | 2/2/79 | All fuel facilities |
| 79-03 | Limitorque Valve Geared Limit Switch Lubricant | 2/9/79 | All power reactor facilities with an OL or a CP |
| 79-04 | Degradation of Engineered Safety Features | 2/16/79 | All power reactor facilities with an OL or a CP |
| 79-05 | Use of Improper Materials in Safety-Related Components | 3/21/79 | All power reactor facilities with an OL or CP |
| 79-06 | Stress Analysis of Safety-Related Piping | 3/23/79 | All holders of reactor OLs or CP |
| 79-07 | Rupture of Radwaste Tanks | 3/26/79 | All power reactor facilities with an OL or CP |
| 79-08 | Interconnection of Contaminated Systems with Service Air Systems Used as the Source of Breathing Air | 3/28/79 | All power reactor facilities with an OL and Pu processing fuel facilities |

| Information Notice No. | Subject | Date Issued | Issued to |
|---|---|---|---|
| 79-09 | Spill of Radioactively Contaminated Resin | 3/30/79 | All power reactor facilities with an OL |
| 79-10 | Nonconforming Pipe Support Struts | 4/16/79 | All power reactor facilities with a CP |
| 79-11 | Lower Reactor Vessel Head Insulation Support Problem | 5/7/79 | All holders of reactor OLs and CPs |
| 79-12 | Attempted Damage to New Fuel Assemblies | 5/11/79 | All fuel facilities, research reactors,and power reactors with an OL or CP |
| 79-13 | Indication of Low Water Level in the Oyster Creek Reactor | 5/29/79 | All holders of reactor OLs and CPs |
| 79-14 | NRC Position on Electrical Cable Support Systems | 6/11/79 | All power reactor facilities with a CP |
| 79-15 | Deficient Procedures | 6/7/79 | All holders of reactor OLs and CPs |
| 79-16 | Nuclear Incident at Three Mile Island | 6/22/79 | All research reactors and test reactors with OLs |
| 79-17 | Source Holder Assembly Damage from Misfit Between Assembly and Reactor Upper Grid Plate | 6/20/79 | All holders of reactor OLs and CPs |
| 79-18 | Skylab Reentry | 7/5/79 | All holders of reactor OLs |
| 79-19 | Pipe Cracks in Stagnant Borated Water Systems At PWR Plants | 7/17/79 | All holders of reactor OLs and CPs |
| 79-20 | NRC Enforcement Policy NRC LIcensed Individuals | 8/10/79 | All holders of reactor OLs and CPs and production licensees with licensed operators |
| 79-21 | Transportation and Commercial Burial of Radioactive Material | 9/7/79 | All power and research reactors with OLs |
| 79-22 | Qualification of Control Systems | 9/14/79 | All power reactor facilities with OLs and CPs |
| 79-23 | Emergency Diesel Generator Lube Oil Coolers | 9/25/79 | All power reactor facilities holding OLs and CPs |
| 79-24 | Overpressurization of Containment of a PWR Plant After a Main Steam Line Break | 9/28/79 | All power reactor facilities with a CP |

ed by a Bulletin, and a written response by the licensee is not required. The licensees may or may not initiate the recommended action. However, if further analysis and/or information regarding the matter indicates increased significance, it may result in the issuance of a Bulletin.

The particular concerns which might require issuance of a Circular include those for which a Bulletin is applicable, except that the impact is of less significance and is not sufficient to warrant specific actions by license or permit holders.

The Information Notice was first put in use in 1979. It is a mechanism by which the NRC is able to rapidly transmit information applicable or potentially applicable to license and permit holders. The information may or may not have been analyzed by NRC. It does not require acknowledgment or response but licensees are instructed to take appropriate action if the information applies to their facility. The concerns which might require issuance of an Information Notice include those for which a Bulletin or Circular may be applicable, but for which significance of the event or condition does not warrant issuance of a Bulletin or Circular. Of course, a Bulletin or Circular may be issued subsequent to an Information Notice on a particular concern as a result of problem evolution and further evaluation. Information Notices may also be used to transmit additional information on previously issued Bulletins or Circulars to license and permit holders.

A listing of the Bulletins, Circulars, and Information Notices issued from January 1, 1979, through September 30, 1979, is included in Table 3 to indicate the types of conditions addressed by these different publications.

## Other Reactive Effort

During fiscal year 1979, the effort expended on reactive inspections, investigations and related work has increased considerably, in addition to that expended on investigation and evaluation of the TMI accident.

Some construction sites have required between 50 and 250 man-days of unplanned reactive effort resulting in some cases in the postponement of routine inspection activities. A considerable amount of this reactive effort relates to inspection, investigation and follow-up effort, associated with allegations, Part 21 Reports and Bulletin, Circular and Information matters. The following construction problems have required substantial reactive effort by both headquarters and regional personnel:

- Pipe support base plate/anchor bolts
- Weld integrity (pipe welds)
- Pump performance
- Piping analysis and as-built conditions
- Steam generators
- Structural concrete
- Foundations

## ENFORCEMENT ACTIVITIES

The regulatory program is designed to assure that licensees perform in accordance with NRC regulations, licenses and permits and with applicable sections of Federal statutes. NRC is empowered to take enforcement action when licensees are not satisfying these requirements or are conducting operations in a way that might endanger the public health and safety or the environment, or adversely affect the common defense and security.

Enforcement action may be taken, for example, when certain significant safety-related matters not meeting NRC requirements have escaped the licensee's attention or when procedures are improperly controlled and the fact is first discovered during an NRC inspection. Such situations reflect adversely on the effectiveness of the licensee's management or quality assurance program. Enforcement action requires the licensee to correct the particular problems and establish measures to preclude reoccurrence—including deficiencies in his quality assurance program if such deficiencies allowed the problem to occur, continue or reoccur.

The severity of NRC enforcement actions varies with the seriousness of the matter and the licensee's previous compliance record. Several levels of NRC action are provided:

- Written Notices of Violation are provided for instances of noncompliance with NRC requirements.
- Civil penalties are considered for licenses who evidence significant or repetitive items of noncompliance, particularly when a Notice of Violation has not been effective. Civil penalties may also be imposed for particularly significant first-of-a-kind violations.
- Orders to "cease and desist" operations, or for modification, suspension, or revocation of licenses, are used to deal with licensees who do not respond to civil penalties or to deal with violations that constitute a significant threat to public health and safety or to the common defense and security. In the latter case, an order may be made effective immediately.

Tables 4 and 5 summarize the enforcement actions taken during the report period.

## Table 4. Civil Penalties Imposed—Fiscal Year 1979

| *Licensee* | *Amount* | *Reason* |
|---|---|---|
| Wisconsin Public Service Corporation Green Bay, Wisconsin (Kewaunee Plant) | $7,000 (reported as Pending in FY 78) | Failure to perform a survey required by regulations to assure control of personnel exposures. Licensee requested a hearing; however, a negotiated settlement was accepted by the licensee and the licensee paid the $7,000 penalty. |
| Jersey Central Power and Light Company Morristown, New Jersey (Oyster Creek Plant) | $26,000 | Failure to follow radiation safety procedures and noncompliance items in the safeguards area. |
| Twin City Testing and Engineering Labs., Inc. St. Paul, Minnesota (Radiographer) | $2,500 | Exposure to the lower back of an individual. Failure to perform necessary radiation surveys. |
| Niagara Mohawk Power Corporation (Nine Mile Point Unit 1) | $18,000 (pending) | Noncompliance items in the physical security area. |
| United Nuclear Corporation Wood River Junction, RhodeIsland (Fuel Processor) | $15,750 (pending) | Noncompliance items in the physical security area. |
| University of Wisconsin Madison, Wisconsin (Academic Broad License) | $2,300 | Inadequate training of personnel, failure to evaluate internal exposures of personnel and releases of airborne material to unrestricted areas. |
| Virginia Electric and Power Company (Surry Unit 2) | $15,000 (pending) | Whole body exposure of an individual and failure to follow procedures. |
| Nuclear Pharmacy, Inc. Milwaukee, Wisconsin (Radiopharmaceutical Distributor) | $24,000 (pending) | Distribution of radioactive material not intended for human use to medical licensees, relabeling and misrepresenting the material as suitable for human use. |
| University of Minnesota Minneapolis, Minnesota | $4,300 | Exposures of three individuals to airborne radioactive material and other noncompliance items in the health and safety area. |

## Enforcement Improvements

The Office of Inspection and Enforcement is seeking continued improvement in enforcement. In December 1979 the enforcement criteria concerning the transportation of radioactive material were upgraded. The Commission also has forwarded to Congress a request to increase NRC's statutory authority to impose civil penalties. If this request is implemented by amendment of the Atomic Energy Act, NRC's maximum allowable penalties will increase from $5,000 to $100,000 for a single violation and from $25,000 to no limit for all violations committed by a licensee within 30 days. Such an increase would provide greater incentives for major NRC licensees to comply with the regulatory requirements. A greater range would also permit the penalties to be imposed by NRC to reflect more equitably the different classes of licensees and the seriousness of offenses. The Commission approved a proposal that copies of escalated enforcement orders and civil penalties be routinely forwarded to State public utility regulatory groups and to State attorneys general for their information. Routine mailing of these communications started in December 1979.

NRC is continuing efforts to develop better methods for the evaluation of the regulatory performance of major licensees. By identifying licensees whose performance may require improvement, NRC hopes to anticipate potential safety and security problems and avert them through prompt remedial action. This would also improve the effectiveness of NRC's use of inspection resources. Identifying valid measures of licensee performance is a complex and controversial process. Measures considered to date include licensees' compliance records, evaluations of licensees by NRC inspectors, and detailed trend analysis of reportable licensee events.

## NRC Operations Center

The NRC Operations Center was activated on three occasions during 1979. This center is the focal point for NRC's initial response to significant incidents involving NRC-licensed activities. The 2,000 square-foot center presently in use includes: a conference room for briefing NRC management; an operations room for monitoring and evaluating information about the incident; a secure communications room; word processing and computer support areas; and a library to house necessary information resources. The center is equipped with a specially-designed communications system and a variety of audiovisual aids.

The first activation occurred in January as a result of an extortion threat against the General Electric Fuel Fabrication Facility in Wilmington, North Carolina. A letter demanded money for return of stolen uranium or the extortionist claimed he would disperse the material in an unnamed U.S. city. Although the Federal Bureau of Investigation (FBI) had the lead in the case, the NRC was concerned about the possible radiological consequences of the threatened act and provided technical support to the FBI. In this case, the FBI quickly apprehended a suspect and located the stolen material.

The other major incident involved the NRC response to the Three Mile Island accident. The center did function as a major focal point for the NRC, as intended, but the limited facilities were quickly overextended during this event. As a result of TMI and increased emphasis on responding to future incidents, major revisions to the NRC incident response program will be made.

The third incident for which the Operations Center was activated occurred in October when a release of radioactive gases from the Prairie Island nuclear plant took place.

The Operations Center is manned 24 hours-per-day by a qualified senior engineer.

# INVESTIGATIONS

An important adjunct to NRC's inspection effort is the investigative program which covers not only in-depth probes of irregularities revealed during inspections, but also investigations of incidents, accidents, allegations or any unusual circumstances occurring at or related to NRC-licensed facilities or activities. A heightened public awareness and interest in nuclear power has resulted in an increase in the number of allegations received by NRC. As each allegation must be carefully investigated to determine its possible impact upon the public health and safety, NRC has more than doubled the number of trained investigators in its employ within the past year.

Investigations are conducted by experienced investigative personnel located in each of the five NRC regional offices. Investigators are assigned to the immediate staff of the regional director, both to emphasize the importance of the investigative program and to provide better support to the various functional branches in the region. Since NRC investigations are usually technical in nature and may involve several scientific or engineering disciplines, the investigator frequently works with and coordinates the activities of technical personnel who may be assigned to provide assistance. Investigators also maintain close liaison with Federal, State and local law enforcement agencies and work closely with them on investigations of mutual interest. Within the past year, IE investigators have provided assistance to agencies having primary jurisdiction in investigations involving the theft of special nuclear material, the intentional damaging of fuel elements at an operating nuclear power plant, the attempted bombing of a nuclear power station, and the falsification of records relied upon by NRC.

## Table 5. Enforcement Orders—Fiscal Year 1979

| *License* | *Date* | *Reason* |
|---|---|---|
| Radioassay Systems, Inc. Southfield, Michigan (Materials Licensee) | 11/30/78 | Order terminating proceedings. Reason: Licensee disposed of all material and requested termination of the licensee. On 7/13/78, the licensee was issued an Order to show cause for processing and distributing material without authorization. |
| Arkansas Power & Light Company Little Rock, Arkansas (Arkansas Nuclear One Unit 1) | 6/15/79 | Order authorizing resumption of operations. Reason: Licensee satisfied the conditions of the 6/2/79 Order. |
| Public Service of Indiana Plainfield, Indiana (Marble Hill Units 1 & 2) | 8/15/79 | Order confirming suspension of construction. Reason: Serious problems with respect to the adequacy of concrete placement and the licensee's quality assurance program. |

Oversight of the NRC investigations program is accomplished by a small investigative staff located at headquarters. During fiscal year 1979, 121 investigations were conducted by inspection and enforcement personnel. Of these, 76 were prompted by allegations dealing with reactor construction or operational events at licensed facilities. Other investigations were conducted into events involving loss or theft of licensed material, overexposures, and general public interest. In 78 of the investigations, licensees were cited for failure to meet NRC requirements.

Significant special investigations conducted during the year are described below.

### Wolf Creek Generating Station

The Wolf Creek Generating Station of the Kansas Gas and Electric Company is located in east-central Kansas in Coffey County. The site is approximately 50 miles south of Topeka and three miles northeast of Burlington, Kansas.

On March 15, 1978, the licensee reported to the NRC that concrete samples tested for compressive strength at the age of 90 days had not all met the specified 5,000 lbs.-per-square-inch (psi) design strength. The samples represented 6,600 cubic yards of concrete placed in a continuous two-day operation to construct the reactor containment building base mat. The licensee initiated a series of studies to determine the cause of the low strength in samples and to determine whether the base mat met the construction permit criteria. The licensee concluded, in a final report in October 1978, that the base mat was acceptable and met specifications for 5,000 psi compressive strength concrete on the basis of supplemental tests. The apparent low-strength samples were attributed to faulty testing procedures by the licensee.

The NRC initiated an investigation that resulted in the licensee's halting the placement of all safety-related concrete on December 19, 1978. Numerous deficiencies which could have contributed to the apparent low-strength concrete were found, as well as quality assurance problems. The NRC concluded that the faulty testing was not the cause of the apparent low-strength samples. Additional studies were initiated by the licensee. The NRC retained an independent consultant and a test laboratory to provide additional information independent of the licensee.

While this work was underway, voids were found in the containment wall in two locations when forms were removed in December 1978. These defects and their causes were reviewed by IE inspectors. Repair was subsequently accomplished, utilizing approved procedures.

On March 6, 1979, after changes related to quality assurance had been made, the licensee was permitted to resume the placement of concrete in safety-related structures except for the reactor containment building. It was not until July 12, 1979 that concrete placement was permitted in the containment, because of the unresolved questions concerning the base mat. Placement was allowed as a result of reanalyses of the reactor containment building by the licensee using the lowered concrete strength values as the actual as-built strength of concrete. The reanalysis showed that enough margin remained in the design to accommodate the low-strength concrete, since the Wolf Creek unit is one of a series of standardized plants which are designed for more severe site conditions than exist at Wolf Creek. As a result of the studies and investigations, greater assurance has been obtained that the structure will perform adequately.

## Marble Hill Nuclear Generating Station

The Marble Hill Nuclear Generating Station of Public Service of Indiana is located in southeastern Indiana in Jefferson County. The site is approximately nine miles northeast of Milton, Ky.

Beginning in April 1979, a series of noncompliances associated with concrete construction were identified by IE inspectors. In May, the NRC met with the licensee to request additional information on in-place concrete. Many of the noncompliances were attributed to inadequate implementation of quality assurance/control programs by the licensee and his contractor. In June, a series of allegations related to concrete construction were made by a former worker at the site. These allegations indicated that voids in the concrete had been found but not properly reported nor properly repaired.

An NRC investigation, with the aid of the worker, found additional areas which were deficient because of voids. In June, the licensee agreed to stop safety-related concrete work until certain QA actions were completed to the NRC's satisfaction. On the basis of observation by IE inspectors of a large non-safety-related concrete placement, safety-related concrete work was allowed to resume.

In July 1979, the National Board of Boiler and Pressure Vessel Inspectors reported several deficiencies at the site (not related to concrete) and recommended suspension of the utility's American Society of Mechanical Engineers Owner's Certificate for apparent Code violations of Section III, Division 1, of the ASME Boiler and Pressure Vessel Code. As a result of further investigation by IE inspectors, which identified contruction management problems, an order confirming the suspension of work was issued on August 14, 1979. A series of corrective actions must be completed before construction of safety-related items will be allowed to resume.

## McGuire Nuclear Plant

The McGuire Nuclear Plant of Duke Power Company is located 17 miles northwest of Charlotte, N.C., adjacent to the Catawba River.

In March 1978, the NRC received telephone calls and a letter from an individual regarding alleged safety problems at the McGuire facility. A meeting of NRC staff with the individual resulted in reduction of the concerns to 12 allegations. IE investigators worked on the case through July and were able to resolve all but one allegation. This one allegation pertained to calculations completed by Duke Power Company to ascertain whether a fuel cask could fall into the spent fuel pool under various hypothetical circumstances. Additional investigatory effort identified a conservative calculation—not previously shown to the NRC—which showed that the cask could enter the spent fuel pool. The license has taken corrective action to prevent the occurrence of such an event.

## Midland Nuclear Plant

The Midland Nuclear Plant, which is owned by Consumers Power Co., is located just south of Midland, Michigan, adjacent to a large industrial complex of the Dow Chemical Co.

In September 1978 the licensee reported greater than expected settlements had occurred in the diesel generator building complex. IE investigations disclosed that many of the commitments the licensee had made at the construction permit stage had been revised without changes in the safety analysis reports. Matters related to revised criteria and remedial action were transferred to NRC's Office of Nuclear Reactor Regulation (NRR).

Joint IE-NRR efforts are still underway to define what corrective measures need to be taken.

## Surry Nuclear Power Station

The Surry Nuclear Power Station, which is owned by the Virginia Electric and Power Company (VEPCO), is located about eight miles south of Williamsburg, Va.

On May 7, 1979, while conducting inspections of new fuel for Surry Unit 2, the licensee found that plastic protective liners on 62 of 64 nuclear fuel assemblies had been tampered with. Further inspection revealed that a white crystalline substance had been poured on the assemblies. Preliminary analysis by VEPCO indicated that the substance was sodium hydroxide. The new fuel is stored in a building which is locked and alarmed, and to which access is controlled by the issuance of specially coded access cards.

Investigation of this incident by the FBI culminated in the surrender of two VEPCO employees to Surry County authorities on June 19, 1979. Charges of breaking and entering with intent to damage electrical facilities (felony) and willful destruction of utility company equipment (felony) were lodged against the two employees. Two additional charges against both men were introduced for conspiracy regarding the two felonies, for a total of four felonies and one misdemeanor against each man. These charges were filed on behalf of the Commonwealth of Virginia, since current Federal statutes do not provide penalties for such acts of vandalism. Pending legislation may result in making intentional damage to a reactor facility a Federal crime.

The two individuals, who were later tried, convicted on charges of willfully destroying utility company equipment and sentenced to two years' imprisonment, claimed that they had damaged the fuel rods to call attention to poor security practices and unsafe conditions at the VEPCO facility. Subsequent to their trial, they were interviewed by NRC investigators and an investigation into their allegations is currently underway.

## Abnormal Occurrences—Fiscal Year 1979

An "abnormal occurrence" is defined in Section 208 of the Energy Reorganization Act of 1974 as "an unscheduled incident or event which the Nuclear Regulatory Commission determines to be significant from the standpoint of public health or safety." The same Act requires that such events be reported quarterly to the Congress by the NRC and also be included in the Annual Report. The four quarterly reports covering fiscal year 1979 are published as NUREG-0900, Vol. 1, No. 4, and Vol. 2, Nos. 1, 2 and 3, and are available from the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 and from the National Technical Information Service, Springfield, Va. 22161.

In reports on the first three quarters of the fiscal year, eight abnormal occurrences were covered, including the accident at Three Mile Island (see Chapter 2). Three additional events were under consideration for reporting in the fourth quarter (July—September 1979) but had not been officially identified as abnormal occurrences at the end of the report period and are, therefore, omitted in the listing below. Abnormal occurrences which took place during the fiscal year at facilities under the jurisdiction of Agreement States are treated in Chapter 8, "State Programs."

**Loss of Containment Integrity.** The occurrence involved a loss of containment integrity at two nuclear power plants—Millstone Unit 2 and Salem Unit 1—reported in July and September of 1978 respectively. The issue is discussed in Chapter 3, under "Mechanical Operability of Containment Purge Valves," in the section, "Other Technical Issues."

**Electrical System Deficiencies.** The occurrence concerned degraded engineered safety features at the Arkansas Nuclear One site, involving both Units 1 and 2 there, and disclosed serious deficiencies in electrical distribution system operation and design. It was reported in September 1978 and is treated in NUREG-0090, Vol. 2., No. 1

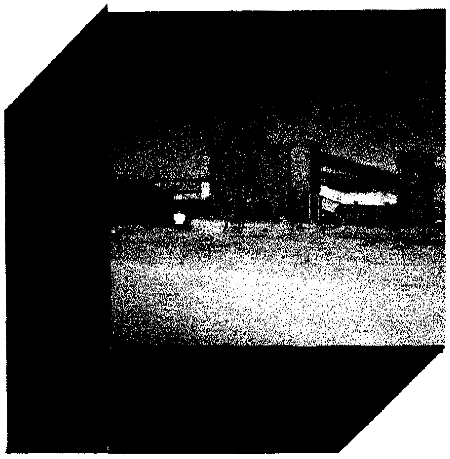
**Piping Reanalysis at Five Plants.** The occurrence derived from the discovery that certain piping systems and pipe suports in five nuclear plants had been constructed according to a faulty calculation. The issue is covered in Chapter 3, under "Shutdown and Seismic Reanalysis of Five Operating Reactors," in the section, "Other Technical Issues."

**Extortion Attempt.** The occurrence arose from an extortion attempt in the form of an anonymous letter sent to officials of the General Electric Company's fuel fabrication facility at Wilmington, N.C., alleging that the sender was in possession of an amount of low enriched uranium oxide and threatening to send portions of it to various persons and to release the materal in certain cities if payment was not made. The extortionist was apprehended and the material recovered. (See Chapter 5, under "Safeguards Events—Fiscal Year 1979.)

**Loss of Feedwater Transient.** The occurrence took place on May 2, 1979, at the Oyster Creek facility, where a loss of feedwater transient resulted in a significant reduction of the water inventory above the reactor core area, as measured by one set of water level instruments. It was later determined that the water level had fallen below the safety limit, but that no part of the core was uncovered and no fuel damage occurred.

**Vandalizing New Fuel Rods.** The occurrence involved the pouring of sodium hydroxide on new fuel assemblies and is discussed above under the heading "Surry Nuclear Power Station."

**Emergency Feedwater Unavailable.** In June 1979 at Unit 1 of Arkansas Nuclear One, an NRC inspector found that, as preparations were made for startup of the facility the controls for the emergency feedwater system were so positioned that the system could not automatically respond if needed. It was later ascertained that there was no procedural requirement that the system status be checked before startup. The plant was returned to cold shutdown for 12 days, until procedures could be reexamined and revised. All holders of reactor operating licenses and construction permits were informed of the event and its implications.



# 8

# State Programs

NRC and State officials dealt with mill tailings matters throughout 1979.

The Three Mile Island nuclear power plant accident in Pennsylvania focused increasing interest by the States on most areas of NRC activity, including reactor regulation, emergency preparedness, and waste disposal. While NRC's contacts with the States are far ranging and involve activities of many of the agency's offices as well as the Commission itself, the principal responsibility for NRC/State interaction is centered in the Office of State Programs.

Highlights of fiscal year 1979 included the negotiation of memoranda of understanding with Indiana and Nebraska, regional meetings with State liaison officers in NRC Regions I and II, a decision to place State liaison officers in all NRC regions, NRC concurrence in five more State plans for response to radiological emergencies, and several regional workshops to develop a more explicit policy for nuclear facility decommissioning.

NRC/State activities discussed in this chapter include (a) the State Agreements Program, under which NRC relinquishes to qualified States the authority to regulate certain kinds and quantities of nuclear materials; (b) assistance to State and local governments in radiological emergency response planning; and (c) cooperative activities regarding NRC responsibilities affecting the States such as licensing, decommissioning, waste management, and transportation of radioactive materials.

## STATE AGREEMENTS PROGRAM

Section 274 of the Atomic Energy Act of 1954, as amended, authorizes the Commission to enter into agreements providing for the assumption by qualified States of regulatory responsibility over byproduct and source material and small quantities of special nuclear material. At the end of 1979, there were 25 Agreement States exercising regulatory authority over some 11,800 nuclear material licenses: Alabama, Arizona, Arkansas, California, Colorado, Florida, Georgia, Idaho, Kansas, Kentucky, Louisiana, Maryland, Mississippi, Nebraska, Nevada, New Hampshire, New Mexico, New York, North Carolina, North Dakota, Oregon, South Carolina, Tennessee, Texas, and Washington. (An agreement concluded in late 1979 with the State of Rhode Island became effective on January 1, 1980.)

### Review of State Regulatory Programs

The NRC conducts a formal annual review of each Agreement State's radiation control program to determine whether it is adequate to protect the public health and safety and is compatible with NRC's regulatory program. The annual reviews assess the State's organization, administration, staffing, regulations, licensing, and compliance functions for the program. Field evaluations of State inspectors are also made. During fiscal year 1979, the NRC conducted 29 such program reviews and one followup review. NRC staff accompanied State inspectors at a number of licensed facilities, including four State-licensed uranium mills.

### Adequacy and Compatibility Findings

During calendar year 1978, NRC found that all 25 Agreement State radiation control programs were adequate to protect public health and safety. The NRC staff did, however, recommend a follow-up review of the Florida program because of a recurring high inspection backlog and staff shortages.

With respect to the compatibility of Agreement State programs with NRC's regulatory program, NRC determined that 23 of the 25 States had compatible programs in calendar year 1978; however, the pro-

NRC staff members meet periodically with representatives of the National Association of State Directors for Disaster Preparedness, the U.S. Civil Defense Council, and the Conference of State Radiation Control to seek their views and assistance in developing and improving Federal programs for radiological emergency response preparedness.

grams of Nevada and New Mexico were found not fully compatible. Compatibility findings for these two States were deferred because they had not adopted regulations fully equivalent to those of the NRC dealing with requirements for notices, instructions, and reports by licensees to workers (10 CFR Part 19 of NRC regulations).

## NRC Technical Assistance

NRC provides technical assistance to the Agreement States in areas such as major licensing actions, health physics, environmental analyses, review of proposed regulations, and guidance for inspection and enforcement actions. NRC is assisting Kansas with the review of a proposed low-level waste repository, and Washington in connection with renewal of a low-level waste disposal license. Nevada asked for and received NRC assistance regarding waste shipments to the burial site at Beatty. New York received NRC help in its review of an environmental report from a manufacturer of devices containing tritium. Texas, New Mexico, Colorado, Washington, and Arizona are receiving NRC assistance connected with uranium milling operations, and NRC is giving a great deal of technical assistance to Arizona in a case involving excessive radioactivity released from a plant manufacturing devices containing tritium.

## Training Offered by NRC

State regulatory personnel regularly attend NRC-sponsored courses to upgrade technical and administrative skills. This training is available to both Agreement and non-Agreement State personnel at no cost.

The following training courses were presented in fiscal year 1979: Safety Aspects of Industrial Radiography at Louisiana State University; Medical Uses of Radionuclides at Baylor College of Medicine in Texas; Health Physics and Radiation Protection at Oak Ridge Associated Universities; Inspection Procedures at NRC Region III Offices; Calibration of Teletherapy Machines at M. D. Anderson Hospital; and Orientation in Regulatory Practices and Procedures at NRC Headquarters. In all, 138 State personnel received 292 student-weeks of training during the year.

## Annual Meeting

The Agreement State radiation control program directors meet each year at NRC headquarters to discuss a wide range of topics. The October 1978 agenda covered the transportation of radioactive materials, high- and low-level waste, generally licensed devices, occupational ALARA (maintaining a level of employees' exposure to radioactivity which is "as low as reasonably achievable"), decommissioning, offshore radiography, and uranium mills. The meeting produced recommendations from State representatives to the NRC about the uniform Federal regulation of all naturally occurring and accelerator-produced radioactive material, as well as "agreement" material; the registration of industrial radiographers; resolution of the nation's waste disposal problem; and NRC's regulations pertaining to transportation.

## Agreement States and Mill Tailings

The Uranium Mill Tailings Radiation Control Act of 1978 requires Agreement States which wish to continue regulating uranium mills and tailings after November 8, 1981, to adopt Federal technical standards and procedures, including the preparation of written analyses, and the provision of opportunities for hearings and public participation in the processing of license applications for these facilities. The Agreement States regulate more than half of the active uranium mills and have a number of abandoned tailings piles within their borders. In conformance with the legislation, NRC will be negotiating amendments to existing agreements with those States that currently license uranium mills and wish to continue such regulation. These agreements, plus technical and financial support in the form of grants, will result in uniform regulation of uranium mills across the nation. (See also Chapter 6.)

## Agreement State Abnormal Occurrences

In early 1977 the Commission directed that abnormal occurrences taking place at facilities of Agreement State licensees should be included in the quarterly report to the Congress (see Chapter 7, "Abnormal Occurrences-Fiscal Year 1979"). The criteria applied in determining that an event at an Agreement State licensee's facility is an abnormal occurrence are the same as those applied to NRC licensees.

During the first three quarters of fiscal year 1979 (October 1978 through June 1979), a total of four abnormal occurrences were reported to Congress which took place in Agreement States. Two additional events were under consideration for reporting in the fourth quarter but had not been officially identified as abnormal occurrences at the end of the report period.

**NRC staff and State representatives exchange information and ideas at the Annual Agreement States meeting held in Silver Spring, Md., in October 1979. The discussion included issues on emergency response, transportation and waste management, regulation of uranium mills and environmental review, and siting of nuclear power plants.**

Two of the occurrences reported from Agreement States took place in the first quarter of fiscal year 1979 and are covered in the quarterly report to Congress, NUREG-0900, Vol. 1 No. 4. One of these involved the overexposure of a radiographer's assistant in Louisiana and the other the transportation of a package of radioactive material whose radiation emission after packaging exceeded limits set out in the license of the sender.

During the third quarter (covered in NUREG-0900, Vol. 2, No. 2), two more events were reported from Agreement States as abnormal occurrences. On March 9, 1979, the Arizona Atomic Energy Commission found several items of noncompliance in the operations of a State licensee engaged in making and distributing to authorized persons various signs and devices using tritium as an activating agent. An unannounced inspection on May 7 revealed continued noncompliance and also the presence of tritium in food prepared in a facility near the licensee for a number of schools in the area. The level of tritium exceeded the EPA standard for tritium concentration in liquids by 180 percent. The company was directed by the State to decommission operations, the tritium on the premises was sealed up and, by order of the Governor, removed to a U.S. Army facility leased for the purpose.

In California, a State licensee was conducting radiography activities at a manufacturing plant on May 22, 1979. The radiographer failed to notice that the radioactive source in his instrument had become disconnected. It was found on the floor by a plant employee who put it in his hip pocket. He later passed it on to another employee of the plant and a number of others also handled the source before it was retrieved by the radiographer. The radiographer did not inform the nine people who had been exposed to the source of its radioactivity and the attendant dangers, nor did he report the incident to either his own or the client's management. The employee who had picked up and pocketed the source was later hospitalized and required surgical repair of ulcerated skin. It is estimated that he had received a dose on the skin surface of 1.5 million rem. Others exposed to the source received radiation doses in the thousands of rem to their hands, and several incurred radiation burns. The State suspended the radiography firm's license and instituted a State Board of Inquiry to investigate the matter. NRC alerted all radiography licensees to the event and to the importance of the training of radiographers, of their performing radiation surveys, and of their promptly notifying responsible management in the event of accidental exposures to radiation.

## EMERGENCY PREPAREDNESS

### Emergency Response Planning

The accident at Three Mile Island (TMI) has greatly intensified interest in emergency preparedness on the part of the public, the Congress, and the NRC. In the past, State and local government efforts in this field have been largely voluntary. There was no requirement that States do such planning either by law or by rule, and no sanction could be visited upon a State or locality which chose to neglect or ignore the subject. In the wake of TMI, there has been widespread recognition that too little attention had been paid to emergency preparedness in the past and that much more time, effort and money must be devoted to it in the future by NRC, other Federal agencies, State and local governments, and the nuclear utilities. In the future, the present voluntary system for reviewing State and local plans may well, and probably will, be replaced by a more formal system, based on legislation or regulations, or both.

**The Procedure.** The responsibilities of Federal agencies for assisting State and local governments in developing plans for responding to radiological emergencies were outlined in a *Federal Register* notice of December 24, 1975, promulgated by the former Federal Preparedness Agency (FPA) of the General Services Administration. The notice, entitled "Radiological Incident Emergency Response Planning; Fixed Facilities and Transportation," gave the "lead agency" role to NRC, while assigning specific support responsibilities to the Environmental Protection Agency (EPA); the Department of Energy (DOE); the Department of Transportation (DOT); the Department of Health, Education and Welfare (HEW); the Defense Civil Preparedness Agency (DCPA); and the Federal Disaster Assistance Administration (FDAA) of the Department of Housing and Urban Development. Under powers granted him by the Congress, President Carter combined three of these agencies (FPA, DCPA and FDAA) into a new Federal Emergency Management Agency (FEMA) on July 15, 1979.

In his statement of December 7, 1979, responding to the report of the President's Commission on the Accident at Three Mile Island, President Carter directed that FEMA: "(1) take the lead in off-site emergency planning and response; (2) complete by June 1980 the review of State emergency plans in those states with operating reactors; (3) complete as soon as possible the review of state emergency plans in those states with plants scheduled for operation in the near future; (4) develop and issue an updated series of interagency assignments which would delineate respective agency capabilities and responsibilities and clearly define procedures for coordination and direction for both emergency planning and response; (5) assure that DOE resources and capabilities for responding to radiological emergencies are made available and augmented as needed to service civilian related radiological emergencies; and (6) assure the development of programs to address the recommendations for additional research and public education needs."

NRC is cooperating fully with all of these efforts of the new agency (see Chapter 1 and Chapter 2).

**Concurrence in State Plans.** Six State plans received NRC concurrence in 1979, bringing to 14 the number of State plans so approved.

## Planning Guidance to States

NRC has been working with the EPA to determine the types of accidents for which radiological emergency plans should be developed by State and local governments. A draft report on this subject (NUREG-0396/EPA 520/1-78-016) was completed by the NRC/EPA Task Force on Emergency Planning and issued for public comment in December 1978. The task force concluded there was no specific accident sequence that could be used for emergency planning because each accident could have different consequences, both in nature and degree. Instead, the task force developed recommendations in an alternative form which would provide State and local governments with a basis on which to formulate emergency plans. The planning basis selected involves a variety of accident consequences. The planning distances, time characteristics, and radiological release characteristics specified in the report provide guidance that scopes the emergency planning effort.

The fundamental recommendation in the NRC/EPA task force report is that Emergency Planning Zones (EPZs) be established around each nuclear power plant for purposes of emergency planning, and that an EPZ of about 10 miles in radius be established for the plume exposure pathway and a second concentric EPZ of about 50 miles in radius be established for the ingestion exposure pathway (milk and agricultural products).

The final report was published for public comment on December 15, 1978. The original 90-day comment period was extended to May 15, 1979 as a result of the Three Mile Island accident. The task force recommendations were submitted to the Commission in July 1979, and Commission action is expected early in fiscal year 1980.

**At DOE's Nevada Test Site, NRC sponsors training in radiological emergency response operations for State and local government personnel who are or may be members of response teams during emergencies. Above, students conduct a survey of contamination resulting from a simulated ground spill, while a faculty member acts as a news correspondent. Below, students "suit up" before entering a contaminated area.**

## Training Program for States

Several years ago, in cooperation with the States and other Federal agencies, NRC identified a number of areas where training was needed for State and local government personnel involved in radiological emergency planning and preparedness. Three training courses are now being offered. Courses dealing with radioactive materials in transit will be developed by DOT during fiscal year 1980, and courses in the medical area are being considered. FEMA is planning courses for "first-at-the-scene" personnel.

The following training is offered free of charge to qualified State and local government personnel:

(1) *Radiological Emergency Response Operations:* This course is now conducted routinely at DOE's Nevada Test Site. It is designed for personnel who are, or will be, assigned to State or local radiological emergency response teams. Sixteen sessions were conducted during fiscal year 1979 for 320 State and local government employees. Eighty Federal employees received training in the same program.

(2) *Radiological Emergency Response Coordination:* This course is designed to help the State radiological emergency response coordinator make decisions on what protective actions to take in the event of an accidental release of radioactive material to the environment from a nuclear facility. The course is conducted on request by the States.

(3) *Radiological Emergency Response Planning:* This course was developed to provide training needed for State and local radiological emergency response planners, and is conducted on request.

(4) *Handling Radioactive Material in Transportation Accidents:* Through the interagency program described in the December 24, 1975 *Federal Register* notice, and in cooperation with NRC, the DOT developed an 8-hour training course on handling radioactive material in transportation accidents. The course is a self-contained package consisting of slides and taped narratives and a student workbook. One package will be made available free of charge to all States by DOT, and NRC and DOT plan to make it available to many local jurisdictions.

## Field Assistance Program

NRC continues to lead and coordinate Federal interagency field reviews of State radiological emergency response plans and critiques of exercises to test these plans. During fiscal year 1979, the regional advisory committees made 35 field review and assistance visits and critiqued 12 radiological emergency response exercises.

## TMI Activities

Like many offices within NRC, the Office of State Programs' staff spent considerable time on Three Mile Island (TMI) activities and subsequent followups. In the early stages of the TMI accident, six health physicists from the Agreement States Program went to the site to assist in a variety of tasks, including environmental sampling, communications, and direct health physics technical support to the State of Pennsylvania. This entire NRC activity is covered in Chapter 2 and in other reports. It is important to note that, as a result of the accident, many States which previously were not actively pursuing concurrence in their radiological emergency response plans are now actively seeking such concurrence. Many meetings were held with States; office personnel testified at several State and Congressional hearings on the subject; and plans and schedules were made to concur in plans of 18 additional States by May 1980. To help with this new workload, personnel were temporarily assigned to the Office of State Programs from the Office of Nuclear Reactor Regulation, and temporary employees and consultants were acquired.

## GAO Report

The General Accounting Office (GAO) published a report March 30, 1979, entitled "Areas Around Nuclear Facilities Should be Better Prepared for Radiological Emergencies." The report made recommendations to the Secretaries of Defense and Energy, the Director of the Federal Emergency Management Agency, and the Chairman of the NRC.

The GAO recommended that no nuclear power plant be allowed to begin operations until State and local emergency response plans contain all the Commission's essential planning elements, and that licensees make arrangements for State and local agency participation in annual emergency drills. The Commission responded that NRC is committed to having effective, tested emergency plans wherever needed and as early as possible, and that the proposed licensing requirement dealing with plans and exercises will be included in an expedited NRC rulemaking procedure.

The GAO recommended that NRC establish the 10-mile emergency planning zone around all nuclear power plants. The Commission has endorsed this concept, as previously mentioned.

The GAO recommended that there be a requirement for people living near nuclear facilities to be given information about the potential hazard, the emergency actions planned, and the proper course of action in case of a radiological release. The Commission response said that action will be taken to implement this recommendation in connection with NRC's ongoing assessment of regulatory requirements.

## Other Emergency Response Activities

(1) Under a contract with DOE, Sandia Laboratories is developing a set of accident scenarios which can be used to test nuclear facility, State and local government emergency plans.

(2) To answer the need for improved emergency planning guidance in the event of transportation accidents involving radioactive materials, an NRC/DOT task force will be established in early 1980 to deal with the subject.

(3) A large step was taken in 1979 to provide more uniformity in reviewing and concurring in State/local plans. At a national meeting of Federal regional personnel involved in the

review process, acceptance criteria were developed for each of the essential elements required for concurrence. These criteria will be used to judge the adequacy of individual elements. Such a system eliminates much of the subjectivity involved in differing interpretations of what constitutes acceptability. The criteria are intended for use by both planners and reviewers.

(4) A draft report called "Beyond Defense In Depth" (NUREG-0553) was published in March 1979. It is a study of the costs of developing and implementing State and local emergency response plans, which are particularly acute at the local government level. It also discusses several methods of funding such plans and recommends that additional funds for emergency planning by State and local governments be raised through the imposition of additional fees on licensees and on applicants for NRC licenses. The final report will be published for public comment. The NRC staff plans to make formal recommendations to the Commission and to the new FEMA concerning the funding problem and possible solutions to it.

## LIAISON AND COOPERATIVE ACTIVITIES

### Transportation Surveillance

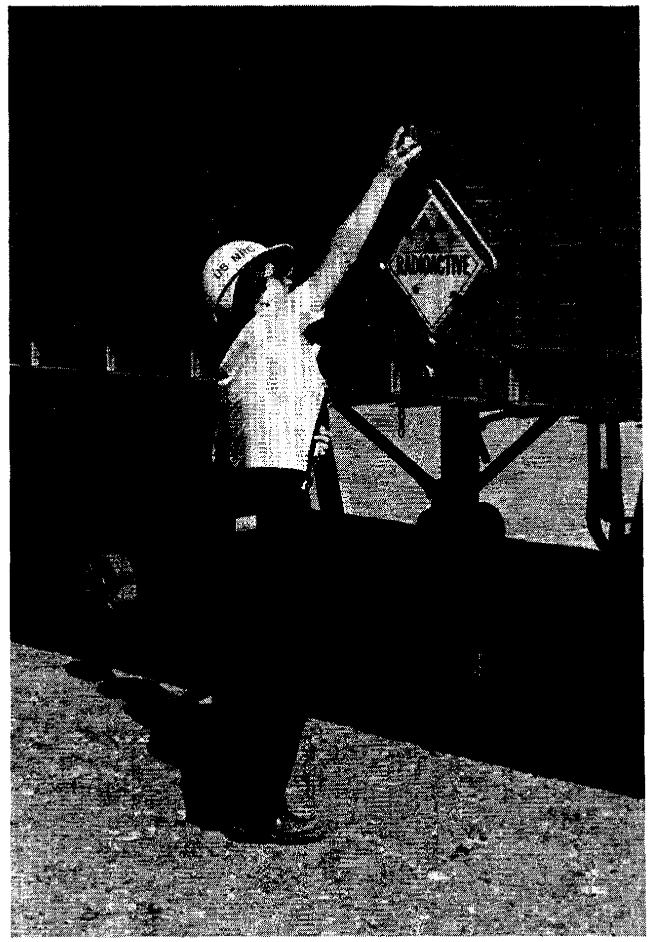
During fiscal year 1979, seven States participated in the NRC/DOT program for the surveillance of radioactive material transported into, within or through their borders. Georgia, Illinois, Michigan, and South Carolina completed 2 years of monitoring. The first-year results of the Illinois program (for the period June 1977 to June 1978) and the Georgia program (August 1977 to September 1978) were published as NUREG/CR-0756 and -0931, respectively. Kentucky will complete its first year of monitoring in December 1979. Washington and Florida began their programs in September.

The program contributes valuable data concerning all aspects of transportation in the respective States; promotes greater familiarity with Federal and State regulations on the part of shippers, carriers, and State personnel; and results in closer adherence to the regulation, thus safeguarding the health and safety of transportation workers and the general public.

### Memorandums of Agreement

In January 1976, NRC and EPA entered into a second memorandum of understanding regarding their respective responsibilities under the Federal Water Pollution Control Act Amendments of 1972 (FWPCA). NRC encourages agreements with States to whom EPA has delegated the National Pollutant Discharge Elimination System (NPDES) permitting authority under section 402 of the FWPCA.

In the recent past, NRC entered into understandings with Virginia, New York, South Carolina and Washington. During fiscal year 1979, NRC concluded memorandums of understanding with Indiana and Nebraska. Discussions continue with several other States.




**An NRC radiation specialist checks a trailer carrying low-level radioactive waste materials. Shipments such as this one from Three Mile Island are checked frequently to ensure that radiation is within safe limits.**

### State Liaison Officers Program

The Governors of all States have appointed liaison officers to maintain direct communication with NRC. There are now a total of 51 State liaison officers to the NRC, from the 50 States as well as the Commonwealth of Puerto Rico.

Regional State Liaison Officers' meetings were held in NRC Region I in October 1978 in King of Prussia, Pennsylvania, and in Region II in December 1978 in Atlanta, to acquaint the States with the NRC regional office operations and to discuss such major issues as the transportation of radioactive waste, decommissioning, waste management, emergency planning and notification. Meetings are also planned for NRC Regions IV and V during the fiscal year 1980.

A pilot program by NRC to place its own State liaison officers in the Philadelphia and San Francisco regional offices was begun by the Commission in 1977. After evaluating the program in July 1979, NRC approved full implementation of this program to all regions. During fiscal year 1980, NRC expects to assign liaison officers to the other three regions.

## National/State Organizations

Throughout 1979, NRC engaged in cooperative efforts with regional bodies such as the Western Interstate Energy Board, and with national State organizations such as the National Governors' Association, National Conference of State Legislatures, National Association of Attorneys General, National Association of Counties, and National Association of Regulatory Utility Commissioners. NRC staff also met with State legislators several times during the year to discuss NRC's programs on radioactive waste management, decommissioning of nuclear facilities and radiological emergency response planning.

There was a great deal of legislation concerned with nuclear power before State legislatures during the 1979 legislative session. Numerous bills dealt with radioactive waste, transportation, emergency response planning, and conditioning nuclear power plant siting to solution of the waste problems. NRC continued to provide comments on proposed legislation when requested and in several instances presented testimony before legislative committees.

## Conference of Radiation Control Program Directors

Together with the Food and Drug Administration and Bureau of Radiological Health of HEW, and the Office of Radiation Programs of the Environmental Protec-tion Agency, the NRC continued its financial and technical assistance to the Conference of Radiation Control Program Directors, Inc. The Conference, composed of the heads of State and major municipal radiological health programs, seeks to promote and coordinate State and local radiological health activities. A major topic of the Conference's 1979 meeting was the sharing of Federal and State experiences as a result of the Three Mile Island accident.

The Interorganizational Committee for Radiological Emergency Response Planning and Preparedness—comprised of representatives of the Conference of Radiation Control Program Directors, the National Association of State Directors for Disaster Preparedness, and the U.S. Civil Defense Council—continued to review Federal guidance publications, training, field assistance and other Federal emergency planning and preparedness efforts, and NRC will continue to look to this group to provide the State and local government comments and suggestions.

## Intergovernmental Personnel Assignments

During 1979 NRC professional employees were serving with the States of Arizona, Georgia, New Mexico and Oregon under the provisions of the Intergovernmental Personnel Act of 1970. Under the same program, a staff member of the New Mexico Environmental Improvement Division was detailed to the NRC.

**The first shipment of low-level radioactive waste from the damaged Three Mile Island Unit 2 arrives at the Nuclear Engineering Company's (NECO) Hanford, Washington, disposal site. Inspecting the load are Beth A. Riedlinger, a radiation specialist from NRC region V office (on step ladder), David Jenkins, Special Assistant to the Governor of Washington (right), and a NECO employee.**

NRC served as host agency for the civil defense director of Westchester County, New York, as part of the Intergovernmental Affairs Fellowship Program. He was assigned by the Office of Personnel Management to the NRC Office of State Programs to help improve radiological emergency response planning guidance for county governments and to assist in setting up an integrated emergency response plan for the five counties surrounding the Indian Point nuclear power facility in the State of New York.

## Workshops

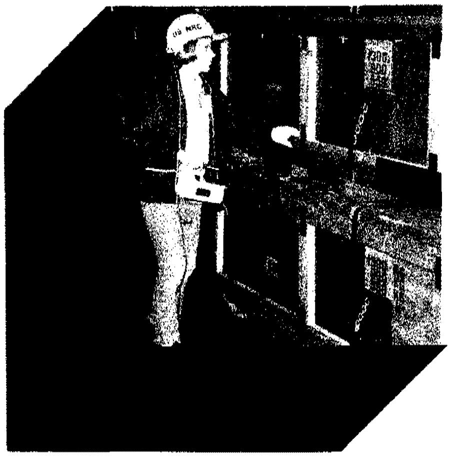
NRC has found State personnel a valuable source of help in the development and, more particularly, the review of policy and regulations. Experience has shown that the regional workshop is a most effective tool in this process.

For some time, NRC has been engaged in the development of a more explicit policy for nuclear facility decommissioning. In September 1978, NRC held a set of regional workshops to review with State officials the specifics of the NRC plan. The plan was modified in response to State comments, reissued, and sent to all the States for review. Followup workshops to discuss (1) the technical studies completed following the 1978 workshops, (2) a preliminary draft of a generic environmental impact statement, and (3) draft proposed rule changes, were held in September 1979 in Columbia, South Carolina, and in Seattle, Washington. These workshops were attended by a broad cross-section of State officials, including legislators as well as energy policy, siting, economic, regulatory, and radiation control officials. More than 150 State officials from 44 States participated.

## Three Mile Island Waste Shipments

The Office of State Programs provides advance notice of radioactive waste shipments from Three Mile Island to all States through which the shipments are routed (the low-level waste is bound for the Hanford, Washington, low-level waste burial site).

Information on the shipments, such as nature of waste, carrier, radiation levels, etc., is conveyed immediately to the State liaison officer and the State radiation control program director in each of the States involved in the routing.



# 9 International Activities

**NRC inspector checks containers destined for overseas shipment for possible escape of radiation.**

The NRC's international activities encompass formal exchanges of information and cooperation with other countries regarding radiological health and safety, administration of nuclear export and import licensing, efforts to deter nuclear proliferation, and the closely related area of international nuclear safeguards.

During fiscal year 1979, the NRC:

- Executed four new agreements with other countries and international organizations for the conduct of experiments or exchanges of information in the nuclear safety research area.
- Maintained bilateral arrangements for regulatory information exchange and cooperation with 17 countries and began or continued negotiations for similar arrangements with seven other countries.
- Held policy and technical meetings with 571 visitors from 32 countries and four international organizations. The increase of nearly 40 percent in the number of foreign visitors during the year was due mainly to concern over the accident at the Three Mile Island nuclear plant in Pennsylvania.
- Issued 678 nuclear export licenses, of which 154 were major licenses, and received 709 new export license applications.
- In its role of implementing the Nuclear Non-Proliferation Act of 1978, reviewed and provided NRC views to the Executive Branch on 13 requests from four countries for approval of retransfer of U.S.-origin spent nuclear fuel to other countries for reprocessing, and consulted with Executive Branch agencies on several cases involving export of technology associated with the production of special nuclear material outside the United States.
- Continued to support domestic and international efforts to develop and operate the nuclear fuel cycle in ways that minimize the risks of nuclear proliferation.
- Worked closely with the Executive Branch to assist the International Atomic Energy Agency in strengthening international safeguards.
- Issued revised proposed regulations to implement the US/IAEA Safeguards Agreement when ratified by the Senate. The Agreement provides for the application to U.S. civil nuclear facilities of IAEA safeguards.

## Information Exchanges

### Bilateral Arrangements

The NRC has entered into regulatory information exchanges and cooperation arrangements with regulatory bodies of 17 countries since 1974: Belgium, Brazil, Denmark, France, the Federal Republic of Germany, Greece, Iran, Israel, Italy, Japan, Korea, the Netherlands, Spain, Sweden, Switzerland, Taiwan, and the United Kingdom. While no new arrangements were concluded during fiscal year 1979, negotiations were either begun or continued with regulatory bodies in Canada, Egypt, Finland, Mexico, the Philippines, Yugoslavia, and the Union of Soviet Socialist Republics.

The objectives of the arrangements are to:

(1) Establish a formal channel of communication with foreign regulatory organizations to assure prompt and reciprocal notification of reactor safety problems that could apply to both U.S. and foreign nuclear facilities.

(2) Form a network for bilateral cooperation related to public health and safety, safeguards, and environmental protection.

(3) Assist in developing an international consensus on regulatory matters and safety standards and experiments.

(4) Provide assistance in improving nuclear health and safety practices of countries importing U.S. reactors.

Provisions of the arrangements typically call for the reciprocal exchange of regulatory information in the form of technical reports, correspondence, newsletters, meetings, training courses, and any other means agreed upon. In some cases, they also provide for future cooperation in reactor safety research and temporary assignments of personnel to agency headquarters and laboratory programs under the sponsorship of both parties.

Such arrangements are effective for 5 years, but may be extended by mutual written consent. Five of NRC's arrangements (those with Japan, France, Spain, Sweden, and Switzerland) were either renewed or were in the process of being renewed in 1979.

## New Research Agreements

During 1979, the NRC executed four new agreements involving nuclear safety research. A tripartite administrative agreement was concluded between the Bundesminister Für Forschung and Technologie of the Federal Republic of Germany (FRG), the Joint Research Center of the Commission of the European Communities, and the NRC to collaborate on a series of molten salt-water experiments. This program will provide data for confirming theoretical models used in the analysis of possible steam explosions in light water reactors.

A 5-year agreement was concluded with the European Atomic Energy Community (EURATOM) covering a broad exchange of research information on both light water and fast reactor topics.

Two research agreements were concluded with the Federal Office of Energy of Switzerland (EAW). The first agreement provides for the exchange of information and EAW's participation in the NRC's Loss-of-Fluid Test (LOFT) program and reciprocal activities

In October 1979, a delegation from the Federal Republic of Germany (FRG), headed by Dr. Volker Hauff, Federal Minister for Research and Technology (far right), visited the NRC to discuss waste management and other nuclear safety topics with Commissioner John Ahearne (far left) and Dr. Joseph D. Lafleur, Deputy Director, Office of International Programs. NRC maintains close working relations with West German government agencies, through bilateral agreements for cooperation and research information exchange, as well as through continuous contacts within the International Atomic Energy Agency.


The accident at Three Mile Island in March 1979 resulted in many requests to visit the site from foreign scientists and engineers. The NRC staff arranged for such visits and accompanied the visitors on several occasions. One such group is shown here about to enter the Middletown, Pa., National Guard Armory for a briefing on the TMI situation.

by NRC in the emergency core cooling thermal-hydraulic experimental program and the loss-of-coolant accident analysis program of the Swiss Federal Institute of Reactor Research. The second agreement provides for EAW participation in the NRC's Heavy Section Steel Technology program in return for NRC participation in the fracture mechanics research program in Switzerland. Each of the agreements has a term of 4 years.

Additionally, the NRC renewed its participation in the Halden Reactor Project for a period of 3 years. The Halden project is sponsored by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD) and is conducted by the Norwegian Institut for Atomenergie. The program covers fuel reliability and safety, fuel performance modeling, and the use of process computers in plant control.

## Increase in Foreign Visitors

During fiscal year 1979, the NRC held policy and technical meetings with large delegations and individual visitors from foreign countries and organizations totaling 571 persons from 32 countries and four international bodies. These included several 2- to 3-day discussions with foreign administrators of information and cooperation agreements with NRC as well as with their designated representatives regarding operational safety, safeguards, and environmental protection. Some visits included tours of U.S. nuclear facilities and national laboratories to observe NRC safety activities and research programs.

An increase of nearly 40 percent in the number of foreign visitors over the total for the preceding year was due to concern over the Three Mile Island nuclear power plant accident. Most of the visitors were from countries with which NRC has bilateral, regulatory and safety, and research arrangements. Numerous delegations of reactor specialists were escorted to the TMI site for discussions with NRC operating personnel.

A number of regulatory officials from other countries participated with the NRC staff in ongoing programs to gain experience in the U.S. regulatory process and to contribute their expertise to various tasks over periods of 1 month to 1 year. Participants in fiscal year 1979 were from Brazil, Israel, Korea, Mexico, the Philippines, and Turkey.

## IAEA Nuclear Safety Program

The NRC assisted the IAEA in augmenting its nuclear safety program following the accident at Three Mile Island. The expanded IAEA program, which was approved by the IAEA Board of Governors in June 1979, involves the addition of IAEA staff specialists to work with developing countries on reactor safety matters, additional safety guidance to be prepared under the IAEA nuclear reactor safety standards program, special meetings to exchange information on significant safety incidents at power reactors, and increased efforts on emergency plans and assistance. The NRC has provided, without cost to the IAEA, the services of a senior staff member for one year to assist in providing reactor safety advice to the regulatory authorities of developing countries with nuclear power programs.

NRC and IAEA will co-sponsor a meeting early in 1980 to discuss the actions being taken in the U.S. and other countries in response to the Three Mile Island accident. Preliminary briefings on the accident were given by the NRC staff to international representatives visiting the U.S. in April and May 1979.

### Technical Assistance through IAEA

In coordination with the IAEA Technical Assistance Program, the NRC continued to provide safety advice and assistance to regulatory authorities of countries embarking on nuclear power programs.

Several short-term reactor safety missions to support the Brazilian National Nuclear Energy Committee, begun in 1978 on behalf of the IAEA, were completed in 1979. The IAEA has requested NRC's continued assistance to Brazil in the areas of operator licensing review, criticality preparations, power ascension, and health physics. Two short-term missions were carried out by NRC staff experts to assist the National Nuclear Safety and Safeguards Commission of Mexico regarding concrete and containment systems.

A safeguards training program for Korea, begun in 1978, was completed in 1979 and developed into a model program for use by any country that may request NRC assistance in strengthening its state system of material accounting and control.

Also, the NRC staff has made available two experts for 1-year assignments as advisors under IAEA auspices to countries undertaking a strengthening of their nuclear regulatory programs. One has been assigned to the Philippines to advise on quality assurance and structural engineering, and the other to Mexico to advise on licensing reviews.

**Training Courses Held.** NRC staff members presented lectures as part of three courses conducted in 1979 on behalf of the IAEA by Argonne National Laboratory's Center for Educational Affairs. These included:

- A 4-week course in safety analysis review, held in Seoul, Korea.
- A 1-month course on nuclear power plant siting, at Argonne Laboratory.
- A 6-week interregional training course at Argonne on safety and reliability in nuclear plant operation, open to candidates from developing countries.

The NRC also provided lectures for an IAEA course at the Karlsruhe Research Center in the Federal Republic of Germany.

### Cooperation with the OECD

NRC is represented on several committees of the OECD's Nuclear Energy Agency, with major effort centered in the Committee on the Safety of Nuclear Installations (CSNI) and its Licensing Subcommittee. NRC helped plan and carry out the CSNI response to the accident at Three Mile Island, including efforts toward establishing an effective system for reporting information concerning safety-significant reactor incidents occurring in any of the 24 member countries. NRC also played a prominent role in an October 1979 Specialists Meeting of Regulatory Review in the Licensing Process, held in Madrid, Spain, and co-sponsored by the CSNI Licensing Subcommittee and the Spanish Junta de Energia Nuclear.

NRC senior staff also participated in activities of the NEA standing committees on Radiation Protection and Public Health and on Waste Management, and of the NEA Ad Hoc Group for the Study of Administrative and Financial Aspects of Long-Term Management of Radioactive Waste.

## Export/Import Matters And Nonproliferation

The NRC continues to perform a vital role in developing, supporting, assessing, and implementing U.S. nonproliferation policy. The Nuclear Nonproliferation Act of 1978 (NNPA) provides a policy framework for the discharge of responsibility by the NRC and the Executive Branch in (1) ensuring that nuclear export activities are conducted promptly and in conformance with national security and specific criteria set forth in the Act, (2) strengthening IAEA safeguards, (3) improving physical protection measures, (4) improving nuclear fuel assurances to other countries, (5) renegotiating Agreements for Cooperation, (6) evaluating alternative fuel cycles, and (7) formulating spent fuel disposition policy.

In addition to NRC's direct export licensing activities, the NNPA requires Executive Branch agencies to consult formally with NRC on nuclear export-related activities under their purview, including:

- Negotiation of new and revised agreements for cooperation (State Department and Department of Energy).
- Nuclear technology exports (DOE).
- Foreign distribution of nuclear material (DOE).
- Negotiation of contracts for the supply of nuclear materials and equipment (including enrichment services to foreign recipients) (DOE).
- Consideration of requests to retransfer U.S.-supplied nuclear material and facilities (DOE).

- Consideration of requests to reprocess irradiated U.S.-supplied nuclear fuel (DOE).
- Other "subsequent arrangements" as defined in section 131 of the Atomic Energy Act of 1954, as amended.
- Exports of nuclear-related commodities by the Department of Commerce.

The NRC's nonproliferation role became increasingly visible during the year, both in overall activities in the international sphere and as a member of the interagency Subgroup on Nuclear Export Coordination (SNEC) which considers many significant or controversial nuclear export matters to facilitate appropriate actions. (NRC participates in SNEC only in an observer capacity when the Executive Branch is formulating its position on individual export license applications filed with NRC.)

## Retransfers for Reprocessing

In its role as a member of SNEC, the NRC in fiscal year 1979 provided its views on 13 requests to retransfer U.S.-supplied nuclear material to other countries for reprocessing: four from Spain, five from Japan, two from Switzerland, and two from Sweden.

While the NRC generally has not opposed such requests, the Commission has stressed strict adherence to NNPA requirements and the Presidential criteria promulgated in the TEPCO and Kansai cases (see 1978 NRC Annual Report, p. 150). It was stated in these cases that such requests could be approved based on the existence of a reprocessing contract that antedates April 1977, or based on the physical need to remove the spent fuel from congested storage pools.

In its review of such requests, the Commission has expressed its concerns to DOE and has sought to encourage development of the overall U.S. policy regarding deferral of reprocessing activities. Upon completion of the International Nuclear Fuel Cycle Evaluation (see discussion later in this chapter) in early 1980, the NRC anticipates active participation in establishing new policy and approval guidelines in this regard.

## Technology Transfers

The NNPA requires the Executive Branch agencies to consult with NRC regarding exports of technology involving directly or indirectly the production of special nuclear material outside the United States, which require specific authorization by the Secretary of Energy and are controlled by DOE under its regulations in 10 CFR Part 810. Under Part 810, all exports of such technology to communist countries or exports related to reprocessing, enrichment, heavy water production, and plutonium fuel fabrication to any destination require specific authorization unless the information is publicly available in published form.

During 1979, the NRC considered several such cases, including the transfer of enrichment technology from West Germany to Brazil and the sale of CANDU reactor components to Romania. NRC is encouraging the Executive Branch to develop an overall policy on the export of laser isotope separation technology.

## Agreements for Cooperation

The renegotiation of agreements for nuclear cooperation, called for in the NNPA, continued in 1979. A principal feature of the renegotiation process has been to make reciprocal for both the U.S. and some of its trading partners the provisions regarding physical security and the storage, retransfer, and reprocesssing of spent fuel.

Under the lead of the Department of State, and in consultation with other U.S. agencies, including the NRC, renegotiated agreements have been concluded with Australia and the IAEA. Agreements with Indonesia, Sweden, Norway, Finland, Peru, Morocco, Columbia, Korea, Japan, and Egypt, were in various stages of completion at the end of the fiscal year.

## Nuclear Fuel Cycle Evaluations

The NRC continued to participate in support of activities associated with both international and domestic evaluations of nuclear fuel cycle systems aimed at reducing proliferation risks. These programs are:

- The International Nuclear Fuel Cycle Evaluation (INFCE), an effort begun in October 1977 and due to end in March 1980 in which 53 countries and four international organizations are evaluating means of developing and operating the nuclear fuel cycle in a manner to minimize the risks of proliferation.
- The Nonproliferation Alternative Systems Assessment Program (NASAP), a related U.S. effort being conducted by DOE which is providing technical data and input to INFCE.

The 1979 NRC Authorization Act (P.L. 96-601) directed the agency to monitor and assist, as requested, both the INFCE and NASAP studies and to report to Congress on their status semiannually through calendar year 1980, and annually thereafter through 1982. The first such report, covering activities through June 30, 1979, was sent to Congress in December 1979. It noted that, because the NASAP and INFCE reports had not yet been released and because of the strain on agency resources caused by the Three Mile Island nuclear plant accident, NRC was able to provide only limited analysis of the status of the evaluations in the first report. With Congressional approval, NRC had delayed much of a carryover of

$800,000 in fiscal year 1978 funds earmarked for alternative reactor and fuel cycle research in order to meet resource needs precipitated by the Three Mile Island accident.

The NRC has reviewed and provided comments on preliminary NASAP documents furnished by DOE on the six principal reactor concepts being considered: light water, heavy water, light water breeder, high temperature gas-cooled, gas-cooled fast breeder, and liquid metal fast breeder reactors. Other DOE documents reviewed included those discussing the associated fuel cycle facilities and safeguards considerations for the alternative cycles.

NRC is sponsoring technical work by outside consultants to better understand licensing issues unique to the NASAP reactor conceptual designs and to obtain information useful for establishing future programs and policies regarding safeguards for fuel cycles developed commercially. In addition to technical work supporting the alternative reactor and fuel cycle effort, NRC maintains a significant research program in fast breeder reactors and a limited research effort in advanced converter reactors.

The main proliferation concerns center on the back end of the fuel cycle—reprocessing and fabrication of recycle fuel, and waste management. All fuel cycles considered, except once-through, require reprocessing and recycling of fissile materials, and more information is needed for appropriate NRC assessment of the relative proliferation risks.

Additional information and analysis also are required to reach judgments on the relative safety, safeguardability, environmental impact, and licensability of the alternative reactor and fuel cycle concepts that have been reviewed.

## Pacific Basin Fuel Storage

The concept of storing spent nuclear fuel in centralized regional facilities as a means of enhancing energy security of countries dependent upon nuclear power and advancing shared nonproliferation objectives is being explored by the United States. Specifically, the Department of State and DOE are continuing to pursue with the government of Japan the issue of a joint feasibility study on an interim spent fuel storage facility in the Pacific Basin.

Observers from the NRC staff attended several interagency planning meetings during the year and contributed to the consideration of technical and regulatory issues raised by other agencies regarding the possibilities of Pacific Basin fuel storage. Discussions with interested Pacific Basin countries are continuing. NRC plans to offer informal staff comments and assistance to the Executive Branch agencies engaged in the study.

## NRC Views on Nonproliferation Role

Section 602 of the Nuclear Nonproliferation Act requires the Commission and DOE to include in their annual reports to the Congress "views and recommendations regarding the policies and actions of the United States to prevent proliferation which are the statutory responsibility of those agencies. . ."

Progress has been made in expediting the processing of export license applications through NRC's revised export/import regulations (10 CFR Part 110, issued in May 1978) which incorporate licensing criteria and requirements of the nonproliferation legislation. However, delays are unavoidably entailed in the review of significant export cases by at least six separate U.S. Government agencies, particularly when complex issues and differing interpretations of the applicable export review criteria are involved. Since enactment of the NNPA, the Commission has:

(1) Delegated to the staff authority to process routine low-enriched uranium fuel export license applications without referral to the Executive Branch.

(2) Established a limited policy to license several nuclear fuel reload exports at one time, thereby reducing the number of applications that need to be filed and processed.

(3) Adopted the approach, authorized by sections 126a(2) of the Atomic Energy Act of 1954, as amended, of issuing export licenses upon a finding of "no material changed circumstances" from those existing at the time the last license application for export to the same country was approved using NNPA procedures, thus eliminating repetitive analyses of some applications.

(4) Approved filing of consolidated export license applications covering more than one shipment of similar material or equipment to various countries and consignees.

(5) Streamlined internal Commission and staff reviews of exports handled both by the NRC and other agencies.

In November 1978, the NRC proposed a standard format for the Executive Branch's analyses of nuclear export license applications in an effort to reach agreement on those matters which the Commission believes should be addressed by the Executive Branch. Discussions were nearing completion at the end of 1979.

The Commission continues to believe further clarification is needed of NRC's consultation role on nuclear export matters under the purview of the Executive Branch—particularly with respect to retransfer requests involving reprocessing, on which the Commission has provided several comments to the Executive Branch.

The issues of the adequacy of IAEA safeguards applied to nuclear exports and NRC needs for more detailed information concerning safeguards implementation abroad continue to concern the Commission. During the year, the NRC worked closely with the Executive Branch in the continuing effort to improve international safeguards. (See discussion below.)

## EXPORT LICENSING ACTIONS

During the fiscal year ending September 30, 1979, the NRC issued 678 export licenses and amendments to existing licenses, and received 709 new export license applications, including requests for amendments. Of the 678 licenses issued, 154 were major licenses which are listed in the accompanying table in three categories: special nuclear material, source material, and reactors. The 524 export licenses considered to be minor included 162 for small quantities of special nuclear material, 39 for source material, 76 for byproduct material, and 247 for components. (NRC also issued 53 import licenses, including amendments , and received 43 new import license applications.)

Sixteen different nations received U.S. shipments of special nuclear material under major export licenses during the year. In addition, seven nations received major quantities of source material, and three nations received a reactor facility. No licenses were issued during the period for the export of large quantities of plutonium.

Two significant export license cases are discussed below.

### Philippines Reactor Project

A reactor export application dating from 1976 and subsequent license applications for the export of associated components and fuel for a nuclear power plant at Napot Point in the Philippines drew substantial public attention—mainly over safety—during 1979.

An application by Westinghouse Electric Corporation (XR-120) for a license to export a nuclear facility to the Philippines was filed in November 1976. In August 1978, Westinghouse filed another application for the export of several reactor components (XCOM-0013) needed to allow completion of the project, on which construction had been started. This action, which formerly would have been handled by the Department of Commerce, was transferred to the NRC under provisions of Section 309 of the Nuclear Nonproliferation Act of 1978 which transferred authority for licensing exports of nuclear components from Commerce to the NRC. In February 1979, Westinghouse also filed a license application (XSNM-1471) with NRC for the export of 121,000 kilograms of low-enriched uranium for the initial core and three reloads for the Philippines facility. The Executive Branch has completed its review of these cases and has recommended that all be approved.

In June 1979, President Marcos of the Philippines, citing concerns that the plant site may be too close to an active earthquake zone and to possibly volcanic areas and also that the Westinghouse design may be faulty, suspended construction of the nuclear plant until issues concerning the safety of the operation could be resolved. Subsequently, the Philippines government established a Commission (the "Puno Commission") to investigate these concerns. In November, the Puno Commission released its findings, which essentially concluded that the plant site was acceptable, but that serious concerns still remained regarding the safety of Westinghouse's design. Based upon these findings, President Marcos continued in force the suspension of construction activities.

On April 19, 1979, the Center for Development Policy, Jesus Nicanor P. Perlas III, and the Philippine Movement for Environmental Protection filed a petition with the Commission requesting a public hearing on the proposed export licenses. Petitioners requested that the hearing focus on issues related to environmental, health and safety impacts that the proposed reactor would have upon the Philippines, including the potential effects on U.S. citizens residing there. On October 19, 1979, the Commission issued an order inviting petitioners and other members of the public to submit written comments to the Commission addressing the issues of:

(1) Whether the Commission has jurisdiction to examine health, safety, and environmental impacts in a foreign country arising from the construction and operation of an exported nuclear reactor.

(2) Whether its health, safety or environmental review of export license applications is limited to the consideration of those issues with U.S. common defense and security implications, or whether the legal principles would permit or require the Commission to examine such matters as part of its licensing review.

(3) What issues arising from the application to export a nuclear reactor to the Philippines should the Commission examine in any future public proceeding.

(4) What procedural format should be adopted for considering such issues if they are found to lie within NRC jurisdiction.

(5) If health, safety and environmental aspects of a U.S.-supplied facility are to be evaluated in the NRC export licensing process, in what manner

# Table 1: Major Nuclear Export Licenses

(Major Licensing Actions Taken by NRC—October 1, 1978 through September 30, 1979)

SPECIAL NUCLEAR MATERIAL (One or more "effective kilograms" as defined in 10 CFR 70.4(t))

| *Licensee* | *Kilograms of Uranium* | *Enrichment %* | *Country of Destination* | *Date Issued* |
|---|---|---|---|---|
| Transnuclear | 15.976 | 93.3 | Netherlands & W. Germany | 10/13/78 |
| Transnuclear | 22.556 | 93.3 | Netherlands | 10/13/78 |
| Transnuclear | 21.053 | 93.3 | Netherlands | 10/13/78 |
| Marubeni America | Increase maximum enrichment | | Japan | 10/31/78 |
| Mitsui & Company | Increase maximum enrichment | | Japan | 10/31/78 |
| Mitsui & Company | Increase maximum enrichment | | Japan | 10/31/78 |
| Marubeni America | Increase maximum enrichment | | Japan | 10/31/78 |
| Transnuclear (rountine reload) | 14,151 | 3.35 | Netherlands | 11/02/78 |
| Westinghouse | 66,752 | 3.49 | Yugoslavia | 11/09/78 |
| Westinghouse | Extend expiration date; change licensee's address; Additional 65,939 | | Rep. of Korea | 11/17/78 |
| General Electric | 243,000 | 4.0 | Switzerland | 11/17/78 |
| | 1.07 gm | 93.5 | | |
| GETSCO | Extend expiration date | | Japan | 11/22/78 |
| General Atomic | 38.675 | 93 | Romania | 11/22/78 |
| | 44.400 | 20 | | |
| | 20 gms | 93 | | |
| GETSCO | Extend expiration date | | Switzerland | 11/22/78 |
| General Electric | Extend expiration date; delete conditions | | Spain | 11/22/78 |
| General Electric | Extend expiration date | | Italy | 11/27/78 |
| General Atomic | Additional | | Romania | 11/29/78 |
| | .245 | 93.3 | | |
| | .930 | 20.0 | | |
| Transnuclear | 35.423 | 93.30 | Japan | 12/01/78 |
| Westinghouse (routine reload) | 109,406.170 | 3.30 | Sweden | 12/07/78 |
| Transnuclear | 107,386 | 3.25 | W. Germany | 12/26/78 |
| Exxon Nuclear | Additional | | Sweden | 12/26/78 |
| | 36,443.1 | 2.90 | | |
| Exxon Nuclear | Additional | | W. Germany | 01/02/79 |
| | 50,037 | 3.20 | | |
| Transnuclear | 198 | 93.3 | Canada | 01/04/79 |
| Transnuclear | 779 | 4.05 | Canada | 02/01/79 |
| Mitsubishi | 36,783 | 3.25 | Japan | 02/02/79 |
| General Electric | 37,160 | 3.65 | Japan | 02/02/79 |
| Mitsubishi | 9,955 | 2.65 | Japan | 02/02/79 |
| Edlow International | 10,270 | 3.15 | Japan | 02/02/79 |
| Mitsubishi | 26,222 | 2.85 | Japan | 02/02/70 |
| Mitsubishi | 9,679 | 3.15 | Japan | 02/02/79 |
| Mitsubishi | 11,191 | 2.85 | Japan | 02/02/79 |
| Transnuclear | Add Intermediate Consignees; delete another | | Netherlands | 02/09/79 |
| General Electric | Increase maximum enrichment | | Spain | 02/16/79 |
| General Electric | Add party to export; change licensee's name | | Japan | 02/16/79 |
| Transnuclear | 11,584.75 | 3.55 | Switzerland | 02/16/79 |
| Transnuclear | Extend expiration date | | W. Germany Netherlands | 02/16/79 |
| Transnuclear | Extend expiration date | | Netherlands | 02/16/79 |
| Westinghouse | Increase maximum enrichment | | Spain | 02/22/79 |
| Transnuclear | 11,056 | 4.335 | France | 02/23/79 |
| Transnuclear | 34,801 | 3.25 | W. Germany | 02/23/79 |
| Transnuclear | 49,656 | 3.30 | France | 02/23/79 |
| Transnuclear | 37,160 | 3.0 | W. Germany | 02/23/79 |
| Transnuclear | 24,062 | 3.4 | W. Germany | 02/23/79 |
| Transnuclear | 22,295.905 | 3.0 | W. Germany | 02/23/79 |
| Transnuclear | 5.297 | 93.3 | Canada | 02/26/79 |

## SPECIAL NUCLEAR MATERIAL (One or more "effective kilograms" as defined in 10 CFR 70.4(t))

| *Licensee* | *Kilograms of Uranium* | *Enrichment %* | *Country of Destination* | *Date Issued* |
|---|---|---|---|---|
| Transnuclear | 10,794 | 3.40 | Belgium | 02/26/79 |
| General Electric | 377,600 | 4.0 | Mexico | 03/02/79 |
| Transnuclear | Delete Other Party to Export; add Shipper of Record | | W. Germany | 03/14/79 |
| Transnuclear | Delete Other Party to Export; add Shipper of Record | | W. Germany | 03/14/79 |
| Transnuclear | Delete Other Party to Export; add Shipper of Record | | W. Germany | 03/14/79 |
| Transnuclear | 4.900 | 93.3 | Japan | 03/14/79 |
| General Electric | Extend expiration date; delete condition | | Italy | 03/19/79 |
| Edlow International | 16,803.6 | 2.71 | India | 03/23/79 |
| Union Carbide | 7.33 | 93.16 | France | 03/26/79 |
| Westinghouse | Extend expiration date | | Brazil | 03/26/79 |
| Westinghouse | 2.48 Pu; 61 natural U | | France | 03/22/79 |
| General Electric | Change name of 2nd Ultimate Consignee | | Mexico | 04/03/79 |
| Transnuclear | 23.058 | 93.3 | France | 04/09/79 |
| Westinghouse | Remove Intermediate Consignee; extend expiration date; delete conditions; Additional | | Switzerland | 04/09/79 |
| | 23,347 | 3.55 | | |
| Transnuclear | 20.050 | 93.3 | Netherlands | 04/11/79 |
| Transnuclear | 19.8 | 93.3 | Netherlands | 04/11/79 |
| Transnuclear | 22.055 | 93.3 | Sweden | 04/11/79 |
| Transnuclear | 17.043 | 93.3 | Sweden | 04/11/79 |
| Mitsui & Company | 1,897 | 3.85 | Japan | 04/13/79 |
| Transnuclear | 22.0 | 93.3 | W. Germany | 04/17/79 |
| General Electric | Add condition | | France | 04/26/79 |
| Transnuclear | 12,714 | 3.25 | W. Germany | 04/27/79 |
| Exxon Nuclear Co. | 55,400 | 3.5 | Belgium | 04/27/79 |
| Transnuclear | 18,951 | 3.35 | W. Germany | 04/27/79 |
| Transnuclear | 10,581 | 3.4 | Belgium | 04/27/79 |
| Transnuclear | 4.23 | 93.3 | Canada | 05/08/79 |
| Edlow International | 61,750 | 3.55 | Sweden | 05/09/79 |
| Transnuclear | Delete Other Party to Export; submit another for transport only | | Japan | 05/09/79 |
| Mitsubishi | 16,341 | 3.33 | Japan | 05/10/79 |
| Exxon Nuclear | Extend expiration date | | W. Germany | 05/24/79 |
| Transnuclear | 22,090.900 | 3.35 | Switzerland | 05/29/79 |
| Transnuclear | 10,225 | 3.0 | W. Germany | 06/05/79 |
| Transnuclear | Extend expiration date | | W. Germany | 06/05/79 |
| Transnuclear | Extend expiration date | | Japan | 06/05/79 |
| General Electric | Extend expiration date | | Japan | 06/06/79 |
| General Electric | 239,000 | 4.0 | Taiwan | 06/08/79 |
| General Electric | Delete Ultimate Consignee | | Taiwan | 06/13/79 |
| Transnuclear | 14,101 | 3.25 | Sweden | 06/20/79 |
| Westinghouse | Extend expiration date; delete condition; change maximum enrichment | | Spain | 06/25/79 |
| Transnuclear | 12,283.11 | 4.3 | France | 06/25/79 |
| GETSCO | Extend expiration date | | Japan | 06/27/79 |
| General Electric | Add conditions | | France | 06/28/79 |
| Transnuclear | Add conditions | | Japan | 06/28/79 |
| Mitsubishi | 11,983 | 2.85 | Japan | 07/03/79 |
| General Electric | 11,965 | 3.1 | Japan | 07/03/79 |
| Transnuclear | 11,585.0 | 3.55 | Switzerland | 07/03/79 |
| General Electric | 5,875 | 3.1 | Japan | 07/13/79 |
| Mitsui & Company | 26,902 | 3.95 | Japan | 07/13/79 |
| Mitsui & Company | Include enriched uranium as $UO_2$ | | Japan | 07/18/79 |
| Edlow International | 13,125 | 4.52 | Italy | 07/20/79 |
| Transnuclear | 22,090.0 | 3.35 | Switzerland | 07/27/79 |
| Mitsui & Company | 157,382 | 2.55 | Japan | 08/02/79 |

# Table 1: Major Nuclear Export Licenses—Continued

**(Major Licensing Actions Taken by NRC—October 1, 1978 through September 30, 1979)**

**SPECIAL NUCLEAR MATERIAL** (One or more "effective kilograms" as defined in 10 CFR 70.4(t))

| *Licensee* | *Kilograms of Uranium* | *Enrichment %* | *Country of Destination* | *Date Issued* |
|---|---|---|---|---|
| Transnuclear | 10,794 | 3.4 | Belgium | 08/08/79 |
| General Electric | 29,526 | 3.65 | Japan | 08/13/79 |
| Marubeni America | 30,382 | 2.97 | Japan | 08/13/79 |
| General Electric | 18,452 | 3.65 | Japan | 08/13/79 |
| Marubeni America | 25,723 | 3.90 | Japan | 08/14/79 |
| Marubeni America | 3,799 | 3.90 | Japan | 08/14/79 |
| Transnuclear | 17,589 | 3.35 | Netherlands | 08/23/79 |
| Transnuclear | 3.810 | 93.3 | Austria | 08/27/79 |
| Edlow International | Extend expiration date | | Sweden | 09/05/79 |
| Edlow International | Extend expiration date | | Sweden | 09/05/79 |
| Edlow International | Extend expiration date | | Sweden | 09/05/79 |
| Mitsui & Company | 3,808 | 3.95 | Japan | 09/06/79 |
| Edlow International | Extend expiration date; delete conditions | | Sweden | 09/06/79 |
| Edlow International | Extend expiration date; delete conditions | | Sweden | 09/06/79 |
| Edlow International | 44,870 | 3.55 | Sweden | 09/06/79 |
| Westinghouse | Extend expiration date | | Sweden | 09/11/79 |
| Westinghouse | 122,220 | 3.55 | Switzerland | 09/11/79 |
| Mitsui & Company | 9,214 | 3.95 | Japan | 09/20/79 |
| | 21,237 | 3.95 | | |
| Exxon Nuclear | Extend expiration date; change licensee's address | | Sweden | 09/21/79 |

## SOURCE MATERIAL

| *Licensee* | | *Material* | *Country of Destination* | *Date Issued* |
|---|---|---|---|---|
| NL Industries | | Additional | Italy | 10/13/78 |
| | 8,164.746 | kgs. depleted uranium | | |
| RMI Company | | Extend expiration date | Canada | 10/16/78 |
| Transnuclear | 304,818 | kgs. depleted uranium | Switzerland | 11/03/78 |
| NL Industries | | Additional | | |
| | 9,072 | kgs. depleted uranium | Canada | 01/03/79 |
| Nuclear Metals, Inc. | 23,000 | kgs. depleted uranium | United Kingdom | 03/01/79 |
| Edlow International | | Extend expiration date | Canada | 03/21/79 |
| Mitsubishi | 538,511 | kgs. natural uranium | Canada | 04/09/79 |
| Army Mat'l. & Mech. | 250 | kgs. depleted uranium | United Kingdom | 05/09/79 |
| RMI Company | | Additional | Canada | 05/09/79 |
| | 90,039 | kgs. depleted uranium | | |
| Aerojet | | Additional | Canada | 05/22/79 |
| | 55,344 | kgs. depleted uranium; extend expiration date | | |
| Rhone-Poulenc | 100,335 | kgs. thorium | France | 05/24/79 |
| | 10,165 | kgs. natural uranium | | |
| Edlow International | 10,000 | kgs. depleted uranium | Rep. of China | 06/08/79 |
| NUS Corporation | 16,800 | kgs. depleted uranium | Rep. of China | 06/08/79 |
| NUS Corporation | | Delete Ultimate Consignee | Taiwan | 06/13/79 |
| Edlow International | | Delete Ultimate Consignee | Taiwan | 06/13/79 |
| Edlow International | 1,000,000 | kgs. natural uranium | United Kingdom | 06/21/79 |
| Edlow International | 129,243 | kgs. natural uranium contained in 152,410 kgs. of $U_3O_8$ | West Germany | 07/27/79 |
| Transnuclear | 125,500 | kgs. depleted uranium | France | 07/27/79 |
| | 500 | kgs. depleted uranium | | |
| U.K. Treasury & Supply Delegation | | Increase material exported to | United Kingdom | 09/25/79 |
| | 1,200 | kgs. depleted uranium | | |
| Transnuclear | | Increase quantity to | France | 09/06/79 |
| | 126,000 | kgs. depleted uranium | | |

## REACTORS

| Licensee | Facility Description | Country of Destination | Date Issued |
|---|---|---|---|
| Westinghouse | Two 2,785 MWT PWR KORI-3 and KORI-4; value of items $200,000,000 | S. Korea | 10/04/78 |
| Westinghouse | Extend expiration date; change licensee's address | S. Korea | 10/16/78 |
| GETSCO | Kaiseraugst Nuclear Power Station/BWR/2894 MWT; value $23,000,000 | Switzerland | 11/20/78 |
| GETSCO | Add Additional Authorized Exporter; delete condition | Japan | 11/27/78 |
| Westinghouse | Add Intermediate Consignee | Spain | 01/03/79 |
| GETSCO | Extend expiration date; change licensee's address | Spain | 01/23/79 |
| General Electric | Add another party to export | Mexico | 02/01/79 |
| General Electric | Add another party to export | Mexico | 02/01/79 |
| Westinghouse | Extend expiration date; increase value to $28,000,000 | Sweden | 02/02/79 |
| Westinghouse | Conform early license with recent licenses | Brazil | 02/02/79 |
| General Electric | Change addresses of Intermediate Consignees; add 2 other parties to export; delete Intermediate Consignee | Japan | 03/26/79 |
| General Atomic | Extend expiration date | Romania | 05/10/79 |
| General Electric | Change addresses of Ultimate and Intermediate Consignees | Japan | 05/14/79 |
| Westinghouse | 2785 MWt PWR Taiwan Power Co. Units 5 & 6 | Rep. of China | 06/08/79 |
| Westinghouse | Delete Ultimate Consignee | Taiwan | 06/13/79 |
| General Electric | Change address of Ultimate Consignee | Spain | 06/27/79 |
| Westinghouse | Extend expiration date | Brazil | 08/23/79 |
| General Electric | Export instruments for 2 BWRs; add other parties to export; change other addresses | Taiwan | 09/11/79 |

should the review be conducted differently from domestic reactor licensing proceedings.

(6) Whether there are any factual or legal considerations which would justify different NRC health, safety or environmental reviews for some export license applications than for others.

After reviewing the submissions received, the Commission is expected to issue a second order which may or may not result in additional proceedings on the merits of the proposed Philippine exports. (See Chapter 1 for further Commission action in January 1980.)

In a separate development in this case, in an effort to force issuance of the license for the export of the nuclear components, Westinghouse filed a suit against the NRC in August 1979. In the suit, Westinghouse argued that NRC's determination not to act on the components license until questions regarding the safety of the site are resolved constituted action unlawfully withheld and unreasonably delayed. On August 30, the U.S. District Court of the District of Columbia ruled in favor of the NRC.

## Tarapur (India) Proceeding

The NRC Annual Reports for 1976, 1977, and 1978 contain detailed discussions of the history of the Commission's deliberations on the license applications to export low-enriched uranium to India for use in the Tarapur Atomic Power Station. In March 1979, by a three to two vote, the Commission authorized issuance of license XSNM-1222 to India, covering 16,804 kilograms of low-enriched uranium. The Commission majority concluded that this license application met all the requirements for issuance under the Atomic Energy Act of 1954. The dissenting Commissioners were of the view that, because of unique features in the United States-India Agreement for Cooperation, India's failure to adopt full-scope safeguards, and the lack of additional assurances covering the eventual fate of U.S.-supplied fuel, a finding was precluded that the United States had adequate assurances that material supplied to India would be kept under IAEA safeguards, not be used to develop nuclear explosive devices, and not be reprocessed or transferred to another nation without prior U.S. approval.

In September 1978, Edlow International filed an application for the export of another 19,858.8 kilograms of low-enriched uranium to India for conversion and fabrication into fuel elements for the Tarapur Station. Favorable Executive Branch views were provided in July 1979. However, in view of subsequent developments in India the Commission has deferred action on this request until the political situation in India has stabilized sufficiently to permit an updated Executive Branch judgment concerning India's nuclear policies and intentions. An additional application (XSNM-1569) for fuel export to India was filed in August 1979. This is still under review by the Executive Branch.

## ENVIRONMENTAL EFFECTS OF EXPORTS

### Unified Interagency Procedures

On January 4, 1979, the President issued Executive Order 12114, "Environmental Effects Abroad of Major Federal Actions," effective September 4, 1979, which directed all Federal agencies to join in developing unified implementing procedures for environmental reviews of certain nuclear exports.

The Order applies to several types of major Federal actions, including: (1) actions significantly affecting the environment of the global commons (e.g., the oceans and Antarctica), (2) actions significantly affecting the environment of a foreign nation not participating in the action, (3) actions affecting the environment of a foreign nation which provide to that nation a product or physical project producing toxic emissions which are strictly regulated in the U.S. (e.g., radioactive substances), and (4) actions which significantly affect natural or ecological resources of global importance. In the nuclear area, the Order specifically applied to Executive Branch actions providing to a foreign nation a nuclear production, utilization, or waste management facility. Exports of nuclear fuel, however, are excluded from the requirements of the Order.

The Order calls for the preparation of certain types of documents to aid in assessing the significance of the environmental consequences stemming from the applicable Federal actions. A draft of proposed interagency procedures was distributed at the end of August to all Federal agencies, including the NRC, for formal review and concurrence or comment. Following approval by Executive Branch agencies, the procedures were issued on November 5, 1979 and published in the *Federal Register* on November 13. The NRC has not formally commented on the procedures.

The first export case for which the State Department has prepared an environmental document consistent with Executive Order 12114 is the reactor export to the Philippines.

### Research and Development Reactor Exports

The NRC assisted the State Department in preparing a document which examined the environmental aspects fo nuclear research and development facility exports. Its purpose was to determine, in light of the ongoing and prospective cooperation in research and development activities, whether the nature and/or extent of environmental impacts on the global commons or the U.S. from such activities require the preparation of environmental documents or impact statements pursuant to the National Environmental Policy Act of 1969. The analysis drew on evaluations already presented in other environmental assessments, particularly the Final Environmental Statement of U.S. Nuclear Power Export Activities (ERDA-1542). The analysis concluded that neither the operation of, nor potential accidents associated with, exported research and test reactors would have a significant impact on the environment of the global commons or the U.S. and that none of the impacts is likely to exceed those associated with the power reactors considered in ERDA-1542.

The report also examined the possible environmental impacts of the program for the return, temporary storage, and reprocessing of foreign spent research and test reactor fuel, and concluded similarly that no significant impacts are likely to result from this program.

# International Safeguards

International safeguards continued to draw substantial attention of the NRC in fiscal year 1979. In addition to responsibilities associated with the licensing of exports of nuclear materials and facilities, which require NRC to consider the implementation of international safeguards in recipient countries, NRC was involved during 1979 with the voluntary application of international safeguards at civil nuclear facilities in the U.S.

### US/IAEA Safeguards Agreement

The NRC devoted further attention in 1979 to activities related to the US/IAEA Safeguards Agreement, the voluntary U.S. offer to permit application of international safeguards by the International Atomic Energy Agency to civil nuclear facilities in the U.S. Under this agreement, the U.S. would provide to the IAEA safeguards information about U.S. civil nuclear facilities "not of direct national security significance." From these, the IAEA would select a number of facilities for the implementation of full safeguards inspections by IAEA inspectors. Implementation of the agreement will fulfill a 1967 Presidential offer to apply

IAEA safeguards to U.S. civil nuclear facilities in order to demonstrate to other nations—particularly the developed nonnuclear weapon states—that the application of international safeguards would not result in commercial disadvantages. The United Kingdom and France, both nuclear weapon states, have made similar voluntary offers.

The two major developments during 1979 were the Senate hearings on the proposed Safeguards Agreement and the republication for public comment of the proposed NRC regulations to implement the treaty with respect to NRC licensees. The Subcommittee on Arms Control and International Operations of the Senate Committee on Foreign Relations held hearings on the US/IAEA Safeguards Agreement on June 22. Officials from the Department of State, Arms Control and Disarmament Agency, Department of Energy, and NRC testified on the intent, scope, and likely impacts of the treaty. Representatives of the U.S. nuclear industry joined the government officials and testified in favor of the treaty. Additional hearings were held by this Subcommittee in December 1979.

Proposed Part 75 of NRC's regulations to implement the Safeguards Agreement, initially published for public comment in 1978, was republished in revised form in July 1979 to afford further opportunity for licensee and public participation. In preparing the revised proposed regulations, NRC endeavored to state the requirements as clearly as possible and to provide for licensees to take an active role, especially in the preparation of "Facility Attachments" (which define the safeguards to be applied at specific facilities), while at the same time making sure that the objectives of the Agreement are implemented effectively.

In 1979, the NRC established an internal Safeguards Agreement Implementation Group to coordinate all agency activities directed at implementing the Agreement. These activities have included an assessment of the costs that the Agreement will impose on licensees, revision of NRC/DOE nuclear material reporting forms and procedures to make them consistent with the requirements of the Agreement, and developmental work to facilitate preparation of Facility Attachments as soon as the Agreement comes into force.

## Export Licensing Information Needs

As discussed in the 1978 Annual Report (pp. 153-154), the NRC safeguards staff has identified a need for additional information on the implementation of international safeguards for use in reviewing export license applications. During fiscal year 1979, the NRC and the Department of State began developing an approach to meet the needs of the Commission which will be consistent with overall U.S. policy on international safeguards and the nonproliferation of nuclear weapons.

The IAEA Safeguards Implementation Report (SIR) for 1978, received by NRC in 1979, identified the safeguards implementation problems that existed during calendar year 1978 and the corrective activities undertaken by the IAEA. A number of the problems were unchanged from those identified in the reports for 1976 and 1977. Reasons for the persistence of these problems include the resource constraints which the IAEA faces. These constraints include both a lack of inspection personnel and equipment, and the difficulties of expanding safeguards implementation apace with the rapidly growing number of nuclear facilities subject to safeguards.

## Support of International Safeguards

During fiscal year 1979, NRC continued to work closely with the Executive Branch on a number of activities designed to assist the IAEA in strengthening international safeguards, including:

- Participation in DOE's Program for Technical Assistance to IAEA Safeguards. NRC's major contributions consisted of technical reviews of the activities and the provision of experts without cost to the IAEA.
- Providing technical assistance to a foreign country in the development of its national system of material accounting and control, and the offer of similar assistance to other countries on request.
- Working with the IAEA and Executive Branch agencies to provide a training course in the U.S. for foreign officials who are responsible for establishing and managing their countries' national systems of material control and accounting.
- Participation in the U.S. Interagency Action Plan Working Group to strengthen IAEA safeguards.
- Provision of direct technical assistance to the IAEA.

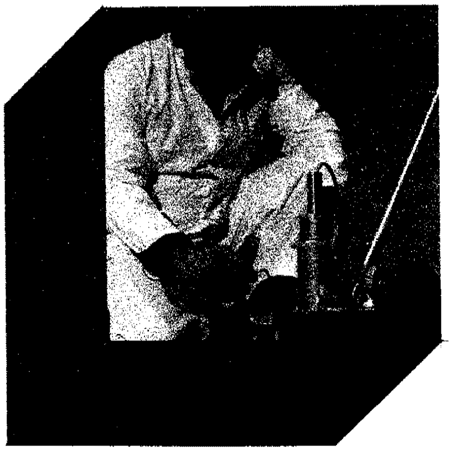
## Other Activities

Other activities related to the areas of international safeguards and physical security of nuclear materials which NRC undertook during the year included:

(1) Participation with other U.S. agencies in evaluating the safeguards aspects of technical papers prepared by International Nuclear Fuel Cycle Evaluation (INFCE) working groups. The NRC also provided experts to the INFCE "Safeguards Crosscut Group" in support of this effort.

(2) Participation in meetings, both in the U.S. and abroad, with foreign experts on international safeguards and physical security matters to exchange views and information.

(3) Assignment on a long-term basis of NRC safeguards technical experts to the IAEA staff in Vienna.
(4) Participation in the U.S. negotiating team on the Convention on the Physical Protection of Nuclear Materials. This treaty, a U.S. initiative, on which negotiations were concluded in October 1979, establishes the agreement of the international community on the appropriate levels of physical protection to be accorded to nuclear materials during international transport. The treaty also is designed to facilitate international cooperation in the physical protection of nuclear material.



# 10 Standards Development

NRC standards include specification and testing criteria for measurement devices.

NRC standards provide for protection of the public and nuclear industry workers from radiation, the safeguarding of nuclear materials and facilities from theft and sabotage, and protection of the quality of the environment in nuclear activities. Thus, the development of standards cuts across the range of the NRC's activities and requires close interaction between the Office of Standards Development and the agency's other program offices.

While many of the standards issued or worked on during fiscal year 1979 are discussed in this chapter, some are discussed elsewhere in this Annual Report under the topics to which they relate (e.g., transportation in Chapter 4 and safeguards in Chapter 5).

## CONCERNS OF HIGH PRIORITY

Current issues of high priority in standards development include:

**Standards Development After TMI.** Lessons learned as a result of the accident at Three Mile Island (TMI) will have a substantial impact on the NRC standards program for fiscal year 1979 and succeeding years. Major efforts to incorporate into NRC standards a consistent treatment of fission product release caused by fuel failure, and on improving NRC standards for emergency planning and for nuclear power plant operations, are underway. In response to recommendations from the NRC staff and from other groups that have been investigating the TMI accident, the NRC is developing new standards, revising existing standards, and working with the national standards development program to incorporate the lessons of TMI into national standards. In addition, the NRC Office of Standards Development has assigned a number of its senior staff to participate in the various TMI investigations and to assist in carrying out agency-wide activities during the post-TMI period. (See Chapter 2.)

**Degraded Core Cooling.** The TMI accident involved a condition of inadequate cooling of the reactor fuel that led to substantial fuel damage and associated release of radioactivity to the reactor coolant and into the containment. Accompanying this condition of degraded core cooling was the generation of substantial quantities of hydrogen, attributable to the reaction of the overheated fuel cladding with the reactor coolant. Furthermore, radioactivity was released into other parts of the plant and the surrounding environment. Degraded core cooling conditions in accidents do not appear to be consistently handled in NRC's regulations and guides. In view of this situation, a group has been established within the Office of Standards Development to reevaluate the regulatory requirements and guidance related to various design and operational aspects of the nuclear power plants that may be affected by degraded core cooling conditions that could occur during severe accidents. The objective of the group is to assess these regulations and regulatory guides in order to identify and implement whatever improvements may be needed to better ensure that any systems and components that may be required to function during or following a degraded core cooling situation are designed, constructed, and operated to do so.

**Emergency Planning.** A proposed rule change was published in the *Federal Register* in December 1979 that would require as a condition of operating license issuance that State and local governmental emergency response plans be submitted to and concurred in by the NRC. In addition, during the 60-day public comment period, several workshops will be held to (a) present the proposed rule changes to State and local governments, utilities, and other interested parties, and (b) obtain comments concerning the costs, impacts, and practicality of the proposed rule change. Comments from the workshops and the public will be considered prior to preparing a final rule in 1980.

## REGULATIONS AND GUIDES

NRC standards are primarily of two types:

- Regulations, setting forth in Title 10, Chapter I, of the Code of Federal Regulations requirements that must be met.
- Regulatory Guides, describing, primarily, methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations.

When a new or amended regulation is proposed, it is normally published in the *Federal Register* to allow interested citizens time for comment before final adoption, in accordance with the Administrative Procedure Act. Following the public comment period, proposed regulations are revised, as needed, to reflect the comments received. If the regulation is adopted by the NRC, it is published in the *Federal Register* in final form with the date it becomes effective. After that publication, rules are codified for inclusion in the annual publication of the Code of Federal Regulations.

Some regulatory guides delineate techniques used by the staff to evaluate specific situations. Others provide guidance to applicants concerning the information needed by the staff in its review of applications for permits and licenses. Many NRC guides refer to or endorse consensus standards (also called "national standards") that are developed by recognized national organizations, often with NRC participation. NRC makes use of a national standard in the regulatory process only after an independent review of the standard has been made by the NRC staff and after public comment on NRC's planned use of the standard has been reviewed.

The NRC encourages comments and suggestions for improvements in regulatory guides at all times, and they are revised to take account of appropriate comments and suggestions and to reflect new information or experience. In its continuing effort to provide for increased public participation in the regulatory process, the NRC instituted a new procedure for developing and issuing guides during this fiscal year. Guides are now being issued for public comment in draft form before complete staff review and before an official NRC staff position has been established.

Copies of draft regulatory guides, together with their value/impact statements, are mailed for comment to many individuals and organizations. The value/impact statement indicates the objective of the guide, its expected effectiveness compared to alternative ways of achieving the objective, and expected impacts on other safety systems, NRC operations, other Government agencies, industry, and the public.

In order to reduce the burden on the taxpayer, the NRC has made arrangements with the U.S. Government Printing Office to become a consigned sales agent for certain NRC publications. Effective November 1, 1979, regulatory guides are being included in this sales program. Draft guides, which are issued for public comment, will continue to receive free distribution. Active guides will be sold on a subscription or individual copy basis. Licensees of the NRC will receive, at no cost, pertinent draft and active guides as they are issued.

Proposed and effective regulations published during fiscal year 1979 are summarized in Appendix 4. Draft and active regulatory guides issued, revised, or withdrawn are listed in Appendix 5.

**Decommissioning.** NRC policy is being reevaluated in this area with a view toward improving standards for all nuclear facilities. Major technical studies are nearing completion on the engineering methodology, radiation risks, and estimated costs of decommissioning light water reactors and other nuclear facilities. A draft generic environmental impact statement (GEIS), to be used in developing appropriate regulations on decommissioning of nuclear facilities, is also nearing completion.

**Spent Fuel Storage.** Revision of proposed licensing requirements for independent spent fuel storage installations is underway to reflect comments received from the public and from the NRC staff. (See Chapter 4.)

**Uranium Milling and Processing.** Proposed and effective rule changes were published in August 1979 to establish specific uranium mill licensing requirements. These amendments are derived from a draft GEIS on uranium milling and the requirements contained in the Uranium Mill Tailings Radiation Control Act of 1978.

**Inservice Inspection.** The NRC is seeking to upgrade the capability of inservice reactor inspection methods to reliably detect and characterize flaws in components of the primary coolant and other safety-related systems. Contract studies are continuing, and a guide on inspection of welds in austenitic piping is under development. A draft guide has been issued on inspection of welds in pressure vessels.

**Siting Policy.** In August 1979, a Siting Policy Task Force, established by the NRC in November 1978, recommended (NUREG-0625) extensive revision of NRC's Reactor Site Criteria (10 CFR Part 100). Similarly, Congressional committees and the President's Commission have recommended amendment of the siting criteria, especially with regard to demographic factors such as population density and distribution around nuclear power plants. Action plans are being developed to implement the recommendations.

**Nuclear Power Plant Simulation.** Concern for the improvement of operator training led to the initiation of contract work with Oak Ridge National Laboratory and Memphis State University Center for Nuclear Studies to investigate the capabilities of nuclear power plant simulators within the United States. In particular, the study is investigating the feasibility of increasing the use of simulators and expanding their role in operator training. A regulatory guide to provide guidance concerning the use of simulators in operator training is planned for the near future.

**Improvement of Regulatory Guidance.** As part of the NRC's evaluation of the TMI accident, review of regulatory guides and rules has been initiated to determine necessary improvements in regulatory guidance. For example, Guide 1.8, on the qualifications of nuclear power plant personnel, and Guide 1.33, on overall quality assurance program requirements for the operational phase of nuclear power plants, were reviewed to determine areas where guidance could be improved. A notice for each of the guides was published in the *Federal Register* requesting specific comments that would improve the recommendations of the guides in these areas. In addition, work was initiated to improve the regulatory guidance presented in Guide 1.97 on instrumentation to assess plant and environs conditions during and following an accident at a nuclear power plant.

**Radiological Health.** Major NRC efforts related to the effects of low-level ionizing radiation included:

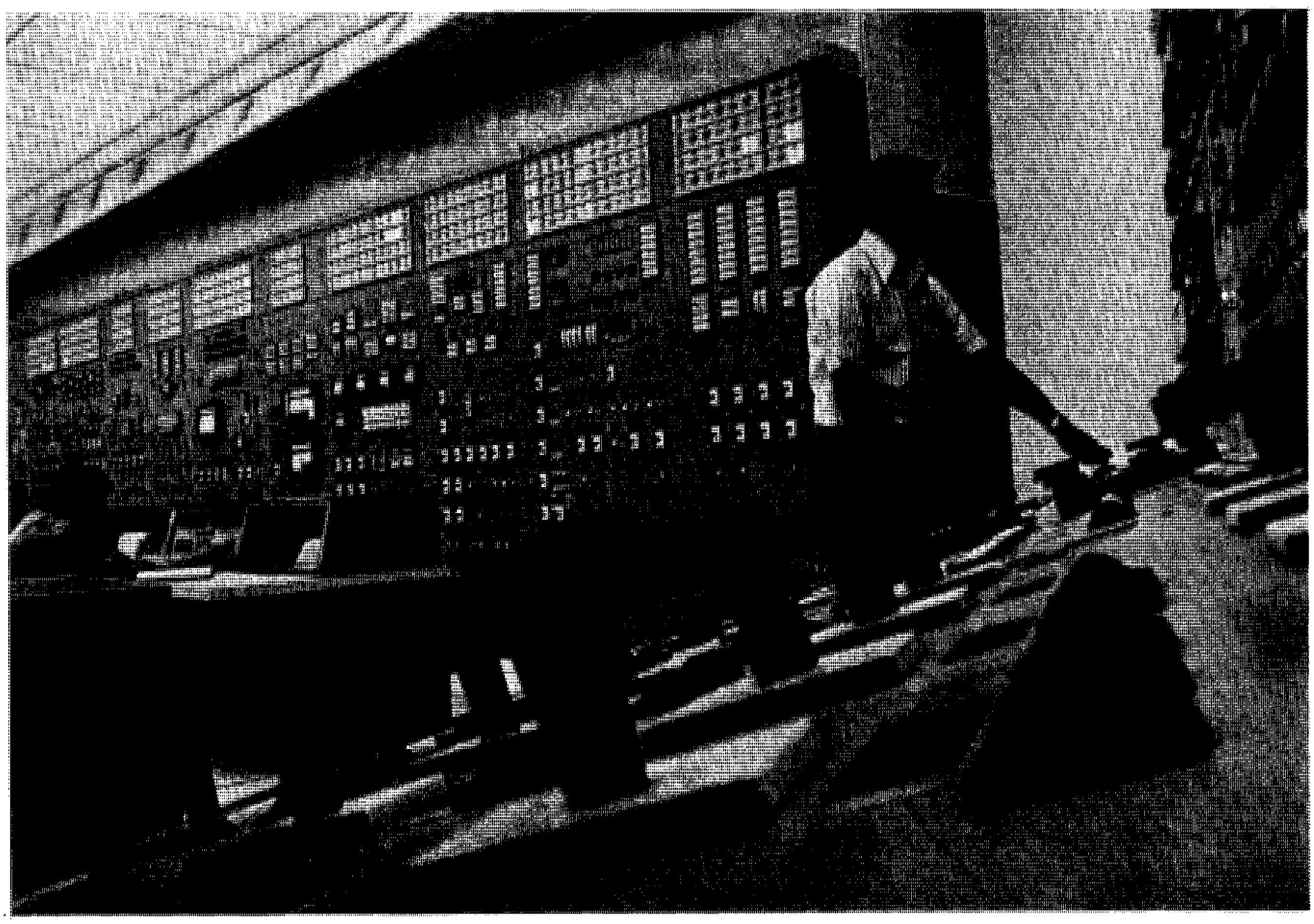
(1) A joint EPA/NRC report was sent to the Congress on the research needs, capabilities, and current programs of the two agencies with regard to health effects of ionizing radiation.

(2) A contract was awarded for the preparation of a study to identify and analyze the feasibility of options for Federal epidemiological studies of populations exposed to low-level ionizing radiation.

(3) NRC staff participated in preparing the report of HEW's (now the Department of Health and Human Services [HHS]) Interagency Task Force on the Health Effects of Ionizing Radiation, issued for comment in April and in final form in June.

(4) A Federal Interagency Task Force on Ionizing Radiation Research was formed by HEW (now HHS) with NRC participation. The NRC staff also participated as members of a Task Force subcommittee formed to evaluate and make recommendations on possible followup studies of the residents in the TMI area.

(5) NRC staff members assisted HEW (now HHS) in designing a questionnaire for a population census in the vicinity of TMI. This effort is ex-



**Standards development work initiated as the result of TMI included the preparation of new NRC guidance on the use of nuclear power plant simulators and a study on simulator capabilities across the nation. This PWR control room simulator is in use at Consolidated Edison Company's Indian Point plant in New York.**

A health physics technician prepares to conduct a contamination survey. The filter paper in her right hand is used to pick up surface dust or other material that may be contaminated. The portable survey meter carried over her shoulder is used to locate and identify sources of radioactivity. Contamination surveys are conducted frequently in facilities where fuel is handled as well as at power reactor plants.

pected to facilitate any future health effects studies that may be performed on this population.

One of the major tasks of the Interagency Task Force on the Health Effects of Ionizing Radiation was to reexamine the organization of Federal radiation protection programs. NRC staff members assisted that Task Force in preparing recommendations to the President on reorganization of Federal radiation protection and radiation research activities. Congressional hearings on this topic were held in May by a subcommittee of the Senate Committee on Governmental Affairs.

**Nuclear Medicine.** A final policy statement and rule changes provide for NRC regulation of the radiation safety of workers, the general public, and patients, but with minimal intrusion into medical judgments affecting patients. Efforts in the regulation of radiopharmacies and on licensee reports of misadministrations in this area are progressing, but these matters remain unresolved.

**Occupational Radiation Protection.** The NRC is considering rule changes to strengthen its requirements that workers' exposures to radiation be kept not only within regulatory limits but "as low as is reasonably achievable" (ALARA) and to make them more readily subject to inspection and enforcement. In addition, the NRC expects to participate in a public hearing jointly with the EPA and the Occupational Safety and Health Administration (OSHA) on new EPA occupational exposure guidance for Federal agencies, the adequacy of current occupational exposure standards, and related matters early in 1980.

**Transportation.** Sandia Laboratories continued work on assessing the environmental impacts resulting from the transportation of radioactive materials through urban areas. Results concerning the consequences of the sabotage of spent fuel shipments influenced the decision to establish interim requirements for the protection of spent fuel in transit. (See Chapters 4 and 5.)

**Safeguards.** Major staff efforts continue to be focused on (1) developing through regulations and guides a material control capability that is both timely and sensitive with respect to material loss, and (2) implementing a material access authorization program for licensees. Other matters of importance include determining the level of safeguards needed to protect SNM in the cores of nonpower reactors and publishing the final rule to implement the US/IAEA Agreement. (See Chapter 5.)

**Radiation Protection Standards.** The NRC has begun to update and restructure its radiation protection standards as contained in 10 CFR Part 20. Public comments are being sought on areas of the present regulations that need improvement.

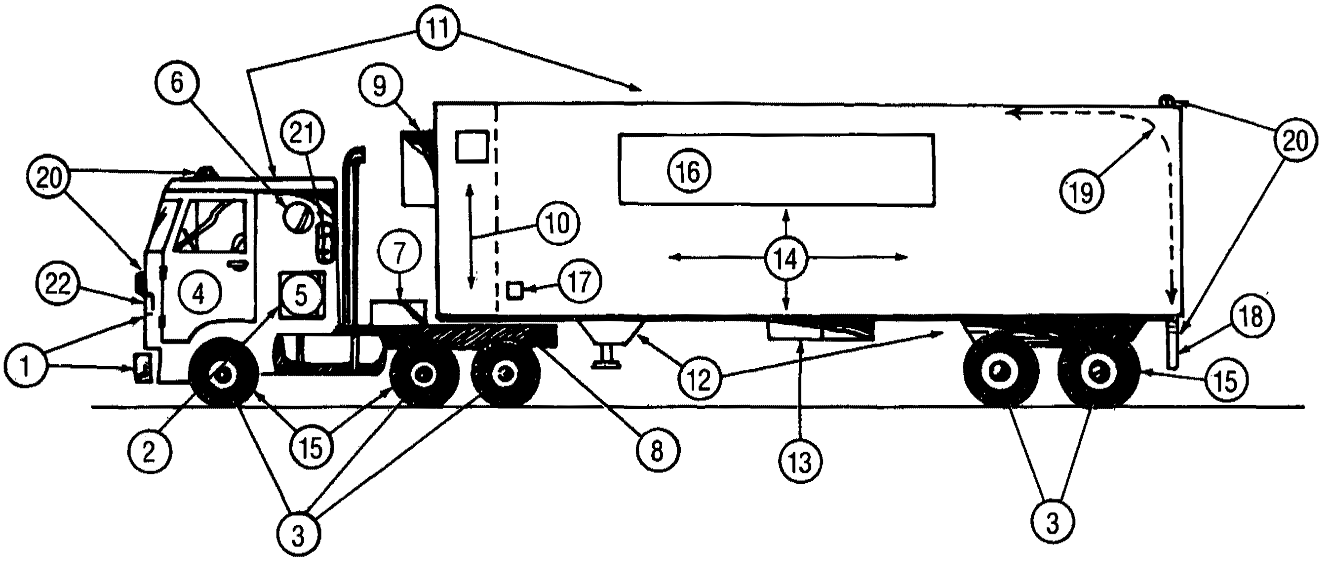
## POWER REACTOR STANDARDS

Development of power reactor standards continued during fiscal year 1979, aimed primarily at protecting the health and safety of the public and secondarily at reducing the regulatory burden.

### Surveillance and Inservice Inspection

Increased emphasis in this area brought about several changes to regulations and the issuance of two draft guides.

Section 50.55a, "Codes and Standards," of 10 CFR Part 50 has been amended to incorporate, by




1. Bumper & Grill Area
2. Engine Area
3. Tire & Wheel Assemblies
4. Cab Area
5. Baggage Compartment
6. Cab Sleeping Area
7. Battery Boxes
8. Fifth Wheel Area
9. Trailer Refrigeration Unit
10. Ice Bunker Compartment
11. Roof-Both Tractor & Trailer
12. Under Entire Tractor-Trailer
13. Spare Part & Chain Compartments
14. Interior of Trailer
15. Wheel Axles
16. Company Sign Panels
17. Document Pouches
18. Bumper on Rear of Trailer
19. Trailer-Upward Sliding Door
20. Light Lenses & Reflectors
21. Externally Mounted Air Filter
22. External Tractor Air Inlets

**The potential for concealing special nuclear material in vehicles is considered in developing safeguards standards. The diagram shown above identifies 22 places for possible concealment.**

reference, the 1977 Edition of the ASME Boiler and Pressure Vessel Code, Division 1 of Section XI, "Rules for Inservice Inspection of Nuclear Power Plants," with certain modifications, and Division 1 of Section III, "Nuclear Power Plant Components," as well as their addenda through 1978. This will result in more flexibility for inservice inspection of pipe welds in facilities under construction and in operation and will avoid potential conflict between the code and the technical specifications concerning examination requirements for steam generator tubing. This regulation was also amended to clarify certain ambiguities in the requirements for inservice inspection of nuclear power plants.

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 were revised with regard to material toughness requirements for bolts and requirements for location and method of attachment of surveillance capsules.

Two draft guides were issued for public comment. The first, issued in May 1979, deals with ultrasonic testing of welds for inservice inspection and defines some ultrasonic testing criteria considered acceptable by NRC; the second, issued in August 1979, pertains to inservice inspection code case acceptability for ASME Section XI, Division 1, and lists the ASME inspection code cases that the NRC accepts.

## Accident Analysis

The NRC is considering modifying the Emergency Core Cooling System (ECCS) Rule (Section 50.46 of 10 CFR Part 50 and Appendix K to 10 CFR Part 50) to take into account experience derived from using the rule in the licensing process, new research information, and reactor operating experience gained since the rule was implemented. In December 1978, the NRC published in the *Federal Register* an advance notice of proposed rulemaking action and invited public comments and recommendations. The NRC is currently evaluating the public comments as well as the lessons learned from the Three Mile Island accident with respect to the proposed ECCS rulemaking.

## Reactor Containment

**Containment Design.** In October 1978, the NRC published a regulation that is expected to significantly reduce the number of plants required to have inert containment atmospheres in order to prevent hydrogen explosions under certain accident conditions. This change takes account of increased conservatism in the revised emergency core cooling system requirements. Revision 2 to Guide 1.7, which describes acceptable methods of implementing the new rule, was issued in December 1978. The NRC is currently reevaluating the regulation in view of the accident at Three Mile Island.

**Concrete Containment and Structures.** NRC endorsement of the ASME Boiler and Pressure Vessel Code's Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," progressed another step with the issuance of Revision 1 in October 1978 and proposed Revision 2 in November 1979 to Guide 1.136 on the subject of materials, construction, and testing of concrete containments. Acceptance of this industry standard will make it possible to withdraw some existing regulatory guides to be covered by this standard.

In concrete containment buildings with prestressing tendons that use a grease-like coating to inhibit corrosion, periodic inspections of these tendons are required. Based on knowledge gained from these testing programs, Guide 1.35, on inservice inspection of ungrouted tendons in prestressed containments, is occasionally revised and updated. Proposed Revision 3, reflecting an increased flexibility in the testing program and a broadening of its scope, was issued in April 1979. Guide 1.35.1, on determining prestressing forces for inspection of prestressed concrete containment, was simultaneously issued for comment to clarify the NRC staff position on how to determine prestressing forces when conducting the inservice inspection program outlined in Guide 1.35.

## System and Component Criteria

**General Design Guidance.** A number of documents have been issued in this category, including Revisions 14 and 15 to Guides 1.84 and 1.85. These guides continue to list acceptable ASME Boiler and Pressure Vessel Code, Section III, Division 1, code cases as they are published. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," was issued in August 1979 when Guide 1.104, on overhead crane handling systems for nuclear power plants, was withdrawn. NUREG-0554 identifies features of the design, fabrication, installation, inspection, testing, and operating of single-failure-proof overhead crane handling systems that are used for handling critical loads.

**Guidance on Specific Systems and Components.** Guidance for the use of spray pond piping made from fiberglass-reinforced thermosetting resin was updated with the issuance of Revision 2 to Guide 1.72. Also issued was Guide 1.130, covering acceptable levels of service limits and appropriate combinations of loadings for Class 1 plate-and-shell-type component supports.

Revision 3 to Guide 1.70, providing the standard format and content of safety analysis reports for light-water-cooled nuclear power plants, was issued in November 1978.

The following guide revisions were issued in October 1979 to reflect public comments: Revision 1 to Guide 1.140, on design, testing, and maintenance criteria for normal ventilation exhaust system air filtration and adsorption units; Revision 1 to Guide 1.143, on design guidance for radioactive waste management systems, structures, and components; and Revision 1 to Guide 1.137, which describes an acceptable method for complying with NRC regulations regarding fuel-oil systems for standby diesel generators and assurance of adequate fuel-oil quality.

A draft guide on the subject of safety-related permanent dewatering systems was issued in September 1979.

A draft guide describing acceptable practices for complying with the NRC regulations with regard to the nuclear analysis and design of concrete radiation shielding for nuclear power plants was issued in February 1979.

Several technical reports developing a decommissioning information base for light water reactors were nearing completion at the end of the fiscal year. (See "Fuel Cycle Plant Standards" later in this chapter.)

## Quality Assurance

Quality assurance requirements for the design, construction, and operation of safety-related structures, systems, and components of nuclear power plants are established in Appendix B to 10 CFR Part 50. During the past fiscal year, the NRC issued new and revised guides concerning the implementation of these requirements. In January 1979, Guide 1.144, on auditing of quality assurance programs for nuclear power plants, was issued for comment. In February 1979, Revision 2 to Guide 1.28, on quality assurance program requirements for the design and construction of nuclear power plants, and a draft guide on the qualification of audit personnel in the quality assurance program for nuclear power plants were issued. Also in February 1979, proposed Revision 2 to Guide 1.8, on the qualifications of nuclear power plant personnel, was issued and, following the Three Mile Island accident, the public comment period was extended to August 1979 to encourage additional public input. The NRC requested specific comments in the following areas covered by Guide 1.8:

(1) Staffing, training, initial qualification, and requalification of operating personnel, supervisory personnel, and technical support personnel.

(2) Use of plant simulators for training, initial qualification, and requalification.

(3) Training of plant operating staff following extended shutdown.

(4) Content of programs for training nuclear power plant personnel.

(5) Use of operating experience information in training nuclear power plant personnel.

In July 1979, proposed Revision 1 to Guide 1.58, on the qualification of inspection, examination, and testing personnel for nuclear power plants, was issued. Proposed Revision 3 to Guide 1.33, on overall quality assurance program requirements for the operational phase of nuclear power plants, was issued in August 1979. The revision to Guide 1.33 was initiated prior to the Three Mile Island accident and provided improved guidance that was developed as a result of a program using feedback from the Office of Inspection and Enforcement to the Office of Standards Development. Specific comments were requested in certain areas, in addition to general comments, which would improve the recommendations of Guide 1.33 in light of the Three Mile Island accident. These particular areas are as follows:

(1) Equipment control during the operational phase, including procedural controls for conducting surveillance tests and inspections.

(2) Procedures for shift and relief turnover at operating nuclear power plants.

(3) Quality assurance for maintenance, repair, and modification activities at nuclear power plants.

(4) Preparation of operating and emergency procedures.

(5) Independence of operating plant personnel performing inspections of operating activities.

In September 1979, proposed Revision 2 to Guide 1.94, on quality assurance requirements for the installation, inspection, and testing of structural concrete and structural steel, soils, and foundations during the construction phase of nuclear power plants, was issued.

## Qualification Testing

**Electrical.** Work continued on the development of standards and guides for the qualification testing of electric equipment used in nuclear power plants. A draft guide on qualification testing of cable penetration fire stops was issued in July 1979. Proposed Revision 1 to Guide 1.131, on the qualification testing of electric cables and field splices, was issued in August 1979.

Supporting research continues at Sandia Laboratories on test source equivalence, synergistic effects in environmental qualification, and accelerated aging. Underwriters Laboratories completed the NRC-sponsored study of the adequacy of IEEE Standard 383-1979 on flammability testing.

The NRC staff continued to participate with national standards committees in developing standards for qualifying specific electric components that are important to safety—including converters, batteries, battery chargers, inverters, motor control centers, electrical connectors, and switchgear—as well as a general standard for qualifying both electric and mechanical equipment. NRC also participated in the updating of existing national qualification standards, including those for qualifying valve operators, penetration assemblies, continuous duty motors, and cables.

**Mechanical.** In February 1979, the NRC issued a draft guide on the subject of functional specification for safety-related valve assemblies in nuclear power plants. It describes a method for complying with requirements of Section III, "Design Control," of Appendix B to 10 CFR Part 50 related to the specifications for valve assemblies whose operability must be ensured.

## Electric Systems and Components

General Design Criterion 17, "Electric Power Systems," of Appendix A to 10 CFR Part 50 includes a requirement that the onsite electric power system have sufficient capacity and capability to ensure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. In November 1978, Revision 1 to Guide 1.9, related to the design and testing of diesel-generator units intended for use as onsite power sources in nuclear power plants, was issued for comment. A draft guide on the protection of electric systems and equipment from the effects of lightning was issued in August 1979.

The NRC staff continued to participate on national standards committees in developing criteria for electric systems and components important to safety, including the updating of standards pertaining to protection system design, emergency power system design, physical independence of redundant systems and equipment, and post-accident monitoring.

## Reporting Defects and Noncompliances

A rule (10 CFR Part 21) requiring certain persons to report to the NRC defects that could create a substantial safety hazard, or failures to comply with regulations relating to substantial safety hazards, became fully effective in January 1978. Following implementation of the regulation, an unintended impact occurred as a result of an interpretation of the term "basic component," as defined by the rule. Revision of the rule was completed in October 1979 to relieve the conditions resulting from that interpretation.

## Protection Against Fire

The extended one-year public comment period for Guide 1.120 ended in November 1978. This guide describes how to implement NRC's requirement that the probability and effects of fire must be minimized through fire prevention, detection, and suppression. It also provides guidelines for designing fire safety features into nuclear power plants. Public comments are being reviewed and resolved in conjunction with the earlier suggestion of the Advisory Committee on Reactor Safeguards that the staff consider a "dedicated shutdown system" instead of some of the individual fire protection items called for in the guide.

As part of the program to develop fire protection guidance for nuclear power plants, Sandia Laboratories, under NRC contract, has completed studies to develop technical bases in four separate areas of fire protection: Task 1—Ventilation; Task 2—Fire Detection; Task 3—Fire Barriers; and Task 4—Fire Hazards Analysis.

The final report for Task 1, NUREG/CR-0636, "Nuclear Power Plant Fire Protection; Ventilation (Subsystems Study Task 1)," was issued in August 1979. The report analyzes the function of ventilation during a fire emergency at a nuclear power plant, its interrelationship with other plant systems, and its role in fire control and extinguishing operations.

The final report for Task 2, NUREG/CR-0488, "Nuclear Power Plant Fire Protection: Fire Detection (Subsystems Study Task 2)," was issued in March 1979. The report examines the adequacy of fire detection in the context of overall nuclear plant safety.

The final report for Task 3, NUREG/CR-0468, "Nuclear Power Plant Fire Protection: Fire Barriers (Subsystems Study Task 3)," was issued in September 1979. The report considers the adequacy of the three-hour fire barrier requirement of Guide 1.120.

The final report for Task 4, NUREG/CR-0654, "Nuclear Power Plant Fire Protection: Fire Hazards Analysis (Subsystems Study Task 4)," was issued in September 1979. The report considers the presently available methods of conducting a fire hazards analysis and suggests using a composite of the best portions of all methods considered.

## Medical Certification

Revision 1 to Guide 1.134, which describes acceptable methods for providing the information needed by the Commission for its evaluation of the medical qualification of applicants for initial or renewal operator or senior operator licenses, was issued in March 1979.

# FUEL CYCLE PLANT STANDARDS

The NRC devoted substantial effort during fiscal year 1979 to the development of standards concerning the safety and environmental impacts of fuel cycle plants.

## Decommissioning

Technical studies for the NRC are continuing at the Battelle Pacific Northwest Laboratories (PNL) to develop a decommissioning information base for light water reactors and other nuclear facilities. This base will be used in developing appropriate regulations and guides. The PNL reports, "Decommissioning of Nuclear Facilities—An Annotated Bibliography" (NUREG/CR-0131), and "Technology, Safety and Costs of Decommissioning a Reference Small Mixed Oxide Fuel Fabrication Plant" (NUREG/CR-0129), were published in October 1978 and in February 1979, respectively. Two PNL reports, "Decommissioning Commercial Nuclear Facilities: A Review and Analysis of Current Regulations" (NUREG/CR-0671) and "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station—Addendum" (NUREG/CR-0130), were published in August 1979. Other PNL reports, "Technology, Safety and Costs of Decommissioning a Reference Low-Level Waste Burial Ground" (NUREG/CR-0570) and "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station" (NUREG/CR-0672), were nearing completion at the end of the fiscal year.

These PNL reports are part of the comprehensive reevaluation of NRC policy related to decommissioning nuclear facilities. The detailed plan and schedule for this reevaluation are described in an NRC staff report entitled "Plan for Reevaluation of NRC Policy for Decommissioning of Nuclear Facilities" (Revision 1 to NUREG/CR-0436), which was published in December 1978. Another NRC report, "Draft GEIS on Decommissioning of Nuclear Facilities" (NUREG-0586), was nearing completion at the end of the fiscal year.

Two regional workshops were held in September 1979 to review the technical reports completed during the fiscal year with State officials and to discuss with them proposed policy and rule changes. Other draft NRC reports prepared for these meetings, and also as a part of the reevaluation, were "Assuring the Availability of Funds for Decommissioning Nuclear Facilities" (NUREG-0584), "Thoughts on Regulatory Changes for Decommissioning" (NUREG-0590), and "Residual Activity Limits for Decommissioning" (NUREG-0613).

In June 1979, the NRC denied a petition by the Public Interest Research Group, et al., to initiate rulemaking now to implement a specific decommissioning funding plan in which nuclear power plant operators post surety bonds to cover decommissioning costs. To the extent that the petitioners' request asked the NRC to reconsider the adequacy of its regulations on decommissioning, their request was granted. However, a decision as to the specific method or methods for funding decommissioning was deferred until the comprehensive reevaluation of NRC policy related to the decommissioning of nuclear facilities is completed.

## Spent Fuel Storage

Staff work is nearing completion on revisions to the proposed 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation," to reflect changes made in response to comments from the public and the staff. Guide 3.44, providing the standard format and content for the safety analysis report to be included in a license application for the storage of spent fuel in an independent spent fuel storage installation (water-basin type), was issued for comment in December 1978. (See Chapter 4.)

## Uranium Milling and Processing

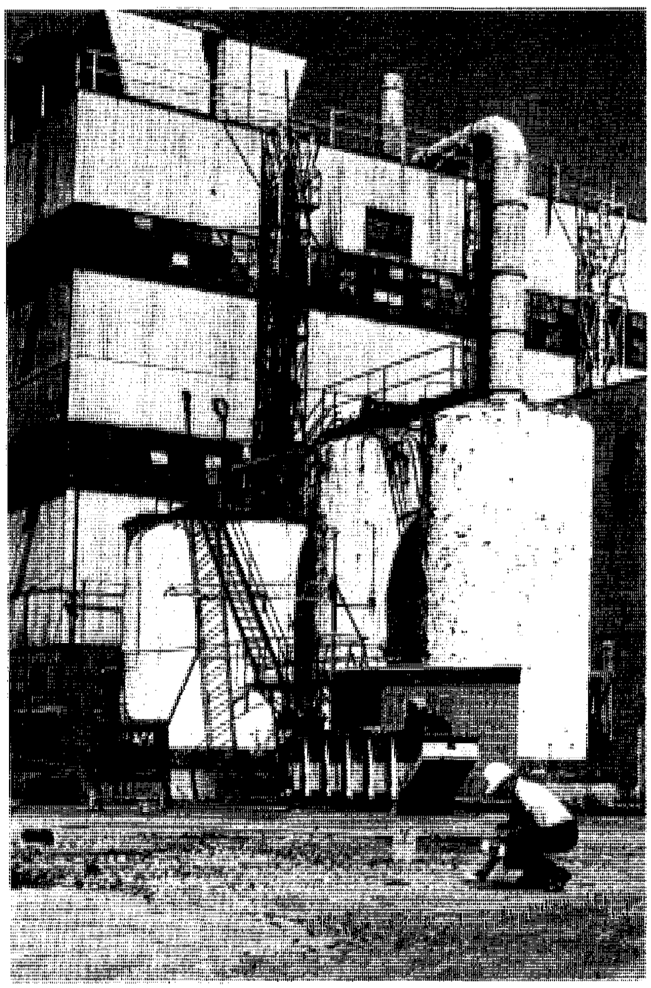
Certain definitive rule changes to the Commission's regulations have been made to establish specific uranium mill licensing requirements, particularly with regard to the tailings or wastes generated during the milling process. These rule changes incorporate into the Commission's regulations the conclusions derived from a generic environmental impact statement (GEIS) on uranium milling and the licensing requirements covering "byproduct material" set forth in the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978. In August 1979, the NRC issued effective rule changes to 10 CFR Part 40 and proposed rule changes to 10 CFR Part 170 to establish specific licensing requirements and fees for uranium mills and mill tailings (i.e., byproduct material as defined by the recently enacted UMTRCA of 1978), amendments to 10 CFR Parts 30 and 70 to require, for sake of consistency, the completion of a final environmental impact asessment prior to commencement of construction of certain types of major plants, and amendments to 10 CFR Part 150 related to byproduct materials in Agreement States.

## Nuclear Criticality Safety

The following guide revisions were issued to reflect public comments: Revision 1 to Guides 3.34 and 3.35, on assumptions used for evaluating the potential radiological consequences of accidental nuclear criticality in a uranium fuel fabrication plant and in a plutonium processing and fuel fabrication plant, respectively, in July 1979; Revision 1 to Guide 3.43, on nuclear criticality safety in the storage of fission materials, in April 1979.

## Plant Safety

Guidance in developing emergency plans for fuel cycle facilities licensed under 10 CFR Parts 50 and 70 is provided by Revision 1 to Guide 3.42, issued in September 1979. This guidance may be further revised as a result of assessment of the lessons of Three Mile Island.



**Standards for the decommissioning of fuel cycle facilities specify exhaustive inspections and tests of buildings, grounds and equipment. Here, an NRC inspector checks for radioactivity in concrete at a defunct fuel cycle facility in Chicago, Ill.**

In order to define an acceptable detailed inservice inspection program for earth and rock-fill embankments used to retain uranium mill tailings, Guide 3.11.1, on operational inspection and surveillance of embankment retention systems for uranium mills, was issued for comment in April 1979. This guide supplements existing Guide 3.11, on the design, construction, and inspection of embankment retention systems for uranium mills.

### Waste Management

During fiscal year 1979, the NRC Standards staff participated in developing policy and proposed rules for the licensing of high-level and low-level radioactive waste management facilities. A draft guide providing the standard format and content of license applications for the disposal of high-level radioactive waste in geologic repositories was nearing completion at the end of the fiscal year.

In November 1978, the NRC published for comment a proposed general statement of policy suggesting licensing procedures for geologic disposal of high-level radioactive wastes. Concurrently, a draft of an implementing regulation was circulated to the States for their review. This was done as part of NRC's policy to seek early input from the States in the waste management area. Comments on the proposed policy statement and comments from the States have been considered. The technical section of a proposed regulation (10 CFR Part 60) for geologic disposal of high-level waste is under preparation; the procedural section of this proposed regulation was issued for public comment in December 1979.

In the area of high-level waste, site suitability criteria were developed for the Technical Requirements of the proposed 10 CFR Part 60. Work was begun on branch technical positions on long-term climatic change, assessment of geochemical retardation, hydrologic assessment for transport modeling, and the use of models in site evaluation. Future work will emphasize specific requirements and guidance for site characterization in safety and environmental analysis reports.

A similar program of regulations and guides is underway to address the management of low-level radioactive wastes. Ongoing efforts include a regulation for licensed disposal of low-level radioactive wastes that addresses waste categorization. Several regulatory guides are planned in support of this regulation.

### General Site Suitability Criteria

A study is being conducted under contract with Argonne National Laboratory to identify and characterize various typical accidents and incidents that have occurred at fuel cycle facilities. The identification of these accidents and their associated exposures of personnel to radioactivity will provide an updated assessment of the impact at each type of facility.

A study was completed under contract with United Engineers and Constructors, Inc., to collect available data and establish cost characteristics for nuclear generating station designs, as they are determined by the most significant site characteristics. The scope of the study included main condenser cooling, transmission, flood protection, site access, and plant/site earthwork.

## SITING STANDARDS

The standards on the siting of nuclear plants deal with procedures for site review, site safety, and protection of the environment.

### Site Review Procedures

**Alternative Sites.** A proposed rule has been prepared to establish regulatory policy and procedures for evaluation of alternative sites for nuclear generating stations under the National Environmental Policy Act (NEPA). The proposed rule is designed to (1) fulfill the NEPA objectives of ensuring that environmental factors have been fully considered in NRC decisionmaking, (2) reduce uncertainty and delay in the decisionmaking process, (3) reduce Federal paperwork in NEPA statements, and (4) limit alternative site review to relevant and material issues. There is an ongoing contract at Brookhaven National Laboratory to survey the range of site screening and selection methodologies, to examine the operational features, and to identify those features that should be looked for in determining whether an applicant has employed an acceptable selection methodology.

**Emergency Planning.** Guide 2.6, on emergency planning for research reactors, was issued for comment in January 1979. The staff participated in the preparation and writing of NUREG-0396, EPA 520/1-78-016, entitled "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants." This report of the NRC/EPA Task Force on Emergency Planning was issued for public comment in December 1978.

The NRC is in the process of reevaluating and revising regulations and regulatory guides in the emergency planning area. Late in 1979, the Commission published proposed amendments to its regulations to provide an interim upgrade of NRC emergency planning rules.

In June 1979, the NRC began a formal reconsideration of the role of emergency planning in assuring the continued protection of the public health and safety in areas around nuclear power facilities. The Commission had begun this reconsideration in recognition of the need for more effective emergency planning and in response to reports issued by responsible offices of the Federal government and its Congressional oversight committees.

On July 17, 1979, the NRC published an advance notice of proposed rulemaking on the subject of State and local emergency response plans and those of licensees. Comments received from the advance notice are being evaluated by the NRC staff for use in possible future rulemaking actions.

By memorandum dated July 21, 1979, the Commission requested that the NRC staff undertake expedited rulemaking on the subject of emergency planning. The proposed amendments (published in the *Federal Register* in December 1979) respond to the Commission's request. Consequently, because these proposed amendments were prepared on an expeditious basis, considerations related to the workability of the proposed rule changes may have been overlooked and significant impacts to NRC, applicants, licensees, and State and local governments may not have been identified. Therefore, the NRC will seek additional public comment by holding workshops prior to the preparation of a final rule to (a) present the proposed rule changes to State and local governments, utilities, and other interested parties, and (b) to obtain comments concerning the costs, impacts, and practicality of the proposed rule changes.

Specifically, NRC is proposing amendments to 10 CFR Part 50 to require that emergency response planning considerations be extended to Emergency Planning Zones (discussed in NUREG-0396). Both the Commission and EPA have formally endorsed the concepts in that EPA/NRC Report, 44 *Federal Register* 61123 (October 23, 1979).

The proposed amendments include, as a condition of operating license issuance, that State and local governmental emergency response plans be submitted to and concurred in by the NRC. If the State and local governmental plans have not received NRC concurrence, the applicant may attempt to demonstrate to the satisfaction of the Commission that deficiencies in the plans are not significant for the plant in question or that alternative compensating actions have been or will be taken.

In addition, the proposed rulemaking revises 10 CFR Part 50, Appendix E, "Emergency Plans for Production and Utilization Facilities," to clarify, expand, and upgrade the Commission's emergency planning regulations.

The Commission is also proposing to amend its regulations in order to require certain special nuclear material facility licensees (for processing and fuel fabrication, scrap recovery, or conversion of uranium hexafluoride) to keep their emergency plans up to date.

### Site Safety

NRC site safety standards are rules and guides for assessing and mitigating adverse effects associated with natural events such as earthquakes, floods, and extreme meteorological conditions and man's activities at and near nuclear sites.

In the field of meteorology, Guide 1.145, on atmospheric dispersion models for potential accident consequence assessments at nuclear power plants, was issued for public comment in August 1979. NUREG/CR-1024, "An Initial Assessment of Flash Density and Peak Current Characteristics of Lightning Flashes to Ground in South Florida," was issued in October 1979. A revision to Guide 1.23, on onsite meteorological programs, is in progress. The staff is continuing data evaluation for the development of standards on extreme windspeeds in coastal areas, extreme snow and ice accumulations, and extreme temperatures, as well as on the hazards associated with lightning and dust and sand storms.

In the geology and seismology area, review of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100 was completed. The results of the staff's review were provided to the Commission on April 27, 1979. Other earth science activities included gathering data for a technical report on dating materials that may be found in fault zones and classifying the faults and fractures of the Appalachian foldbelt. A study on siting nuclear facilities in areas susceptible to ground collapse is still underway.

In the hydrology area, work on a revision to Guide 1.135, on normal water level and discharge, is in progress. ANSI committee and IAEA work on surface water, ground water, and radionuclide transport is also continuing through the preparation of new standards and revision of existing standards and guides. Input was provided to the reassessment effort on low-level waste regulations, and work is continuing on the hydrologic aspects of siting criteria.

## ENVIRONMENTAL PROTECTION

Environmental protection standards are concerned with the protection of the public and the environment from both radiological and nonradiological impacts of nuclear facilities. This includes assessment of environmental impacts, control of effluents, and monitoring of the environment around the facilities.

Revision 1 to Guide 4.15, on quality assurance for radiological effluent and environmental monitoring for normal operations, was issued in February 1979. An NRC Task Group issued its report on "Radiological Monitoring by NRC Licensees for Routine Operations of Nuclear Facilities," NUREG-0475, in October 1978.

A regulation was proposed to revoke Section 20.304 of 10 CFR Part 20, which currently allows licensees to bury small quantities of radionuclides without notifying NRC.

A substantial effort is now being devoted to the environmental aspects of uranium milling, decommissioning and decontaminating nuclear facilities, and radioactive waste disposal.

## Interagency Coordination

NRC has the responsibility for implementing both EPA's guidance and generally applicable environmental standards for protection against radiation. During 1977, EPA published standards (40 CFR Part 190) that limit releases of radioactive material and resulting doses to the public from the operation of various nuclear facilities associated with the uranium fuel cycle. An NRC task force, which includes EPA staff members, is completing the program for implementing these standards.

The Clean Air Act Amendments of 1977 include provisions for EPA to develop emission standards for radioactive materials from NRC-licensed facilities. EPA is expected to issue its first standard in 1980. The NRC and EPA staffs have been working together on the development of these standards and on an interagency agreement to minimize duplication of effort in implementing the Clean Air Act.

The NRC became a member, along with 15 other major Federal agencies, of the Toxic Substances Strategy Committee, formed under the leadership of the Council on Environmental Quality. NRC staff members served on seven of the task groups of this committee, which is to recommend strategies to be used by the Federal Government for the control of toxic and hazardous substances. Although radioactive materials have been excluded from this report, the principles for controlling cancer-causing materials would be expected to affect radiation control strategies, and the expertise gained by NRC in controlling radiation is directly applicable to some aspects of controlling other carcinogens. The draft report of the Committee has been issued for public comment. The final version is expected to be transmitted to the President by the end of the year.

The NRC staff has reviewed the proposed Hazardous Waste Guidelines and Regulations issued by the EPA Division of Hazardous Waste, and a series of meetings has been held with the EPA staff with a view to clarifying apparent differences in approach for disposal of EPA-regulated hazardous wastes and NRC-regulated low-level radioactive wastes. A memorandum of understanding is being developed to provide guidance for those instances when EPA-regulated hazardous wastes are mixed with radioactive wastes.

The Departments of Interior and Commerce have published proposed regulations implementing the Fish and Wildlife Coordination Act. These proposed regulations could require significant revisions in NRC's operating procedures. The NRC staff has commented on the proposed regulations and, depending on the final form of the regulations, will develop implementing procedures consistent with the NRC's position as an independent regulatory agency. Continuing discussions are expected among the three agencies to determine how this can be done in a mutually acceptable manner.

# RADIOLOGICAL HEALTH STANDARDS

### Low-Level Radiation Effects

Significant progress was made during the year in carrying out the mandates of Public Law 95-601, which directed the NRC and the EPA, in consultation with the Secretary of Health, Education and Welfare (HEW, now HHS), to conduct preliminary planning and design studies for epidemiological studies. A joint EPA/NRC report on the health effects research needs, capabilities, and current programs of the two agencies was prepared and transmitted to the Congress. A contract was awarded for the preparation of a plan for identifying the options for Federal epidemiological research and for assessing the feasibility of these options. A progress report to the Congress on these activities was prepared and transmitted at the end of the fiscal year.

The report of the Interagency Task Force on the Health Effects of Ionizing Radiation was issued by HEW. Drafts of the various work group reports were issued for public comment in April, and the final report was issued in June. The task force examined information on the health effects of low-level ionizing radiation and the problem of coordination of Federal radiation protection and radiation research activities. The Director of the Office of Standards Development represented NRC on the task force, and NRC staff members participated actively on the various working groups. The report was the subject of hearings held on May 8, 1979, before the Subcommittee on Energy, Nuclear Proliferation, and Federal Services of the Senate Committee on Governmental Affairs. The NRC was represented at this hearing and contributed suggestions for improving Federal radiation protection and radiation research programs.

NRC staff members were also active participants in the efforts of the Interagency Task Force on Ionizing Radiation Research to improve the coordination of Federal research programs, to conduct a comprehensive review of these programs, and to establish a comprehensive program of research into the biological effects of low-level ionizing radiation. The task force was convened by the Secretary of Health, Education, and Welfare to assist in carrying out the requirements of Title II of Public Law 95-622, the Biomedical

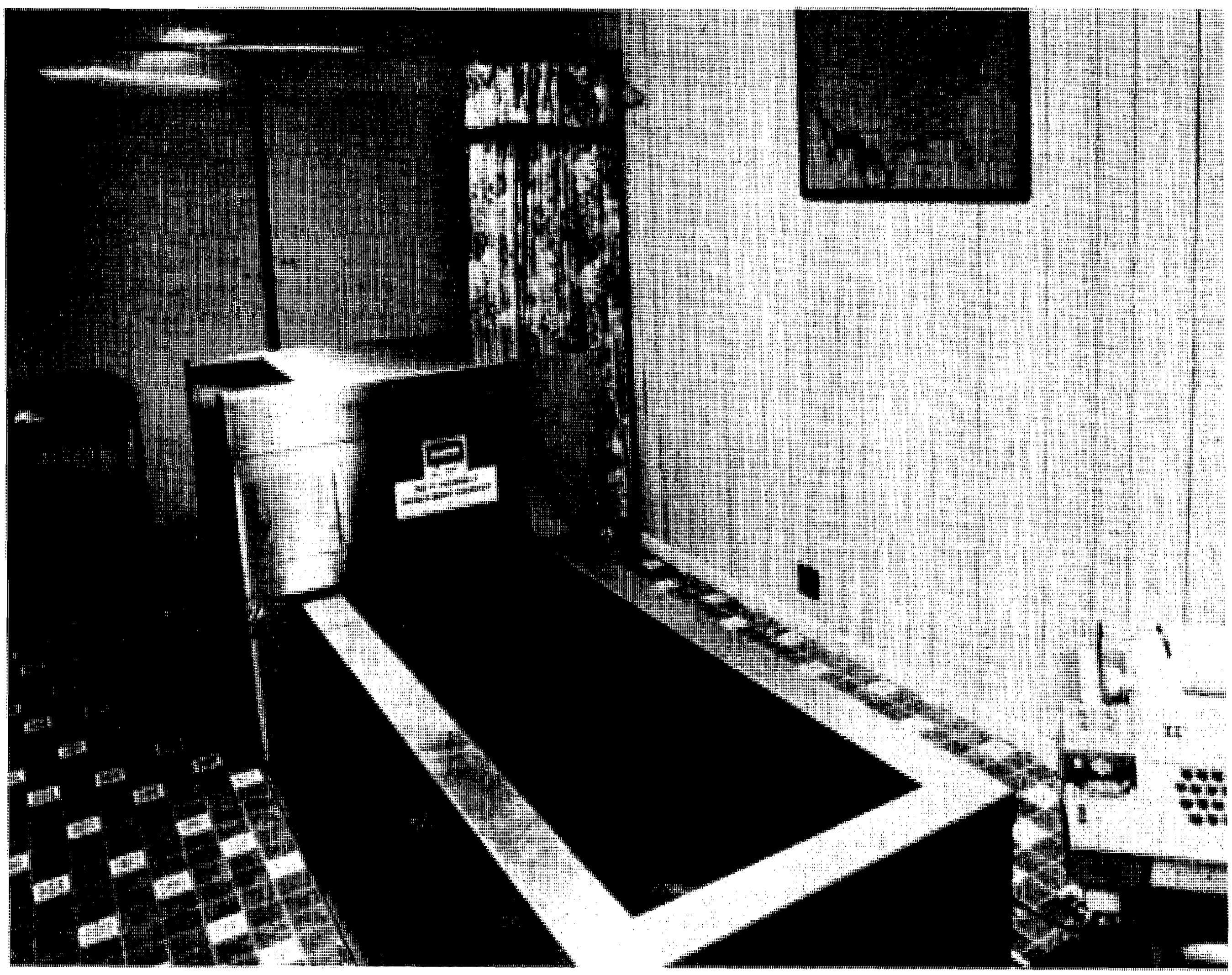
Research and Research Training Amendments of 1978. A special subcommittee of this task force was formed after the accident at the Three Mile Island Nuclear Station to evaluate proposed studies and to recommend possible Federal studies of the health and psychological impact of the accident. NRC staff were members of this subcommittee and provided information on the radiation levels and doses received by off-site residents and workers at the facility.

A final report was published on an NRC-sponsored epidemiological study of residents in Mesa County, Colorado, where extensive uranium mining and milling operations are located. This study, conducted by the Colorado Department of Public Health, found no relationship between exposure to mill tailings used in residential construction and an observed excess of leukemia in Mesa County.

Three contracts for independent analyses of data on the radiation exposures and their effects on the health of the Hanford worker population were also completed. The results of this work will be useful in support of the epidemiology feasibility planning studies described above. Valuable insights were obtained into the complexities of handling and analyzing data bases of this nature in investigating possible causal relationships between radiation exposures and health effects.

## Nuclear Medicine

In fiscal year 1979, the NRC issued a final medical policy statement, a final rule requiring radiation surveys of certain therapy patients prior to discharge, and a final rule requiring annual calibrations of teletherapy units. The two rules resulted from reports of medical misadministrations of radiopharmaceuticals. The NRC also issued a proposed rule requiring each medical licensee to appoint a radiation



**In April 1979 the NRC Office of Nuclear Reactor Regulation began a public whole body counting program at Three Mile Island using this "do-it-yourself" counter. Of more than 700 counts completed none identified radioactivity which could have resulted from the accident. The radiation detector, housed in the box-like structure, passes over the examinee, on signal.**

safety committee and one requiring medical licensees to test certain radiopharmaceuticals for radionuclide contaminants. A final rule change, effective in June 1979, requires all materials licensees, including medical institutions, to inform the NRC when they decide to give up their licenses. Another final rule change, effective in September 1979, added veterinarians to the groups authorized to use byproduct material under general license for *in vitro* clinical or laboratory testing.

## OCCUPATIONAL HEALTH STANDARDS

### Respiratory Protection

During fiscal year 1979, the NRC continued to develop and provide the information necessary to ensure the adequacy of licensees' respiratory protection programs. Two Respirator Users' Warnings from the National Institute for Occupational Safety and Health (NIOSH) were circulated to advise licensees of actions to take concerning potential failures of certain self-contained breathing apparatus (SCBA) (Inspection and Enforcement Circulars 79-09 and 79-15).

Under a technical assistance contract with the Los Alamos Scientific Laboratory (LASL), a videotape was produced for training users in proper respirator fitting methods. Under a separate research contract with LASL, measurements were made to determine the protection provided by open-circuit SCBA. These data, along with additional information from LASL and other sources, will be used to update the NRC's guidance to licensees on acceptable respiratory protection programs.

LASL is also providing technical assistance for the NRC's Three Mile Island post-accident operations. An expert on respiratory protection provided onsite assistance to the NRC. Special studies were carried out at the laboratory to obtain information on the performance of new positive-pressure closed-circuit SCBA devices and on the efficacy of air-purifying respirator canisters for use against airborne radioiodines.

The NRC continued its cooperation with other governmental and nongovernmental agencies toward the development of needed improvements in respiratory protection.

### Testing for Personnel Dosimetry

Evaluations of the degree of accuracy that is provided by personnel dosimetry processors in the United States indicate that improved performance of some processors is needed. Personnel dosimetry devices are used to measure the radiation dose received by workers in NRC-licensed facilities. To obtain more accurate processing of dosimeters, the NRC staff is working on a requirement that personnel dosimetry results be accepted only from a processor who has successfully passed certain prescribed accuracy tests. The test criteria would be adapted from a consensus standard developed by ANSI.

In preparation for the new regulation, the NRC funded a two-year pilot study, which was conducted by the University of Michigan. The objectives of the pilot study were:

(1) To test the consensus standard for practicality as well as for degree of difficulty.

(2) To provide processors an opportunity to correct any process problems that they may have prior to publication of the new regulation in effective form.

(3) To develop a detailed procedures manual for use by future testing laboratories.

The study was completed in September 1979. Fifty-eight processors participated in two rounds of testing in various radiation categories. In Test #1, only 22 percent of the category tests were passed, using criteria published in the ANSI standard (N13.11, July 1978). In Test #2, the percentage of category tests passed rose to 38 percent. Thus, while considerable improvement was experienced, the overall performance of some processors is evidently in need of improvement. The NRC staff has initiated work on the necessary amendment of 10 CFR Part 20. The pilot study indicated that the ANSI standard, with minor modifications, is suitable for the regulatory purposes of the NRC.

### Personnel Monitoring Reports

In September 1978, the NRC published an amendment to 10 CFR Part 20 to extend to all NRC licensees the requirement for annual statistical summary reports on workers' radiation exposures. Under the previous regulation, only four categories of licensees were required to submit an annual statistical summary of monitored whole-body exposures, i.e., the number of people in each of 18 prescribed ranges of radiation exposure.

The amendment to Part 20 extends this statistical summary reporting requirement to all NRC specific licensees for a period of two years. After evaluating the data for 1978 and 1979, the NRC will consider whether it will extend or modify the reporting requirement. The four categories of licensees previously covered will continue to be required to report in any event. The amendment does not affect existing requirements for the provision and use of personnel monitoring equipment or for the recording of personnel monitoring data but relates solely to the reporting of data already recorded.

## Worker Exposure to Neutrons

The use of NTA film has come into question as an adequate dosimeter for monitoring workers' exposures to neutrons at pressurized water reactors (PWRs). The NRC has contracted with the Battelle Pacific Northwest Laboratories (PNL) and the Rensselaer Polytechnic Institute (RPI) independently to examine this problem. PNL and RPI are conducting surveys both of the neutron spectra in work spaces at PWRs and of the response of various types of dosimeters to such neutrons. The preliminary results of these studies indicate that NTA film might not be a good dosimeter for these neutrons because it responds poorly to the neutrons of less than 0.7 Mev that constitute a portion of the neutrons in PWR work spaces. Improved albedo dosimeters might provide better measurements of these neutron exposures. Therefore, the NRC undertook a revision of Guide 8.14 in 1979. The purpose of this revision is to advise licensees of the inadequacy of NTA film for neutron dosimetry for routine operations at nuclear power plants. In revising the guide, the NRC will address the "state of technology" in personnel dosimeters for neutrons and will provide guidance on acceptable methods for estimating workers' exposures to neutrons.

## Calibration of Air Sampling Instruments

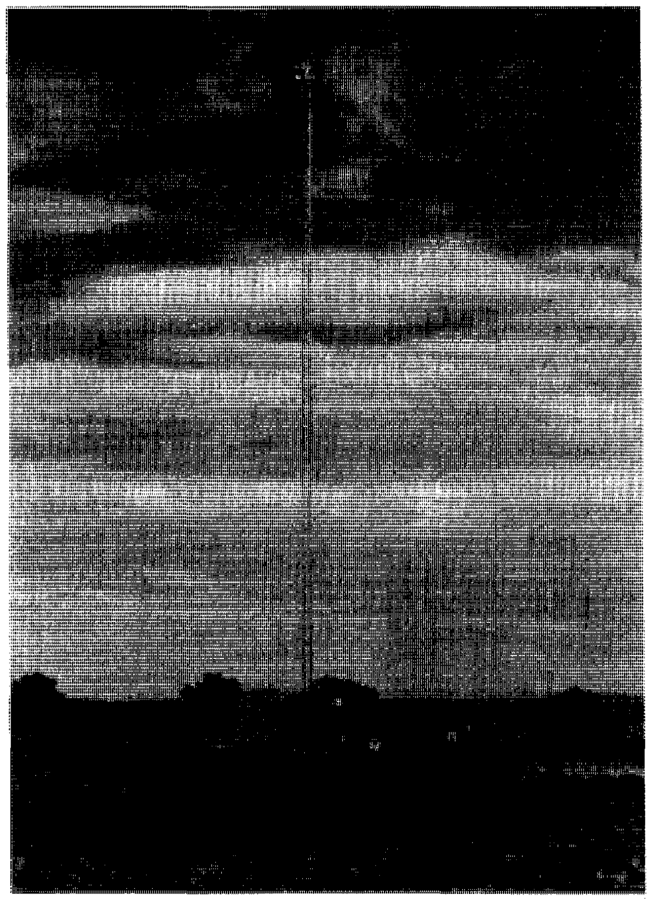
In October 1979, a draft regulatory guide was issued on acceptable methods of calibrating air sampling instruments to more accurately determine the volume of air sampled. In addition, a frequency of calibration, an error limit, and documentation requirements are specified.

The guide is expected to improve licensees' air monitoring programs and estimates of workers' exposures to airborne radioactive material.

## Transient Worker Radiation Protection

In June 1979, the NRC published amendments to 10 CFR Parts 19 and 20 that are intended to minimize the possibility of overexposure of (1) short-term workers, sometimes called "transient workers," and other individuals who may be employed by or work in the restricted areas of more than one licensee within a single calendar quarter, and (2) individuals who may work for more than one licensee at a time (moonlighters).

The amendments, which became effective on August 20, 1979, require NRC licensees to control the total occupational radiation dose of individuals who work in NRC-licensed activities. The changes require licensees (1) to obtain information from a prospective employee on occupationally related doses received during a current calendar quarter from sources outside the licensee's control if there is a chance that the employee may subsequently receive a dose in excess of 25 percent of the regulatory standards in the facility of the new employer; (2) to furnish prompt estimates of occupational dose, at the request of the individual, upon termination of work; and (3) to keep associated records.



**This 100-foot radon sampling tower near a mine in New Mexico measures the speed, direction and stability of winds, temperature and the atmospheric pressure, and takes ambient radon concentrations measurements at five different elevations. This program is conducted for NRC's Office of Nuclear Materials Safety and Safeguards by Argonne National Laboratory.**

## Medical Institutions

Two guides specific to occupational radiation protection in medical institutions were issued for comment during fiscal year 1979: Guide 8.23, on radiation surveys in medical institutions, and Guide 10.8, on medical licensing. Guide 10.8 explains the information to be submitted in an application for a license to use byproduct radioactive materials in diagnostic and therapeutic medical applications, provides a simpler form (NRC-313M) for completing the required entries, and provides acceptable methods and statements related to radiation safety and user qualifications.

As reported last year, Guide 8.18 and a companion report (NUREG-0267) that provides more detailed information and references were issued for comment in January 1978. During fiscal year 1979, these two documents have provided broad interim guidance and information for establishing acceptable occupational radiation safety programs in medical institutions. Both documents have received a generally favorable reception from the medical and medical physics communities, but there have been a number of suggestions for improvement, additions, or deletions that will require a careful balancing of the various viewpoints in preparing the final versions. Extensive comments have also been received on Guides 8.23 and 10.8. Since the four documents now published for medical institution radiation safety and licensing have some overlapping subject areas of guidance, a major effort will be made during the early part of fiscal year 1980 to revise all of these documents. The effort will include consideration of the advice of interested professional and public groups who may be interested in or affected by the guidance in these documents.

## Bioassays

Revision 1 to Guide 8.20, providing guidance for I-125 and I-131 bioassay, was issued in September 1979. A draft guide on bioassay programs and methods for fission and activation products was issued in August 1979. For the most part, this draft guide adopts the recommendations of a recent standard issued by the American National Standards Institute (ANSI). The NRC staff participated in the development of the ANSI standard. These documents supplement three previous guides that provided general information on acceptable methods of interpreting bioassay results and gave specific guidance on interpreting uranium bioassays. The new documents provide guidance to management on the levels of radioactivity or working conditions under which bioassay should be performed. They also specify on whom the assays should be performed and when investigative or corrective measures should be taken. The iodine and fission product bioassay guidance takes into consideration the amounts of each radionuclide above which exposure potential becomes appreciable, as indicated by industrial and medical experience.

## Health Protection at Uranium Mills

A memorandum of understanding to ensure consistency of regulatory actions is still being developed between the NRC and the Mine Safety and Health Administration (MSHA) of the Department of Labor. The Federal Mine Safety and Health Act of 1977 gives MSHA jurisdiction with respect to protection of uranium mill workers similar to that given NRC under the Atomic Energy Act of 1954, as amended.

Reports were published during this fiscal year on the solubility of yellowcake in the human body (NUREG/CR-0414 and NUREG/CR-0530) and on *in vivo* counting for uranium and radium in the body (NUREG/CR-0841).

A regulatory guide on health physics surveys at uranium mills is in preparation.

**Occupational ALARA at Uranium Mills.** In 1979, the NRC undertook the development of a guide intended to provide operators of uranium recovery facilities with the specific information needed to ensure that occupational exposures are as low as reasonably achievable (ALARA) for workers in uranium mills. In providing adequate occupational health protection for uranium mill workers, it is important to consider the chemical toxicity of uranium to the kidney, as well as radiologic effects. The major health effect for uranium mill workers results from the inhalation of suspended particulates. Thus, the ALARA guide will provide both operational procedures and design specifications for ventilation equipment to minimize suspended particulate materials in work areas. Of particular concern are those areas associated with the yellowcake dryer and with the packaging and shipping of dried yellowcake. These locations are important because yellowcake is a biologically soluble compound and has a proven chemotoxic effect on the kidney, as well as a radiotoxic effect. The guide is scheduled for issuance for public comment by mid-1980.

## Effecting Occupational ALARA

The NRC staff has completed consideration of methods for quantifying ALARA requirements under proposed amendments to NRC regulations previously developed. These proposed amendments would require the development and implementation of ALARA programs by all licensees who are required by the NRC to perform personnel dosimetry, air sampling, or bioassays for worker protection. The staff is proposing the development of several guides to complement the proposed amendments. The guides would be specific to several different types of licensees and would provide descriptions of ALARA programs that would be acceptable under the proposed amendments.

## Surveys During U-235 Processing and Fuel Fabrication

Guide 8.24, issued for comment in November 1978, identifies the types and frequencies of health physics surveys that are acceptable to the NRC staff for use in plants licensed by the NRC for the processing of enriched uranium-235 and for the fabrication of uranium fuel. Revision 1 to Guide 8.24 was issued in October 1979.

## Radiation Safety Training at LWRs

A draft guide on radiation protection training for personnel at light water reactor (LWR) facilities was issued in August 1979. The draft guide describes radiation protection training consistent both with maintaining occupational doses at LWRs as low as is reasonably achievable and with the requirement of 10 CFR Part 19 for training individuals who enter restricted areas at nuclear power plants.

## Gamma Irradiators

A draft guide on the preparation of license applications for the use of gamma irradiators was issued in March 1979. This guide informs people who wish to apply for such licenses of the information that the NRC requires for reviewing and acting upon such applications. The guide reflects the new requirements for irradiators in 10 CFR Part 20, effective in 1978.

## Industrial Radiography Safety

New amendments to 10 CFR Part 34 on safety in industrial radiography were published in effective form in September 1979.

A petition for rulemaking to have the NRC license individual radiographers is under consideration. The petition states that safety in industrial radiography could be improved by making individual radiographers more directly responsible for their actions.

A draft regulatory guide on the use of audible-alarm dosimeters for improving radiography safety was issued in August 1979. Battelle Pacific Northwest Laboratories is conducting tests of such dosimeters to determine their reliability, and a report on their first series of tests has been published (NUREG/CR-0554).

## Petition on "Radiation Area"

A petition received in October 1978 requested that the NRC amend the definition of "radiation area," as set forth in §20.202 (b)(2) of 10 CFR Part 20, to specify dose rates comparable to those set forth in §20.105, rather than the rates of 5 millirems in an hour and 100 millirems in any 5 consecutive days which are currently specified in §20.202(b)(2). The proposed change would require any area that could not qualify as an unrestricted area to be posted as a radiation area.

After due consideration of the petition and the bases for the proposed change provided by the petitioner, the comments received following publication of notice of receipt of the petition, and other factors involved, the petition was denied on September 26, 1979. The principal reason for the denial was that there does not appear to be any reduction in risk associated with the petitioned change. Indeed, there is a potential that unnecessary exposure of workers might result.

# SAFEGUARDS STANDARDS

The NRC devoted substantial standards effort during fiscal year 1979 to the safeguarding of nuclear materials and facilities against theft and diversion. Development of regulations in this area is discussed in Chapter 5.

## Physical Protection

In support of existing regulations and of the proposed and newly adopted safeguards regulations for physical protection of SNM discussed in Chapter 5, the NRC has issued several regulatory guides and technical reports. These include:

(1) Proposed Revision 1 to Guide 5.7, describing measures acceptable to the NRC staff for implementing entry/exit control requirements at facilities other than nuclear power plants, was issued in May 1979.

(2) Proposed Revision 1 to Guide 5.14, describing measures acceptable to the NRC staff for surveillance or observation of individuals within material access areas in order to strengthen the safeguarding of strategic special nuclear material, was issued in May 1979.

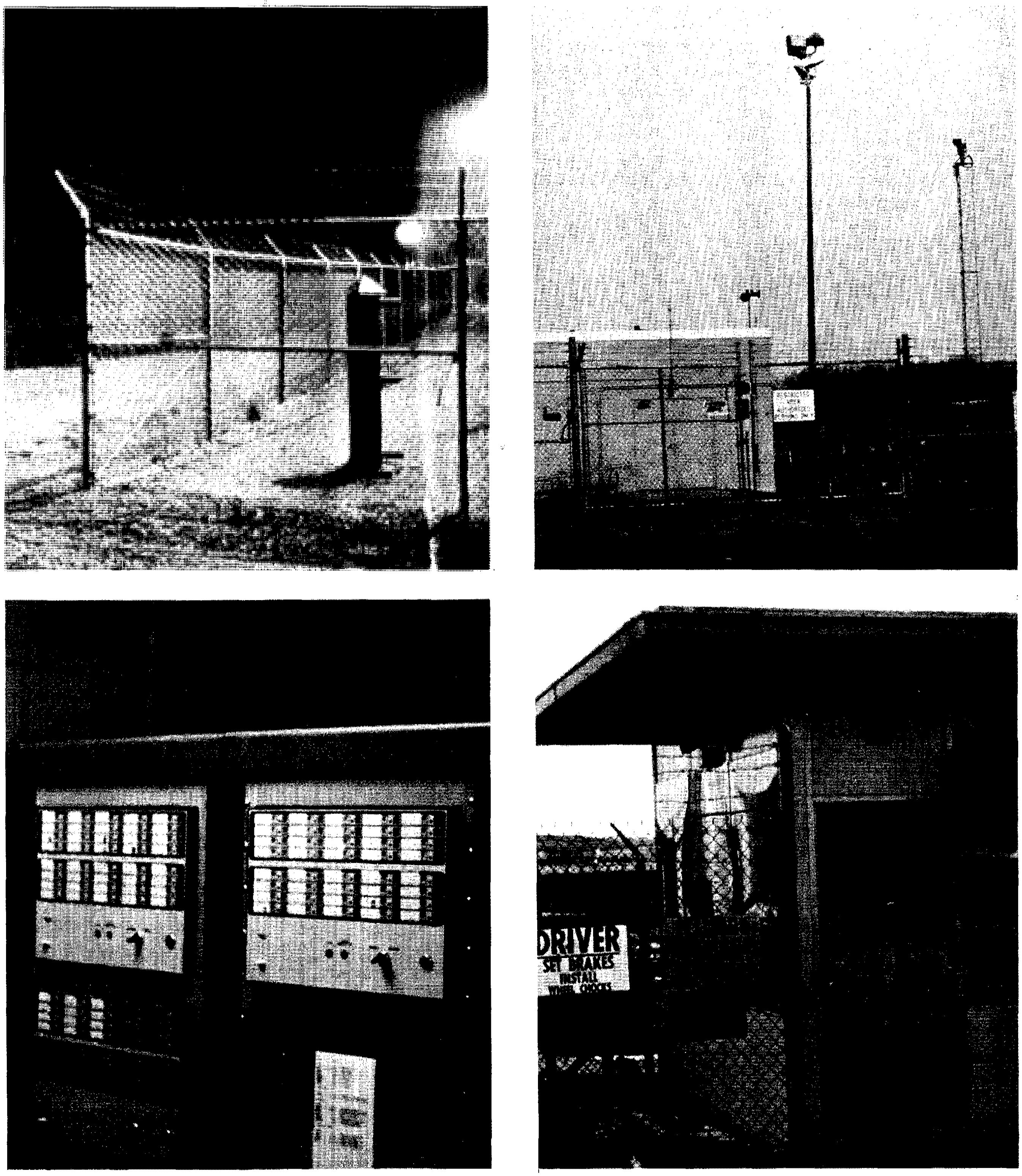
(3) Proposed Revision 2 to Guide 5.44, describing six types of perimeter alarm systems for detecting intrusion into plants that use or process highly enriched uranium, uranium-233, or plutonium, was issued in May 1979. This guide also sets forth criteria that are acceptable to the NRC staff for the systems' performance and use.

(4) Proposed Revision 1 to Guide 5.57, describing procedures acceptable to the NRC staff with regard to the protection of strategic special nuclear material during preparation for shipment, transfer between licensees, and on receipt by a licensee, was issued in May 1979.

(5) A draft guide providing a standard format and content for licensees' plans submitted in response to requirements for the protection of Category II and III type materials against theft was issued in August 1979.

(6) A draft guide providing a logical scheme for determining when a safeguards event should be reported to the NRC, in addition to a partial list of the kinds of events that should be reported, was issued in October 1979.

(7) NUREG/CR-0464, "Site Security Personnel Training Manual," and NUREG/CR-0465, "Transportation Security Personnel Training




NRC safeguards regulations for fuel cycle facilities specify many of the techniques and equipment to be used for the prevention of intrusion by unauthorized people. Some devices used for this purpose are shown here. Clockwise from top left: **a fenced, protected area features nighttime illumination and infrared intrusion detection devices**; closed-circuit TV surveillance equipment scans protected areas; access control points feature shielded doors/windows, two-way remote communications, internal controls for lighting systems and gatelocks, as needed; a central alarm control panel "oversees" an entire facility.

Manual," assist licensees in developing effective security personnel training and qualification programs, as required by 10 CFR Part 73. The manuals identify the level and scope of training of security personnel assigned to perform specific tasks and job duties to protect special nuclear material, nuclear facilities, and shipments.

(8) NUREG/CR-0484, "Vehicle Access and Control Planning Document," provides guidance in planning a vehicle access and control system at nuclear fixed site facilities.

(9) NUREG/CR-0485, "Vehicle Access and Search Training Manual," provides training guidance to help NRC licensees better perform vehicle access and search operations at nuclear fixed site facilities.

(10) NUREG/CR-0509, "Emergency Power Supplies for Physical Security Systems," includes basic information useful in the planning, design, and implementation of emergency electric power systems for safeguard security systems.

(11) NUREG/CR-0510, "Duress Alarms for Nuclear Fixed Site Facilities," identifies and discusses different types of devices currently available that could be used as duress alarm systems.

(12) NUREG/CR-0543, "Central Alarm Station and Secondary Alarm Station Planning Document," provides planning guidance for the siting, construction, and equipping of a central alarm station and a secondary alarm station as required by the NRC.

## Material Control and Accounting

In support of existing requirements and the strengthened regulations for material control and accounting of special nuclear material (SNM), discussed in Chapter 5, the NRC issued several regulatory guides and technical reports. These include:

(1) Guide 5.58, issued for comment in November 1978, presents conditions and procedural approaches acceptable to the NRC staff for establishing and maintaining traceability of SNM control and accounting measurements.

(2) NUREG/CR-0061, "Preparation of Working Calibration and Test Materials: Plutonium Oxide," provides guidance for preparing plutonium dioxide reference materials used to calibrate and to maintain quality control over methods of analysis for plutonium.

(3) NUREG/CR-0139, "Preparation of Working Calibration and Test Materials: Mixed Oxide," provides guidance for preparing mixed-oxide reference materials used to calibrate and to maintain quality control over methods of analysis for plutonium and uranium in mixed-oxide powders and pellets.

(4) NUREG/CR-0515, "Methods for the Accountability of Reprocessing Plant Dissolver and Waste Solutions," gives detailed methods for the accountability of uranium and plutonium solutions resulting from the dissolution of spent light water reactor fuel.

(5) NUREG/CR-0562, "Specifications for Germanium Radiation Detectors Used for Gamma Ray Assay in Safeguards Applications," discusses the physics and engineering of radiation detector fabrication and operation and the basic principles of assay by gamma-ray spectroscopy.

(6) NUREG/CR-0591, "Current Usage of Containers for SNM Storage, Transfer and Measurements," is an interim report that surveyed the types, sizes, and materials of containers used to store, transfer, or measure SNM. Information gained from this report will serve as the basis for standardization of SNM containers with respect to size and fabrication specifications in the nuclear industry, reducing cost while improving nondestructive assay measurement performance.

(7) NUREG/CR-0602, "Active Assay Handbook," contains principles of active nondestructive assay and associated measurement procedures for a variety of SNM samples.

(8) NUREG/CR-0683, "Statistical Methods for Evaluating Sequential Material Balance Data," evaluates and compares four material loss estimators for two different loss mechanisms and for four different time periods using the criterion of "power of the test" for detecting a loss.

(9) NUREG/CR-0772, "Auditing of Measurement Control Programs," documents concepts and factors associated with auditing a measurement control system that licensees may use to assist them in auditing their measurement control program.

(10) NUREG/CR-0773, "Training and Qualifying Personnel for Performing Measurements for the Control and Accounting of Special Nuclear Material," documents procedures that licensees may use to assist them in meeting the present requirements contained in the NRC regulations.

# RADIOISOTOPES IN INDUSTRY

## Products Containing Radioactive Material

In fiscal year 1979, the NRC issued two reports on radiation doses associated with consumer products containing radioactive materials. These reports, NUREG/CR-0215 and NUREG/CR-0216, present estimates of potential radiation doses to members of the general public from exposure to timepieces containing radioactive material. This radioactive material is either tritium or promethium-147 mixed with a phosphor used in luminous paint or tritium contained in sealed-glass ampules to be used as a radioluminescent source. The conclusion that can be drawn from the results presented in the reports indicates that the use of these radioactive materials in timepieces does not constitute a significant health hazard.

In November 1978, the NRC issued a report, NUREG/CP-0001, entitled "Radioactivity in Consumer Products." This report is a compilation of information dealing with radioactivity in products available in the market place to the general public. The report is based on papers presented at the February 1977 symposium on public health aspects of radioactivity in consumer products.

In July 1979, the NRC issued a supplement to NUREG-0060, "Generic Environmental Statement on Routine Use of Plutonium-Powered Cardiac Pacemakers." The supplement provides updated information on alternative power sources, particularly lithium batteries, and considers the current extent of pacemaker use and the makeup of the patient population. The supplement concludes that the results of the original environmental statement are still valid, i.e., the routine use of plutonium-powered pacemakers should be authorized. The supplement further states that the plutonium-powered pacemaker provides physicians with an alternative choice of medical treatment for patients who may require long-term pacing, but who are ill-suited for replacement operations. The NRC is interested in receiving additional public comments on both documents, NUREG-0060 and its supplement, before considering a final regulation on the routine use of the plutonium-powered pacemakers.

In 1979, a standard on radiological safety in the design and construction of apparatus for gamma radiography was being developed by a subcommittee of the American National Standards Institute with NRC participation. The standard provides guidance for persons responsible for the design and construction of apparatus for industrial gamma radiography using radioactive materials as the energy source. Criteria for the design of new devices and for qualifying prototypes to performance standards are included. At year-end, development was not completed, and the NRC was considering rulemaking proceedings in this area.

### Licensing Matters

In October 1978, the NRC issued for comment Guide 6.8, on identification plaques for irretrievable well-logging sources. This guide describes methods that would be acceptable to the NRC staff for meeting the proposed requirements with respect to the characteristics and the mounting of permanent identification plaques at the surface of a well that contains an irretrievable well-logging source. These identification plaques will serve as one aspect of continuing control and are intended to be a long-term indication of a sealed radioactive source downwell. Thus, any persons planning to reenter the well for additional operations will be alerted to the existence of a source downwell. The guide was issued for comment concurrently with the proposed rule changes to 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," that will ensure continuing control of byproduct and special nuclear materials.

## NATIONAL STANDARDS PROGRAM

The national standards program is conducted under the aegis of the American National Standards Institute (ANSI). ANSI acts as a clearinghouse to coordinate the work of standards development in the private sector.

The NRC staff is active in the national standards program, particularly with respect to setting priorities so that regulatory views are known regarding the standards that can be most useful in protecting the public health and safety. NRC participation is based on the need for national standards to define acceptable ways of implementing the NRC's basic safety regulations.

The actual drafting of standards is done by experts, most of whom are members of the pertinent technical and professional societies. Approximately 230 NRC staff members serve on working groups organized by technical and professional societies. These societies are listed in the accompanying table. National standards are used in the regulatory process only after independent review for suitability by the NRC staff and after public comments on their intended use have been solicited and considered.

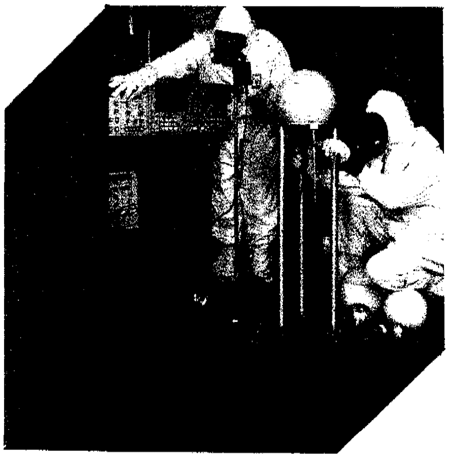
## IAEA REACTOR SAFETY STANDARDS

NRC has continued its lead role in organizing and carrying out U.S. participation in the IAEA program to develop safety codes of practice and safety guides for nuclear power plants. The NRC coordinates U.S. technical activities associated with this program. The codes and guides will provide a basis for national regulation by developing countries of the design, construction, and operation of nuclear power plants. NRC staff members continued to represent the United States on the IAEA Senior Advisory Group (SAG) that oversees the program and on the Technical Review Committees working in the five areas of primary interest: governmental organization, siting, design, operation, and quality assurance. The Director of the NRC's Office of Standards Development is the U.S. member of the SAG.

During 1979, the Senior Advisory Group, Technical Review Committees, and working groups under them drafted four new guides and completed seven safety guides that were forwarded to the Director General of the IAEA with the recommendation that they be issued. About 40 of the approximately 50 safety guides planned to date have been drafted and are undergoing review. During the drafting process, the NRC standards staff coordinated the reviews within the U.S., soliciting comments from interested members of the public, industry, and other government agencies. (See also Chapter 9.)

### SOCIETIES SPONSORING NUCLEAR STANDARDS DEVELOPMENT ACTIVITIES IN WHICH NRC STAFF MEMBERS PARTICIPATE

American Association of Physicists in Medicine
American Concrete Institute
American Conference of Governmental Industrial Hygienists
American Institute of Chemical Engineers
American Institute of Steel Construction
American Insurance Association
American National Standards Institute
American Nuclear Society
American Society of Civil Engineers
American Society of Mechanical Engineers
American Society for Nondestructive Testing
American Society for Testing and Materials
American Welding Society
Health Physics Society
Institute of Electrical and Electronics Engineers
Institute of Nuclear Materials Management
Instrument Society of America
Metals Properties Council
National Council on Radiation Protection and Measurements
National Fire Protection Association
National Sanitation Foundation
Society of Naval Architects and Marine Engineers
Welding Research Council



# 11 Regulatory Research

**NRC research continued to address the effectiveness of radiation measurement techniques in 1979. Here instruments are set up in a reactor containment to measure neutron spectra and fluxes.**

The lessons for NRC research from the Three Mile Island accident range from the obvious need to study the plant itself to the suggestion that a rethinking of research priorities and policies is in order. By the end of fiscal year 1979, several key reactor safety research programs had been redirected and the Commission had under study a number of proposals to rechannel or supplement resources for TMI-related research.

It was clear, for example, that NRC's earlier concentration of research resources on design basis accidents, such as massive loss-of-coolant accidents (LOCA), must be re-evaluated and those resources either redistributed or augmented to permit greater emphasis on the smaller kinds of LOCAs and transients and on the operator deficiencies that characterized the TMI event. Better computer codes will be needed to permit study of the many variations that can occur in small LOCAs and transients and to predict with improved precision the behavior of plants in such situations. The availability of those codes also will aid in studies designed to enhance operator capabilities.

Soon after the TMI accident, NRC began to reorient its 1979 research programs to increase the emphasis in TMI-related areas, with some $12 million of 1979 program support funds rechanneled into this immediate research. In addition, fiscal year 1980 research funds are being reprogrammed, and supplemental funds of $20.9 million are being sought for the TMI-related effort. With the supplemental funds, the total TMI research effort in fiscal year 1980 will be about $55.4 million.

Except for the safeguards research program (discussed this year in the separate Chapter 5, "Safeguards"), all NRC research activities felt the impact of TMI. In water reactor safety research, the ongoing systems engineering projects using the LOFT and Semiscale facilities in Idaho were examined and, where appropriate, deferred to accommodate a new emphasis on the spectrum of accidents which lies between "design basis" accidents and "core melt" accidents. The accident at TMI, of course, was in this in-between spectrum.

TMI also pointed up the need for new research on coolant chemistry—particularly hydrogen evolution—after severe fuel damage, and on the impact of severe accidents on plant components, including the equipment qualification and testing techniques used to evaluate them.

In the area of human engineering, new or revised safety analysis codes now being developed will allow studies of simulators which could be used in training plant operators and of instrumentation and control room display and diagnostic equipment, toward improving operator response to the full spectrum of reactor accidents. In risk assessment research, new event trees are needed to address accident sequences involving severe core damage and to guide research into the effect of human error on the course of such sequences.

In parallel with all these studies it is necessary to continue investigations of improved plant design features, including decay heat removal and emergency core cooling systems, vented containment concepts, etc., and to intensify ongoing site safety research with new emphasis on population projections and other socioeconomic considerations.

These activities are described in the paragraphs which follow. In addition, details of NRC's implementation of the Congressionally mandated Research to Improve Reactor Safety are discussed in a required annual report summary at the end of the chapter.

## Water Reactor Safety Research

### SYSTEMS ENGINEERING

In early 1979, prior to the Three Mile Island accident, the NRC was beginning to phase down large-break LOCA research to increase the emphasis on small breaks and other reactor transients. The Blowdown Heat Transfer (BDHT) program at Oak Ridge, for example, was restructured to obtain thermal-hydraulic data under slower transient conditions than those previously studied, BDHT instrumentation was modified to cover the slower process, and natural circulation was studied in a small-scale apparatus. After the TMI accident, many projects were further modified to include other concerns about small breaks. Under the Semiscale program, expedited tests were conducted to simulate TMI-type accident conditions at the height of the crisis, and the facility is now undergoing major modifications to serve as the major small-break test facility in the United States. Numerous other programs are being similarly redesigned, and these are discussed below.

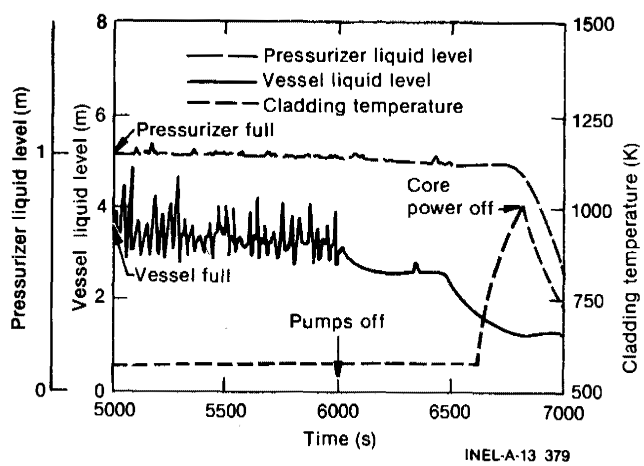
## Integral Systems Tests

NRC's major integral systems test programs are built around the Loss of Fluid Test (LOFT) facility, a 50 MWt experimental pressurized water reactor (PWR) which accomodates both non-nuclear and nuclear tests, and Semiscale, a smaller scale non-nuclear test facility. Both are located at DOE's Idaho National Engineering Laboratory.

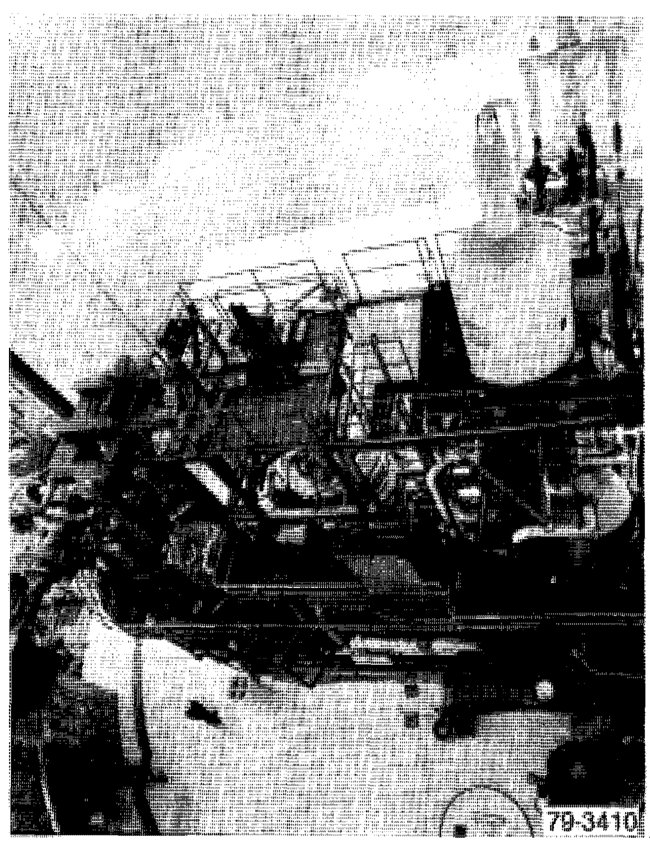
**Loss-of-Fluid-Test (LOFT) Program.** LOFT provides experimental information that is used to assess the analytical models that evaluate the safety of commercial reactors and their emergency core cooling systems (ECCS).

Significant LOFT events in 1979 included the second large-break LOCA nuclear experiment and a preliminary small-break LOCA experiment, both performed in May. (The first nuclear experiment was described on p. 182, 1978 Annual Report.)

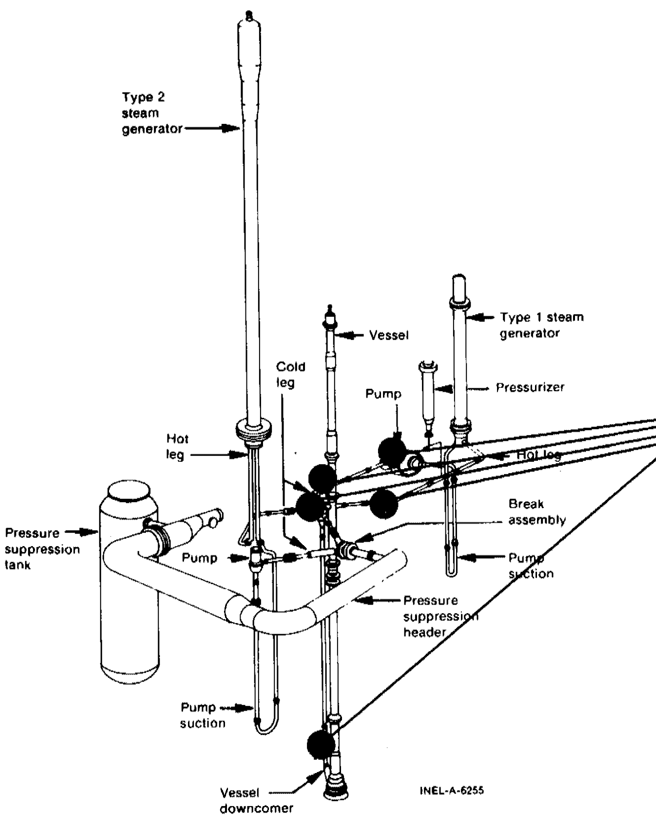
The large-break experiment on May 12, 1979, was similar to the first nuclear test in that it began with the rapid opening of blowdown valves to simulate the instant shearing of the major reactor coolant pipe. The experiment was conducted at a power level about 1/80th that of a commercial power reactor, yet the core heat generation rate and coolant pressures and temperatures were the same as those of a commercial reactor. The emergency-core-cooling system functioned as expected, although measured fuel cladding temperatures were lower than predicted. This was caused by a slug of water passing upwards through the core during system depressurization, prior to delivery of emergency coolant.

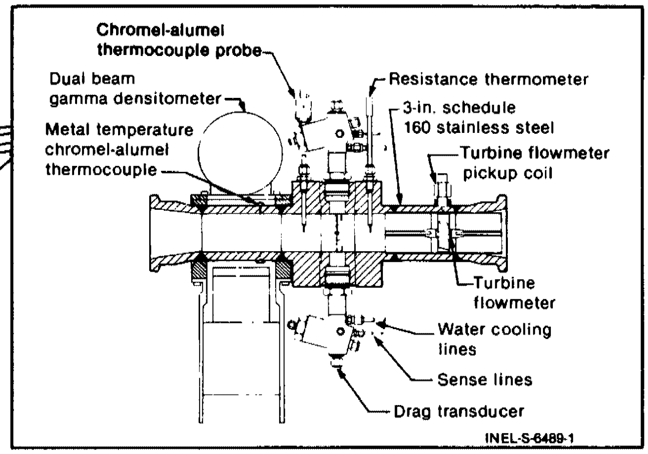



**NRC's Semiscale facility in Idaho, a small-scale, non-nuclear model reactor, was called on early in the accident sequence at TMI to simulate plant conditions as an aid in bringing the plant under control. This representation of a relief valve leak experiment shows the overlays for pressurizer and vessel liquid levels and for core cladding temperatures in the time-frame of 5000-7000 seconds (1 hr. 24 min.—1 hr. 56 min.) after trip. Oscillations shown for the pressurizer liquid level were caused by pump cavitation as indicated by the fact that the oscillations subsided when the pumps were turned off. Shortly afterwards, the vessel began draining and the vessel liquid level passed below midcore level. When this occurred, the cladding temperatures began to rise, indicating void formation even though the pressurizer liquid level was indicated to be full.**




**This overhead view of the LOFT test assembly at the Idaho National Engineering Laboratory (INEL) gives a good indication of the size of the facility. The apparatus contains a 50 MW(t), volumetrically scaled PWR system which permits NRC researchers to study the response of the engineered safety systems to loss-of-coolant accidents. Fiscal year 1979 saw a considerable reorientation of LOFT experiments toward the study of small-break accident phenomena such as those prevalent at Three Mile Island.**




**The Semiscale Mod-3 features a 3.658-meter core, upper-head injection capabilities, active broken loop components, a full-length U-tube broken loop steam generator, thru-core densitometers, and other advanced instrumentation with more than 700 potential measurement locations. One type of instrumentation is the intact loop spoolpiece, shown in cross section. The spoolpiece consists of a bidirectional turbine, a resistance bulb thermometer, a water-cooled pressure sense line, a fluid thermocouple and metal thermocouple, a two-beam cesium source densitometer, and either a single water-cooled transducer drag disc or a three-pin water-cooled transducer drag screen. It can be installed at various points along the intact loop, as shown.**

The first two large-break nuclear experiments yielded results that were quite similar. What differences there were stemmed from performing the second test at a core power 50 percent greater than that used for the first test, since this increased the coolant temperature rise in the core some 45 percent.

Temperature response of the fuel in the two large-break nuclear experiments, and the post-test core requalification, indicate there was no fuel damage in either test, hence, that fuel is available for future small-break tests. As noted earlier in this chapter, small-break LOFT experiments were planned to help answer questions related to the small-break accident at Three Mile Island.

The first such experiment, conducted on May 31 with the nuclear core at zero power, yielded data which were needed to prepare for the first nuclear small-break test in November, 1979. It consisted of a slow depressurization during which the ECCS performed as expected, and it provided new information on accumulator injection and natural circulation. The staff expects the small-break and anomalous transient tests, using more sophisticated control-room diagnostic instruments, to reveal new information about the response of a reactor facility to accidents similar to the one at Three Mile Island. The information should also suggest what operator actions are necessary to mitigate the consequences of such accidents.

Foreign participation in the LOFT program continued with representatives from Austria, the Federal Republic of Germany, Japan, the Netherlands, Switzerland, and Scandinavia actively participating in the program.

**Semiscale Program.** The Semiscale facility nuclear core is simulated by electrically heated rods whose geometry corresponds to the nuclear fuel rods in a typical PWR. Using the Mod-3 configuration (see p. 150, 1977 NRC Annual Report), the Semiscale facility was modified quickly during the recovery from the accident at Three Mile Island and used to simulate conditions in the plant in evaluating alternatives for securing the reactor. During critical phases of the TMI accident, Semiscale tests provided significant insights into the growth and movement of noncondensible bubbles, such as those caused by hydrogen. In post-accident analyses, Semiscale test are being used to model the sequence of events in continuing investigations of TMI-type behavior. In all, ten tests were done to furnish additional information on the Three Mile accident.

Other highlights of the Semiscale Program during 1979:

- Completed small break tests to assess licensing codes used to analyze a postulated accident. (This is the test series which was expanded to include the Three Mile Island accident phenomena).

- Completed baseline tests on the Mod 3 facility to establish performance parameters of the system in evaluating different phases of a LOCA. Nine published NRC reports summarized these tests.
- Tested the capability of a prototype optical probe to characterize water mixing behavior in the vessel downcomer. It was successful.
- Completed and published the results of studies scaling Semiscale MOD-3 to PWR, LOFT to PWR and Semiscale MOD-1 to LOFT. Results confirmed the rationale earlier employed in scaling LOFT from PWR and Semiscale from LOFT.
- Published the report on Semiscale tests and analyses performed in support of the Three Mile Island evaluation effort.

## Separate-Effects Experiments

Major test facilities for NRC's separate effects experiments are the Thermal Hydraulic (Blowdown) Test Facility (THTF) at Oak Ridge National Laboratory, Tennessee; FLECHT SEASET* at Westinghouse in Pittsburgh; the Two-Loop-Test-Apparatus (TLTA) and Sector Steam Test Facility (SSTF) at General Electric in California, and the steam-water mixing facilities at Battelle Memorial Institute in Ohio. Bench-scale tests and instrument development programs also are being conducted at several university laboratories. A table summarizing the location and capabilities of these facilities is on p. 151, 1977 NRC Annual Report.

**Two Loop Test Apparatus (TLTA).** TLTA tests during 1979 showed that the injection of emergency core cooling water cools the bundle during the early phases of the postulated accident more effectively than previously believed. Studies of later phases of the LOCA and tests of other, more probable accident sequences are planned for 1980 and beyond.

**Steam Water Mixing Tests.** The studies of steam-water mixing effects on the penetration of cooling water in models of PWR vessels conducted over the past five years in the small 1/15 and 2/15 scale models at Battelle Columbus Laboratories and at Creare, Inc. were largely completed during fiscal year 1979. A major result has been the accumulation of additional evidence of the conservatism in models used in the licensing process. Knowledge gained from these small-scale tests now will be used in the planning and analysis of full-scale penetration tests to be conducted in the Upper Plenum Test Facility in Germany. (See "3D Program," under "Research Support," later in this chapter.)

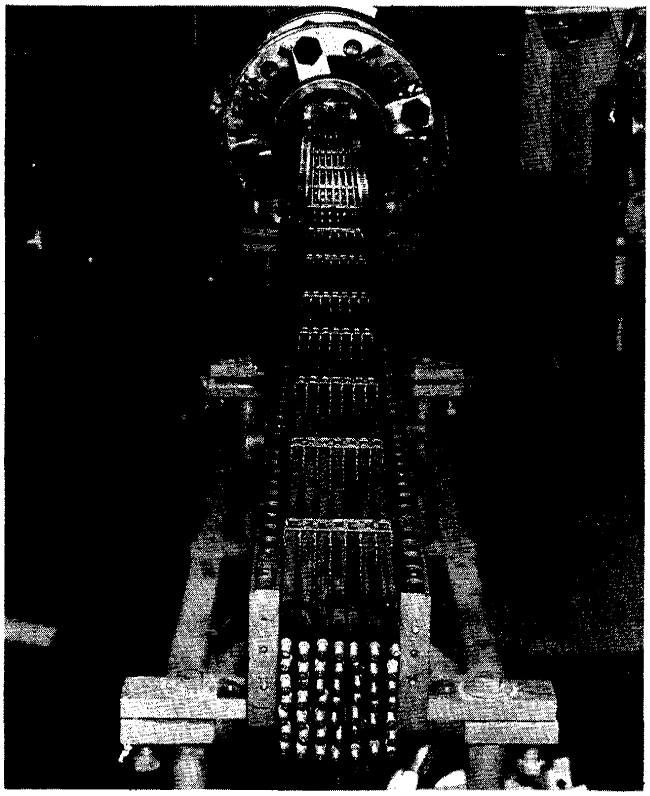
* Full Length Emergency Cooling Heat Transfer Separate Effects/System Effects Tests.

**Counter Current Flow Limit (CCFL) Refill/Reflood Program.** This project, sponsored jointly by NRC, the Electric Power Research Institute (EPRI), and General Electric, was initiated in 1979 and features the Sector Steam Test Facility (SSTF), a full-scale model of a 30 degree sector of a boiling water reactor (BWR). Tasks undertaken to date include investigations of the distribution of cooling water sprayed over the top of a core and how that cooling water penetrates fuel bundles. A modeling effort is in process to aid the development of a BWR version of the TRAC code (see "Code Development," below).

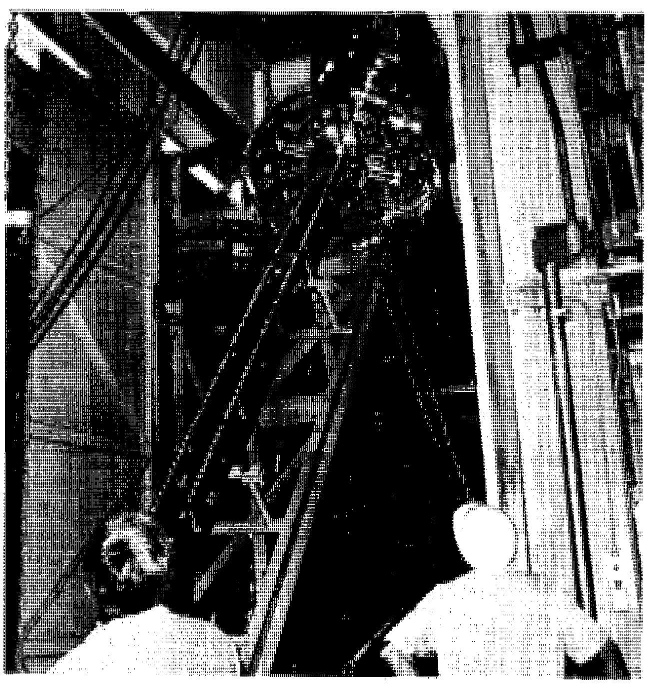
**FLECHT-SEASET Program.** This program is described in detail on p. 184, 1978 NRC Annual Report. In 1979, Westinghouse completed separate effects studies of both steam generator and core behavior during reflood, and continued the studies of core blockage using a small 21-rod bundle. The latter program will be expanded to use a 161-rod facility to investigate bypass as well as blockage geometry, and to permit comparisons of results with core blockage data from the Cylindrical Core Test Facility in Japan (see "3D Program," below). These tests should clarify some uncertainties regarding the conservatism of heat-transfer-rate and flow-regime criteria now used in licensing regulations. The system components (core rod bundles and steam generators) which have been studied separately will be integrated to investigate system interactions during various post-accident cooling tests. These system tests will focus on heat transfer mechanisms that are important in both large and small-break LOCAs. Steam heat transfer data from the 161-rod facility were used in TMI-related activities.

**PWR Blowdown Heat Transfer Program (BDHT).** The PWR Blowdown Heat Transfer Program at Oak Ridge was redirected in 1979 toward gaining a better understanding of fluid conditions in a PWR core during slower depressurization such as that characteristic of TMI. After modifications, ORNL ran a series of tests with the electrically heated 7x7 fuel-rod bundle (Bundle No. 2) to simulate various transient conditions. Initial tests using Bundle No. 3 were run in December 1979. This bundle features much-improved instrumentation and should produce a considerably improved understanding of pressure, flow, power, and temperature behavior. Tests with Bundle No. 1 were described on p. 183, 1978 Annual Report.

**Model Development Studies.** NRC continues to use small scale tests to study the various phenomena associated with steam-water flow. Each test is directed toward a particular effect to produce improved models or better data for input to code calculations. In 1979, Argonne National Laboratory completed experiments on the effect of controlled oscillations on reflood heat transfer; Massachusetts Institute of



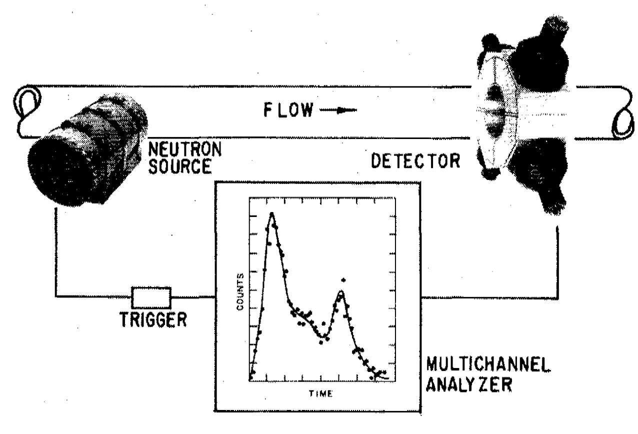
NRC's PWR Blowdown Heat Transfer Program at Oak Ridge, Tennessee, entered a new phase in 1979 as tests with fuel-rod simulator Bundles 1 and 2 were completed. Shown above is Bundle 1 after removal from the Thermal Hydraulic Test Facility (THTF). Below: Bundle No. 3, an 8x8 rod array, is lifted into a vertical position prior to installation in the THTF. Bundle 3 has more than 1200 instrument sites which can be monitored 20 times per second during experiments.



Technology completed work on spacer-grid effects on reflood and on natural circulation flows; Northwestern University continued its experiments on the rate of steam condensation, and Brookhaven National Laboratory continued its study of vapor generation rates in depressurization situations. In addition, Rensselaer Polytechnic Institute and Argonne National Laboratory undertook the development of models dealing with void fractions in reactor coolant flow during accidents, and the State University of New York at Stony Brook began work on models for the amount and character of droplet flow in and above a reactor core.

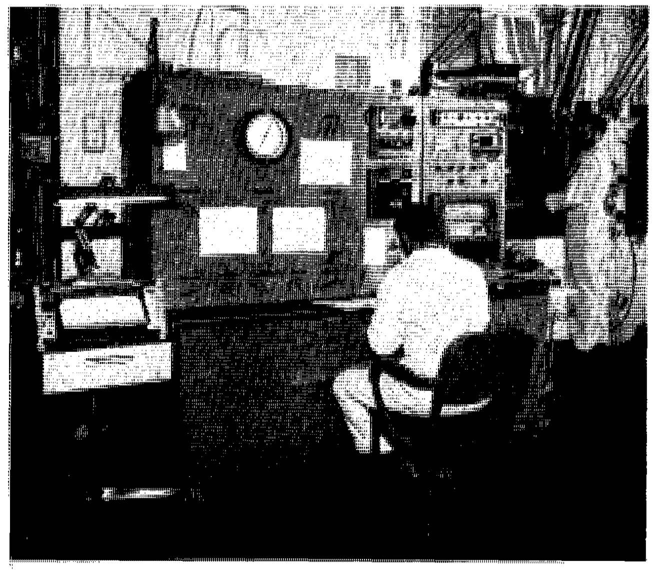
**Instrument Development.** Test facilities require many sophisticated instruments which are not commercially available. NRC's advanced instrumentation program, described in the 1977 and 1978 Annual Reports, is designed to fill this void. Progress during 1979 was reported in July at the Reactor Safety Instrumentation Review Group Meeting (NUREG/CP-0007). A summary of that report follows:

A technique called "pulsed neutron activation" was developed by Argonne National Laboratory to measure the velocity and density of steam/water flow. The technique will be used as a standard calibration for other instruments. Sandia Laboratories developed a portable neutron generator as part of the U.S. contribution to the international "3D" program (see below). Flow-measuring instruments, including turbine, drag disk and gamma densitometers have been developed and improved by INEL, and impedance

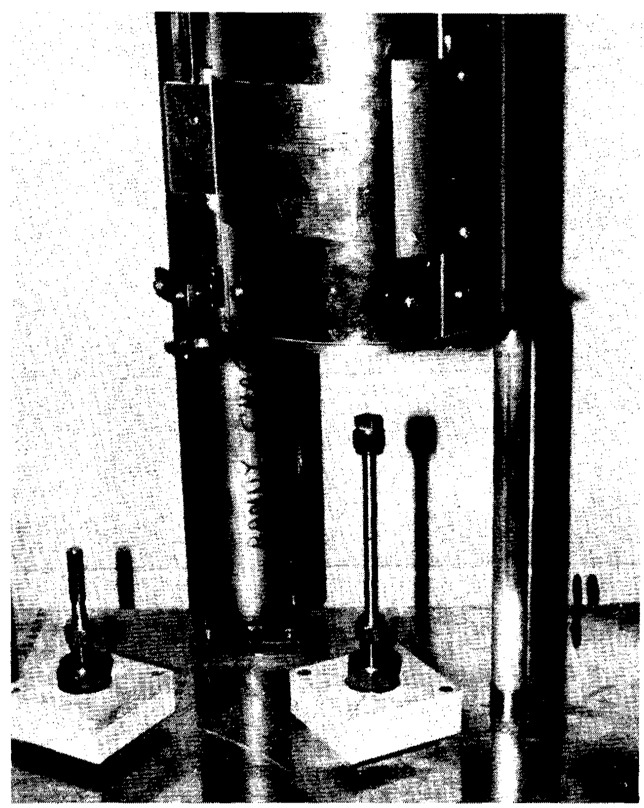



A July 1979 progress report on the NRC advanced instrumentation program described the pulsed neutron activation (PNA) technique for measuring velocity and density of steam-water flow. The technique involves: activating fluid with a neutron pulse from a portable neutron source (left); detecting activated nuclei a known distance downstream, and plotting the transit-time; determining average mass flow velocity and the average density of the fluid, and deriving mass flow from the latter two averages. The torus detector (right) features sodium iodide scintillators mounted to photomultiplier tubes. The plot at center is reproduced from PNA measurement of an air-water mixture. PNA equipment has been installed at the Semiscale and the LOFT facilities, and will be used in tests with other devices in 1980.

probes have been developed by ORNL. Other advanced instrumentation for flow measurements include optical probes, ultrasonic densitometers, and a stagnation probe, developed by INEL for measurements in LOFT. Rensselaer Polytechnic Institute, State University of New York at Stony Brook, Lehigh University, and Northwestern University continued their development of void fraction probes, film probes, special thermocouples, laser doppler techniques, and pitot tubes to better measure the thermal hydraulic properties of water and steam under accident conditions. So far, all of this advanced instrumentation technology has been applied in research facilities. Evaluation of its applicability to commercial power plants continued in 1979 and into 1980, particularly in light of an important lesson learned from the Three Mile Island accident—the need for instrumentation to measure water levels in reactor cores.



Tube burst test equipment at the Battelle Columbus Laboratory in Ohio is used to determine changes in fuel rod cladding burst properties resulting from irradiation in a commercial power reactor. Above, technician (seated) operates the tube burst console which programs, reads and records test parameters, while a technician (right) uses remote manipulators to load a radioactive specimen into the test stand in a hot cell. Below, a specimen is mounted in one of two test fixtures. A furnace for conducting elevated temperature burst tests is shown above the test specimen.



## FUEL BEHAVIOR

NRC's fuel behavior research programs in 1979 included cladding experiments, in-reactor tests, fuel meltdown and fission product transport tests, and fuel code development. These are described below. Data produced in these programs in previous years contributed significantly to improved understanding and analyses of the TMI-2 accident, particularly in the areas of oxidation, embrittlement and ballooning of the cladding and the release of fission gases from the fuel. As technical data from post-TMI analyses became available, the research staff began to recast some of those programs toward the study of fuel behavior when fuel rods are uncovered or severely damaged. In some tests, fuel assemblies will be allowed to boil dry to ascertain whether or how best they can be cooled. In others, the release and transport of fission products from damaged fuel will be studied. Some of the facilities and techniques which will be used for these TMI-related projects are described in the following summary of ongoing fuel-behavior research activities.

### Cladding Experiments

**Multirod Burst Test (MRBT) Program.** The MRBT program at ORNL has two main objectives. The first is to better define the deformation behavior of unirradiated Zircaloy cladding under conditions postulated for a loss-of-coolant accident (LOCA). The second is to provide a data base for use in assessing geometrical changes which occur in fuel rod cladding and the extent of coolant flow restrictions that the changes (such as ballooning and rupture) might produce.

Data accumulated in the MRBT program have come from unheated-shroud single-rod tests and from heated-and unheated-shroud multirod (4 x 4 bundle) tests in which the shroud temperature at the time of burst was no closer than 80°C from the fuel bundle temperature, and in which the rods were not restricted in their outward movement. The importance of closely simulating these thermal and confinement factors is not clear at this time, but the data base will be increased by investigating these concerns in tests. One conclusion that has held up throughout the testing is that, at a given temperature, the temperature gradients determine the extent of deformation, i.e., the more uniform the temperature distribution, the greater and more uniform the deformation.

**Mechanical Properties of Zircaloy.** Work in 1979 at Argonne National Laboratory produced information which will lead to better criteria for predicting damage to cladding embrittled by oxidation at high temperatures. These new criteria are stated in terms of measurable mechanical properties which can be used to define the oxidation limits that will permit the cladding to survive such conditions as thermal shock, impact and physical damage, and the loads on cladding resulting from the interaction of reflooding water and steam.

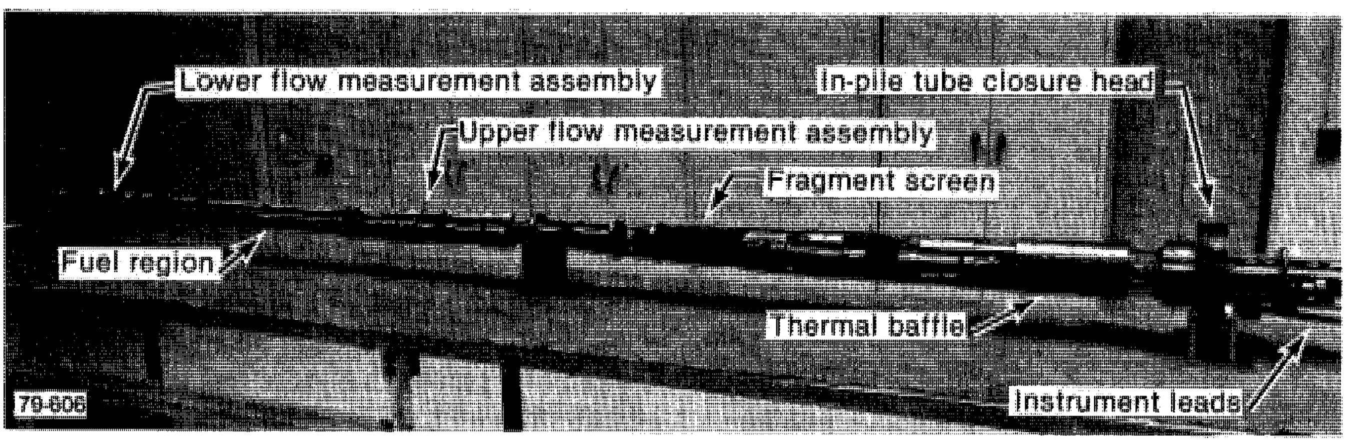
## In-Reactor Testing

**Power Burst Facility (PBF).** At the PBF in Idaho, research continued in 1979 on the behavior of fuel rods under various operating and accident conditions. (See page 154, 1977 Annual Report.) The "reactivity initiated accident" (RIA), described on p. 188, 1978 Annual Report involves a burst of power generated when a control rod is ejected from the core. Early in fiscal year 1979, two RIA experiments were performed in the PBF test reactor. In the first test, two fresh fuel rods and two pre-irradiated fuel rods were exposed to a power burst that in earlier experiments had done substantial structural damage to the test fuel. In some regions of the pre-irradiated rods the fuel swelled more than expected, and the swollen fuel and debris from adjacent parts of a rod blocked the flow shroud more than expected.

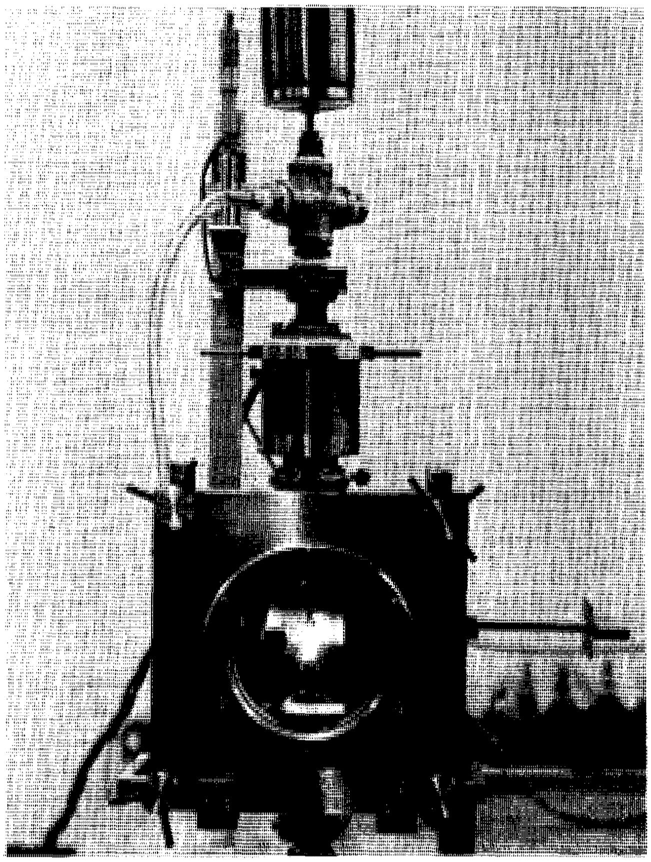
The second PBF RIA test was performed with four pre-irradiated rods previously exposed to a lower power burst near the level at which cladding could be expected to fail and to release fission products. Only one of the four rods failed, but a series of small longitudinal cladding cracks appeared which resembled the kind of pellet-cladding interaction which usually induces cladding failure. Since this rod had not been opened after pre-irradiation, and the companion rods had been, these effects will be studied in future RIA damage threshold tests using unprocessed pre-irradiated rods.

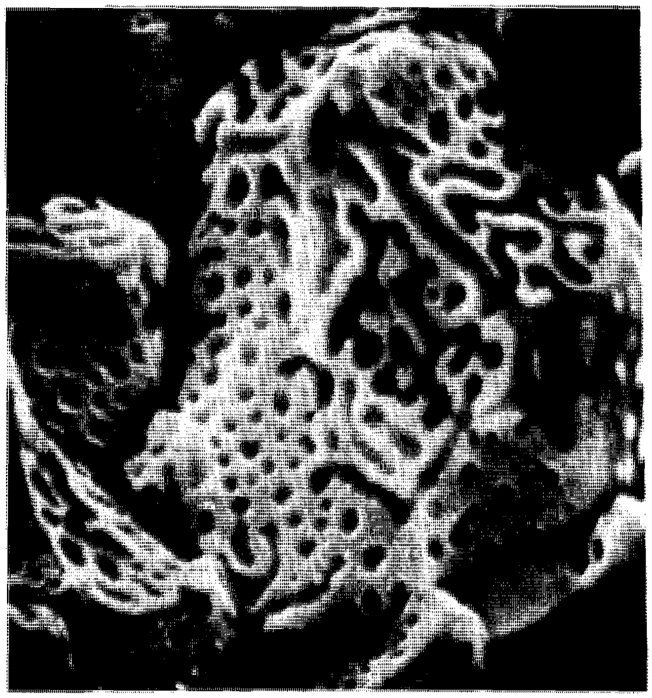
Later in the year, three LOCA blowdown tests were performed to confirm that planned LOFT tests with unpressurized rods could be run without jeopardizing the fuel bundles.

The final two PBF tests of 1979 used both fresh and pre-irradiated rods, pressurized to match the fill gas pressures of commercial rods when new and at or near the end of life, to examine the "ballooning" associated with cladding rupture with the peak temperatures in the 1050°K-1500°K (1430°F-2240°F) range. In one test, circumferential ballooning as high as 50 percent was observed. (NOTE: More than 70 percent ballooning of four adjacent rods is required to block a flow channel in a commercial power reactor, and some cooling water flow through any cladding ruptures is possible even then.) The second test was performed in the last months of fiscal year 1979, and the rods had not been examined at year's end.




**This LOFT Lead Rod Test Assembly, fabricated by EG&G Idaho, Inc. for testing in the PBF, contains four individually shrouded and instrumented PWR-type fuel rods. The apparatus is used to collect data for the assessment of predicted fuel behavior during loss-of-coolant accident testing in LOFT.**





A combined experimental/analytic program at Argonne National Laboratory, aimed at better understanding the effects of fission gas releases on irradiated fuel, employs this direct electric heating apparatus (left) and predictive models of the GRASS family of computer codes. In the experimental effort, fuel pellets are heated to temperatures similar to nuclear heating, then studied in various ways, including the use of scanning electron microscope photography. The photograph reveals an interconnected network of tunnels on the grain boundaries of irradiated $UO_2$ fuel.

## Fuel Meltdown/Fission Product Release and Transport

**Fission Product Release and Transport** research continued at two laboratories:

- Oak Ridge National Laboratory completed an experimental program to determine the quantity and type of fission products which might escape during reactor accidents. In four high-temperature tests, segments of irradiated commercial fuel rods were subjected to peak temperatures (1200° to 1600°C) well above those predicted for successful control of a LOCA. These tests, conducted in a flowing-steam environment, indicated that the release of fission products iodine and cesium increased tenfold and of krypton by a factor of five within the temperature range 1350°C to 1400°C. The results of these tests were helpful in estimating the fuel temperatures attained during the TMI-2 accident.
- Battelle Columbus Laboratories published a user's manual for the TRAP-MELT code, which models the transport behavior of fission products in primary coolant systems of LWRs in various accident conditions, including fuel meltdown.

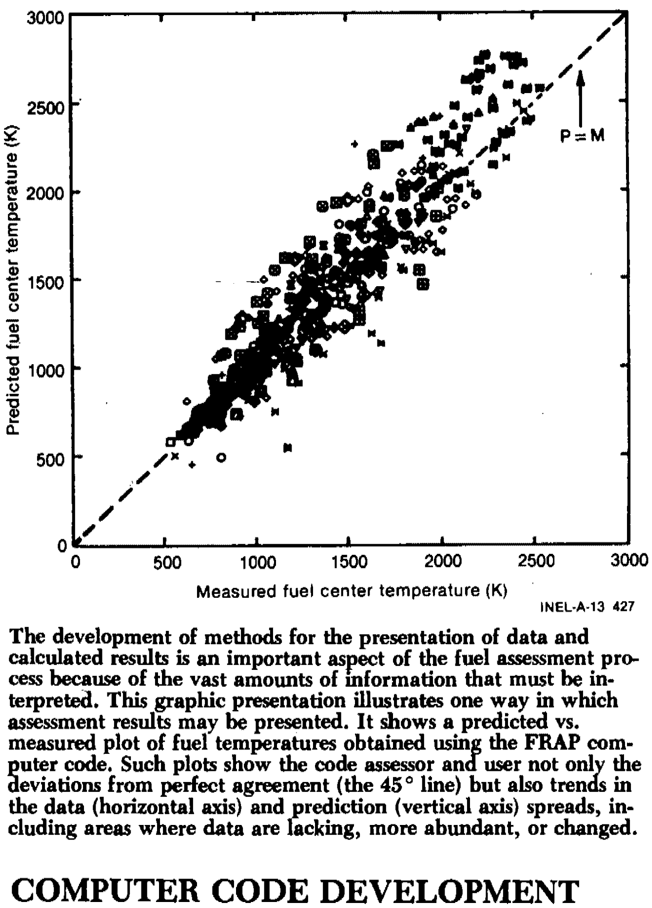
**Core-Melt Research.** Sandia Laboratories continued investigating the potential for thermal explosions, if molten core materials were to contact water, with a series of experiments dealing with the efficiency of converting the thermal energy of the melt into mechanical energy. In a series of 48 tests using molten iron/alumina with masses up to 27Kg, the maximum efficiency was measured at 1.34 percent, a factor of about 20 less than the maximum theoretical efficiency for thermal interactions.

Another Sandia Laboratories investigation explored the interaction of molten core materials and concrete, producing important data on the gases and aerosols generated, the penetration rate of the melt into the concrete, and the rate of fission product evolution from the melt. This information was used to develop an advanced melt/concrete computer code called CORCON.

## Fuel Behavior Codes

**Fuel Rod Analysis Program (FRAP).** Information from the PBF, LOFT (see above) and Halden Reactor Project (see p. 189, 1978 Annual Report) programs is used in developing and assessing NRC codes "FRAP-CON," used for the steady-state analysis of fuel rod response during normal reactor operation, and FRAP-

T, used for transient analysis of fuel rod response during off-normal reactor conditions. During 1979, improved models and correlations gave FRAP-T a capability to more accurately predict fuel rod temperatures, deformations, and possible failures during all phases of a LOCA. A report on the revised library of materials properties (MATPRO-11) needed by the two codes was issued (NUREG/CR-0497), and at the end of the reporting period, both FRAPCON-1 and FRAP-T5 were available at the National Energy Software Center for distribution.




**The development of methods for the presentation of data and calculated results is an important aspect of the fuel assessment process because of the vast amounts of information that must be interpreted. This graphic presentation illustrates one way in which assessment results may be presented. It shows a predicted vs. measured plot of fuel temperatures obtained using the FRAP computer code. Such plots show the code assessor and user not only the deviations from perfect agreement (the 45° line) but also trends in the data (horizontal axis) and prediction (vertical axis) spreads, including areas where data are lacking, more abundant, or changed.**

## COMPUTER CODE DEVELOPMENT

### Code Development

Research results from the reactor safety research program are used to develop computer codes and assess their accuracy in analyzing nuclear plant accident. Thus, code development is a focal point for reactor safety research results.

The reference computer code (RELAP) for analysis of reactor system accident behavior is in worldwide use. Development and improvement work on RELAP is being phased out, and the major code effort now is addressed to the advanced system code, TRAC. TRAC contains better and more complete descriptions of physical processes, which makes the code less susceptible to scaling errors in extrapolating from results of small-scale research tests to the behavior of full-scale power reactors. In 1979, the initial version of the TRAC code (applicable to pressurized water reactor) was sent to the National Energy Software Center. Work to improve that version and work on the boiling water reactor version continued into 1980. In addition, the TRAC system code is being linked with more detailed component codes such as fuel element and fuel bundle codes.

Development of the WRAP code (see below) package used in licensing audit calculations for LWR plants was completed.

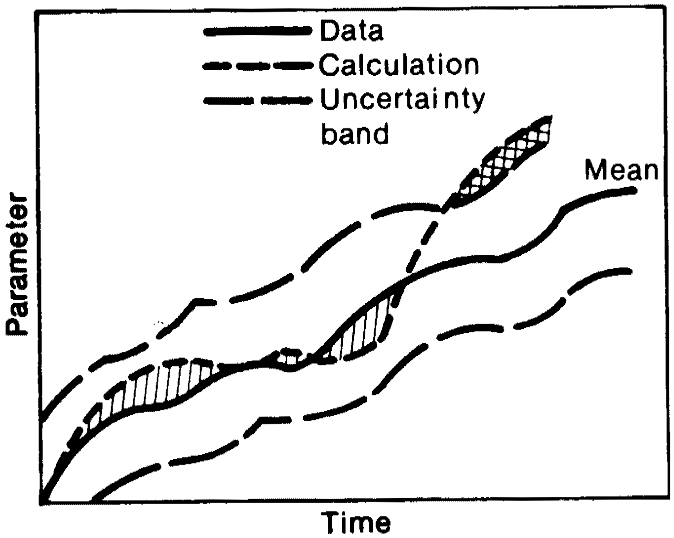
**RELAP-4.** MOD 7 of this system's LOCA code has been applied extensively in analyzing the TMI accident, as well as in performing TMI sensitivity studies, and in auditing small-break calculations developed by various steam system suppliers. The primary conclusions from this analysis were as follows:

- The RELAP-4/MOD 7 computer code addresses most of the phenomena thought to occur in a full-scale pressurized water reactor under TMI-type conditions.
- A detailed input model was needed to calculate the phenomena occurring during the TMI accident, and this resulted in large computer calculational times.

**WRAP.** The Water Reactor Analysis Package (WRAP) is a set of evaluation-model computer codes linked together to perform accident calculations in the licensing review process. WRAP-BWR model was tested in 1979 in a calculation of a typical BWR plant and on test data from the Two Loop Test Apparatus. A computational module for the refill portion of a postulated LOCA was added to WRAP-PWR model, and the complete PWR package was exercised in a calculation of a typical PWR plant, and on test data from the LOFT facility.

**TRAC.** The Transient Reactor Analysis Code (TRAC), an advanced best-estimate computer program, is designed to predict the thermal and hydraulic responses of LWRs to LOCAs and other transients. (For a detailed description of TRAC, see Page 159 of the 1977 Annual Report).

TRAC was used extensively in analyzing the Three Mile Island Accident. The scenario of events and accident parameters—to the extent they were known—were modeled, and these factors were then varied in additional calculations to learn what would have happened had different decisions been made during the accident. The TRAC computations indicated that the plant safety systems themselves were adequate to prevent the core from being uncovered and causing core damage, and that if the high pressure coolant injection system had been left on, the core would have stayed covered with water. Other activity involving




One of the objectives of code assessment is to determine the accuracy of a code prediction (or the capability of the code to make a prediction). The procedure illustrated here, which is under development at EG&G Idaho, Inc., shows experimental data in terms of mean value and uncertainty bounds. The calculation, a single-value curve, is compared with the data range. The accuracy of the calculation is determined by integrating (summing) the deviations of the calculation from the mean and normalizing (dividing) the total error by the width of the uncertainty band of the data. Shaded areas illustrate ways in which the calculation may deviate from the data.

TRAC included publication of Volume II of the TRAC-P1A manual. It details test data analyzed by the code during its development.

Work also was started on a much faster version of TRAC, TRAC-PF1, scheduled for release in 1980. This version can reduce the number of computational nodes required for a given calculation, thus speeding the running time. Also, a revised numerical method of solving differential equations will accommodate larger time-steps during slower transients similar to the one that occurred at TMI.

TRAC calculations were performed both in pretest predictions and post-test analyses on tests performed in the LOFT, SEMISCALE, and other experimental U.S. facilities, and in support of German and Japanese facilities as part of the International 2D/3D Refill/Reflood Program.

**COBRA.** The COBRA-TF multidimensional vessel code (described on p. 191 of the 1978 Annual Report) has been linked as a vessel module to the TRAC code to obtain self consistent vessel boundary conditions during the course of a postulated accident. The linked code has been named COBRA-TRAC.

**RAMONA-III.** This code, developed by SCANDPOWER, Kjeller, Norway, is being employed at Brookhaven National Laboratory (BNL) in calculating BWR operational transients. The code accommodates up to 100 parallel thermohydraulic channels with different flows, voids, and conditions. While working with RAMONA-III, BNL also developed improved models for analyzing fuel rods, jet pumps, and steam lines. In a companion move, NRC discontinued its development of the advanced system code "THOR" (See p. 193, 1978 Annual Report) following a review of three candidates (THOR, RELAP-5, and simplified TRAC) for a fast running systems code.

## Code Assessment

A number of codes were assessed during the fiscal year. An assessment of RELAP-4/MOD 6 was completed at INEL with the publication of an assessment report in December 1978, and an addendum report in September 1979. An assessment procedure and user guidelines were developed, and multiple comparisons between calculations and data were obtained for the Semiscale MODs 1 and 3, LOFT, PKL, Marviken, PBF, THTF, FLECHT and FLECHT-SEASET facilities. In systems comparisons, clad temperature calculations tended to be too high when compared to experimental data, indicating need for improvements in calculations of certain heat transfer and fluid flow pheonmena. These are being addressed in the RELAP-4/MOD 7 code version.

A coordinated assessment program for the TRAC-P1A code was initiated at LASL, BNL and INEL, in which particular emphasis is placed on "blind" predictions—predictions in which the result of the experiment is not known to the analyst at the time of preparation of the code input. As an example, all LOFT nuclear test applications in the assessment of RELAP-4/MOD 6 and TRAC-P1A have been done using blind predictions.

The assessment of the TRAC code is a long-term project and a plan has been prepared based on three types of tests: Integral Systems Tests (for example LOFT and Semiscale), Separate Effects Tests (separate components of the reactor system), and Basic Tests (designed to explore basic thermohydraulic phenomena that are of importance during a LOCA). Specific experiments have been selected and many of these were run during 1979. The assessment program will continue in 1980.

# METALLURGY AND MATERIALS

NRC's Metallurgy and Materials Research Program is designed to provide confirming information regarding the integrity of water reactor vessels, tubes and piping systems. It encompasses research in fracture mechanics including design criteria for fracture prevention, flaw detection and evaluation, the growth and arrest of cracks; radiation embrittlement; corrosion, and the uses of acoustic, electro-chemical and other techniques to help assure component integrity.

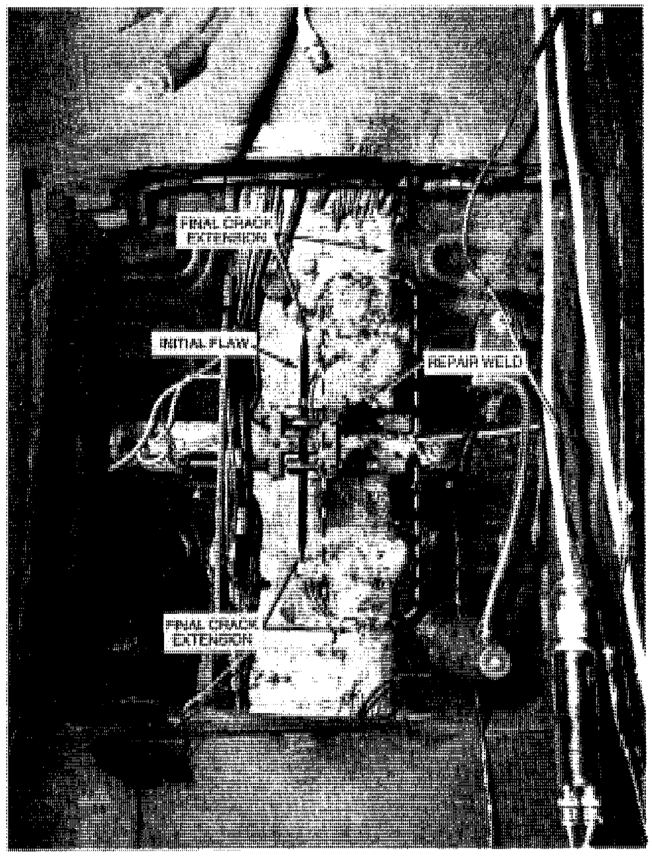
Large vessel tests in 1979 continued to validate the design conservatisms in reactor pressure vessels. In addition, fracture mechanics research demonstrated the integrity of weld repair techniques and identified conditions under which thermal shock will not propagate cracks.

The impact of Three Mile Island on these programs will be known in detail once cleanup plans for the plant are developed. Steps already are being taken, however, to integrate NRC research requirements with those of DOE and other agencies in post-accident examinations of the containment building and safety systems and components.

## Fracture Mechanics

Elastic-plastic Fracture Mechanics. Reactor vessel steels are designed to be extremely tough in reactor thermal environments so that sudden, rapid, brittle fracture will not occur. If there is a crack in the vessel wall, however, possibly caused by corrosion, there is a chance that the crack will grow through a process called ductile tearing. Such tearing is "stable," that is, it will continue slowly if the internal pressure is high enough to sustain it. The tearing ceases at lower pressures. It is important to study the nature of these phenomena since conditions may develop where pressure loads are high enough and cracks have become large enough to make the crack growth "unstable," i.e., the ductile tearing propagates at a high rate and endangers the integrity of the vessel or pipe.

The "tearing-instability" method used for structural analysis, which was developed at Washington University in St. Louis, has been found useful in determining whether pipe and pressure vessels cracks will grow in a stable manner when subjected to severe loads such as those generated by earthquakes, water hammer effects, etc. It also will permit characterization of the resistance to crack growth of irradiated materials. These phenomena are discussed below. A parallel effort at the Naval Ship Research and Development Center in Maryland is the development of a new technique for measuring "J-resistance" curves (fracture toughness resistance to ductile tearing for piping and vessel steels), using the tearing-instability concept.




**Researchers at Oak Ridge have performed simulated weld repairs on thick-walled cylinders and pressure vessels as part of NRC's Heavy-Section Steel Technology Program to assess a method recommended in American Society of Mechanical Engineers (ASME) codes. The method is designed for in-service repair of components when it is not practical to apply the high-temperature stress-relieving process usually employed. The pressure vessel shown here was pressurized to failure with a large crack deliberately implanted where crack propagation was most likely under combined repair and pressure-loading stresses. The test demonstrated the importance of residual stresses to vessel integrity, but it also showed that failure under complex loading conditions can be predicted, a finding which improves the system of techniques available for evaluating pressure vessel safety. The tests have also demonstrated that the structure itself, under some circumstances, will arrest rapid propagation of a crack.**

**Fracture Toughness of Irradiated Materials.** During the life of a power-reactor pressure vessel, it is bombarded by high-energy neutrons, which reduces the fracture toughness of the vessel steel. An NRC program at Oak Ridge attempts to quantify loss of toughness to establish margins of safety against fracture. This has involved the irradiation of several hundred specimens of reactor steels and weld materials, ranging in thickness from one-half inch to four inches. Approximately 50 specimens of weld material have been tested, and it is anticipated that the results will influence licensing decisions and safety evaluations of commercial reactors, as well as contribute to the resolution of the generic issue addressed in Task A-11 of the unresolved safety issues. (See Chapter 3.)

**Thermal Shock.** The fifth in a series of experiments at Oak Ridge was run in 1979 to demonstrate the structural integrity of a reactor pressure vessel when subjected to the emergency injection of cold water during a postulated accident. This experiment indicated that an existing crack was arrested, without penetration of the outer cylinder surface. Further tests with large diameter cylinders to more closely approximate the performance of full-sized reactor vessels will be conducted in 1980.

**Reliability of Piping Systems.** The continuing incidence of cracks in piping has demonstrated the need for research to quantify and upgrade pipe-system reliability. Programs initiated during the year include development of toughness and crack growth rate data on typical piping; and estimates of the size, distribution and detection efficiency of flaws and reevaluation of stresses, loads, and potential pipe-break locations. At year's end, data were being combined for a probabilistic analysis of NRC pipe design and operation criteria and for reconfirming the reliability of nuclear piping systems.

## Steam Generator Tube Integrity

**Stress Corrosion Cracking of PWR Steam Generator Tubing.** The NRC research program initiated at Brookhaven National Laboratory in 1978 to develop a data base for predicting stress corrosion cracking in steam generator tubing was supplemented in 1979, using tubing that cracks abnormally quickly in addition to the regular tubing used in actual steam generators. This supplemental testing will permit more rapid validation of the predictive models now being developed. Also in fiscal year 1979, other baseline data were under development which would allow predictions of cracking as a time-function of temperature in high purity water. Tests of various water chemistry effects also were started.

**Integrity of Flawed Tubing.** Battelle Pacific Northwest Laboratory's program to investigate burst/collapse strength of flawed steam generator tubing contributes to Tasks A-3, 4 and 5 of the unresolved safety issues. (See Chapter 3.) The first of three major phases was completed in 1979 with results showing that even grossly defected tubing retains considerable strength. Although some eddy current tests showed that the techniques used may not accurately characterize the tubing defects, the overall conclusions were that pertinent NRC criteria are appropriately conservative. Research thus far has used mechanically defected tubes, but plans were being made at the end of 1979 to use service defected tubes.

Research at ORNL to improve eddy current test methods, and instrumentation for use during inservice inspections of tubing has already produced computer codes that can optimize examination instrument designs. Prototypes of these designs were being evaluated on machine-flawed samples at the end of the period.

## Radiation Embrittlement

**Irradiation-Anneal-Reirradiation Program.** Research on periodic heat treatment (annealing) of irradiated pressure vessel steel has been reported in NRC Annual Reports for several years (see p. 196, 1978 report). Work performed this year confirmed that periodic, one-week heat treatments of irradiated steel at 200°F above the normal operating temperature will give continuing relief from radiation embrittlement, but the same treatment at only 100°F above normal gives only temporary relief. The implications of heat treatment at this temperature were under study at the close of 1979.

**Irradiation Surveillance Dosimetry.** To assess the significance of radiation embrittlement caused by neutrons emitted by reactor fuel during operation one has to know the numbers and damaging effects of the neutrons. At Hanford Engineering Development Laboratory in Washington, with major contributions from Oak Ridge National Laboratory and the National Bureau of Standards, a benchmark test facility simulating a pressure-vessel wall was completed in 1979 and experiments dealing with the numbers of energy levels of neutrons at specific locations were conducted. The results showed very good agreement between measurements and calculations, and this will aid in future estimates of reactor vessel lifetimes, and in the preparation of standards for calculating and predicting neutron flux and spectrum in operating power reactors.

## Flaw Detection

**Acoustic Monitoring.** Continuous on-line monitoring of reactor components during operation using the acoustic emission technique has been under study at Battelle Pacific Northwest Laboratories (PNL). Acoustic emission "hears" the waves produced by a growing crack in a reactor component, and by using several sensors distributed over the component, the crack can be located. In 1979 capabilities also were developed to evaluate the severity of a flaw and to distinguish different flaw types from acoustic emission signals. In 1980, these developments will be examined in a large structural test that will simulate all important reactor operations. Later a commercial reactor will be instrumented for final validation and acceptance of the techniques.

The GARD, Inc., program to develop acoustic emission techniques for the detection and location of flaws produced during welding (p. 197, 1978 Annual Report) was completed in 1979. The resulting field equipment will detect and locate flaws during welding, as well as distinguish among different types of flaws and evaluate their severity.

**Vibratory Excitation.** Another continuous monitoring technique under study utilizes internal friction of a material to detect certain kinds of cracking in stainless-steel reactor pipes. This technique, developed by Daedalean Associates of Woodbine, Md., depends on identification of microscopic changes in the material as the precursors to cracking.

# RESEARCH SUPPORT

Research Support in 1979 encompassed Operational Safety Research work, and Program Management Support and Technical Support activities. Operational safety research includes fire protection, qualification testing evaluation, noise diagnostics, human factors, and assessment of the behavior of safety and relief valves under certain postulated accident conditions (now particularly oriented to a TMI-type of accident). Program Management Support is provided for certain areas of international research. Technical Support includes the dissemination of safety research information and computer codes.

## Operational Safety Research

**Fire Protection.** NRC's fire protection research was substantially reoriented in 1979 with project research personnel shifting to work on a series of full-scale fire tests. These tests will mock up portions of actual plant facilities (such as a cable spreading room) and subject them to design basis fires to test detection and suppression systems. The first full-scale test will take place in 1980.

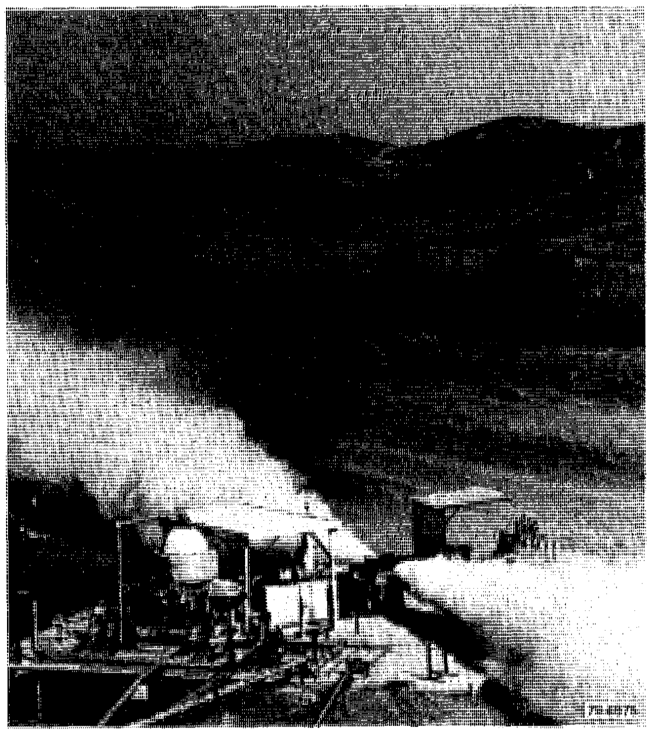
In a fire confinement study, six tests dealing with the relationship between ceiling/wall materials and a fire in a tray of electrical cables showed that the closer the tray is to ceiling/wall corner, the hotter the cable will be, and the greater its weight loss. Also there is more chance of fire propagation between cable trays.

The new fire suppression test facility at Sandia Laboratories in New Mexico was finished in 1979, and checkout tests have been scheduled. The facility will be used to study the effectiveness of fire suppression agents (Halon, water and carbon dioxide).

**Qualification Testing Evaluation.** In 1979 Sandia Laboratories completed a new facility to test qualification methodologies. The facility features irradiation, steam, chemical, pressure and temperature capabilities. A series of facility checkout tests as well as the first test of electrical cable connectors from a commercial nuclear power plant already have been run. Plans are under way to test additional safety-related equipment under LOCA and MSLB conditions. The adequacy of the test methodology will be evaluated in these tests. In another study completed under this program, final calculations were made on the capability of a cobalt-60 radiation source to simulate LOCA accident radiation effects.

Other research activities included the continuing assessment of methods used to predict the behavior of electrical cables when "aged" by exposure to heat, humidity and radiation. A model has been developed that can be used to predict useful material life by accelerated thermal and radiation aging, including the synergistic effects of their combined application. Plans have been made under this program for participation in the examination of safety-related electrical equipment from TMI-2.

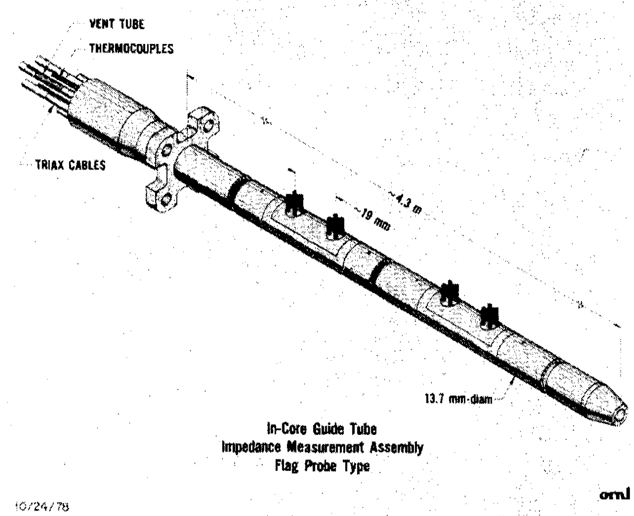
**Noise Diagnostics.** Noise diagnostics research aims to improve the monitoring of the behavior of a nuclear power plant by measuring the noise associated with such signals as vibrations, power oscillations, loose parts, etc. In 1978, researchers at Oak Ridge completed a code that statistically combines noise sources and that has done a reasonably good job of predicting power/noise spectra (signal as a function of frequency) in a nuclear power plant. It is known that anomalies such as instrument tube vibration or core barrel motion changes the spectrum of reactor noise. Once a noise change is understood, one can use computer models to ascertain the abnormality causing it.

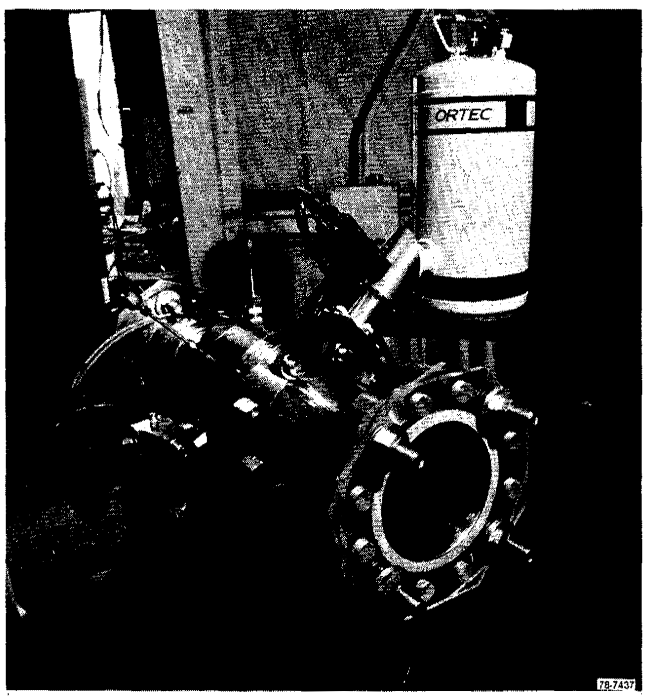


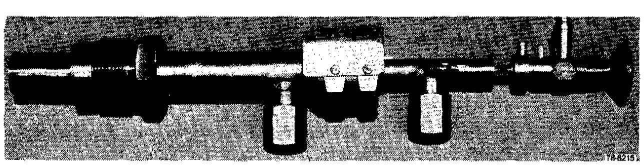
Government testing in high technologies may be conducted at very elaborate and expensive facilities, such as the FFTF, or on test-beds designed for specific short-term experiments such as the apparatus shown here. The photo, taken during the conduct of an EG&G Idaho, Inc., test, shows the government-owned Transient Test System (TTS) located at Wyle Laboratories, Norco, Cal., a facility designed to test flow-measurement instrumentation under transient two-phase flow conditions identical to those of the LOFT system.

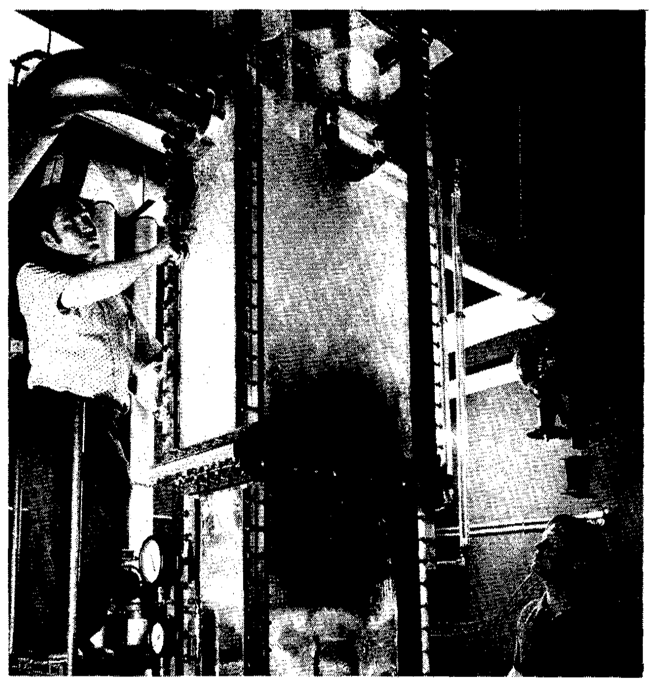
**Human Factors.** In 1979, researchers at Oak Ridge completed the first phase of a study of operator response to accidents. Study of events at several plants show that adjustable time margins may provide the most reasonable criteria for operator response, since experienced operators tend to overestimate their ability to respond quickly. This work now includes various post-TMI studies of the use of control-room simulators in training reactor operators. A report including recommendations for the future use of simulators will be published in 1980.

**Safety/Relief Valve Behavior.** A literature study to determine the flow behavior of safety and relief valves under ATWS conditions (See Chapter 3) was published in March 1979 by the Energy Technology





NRC programs at the Idaho and Oak Ridge National Laboratories include the development of instrument systems for international (3D) activities in Germany and Japan. Some instruments and testing activities under these programs are: (1) An in-core guide-tube impedance measurement assembly developed at Oak Ridge for the PKL facility in West Germany. (2) Testing the assembly on a steam-water test stand at Oak Ridge. (3) Instrumented spool-piece, featuring 11 transducers, developed by EG&G in Idaho for the PKL. Four such devices have been sent to Germany and eight to Japan for the Cylindrical Core Test Facility. (4) Design development and calibration of instrumentation for the German Upper Plenum Test Facility (UPTF) entails testing in a "see-through" representation of the UPTF at Oak Ridge. A similar facility is used at EG&G. (5) A video camera can be coupled to the EG&G Imaging Optical Probe to record the flow of high-temperature and pressure steam-water in nuclear reactor piping.

Engineering Center at Canoga Park, California. Results appeared to confirm the conservatism of the present model (i.e., that the flow through the valve is greater than predicted in the NRC analysis model). However, tests may still be needed.

### Program Support

As part of NRC's participation in the 3D international program (see p. 199, 1978 Annual Report), researchers at the Idaho National Engineering Laboratory and the Oak Ridge National Laboratory began the loan of advanced instruments to West German and Japanese counterparts for use in their facilities, and scientists at Los Alamos began using the TRAC code to analyze those test facilities. During 1979, the Japanese completed the large 2000-rod electrically heated Cylindrical Core Test Facility, and checkout tests have been completed. Work continues in Japan on the Slab Core Test Facility, expected to be operational in 1980. Both facilities will provide improved understanding of steam/water behavior in a simulated nuclear core. In Germany, researchers continued design work on the Upper Plenum Test Facility, which will experimentally model the behavior of steam and water droplets in the upper plenum of a PWR during refill and reflood stages of a postulated LOCA. This cooperative program provides NRC with better understanding of the physics of accident behavior and an advanced accident-prediction code (TRAC) at about one-third the cost of doing the whole job alone.

### Technical Support

Under the Technical Support Program, NRC shares sponsorship with the Department of Energy of the Nuclear Safety Information Center at Oak Ridge and the National Energy Software Center at Argonne in Illinois.

**Nuclear Safety Information Center.** The Nuclear Safety Information Center (NSIC) at Oak Ridge provides a focal point for safety information on reactors and other nuclear facilities. Technical experts who are cognizant of the literature in each area of specialty provide replies to questions from NRC, DOE and the nuclear community. Information is provided to non-exempt customers on a cost recovery basis. Seventeen reports, in addition to the bimonthly review, *Nuclear Safety*, were published in 1979. The NSIC also gave significant support to the ACRS, and its consultants during its review of the Licensee Event Reports.

**National Energy Software Center.** The National Energy Software Center (NESC) at Argonne National Laboratory is partially funded by NRC to make NRC-sponsored computer codes available to the public. Between October 1978 and September 1979 the Center distributed 1180 copies of the software packages in response to requests from NRC and DOE offices, the Nuclear Energy Agency Data Bank in France, other U.S. government agencies, universities, and commercial and industrial organizations. On September 30, 1979, the NESC list of software packages available for distribution contained 45 items (codes) specifically sponsored by NRC.

**Water Reactor Safety Information Meeting.** The NRC held its sixth Water Reactor Safety Research Information Meeting November 6-9, 1978, at the National Bureau of Standards, Gaithersburg, MD. One hundred sixteen papers were presented, including 12 invited papers dealing with foreign safety research programs. More than 700 persons attended including 175 foreign visitors.

# Advanced Reactor Research

NRC's Advanced Reactor Safety Research program focuses on two reactor concepts: High Temperature Gas-cooled Reactors (HTGR), and Liquid Metal-cooled Fast Breeder Reactors (LMFBR). Fiscal year 1979 activities in each of these areas are outlined below.

## HIGH TEMPERATURE GAS-COOLED REACTORS

The President's fiscal year 1980 budget eliminated funds for gas-cooled reactor research, and NRC programs in that area were targeted for termination in the first several months of that year. Some tasks had been discontinued by mid-1979. Four DOE laboratories were affected by the terminations. To assure an orderly termination and provide all possible allowances for possible future resumption of the research, guidelines were provided for actions in which: (1) items of particular significance to the operation of the Fort Saint Vrain reactor (FSV) in Colorado were identified; (2) distinctions were made between programs which would be costly to discontinue in terms of data-loss, restart expertise, etc., and those which could be resumed fairly easily, and (3) the impact on contractor personnel would be minimized, with assurances that cadres of expertise can be maintained at each laboratory in case of a decision to resume the research. Within these guidelines, a number of programs could be carried out, although at very low levels of activity. Some examples: the metals and graphite programs were kept going at Brookhaven National Laboratory; transient analysis and seismic core modeling of Fort Saint Vrain continued at Los Alamos; low-level efforts continued at Oak Ridge on FSV-related heat transfer, and at Battelle's Pacific Northwest Laboratory (PNL) on graphite inspection techniques. Other than these greatly curtailed activities, however, NRC research in advanced reactor concepts will be confined to the development of computer codes and models for future use in safety investigations.

## LIQUID METAL-COOLED FAST BREEDER REACTORS

The LMFBR program is subdivided into five areas: analysis, safety test facilities, materials interactions, aerosol release and transport, and systems integrity. Progress during 1979 in each of these sub-programs is discussed below.

### Analysis Program

Most of the work in this area was performed at Argonne, Brookhaven, Los Alamos, and Sandia Laboratories, as follows:

**Argonne National Laboratory** completed an analysis of critical experiments dealing wth LMFBR safety using the VIM Monte Carlo code and the Zero Power Reactor-9 (ZPR-9) facility. Results should aid in validating the neutronics computer codes used in LMFBR accident analysis. Other ANL code work resulted in improved calculational efficiencies (by factors up to five) using the COMMIX (Component Mixing) and BODYFIT-1 (Boundary-Fitted Transformation) codes. ANL work in the cooperative studies with EURATOM and the United Kingdom featured calculations with the SAS3D/EPIC code to quantify consequences of various accident phenomena, as well as a study of fuel-pin behavior.

At **Brookhaven National Laboratory**, work on the "Super System Code" (SSC) continued during 1979 (see p. 172, 1977 NRC Annual Report), and a version modeling the Fast Flux Test Facility reactor (FFTF) was completed. Startup tests planned for the FFTF were precalculated and will be compared with operating data when it becomes available next year.

**Los Alamos Scientific Laboratory's** analysis program on hypothetical core disruptive accidents (HCDA) in breeder reactors has been shifted from rapid, energetic accidents to those that develop more slowly. A key concern in such hypothetical accidents is the transition phase in which the core begins to melt and core materials begin to move. LASL completed the first consistent analysis of the transition phase using the SIMMER computer code. SIMMER combines calculations of neutronics, fluid dynamics, thermodynamics and the interactions of these factors with one another during an accident. Also the portion of the code that calculates neutronics phenomena was completely revised and the revised code was tested against experiments performed at Argonne. The fluid dynamics and thermodynamic models of SIMMER had been tested earlier with generally good results. The code was made available in 1979 for use in the United Kingdom, Germany, and the European Economic Community Research Center in Italy.

**Sandia Laboratories** completed a preliminary version of a computer code called CONTAIN, for use in analyzing the responses of advanced reactor containment systems to postulated accident threats. The code will compute the structural and radiological interactions when core material drops from a primary reactor vessel onto the containment floor, and will assess the character of the residual mass.

**The University of Arizona** completed the BRENDA code used for dynamic similation of transients in loop type LMFBRs.

### Safety Test Facilities

Following the upgrading of the Annular Core Research Reactor (ACRR) at Sandia Laboratories, NRC's safety test facility work has consisted of installing a new diagnostics system in ACRR, and of implementing the ACRR-CABRI collaboration in fast-reactor safety experiments. (Both activities were described in the 1978 NRC Annual Report, p. 202.)

### Materials Interactions

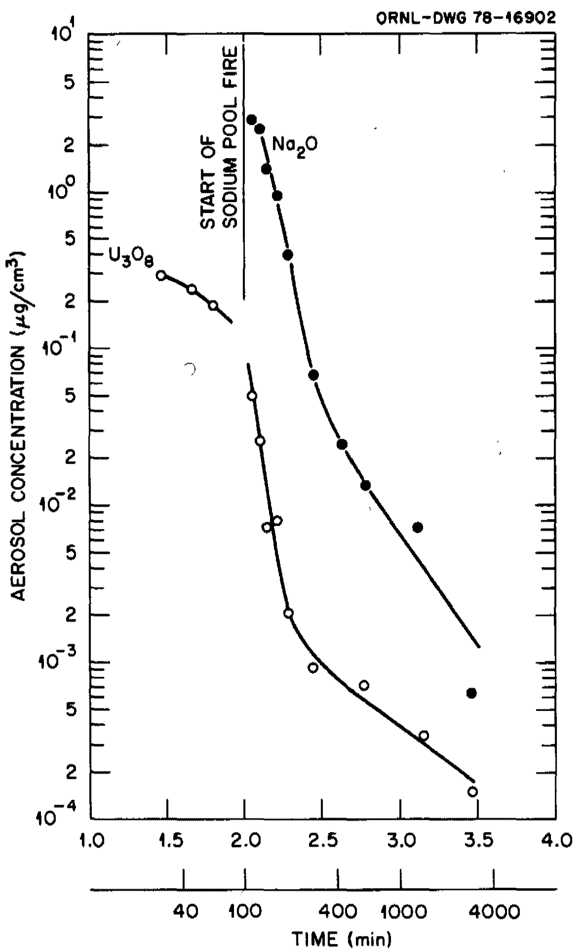
Experiments and analytical model development on the energetics of severe accidents and on the melt-through potential of post-accident core debris continued at Sandia Laboratories in 1979. Analysis of previous ACRR experiments on fuel pellets under accident conditions showed that rapid fuel swelling resulted from fission gas production, and this could not be explained by existing analytical models. As a result, new models of this phenomenon were developed and an improved ACRR series of experiments on fuel disruption (FD-2) was begun. Experiments on the disruption of irradiated fuel and its sweep-out from the core by coolant in LMFBR accidents will use the ACRR's new fuel-motion diagnostics system. A possible spin-off from this work is a suggested series of tests on light water reactor fuel to examine details of fuel failure in conditions such as TMI-type accidents.

In Prompt-Burst Energetics work at Sandia, experiments on the damage potential of severe power excursions (prompt bursts) were resumed in the upgraded ACRR. In these experiments an LMFBR fuel pin contained in a sodium-filled capsule is placed in the ACRR experiment cavity and exposed to an intense, short burst of neutrons that melts and may even partially vaporize the fuel. The resulting pressure and mechanical damage potential are measured and used in constructing analytical models for assessing the threat of such power excursions to the integrity of reactor vessels and piping. The experiments are showing considerably less damage potential than previously considered possible.

### Aerosol Release and Transport

Tests of sodium/uranium oxide aerosols in the Nuclear Safety Pilot Plant (NSPP) at Oak Ridge (see p.

173, 1977 Annual Report) were directed toward areas recommended by NRC's Advisory Committee on Reactor Safeguards. An extensive matrix of such aerosols was examined in a wide variety of conditions in the first phase of the test program. It is scheduled to continue through most of fiscal year 1980. The NSPP test results are used in the assessment of the HAARM-3 aerosol transport computer code. The code assessment is being performed by Battelle Columbus laboratories with the assistance of the University of Missouri—Columbia.




Mixed Oxide Aerosol Concentration – Run 301

**As part of the Aerosol Release and Transport program at Oak Ridge, uranium-oxide ($U_3O_8$) aerosols and sodium-oxide ($Na_2O$) aerosols are mixed to permit study of their interactions and behavior as a function of time. (The $U_3O_8$ aerosols are produced by burning uranium in a plasma torch and the $Na_2O$ aerosols by burning sodium in pools or sprays.) Mixing in the Nuclear Safety Pilot Plant (NSPP) permits study of behavior under conditions in which both aerosols coexist in the secondary containment and are presumed to act together. The experiment diagramed here involved the introduction of $Na_2O$ aerosols into an existing concentration of $U_3O_8$ aerosol, causing a marked increase in the $U_3O_8$ removal rate due to gravitational settling. The experiment tended to confirm that sodium oxide aerosols decrease the concentration of nuclear fuel (uranium) aerosols in a vessel under postulated LMFBR accident conditions.**

Another study at ORNL involves the transport of $UO_2$/sodium aerosols through overlying sodium. This study uses the Fuel Aerosol Simulant Test (FAST) facility, and trial tests using water instead of sodium to establish facility characteristics were under way at the end of 1979. Analytical assistance is provided to this program by the University of Virginia.

### Systems Integrity

Debris-bed behavior modeled in the first three in-reactor tests at Sandia Laboratories, using the Annular Core Research Reactor (see p. 176, 1977 Annual Report and p. 204, 1978 Annual Report) was used in coolability analyses of the Three Mile Island accident. Following that work, a fourth test (of a planned 16-test matrix) dealing with coolability as a function of sub-cooling was performed, with results indicating that self-rearrangement of the debris bed enhances its coolability.

A special series of large-scale sodium concrete interaction tests was completed in support of the NRC's final safety evaluation of the Fast Flux Test Facility (FFTF). The results of the test confirmed the staff position regarding FFTF containment margins. Another test program was initiated to study the interaction of molten fuel materials with candidate materials that could be used in place of concrete to contain core debris from a postulated core meltdown. A large fuel melt test facility is under construction at Sandia. The facility will be used to conduct tests containing up to one half ton of molten fuel in contact with structural materials to confirm analytical methods for predicting containment system response under postulated accident conditions.

# General Reactor Safety Research

NRC's General Reactor Safety Research comprises three areas: site safety research, mechanical engineering research, and structural engineering research. These were described in the 1978 NRC Annual Report (see pp. 206-208).

## SITE SAFETY RESEARCH

Site safety research is generic research directed toward estimating the effects on nuclear facilities of earthquakes, floods, and tornadoes and other severe phenomena, understanding the distribution of those severe natural phenomena, and providing information on meteorology affecting the atmospheric dispersion of radionuclides under postulated accident conditions.

### Geology and Seismology

A magnitude-4 earthquake occurred about six miles from the Maine Yankee nuclear power plant near

Wiscasset, Maine on April 18, 1979. Following that event, Boston College and the Maine State Geologist's Office cooperated in installing a dense array of portable seismographs to record aftershocks which would locate the causative fault. The aftershock pattern suggested a north-south trend clustered around the epicenter of the main shock. The last aftershock was recorded on June 21, 1979.

In the Charleston, S.C., region, high-precision vertical seismic reflection profiling has revealed a probable fault in deep subsurface rock near the center of the large 1886 earthquake. (See p. 206, 1978 Annual Report.) This is the first direct evidence of a possible causative geologic structure for that earthquake. Studies are continuing to determine the extent, history of movements, and tectonic relationships of the fault in order to assess its potential earthquake hazard.

In other 1979 activities under this program:

- A new arrangement for direct cooperation with the Canadian Department of Energy, Mines and Resources will add 12 high quality seismograph stations to "look" southward into areas of interest in the Northeastern U.S. This also will improve U.S. capabilities to evaluate earthquake regions along the St. Lawrence River and in the northwest extension of the problematic "Boston-Ottawa seismic trend.
- Summaries and interpretations of known data bearing on earthquake hazard assessment in areas of the Northeastern U.S., in the New Madrid, Mo., region and in the midcontinent region were published during 1979.
- Studies of the response of soil foundations to earthquake motions are important to earthquake design of power plants and other structures. In 1979 NRC-supported studies of the foundations of important accelerograph stations were completed. Other geotechnical studies resulted in publication of a technical manual describing equipment and operations for determining dynamic soil properties in place.

## Meteorology and Hydrology

Projects in this research field included the following:

**Severe Storms.** Damage surveys of the December 4, 1978 tornado that struck Bossier City, La. and the tornado that devastated Wichita Falls, Tx., on April 10, 1979, were performed. Six 750-pound wide-flange steel beams, 18- and 24-ft. long, were hurled up to 300 yards in the Bossier City tornado, two of them penetrating the ground about eight feet. The most significant aspect of the Wichita Falls tornado was its size—up to one mile wide and more than 40 miles long. Information compiled from these surveys provide authoritative data against which to evaluate the design criteria developed for nuclear power plants and fuel cycle facilities.

**Flooding.** A research program was initiated in 1979 to quantify the safety margins used in flood-related design criteria for nuclear power plants. Flood probabilities, as a function of geographic location, and with particular reference to coastal phenomena, will be determined. A numerical simulation of the November 1975 tsunami in Hawaii was completed during the year, and the storm surge and wave height associated with the passage of Hurricane David along the east coast of Florida in September were measured.

**Atmospheric Diffusion.** The NRC-supported atmospheric dispersion research program featured full-scale field tests and wind tunnel simulations of building-wake-dispersion characteristics; the use of gaseous tracer and lidar technologies to measure vertical diffusion over different terrains, and determinations of thermal performance of cooling and fixed-spray ponds used as heat sinks. Planning also began late in the year for a comprehensive field and modeling program to study atmospheric diffusion in a complex shoreline environment.

# MECHANICAL ENGINEERING RESEARCH

NRC's new Mechanical Engineering Research Program, initiated in 1978, provides the licensing staff with improved methods for evaluating the safety and structural integrity of systems, components, and equipment under normal and accident conditions in terms of margins of safety and probabilities of failure. Major sub-programs are:

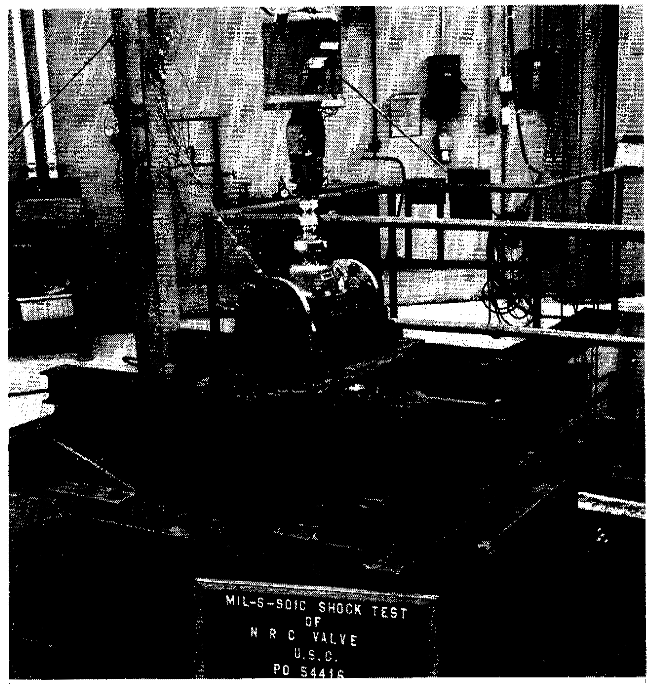
**Seismic Safety Margins Research Program (SSMRP).** This multi-discipline program at Lawrence Livermore Laboratory is designed to estimate the conservatisms in the seismic safety requirements stated in the NRC licensing standard review plan, and to improve those requirements. The approach is to develop a probabilistic methodology that can realistically estimate the behavior of buildings and components of a nuclear plant during an earthquake. The first phase of the program will be completed in 1980.

**Nonlinear System Modeling Program.** A simplified computer code for the analysis of piping systems was completed; a mathematical model of a simplified mechanical system (typical of those in a nuclear plant) was validated; and design charts were issued for use in describing the motion of mechanical equipment, piping and components. There is a need for further research to better characterize the dynamic response of valves in nuclear plants and to better model and scale mechanical systems and equipment.

**PARET Program.** PARET is a systems identification technique to determine frequencies, mode shapes and damping values for complex systems. The design of basic techniques and a computer program were completed during the year. Some test plans and recommendations for experiments to aid in determining certain dynamic parameters of nuclear power plants also were developed. These parameters will be used to confirm safety evaluations for earthquakes, blowdowns, and other accident or environmental events.

**Piping Benchmarks.** Several piping benchmarks were developed at Brookhaven National Laboratory. Benchmarks are used to validate the computer programs used in the dynamic analysis of power plant piping systems. Some of the benchmark problems from of this project were used in evaluating five nuclear power plant shutdowns (see Chapter 3) which occurred in March 1979 when errors were discovered in the analytical computer codes. Licensee codes were checked against benchmark problems, and agreement between them was required before the plants could resume normal operation.

**Load Combinations.** A probabilistic study dealing with load combinations was undertaken in 1979 which will provide guidance on design requirements for simultaneous occurrences of LOCA's and earthquakes, and will develop dynamic response combination methodologies.




**Development of mathematical models to represent the response of reactor components to potential accident conditions begins with actual tests and simulations. Here a nuclear-type valve is subjected to various patterns of seismic shock excitation to permit researchers to derive dynamic response and behavior characteristics for code use. The "shake table" is located at Hughes Aircraft Company, Fullerton, California.**

## STRUCTURAL ENGINEERING RESEARCH

Structural Engineering Research is generic research to assess the safety of nuclear plant structures subjected to extremes of natural events in all possible operating and accident conditions. Primary areas of exploration during 1979 were the definition of design loads on nuclear plant structures and the behavior of such structures. Individual projects in these areas are the following:

**Soil-Structure Interaction (SSI).** A simple computer code for the evaluation of SSI has been completed. Two other codes will permit independent evaluation by the licensing staff of SSI analyses submitted by applicants. These appear to promise a reduction in licensing time heretofore spent verifying licensee calculations.

**Seismic Design Criteria.** This program deals with an unresolved safety issue for which research support is provided to the licensing staff (see Chapter 3). Reports of research projects on the quantification of seismic conservatisms, elasto-plastic seismic analysis, site specific spectra, seismic input and SSI, nonlinear structural dynamic analysis procedure have been completed. Studies in the modeling of earthquakes and analyses of near-field ground motion continued at the end of the year, and recommendations on modifying seismic design criteria were almost finished.

**Seismic Shear Transfer.** This comprehensive program encompasses: scale model testing at Cornell University; full scale tests of wall segments at the Portland Cement Association; and analytical efforts at the Massachusetts Institute of Technology, to provide a better basis for the safety assessment of containment buildings. Several tests and analyses had been completed by year's end.

**Methods of Seismic Qualification.** A report, NUREG/CR-0345, "An Evaluation of Seismic Qualification Tests for Nuclear Power Plant Equipment," compared the results of a series of six tests of the performance of a typical nuclear power plant electrical cabinet in seismic events. Improvement of the criteria defining vibratory input was recommended.

**Water Hammer Effects.** This research evaluates the effects and safety significance of water hammer phenomena in nuclear power plants. A technical report, NUREG-0582, "Water Hammer in Nuclear Power Plants," published in July, provides the results of a staff review of water hammer events and states current staff licensing positions on the topic.

In addition, seven technical reports on water hammer were issued. Together they provide a review and evaluation of actual and potential water hammer events in nuclear power plants, analytical methods and calculation procedures to be used in evaluating water hammer incidents, and state-of-the-art information on water hammer. (See Chapter 3, "Water Hammer.")

**Tornado Generated Missile.** Research in this area was oriented in 1979 to automobiles, considered among the objects most likely to become airborne and threaten nuclear power plants in a tornado. Current practice in impact calculation assumes that the automobile is a rigid mass and this results in the imposition of ultra-conservative design requirements. NRC's reoriented tornado missile project examines the behavior of automobiles at high-impact velocities, an approach which promises more realistic plant design requirements.

# Fuel Cycle, Environmental and Waste Management Research

NRC's Fuel Cycle, Environmental and Waste Management research aims to confirm the basic data and predictive models used in assessing safety in the routine operations of reactors, fuel cycle facilities, transportation of radioactive materials, and disposal of radioactive wastes.

## FUEL CYCLE RESEARCH

Fuel cycle research in 1979 focused largely on commercial operations associated with the milling of uranium ore, the fabrication and storage of power reactor fuels, and the transportation of radioactive materials. Earlier research in the uranium fuel recycle area was canceled as a matter of national policy. The following were 1979 highlights:

- Experiments to determine the rate at which uranium oxide, or "yellowcake" dissolves in simulated human digestive juices and lung fluids indicated that it dissolves more rapidly than previously believed. A follow-on study was undertaken to verify these results in rats, dogs, and monkeys.
- The effectiveness of so called "high efficiency" filters used to reduce airborne releases to the environment was tested under simulated tornado conditions. While the filters did not survive the test conditions of NRC Region I tornadoes (maximum differential pressure = 3 psi), they do survive reduced pressures and air-flows. This information will be useful in designing future plant ventilation systems.
- Criticality experiments provided benchmark data related to fuel-element storage and shipping configurations. Data from these experiments were used to validate NRC methods of analyzing licensee criticality safety programs.
- Transportation research programs were directed toward providing verified codes for analyzing damage to large shipping packages during transportation accidents. At the end of the year puncture resistance assessments had been completed using laminated plates to represent large shielded shipping cask end-plates, and a computer model to assess the shocks experienced by shielded casks during normal rail transport was being checked.
- A new program was developed to establish the response of spent-fuel casks to acts of sabotage involving explosive threats and the potential consequences of such acts on the public.

## ENVIRONMENTAL RESEARCH

Environmental Effects Research is concerned with the radioactive chemicals and heat which are released from nuclear-power plants, with their movement through the environment, and with improving methods of predicting their impacts on people and the environment—impacts which include not only health and safety, but the social and economic effects from nuclear power plants, as well.

### Radiation Dosimetry and Health Effects

A method for assuring that occupational radiation exposures at nuclear power reactors are kept as low as is reasonably achievable (ALARA) was developed and documented in 1979 with the issuance of NUREG/CR-0446, "Determining the Effectiveness of ALARA Design and Operational Features." In addition, a study of animals continuously exposed to low levels of gamma radiation has led to new research to determine if an animal's susceptibility to radiation-induced leukemia is affected by genetic disorders. This program may ultimately result in better assessment of the effects of human exposure to radiation.

### Ecological Impact Studies

NRC is investigating the effects of chlorine compounds released from power plants on important fish species. Chronic toxicity studies to date have indicated that rainbow trout are not adversely affected by the amounts of chlorine normally discharged from nuclear stations, that chlorine byproducts (chloroform, bromoform, etc.) have little impact on shrimp, oysters, and other fish species. A mathematical model developed in 1979 to predict concentrations of chlorine compounds in discharged cooling water permits a more precise estimate of their effects. Validation of the model, using field data from an operating nuclear station, will be undertaken in 1980. Other studies at nuclear power stations have shown that copper is released to the environment during plant operations, and measurements of the effects of this metal on various fish were undertaken at coastal and estuarine sites toward the end of the period. NRC has also investigated the release of asbestos fibers to the environment from power plant cooling towers, and found that the water from cooling towers which have asbestos-fill contains asbestos fibers. This was determined to be of no immediate significance to the public health but worthy of continued monitoring.

## Environmental Transport and Effluent Monitoring

Field studies were completed at the West Valley, N. Y., waste burial site to validate a transport model simulating radionuclide movement in river systems. The studies included measurements of data on channel characteristics, water flow rates, and radionuclide concentrations in Cattaraugus and Buttermilk Creeks extending from West Valley to Lake Erie.

Other studies were done to measure the levels of radioiodine, carbon-14, and tritium in the environment adjacent to the Quad Cities nuclear station. Measurements thus far show no detectable quantities of tritium or carbon-14 within 5 kilometers from the site. Trace amounts of radioiodine, xenon, and krypton were measured at the same locations, and radioiodine transport mechanisms are being evaluated from those measurements. Follow-on studies are in progress on radioiodine behavior.

Independent NRC measurements of radioactive materials in liquid and gaseous effluents at four operating PWRs (Fort Calhoun, Zion, Turkey Point, and Rancho Seco) show generally good agreement both with earlier measurements made by plant operators and with estimates by the NRC staff. This information is used by NRC to assure that releases are maintained at levels as low as is reasonable achievable. The effects of atmospheric pollutants such as nitrogen and sulfur oxides also have been measured, and the data obtained were being evaluated for use in establishing filter replacement schedules for nuclear plants.

To support the development of NRC standards for reactor decommissioning, research has been initiated to determine both the sources of long-lived radioactive products in structural and shielding materials and the nature and distribution of radioactive contaminants within such plants. The costs and methods of removing such contamination also will be studied.

## Socioeconomic Impacts and Regional Siting

A method for obtaining quantitative estimates of the visual impact of different nuclear power plant cooling systems have on communities has been completed, and the results may enhance NRC's ability to weigh the costs and benefits of alternative cooling systems.

NRC also undertook to develop a modeling system to assist State authorities in assessing of environmental impacts and siting alternatives. The project, now in Phase II in New England, consists of a regional electric energy demand model, a power-generation mix model, and a power facility siting model. When completed, the system will give State regulatory bodies access to the quantitative estimates of energy/environment tradeoffs and other information needed for informed siting decisions.

A project begun in 1978 to study the socieconomic impacts of the construction and operation of nuclear power plants was modified and expanded during 1979 to assess the impacts of the Three Mile Island accident. Data input to the study from Three Mile Island will include economic costs (evacuation costs, loss of production, and costs to local governments), the sociological impact of the incident, and the effect of the incident on land use and land values.

# WASTE MANAGEMENT RESEARCH

NRC's waste management research program was set up to establish an independent data base for licensing decisions on high-level waste repositories, shallow land burial sites, and uranium-mill tailing operations.

## High-Level Radioactive Waste Research

High-level waste research in 1979 was divided into the Material Science Programs and the Geotechnical and Sciences Program:

**Material Science Programs.** Quantitative relationships between solid (glass-like) waste forms and a limited number of underground environmental and other parameters were investigated in 1979. The effects of temperature, pressure and chemical influences on interactions between waste forms and surrounding media were also investigated and a preliminary analytical model was developed for predicting the long-term performance of silicate glasses.

**Geotechnical and Sciences Program.** This program deals with the molecular movement of dissolved solids; flow of ground water; fundamental concepts of flow in fractures (as opposed to flow between grains); development of numerical predictive models; and the residence time of ground water. The potential escape of waste materials is being addressed through research into the sealing of construction-induced openings and natural fractures using man-made materials. Indirect determinations of the geologic structure of waste burial site media by geophysical methods are being studied toward minimizing the need for exploratory physical excavation or core boring in such rock.

### Low-Level Radioactive Waste Research

Studies continued in 1979 to assess the migration of wastes at the shallow land burial sites at Maxey Flats, Kentucky; West Valley, New York; and Sheffield, Illinois. At the end of the period, radioactive and chemical substances in trench water had been characterized and potential migration pathways were being analyzed. A laboratory test program on soil retention of radionuclides and the uptake of radionuclides by agricultural crops is being extended to include field studies at the Maxey Flats site. These are being done in coordination with a complementary program of the U.S. Geological Survey. Results will be used to assess the applicability of the laboratory studies to site characterization and predictions of waste mobility. Such information is needed in making decisions about decommissioning existing burial sites and to improve criteria for the design, operation and monitoring of future low level radioactive waste disposal facilities. Tests of the characteristics of solidified low-level wastes were described in the NRC topical report, "Properties of Radioactive Waste and Containers" (NUREG/CR-0619).

### Uranium Milling Research Program

A phase of the field research program at operating mills to support licensing activities and the development of the "Generic Environmental Impact Statement on Uranium Milling," was completed and the report (NUREG-0511) was published in April 1979. The program included field tests to characterize the nature and extent of airborne contaminants at active mills and to assess their potential impacts on air and water resources. In 1980, the program emphasis will shift to the development of information relative to such mitigative measures as groundwater protection and radon attenuation.

## Risk Assessment Research

NRC risk assessment research embraces the development of methods and data for probabilistic nuclear safety analysis, reliability analysis, and the prediction of risks. Activities during 1979 included the development of improved techniques to predict nuclear accident consequences; reactor risk assessment and licensing support; fuel cycle risk assessment, and the development of statistical methods and data-bases necessary for risk assessment.

### Lewis Group Review of WASH-1400

At the time of publication of the 1978 NRC Annual Report, the Commission was studying the report of a special review group headed by Professor Harold W. Lewis of the University of California which was chartered to: (1) clarify the achievements and limitations of the Reactor Safety Study (WASH-1400, issued in 1975, also known as the "Rasmussen Study"); assess peer comments on it, and the response to those comments; (2) study the present state of risk assessment methodology; and (3) recommend how and whether such methodology can be used in the regulatory process. In general, the Lewis report agreed with much of the criticism that had been expressed of the Reactor Safety Study, particularly of the Executive Summary of the study, while endorsing the basic fault tree/event tree methodology that was employed in the study. (See 1978 NRC Annual Report, p. 213, for summary of Lewis report finding.)

In January 1979, the Commission issued a policy statement accepting the findings of the review group. The Commission, among other things, withdrew any explicit or implicit past endorsement of the Executive Summary of WASH-1400, which the Lewis group concluded had lent itself to misuse in the discussion of reactor risks; agreed that the peer review process followed in publishing WASH-1400 was inadequate; and accepted the Lewis group report's conclusion that absolute values of the risks presented in WASH-1400 should not be used uncritically either in the regulatory process or for public policy purposes. In particular, in light of the review group's conclusions on accident probabilities, the Commission said it does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accidents. The Commission noted, however, that it supports the extended use of probabilistic risk assessment in regulatory decision-making, and provided the staff with additional detailed instructions on use of risk assessment techniques and results.

The events at TMI showed serious problems in many areas that were noted as problems in WASH-1400, namely, transient events, small LOCA response, and human error. This emphasized the need to use the valuable tool of probabilistic risk assessment in the regulatory process. Inquiries into the TMI accident have also urged NRC to do this.

Probabilistic risk assessment techniques are now being used much more widely in the NRC. They were used recently, for example, to compare generic safety issues according to their contribution to risk, and this has allowed the Commission to focus its efforts on those 18 issues which pose the highest risk. The remaining 115 items will be addressed later (see "Unresolved Safety Issues," in Chapter 3).

## Reactor Accident Consequence Analysis

A program to develop a site-specific consequence model recommended by the Lewis group was undertaken to update the Calculations of the Reactor Accident Consequences (CRAC) model, developed for the Reactor Safety Study. Improvements, including more realistic treatment of population movements and emergency response and a new meteorological sampling technique, should give a more realistic prediction of the consequences of accidental releases of radioactive materials at specific locations. When completed, the consequence model will be able to differentiate between acceptable and unacceptable sites based on public risk criteria.

## Reactor Risk Assessment And Licensing Support

NRC continued in 1979 to expand the application of risk methodology to a broader spectrum of LWR safety issues, and to apply the methodology and related engineering insights to issues of immediate concern to the licensing staff. This included special efforts to analyze the accident at Three Mile Island.

Activities included: sensitivity studies on potential core meltdowns, (including the Three Mile Island accident) to aid in setting priorities for meltdown-accident research; applying fault-tree and event-tree methodology to other LWR design concepts to broaden engineering insights; evaluating risks in LWR accidents that do not lead to core melting; assessing risks to the public from radioactive contamination of the hydrosphere as a result of core-melt accidents; and assessing the impact of external events (such as transportation accidents) on nuclear plants. In the category of reactor licensing support, a ranking of generic safety issues from a risk perspective was published (see "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," NUREG-0510, and p. 19, 1978 Annual Report). Analyses related to Three Mile Island included a cooperative effort with licensing engineers to evaluate and improve the reliability of auxiliary feedwater systems in operating nuclear plants. Other efforts included an event-tree analysis and recommendations to deal with anticipated-transients-without-scram (see Chapter 3); a value/impact study of the standard review plan for BWR's, and continued evaluation and recommendations for the use in licensing of test intervals for certain plant components such as valves and pumps. Probabilistic assessments of direct current power supplies and of protective measures against loss of alternating current power were also initiated.

## Fuel Cycle Risk Assessment

The objective of fuel cycle risk assessment is to identify the important contributors to risk from nuclear fuel cycle activities, other than reactors, using consequence and probability models. In 1979 a major effort using such models was made at Sandia Laboratories to develop risk methodology to examine deep geologic isolation of high-level radioactive waste in bedded salt. Work in 1980 will focus on licensing questions and on expanding the methodology on other isolation media. Several documents on waste isolation risk methodology were published: "Risk Methodology for Geologic Disposal of Radioactive Wastes: Interim Report" (NUREG/CR-0458), "Risk Methodology for Geologic Disposal of Radioactive Waste: Sensitivity Analysis Techniques" (NUREG/CR-0394), and "Risk Methodology for Geologic Disposal of Radioactive Waste: The Sandia Waste Isolation Flow and Transport (SWIFT) Model" (NUREG/CR-0424). In order for the NRC staff to build up expertise in the use of the risk methodology, the Interoffice Waste Management Modeling Group (IWMG) was formed. By documenting the insights gained from exercising the models on a series of increasingly complex waste repository assessment problems, the IWMG work is designed to improve understanding of the strengths and weaknesses of the risk assessment methodology, and to assist in the licensing decision process.

Planning was completed for a three-year fuel cycle project review to obtain a balanced, independent, multidisciplinary appraisal of results from the waste-isolation risk methodology program. Other 1979 activities included initial steps to develop methodologies for assessing spent-fuel isolation risks and for examining the management of certain radioactive gases emitted from fuel-cycle facilities. The Environmental Protection Agency is investigating the use of probabilistic risk analysis in its regulatory activities, and the NRC staff and consultants reviewed EPA-sponsored work in the assessment of waste isolation. The review group recommendations are being implemented in both NRC and EPA risk methodology programs.

## Methodology Development

NRC programs to develop methodology for probabilistic safety analysis and risk assessment deal with the hazards to nuclear plant operations from fires and floods, and with the impact of testing schedules on the reliability of engineered safety features. New initiatives in fire and flood risk assessment were taken in 1979. The impact of testing schedules on the reliability of safety systems is studied using a computer program named FRANTIC (see p. 180, 1977 NRC Annual Report). In 1979 the program was extended and improved to better portray time-dependent effects on system reliability.

Programs for the collection and analysis of statistical data initiated in 1979 addressed the reliability of nuclear plant operators, maintenance personnel, and a wide variety of safety-related components. Data on equipment failures and human errors were drawn from licensee event reports, power plant log books, and the records in the Nuclear Plant Reliability Data System data banks. The data were analyzed to estimate average error or failure rates; check the consistency of the data sources; identify trends or patterns in the data (time trends, variations from plant to plant, etc.), and study multiple failures of common cause. Work continued on a handbook describing the principal factors governing the reliability of nuclear plant operators to serve as a guide to safety analysts and risk assessment practitioners in assessing human error contributions to risks.

# Research to Improve Reactor Safety

An amendment (P.L. 95-209) to the Energy Reorganization Act of 1974 directs NRC to "develop a long-term plan for projects for the development of new or improved safety systems for nuclear power plants" and requires that the plan be updated annually and submitted to the Congress. The Congressional intent behind this effort is "the improvement of reactor safety and not the enhancement of the economic attractiveness of nuclear power versus alternative energy sources.

In April 1978 NRC submitted to Congress a "Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants" (NUREG-0438), which presented an evaluation of concepts proposed to improve safety and recommended a three-year, $14.9 million research program. The objectives are to determine the feasibility of achieving particular improvements in safety, to evaluate the safety significance of proposed changes and to propose regulatory requirements where implementation is determined to be desirable, without preparing detailed designs. Five research topics and two general studies were suggested:

*Alternate containment concepts*—especially vented containments—to mitigate the consequences of postulated core meltdown accidents. This is accomplished by improving control of the release of radioactivity to the environment.

*Alternate decay heat removal concepts*—especially add on, bunkered systems—to reduce the probability of core meltdowns by increasing the reliability of systems designed to remove heat from the reactor core after fission ceases.

*Alternate emergency core cooling concepts*—to develop simpler and more clearly demonstrable systems to prevent fuel overheating in the event of pipe rupture.

*Improved in-plant accident response*—to reduce the risk from human error by enhancing the quality of the operator-machine interface and by helping operators make correct decisions during accidents.

*Advanced seismic design*—to reduce the vulnerability of plants to earthquakes by decoupling or strengthening components against seismic forces.

*Scoping studies of other concepts*—to determine their potential for improving safety and to assess the need for further research. The studies address protection against sabotage, better ways to monitor the condition of the plant, new siting concepts and ways to reduce occupational exposure without increasing public risk.

*Improved evaluation methodology*—to assist in making more rigorous and thorough assessments of the values and impacts associated with these concepts, and in planning future safety research programs.

Operating experience accumulated since NUREG-0438 was issued, including the events at Three Mile Island in March 1979, reinforces the judgments expressed therein. Many individuals and organizations have submitted additional recommendations for improving safety since TMI, and these also tend to support the original judgments, especially the high priorities assigned to improved in-plant accident response, alternate containment concepts and alternate decay heat removal systems. Within the high priority areas, research toward enhancing the capabilities of reactor operators and improving the quality of the operator-machine interface has been accelerated.

In fiscal year 1979 Congress authorized the expenditure of $1,500,000 to implement the research plan, but appropriated no funds for the purpose. To accommodate this mismatch, the Commission sought and received from Congress reprogramming approval in two separate actions totaling $800,000. The time required to complete these actions delayed the initiation of technical work until the latter half of the fiscal year. Furthermore, the reduced amount of funds made it necessary to restrict studies to a few specific concepts judged to have the highest potential for risk reduction. These factors are reflected in the limited results reported here.

In 1979 research focused on improved in-plant accident response, vented containment, and add-on decay heat removal systems. The status and direction of programs underway at the end of the fiscal year are summarized below.

## Improved In-Plant Accident Response

A study was initiated at Idaho National Engineering Laboratory to identify the information required by an operator to determine unambiguously the status of his plant. Accident sequences having relatively high probabilities of leading to core damage are being analyzed to identify how the plant might respond and what measurements are needed to accurately and uniquely characterize that response. The study indicates what range of physical parameters should be measurable and for which parameters direct measurement might be preferable to indirect measurement. The results are helping establish regulatory positions regarding instrumentation required to monitor the course of an accident. They are also useful in determining what data might be transmitted to remote monitoring and technical support centers.

The simultaneous influx of alarms and data during a reactor accident can overwhelm even a well-trained, experienced operator. A potential solution lies in the application of computer technology to the diagnosis of plant disturbances. Oak Ridge National Laboratory is reviewing the state-of-the-art in computerized disturbance-analysis systems and audio-visual displays with a view toward transferring the applicable technology to nuclear plants.

Oak Ridge is also examining key aspects of the technical basis for disturbance analysis systems, including the reliability of the hardware and software and the quality of the plant systems analysis. These insights will enable the NRC to establish regulatory requirements and to evaluate related efforts by the nuclear industry.

Planning for a conference jointly sponsored by the NRC and the Institute of Electrical and Electronics Engineers was initiated in fiscal year 1979 to identify the extent to which technology from aerospace, defense and other industries may be useful in these NRC programs.

## Alternate Containment Concepts

Risks to the public from nuclear reactor accidents are dominated by sequences in which the core melts and the containment ruptures above ground level because of overpressurization. Concepts have been proposed which would reduce the pressure in the containment during accidents while channeling the containment atmosphere through filter media to retain the radioactive materials it might contain.

In a Sandia Laboratories investigation of the technical feasibility and risk reduction potential of these systems, the state-of-the art in containment design and filter technology has been surveyed and incorporated into a program plan. Conceptual designs fitting vent filter systems to several existing containment types are being developed, and computer codes to analyze their effectiveness are being modified as necessary. The goal is to propose regulatory requirements for vent filter systems and to assess the values and impacts of their implementation on existing and future reactors.

## Alternate Decay Heat Removal Systems

After the nuclear chain reaction in a power reactor ceases, residual heat produced by the decay of radioisotopes must be removed over an extended period of time. This is the function of decay heat removal systems. Improvements in the reliability of these systems offer relatively high potential for risk reduction.

A study was initiated at Sandia Laboratories to identify ways to enhance that reliability and to quantify the risk reduction potential inherent in such improvements. The study includes review of existing criteria for the design of decay heat removal systems and estimates of the resulting reliability. Preliminary performance and safety design requirements are being generated to guide the development of detailed designs by the Department of Energy. The reliability of the improved system will be estimated and the information used to identify the need for and nature of revised regulatory requirements.

## Improved Methodology

Assignments of priorities for safety research and decisions on the adoption and implementation of regulatory positions require judgments regarding the values and impacts of alternatives. A study was initiated at Pacific Northwest Laboratory in conjunction with Battelle Columbus Laboratory to develop more objective and precise methods for making value/impact assessments. Initial emphasis is on better ways to quantify the risk reduction potential of proposed changes in design or operation. Longer term activities include improving the methods used to address the uncertainty in risk estimates and to integrate risk reduction potential with other values and impacts affecting a decision. The results would also pertain to activities beyond the scope of the improved safety research, such as confirmatory research and adoption of regulatory standards.

No work was initiated to study containment concepts other than vented containment or decay heat removal concepts other than a dedicated add-on system. No work was initiated to investigate advanced seismic designs or alternate emergency core cooling concepts. These are being deferred pending the outcome of confirmatory research on the effectiveness of current designs. Needs for research on concepts other than those identified above (e.g., protection against sabotage, improved techniques for nondestructive examination, improved emergency planning, reduced occupation exposure) are being reviewed continually by the NRC staff within the context of confirmatory research and the development of requirements by the regulatory staff. Such needs will be addressed in 1980 and beyond, as resources allow.

In addition to conducting its own research, NRC has provided guidance to the DOE regarding its efforts to improve reactor safety, including recommendations that DOE initiate more detailed examinations and design studies on computerized disturbance analysis techniques, vented containment, hydrogen control technology, add-on decay heat removal systems and seismic decoupling. In addition, NRC asked DOE to take the lead in assessing the cost impacts of design improvements, including implications for retrofit. This cooperation should ease the transition of new concepts into commercial application.

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# Informing and Involving the Public

**NRC's Harold Denton(left) kept press and public informed during the early weeks of the TMI recovery.**

Many actions are taken annually by the NRC to inform the public directly and to make information available regarding nuclear regulation. These take the form of public announcements and *Federal Register* notices; publication of staff and contractor reports; providing access to documents in localities across the country; holding meetings and workshops with public, State and local representatives on issues of widespread interest; responding to public and Congressional inquiries; opening to public observation Commission, staff and advisory committee meetings; and many public hearings on rulemaking and licensing.

Members of the Commission and the NRC staff also participate in press conferences and public meetings and, where special interest warrants, testify before committees of the Congress.

As the most direct means of communicating to the public, the NRC issues announcements on a wide range of topics from headquarters and the five regional offices to some 5,000 members of the news media, industry, the scientific community and the general public.

## Making Documents Available

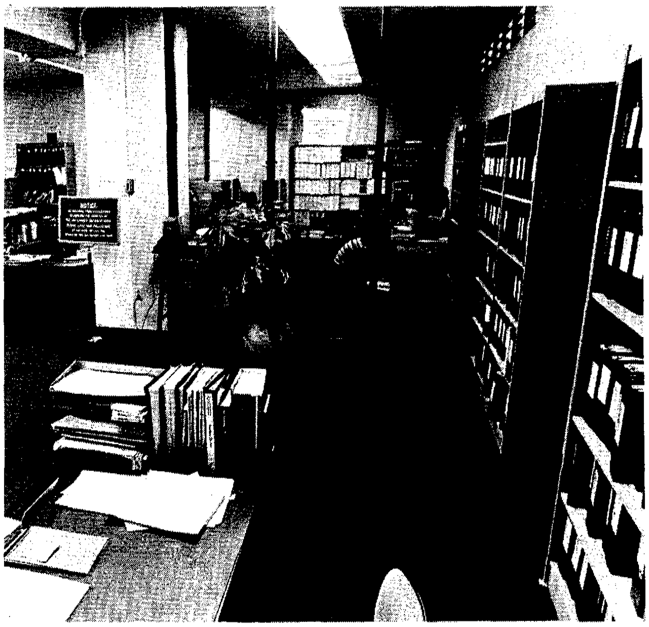
The NRC maintains its principal Public Document Room (PDR) at 1717 H Street, N.W., Washington, D.C., and has established more than 130 public document rooms throughout the country. The local PDRs are typically located in libraries in cities and towns near proposed and actual nuclear plant sites, and contain detailed information specific to the nearby facilities which are either licensed or under regulatory review. (See Appendix 3 for a list of all local PDRs.)

NRC currently makes publicly available through its main PDR in Washington approximately 320 new documents each day. The main PDR contains about 880,000 documents in hard copy or in microfiche. These documents pertain to the licensing of source material, production and utilization facilities, special nuclear material, transportation of radioactive materials, nuclear exports and imports, research and technical assistance reports, reports on generic technical issues, rules and regulations, Commission correspondence, transcripts of Commission meetings, minutes and reports of NRC's advisory committees and other material relating to the responsibilities and operation of the Commission.

The PDR services a diverse clientele. About 42 percent of its users are from industries directly affected by the Commission's activities (utilities, vendors, insurers, manufacturers), 12 percent from companies peripherally affected, 9 percent from law firms, and 37 percent from educational institutions, media, congressional and Federal agencies, public interest groups, and private citizens.

Members of the public may visit the PDR and examine any document in the facility, which furnishes reference assistance, copying services, and micro-fiche reader/printers. In fiscal year 1979, an on-line bibliographic retrieval system was designed and installed in the main PDR to increase accessibility to NRC information. During an average month in the report period, visitors to the Washington PDR requested access to about 7,000 files. The PDR staff also responded to an average of 90 letters a month. More than 1.7 million pages of documents and 13,500 microfiche cards were reproduced for the public during the year.

NRC publishes a Daily Accession List providing a bibliographic description of the documents placed in the PDR. A copy of this list may be obtained by writing the Division of Technical Information and Document Control. The PDR also provides limited free distribution of press releases and Commission/Board orders and issuances.



The NRC's main Public Document Room (PDR) in Washington, shown here, contains about 880,000 documents and makes available to the public more than 300 new documents per day. Members of the public are encouraged to visit the PDR, which offers reference, copying and reader/printer services.

**TMI-2 Investigation Center.** In the aftermath of the accident at Three Mile Island Unit 2 (TMI-2), Senator Gary Hart, Chairman of the Subcommittee on Nuclear Regulation of the Committee on Environment and Public Works, initiated an investigation into the event. To support this special investigation, Senator Hart requested NRC to provide facilities for access to all the documentation related to the accident and other information concerning other operating nuclear power plants of Babcock and Wilcox (B&W) design. On June 2, 1979, the Three Mile Island Documentation Investigation Center was established at the Commission's Washington, D.C., office. The NRC's Division of Technical Information and Document Control collected the documentation and staffed the center, which was also utilized by staff of the President's Commission on the Accident at Three Mile Island.

A pre-incident and post-incident file was established for all documentation concerning Three Mile Island Unit 2, including letters, memoranda, technical and safety analysis reports, pictures, engineering drawings, and depositions taken by investigators. Tapes of conversations between the NRC Operations Center and the TMI site and interviews of key participants were also made available.

The related information on the operating B&W facilities (Arkansas Nuclear One-Units 1 and 2, Crystal River, Davis-Besse, Rancho Seco, Oconee Units 1, 2, and 3, and Three Mile Island Unit 1) included Preliminary Safety Analysis Reports, Final Safety Analysis Reports, Environmental Reports, Safety Evaluation Reports, inspection reports, reportable occurrences, monthly operating and annual environmental reports. A complete set of B&W topical reports was also available in microfiche. Reference information included NRC Regulatory Guides, the nuclear power Standard Plan, and the results of extensive computer-generated bibliographic searches done on the Lockheed, RECON, and other energy data bases.

A sophisticated array of document management technology was made available to provide the investigators with prompt access to the large volume of documentation, which exceeded 2,000,000 pages. The Commission provided the investigators use of the NRC's automated Document Control System for on-line, computer-assisted searches of NRC documentation since 1978, and all TMI-related documents. In addition, a professional technical librarian and a technical information assistant handled investigators' requests for specific information.

**Freedom of Information Act.** The NRC continues to fulfill its obligation to make available records in its possession to interested members of the general public who request them under the Freedom of Information Act (FOIA). By making identifiable records available in all cases where the requested information is not exempt from production, the NRC is contributing to full and fair debate of public issues.

Some categories of records deemed exempt by the statute consist of documents properly classified under Executive Order 12065 (national security matters, trade secrets and commercial or financial information); some types of investigatory files; and certain interagency or intraagency memoranda of a pre-decisional nature.

The NRC continues to place material released in response to FOIA requests in the Headquarters Public Document Room, where the public will have full access. Additionally, documents released under the FOIA which pertain to a particular licensed facility or one under licensing review are furnished to the NRC Local Public Document Room serving that facility.

The number of FOIA requests received during fiscal year 1979 rose by 70 percent over 1978 to 503. This was due both to the public interest generated by the accident at Three Mile Island and growing public awareness and concern regarding various nuclear power issues. Some 16,800 staff man-hours were devoted to supplying requesters with information requested under the FOIA, and more than 80,000 pages of documents were released.

**The Privacy Act of 1974.** This law, which became effective in 1975, provides that individuals have the right to determine the existence of agency records concerning themselves, to seek access to them and to correct any errors that may exist. Agencies are obligated to keep timely, accurate and complete records for agency purposes, and to advise individuals from whom information is solicited how that information is to be used. During fiscal year 1979, the NRC received 40 Privacy Act requests, compared with 37 received in fiscal year 1978. As was the case last year, most of these requests came from agency employees seeking access to personnel security records about themselves.

## INVOLVING THE PUBLIC

The Commission took additional steps during 1979 to facilitate more meaningful and practical involvement of the public in regulatory affairs, both informally and formally.

### Informal Participation

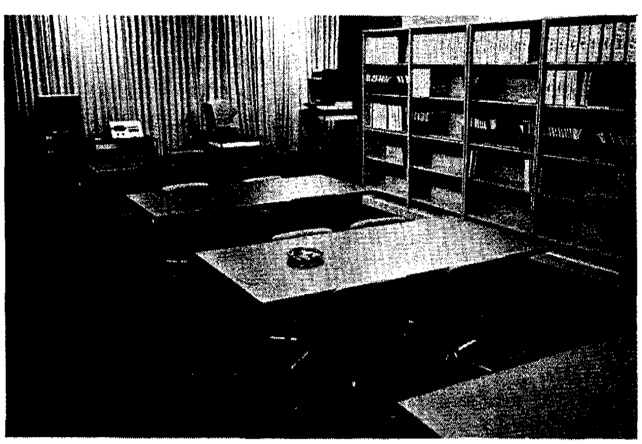
While opportunities for formal public participation in nuclear regulatory proceedings have been provided from the beginning and expanded periodically, the NRC continues to seek practical means of involving the public informally in its deliberations. In 1979, the NRC staff sponsored conferences, workshops and regional meetings on public issues. Some of these activities were:

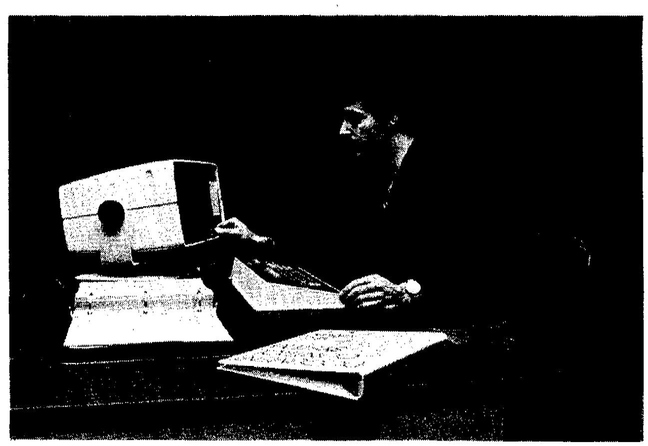
- A meeting on the effects of low-level radiation, held at the NRC's office in Silver Spring, Md. (January 1979).
- A workshop to discuss a proposed rule on alternate sites under the provisions of the National Environmental Policy Act, held at the Mitre Corporation in Virginia (March 1979).
- Workshops to review NRC policy on decommissioning, held in Columbia, S. C., and Seattle, Wash. (September 1979).

In addition, the NRC conducted more than 20 public meetings from late September through December 1979 on emergency response plans for nuclear power plants in operation or expected to be ready for operation in the near future. The NRC sent emergency preparedness teams to meet in the neighborhood of each plant with representatives of the licensee and State and local officials, after notification in the local press. An additional meeting was held to invite public comment and questions on emergency preparedness, lasting until everyone had an opportunity to be heard. Also, in January 1980 a series of four regional meetings to discuss proposed new emergency response regulations were scheduled to be held in New York, San Francisco, Atlanta, and Chicago.

In addition to engaging in cooperative efforts with a broad spectrum of national and regional bodies of State and local representatives (see Chapter 8, "State Programs"), the NRC staff met with State legislators numerous times during 1979 to discuss the agency's programs on nuclear waste management, decommissioning of nuclear facilities, and radiological emergency response planning. In connection with the large volume of legislation on nuclear power before State legislatures during the year, the staff provided comments when requested and presented testimony before legislative committees on several occasions.

The event that highlighted the NRC's public affairs in 1979, however, was the accident at Three Mile Island. The NRC had assumed responsibility for informing the public on the condition of the plant. Three days after the accident, press centers were set up in the NRC's office in Bethesda, Md., and in Middletown, Pa. By April 1, seven NRC public affairs officers were stationed at Middletown, where it had been decided to concentrate all news media activities. The following day, more than 300 national and international news





**The TMI-2 Investigation Center (above left) at the NRC headquarters in Washington, D.C., housed all pre- and post-accident documentation concerning the plant. At right, a technical information assistant uses the automated Document Control System to search out information needed by an investigator.**

Shortly after the TMI accident, press centers were set up in Middletown, Pa., and at the NRC's Bethesda, Md. office. In the photo above, Harold Denton, Director of the Office of Nuclear Reactor Regulation, addresses journalists in Middletown. At his left is Joseph Fouchard, Director of the Office of Public Affairs.

media correspondents were gathered at the press center for daily briefings on the status of the plant.

**Staff Reviews Opened to Public.** The use of informal meetings has significantly expanded the opportunities for public observance of and participation in the early non-adjudicatory stages of the licensing process. An NRC staff policy, which became effective in June 1978, provides that all meetings conducted by the staff as part of the review of any proposed licensing action are open to the general public as observers, including parties to a hearing and petitioners for leave to intervene. In addition, selected meetings are planned specifically to provide information to the public and are held near the proposed facility site. Suitable advance arrangements and notification are made, and public participation (in the form of questions and comments) is sought during these meetings.

The following are cases where such meetings were held in fiscal year 1979 concerning nuclear power plants. With regard to an application for a construction permit for Palo Verde Units 4 and 5, public meetings were held in Phoenix, Ariz., on environmental matters (October 12 and 13, 1978) and on safety matters (October 17 and 19, 1978). On full power operation of Fort St. Vrain, open meetings were held in Denver, Colo., on November 3 and 4, 1978. A program for repair of the recirculation inlet safe ends at the Duane Arnold Energy Center was the subject of a public meeting in Cedar Rapids, Ia., on November 14, 1978. In connection with an application for a construction permit for the New Haven Nuclear Station, a public meeting was held in New Haven, N. Y., on environmental aspects on December 13, 1978. A public meeting was held in Burlington, Kan., on May 15, 1979, regarding the concrete strength of the containment base mat of the Wolf Creek nuclear plant under construction.

Public attendance at individual sessions ranged from four to over 200, with press and TV coverage fairly extensive at several of the meetings. The meetings appear to have a positive effect in permitting the public to judge for itself the effectiveness of nuclear regulation. The NRC intends to hold this type of open forum whenever there is a significant technical issue with considerable public interest or a nuclear power plant is being considered in a vicinity for the first time. Ways to improve anticipation of public interest continue to be explored by the staff.

**"Government in the Sunshine."** During fiscal year 1979, the Commission opened two-thirds of its meetings to public observation in compliance with the Government in the Sunshine Act. The statute, which became effective on March 13, 1977, regulates the conduct of meetings of collegial agencies like the NRC, and makes their deliberative processes more accessible to the public.

The Commission's regulations implementing the Sunshine Act (10 CFR Part 9, Subpart C) specify procedures for deciding whether to close a meeting, what records will be kept, and other administrative details. The law requires the Commission to open all of its meetings to public attendance unless one or more of 10 exemptions applies. The exemptions are designed to permit closed discussion of specified matters. However, transcripts or recordings must be made of most closed meetings and are released to the public when appropriate. In closing one-third of its meetings in fiscal year 1979, the Commission primarily cited four of the 10 exemptions: 1 (classified information), 2 (internal rules and practices), 6 (personnel) and 10 (adjudicatory/litigation). The regulations also specify that advance notices of meetings be published in the *Federal Register*, placed in the Public Document Room, and mailed directly to individuals and organizations on request.

The Commission firmly supports the principles of open government enunciated in the Sunshine Act and has voluntarily chosen to go beyond the literal requirements of the Act to adopt policies that advance its purposes. For example, staff papers and documentation pertaining to the proposed issuance of export and import licenses are made available in the PDR; some Commissioners' correspondence is placed in the PDR;

staff papers discussed in public Commission meetings are placed in the PDR, together with handouts and visual material presented at the meeting; radio coverage, television coverage and tape recordings of Commission and licensing board meetings are permitted; and, in cases of general interest, the public has been permitted to attend Commission adjudicatory sessions that could have been closed under Exemption 10 of the Sunshine Act. In addition, the Commission has a continuing program for reviewing transcripts of closed meetings so they may be released to the public. Since the enactment of the Sunshine Act, the Commission has released 325 transcripts of closed meetings, including adjudicatory minutes.

## Formal Public Participation

NRC regulations provide for formal participation by members of the public as parties in rulemaking, licensing and other proceedings. Opportunities for hearings are indicated in the accompanying table.

Commission regulations require that a public hearing on each application for a major nuclear facility construction permit be conducted by an Atomic Safety and Licensing Board (see Chapter 13). The hearing is announced well in advance in the *Federal Register* and posted in a public document room near the proposed construction site, together with a copy of the application. Local newspapers also carry notice of the hearing. Interested persons or groups are invited to participate in the hearing by: (1) submitting a written statement at the hearing; (2) making an oral presentation at the hearing; or (3) petitioning the licensing board for the right to become an "intervenor" in the proceeding with full participatory rights, including cross-examination of other participants. Intervenors participate fully in prehearing conferences with other interested parties for the exchange of data and identification of issues in contention.

If the licensing board disallows a petition, the petitioner may appeal to the Atomic Safety and Licensing Appeal Board (see Chapter 13). In some instances, the Commission may rule on a petition. Ultimately, a petitioner may seek a ruling in the appropriate Federal Court of Appeals and the Supreme Court of the United States.

The same rights and procedures for public participation apply to hearings on applications for operating licenses, with the difference that such hearings are not mandatory and need not take place unless requested by one or more interested parties.

To facilitate public participation, hearings of the licensing boards, with rare exceptions, are held in communities near each proposed facility site. To enhance public participation in the review and hearing process for facility license applications, and to improve coordination with States, counties, and municipalities, the NRC's basic rules of practice (10 CFR Part 2) contain the following key elements:

- Interested persons may make limited appearances at prehearing conferences.
- Interested cities, counties and local government agencies may participate in licensing proceedings without taking a position on the issues, a privilege also accorded to States.
- Interested States, counties, cities and/or agencies thereof may file proposed findings of fact and conclusions of law, exceptions to initial decisions, and petitions for Commission review.
- Procedures have been established for amicus participation in appeals before licensing boards or the Commission.
- Motions for summary disposition are no longer limited to initial licensing proceedings.
- Licensing boards have authority to consolidate two or more proceedings for hearing.
- Joint hearings with States or other Federal agencies are authorized on matters of concurrent jurisdiction; however, NRC rules of practice may not be waived, and the action must be conducive to the proper dispatch of Commission business and the ends of justice.

NRC's Director of Nuclear Reactor Regulation, Harold Denton, responds to questions from the media in Middletown, Pa., a few days following the onset of the TMI accident.

## ENHANCING INTERNAL COMMUNICATION

Since the NRC's inception, its Chairmen have supported an "open door" policy for the consideration of the views of all employees which extends up through the management chain to the Commissioners' offices. Safety matters may also be discussed with the independent Advisory Committee on Reactor Safeguards. However, in a July 1978 memorandum to all employees, Chairman Hendrie asked for comments and suggestions which might help "to make the 'open door' policy more of a reality both in concept and in practice."

Included in the memorandum were the results of a survey of policies and procedures used or considered by a number of Federal agencies, business corporations, professional societies and other private organizations for bringing differing professional views to the attention of management, and for appropriate management response (NUREG-0500).

The survey described concepts that NRC planned to use in developing formal procedures for making known to management employees' opinions on any substantive matter within the agency's purview that differ from an existing policy or a proposed staff position on the matter. It identified and discussed procedural steps that could provide for: making employee differences known to management, management response, alternatives if the employee is dissatisfied with the response, follow-up on resolution of the issue, and follow-up to ensure that the employee is not subjected to retaliatory actions. In addition, the survey described criteria that could be used to judge the effectiveness—and perhaps the acceptability—of any mechanism designed to handle differing professional opinions.

Comments were solicited from both NRC employees and the public for consideration in developing an agency-wide plan (NUREG-0567). In late 1979, a supplemental statement to this revised plan on differing professional opinions was sent to all NRC employees for comment in early 1980.

## CONGRESSIONAL OVERSIGHT

The number of hearings by the several Congressional committees exercising jurisdiction over NRC activities continued to increase in 1979. During the fiscal year, NRC witnesses testified a total of 42 times before 18 committees or subcommittees on such subjects as the Three Mile Island accident, the health effects of radiation, the shutdown of five facilities due to seismic problems, emergency preparedness, safeguards, and waste management. The NRC testified at an additional 13 hearings in October through December 1979.

The following list shows the date, committee, and subject of each hearing.

10/ 3/78—House Committee on Foreign Affairs, Subcommittee on International Economic Policy and Trade (Nuclear Fuel Transfer for Reprocessing)

1/26/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Spent Fuel)

2/ 5/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (NRC Authorization)

2/22/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (NRC Authorization)

2/26/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and Environment (Reactor Safety Study)

2/27/79—House Committee on Interstate and Foreign Commerce, Subcommittee on Energy and Power (NRC Authorization)

3/ 5/79—House Committee on Appropriations, Subcommittee on Energy and Water Development (NRC Appropriations)

3/13/79—Senate Committee on Appropriations, Subcommittee on Energy and Water Development (NRC Appropriations)

3/14/79—Senate Committee on Governmental Affairs, Subcommittee on Energy, Nuclear Proliferation and Federal Services (Disposal and Storage of Spent Nuclear Fuel and Waste)

3/16/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (NRC Shutdown of Five Nuclear Reactors)

3/19/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (NRC Shutdown of Five Nuclear Reactors)

3/21/79—Committee on Appropriations, Subcommittee on Energy and Water Development (NRC Shutdown of Five Nuclear Reactors)

3/27/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (NRC Authorization)

3/29/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Three Mile Island Accident)

4/ 4/79—Senate Committee on Labor and Human Resources, Subcommittee on Health and Scientific Research (Three Mile Island Accident)

4/10/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Three Mile Island Accident)

## OPPORTUNITIES FOR FORMAL PUBLIC HEARINGS IN NRC PROCEEDINGS

| *Type of Proceeding* | *Opportunity for Hearing* | *Purpose of Hearing* | *Criteria for Granting Hearing* | *Unit Deciding To Hold Hearing* |
|---|---|---|---|---|
| RULEMAKING Proceeding | Prior to issuance of final rule. | To determine whether a proposed rule should be adopted. | At the discretion of the Commission. | Commission (which may decide to hold informal of "hybrid" hearing). |
| MANUFACTURING LICENSE Proceeding* | Mandatory hearing prior to issuance of manufacturing license. | To determine whether a license authorizing the manufacture of a production or utilization facility of a particular design should be issued. | Mandatory hearing on safety and environ- Board. | Mandatory hearing before Licensing Board. |
| CONSTRUCTION PERMIT Proceeding* | Mandatory hearing prior to issuance of construction permit. | To determine whether a particular production or utilization facility should be constructed at a particular site and, where indicated, to resolve adverse antitrust matters. | Mandatory hearing on safety and environmental issues; on antitrust matters, upon request by interested persons or Attorney General or at discretion of Commission. | Mandatory hearing before Licensing Board. |
| OPERATING LICENSE Proceeding* | Prior to issuance of operating license. | To determine whether a particular production or utilization facility should be permitted to operate; antitrust review where significant changes have occurred since previous antitrust review. | Request by any person whose interest may be affected by proceeding who raises genuine issue of material fact, and at discretion of Commission; in addition, in the case of antitrust review, there must be determination by the Commission that significant changes have occurred. | Commission, Appeal Board or Licensing Board, as appropriate. |
| MATERIALS LICENSE Proceeding | Either prior to or after issuance of materials license. | To determine whether a particular materials license should be issued or remain in effect. | Request by any person whose interest may be affected by proceeding and at discretion of Commission. | Commission, Appeal Board, Licensing Board or Administrative Law Judge, as appropriate. |
| SHOW CAUSE Proceeding (to modify, suspend or revoke a license or for other appropriate action). | Prior to issuance of final Commission Order. | To determine appropriate action to be taken. | Upon demand by person cited in Show Cause Order or by request of other persons whose interest may be affected, upon making requisite factual showing. | Commission. |

* An opportunity for hearing is also provided prior to issuance of amendments to manufacturing licenses, construction permits and operating licenses which involve significant hazards considerations. If there are no significant hazards considerations, opportunity for hearing may be provided after such amendments are issued.

4/30/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Three Mile Island Accident)

5/ 8/79—Senate Committee on Governmental Affairs, Subcommittee on Energy, Nuclear Proliferation, and Federal Services (Libassi Report on Low-Level Radiation)

5/ 9/79—Senate Committee on Government Affairs, Subcommittee on Energy, Nuclear Proliferation, and Federal Services (Federal and State Radiation Monitoring at TMI and Issues Relating to the Siting of Nuclear Reactors)

5/10/79—Senate Committee on Energy and Natural Resources (Nuclear Waste Policy Act—S.685)

5/14/79—Committee on Government Operations, Subcommittee on Environment, Energy and Natural Resources (Emergency Preparedness at NRC-Licensed Nuclear Facilities)

5/16/79—House Committee on Armed Services, Subcommittee on Military Installations and Facilities (Civil Defense Preparedness for Three Mile Island)

5/21/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (TMI and Nuclear Reactor Safety)

5/22/79—House Committee on Science and Technology, Subcommittee on Energy Research and Production (Safety Designs of Nuclear Power Plants)

5/23/79—House Committee on Science and Technology, Subcommittee on Energy Research and Production (Safety Designs of Nuclear Power Plants)

5/31/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (NFS West Valley Issues)

6/ 2/79—House Committee on Science and Technology, Subcommittee on Natural Resources and Environment (Health Effects of TMI)

6/ 4/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Nuclear Regulation)

**At the outset of an April 10, 1979 hearing on the TMI accident, Senator Gary Hart, Chairman of the Subcommittee on Nuclear Regulation of the Committee on Environmental and Public Works, swears in the five NRC Commissioners. NRC witnesses appeared at more than 40 hearings during fiscal year 1979.**

6/ 6/79—House Committee on Post Office and Civil Service, Subcommittee on Investigations (Reactor Construction Sites—Manpower Utilization)
6/11/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and Environment (Domestic Nuclear Industry Security)
6/13/79—House Committee on Science and Technology, Subcommittee on Natural Resources and Environment, and Subcommittee on Energy Research and Production (Low-level Ionizing Radiation)
6/14/79—Senate Committee on Environment and Public Works (Confirmation of Victor Gilinsky)
6/22/79—Senate Committee on Foreign Relations, Subcommittee on Arms Control, Oceans and International Operations and Environment (IAEA: Safeguards)
6/27/79—House Committee on Interstate and Foreign Commerce, Subcommittee on Energy Power (Storage and Disposal of Spent Nuclear Fuel)
6/28/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Public and State Involvement in Waste Management)
7/ 9/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Price-Anderson Act and Liability for Nuclear Incidents)
7/19/79—Senate Committee on Commerce, Science and Transportation, Subcommittee on Science, Technology and Space (Safety in the Transportation of Radioactive Shipments)
7/19/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (State Regulations of Nuclear Activities)
7/26/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Nuclear Proliferation)
8/10/79—House Committee on Interior and Insular Affairs, Subcommittee Oversight and Investigations (Waste Isolation Pilot Project)
9/11/79—Senate Committee on Environment and Public Works Subcommittee on Nuclear Regulation (Nuclear Waste Management)
9/19/79—House Committee on Science and Technology, Subcommittee on Energy Research and Production (Improved Safety of Nuclear Power Plants)
10/ 2/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (TMI Investigation)
10/ 3/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (TMI Investigation)
10/ 5/79—Senate Committee on Foreign Relations (U.S.—Australian Agreement for Cooperation)
10/11/79—House Committee on Foreign Affairs, Subcommittee on International Economic Policy and Trade (U.S.-Australian Agreement for Cooperation)
10/22/79—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Uranium Mill Tailings Disposal at Church Rock)
11/ 1/79—House Committee on Government Operations, Subcommittee on Environment, Energy and Natural Resources (Emergency Preparedness at TMI)
11/ 5/79—House Committee on Interstate and Foreign Commerce, Subcommittee on Energy and Power (Kemeny Commission Report—TMI)
11/ 7/79—House Committee on Science and Technology, Subcommittee on Energy Research and Production (Low-Level Radioactive Waste Disposal)
11/ 8/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Kemeny Report)
11/ 9/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Kemeny Report)
11/14/79—House Committee on Science and Technology, Subcommittee on Energy Research and Production (TMI—Kemeny Report)
11/27/79—House Committee on Government Operations, Subcommittee on Environment, Energy and Natural Resources (Marble Hill)
12/11/79—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Waste Management)

## Reports to Congress

The NRC must keep committees having jurisdiction over its functions under rules of the Senate and the House "fully and currently informed" regarding the Commission's activities. Information on significant developments is forwarded routinely to the appropriate committees, and special reports are issued in response to inquiries by committees and individual members of Congress.

Periodic reports to Congress or Congressional committees are required by law on the following matters:

- NRC Annual Report to the President, for his transmittal to the Congress on a fiscal year basis.
- Abnormal occurrences in regulated nuclear activities (quarterly).
- Indemnity activities under the Price-Anderson Act (annual; now being incorporated in the overall Annual Report).
- Administration of the Freedom of Information Act (annual).
- Implementation of the Government in the Sunshine Act (annual).
- Printing plant report (annual).
- Annual plant inventory (annual).
- Major organizational components and numbers of employees (annual).
- Steps to meet provisions of Equal Opportunity Act (quarterly).
- Progress on resolving generic safety issues related to nuclear power plants (annual; being incorporated in the NRC Annual Report).
- Updating of long-term research plan for projects to develop new or improved safety systems for nuclear power plants (annual).
- Commission's views and recommendations on U.S. policies and actions to prevent proliferation (annual).
- ACRS report concerning nuclear reactor safety research program (annual).

In addition, the fiscal year 1979 NRC Authorization Act contained provisions affecting the NRC's activities and authority, and required new reports to Congress, both through new mandates and through amendments to the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974. The principal new reporting requirements are as follows:

- Agency capabilities and research regarding the health effects of low-level radiation (April 1979), and options for Federal research in this area (September 1979).
- Status of domestic safeguards matters during previous fiscal year (annual; incorporated in overall Annual Report).
- Fuel cycle systems evaluation (semi-annual; annually in 1981 and 1982).
- Radioactive waste storage or disposal activities (March 1979).
- Agency use of contractors, consultants, and national laboratories (annual).
- Review of selection and training for members of Atomic Safety and Licensing Boards (1979).

**GAO Reports.** A number of other Congressional reports are issued as the result of studies by the General Accounting Office under its broad authority to assist Congress, its committees, and individual members in carrying out their legislative and oversight responsibilities.

An agency which is the subject of GAO reports recommending corrective actions is required by law to report within 60 days to the Government Operations Committees of the House and Senate on steps taken or planned to implement the recommendations. During fiscal year 1979, the GAO issued seven reports covering various aspects of NRC activities. Four more were issued in October through December 1979. NRC responses to GAO recommendations are available in the main NRC Public Document Room in Washington, D.C. GAO reports issued during the year are:

12/18/78—"Nuclear Diversion in the U.S.? 13 Years of Contradiction and Confusion."

1/23/79—"Automated Systems Security—Federal Agencies Should Strengthen Safeguards Over Personal and Other Sensitive Data."

1/26/79—"Reporting Unscheduled Events at Commercial Nuclear Facilities: Opportunities to Improve Nuclear Regulatory Commission Oversight."

2/16/79—"Higher Penalties Could Defer Violations of Nuclear Regulations."

3/ 8/79—Letter report on NRC's use of DOE Laboratories and of outside contractors and consultants.

3/30/79—"Areas Around Nuclear Facilities Should Be Better Prepared for Radiological Emergencies."

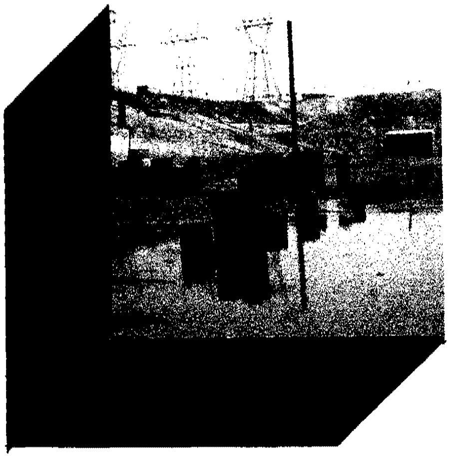
5/ 7/79—"Federal Actions Are Needed to Improve Safety and Security of Nuclear Materials Transportation."

10/ 2/79—"Emergency Preparedness Around the Rancho Seco Nuclear Powerplant: A Case Study."

10/10/79—"Nuclear Construction Times for the Second and Subsequent Plants at a Multi-Plant Site are Overstated."

11/15/79—"Placing Resident Inspectors at Nuclear Power Plants: Is It Working?"

12/ 4/79—"Radiation Control Programs Provide Limited Protection."



# 13 Proceedings and Litigation

**Structural pilings were the issue in the latest hearings on the Bailey Generating Station in Indiana.**

The following are accounts of adjudicatory activity of the NRC during fiscal year 1979, with highlights through December, covering specifically activities of the Atomic Safety and Licensing Boards, the Atomic Safety and Licensing Appeal Boards, and the Commission. In addition, brief reviews are presented of Federal court actions in which the NRC was a party or had an interest.

The most significant developments affecting the adjudicatory phase of the licensing process were delays in several proceedings before licensing boards pending the NRC staff's evaluation and the boards' review of the Three Mile Island nuclear power plant accident, and the Commission's announcement of October 5 that no licensing board decisions authorizing issuance of a construction permit, limited work authorization or operating license would be issued except after further order of the Commission itself. The Commission action was part of an interim policy statement setting out procedures to be followed while the Commission considers a range of options dealing with the extent to which its regulatory structure should be modified as a result of the TMI accident. (See Chapters 1 and 2.)

**Study of the "Immediate Effectiveness Rule."** In its January 1978 Seabrook decision, the Commission spoke of the need to reassess the wisdom of its immediate effectiveness rule (10 CFR 2.764) which permits construction or operation to begin at a nuclear plant immediately after a favorable licensing board decision even though an appeal from that decision may be pending before an appeal board or before the Commission. In January 1979 the Commission established a ten-member advisory committee, chaired by Professor Gary Milhollin of the University of Wisconsin School of Law, to study the immediate effectiveness rule and to report on alternatives. The committee held more than a dozen public meetings and obtained public views through a workshop and a Federal Register solicitation of comments. In December 1979 the advisory committee issued its final report and briefed the Commission on its findings and recommendations. The Commission accepted the recommendations of the committee and directed the Office of the General Counsel to work with Professor Milhollin to prepare a Notice of Proposed Rulemaking to seek public comment on the three alternatives to the rule which were proposed by the committee.

**Study of the NRC Appellate System.** In fall of 1978, the Chairman requested that the Office of the General Counsel prepare a study of the Commission's appellate adjudicatory system in order to analyze the merits of proposals to abolish the Atomic Safety and Licensing Appeal Panel or to combine the functions of that with those of the Atomic Safety and Licensing Board Panel. The General Counsel's study was completed in December 1979, and the Commission was briefed on the results of the study in January 1980. The study examined the History of the Appeal Panel, the role of the Appeal Panel in the Commission's overall adjudicatory system, the Panel's workload, and the practices of other Federal agencies with similar responsibilities. A number of alternatives to the present adjudicatory system were identified and analyzed in the study. The General Counsel's conclusion was that the current adjudicatory system, including the Appeal Panel, should be retained but with several modifications designed to increase the opportunity for early Commission-level involvement in proceedings involving significant new issues of law or policy.

## ATOMIC SAFETY AND LICENSING BOARDS

Public participation in the licensing process is apparent in proceedings conducted by Atomic Safety and Licensing Boards, for it is here that individuals may voice their interests about a particular licensing issue before an independent tribunal that will consider their concerns before rendering a decision.

The Atomic Energy Act of 1954 requires that a public hearing be held on every application for a construction permit for a nuclear power plant or related facility. An independent Atomic Safety and Licensing Board conducts this hearing. This board issues a decision on the application (known as an "Initial Decision"), which, subject to the NRC's review and appellate procedures, may become the final NRC decision. The hearing announcement, which invites public participation, is published shortly after receipt of a construction permit application so that interested parties may be aware of the proceeding at an early stage. The announcement is given to appropriate State and local agencies, as well as other interested groups. Commencement of the hearing itself must await the completion of the NRC staff's safety or environmental review.

The Atomic Energy Act requires that a second opportunity for hearing be provided before a license may be issued to operate a facility. A similar opportunity is provided before certain license amendments may be issued. Public participation is also invited in proceedings instituted by the NRC staff.

The Atomic Energy Act also requires that, prior to the issuance of a construction permit for a nuclear power plant or related facility, a determination be made by NRC as to whether the activities licensed by it would create or maintain a situation inconsistent with the antitrust laws. While the procedures for this review are more complex than those for other reviews, an opportunity to request a hearing before a licensing board is provided to those whose interests may be affected.

Each of the licensing boards consists of three members drawn from the membership of the Atomic Safety and Licensing Board Panel—a body of legal, technical, environmental, and other experts appointed by the Commission. As of September 30, 1979, the Panel included 13 full-time and 40 part-time members. Of these 53 members, 18 are lawyers, 16 environmental scientists, 10 engineers, 7 physicists, 1 economist and 1 chemist. (See Appendix 2 for names of members.) The Commission appoints members to the Panel based upon recognized experience, achievement, and independence in the appointee's field. In assigning individuals to a licensing board, consideration is given to the kinds of issues involved in the proceeding before that board. Generally, boards consist of a lawyer-chairman, a nuclear engineer or reactor physicist, and an environmental scientist. However, antitrust problems are heard and decided by a board of three antitrust experts.

Aside from the hearing on antitrust matters, a hearing on a particular application may be divided into two phases—one concerning the health and safety, and common defense and security aspects of the application, as required by the Atomic Energy Act; and the other concerned with the environmental considerations required by the National Environmental Policy Act. Separate Initial Decisions covering these matters may be issued.

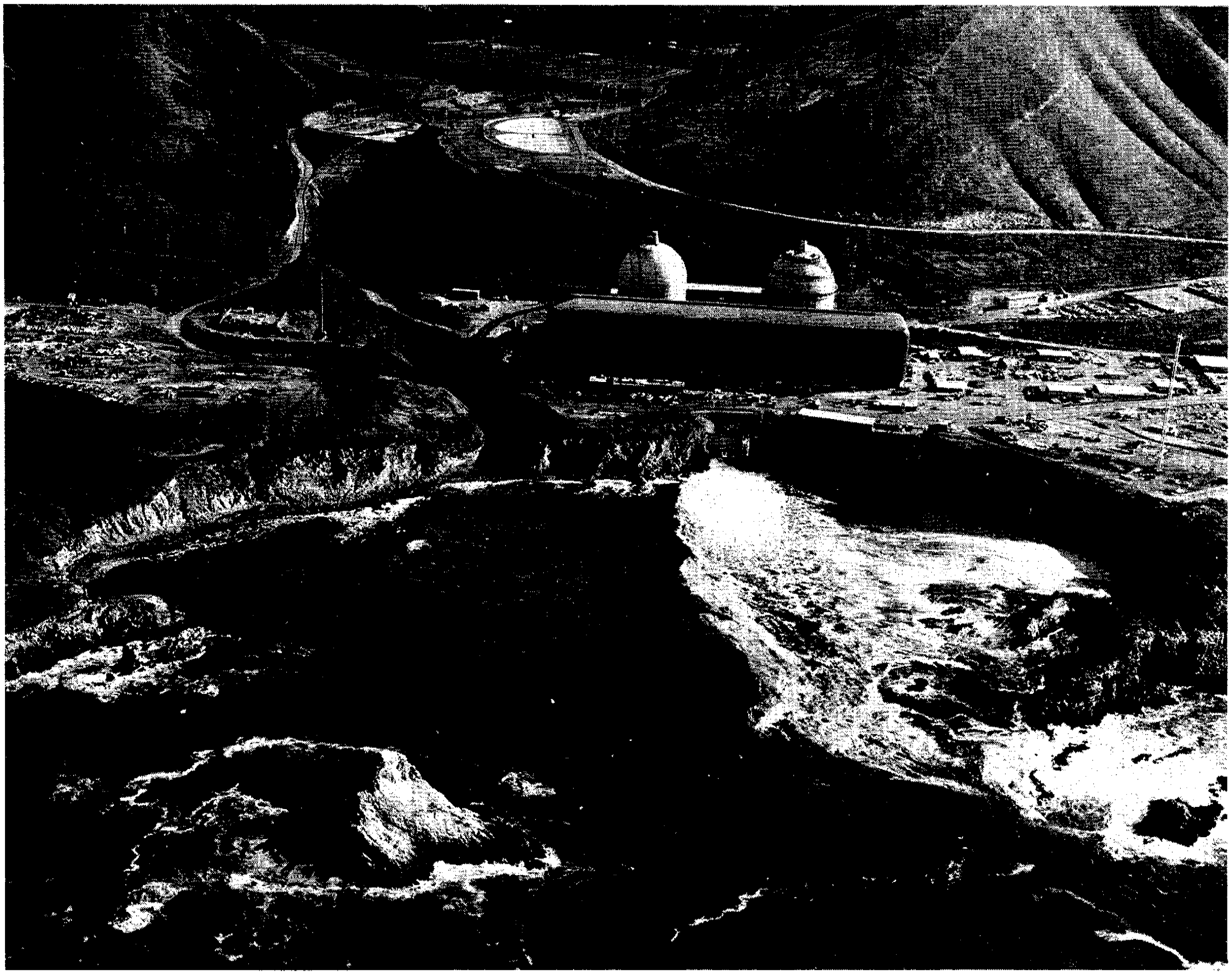
During the period covered by this report, boards were increasingly occupied by operating license and license amendment proceedings and less occupied by construction permit proceedings. Only two construction permit decisions, authorizing four units (*Yellow Creek 1 and 2* and *Jamesport 1 and 2*), were issued. At the same time, petitions to intervene and requests for hearing were received in 12 of 31 operating license and license amendment cases which were noticed. Of these, 10 resulted in hearings being ordered, one was denied, and one was withdrawn. One operating license decision was issued that authorized the operation of the McGuire station's two units. The authorization was, however, stayed pending receipt and review by the board of a supplement to the staff's Safety Evaluation Report.

The *Diablo Canyon* partial initial decision covering seismic, security, and potential aircraft crash issues, fell into this category because it withheld ruling on the operating license pending receipt of the staff's report.

After the Three Mile Island accident, construction permit and operating license proceedings nearing decisions were held up awaiting the staff's evaluation and the board's review of that accident. Boards, however, approved two operating license amendments authorizing expansion of reactor spent fuel pools. Additionally, four proceedings were pending before boards where applicants sought to obtain an early hearing and decision on certain site-related matters. Two of these were commenced during the report period.

Important issues heard and decided by boards during the report period included the following:

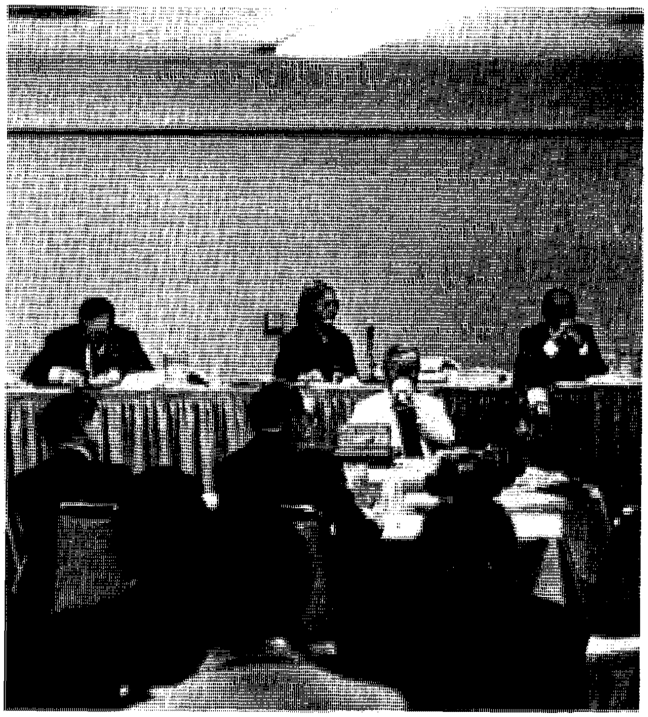
- In *Trojan*, it was discovered during the course of certain modifications to the plant that some structural features did not meet seismic requirements. Pursuant to a request from members of the public, a hearing was held before a board to determine whether interim operation of the plant could be permitted pending approval and implementation of the necessary modifications. After hearing all the evidence, the board permitted operation to resume on an interim basis subject to certain conditions. The board has scheduled a hearing to review the modifications proposed to meet seismic requirements for April 1980.
- In the proceeding on Offshore Power Systems' application to manufacture floating nuclear power plants, the board ruled that an environmental impact statement covering the entire manufacturing program for the plants was not required. Under



The Pacific Gas and Electric Co.'s Diablo Canyon Nuclear Power Plant, is situated on the Pacific Coastline near San Luis Obispo, California. It was the subject of public hearings early in 1979 on plant seismic capabilities in light of its location less than five miles from an offshore geologic fault. At year's end, Unit 1 had been completed and Unit 2 was about 98% complete, with completion of modifications based on the seismic reevaluation expected by mid-1980.

the Supreme Court's ruling in *Kleppe v. Sierra Club,* the board held that the environmental impact statement need cover only the specific proposal to manufacture eight plants, noting the Commission staff's statement that a further impact statement would be prepared on any proposal to manufacture additional plants.

- In *McGuire,* an organization petitioned to intervene as a representative of its members in a license amendment proceeding. In its petition, the organization challenged the requirement that it identify certain of its members who met the legal requirements for participation in the proceeding on the ground that to do so would violate certain of these individuals' rights. The board reaffirmed the requirement that at least one such member be identified.
- In another intervention ruling, involving Unit 2 of the Washington Public Power Supply System, a board was presented with a petition by an organization which did not meet the legal requirements for intervention because it lacked members residing in the vicinity of the plant concerned. The board ruled that this defect could not be cured by the organization's acquisition of such members well after the time for filing of petitions had expired.
- In the *Fermi 2* proceeding, the board was faced with the applicant's request for discovery of the facts relating to the legal basis for a petitioner's right to participate in the proceeding. This request was made prior to the board's ruling on the petition. The board ruled that such discovery was not authorized by the rules.



An NRC Atomic Safety and Licensing Board conducted public hearings in California during Fiscal Year 1979 on the seismic capabilities of the Diablo Canyon Nuclear Power Plant. The hearings followed extensive review by the NRC staff and Advisory Committee on Reactor Safeguards of considerations associated with the Hosgri Fault, located some 3.5 miles offshore from the plant ASLB members Elizabeth Bowers (presiding), William Martin (left), and Glenn Bright are shown.

Other rulings of interest include ones which passed on the qualifications of an expert witness (*Diablo Canyon*); which held that fuel could be delivered to the plant site prior to approval of the operating license (*Zimmer*); and which refused to prevent processing of an early site review application, pursuant to the Commission's rules (*Fulton*). Boards also issued five orders ruling on motions for summary disposition in which they determined whether issues which had been raised in proceedings could be decided without the necessity of an evidentiary hearing. In these orders, the boards were able to decide some issues without a hearing, while setting others down for evidentiary presentations because the nature of the facts surrounding these issues required further elucidation.

At year-end, there were five action antitrust proceedings pending before boards, four in the prehearing discovery phase and one the subject of settlement-negotiations among the parties.

## ATOMIC SAFETY AND LICENSING APPEAL BOARDS

Atomic Safety and Licensing Appeal Boards, consisting of three members each, perform the Commission's review functions in facility licensing proceedings and in such others as the Commission may specify. Board membership for each proceeding is selected from among the members of the Atomic Safety and Licensing Appeal Panel by the Chairman of the Panel. (See Appendix 2 for membership of the Panel.)

Appeal boards entertain appeals from Initial Decisions of licensing boards and certain licensing board orders pertaining to petitions by members of the public seeking to intervene in NRC licensing proceedings. They also review Initial Decisions on their own initiative and sometimes consider questions on rulings referred by a licensing board while the proceeding before it is still in progress. Appeal boards occasionally conduct evidentiary hearings as part of their appellate review functions or as directed by the Commission. The appeal board is the highest level within the Nuclear Regulatory Commission at which a party may seek administrative review as a matter of right. Parties are permitted, however, to seek discretionary Commission review of certain appeal board rulings. The Commission also may itself decide to review an appeal board action. If the Commission does not review a decision, the decision of the appeal board becomes the final order of the Nuclear Regulatory Commission, subject to review in a Federal court of appeals.

During fiscal year 1979, the appeal boards issued 64 decisions and orders, which were included in the *Nuclear Regulatory Commisson Issuances*, the publication containing the adjudicatory issuances of the NRC. Numerous other unpublished orders, generally procedural in nature, were also issued by the appeal boards in the course of conducting the proceedings before them.

The past year presented the appeal boards with many issues concerning environmental acceptability and compatibility with public health and safety. In the environmental area, developments in the much-publicized *Seabrook* (New Hampshire) proceeding were of particular significance. The appeal board held hearings near the Seabrook site on alternate locations for the Seabrook facility. Subsequently, the Court of Appeals for the First Circuit in *New England Coalition on Nuclear Pollution v. NRC* (582 F.2d 87), issued its decision upholding an earlier Commission ruling that in comparing construction costs at the Seabrook site with those at alternate sites, actual costs to complete the facility should be used. The Court's decision ended the need for further appeal board review of the alternate site issue.

Significant environmental issues were also raised in other proceedings. The *Hartsville* (Tennessee) proceedings, involving a facility of the Tennessee Valley Authority, dealt with the effects of a discharge diffuser in the Cumberland River on an endangered species of mussels (*Lampsilis orbiculata*). The diffuser will carry waste and service water from the plant to the river where the mussels live. Following agreement among all parties on a monitoring plan to protect the mussels during construction of the diffuser, the appeal board issued a decision allowing its construction. *Phipps Bend* (Tennessee), involving another TVA facility, concerned the respective responsibilities of TVA and the NRC for satisfying requirements of the National Environmental Policy Act (NEPA). The appeal board affirmed a licensing board ruling that, as the licensing agency, the NRC has the jurisdiction to impose license conditions designed to mitigate adverse environmental effects resulting from the construction and operation of a TVA plant. And *Yellow Creek* (Tennessee), another proceeding involving a TVA facility, raised another question concerning the respective responsibilities of two Federal agencies under the environmental laws—in this instance, the responsibilities of the Environmental Protection Agency and the NRC under NEPA and the Federal Water Pollution Control Act (FWPCA). The appeal board ruled that the NRC may not incorporate in licenses to build nuclear power plants conditions that call for a review of the adequacy of water quality requirements previously established by EPA under the FWPCA.

In *Marble Hill* (Indiana), the issue was which of two States, Kentucky or Indiana, was the proper State for Federal Water Pollution Control Act certification purposes. This issue arose as a result of a conflict between the two States over the boundary line separating them along the Ohio River, close to the plant. The appeal board decided that the boundary line placed the plant's discharge pipe within Indiana's borders, and therefore, the applicant's certification from that State met the Water Act's requirements.

The environmental and health effects of radon (Rn-222) releases produced in the mining and milling of uranium continued to require the attention of the appeal boards in the *Peach Bottom* (Pennsylvania) and a number of other proceedings. The appeal boards jointly handed down orders refining the procedural framework established earlier for resolving the issue without having to hold separate repetitive trials in each pending proceeding. This effort led to a consolidation of several of the cases and disposition, without hearing, of a number of items in dispute. Three members of the Appeal Panel were selected by the Panel membership to take evidence on the remaining, unresolved items. The need for consideration of this matter stemmed from an April 1978 issuance in which the Commission decided that the effects of radon considered by licensing boards in each of 17 earlier proceedings should be reevaluated, because the licensing board evaluations had been made on the basis of a rule that the Commission later found to be in error.

Evidentiary hearings were held by an appeal board in a *North Anna* (Virginia) proceeding on two safety questions, one of which was raised by the board in its review of the licensing board decision authorizing an operating license for the plant. It concerned the plant's ability to withstand damage from missiles generated inside or outside the plant. The second issue, raised by an intervenor, involved the settlement of the land under parts of the site. Decision on these issues remained pending at year-end.

Another proceeding in which evidentiary hearings were conducted by an appeal board involved the question of whether a cloud of flammable vapor from a tanker accident on the Delaware River might reach the *Hope Creek* (Pennsylvania) plant. The appeal board ruled that the probability of such an occurrence was so small that it need not be considered in the design of the plant.

The appeal board in *Callaway* (Missouri) ruled that the NRC could suspend the construction permit for the plant for failure of the construction contractor to cooperate in NRC's investigation into the firing of a construction worker, allegedly for "whistle blowing" on unsafe construction practices. The appeal board held argument on this appeal in the Moot Court Room of the Howard University School of Law in Washington, D.C., with students and faculty members of the school's administrative law and environmental law classes in attendance.

During the year the appeal boards elaborated on the requirements that persons and organizations desiring to intervene in NRC licensing proceedings must meet. In the *North Anna* (Virginia) operating license amendment proceeding in which the licensee was seeking permission to expand the capacity of the facility's spent fuel pool, a petitioner's close proximity to the facility was held to be enough to establish the requisite interest for intervention. *Allens Creek* (Texas) dealt with the appeal of various persons and organizations whose intervention petitions in this reactivated construction permit proceeding had been denied by the licensing board. In ruling on the petitions under appeal, the appeal board detailed the requirements for an organization whose standing to intervene is dependent upon at least one of its members. And in *Skagit* (Washington), with one member dissenting, the appeal board affirmed the licensing board's denial of a late intervention petition of three Indian tribes. Earlier, the licensing board had granted the Indians intervention on the ground that the special relationship between the United States and Indians compelled the grant of the petition even though it was more than three years late. The appeal board held this ruling to be in error and set the decision aside. Upon reconsideration, the licensing board denied the petition and this action was affirmed by the majority of the appeal board. This matter is now being considered by the Commission.

The past year also presented the appeal boards with important questions about the NRC's subpoena authority. In the *Stanislaus* (California) antitrust proceeding, the appeal board was concerned with whether a *subpoena duces tecum* could be issued to one who was not a party to the proceeding. The appeal board held that such a subpoena was authorized by the Commission's regulations and the Atomic Energy Act. It further held that the Commission could require the party requesting this subpoena to reimburse the subpoenaed person for its cost in reproducing the documents demanded in the subpoena, but it denied the request by the subpoenaed company for reimbursement of its search costs on the ground that those costs were reasonably incident to the conduct of its business. *Diablo Canyon* (California) raised the question whether "exceptional circumstances" existed to justify the issuance of subpoenas, pursuant to 10 CFR 2.720, compelling the testimony of two consultants to the Advisory Committee on Reactor Safeguards. In rendering advice to the ACRS, these consultants had expressed views contrary to the opinion adopted by the ACRS. In the circumstances involved (significant intervening seismic-related developments since construction of the plant was first approved), the appeal board concluded that the subpoenas were appropriate.

In a civil penalty proceeding involving the *Atlantic Research Corporation*, the appeal board overturned the decision of the NRC administrative law judge affirming the imposition of a total of $8,600 in civil penalties against the company. The penalty had been imposed by NRC's Director of the Office of Inspection and Enforcement largely because of the radiation exposure suffered by a company radiographer due to the employee's own unauthorized acts. In setting aside the civil penalties in their entirety, the appeal board found no evidence of any shortcoming by the licensee, and, in the circumstances involved, it perceived no discernible remedial purpose that would be served by the civil penalties.

## COMMISSION DECISIONS

### Final Fuel Cycle Rule Promulgated

On July 27, 1979, the Commission promulgated a final rule which sets out a table of revised environmental impact values for the uranium fuel cycle ("Table S-3"). These values are to be included in environmental reports and impact statements for individual light water nuclear power reactors. (See 10 CFR Part 51 and 44 *Fed. Reg.* 45362 of August 2, 1979.) The revised Table S-3 replaces an interim Table S-3 promulgated for a limited period on March 14, 1977. The final rule became effective on September 4, 1979.

This final rule is the product of an extensive rulemaking which began May 26, 1977, and included public hearings before a three-person hearing board and oral presentations to the Commission itself. The scope of these hearings was limited to the environmental effects of spent fuel reprocessing and radioactive waste management. Environmental impacts of the entire fuel cycle are being studied by the NRC staff in a program to revise and update fuel cycle impact values, including radon releases from uranium mining and milling and technetium releases from reprocessing and waste management. This program has not yet reached the stage where specific further amendments to Table S-3 are ready to be proposed. Because radon and technetium releases are not now included in Table S-3, the amount and significance of these releases may be litigated in individual reactor licensing proceedings.

In promulgating the new S-3 rule, the Commission noted that the matters set out in Table S-3 do not, by themselves, convey the environmental significance of uranium fuel cycle activities. The focus of interest is the health effects resulting from the effluents set out in Table S-3. Accordingly, the Commission directed the staff to prepare for public comment in a further rulemaking an explanatory narrative addressing the significance of fuel cycle releases. Pending adoption of such a narrative, the Commission directed the NRC staff to continue presenting in individual proceedings an evaluation of dose commitments and health effects from fuel cycle releases. In these proceedings, the staff will also address economic, socioeconomic, and possible cumulative impacts of fuel cycle activities and such other impacts as may reasonably appear to have significance for NEPA purposes.

The Commission stressed that the fuel cycle rule has a limited purpose. It applies only to environmental cost-benefit balances considered in reactor licensing proceedings and is in no way intended to be a tool for choosing among alternative uranium fuel cycle technologies.

### Three Mile Island Unit 1

At the time of the March 28, 1979 accident at Unit 2 of the Three Mile Island nuclear station, Unit 1 was shut down for refueling. Unit 1 is essentially identical to Unit 2, and is owned and operated by the same licensee. During the period immediately after the accident, the licensee was instructed by the NRC staff not to resume operation of Unit 1 pending staff approval. On July 2, 1979, the Commission ordered that the facility must remain in a cold shutdown condition until further order of the Commission, and that a hearing must precede restart.

On August 9, 1979 the Commission specified in an Order the issues to be considered in that hearing and the procedures to be applied. The Order provided that satisfactory completion of certain "short-term actions" and reasonable progress toward completion of certain other "long-term actions" are required to provide reasonable assurance of adequate protection of the public health and safety. The short-term actions included a number of mechanical and technical changes that have been required of all similar nuclear facilities. The Order also required other short-term actions resulting from: (1) potential interaction between Unit 1 and the damaged Unit 2; (2) questions about the management capability of the TMI licensee in view of the accident at Unit 2; and (3) recognized deficiencies in the licensee's operating procedures and emergency plans.

The Commission formed an Atomic Safety and Licensing Board to conduct a hearing mandated by the August 9 Order and to determine whether the short-term and long-term actions specified in the Order are necessary and sufficient to ensure safe operation of Unit 1. The Board was instructed to proceed expeditiously in conducting a fair and thorough hearing and in arriving at a decision. Finally, the August 9 Order also provided a special procedure for expedited Commission determination of whether a board decision authorizing resumption of operation should become effective pending completion of Commission review of that decision.

## Offshore Power Systems

On September 14, 1979 the Commission issued a Memorandum and Order in the Offshore Power Systems proceeding which affirmed a decision by an Atomic Safety and Licensing Appeal Board that the probability and consequences of a "Class 9" accident are proper subjects for consideration in the Commission's environmental analysis of Offshore Power System's application for a manufacturing license to build floating nuclear power plants. The Commission's decision was based on the narrow ground that it was not apparent that floating nuclear power plants had been considered in the formulation of the accident classification system under which it was determined that Class 9 accidents need not be considered. Further, the NRC staff had already prepared an environmental impact statement analyzing the Class 9 issue for the OPS proceeding and that statement called for imposition of specific licensing conditions to mitigate the consequences of such an accident. Given those facts, it was felt that there would be a conflict with the philosophy of NEPA for the Commission to blind itself to the existence of the staff's analysis or to refuse to consider it in licensing floating nuclear power plants.

The Commission did not directly address the broader question of consideration of Class 9 accidents for land-based plants, but it instructed the NRC staff to move ahead with completion of a long-dormant rulemaking on that subject. The Commission also directed the staff to prepare interim guidance in this area pending completion of that rulemaking, and to bring to the Commission's attention any individual cases in which the staff believes the environmental consequences of Class 9 accidents should be considered as part of the licensing process.

## Waste Confidence Rulemaking

On October 25, 1979, the Commission published a notice of proposed rulemaking in the *Federal Register* which announced the Commission's intent to conduct a reassessment of its degree of confidence that high level radioactive wastes produced by civilian nuclear facilities will be safely disposed of, to determine when such disposal will be available and whether such wastes can be safely stored until they are safely disposed of. The rulemaking was initiated in response to the decision in *State of Minnesota v. NRC*, 602 F.2d 412 (D.C. Cir. 1979), but it is also a continuation of previous proceedings conducted by the Commission in the waste area.

The notice described the hybrid rulemaking procedures the Commission has determined to use, including the possibility that a legislative type hearing may ultimately be held before the Commission itself. The preliminary stages of the proceeding will be conducted by a presiding officer. To aid the participants in the proceeding, the NRC staff will compile an extensive data bank on the subject of high-level waste, supplemented by an extensive bibliography. The tentative schedule announced by the Commission would lead to promulgation of a final decision on the waste confidence question in early 1981.

## Bailly Short Pilings Petitions Denied

On December 21, 1979, the Commission denied two petitions relating to the installation of pile foundations for the Bailly generating station (Nuclear 1), located on the shores of Lake Michigan, near Gary, Indiana. Commissioner Bradford dissented, and Commissioner Gilinsky filed separate views.

The Bailly licensee, Northern Indiana Public Service Company (NIPSCO), received its construction permit in 1974. In 1978, it submitted to the NRC staff a plan detailing its intent to install foundation piles to be embedded in the glacial lacustrine deposits underlying the site. The petitioners asserted that this plan represented a change from the plan approved at the construction permit proceeding, and that a construction permit amendment (with attendant rights to a hearing) was therefore required.

Before reaching a decision on the petitions, the Commission solicited the views of the parties and of the Advisory Committee on Reactor Safeguards. In particular, the Commission asked the ACRS to assess the significance, from the standpoint of engineering and of safety, of a decision to use pilings embedded in glacial lacustrine deposits, rather than set on bedrock or in the glacial till just above bedrock. By letter of July 16, 1979, the ACRS replied that the use of shorter pilings was not a significant design change from the standpoint of engineering, would not require significant alteration of other aspects of the design, and would not affect the safety of the facility, provided that stated conditions were observed.

In denying the petitions, the Commission stated that it found the short pilings plan to be less a change from an earlier design than a proposed resolution of a matter that at the construction permit stage had properly been left for later resolution. The Commission pointed to references in the record indicating that resolution of the pilings issue was contingent on pile tests to be conducted only after the issuance of the construction permit. The Commission stated that in reaching its decision as to whether to order a hearing, it had addressed the safety issue, on which question the views of the ACRS were particularly useful. The Commission concluded that as a matter of law, no construction permit amendment was needed, and that a discretionary hearing would serve no useful purpose. The Commission explicitly did not address the question of what types of design changes require construction permit amendments.

Writing in dissent, Commissioner Bradford took the position that the hearing record established a clear commitment by the licensee to drive pilings to bedrock; thus a construction permit amendment was needed as a matter of law. Commissioner Gilinsky, in a statement of separate views, concluded that in view of the ACRS judgment as to the significance of the change to shorter pilings, no construction permit amendment was needed, but he would have favored granting a hearing in any case, owing to the novelty of the issue and the possibility that a hearing would develop additional information.

(The Commission's decisions on export licensing cases are discussed in Chapter 9.)

# JUDICIAL REVIEW

## Significant Cases

*Westinghouse Electric Corporation v. NRC* (3rd Cir., Nos. 78-1188, 78-1189).

*Exxon Nuclear Company, Inc. v. NRC* (9th Cir., No. 78-1403) (3rd Cir., No. 78-1840) (dismissed by Exxon, August 11, 1978).

*Allied-General Nuclear Service v. NRC* (D.C. Cir., Nos. 78-1144, 78-1422).

*Scientists and Engineers for Secure Energy, Mid-Atlantic Legal Foundation, and Capital Legal Foundation v. NRC* (3rd Cir., No. 78-1204).

This series of cases challenged the Commission's December 23 order which terminated GESMO and related proceedings. The cases were consolidated in the Third Circuit for argument and decision. Petitioners argued that completion of an EIS was necessary to terminate the proceedings, that the Commission showed too great a deference to the President's foreign policy judgments, and that the Commission is obliged to pass upon all license applications under Atomic Energy Act standards. On April 19, 1979, the Third Circuit affirmed the Commission's decision terminating its GESMO review. In doing so, the court agreed that the Commission could rightly give great weight to the views of the President on foreign policy issues, was empowered to freeze license proceedings when considering overall policy issues, and had correctly determined that NEPA's EIS requirements did not extend to a non-merits freeze decision. 598 F.2d 759.

*Seacoast Anti-Pollution League, et al. v. NRC* (1st Cir., No. 78-1172).

This NEPA case was brought by two environmental groups challenging the consideration of alternative sites in connection with the Seabrook application. On May 30, 1979, the First Circuit affirmed the NRC decision that the alternative site investigation for the Seabrook facility complied with NEPA. 598 F.2d 1221.

The only Seabrook case that was still pending at the end of the report period was the challenge to use of the S-3 rule. (*New England Coalition v. NRC*, 1st Cir., No. 76-1525.) The First Circuit is apparently awaiting the outcome of the S-3 litigation in the D.C. Circuit.

*People of the State of Illinois v. NRC, et al.* (7th Cir., No. 78-1171).

Illinois petitioned the Court of Appeals to review the denial of its request for enforcement action (under Commission regulation 10 CFR 2.206) on the General Electric facility at Morris, Ill.

Petitioner alleged that the Morris facility had been "converted" to long-term storage for radioactive waste without preparation of an impact statement and without an evidentiary hearing. On January 10, 1979, the Seventh Circuit upheld NRC's denial of a petition for relief under 10 CFR 2.206, holding that NRC was not required to hold a formal hearing before it could rule on an enforcement petition. 591 F.2d 12.

*Mississippi Power and Light Company, et al. v. NRC, et al.* (5th Cir., No. 78-1565).

*Nuclear Engineering Company v. NRC, et al.* (5th Cir., No. 78-1871).

*Chem-Nuclear Systems v. NRC, et al.* (5th Cir., No. 78-2200).

A number of utilities sued the NRC on its February 9, 1978 license fee rule. The utilities alleged that NRC exceeded its statutory authority in setting the fees. They sought a declaration that the fee schedules are invalid, a suspension of collections in the interim, and a refund of all fees collected under the rule and its 1973 predecessor. The Fifth Circuit affirmed the NRC schedule generally and as against each specific challenge on August 24, 1979. 601 F.2d 223. The utilities have petitioned the Supreme Court to hear this case. (See also Chapter 14.)

*State of Minnesota, By the Minnesota Pollution Control Agency v. NRC and the United States* (D.C. Cir., No. 78-1269).

*New England Coalition on Nuclear Pollution v. NRC* (2d Cir., No. 78-4103).

Minnesota sought review of the Appeal Board's decision in ALAB-455 *(Norther States Power Company)*, which authorized expanded spent fuel storage at the applicant's Prairie Island facility. The New England Coalition also sued to review ALAB-455, claiming spent fuel storage at the applicant's Prairie Island facility. The New England Coalition also sued to review ALAB-455, claiming in connection with a spent fuel pool expansion proceeding. On May 23, 1979, the D.C. Circuit remanded to the Commission the Appeal Board's decision in ALAB-455 for the Commission to decide whether, in light of recent events such as the report of the Interagency Review Group on Nuclear Waste Management, there is reasonable assurance that off-site storage would be available for spent fuel at the end of the license period. 602 F.2d 412. The proceedings ordered by the D.C. Circuit have been initiated by the Commission.

*Ft. Pierce Utilities Authority of the City of Ft. Pierce, et al. v. United States, et al.* (D.C. Cir., Nos. 77-2101, 77-1925).

In *Ft. Pierce Utilities*, petitioners asked the Court of Appeals to review two related Commission actions denying an antitrust hearing. Petitioners argued that a Commission antitrust review may be initiated at any time, independent of licensing reviews. On March 23, 1979, the Court of Appeals affirmed the NRC determination that the Section 105 antitrust amendments of 1970 to the Atomic Energy Act exempted existing facilities then under construction from antitrust review at the operating license stage. 606 F.2d 986. The court did not reach the broader issue of whether those amendments were the sum total of the Commission's antitrust authority. The Supreme Court declined to review the decision on October 1, 1979. 100 S. Ct. 83.

*Westinghouse Electric Corp. v. Hendrie* (D.D.C., No. 79-2060, on appeal, D.C. Cir., No. 79-2069).

*Westinghouse Electric Corp. v. Vance* (D.D.C., No. 2110, on appeal, D.C. Cir., No. 79-2070).

The Westinghouse Corporation sued the NRC and the Department of State, alleging unreasonable delay in the processing of its licenses to export a reactor and components to the Philippines. On August 31, Judge June Green denied the Westinghouse motion for injunction, while bypassing NRC's jurisdiction arguments, and found that the NRC delay was not unreasonable given the important health and safety considerations implied by the application. After the denial of its motion, Westinghouse appealed to the D.C. Circuit and is in the process of having the record certified. (See Chapter 9.)

*Natural Resources Defense Council, Inc., et al. v. Nuclear Regulatory Commission, et al.* (U.S.C.A., D.C. Cir., No. 74-1385, and related cases Nos. 73-2266, 73-1776, 73-1867, 74-1586, and 77-1905).

These cases are challenges to the Commission's fuel cycle rule (Table S-3) on remand from the Supreme Court. They have been held in abeyance pending completion of the Commission's rulemaking.

*State of New York v. NRC* (D.C. Cir., No. 79-2110).

*Natural Resources Defense Council, et al. v. NRC* (D.C. Cir., No. 79-2131).

These two cases are challenges to the new Commission fuel cycle rule (Table S-3) which became effective on September 4, 1979.

*City of Lancaster v. NRC* (D.D.C., No. 79-1368).

*Susquehanna Valley Alliance v. NRC* (M.D. Pa., No. 79-658).

In separate actions filed in May 1979, plaintiffs sued to bar use of the EPICOR-II demineralizer facility and discharge of waste water from the Three Mile Island nuclear power facility in Pennsylvania pending completion of an environmental impact statement and license amendment proceedings.

On October 12, 1979, the District Court in Harrisburg dismissed the Susquehanna Valley Alliance complaint for lack of subject matter jurisdiction, noting that plaintiffs had made no effort to employ the procedures set out in 10 CFR 2.206, and concluded that they had therefore failed to exhaust their administrative remedies. The Court also found that neither of the two exceptions to the exhaustion doctrine were met in this case since plaintiffs had not shown that recourse to the Section 2.206 remedy would be futile, nor had they shown that NRC had violated a clear, nondiscretionary legal duty. Finally, the District Court noted that, should plaintiffs seek agency relief pursuant to Section 2.206, any final NRC decision on their request would be reviewable exclusively in the courts of appeals.

An appeal to the Third Circuit Court of Appeals followed. On October 18, the Third Circuit denied SVA's motion for injunction pending appeal. The case was argued on the merits in November.

In *City of Lancaster*, the NRC's motion to dismiss has been set for argument in January 1980.

*Three Mile Island Litigation* (M.D. Pa., No. 79-432).

These are consolidated cases seeking damages for injuries claimed to arise from the TMI Unit 2 accident . The Commission, which is not named as a defendant, is participating as a friend of the court in the litigation. The cases are at a preliminary discovery stage and have not yet been certified as a class action.

*Friends of the Earth v. NRC* (9th Cir., No. 79-7311).

Petitioner seeks review of the NRC decision to restart the Rancho Seco facility after it was shut down for modifications following the accident at Three Mile Island. Briefing is now underway.

*Akron, Canton & Youngstown Railroad Company, et al. v. Interstate Commerce Commission, et al.* (6th Cir., No. 78-3425).

Twenty-two railroads petitioned the Sixth Circuit to set aside an order of the Interstate Commerce Commission (ICC) in five consolidated cases. The railroads seek a declaration that they are not common carriers of highly radioactive nuclear materials. The NRC filed a limited appearance before the ICC to argue that ICC lacked jurisdiction to examine the health and safety aspects of nuclear materials transportation. The NRC moved to intervene in the case. The case was argued in February 1979 and is awaiting the Court's decision.

*Natural Resources Defense Council, Inc., et al. v. Robert C. Seamans, Jr., et al.* (D.D.C., No. 76-1691, on appeal, D.C. Cir., Nos. 78-1576, 78-1698).

*Natural Resources Defense Council, Inc. v. NRC* (D.C. Cir., No. 77-1489).

*In re Robert W. Fri, Acting Administrator of ERDA* (D.C. Cir., No. 77-121D).

NRDC and other environmental groups sued ERDA and NRC seeking to block construction of the waste tanks intended for the Hanford and Savannah River facilities. The complaint alleged that ERDA (now the Department of Energy) had failed to comply with NEPA by not issuing an environmental impact statement for the waste tank construction, and had failed to obtain licenses from NRC under Section 202(4) of the Energy Reorganization Act. NRC was named a defendant because plaintiffs sought a declaratory judgment that NRC has licensing authority in this matter.

On May 8, 1978, the District Court Judge issued a 34-page opinion upholding NRC's position that it lacks licensing authority over the Hanford and Savannah River storage tanks, but found that DOE erred in not preparing project-specific environmental impact statements for the waste tanks. Cross-appeals were filed. On August 17, 1979, the Court of Appeals reversed the District Court's determination that it had subject matter jurisdiction to review the NRC decision, finding that the NRC decision was a licensing determination and thus exclusively reviewable in the courts of appeals. The D.C. Circuit affirmed the Commission's determination that, pursuant to Section 202(4) of the Energy Reorganization Act, it lacked licensing jurisdiction over 22 ERDA high-level radioactive waste tanks under construction at Hanford and Savannah River. 606 F.2d 1261.

*Porter County Chapter of the Izaak Walton League of America, et al. v. NRC* (D.C. Cir., No. 78-1556).

*People of the State of Illinois v. NRC* (D.C. Cir., No. 78-1599).

*The City of Gary, Indiana v. NRC* (D.C. Cir., No. 78-1560).

*The Lake Michigan Federation v. NRC* (D.C. Cir., No. 78-1561).

These petitions sought review of the Commission's April 20, 1978 decision affirming the denial by the Director, NRC Office of Nuclear Reactor Regulation, of a 2.206 enforcement request relating to the Bailly Generating Station. The cases were consolidated on June 23. The D.C. Circuit, on September 6, 1979, affirmed the Commission's denial of the petitioners' request for remedial relief under 10 CFR 2.206. In its decision, the Court gave strong support to a wide-ranging role for the NRC staff in nuclear regulation, to the Commission's procedures for responding to 2.206 requests, and to the operating license review as the usual forum for addressing issues that arise during plant construction that relate to whether the completed plant can be operated safely. 606 F.2d 1363.

*Hunt, et al. v. NRC, et al.* (N.D. Okla., No. 79-C-122-C, *aff'd.*, 10th Cir., No. 79-1647).

Plaintiffs, observers at the Black Fox proceeding, challenged under the Sunshine Act a licensing board's ability to conduct a closed session to discuss a report containing proprietary data. The District Court denied a temporary restraining order, and NRC filed a motion to dismiss the action on February 26, 1979. The District Court agreed with NRC's interpretation of the Sunshine Act—that it applied to quorums of the Commission and not sessions of the NRC licensing board. 468 F.Supp. 817. On August 24, 1979, plaintiff appealed to the Tenth Circuit. The Court of Appeals for the Tenth Circuit affirmed the lower court's decision, holding that "sessions of the ASLB are not covered by the Sunshine Act. . . ." No. 79-1647, November 23, 1979.

## Concluded Cases

*Basdekas v. NRC, et al.* (D.D.C., No. 78-465.)

On March 17, 1978, an NRC employee sued to compel disclosure of documents under the FOIA and Privacy Act. The documents were an investigative report of the Commission's Office of Inspector and Auditor and two memoranda from the Office of the General Counsel to the Commission, relating to a 1976 OIA investigation. The NRC asserted that portions of the documents are exempt from disclosure under Exemptions 5 and 6. On December 26, 1978, Judge Green ordered the release of an OGC memorandum but upheld the NRC's claims of exemption for the other two, including the statements of witnesses provided under a pledge of confidentiality. NRC settled the remaining issues in the case.

*Martin Hodder, et al. v. NRC, et al.* (D.C. Cir., Nos. 76-1709, 78-1149) (S. Ct., No. 78-1652).

Petitioners brought two petitions for review of the administrative decisions in this case, challenging the NRC construction permit for St. Lucie Unit 2 on the east coast of Florida. The three issues raised by the case were whether the Commission's treatment of Class 9 accidents satisfied NEPA, whether the St. Lucie Unit 2 site complied with Part 100 of the Commission's regulations, and whether the comparison of alternative sites was sufficient to support a decision to build the reactor at St. Lucie. In a memorandum decision on December 22, 1978, the D.C. Circuit upheld the NRC's decision to permit construction of the St. Lucie 2 reactor, holding that NRC was not legally required to evaluate Class 9 accidents for NEPA purposes, that the Part 100 siting regulations were properly applied, and that the investigation of five alternative sites satisfied NEPA. 589 F.2d 1115 (Table). On October 1, the Supreme Court denied the petition for writ of *certiorari.* 100 S. Ct. 55. The Court subsequently denied a petition for rehearing. 48 U.S.L.W. 3357, November 27, 1979.

*Chauncey Kepford v. NRC* (D. C. Cir., No. 78-1933).

On September 21, 1978, petitioner sued the NRC for review of ALAB-480, an appeal board decision which established a procedure to conduct evidentiary hearings on the radon issue in cases pending before the board. Petitioner sought review only insofar as ALAB-480 affects the Three Mile Island proceeding. On May 14, 1979, the court dismissed this petition to review ALAB-480 as not a final order for the purpose of judicial review.

*Pennsylvania Power & Light Co. v. Gulf Oil Co.* (Ct. of Common Pleas, Lehigh County, Pa., Equity Action No. 75-453).

**Two thirds of the Commission meetings in fiscal year 1979 were open to the public, and many drew large audiences, particularly after the Three Mile Island accident. Shown is part of an attentive crowd at NRC headquarters during a briefing of the Commission by one TMI investigating group.**

On October 16, 1978, Gulf Oil Company subpoenaed NRC records in its effort to defend a breach of contract effort over uranium for the Susquehanna station brought by the applicant, Pennsylvania Power & Light Co. NRC treated this request as if filed under the FOIA, and made some documents publicly available. At NRC's request, Gulf voluntarily withdrew its subpoena and its discovery requests since all documents in their possession are now in the Public Document Room and available to them.

*Robert Fendler v. NRC, et al.* (D. Ariz., No. 79-239-PHX).

In a near duplicate of a case brought several years earlier, plaintiff sued the NRC to compel preparation of a new environmental impact statement for the Palo Verde plant. This case was dismissed on August 6, 1979.

*Virginia Sunshine Alliance v. NRC, et al.* (D.D.C., No. 79-1989, on appeal, D.C. Cir., No. 79-2060).

On July 31, 1979, plaintiff sued to block the shipment of spent fuel from foreign research reactors through Portsmouth, Virginia, alleging the possibility of sabotage to the shipment. On August 3, federal District Court Judge Penn denied plaintiff's request to preliminarily enjoin spent fuel shipments through Norfolk, Virginia, finding that the Commission's new safeguards rule provided adequate protection against sabotage threats. The denial of the injunction is now on appeal.

*Township of Lower Alloways Creek v. NRC* (D.N.J., No. 79-1129.)

*County of Ocean v. NRC* (D.N.J., No. 79-1800).

Plaintiffs sought to enjoin expansion of the spent fuel storage pools at the Salem, Oyster Creek, and Forked River facilities. The cases were consolidated for the purposes of argument, which was held in Camden in September 1979. On December 11, 1979, NRC's motion to dismiss these consolidated cases for lack of subject matter jurisdiction, failure to exhaust administrative remedies, and (in the case of Forked River) the absence of a valid case or controversy, was granted.

*United States of America and the Trustees of Columbia University in the City of New York v. City of New York, et al.* (S.D.N.Y., 77 Civ. 3485, on appeal, 2d Cir., No. 79-6023).

The United States, on behalf of NRC and ERDA (now DOE), and Columbia University, filed a joint complaint against the City of New York asserting that the city's refusal, on radiological health and safety grounds, to permit an NRC-licensed reactor to operate violates the supremacy clause of the United States Constitution. The complaint sought a declaration and injunction against enforcement of section 105.107(c) of the City's Health Code which purports to require a City radiological health and safety review and permit for operation of an NRC-licensed reactor. NRC contended that the Atomic Energy Act preempts local authorities from regulating the health and safety aspects of nuclear reactor operation.

On December 27, 1978 the District Court invalidated New York City's requirement of a separate City permit for the Columbia Triga Reactor, finding that New York City's proposed reassessment of radiological safety conflicts with the Atomic Energy Act's exclusive grant of authority in this field to the Federal Government. 463 F. Supp. 604. An appeal was filed with the Second Circuit. After Columbia cancelled its plans for the Triga Reactor, the Second Circuit ordered the District Court to dismiss the case as moot.

*Mid-America Coalition for Energy Alternatives, Inc. v. NRC* (D.C. Cir., No. 78-1294).

Petitioner sought review of the appeal board's March 9, 1978 decision in ALAB-452 which affirmed the licensing board's authorization of a construction permit for Wolf Creek Generating Station, Unit 1. On April 24, petitioner sought a stay from the Court of Appeals, which NRC opposed. On July 6, the Court of Appeals denied the stay but ordered that the case should be set for argument as soon after the filing of the NRC brief as business permits. On January 15, 1979, the D.C. Circuit affirmed NRC's decision to permit construction of the Wolf Creek Station. The Court held that petitioner's proposed alternative was raised too late and that the NRC gave proper independent consideration to the alternatives in its environmental statement. 590 F.2d 356 (Table).

*Jeannine Honicker v. Joseph Hendrie, Chairman, NRC, et al.* (M.D. Tenn., Civ. No. 78-3371-NA-CV) (6th Cir., No. 78-1405, on appeal, 6th Cir., No. 79-1132) (S. Ct., No. 79-710).

Plaintiff sued the NRC for injunctive relief, alleging that the NRC had permitted nuclear power reactors to operate while underestimating the magnitude of health effects of the nuclear fuel cycle. Plaintiff sought revocation of all licenses and dismantling of all existing fuel cycle facilities. The District Court, on September 6, denied a temporary restraining order. Plaintiff appealed the denial of the temporary restraining order to the Sixth Circuit. Not receiving an immediate decision, she brought her request to the Supreme Court. Mr. Justice Stewart denied her request on September 14, 1978. Subsequently, the Sixth Circuit denied emergency relief on November 6. *Honicker v. Hendrie, et al.*, 6th Cir., No. 78-1405.

On NRC's motion, the District Court dismissed the case for lack of jurisdiction on January 16, 1979. 465 F. Supp. 414. The District Court found that plaintiff had failed to exhaust her available administrative remedies and that, in view of the scientific controversy, the case was appropriate for invocation of the doctrine of primary jurisdiction. Then, because review of such an NRC decision was vested exclusively in the courts of appeals, the District Court dismissed the case. The plaintiff appealed to the Sixth Circuit which, on August 7, 1979, affirmed the District Court's order of dismissal, "[f]or the reasons set forth in the district court's thorough and well-reasoned memorandum opinion." 605 F.2d 556. A petition for writ of certiorari is pending.

*Gilbert, et al. v. NRC* (S.D. Texas, No. H-78-2192).

Plaintiffs challenged various aspects of the NRC procedures to be followed in proceeding on the Allens Creek application for a construction permit. The NRC moved to dismiss this case on grounds of lack of subject matter jurisdiction. On January 23, 1979, the District Court entered a brief order dismissing plaintiffs' complaint, finding, in effect, that petitioners had not exhausted their administrative remedies and that no final order had been entered in the case.

*Tibor Fischer v. Nuclear Engineers, et al.* (E.D. N.Y., Civ. No. 78C-259).

On March 7, 1978, plaintiff sought "discharge and release from scientific talks and burning pressures." On NRC's advice, the U.S. Attorney moved for dismissal on the basis that the plaintiff failed to state a cause of action. On May 14, 1979, the District Court granted the motion to dismiss this complaint.

*Radiation Technology v. NRC* (D.N.J., No. 79-753).

Plaintiff sought money damages under the Federal Tort Claims Act for costs flowing from the suspension of this materials license. NRC's response alleged that counts 1 and 2 of the complaint were time-barred under the Tort Claims Act, and disputed the facts of the remaining claim. Judge Stern granted NRC's motion to dismiss counts 1 and 2 on statute of limitations grounds; the remaining claim is in the process of being settled.

*Natural Resources Defense Council, Inc., et al. v. NRC, et al.* (D. New Mexico, No. 77-240-B).

This case, brought by the Natural Resources Defense Council, challenged operations of a uranium milling facility in New Mexico. On May 3, 1977, NRDC, the Central Clearinghouse of New Mexico, and two individuals filed suit against NRC and the New Mexico Environmental Improvement Agency (NMEIA), to enjoin operations of United Nuclear's Church Rock Mill, which NMEIA licensed, alleging violations of NEPA and the Atomic Energy Act. The gist of the complaint was that neither NRC nor New Mexico prepared an environmental impact statement for the Church Rock Mill. Plaintiffs contended that New Mexico, as signatory to a Section 274 State Agreement to regulate redioactive materials, exercised Federal power and therefore must comply with NEPA, and the NRC's continuing review powers over State programs constituted sufficient Federal involvement to call for preparation of an environmental impact statement (EIS). Second, plaintiffs argued that, in order to comply with section 274, State programs must be "compatible" with the NRC program and that compatibility required preparation of an EIS where NRC would prepare one in a non-Agreement State. NRC currently prepares an EIS for each new milling license and first renewal. A similar petition was filed June 30, 1977, in the D.C. Circuit naming only NRC as a respondent (No. 77-1570).

The D.C. Circuit on January 6, 1978, issued an order which rejected NRDC's theory that New Mexico as an Agreement State is exercising delegated Federal power. The Court also found that NRDC's allegations that the NRC has been "intimately involved" with the licensing of Church Rock demonstrate, if true, only State-Federal cooperation rather than a final decision-making authority retained by NRC. The Court took no view on whether the New Mexico regulatory program is compatible with the Federal regulatory framework. That order brought the proceedings before the D.C. Circuit to a close. 8 ELR 20153.

Motions for summary judgment were filed before the District Court for the District of New Mexico both by NRDC and by the NRC; trial was scheduled early in 1979. The Kerr-McGee Nuclear Corporation intervened before the District Court, after the June 15, 1978 decision of the Tenth Circuit, on the appeal of Kerr-McGee and the American Mining Congress, which reversed the District Judge's decision denying them intervention. 10th Cir., 578 F.2d 1341.

The District Judge approved a stipulation on July 13, 1979, dismissing this case in light of the Uranium Mill Tailings Radiation Control Act of 1978, which changed the regulatory control over the tailings.

## Opened Cases

*Thot-Thompson v. McVeagh* (D. Md., No. B-79-1703).

Plaintiff sued defendant for damages alleged to be the result of slander. Both are NRC employees. Defendant is plaintiff's supervisor and is alleged to have slandered plaintiff by commenting upon plaintiff's job performance. The NRC position is that the defendant was acting within the scope of his employment when he made statements about the plaintiff. A petition for removal from State court to Federal District Court was filed on September 13. The U.S. Attorney will represent the defendant.

*Gentry, et al. v. United States, et al.* (N.D. Ala., No. CA79-L-5181-NE).

Plaintiff seeks money damages for injuries allegedly caused by radiation exposure during the years 1954-1964 while plaintiff was working for Thiokol Corporation on various government projects. The

Justice Department will have lead responsibility for the defense of this action, and has filed a motion to dismiss, or in the alternative, for summary judgment.

*Kertis v. United States* (W.D. Pa., No. 77-1259).

Plaintiff seeks recovery as administratrix of her husband's estate. Plaintiff's decedent, who died of leukemia in 1974, was a worker in the Westinghouse Cheswick facility engaged in repair of Navy submarine pumps. A similar lawsuit was dismissed in 1976 as plaintiff was limited to workmen's compensation against Westinghouse under State law. NRC assists the U.S. Attorney in responding to requests for discovery in this case. The NRC involvement is extremely limited, since the specific NRC license includes only a limited amount of radioactive material.

*Lawrence Skinner v. NRC* (N.D. Calif., No. C-79-1231-WAI).

Plaintiff seeks tort relief from the NRC for injuries alleged to have resulted from atomic weapons testing in the 1950s. This case is being handled through the U.S. Attorney's office. NRC's position is that DOE, and not NRC, is the responsible agency here. Even if NRC were the proper defendant, plaintiff has not exhausted his administrative remedies. A motion to dismiss was filed on August 9, 1979.

*Broudy v. United States* (C.D. Calif., No. 79-02626 LEW(GX)).

This is a tort action for damages against the United States and various government agencies for wrongful death of Broudy, a Marine participant in an AEC weapons test in Nevada in 1957. The NRC position is similar to that taken in *Punnett* and *Skinner* (above). The NRC is not responsible in any way for weapons programs, that function being transferred to the Department of Energy. The Justice Department is handling this defense.

*Life of the Land v. Adams* (D. Hawaii, No. 79-0249).

Plaintiffs challenged the transport of two shipments of spent fuel from Japan through Hawaiian waters and the port of Honolulu, seeking preparation of an environmental impact statement and compliance with NEPA and the Ports and Waterways Safety Act. The application for injunction on the first of these shipments was denied on June 7, 1979, and upheld by the Ninth Circuit on June 8, 1979. The governor closed the port to both shipments. One was permitted to refuel at Pearl Harbor on an emergency basis; the other refueled in non-Hawaiian waters. Because no more shipments were scheduled, the Justice Department filed a motion to dismiss on grounds the case was moot.

*Won-Door Corporation v. United States* (Court of Claims, No. 109-79L).

Won-Door sued the United States for compensation for an alleged taking of its property by virtue of radon contamination from the adjoining Vitro uranium mill tailing site. The Department of Justice is handling the defense of this action. The NRC advised the Justice Department of its opinion that no taking of the plaintiff's property occurred through regulatory actions. The answer to the petition was filed June 11, 1979. NRC responded to interrogatories in August 1979.

*State of Illinois v. General Electric* (N.D. Ill., No. 79-C-1427).

The State seeks a declaration that General Electric is the sole and exclusive owner of the Morris site and that the prior arrangements for perpetualcare of the site by the State under Illinois law are invalid. The NRC is a named defendant in this lawsuit although the principal dispute concerns General Electric and the State. The Illinois Attorney General seeks to invalidate the Illinois Radioactive Waste Act as preempted by the Federal Atomic Energy Act. It is NRC's position that the acts are consistent, and the agency filed a motion to dismiss in July 1979. As to claims that the law is invalid under the Illinois constitution, the NRC would defer to the Illinois State courts now considering this issue. The motion to dismiss is pending.

*Upper Skagit Indian Tribe, Suak-Suiattle Indian Tribe and Swinomish Tribal Community v. NRC and United States of America* (D.C. Cir., No. 79-2277).

On October 26, 1979, three American Indian tribes, the Upper Skagit Indian Tribe, the Suak-Suiattle Indian Tribe, and the Swinomish Tribal Community sued the NRC concerning Appeal Board decisions denying their petition to intervene, which was filed 3-1/2 years late into the Skagit construction permit proceeding. NRC has moved to hold the petition in abeyance pending the outcome of the administrative proceedings.

*Susquehanna Valley Alliance v. NRC* (3rd Cir., No. 79-2800).

On December 13, 1979, petitioner filed a request for review of the October 16, 1979, order directing use of the EPICOR-II decontamination system at TMI-2.

## Pending Cases

*Minnesota Environmental Control Citizen's Association, et al. v. Atomic Energy Commission, et al.* (D. Minn., No. 4-72-109).

Plaintiff, a citizen's association, sought to enjoin further development and operation of Northern States Power Company's Monticello and Prairie Island facilities on the ground that the Prairie Island construction permit and the Monticello provisional operating license were issued without preparation of an an environmental impact statement. On July 28, 1972, the District Court issued an opinion refusing to enjoin the construction or provisional operation, but holding that before full operating permits for these facilities could be granted, a full NEPA review was required. The Court retained jurisdiction over the matter to ensure that such a review was performed. During the past six years, the Commission has undertaken this environmental review, and both licensing proceedings are nearing completion. Once these are completed, NRC intends to move to dismiss the complaint on the grounds of mootness, as well as the statutory mandate that only a court of appeals shall review final orders of the Commission.

*West Michigan Environmental Action Council, Inc. v. AEC, et al.* (W.D. Mich., No. G-58-53).

Citizen-group plaintiffs sought an injunction against increased use of mixed oxide fuel in Consumers Power Company's Big Rock Point power reactor. In June 1974, the Court placed the case in abeyance pending the outcome of the GESMO proceedings and NRC review of Executive Branch comments. The utility has not pressed its application nor prepared the required environmental report, so the case may eventually be moot.

*Chauncey Kepford v. NRC, et al.* (D.C. Cir., Nos. 78-1160, 78-2170).

In No. 78-1160, petitioner sued the NRC to stay operation of the Three Mile Island, Unit 2 facility, primarily because of the level of radon-222 releases from tailings produced in uranium mining and milling. On March 8, 1978, the Court denied petitioner's motion for a stay. On March 22, 1978, the Court, on its own motion, held further review in abeyance pending completion of the administrative appeals and ordered NRC to file periodic status reports with the Court.

In No. 78-2170, petitioner sued the NRC on November 13, 1978, for review of the Commission's affirmation of an appeal board decision that involved all but two of the issues associated with the *Three Mile Island* facility and which permitted its continued operation. On November 30, 1978, the NRC moved to hold the petition for review in abeyance pending the outcome of administrative hearings into one of the issues raised by the petitioner; that is, the probability that a very large aircraft will crash into the reactor. On May 11, 1979, the court granted the motion, and this case is now being held in abeyance.

*Ecology Action of Oswego, New York v. NRC, et al.* (D.C. Cir., No. 78-1855).

Petitioner sued the NRC to set aside the construction permit for the Sterling nuclear facility. Petitioner appeals from the denial of the application for a stay before the appeal board. Briefing was completed on March 14, 1979. Petitioner waived argument and the case awaits decision on the briefs and record.

*Detroit Edison Company, et al. v. NRC* (6th Cir., No. 78-3187).

*National Association of Regulatory Utility Commissioners v. NRC* (6th Cir., No. 78-3196).

These cases involved challenges to the Commission's denial of a rule-making petition filed by the Detroit Edison Company. Detroit Edison had requested that the NRC amend its regulations to provide that NRC lacked authority to require rerouting of transmission lines associated with nuclear plants. This case involves the same legal issue that the First Circuit decided favorably to the Commission on June 21, 1978, in *Public Service Company of New Hampshire v. NRC*, No. 77-1419. 582 F.2d 77, *cert. denied* 99 S. Ct. 721. The cases have been briefed and still await argument.

*State of New York v. NRC* (2d Cir., No. 75-4278).

*Natural Resources Defense Council, Inc., et al. v. NRC, et al.* (2d Cir., No. 75-4276).

*Allied-General Nuclear Service, et al. v. NRDC, et al.* (Sup. Ct., No. 76-653).

*Commonwealth Edison Company, et al. v. NRDC, et al.* (Sup. Ct., No. 76-762).

*Baltimore Gas & Electric Company, et al. v. NRDC, et al.* (Sup. Ct., No. 76-774).

*Westinghouse Electric Corporation v. NRDC, et al.* (Sup. Ct., No. 76-769).

In these cases, New York State and certain citizen groups sought review of the Commission's November 14, 1975 *Federal Register* notice, which set forth procedures for hearings on the Generic Environmental Statement on Mixed-Oxide Fuel (GESMO) and outlined agency standards for licensing activities related to the use of mixed oxide fuel prior to a Commission decision on wide-scale use of plutonium recycle. On May 26, 1976, the Court of Appeals for the Second Circuit issued its decision upholding, in full, both the GESMO hearing procedures and associated individual licensing procedures. However, interim licensing (except for "experimental and feasibility purposes") was forbidden. The Supreme Court granted petitions for *certiorari* by a number of utilities and a manufacturer. On December 23, 1977, however, the Commission voted to terminate the GESMO proceedings.

In January 1978, the Solicitor General filed a suggestion of mootness on behalf of NRC with the Supreme Court, submitting that the Commission's decision in December of 1977 on mixed oxide fuel rendered the Second Circuit's decision moot, and that the opinion should be vacated and the case remanded with instructions to dismiss. On January 16, 1978, the Supreme Court vacated the Second Circuit's judgment and remanded the case to the Second Circuit "to consider the question of mootness." Those cases are still pending in the Second Circuit.

*John Abbotts, et al. v. NRC* (D.D.C., No. 77-624).

John Abbotts, the Public Interest Research Group, and the Natural Resources Defense Council, Inc., brought a Freedom of Information Act suit challenging an NRC decision to withhold certain safeguards documents. The documents involved fall into three categories: (1) records related to the NRC program for onsite review of SSNM facilities initiated in early 1976; (2) records concerning the NRC investigation and review of conditions at the Nuclear Fuel Services facility in Erwin, Tennessee, in late 1975 and early 1976; and (3) studies done for or related to NRC's Special Safeguards Study and the Draft Safeguards Supplement. Parties cross-moved for summary judgment in early 1978. After further review of the documents by NRC, the parties narrowed the dispute to two small portions of two documents, concerning "baseline threat level" information, specifically contesting the proper classification of that information. Supplemental cross-motions for summary judgment have been filed in 1979. The Court must now decide whether to review the documents *in camera* and whether there is a valid exemption claim by NRC.

*Natural Resources Defense Council, Inc. v. NRC, et al.* (D.C. Cir., No. 77-1448).

On May 13, 1977, NRDC filed a petition for review of the NRC's March 14 *Federal Register* notice promulgating an interim rule quantifying the environmental effects of the uranium fuel cycle. On July 5, 1977, NRDC requested that the D.C. Circuit hold the case in abeyance until the Supreme Court reaches a decision in the *Vermont Yankee* fuel cycle case. NRC consented to that motion. On June 7, 1978, the D.C. Circuit requested the parties' views on how to dispose of this other fuel cycle case. On June 27, 1978, NRC advised the Court that the interim rule case should be dismissed or held in abeyance pending a challenge to the final fuel cycle rule. The Court is holding the case in abeyance and considering a new scheduling order in light of the issuance of final rule.

*United States v. New York City* (S.D.N.Y., No. 76 Civ. 273).

On January 15, 1976, the plaintiffs—the NRC, ERDA, and Department of Transportation (DOT)—sought a judgment declaring a New York City Health Code provision dealing with the transportation of nuclear materials through the city to be inconsistent with the Federal statutory scheme governing the transportation of hazardous materials. The government's request for a preliminary injunction against enforcement of the Health Code provision was denied on January 30, 1976, the Court finding that no irreparable injury would occur pending a decision on the merits of the case. DOT has published regulations under the Hazardous Materials Transportation Act (which became effective January 1977) which allow interested persons to seek a ruling that a local ordinance is inconsistent with DOT regulations. On February 28, 1977, Brookhaven filed its request for such a regulation with DOT, arguing that the city's restrictions on shipping new and spent fuel were inconsistent with DOT's regulations. NRC and ERDA (now DOE) wrote DOT in support of Brookhaven's position. On April 4, 1978, DOT ruled that the city ordinance was not inconsistent with DOT policy, but that a rule-making would be held to consider whether regulations regarding the routing of nuclear materials by road are warranted. DOE has asked the Department of Justice to again bring a lawsuit. On February 3, 1979, NRC advised the Department of Justice that it did not favor such a suit.

*Hope Punnett v. Jimmy Carter, et al.* (E.D. Pa., No. 79-29).

Plaintiff seeks notification to soldiers involved in nuclear weapons testing programs and to their families of potential risks arising out of that program. The Department of Justice is the lead agency in this action, and has pending a motion to dismiss on the basis of sovereign immunity and no duty to notify. Discovery is now taking place. Plaintiff amended its complaint to include a request for damages; but the class action nature of the complaint has not yet been certified. Plaintiff's motion for a preliminary injunction was denied on March 30, 1979.

*Peshlakai v. Schlesinger, et al.* (D.D.C., No. 78-2416).

Plaintiff alleges that there is both a national and a regional program to develop and process uranium "yellow cake" being carried out in violation of NEPA. The Department of Justice is coordinating the defense of this lawsuit. On September 5, 1979, the Court denied plaintiff's motion for a preliminary injunction on the Mobil *in situ* pilot project.

*Charles Eason v. NRC* (D.D.C., No. 79-0845).

Plaintiff in this FOIA lawsuit seeks access to the *Media Monitor*, a copyrighted document produced by an outside contractor for the NRC. This case has progressed through discovery and is pending before the Court on NRC's motion for summary judgment. NRC's position is that plaintiff was never denied access to the document since it is available in the NRC Public Document Room for inspection. Such availability does not constitute a denial under the FOIA.

*Commonwealth of Kentucky v. NRC* (D.C. Cir., No. 78-1369).

The Commonwealth of Kentucky seeks review of the appeal board's decision of February 16, 1978, and of a licensing board decision of April 4, 1978, defining the Kentucky/Indiana border for purposes of deciding from which State the utility license applicant must obtain a Section 401 water quality certificate for its Marble Hill facility. On June 27, 1978, the D.C. Circuit dismissed the petition for review insofar as it related to the licensing board decision, but retained jurisdiction over the petition to the extent it sought review of the appeal board's February 16 decision. This case has been argued and awaits decision.

*A. R. Martin-Trigona v. Department of Justice, et al.* (S.D. Ill., No. 78-4006).

On January 30, 1978, plaintiff sued the Justice Department, Commonwealth Edison, and the NRC concerning the withholding under the FOIA of documents pertaining to the Quad Cities power station. NRC is asserting Exemption 7 as grounds for withholding the documents. A motion to dismiss was filed on April 21, 1978; the Court has not yet acted on that motion.

*Coalition for the Environment, St. Louis Region and Utility Consumers Council of Missouri v. NRC* (D.C. Cir., No. 77-1905).

On October 5, 1977, petitioners sued to suspend the construction permit for the Callaway Nuclear Plant based on a challenge to the Commission's interim fuel-cycle rule. On December 1, 1977, the Court held this case in abeyance until 30 days after the Supreme Court's decision in the *Vermont Yankee* case. On June 7, 1978, the Court requested the parties' views as to how this and the other fuel cycle cases should be handled. NRC advised that it should be dismissed or held in abeyance pending a final fuel cycle rule. The case is still in abeyance, as of the close of the report period.

*Rosanna Kelly v. Hendrie, et al.* (D.D.C., No. 79-1550).

Plaintiff alleges that, in her efforts to be promoted, she has suffered age and sex discrimination and that she has experienced retaliation as a result of initiating EEO procedures. Plaintiff seeks retroactive promotion and an injunction against discrimination. NRC's answer, filed in September 1979, denies the substantive allegations of her complaint.

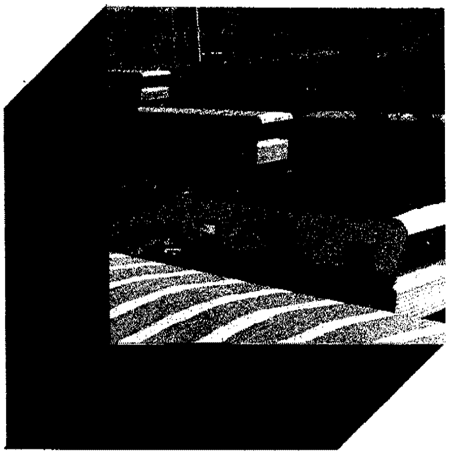
*State of New York v. NRC, et al.* (S.D.N.Y., No. 75 Civ. 2121).

*State of New York v. NRC, et al.* (2d Cir., Nos. 75-6115, 76-6022 and 76-6081).

*State of New York and State of Illinois v. NRC, et al.* (S.D.N.Y., 79 Civ. 4568).

State seeks to stop air shipment of plutonium, pending preparation of environmental impact statement. In No. 79 Civ. 4568, the two States challenge the adequacy of the NRC's environmental statement on transportation (NUREG-0170).

Motion for preliminary injunction was denied in September 1975. Appeal was taken to Second Circuit. State's motion for summary judgment was filed on December 11, 1975. The State's motion for summary judgment was denied, and it sought review of that decision in the Second Circuit. The two appeals were consolidated, as was a third appeal taken from the District Court's order dismissing the Civil Aeronautics Board and the Customs Service as parties to the litigation. Oral argument was heard on July 21, 1976. On February 14, 1977, the Second Circuit issued a 39-page opinion affirming the District Court's denial of plaintiff's motion for a preliminary injunction. The Court reasoned that plaintiff had failed to prove irreparable injury in view of the remoteness of a transportation accident and the absence of an agency commitment of resources to a particular transportation mode. The Second Circuit also dismissed, on procedural grounds that a final appealable order was lacking, the appeals from the District Court's dismissal of the CAB and Customs Service, and the denial of plaintiff's summary judgment motion. The Second Circuit noted NRC's expectation (stated in a November 17, 1976 letter to the U.S. Attorney) that the Commission's impact statement would be published early in February 1977. In fact, NUREG-0170 was issued in December 1977. On September 6, 1978, New York filed its amended complaint, challenging the adequacy of the environmental statement. On October 27, 1978, NRC filed the answer. With the filing of No. 79 Civ. 4568, these cases are being dismissed by stipulation of the parties. NRC's answer to No. 79 Civ. 4568, along with interrogatories and a request for documents, was filed on October 22, 1979.



# 14 Administration and Management

NRC began setting up its operations and support headquarters at the TMI site as soon as the scope of the accident was realized.

The NRC began fiscal year 1979 with an authorized full-time permanent personnel strength of nearly 2,800 and funding of $326 million. In response to the Three Mile Island accident, Congress approved in July an additional 100 positions and $5 million to be utilized by NRR, bringing authorized full-time permanent personnel strength to 2,888 and funding to $331 million by the end of fiscal year 1979. NRC Headquarters activities remained dispersed in 10 buildings throughout the District of Columbia and Maryland, with consolidation currently under review by the Senate Committee on Environment and Public Works. The Civil Service Reform Act of 1978, enacted October 13, 1978, has impacted employee and management practices in all Federal agencies. These and other management and administrative support developments, including organizational, personnel, and fiscal matters are discussed below.

## PERSONNEL AND ORGANIZATION

Augmented by an additional 100 positions from Congress during the fiscal year, NRC's authorized full-time permanent personnel strength had increased at year-end to 2,888—more than 6 percent above the fiscal year 1978 level of 2,723. Over 70 percent of NRC employees are located in the major program offices, about 20 percent in program direction and administration, and the remainder are employed in the Commission, Commission staff, and the independent advisory and adjudicatory bodies.

Of the 70 percent of NRC's employees holding college degrees, over 25 percent have masters degrees, almost 4 percent have professional (mostly law) degrees, and more than 9 percent have doctorates. Employees trained as scientists or engineers comprise over half of the NRC's work force.

### Commission and Office Director Appointments

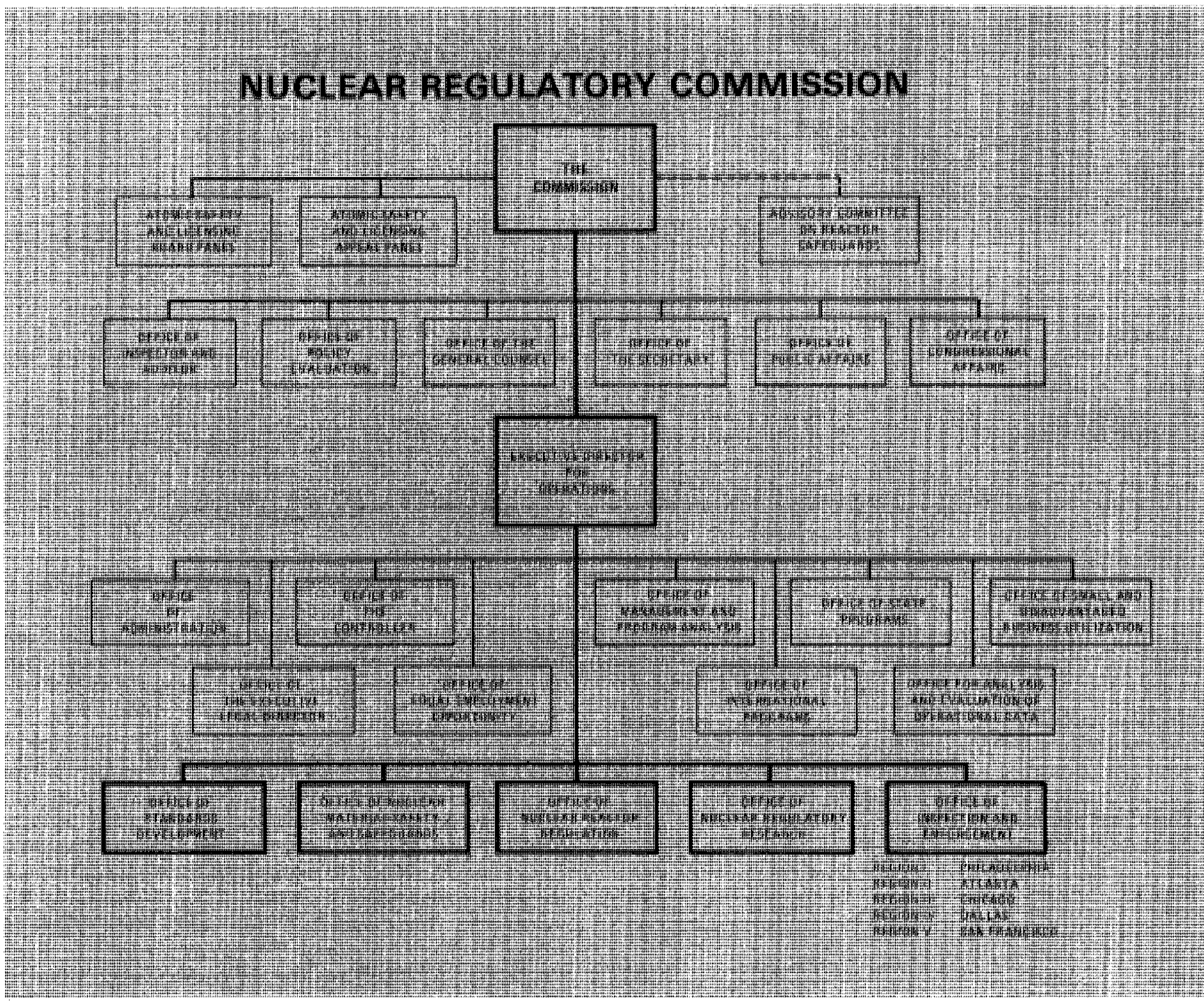
Victor Gilinsky was reappointed by the President to serve a second term as a member of the Commission. Mr. Gilinsky's reappointment is the first to the Commission, and extends from July 1, 1979 through June 30, 1984. The Commission thus continued at full five-member strength. On December 7, 1979, President Carter designated Commissioner John F. Ahearne to serve as Chairman pending selection of a new Chairman from outside the agency. (See Chapters 1 and 2.)

In the upper management of the program offices, Victor Stello, Jr., the former director of the Division of Operating Reactors, Nuclear Reactor Regulation, was named director of the Office of Inspection and Enforcement. This post had previously been filled by John G. Davis following the departure of Dr. Ernst Volgenau. In the Commission staff offices, Leonard Bickwit was named General Counsel, filling a post which had been vacated by Jerome Nelson, and Albert P. Kenneke was named acting director of the Office of Policy Evaluation on the departure of Kenneth S. Pedersen. In November 1979, Edward Hanrahan became director of the Office of Policy Evaluation.

Other key management changes included the following three appointments: Max W. Carbon was named chairman of the Advisory Committee on Reactor Safeguards; Robert M. Lazo, acting chairman of the Atomic Safety and Licensing Board Panel; and E. Kevin Cornell, Deputy Executive Director for Operations.

### Organizational Changes

The organization and normal functions of many NRC units were disrupted in the wake of the Three Mile Island accident in March 1979 as resources were reallocated and priorities shifted, both in direct support of recovery efforts at the site and in related ac-




tivities to pursue near-term and long-range objectives whose needs were revealed by the event.

The organizational structures of the Offices of Nuclear Reactor Regulation (NRR) and Inspection and Enforcement (IE) were particularly affected as several task groups launched activities on safety issues of the highest priority. In May, NRR revamped the priorities assigned to its activities and realigned its structure into an interim organization designed for optimum use of available resources. This required the temporary reassignment of many personnel and a realignment of managerial and technical talents which were expected to continue well into 1980.

Staff from these and many other offices were used from time to time at the Three Mile Island site—many on a voluntary basis—for support during the first few weeks following the accident. For most of the year, the NRC Recovery Operations Office, manned by technical and some administrative staff, operated from mobile office space at the TMI site. In December, efforts were in progress to lease office space in Middletown, Pa., for a projected long-term stay of NRC staff in conjunction with decontamination efforts of the licensee at TMI Unit 2.

Other staff dislocations included the assignment of 55 personnel from throughout the NRC to support the agency's Three Mile Island Special Investigation Group, formed in June, whose report to the Commission was expected to be issued in January 1980.

**Reorganization Impending.** President Carter, in his December 7, 1979 statement on the Kemeny Commission report concerning the TMI accident, announced he would send to Congress a reorganization plan to strengthen the role of the NRC Chairman, and that a new Chairman would be designated from outside the agency. In the meantime, he asked Commissioner Ahearne to serve as the NRC Chairman. (See Chapters 1 and 2.) The President also directed the Federal

rights and employee protections.
the establishment of merit prin-
ersonnel practices and actions, an
Labor Relations Authority, and
Service (SES).
pports the CSRA, with 98 percent
erting to the SES system. The Ex-
ard, a management structure to
S program at the NRC, was
ommission in August 1979. It is
e Directors, is chaired by the Ex-
Operations, and is organized into
ng Committee, the Performance
xecutive Resources Management
Executive Development Commit-

e CSRA, such as the merit pay
rs and probationary periods for
agers, were not required of the
excepted status of its personnel
is, however, undertaking studies
r to participate in these and other

**tation Election.** In late 1978, the
mployees Union (NTEU) petition-
of Labor seeking to represent
loyees in the five NRC regional of-
t election resulted in the certifica-

| TEMBER 30, 1979 | | |
|---|---|---|
| | WOMEN | |
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| 31 | 26 | 8 |
| 14 | 43 | 9 |
| 16 | 47 | 14 |
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rative determination.

tion of the NTEU, on January 26, 1979, as the exclusive representative of both headquarters and regional office employees in one bargaining unit.

**Subsequent Negotiations.** Soon after certification, the NTEU proposed the negotiation of an interim procedure for resolving employee grievances and grievances which may be brought by the union or by NRC management as an institution. These negotiations began on February 27, 1979 and were concluded on June 6, 1979, with full accord on an interim agreement governing the disposition of grievances and procedures for invoking arbitration.

On June 27, 1979, the NTEU forwarded proposals to NRC for a comprehensive bargaining agreement covering the full range of personnel policies and practices. These proposals were under study at year-end, with negotiations anticipated early in fiscal year 1980.

## Equal Employment Opportunity

**Progress at GS-11 and Up.** In fiscal year 1979, the NRC continued to submit quarterly reports to the Congress on minorities and women hired and promoted, and on other actions to improve the agency's EEO posture, with main emphasis on grade levels GS-11 and above. Of all new employees hired at GS-11 and higher grades during fiscal year 1979, 11.5 percent were from minority groups and 11.5 percent were women. Promotions to grades GS-11 and above numbered 374, with minority personnel representing 10.2 percent and women constituting 17.6 percent. On September 30, 1978, there were 126 minority and 143 women employees at GS-11 and above. By September 30, 1979, these figures had changed to 174 and 178, respectively.

**Recruiting Emphasis.** Recruiters visited 25 colleges and universities during the year to attract candidates for the NRC Intern, Cooperative Education, and Summer Intern Programs. Visits to nine of these schools were for the primary purpose of recruiting minority and women employees. Of the 145 persons hired under these programs, 81 were minority personnel and/or women. Continued EEO emphasis is expected to significantly increase the future representation of minorities and women in the higher grades.

**Special EEO Programs.** The special EEO programs highlighting Hispanic heritage, Black history, awareness training, and career counseling continued during fiscal year 1979. In addition, the NRC observed the first National Asian/Pacific Heritage Week with activities featuring contributions of Asian/Pacific Americans to the development of the United States.

**Women's Programs.** Throughout fiscal year 1979, the Federal Women's Program Advisory Committee continued meetings with FWP members and key office directors to discuss questions raised by female employees. FWP managers were appointed to each of the NRC's five regional offices. The FWP also conducted eight weekly workshops on "Leadership for Women," which examined the various techniques, characteristics, and problems of leadership and success.

In June 1979, the NRC's FWP headquarters manager was elected president of the Suburban Maryland Chapter/Federally Employed Women, Inc., while the legal advisor to the FWPAC received FEW's Outstanding Achievement Award. FWP members also participated in the United Nations Mid-Decade Conference for Women, the first Minority Women's Conference, the Image Convention, the Women's Network Reception for U.S. Congresswomen, the Tenth Anniversary of FEW, Inc., receptions for the Women's Airforce Service Pilots, salute to Asian/Pacific American Women, and conferences for FEW and the Society for Women Engineers.

**Class Actions.** Four class action complaints are pending against the NRC alleging sex discrimination. An Equal Employment Opportunity Commission

**The Howard University Jazz Ensemble played to a good "house" during noon-hour performances for members of the NRC staff. The group was featured as part of NRC's Black History observance in 1979.**

(EEOC) complaints examiner has been appointed in the first case to conduct a hearing on the issue of whether NRC's policy of auditing positions in response to requests for promotions has had a discriminatory impact on females. The three remaining complaints are pending EEOC's recommendations regarding procedural sufficiency and a determination as to whether the four cases should be consolidated.

## INSPECTION AND AUDIT

The Office of Inspector and Auditor (IA) conducts audits, investigations and inspections to assure the effectiveness, efficiency and integrity of NRC operations. Its responsibilities have included reviews of employee complaints and financial, compliance and management audits, as well as liaison functions with the General Accounting Office and Department of Justice. Some of the more important IA activities are summarized below.

### Procurement of Goods and Services

At the request of the Executive Director for Operations, IA performed an inspection to determine whether NRC's procurements of goods and services were accomplished properly and whether such goods and services were fully received. This review was both timely and significant in light of the recent investigations of procurement abuses in the Federal government. IA's report of June 1979 evaluated internal controls and procedures, reviewed the certification of payment processes, and commented on the custody and accountability of certain goods ordered and received. Based on selected tests and observations, IA determined that most goods were received and services rendered. However, some discrepancies existed. The IA report pointed out a need for better planning to avoid waste. IA believes that the EDO's proposed corrective actions should resolve the problems noted in the report.

### Resident Inspection Program

The accident at Three Mile Island (TMI) altered the original scope of this review to include assessment of problems and issues which might occur if implementation of the Resident Inspection Program at reactor facilities was accelerated. The report, issued in June 1979, concludes that accelerating this program would be difficult and would reduce the scope and quality of the inspection effort now being performed.

### Guidance for Safeguards Upgrade Rule

A report issued in December 1979 reviews the planning and development of guidance to licensees in response to the July 24, 1979, proposed rule to upgrade physical security at fuel cycle facilities. In December 1976, the Commission endorsed a general upgrading based on the recommendations of a Joint ERDA/NRC Task Force; since January 1977, the safeguards staffs have been in the process of developing the rule and the guidance necessary to implement it.

The IA report discloses that a fragmented approach to the planning and development of safeguards information has resulted in nearly $400,000 worth of technical reports that will not be used as guidance, and in the use of outside assistance for the development of information that could have been done in-house. The report also identifies the NRC's need to improve safeguards interagency coordination to help prevent possible duplication of safeguards information in areas where other agencies have parallel interests and similar ongoing projects. It further criticizes the usefulness of certain documents in a draft compendium of existing NRC and non-agency guidance related to the upgrade rule. The compendium was first released in March 1979, and contains language that may be confusing or misleading to licensees, as well as guidance documents and evaluation tools which may be crucial to licensees in upgrading their physical security systems, but which may not be available during the rule's required implementation period.

### Automatic Data Processing

Two reports were issued which review accountability of automatic data processing equipment and requirements, and management of NRC's ADP systems. In March 1979, IA issued a report assessing NRC's ability to determine long-range ADP requirements. This contained recommendations to strengthen controls over the procurement of and accountability for ADP equipment. The second report, issued in August 1979, concentrated on NRC's management and organizational structure for ADP. As in the first report, IA made recommendations for a more economical and efficient operation.

### Review of Semiscale Program

A report on IA's review of the Semiscale research program was issued in September 1979. The review confirms that this program and related activities are being carried out in an efficient and effective manner;

however, it also reveals a need for improvement in the areas of analysis of direct and indirect costs and in the dissemination and use of Semiscale technical reports. The report makes appropriate recommendations.

### Levels of Review of a Significant Report

IA conducted an investigation to chronologically reconstruct, up until the March 28, 1979 accident at TMI, the events and levels of review in the analysis of the report, "Decay Heat Removal During a Very Small Break LOCA for a B&W 205 Fuel-Assembly PWR." This report, which was prepared in 1977 by an employee of the Tennessee Valley Authority, raised concerns of apparent relevance in the wake of the accident at TMI. (See also Chapter 2.) IA's investigation identified the extent of the NRC review process, and the rationale for certain actions taken by cognizant individuals in the chain of review.

## FUNDING AND BUDGET MATTERS

NRC resource charts and financial statements appear at the end of this chapter. These charts show allocations of authorized personnel and funds to the various NRC activities carried out during fiscal year 1979, and to those projected for fiscal year 1980.

The increase in personnel for fiscal year 1980 is mainly in support of the expanded resident inspector program, and to provide an additional staff of 100 to NRR for non-TMI related licensing activities. These positions were allocated by Congress after budget approval, since the TMI accident had not yet occurred.

The increase in funding for fiscal year 1980 results primarily from emergency core cooling system experiments for light water reactors requiring increased funds to maintain the research efforts initiated in prior years and the first full year of LOFT nuclear testing. Also, increased personnel compensation funding is required to support the added personnel noted earlier.

The financial statements following the charts are self-explanatory.

### Contracting and Reimbursable Work

Most of the NRC's operating funds are expended in reimbursable arrangements with other agencies and contracts for confirmatory research and technical assistance in every major area of the agency's activity.

Approximately $196 million was allocated to program support during fiscal year 1979, of which $171 million went for reimbursable work performed for the NRC by other Federal agencies. The Department of Energy's share of this was approximately $169 million for work performed in DOE's national laboratories and other facilities. This work included major research projects and experiments at the Loss-of-Fluid Test (LOFT) Facility, the Power Burst Facility, and the Semiscale Facility. (Specific research programs are described in Chapter 11.)

Technical and administrative assistance contracts (except work performed through DOE), as well as general purchases of all kinds, are administered through the Division of Contracts in the Office of Administration. Such contracts totaled more than $44 million during fiscal year 1979.

Major activities in the Division of Contracts to improve procurement practices during the fiscal year have focused on (1) publication of a final rule on the avoidance of organizational conflicts of interest; (2) promulgation of an NRC manual chapter on the treatment of unsolicited proposals in the agency; (3) development of internal procedures for the procurement of goods and services; (4) implementation of a bidder's mailing list system for firms who desire to furnish goods and services to the NRC; and (5) development of a management information system which will provide the data required by the Federal Procurement Data System.

Additionally, source selection criteria were formalized for determining the most appropriate source for fulfilling NRC requirements for goods and services.

**Representatives of TERA Advanced Services Corporation, contractor for the installation and operation of NRC's automated document control system, spent many hours in 1979 working with NRC personnel on use of the system. This photo shows TERA official Jim Long conducting a typical DCS training session.**

## AUTOMATED CONTROL OF DOCUMENTS

The NRC's computerized Document Control System (DCS) is operated under contract, and is used to index, store, and retrieve NRC documents. Each document received or generated by the NRC is indexed by bibliographic elements and microfiched for storage and remote access. This allows up to 12 index searches of the data base by computer and production of page images on remote video screens. The DCS is also providing periodic indexes and title list reports, including a daily accession list of documents placed in the Public Document Room, that were formerly produced under high cost time-sharing contracts.

The monthly *Title List of Documents Made Publicly Available* was first published in January 1979. This DCS computer-produced document replaces *Power Reactor Docket Information* (PRDI), which was issued through DOE's Technical Information Center, and contained only selected docketed information. The final issue of PRDI, also published in January 1979, was an annual cumulation of the information for 1978. The new publication includes docketed material associated with civilian nuclear power plants and other uses of radioactive material, as well as non-docketed material received and generated by the NRC pertinent to its regulatory role.

Following the Three Mile Island (TMI) accident, the contractor processed on an accelerated schedule all pre- and post-incident TMI-2 documents held by the NRC, and produced a titled microfiche for each. The DCS was then used to print complete title listings on request, cumulative through any date, to assist in the ongoing reviews of the accident. The search, indexing, and microfiche features of the DCS contributed significantly to the staff's efforts following the accident, and prompted a rescheduling of the major contract tasks to permit an accelerated backfit of other power plant docket files onto the data base.

The DCS facility is located near NRC headquarters in Bethesda, Md., and houses the contractor's staff of engineers, computer specialists, indexers, technical coders, and the computer and microfiche equipment. Approximately 1,000 documents per day are being processed through the facility.

## PHYSICAL FACILITIES

During fiscal year 1979, the NRC continued to house approximately 2,400 headquarters employees in 10 buildings—one located in the District of Columbia, and nine in the Maryland suburbs.

In October 1977, the House Committee on Public Works and Transportation approved a General Services Administration (GSA) report which proposed consolidation of NRC headquarters in a single facility to be constructed on an urban renewal site in Washington, D.C. (See 1977 NRC Annual Report, page 208.) However, as a result of differing views presented by employee representatives and Maryland officials and legislators at an April 1978 hearing of the Subcommittee on Public Buildings and Grounds of the Senate Committee on Environment and Public Works, that committee requested GSA to analyze several additional factors. One of the main factors to be considered was the feasibility of consolidating in Montgomery County, Maryland.

GSA's expanded study, submitted to the Senate committee on July 25, 1978, recommended that two locations in the District of Columbia and three sites in Montgomery County be considered. Prior to final action, however, the committee imposed a moratorium on approval of all pending leasing and construction activities. The moratorium had not been lifted by the end of the fiscal year.

In late 1979, GSA began amending the original report to Congress in order to update the factual information made obsolete by increased costs and by the NRC's expanded responsibilities resulting from the Three Mile Island accident. Congressional consideration of lifting the construction moratorium, and particular attention to alleviating NRC's headquarters dispersion, is expected.

## NRC LICENSE FEES

On March 23, 1978, the Commission adopted a revised schedule of license fees. (See 1978 Annual Report, pp. 253-255.) The schedule increased fees in several categories of applications and licenses, and established additional categories of cost recovery for government services. These new categories included inspections, amendments, applications filed by vendors and architect-engineers for approval of standardized designs, and renewals.

The NRC expects to recover less than 3 percent of its budget through the increased fees. Those collected in fiscal year 1979 amounted to $12.5 million, of which $2.6 million is held in a suspense account by the Department of Treasury until calculations of actual costs after action on the permit or license involved have been completed. The total collected since fees were first imposed in 1968, through September 1979, was $101.4 million, of which $6.5 million has been refunded.

On February 5, 1979, the United States Court of Appeals for the Fifth Circuit heard oral arguments by 17 utilities and 2 waste disposal licensees challenging the NRC's interpretation of the Independent Offices Appropriation Act of 1952, which is the statutory authority for the Commission schedule of fees in 10 CFR Part 170.

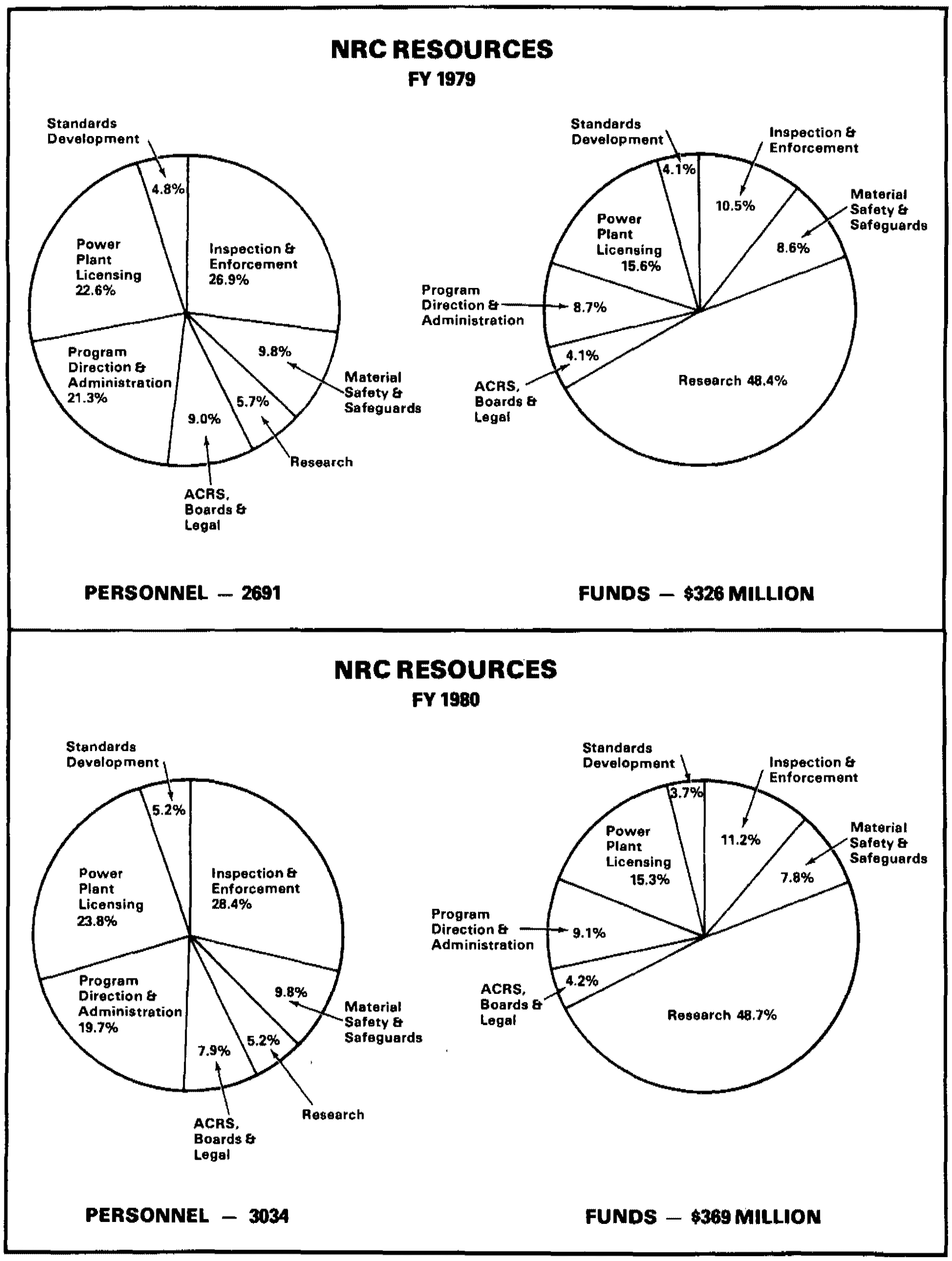
On August 24, 1979, the Fifth Circuit issued its opinion upholding in all respects the Commission's February 9, 1978 license fee schedule and its guidelines for fees. The Court specifically held that the NRC: (1) has the authority to recover the full cost of providing services to identifiable beneficiaries; (2) may recover the costs it incurs in conducting routine inspections, complying with the National Environmental Policy Act, conducting uncontested hearings, and providing administrative and technical support services; and (3) may assess fees to recover the costs of services it provides to Agreement State licensees who may also require an NRC license for certain activities.

## NATIONAL EMERGENCY PREPAREDNESS

During early 1979, efforts were continued toward insuring NRC capabilities in the event of a national emergency, including an attack on the United States. The accident at Three Mile Island, however, interrupted the work on national emergency preparedness as staff efforts were increased to assist State and local governments in their radiological emergency preparedness.

An agreement was reached to permit NRC to share the use of the Department of Housing and Urban Development's alternate emergency operating facility. This facility would be the place of operations for the NRC "Charlie" team, if the NRC management capability needed to be dispersed in an emergency.

After the Federal Emergency Management Agency was formed in July 1979, contacts and lines of communication were confirmed with the new agency for continued cooperation in national emergency preparedness.



# Fiscal Year 1979—NRC Financial Statements

## Balance Sheet (in thousands)

| Assets | September 30, 1979 | September 30, 1978 |
|---|---|---|
| Cash: | | |
| Appropriated Funds in U.S. Treasury | $ 146,257 | $ 129,149 |
| Other (See Notes 1 and 3) | 10,426 | 10,841 |
| | 156,683 | 139,990 |
| Accounts Receivable: | | |
| Federal Agencies | 222 | 142 |
| Miscellaneous Receipts (Note 2) | 5,986 | 3,259 |
| Other | 272 | 85 |
| | 6,480 | 3,486 |
| Plant: | | |
| Completed Plant and Equipment (Note 5) | 7,462 | 5,716 |
| Less—Accumulated Depreciation (Note 5) | 1,314 | 1,428 |
| | 6,148 | 4,288 |
| Advances and Prepayments: | | |
| Federal Agencies | 171 | 129 |
| Other | 1,304 | 984 |
| | 1,475 | 1,113 |
| Total Assets | $ 170,786 | $ 148,877 |

| Liabilities and NRC Equity | September 30, 1979 | September 30, 1978 |
|---|---|---|
| Liabilities: | | |
| Funds held for Others (See Notes 1 and 3) | $ 10,426 | $ 10,841 |
| Accounts Payable and Accrued Expenses: | | |
| Federal Agencies | 42,884 | 39,176 |
| Other | 20,323 | 15,318 |
| Accrued Annual Leave of NRC Employees | 6,285 | 5,552 |
| Deferred Revenue (Note 3) | 1,330 | 2,067 |
| Total Liabilities | 81,248 | 72,954 |
| NRC Equity: Balance @ October 1, 1978 | 75,924 | 68,914 |
| Additions: | | |
| Funds Appropriated-net | 326,601 | 290,023 |
| | 402,525 | 358,937 |
| Deductions: | | |
| Net Cost of Operations (Note 5) | 305,865 | 273,153 |
| Funds Returned to U.S. Treasury (Note 2) | 7,122 | 9,861 |
| | 312,987 | 283,014 |
| Total NRC Equity | 89,648 | 75,923 |
| Total Liabilities and NRC Equity | $ 170,786 | $ 148,877 |

Note 1. As of September 30, 1979, includes $2,504,163.30 of funds received under cooperative research agreements involving NRC, DOE, Federal Republic of Germany, Japan, Austria, and the Netherlands. Included also is $7,117,130.00 of funds received from deferred revenue billings. These funds will be refunded and/or recorded as earned revenue after the cost of processsing the applicable application has been finalized and, accordingly, are not available for NRC use. (See Note 3.)

Note 2. These funds are not available for NRC use.

Note 3. On March 24, 1978, 10 CFR Part 1 was revised. Contained therein by category of license are maximum fee amounts to be paid by applicants at the time a facility or material license is issued. Also, after the review of the license application is complete (generally after license has been issued), the expenditures for professional manpower and appropriate support services are to be determined and the resultant fee assessed. In no event will the fee exceed the maximum fee for that license category which generally has been paid. This could involve the refunding of a significant portion of the initial amount paid. Therefore, the revenue is recorded in a deferred revenue account at the time of billing and is removed from this account and recorded in Funds Held for Others when the bill is paid. The balance in the Deferred Revenue account consists of deferred revenue on billings issued but not collected. (See Note 1.)

## Fiscal Year 1978/1979 Statement of Operations (in thousands)

| | Fiscal Year 1979 (October 1, 1978, thru September 30, 1979) | Fiscal Year 1978 (October 1, 1977, thru September 30, 1978) |
|---|---|---|
| Personnel Compensation | $ 85,144 | $ 77,144 |
| Personnel Benefits | 7,649 | 7,172 |
| Program Support | 181,950 | 161,817 |
| Administrative Support | 27,325 | 19,120 |
| Travel of Persons | 6,123 | 5,378 |
| Training (Technical) | 585 | 759 |
| Equipment (Technical) (See Note 4) | 6,545 | 7,687 |
| Construction (See Note 4) | 10 | 1,672 |
| Taxes and Indemnities | 3 | 5 |
| Refunds to Licensees | 180 | 189 |
| Representational Funds | 9 | 9 |
| Reimbursable Work | 367 | 273 |
| Increase in Annual Leave Accrual | 733 | 709 |
| Depreciation Expense | 547 | 469 |
| Equipment Write-offs and Adjustments | 26 | 229 |
| Total Cost of Operations | $ 317,403 | $ 282,632 |
| Less Revenues: | | |
| Reimbursable Work for Other Federal Agencies | 367 | 273 |
| Fees (deposited in U.S. Treasury as Miscellaneous Receipts (See Note 2)) | | |
| Indemnity | 1,035 | 1,793 |
| Material Licenses | 1,605 | 321 |
| Facility Licenses | 7,810 | 7,383 |
| Other | 137 | 1,188 |
| Total Revenue | 10,954 | 10,958 |
| Net Cost of Operations before Prior Year Adjustment | 306,449 | 271,674 |
| Prior Year Adjustment (See Note 5) | 584 | 1,479 |
| Net Cost of Operations | $ 305,865 | $ 273,153 |

## U.S. Government Investment In The Nuclear Regulatory Commission

### (From January 19, 1975, Through September 30, 1979—in thousands)

| | |
|---|---|
| Appropriation Expenditures: | |
| Fiscal Year 1975 (January 19, 1975, through June 30, 1975) | $ 52,792 |
| Fiscal Year 1976 (July 1, 1975, through September 30, 1976) | 226,248 |
| Fiscal Year 1977 (October 1, 1976, through September 30, 1977) | 230,559 |
| Fiscal Year 1978 (October 1, 1977, through September 30, 1978) | 270,877 |
| Fiscal Year 1979 (October 1, 1978, through September 30, 1979) | $ 309,493 |
| | 1,089,969 |
| Unexpended Balance of Appropriated Funds in U.S. Treasury, September 30, 1978 | 146,257 |
| Transfer of Refunds Receivable from Atomic Energy Commission, January 19, 1975 | 429 |
| Total Funds Appropriated | $1,236,655 |
| Less: | |
| Funds returned to U.S. Treasury (See Note 2) | 50,741 |
| Assets and Liabilities transferred from Other Federal Agencies without Reimbursement | 2,018 |
| Net Cost of Operations from January 19, 1975, through September 30, 1979 | 1,094,358 |
| Total Deductions | 1,147,117 |
| NRC Equity at September 30, 1979, as shown on Balance Sheet | $ 89,538 |

Note 4. Represents current year cost of plan and equipment acquisitions for use at DOE facilities.

Note 5. During fiscal year 1979, net cost of operations was reduced by $584,478.98 for net write-ons of capital equipment resulting from the reconciliation of the November 1977 physical inventory.

# Appendix 1

# NRC ORGANIZATION

(As of September 30, 1979)

## COMMISSIONERS

Joseph M. Hendrie, Chairman*
Victor Gilinsky
Richard T. Kennedy
Peter A. Bradford
John F. Ahearne

### The Commission Staff

General Counsel, Leonard Bickwit
Office of Policy Evaluation, Albert P. Kenneke, Acting Director**
Office of Public Affairs, Joseph J. Fouchard, Director
Office of Congressional Affairs, Carlton C. Kammerer, Director
Office of Inspector and Auditor, James J. Cummings, Director
Secretary of the Commission, Samuel J. Chilk

### Other Offices

Advisory Committee on Reactor Safeguards, Max W. Carbon, Chairman
Atomic Safety & Licensing Board Panel, Robert M. Lazo, Acting Chairman
Atomic Safety & Licensing Appeal Panel, Alan S. Rosenthal, Chairman

## EXECUTIVE DIRECTOR FOR OPERATIONS

Executive Director for Operations, Lee V. Gossick
Deputy Executive Director for Operations, E. Kevin Cornell
Technical Advisor, Vacant

### Program Offices

Office of Nuclear Reactor Regulation, Harold R. Denton, Director
Office of Nuclear Material Safety and Safeguards, William J. Dircks, Director
Office of Nuclear Regulatory Research, Saul Levine, Director
Office of Standards Development, Robert B. Minogue, Director
Office of Inspection and Enforcement, Victor Stello, Jr., Director

### Staff Offices

Office of Administration, Daniel J. Donoghue, Director
Executive Legal Director, Howard K. Shapar
Controller, Learned W. Barry
Office of Equal Employment Opportunity, Edward E. Tucker, Director
Office of Management and Program Analysis, Norman M. Haller, Director
Office of International Programs, James R. Shea, Director
Office of State Programs, Robert G. Ryan, Director
Office for Analysis and Evaluation of Operational Data, C. J. Heltemes, Interim Director

### Regional Offices

Region I Philadelphia, Pa., Boyce H. Grier, Director
Region II Atlanta, Ga., James P. O'Reilly, Director
Region III Chicago, Ill., James G. Keppler, Director
Region IV Dallas, Texas, Karl V. Seyfrit, Director
Region V San Francisco, Calif., Robert H. Engelken, Director

*On December 7, 1979, President Carter designated Commissioner Ahearne to serve as Chairman.

**Edward Hanrahan was named Director of PE in November 1979.

**The NRC** is responsible for licensing and regulating nuclear facilities and materials and for conducting research in support of the licensing and regulatory process, as mandated by the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and the Nuclear Non-proliferation Act of 1978; and in accordance with the National Environmental Policy Act of 1969, as amended, and other applicable statutes. These responsibilities include protecting public health and safety, protecting the environment, protecting and safeguarding materials and plants in the interest of national security; and assuring conformity with antitrust laws. Agency functions are performed through: standards-setting and rulemaking; technical reviews and studies; conduct of public hearings; issuance of authorizations, permits and licenses; inspection, investigation and enforcement; evaluation of operating experience, and confirmatory research. The Commission itself is composed of five members, appointed by the President and confirmed by the Senate, one of whom is designated by the President as Chairman. The Chairman is the principal executive officer and the official spokesman of the Commission.

**The Executive Director for Operations** directs and coordinates the Commission's operational and administrative activities and the development of policy options for Commission consideration.

**The Office of Nuclear Reactor Regulation** licenses nuclear power, test and research reactors under a two phase process. A construction permit is granted before facility construction can begin and an operating license is issued before fuel can be loaded. NRR reviews license applications to assure that the proposed facility can be built and operated without undue risk to the health and safety of the public and with minimal impact on the environment. NRR monitors operating reactor facilities during their lifetime through decommissioning. NRR also reviews the financial responsibility of each applicant for a construction permit, confirms that each applicant is properly indemnified against accidents, and verifies that the applicant(s) is not in violation of antitrust laws.

**The Office of Nuclear Material Safety and Safeguards** is responsible for ensuring public health and safety, and protection of national security and environmental values in the licensing and regulation of facilities and materials associated with the processing, transport, and handling of nuclear materials. NMSS reviews and assesses safeguards against potential threats, thefts, and sabotage, and works closely with other NRC organizations in coordinating safety and safeguards programs and in recommending research, standards, and policy options necessary for their successful operation.

**The Office of Nuclear Regulatory Research** plans and implements research programs of nuclear regulatory research which are deemed necessary for the performance of the Commission's licensing and regulatory functions. Research programs cover reactor safety areas such as materials behavior, site safety, systems engineering, and computer code development and assessment. Research is also performed on safeguards, health effects associated with the nuclear fuel cycle, environmental impact of nuclear power, waste treatment and disposal, and transportation of radioactive materials.

**The Office of Standards Development** develops regulations, guides, and other standards needed for regulation of facilities and materials with respect to radiological health and safety and environmental protection, for materials safeguards and plant protection, and for antitrust review. The Office also coordinates NRC participation in national and international standards activities.

**The Office of Inspection and Enforcement** inspects nuclear facilities and materials licensees to determine whether facilities are constructed and operations are conducted in compliance with license provisions and Commission regulations, and to identify conditions that may adversely affect the protection of nuclear materials and facilities, the environment, or the health and safety of the public; inspects applicants and their facilities to provide a basis for recommending issuance or denial of licenses; investigates accidents, incidents, and allegations of improper actions that involve nuclear material and facilities; and enforces NRC regulations and license provisions. IE, on behalf of NRC, manages and directs the Commission's five regional offices, located in Philadelphia, Pa., Atlanta, Ga., Chicago, Ill., Dallas, Texas, and San Francisco, Calif.

## THE COMMISSION STAFF

**The Office of the Secretary** provides secretariat services for the conduct of Commission business and implementation of decisions, including planning meetings and recording deliberations, manages the staff paper system, monitors the status of actions, and maintains the Commission's official records. The office also processes institutional correspondence, controls the service of documents in adjudicatory and public proceedings, supervises the Washington, D.C. Public Document Room, administers the NRC historical program, and provides administrative support for the Commission.

**The Office of General Counsel** serves the Commission in a variety of legal capacities. The Office assists the Commission in the review of Appeal Board decisions, petitions seeking direct Commission relief, and rulemaking proceedings, and drafts legal documents necessary to carry out the Commission's decisions. The General Counsel provides a legal analysis of proposed legislation affecting the Commission's functions and assists in drafting legislation and preparing testimony. The General Counsel also represents the Commission in court proceedings, frequently in conjunction with the Department of Justice.

**The Office of Policy Evaluation** plans and manages activities involved in performance of an independent review of positions developed by the NRC staff which require policy determinations by the Commission. The Office also conducts analyses and projects which are either self-generated or requested by the Commission.

**The Office of the Inspector and Auditor** investigates to ascertain the integrity of all NRC operations; investigates allegations of NRC employee misconduct, equal employment and civil rights complaints, and claims for personal property loss or damage; conducts the NRC's internal audit activities; and hears individual employee concerns regarding Commission activities under the agency's "Open Door" policy. The office develops policies governing the Commission's financial and management audit program and is the agency contact with the General Accounting Office on this function. Refers criminal matters to the Department of Justice and maintains liaison with law enforcement agencies.

**The Office of Public Affairs** plans and administers NRC's program to inform the public of Commission policies, programs and activities and keeps NRC management informed of public affairs activities of interest to the Commission.

**The Office of Congressional Affairs** provides advice and assistance to the Commission and senior staff on congressional matters, coordinates NRC's congressional relations activities, and maintains liaison for the Commission with congressional committees and members of Congress.

## SUPPORT STAFF

**The Office of Administration** directs the agency's programs for organization and personnel management; security and classification; technical information and document control; facilities and materials license fees; contracting and procurement; rules, proceedings and document services; data processing; management development and training; and other administrative housekeeping and special services.

**The Office of the Controller** develops and maintains the Commission's financial management program, including accounting, budgeting, pricing, contract finance, automatic data processing equipment acquisition, and accounting for capitalized property. Prepares reports necessary to the management of NRC funds. Maintains liaison with the General Accounting Office, Office of Management and Budget, Congressional Committees, other agencies, and industry. The Controller also prepares the NRC Five-Year Plan and performance resource evaluation studies.

**The Office of the Executive Legal Director** provides legal advice and services to the Executive Director for Operations and staff, including representation in administrative proceedings involving the licensing of nuclear facilities and materials, and the enforcement of license conditions and regulations; counseling with respect to safeguards matters, contracts, security, patents, administration, research, personnel, and the development of regulations to implement applicable Federal statutes.

**The Office of Equal Employment Opportunity** develops and recommends overall policy providing for equal employment opportunity, recommends improvements or corrections to achieve this goal, and monitors the agency's affirmative action program.

**The Office of International Programs** plans and implements programs of international nuclear safety cooperation, creating and maintaining relationships with foreign regulatory agencies and international organizations; coordinates NRC export-import and international safeguards policies; issues export and import licenses; and coordinates responses by NRC to other agencies related to export-import actions and issues.

**The Office of Management and Program Analysis** provides NRC staff with management information and program analyses; identifies and analyzes major NRC policy, program and management issues and conducts long- and short-range planning to assist NRC operating officials; develops and implements management information and control systems and recommends policy on use of such systems for agency-wide applications; develops and implements application of sound statistical practices within NRC; and coordinates special information projects on overall NRC policies and programs.

**The Office of State Programs** directs programs relating to regulatory relationships with State governments and organizations and interstate bodies; manages the NRC State Agreements program; and provides Federal agency leadership in assisting State and local governments in radiological emergency response planning.

**The Office for Analysis and Evaluation of Operational Data** assures the proper analysis of operational data associated with all NRC-licensed activities and the feedback of such analyses to improve safety. The office identifies key analyses to be conducted, taking into account such factors as postulated accident sequences and data availability; selects appropriate analytical techniques and proposes data gathering mechanisms for data not currently available; conducts systematic safety analyses and evaluations of operational data to seek trends that would forecast a potential problem; develops recommendations to resolve problems revealed by operational data analyses and evaluations; provides analytical guidance to, accepts technical input from, and coordinates efforts of operational data analysis groups in other NRC offices; reviews overall NRC and industry response to assess implementation of recommended actions; and serves as focal point for interaction with the ACRS and industry groups involved in operational data analysis and evaluation.

## OTHER OFFICES

**Advisory Committee on Reactor Safeguards.** A statutory committee of 15 scientists and engineers, advises the Commission on the safety aspects of proposed and existing nuclear facilities and the adequacy of proposed reactor safety standards, and performs such other duties as the Commission may request.

**Atomic Safety and Licensing Board Panel.** Three-member licensing boards drawn from the Panel-made up of lawyers and others with expertise in various technical fields-conduct public hearings and make such intermediate or final decisions as the Commission may authorize in proceedings to grant, suspend, revoke, or amend NRC Licenses.

**Atomic Safety and Licensing Appeal Panel.** Three-member appeal boards selected from the Panel exercise the authority and perform the review functions which would otherwise be carried out by the Commission in licensing proceedings. ASLB decisions are reviewable by an appeal board, either in response to an appeal or on its own initiative. The appeal board's decision also is subject to review by the Commission on its initiative or in response to a petition for discretionary review.

# Appendix 2

# NRC Committees and Boards

**ɨeactor Safeguards**

statutory committee in 1957 by Sec-
ergy Act of 1954, as amended. The
studies and facility license applica-
ɹrdance with the Atomic Energy Act
ation Act and makes reports thereon
he public record of the proceeding.
dvice with respect to the hazards of
cilities and the adequacy of related
mittee also performs such other addi-
ission may request. The members are
erms by the Commission. The com-
wn chairman and vice chairman. As
e members were:

*Chairman*, Professor and Chairman
Department, University of Wiscon-

or, Engineering Division, Oak Ridge
ak Ridge, Tenn.
red Head Nuclear Engineer, Division
, Tennessee Valley Authority, Knox-

ON, Consulting Engineer (Mechani-
g), Jupiter, Fla.
R, Professor of Nuclear Engineering,
Memorial-Phoenix Project, University
or, Mich.
OSKI, Senior Engineer, Chemical
, Argonne National Laboratory,

IS, Department of Physics, Universi-
Barbara, Calif.
Retired Division Leader, Los Alamos
Los Alamos, N.M.
, Retired Director, Planning, United
., Richland, Wash.
ER, Chairman, Department of En-
Sciences, School of Public Health,
oston, Mass.
Professor, School of Engineering and
versity of California, Los Angeles,

SET, Professor, Department of Engi-
meritus, California Institute of
, Calif.
Retired Chief Electrical Engineer,
Company, Philadelphia, Pa.
ION, Professor, Chairman of Metal-
Department, Ohio State University,

S, Professor, Head of Civil Engineer-
ersity of Illinois, Urbana, Ill.

**Atomic Safety and Licensing Board Panel**

Section 191 of the Atomic Energy Act of 1954 authorizes the Commission to establish one or more atomic safety and licensing boards, each comprised of three members, one of whom is to be qualified in the conduct of administrative proceedings and two of whom will have such technical or other qualifications as the Commission deems appropriate to the issues to be decided. The boards conduct such hearings as the Commission may direct and make such intermediate or final decisions as it may authorize in proceedings with respect to granting, suspending, revoking, or amending licenses or authorizations. The Atomic Safety and Licensing Board Panel (ASLBP) Office—with a permanent chairman who coordinates and supervises the ASLBP activities—serves as spokesman for the panel, and makes policy recommendations to the Commission concerning conduct of hearings and hearing procedures. Pursuant to subsection 201 (g)(1) of the Energy Reorganization Act of 1974, the functions performed by the licensing boards were specifically transferred to the Nuclear Regulatory Commission. As of September 30, 1979 the ASLBP was composed of the following members and professional staff ("*" denotes full-time ASLBP members and staff):

ROBERT M. LAZO, *Acting Chairman*, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.*
DR. GEORGE C. ANDERSON, Department of Oceanography, University of Washington, Seattle, Wash.
CHARLES BECHHOEFER, ASLAB Attorney, Bethesda, Md.*
ELIZABETH S. BOWERS, ASLBP Attorney, Bethesda, Md.*
JOHN H. BREBBIA, Attorney with law firm of Alston, Miller & Gaines, Washington, D.C.
GLENN O. BRIGHT, ASLBP Engineer, Bethesda, Md.*
DR. A. DIXON CALLIHAN, Retired Physicist, Union Carbide Corporation, Oak Ridge, Tenn.
DR. E. LEONARD CHEATUM, Retired Director of Institute of Natural Resources, University of Georgia, Watkinsville, Ga.
HUGH K. CLARK, Retired Attorney, E. I. duPont de Nemours & Company, Kennedyville, Md.
DR. RICHARD F. COLE, ASLBP Environmental Scientist, Bethesda, Md.*
DR. FREDERICK P. COWAN, Retired Physicist, Brookhaven National Laboratory, Stuart, Fla.
VALENTINE B. DEALE, Attorney at Law, Washington, D.C.
RALPH S. DECKER, Retired Engineer, U.S. Atomic Energy Commission, Cambridge, Md.
DR. DONALD P. DE SYLVA, Professor, Biology and Living Resources, School of Marine and Atmospheric Science, University of Miami, Miami, Fla.

MICHAEL A. DUGGAN, College of Business Administration, University of Texas, Austin, Tex.
DR. GEORGE A. FERGUSON, Professor of Nuclear Engineering, Howard University, Washington, D.C.
DR. HARRY FOREMEN, Director, Center for Population Studies, University of Minnesota, Minneapolis, Minn.
JOHN H. FRYE, III, ASLBP Legal Counsel, Bethesda, Md.*
MICHAEL GLASER, Partner, law firm of Glaser and Fletcher, Washington, D.C.
ANDREW C. GOODHOPE, Retired Administrative Law Judge, Federal Trade Commission, Wheaton, Md.
HERBERT GROSSMAN, ASLBP Attorney, Bethesda, Md.*
DR. DAVID B. HALL, Los Alamos Scientific Laboratory, Los, Alamos, N.M.
DR. CADET H. HAND, JR., Director, Bodega Marine Laboratory, University of California, Bodega Bay, Calif.
DR. DAVID L. HETRICK, Professor, Nuclear Engineering Department, University of Arizona, Tucson, Ariz.
ERNEST E. HILL, Engineer, Lawrence Livermore Laboratory, University of California, Livermore, Calif.
DR. ROBERT L. HOLTON, School of Oceanography, Oregon State University, Corvallis, Ore.
DR. FRANK F. HOOPER, Chairman, Resource Ecology Program, School of Natural Resources, University of Michigan, Ann Arbor, Mich.
ELIZABETH B. JOHNSON, Engineer, Oak Ridge National Laboratory, Oak Ridge, Tenn.
DR. WALTER H. JORDAN, Retired Senior Research Advisor & Physicist, Oak Ridge National Laboratory, Oak Ridge, Tenn.
DR. JAMES C. LAMB, III, Department of Environmental Sciences & Engineering, University of North Carolina, Chapel Hill, N.C.
MARGARET M. LAURENCE, Partner, law firm of Laurence, Stokes and Neilan, Arlington, Va.
DR. J. V. LEEDS, JR., Professor, Environmental and Electrical Engineering, Rice University, Houston, Tex.
GUSTAVE A. LINENBERGER, ASLBP Physicist, Bethesda, Md.*
DR. LINDA W. LITTLE, Research Triangle Institute, Research Triangle Park, N.C. Department of Environmental Sciences & Engineering, University of North Carolina, Chapel Hill, N.C.
DR. M. STANLEY LIVINGSTON, Retired Associate Director, Atomic Energy Commission National Accelerator Laboratory, Santa Fe, N.M.
DR. EMMETH A. LUEBKE, ASLBP Physicist, Bethesda, Md.*
DR. WILLIAM E. MARTIN, Senior Ecologist, Battelle Memorial Institute, Columbus, Ohio
DR. KENNETH A. McCOLLOM, Dean, Division of Engineering, Technology and Architecture, Oklahoma State University, Stillwater, Okla.
GARY L. MILHOLLIN, University of Wisconsin Law School, Madison, Wis.
MARSHALL E. MILLER, ASLBP Attorney, Bethesda, Md.*
DR. OSCAR H. PARIS, ASLBP Environmental Scientist, Bethesda, Md.*
DR. HUGH PAXTON, Los Alamos Scientific Laboratory, Los Alamos, N.M.
DR. PAUL W. PURDOM, Director, Environmental Studies Institute, Drexel University, Philadelphia, Pa.
DR. FORREST J. REMICK, Director, Institute of Science and Engineering, Pennsylvania State University, University Park, Pa.
DR. DAVID R. SCHINK, Department of Oceanography, Texas A&M University, College Station, Tex.
FREDERICK J. SHON, ASLBP Physicist, Bethesda, Md.*
IVAN W. SMITH, Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.*
DR. MARTIN J. STEINDLER, Chemist, Argonne National Laboratory, Argonne, Ill.
DR. QUENTIN J. STOBER, Research Associate Professor, Fisheries Research Institute, University of Washington, Seattle, Wash.
JOSEPH F. TUBRIDY, Attorney at Law, Washington, D.C.
SEYMOUR WENNER, Retired Administrative Law Judge, Postal Rate Commission, Washington, D.C.
JOHN F. WOLF, Attorney, law firm of Lamensdorf, Leonard & Moore, Washington, D.C.
SHELDON J. WOLFE, ASLBP Attorney, Bethesda, Md.*

**Atomic Safety and Licensing Appeal Panel**

An Atomic Safety and Licensing Appeal Board, established effective September 18, 1969, was delegated the authority to perform the review function which would otherwise be performed by the Commission in proceedings on applications for licenses or authorizations in which the Commission had a direct financial interest, and in such other licensing proceedings as the Commission might specify.

In view of the increase in the number of proceedings subject to administrative appellate review, the Atomic Safety and Licensing Appeal Panel was established on October 25, 1972, from whose membership three-member appeal boards could be designated for each proceeding in which the Commission had delegated its authority to an appeal board. At the same time, the Commission modified its rules to delegate authority to appeal boards in all proceedings involving the licensing of production and utilization facilities (for example, power reactors).

Pusuant to subsection 201 (g)(1) of the Energy Reorganization Act of 1974, the functions performed by appeal boards were specifically transferred to the Nuclear Regulatory Commission. The Commission appoints members to the Appeal Panel, and the Chairman of the panel (or, in his absence, the Vice Chairman) designates a three-member appeal board for each proceeding. The Commission retains review authority over decisions and actions of appeal boards. The appeal board panel, on September 30, 1979, was composed of the following full-time members and professional staff:

ALAN S. ROSENTHAL, Appeal Panel *Chairman*, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DR. JOHN H. BUCK, Appeal Panel *Vice Chairman*, U.S. Nuclear Regulatory Commission, Bethesda, Md.
MICHAEL C. FARRAR, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
RICHARD S. SALZMAN, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JOHN CHO, Counsel, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.

CARDIS L. ALLEN, Technical Advisor, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
LINDA S. GILBERT, Legal Intern, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
ROBERT S. PERLIS, Legal Intern, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.

During fiscal year 1979, the Appeal Panel also included the following part-time members:

DR. LAWRENCE R. QUARLES, Dean Emeritus, School of Engineering and Applied Science, University of Virginia, Charlottesville, Va.
DR. W. REED JOHNSON, Professor of Nuclear Engineering, University of Virginia, Charlottesville, Va.

**Advisory Committee on Medical Uses of Isotopes**

The Advisory Committee on Medical Uses of Isotopes was established in July 1958. The ACMI, composed of qualified physicians and scientists, considers medical questions referred to it by the NRC staff, and renders expert opinion regarding medical use of radioisotopes. The ACMI also advises the NRC staff, as requested, on matters of policy. Members are employed under yearly personal services contracts. The Deputy Director, Division of Fuel Cycle and Material Safety, serves as Committee Chairman. As of September 30, 1979 the members were:

RICHARD E. CUNNINGHAM, *Chairman*, ACMI, Deputy Director, Division of Fuel Cycle and Material Safety, U.S. Nuclear Regulatory Commission, Silver Spring, Md.

DR. FRANK H. DE LAND, Chief, Nuclear Medicine Department, Veterans' Administration Hospital, Lexington, Ky.

DR. EDWARD W. WEBSTER, Director, Department of Radiation Physics, Massachusetts General Hospital, Boston, Mass.

DR. JOSEPH B. WORKMAN, Associate Professor of Radiology, Duke University Medical Center, Durham, N.C.

DR. VINCENT P. COLLINS, Medical Director, Houston Institute for Cancer Research, Diagnosis and Treatment, Houston, Tex.

DR. MELVIN L. GRIEM, Professor and Director, Chicago Tumor Institute, University of Chicago, Chicago, Ill.

DR. SALLY DENARDO, Director, Nuclear Hematology-Oncology, Department of Nuclear Medicine, University of California-Davis Medical Center, Sacramento, Ca.

DR. JACK GOODRICH, Radiology Associates of Erie, Hamot Medical Center, Erie, Pa.

DR. B. LEONARD HOLMAN, Chief, Clinical Nuclear Medicine, Department of Radiology, Peter Bent Brigham Hospital, Boston, Ma.

DR. DAVID H. WOODBURY, Director, Nuclear Medicine, Wayne County General Hospital, Eloise, Mi.

# Appendix 3

# Public Document Rooms

Most documents originated by NRC, or submitted to it for consideration, are placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C., for public inspection. In addition, documents relating to licensing proceedings or licensed operation of specific facilities are made available in local public document rooms established in the vicinity of each proposed or existing nuclear facility. The locations of these local PDRs as of December 1979, and the name of the facility for which documents are retained, are listed below. (NOTE: Updated listings of local PDRs may be obtained by writing to the Local Public Document Room Branch, Division of Rules and Records, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.)

## ALABAMA

- Mrs. Maude S. Miller
  Athens Pubic Library
  South and Forrest
  Athens, Ala. 35611
  Browns Ferry Nuclear Plant
- Mr. Wayne Love
  G.S. Houston Memorial Library
  212 W. Burdeshaw Street
  Dothan, Ala. 36303
  Farley Nuclear Plant
- Mrs. Joanne Wyatt
  Clanton Public Library
  100 First Street
  Clanton, Ala. 35045
  Barton Nuclear Plant
- Mrs. Peggy McCutchen
  Scottsboro Public Library
  1002 South Broad Street
  Scottsboro, Ala. 35768
  Bellefonte Nuclear Plant

## ARIZONA

- Mrs. Mary Carlson
  Phoenix Public Library
  Science and Industry Section
  12 East McDowell Road
  Phoenix, Arix. 85004
  Palo Verde Nuclear Plant

## ARKANSAS

- Mr. Vaughn
  Arkansas Polytechnic College
  Russellville, Ark. 72801
  Arkansas Nuclear One

## CALIFORNIA

- Mr. C. Combs
  Kern County Library
  1315 Truxtun Avenue
  Bakersfield, Calif. 93301
  San Joaquin Nuclear Plant
- Mrs. Alice Rosenberger
  Palo Verde Valley District Library
  125 West Chanslorway
  Blythe, Calif. 92255
- Mr. William B. Rohan
  San Diego County Law Library
  1105 Front Street
  San Diego, Calif. 92101
  Sundesert Nuclear Plant
- Mrs. Eileen Danforth
  Mission Viejo Branch Library
  24851 Chrisanta Drive
  Mission Viejo, Calif. 92676
  San Onofre Nuclear Plant
- Mr. Chi Su Kim
  Documents and Maps Department
  California Polytechnic State University Library
  San Luis Obispo, Calif. 93407
  Diablo Canyon Nuclear Plant
- Mrs. Judy Klapprott
  Humboldt County Library
  636 F Street
  Eureka, Calif. 95501
  Humboldt Bay Nuclear Plant
- Mrs. Dorothy Harvey
  Business & Municipal Department
  Sacramento City-County Library
  828 I Street
  Sacramento, Calif. 95814
  Rancho Seco Nuclear Plant
- Stanislaus County Free Library
  1500 I Street
  Modesto, Calif. 95345
  Stanislaus Nuclear Plant

## COLORADO

- Miss Ester Fromm
  Greeley Public Library
  City Complex Building
  Greeley, Colo. 80631
  Fort St. Vrain Nuclear Plant
- Mrs. Elizabeth Morrissett
  Acquisitions Department
  Auraria Library
  University of Colorado at Denver
  Lawrence and 11th
  Denver, Colo. 80204
  Atlas Corp. Uranium Mill

## CONNECTICUT

- Mrs. Judy Liskov
  Waterford Public Library
  Rope Ferry Road—Route 156
  Waterford, Conn. 06385
  Millstone Nuclear Plant
- Mr. Stewart Porter
  Russell Library
  119 Broad Street
  Middletown, Conn. 06457
  Haddam Neck Nuclear Plant

## DELAWARE

- Mrs. Yvonne Puffer
  Newark Free Library
  750 East Delaware Avenue
  Newark, Del. 19711
  Summit Nuclear Plant

## FLORIDA

- Ms. Sally Litton
  Jacksonville Public Library
  122 North Ocean Street
  Jacksonville, Fla. 32204
  Offshore Power Systems Manufacturing Facility
- Mrs. R. Scott
  Indian River Community College Library
  3209 Virginia Avenue
  Ft. Pierce, Fla. 33450
  St. Lucie Nuclear Plant
- Mrs. Rene' Daily
  Environmental and Urban Affairs Library

Florida International University
Miami, Fla. 33199
Turkey Point Nuclear Plant
- Mrs. Bonsall
Crystal River Public Library
668 N.W. First
Crystal River, Fla. 32639
Crystal River Nuclear Plant

## GEORGIA

- Mrs. J. W. Borom
Burke County Library
Fourth Street
Waynesboro, Ga. 30830
Vogtle Nuclear Plant
- Ms. Annette Osborne
Appling County Public Library
103 City Hall Drive
Baxley, Ga. 31513
Hatch Nuclear Plant

## ILLINOIS

- Mr. Ed Anderson
Illinois Valley Community College
Rural Route #1
Oglesby, Ill. 16348
LaSalle Nuclear Plant
- Mrs. Pam Wilson
Morris Public Library
604 Liberty Street
Morris, Ill. 60451
Dresden Nuclear Plant
Midwest Fuel Recovery Plant
- Mrs. Marie Hoschied
Moline Public Library
504 17th Street
Moline, Ill. 61255
Quad Cities Nuclear Plant
- Ms. Jo Ann Ellingson
Zion-Benton Public Library
2600 Emmaus Avenue
Zion, Ill. 60099
Zion Nuclear Plant
- Mrs. M. Evans
Vespasian Warner Public Library
120 West Johnson Street
Clinton, Ill. 61727
Clinton Nuclear Plant
- Mrs. Penny O'Roarke
Byron Public Library
Third and Washington Streets
Byron, Ill. 61010
Byron Nuclear Plant
- Mr. Thomas Carter
Wilmington Township Public Library
201 S. Kankakee Street
Wilmington, Ill. 60481
Braidwood Nuclear Plant
- Savanna Township Public Library
326 Third Street
Savanna, Ill. 61074
Carroll Nuclear Plant

## INDIANA

- West Chester Township Public Library
125 South Second Street
Chestertown, Ind. 46304
Bailly Nuclear Plant
- Ms. Carol Cowles
Madison-Jefferson County Public Library
420 West Main Street
Madison, Ind. 47250
Marble Hill Nuclear Plant

## IOWA

- Miss Kay Burke
Reference Service
Cedar Rapids Public Library
428 Third Avenue, S.E.
Cedar Rapids, Ia. 52401
Duane Arnold Nuclear Plant

## KANSAS

- Mr. Jack Scott
Coffey County Courthouse
Burlington, Kans. 66839
Wolf Creek Nuclear Plant

## KENTUCKY

- Mr. Clarence R. Graham
Louisville Free Public Library
4th and York Streets
Louisville, Ky. 40203
Marble Hill Nuclear Plant

## LOUISIANA

- Business & Science Division
New Orleans Public Library
219 Loyola Avenue
New Orleans, La. 70140
Offshore Power Systems
Manufacturing Facility
- Mr. Ken Owen
University of New Orleans Library
Louisiana Collection, Lakefront
New Orleans, La. 70122
Waterford Nuclear Plant
- Mrs. Freeda Fisher
Audubon Library
West Feliciana Branch
Ferdinand Street
St. Francisville, La. 70775
- Mr. Jimmie H. Hoover
Government Documents Department
Louisiana State University
Baton Rouge, La. 70803
River Bend Nuclear Plant

## MAINE

- Mrs. Barbara Shelton
Wiscasset Public Library
High Street
Wiscasset, Me. 04578
Maine Yankee Nuclear Plant

## MARYLAND

- Mrs. Elizabeth Hart
Charles County Library
Garrett and Charles Streets
La Plata, Md. 20646
Douglas Point Nuclear Plant
- Mrs. Marie Barrett
Calvert County Library
Prince Frederick, Md. 20678
Calvert Cliffs Nuclear Plant
- Ms. Pamela R. Schott
Harford Community College
401 Thomas Run Road
Bel Air, Md. 21014
Perryman Nuclear Plant

## MASSACHUSETTS

- Mrs. Margaret Howland
Greenfield Community College
One College Drive
Greenfield, Mass. 01301
Yankee Rowe Nuclear Plant
- Mr. Mark Titus
Plymouth Public Library
North Street
Plymouth, Mass. 02360
Pilgrim Nuclear Plant
- The Carnegie Library
Avenue A
Turner Falls, Mass. 01376
Montague Nuclear Plant

## MICHIGAN

- Mrs. Diana Shamp
Reference Department
Kalamazoo Public Library
315 South Rose Street
Kalamazoo, Mich. 49006
Palisades Nuclear Plant
- Mrs. Katherine Thomson
St. Clair County Library
210 McMorran Boulevard
Port Huron, Mich. 48060
Greenwood Nuclear Plant
- Mrs. M. B. Wallick
Charlevoix Public Library
107 Clinton Street
Charlevoix, Mich. 49720
Big Rock Point
- Mrs. Averill Packard
Grace Dow Memorial Library
1710 West St. Andrews Road
Midland, Mich. 48640
Midland Nuclear Plant
- Ms. Ann Stobbe
Maude Preston Palenske
Memorial Library

500 Market Street
St. Joseph, Mich. 49085
D.C. Cook Nuclear Plant

- Mrs. Marcia Learned
Reference Department
Monroe County Library System
3700 South Custer Road
Monroe, Mich. 48161
Fermi Nuclear Plant

## MINNESOTA

- Mrs. Copeland
Environmental Conservation Library
Minneapolis Public Library
300 Nicollet Mall
Minneapolis, Minn. 55401
Monticello Nuclear Plant
Prairie Island Nuclear Plant

## MISSOURI

- Mrs. Ladonna Justice
Fulton City Library
709 Market Street
Fulton, Mo. 65251
- Mrs. Ranata Rotkowicz
Olin Library of Washington University
Skinker & Lindell Boulevard
St. Louis, Mo. 63130
Callaway Nuclear Plant

## MISSISSIPPI

- Mrs. Stella Jennings
Clairborne County Chancery Clerk
Clairborne County Courthouse
Port Gibson, Miss. 39150
Grand Gulf Nuclear Plant
- Mr. William McMullin
Corinth Public Library
1023 Fillmore Street
Corinth, Miss. 38834
Yellow Creek Nuclear Plant

## NEBRASKA

- Mr. Frank Gibson
W. Dale Clark Library
215 South 15th Street
Omaha, Neb. 68102
Ft. Calhoun Nuclear Plant
- Mrs. Loy Mowery
Auburn Public Library
118 15th Street
Auburn, Neb. 68305
Cooper Nuclear Plant

## NEW HAMPSHIRE

- Miss Pamela Gjettum
Exeter Public Library
Front Street
Exeter, N.H. 03883
Seabrook Nuclear Plant

## NEW JERSEY

- Stockton State College Library
Pomona, N.J. 08240
Offshore Power Systems Manufacturing Facility
Atlantic Nuclear Plant
- Miss Elizabeth Fogg
Salem Free Public Library
112 West Broadway
Salem, N.J. 08097
Salem Nuclear Plant
Hope Creek Nuclear Plant
- Mrs. Gail Colure
Ocean County Library
Brick Township Branch
401 Chambers Bridge Road
Brick Town, N.J. 08723
Oyster Creek Nuclear Plant
Forked River Nuclear Plant

## NEW MEXICO

- Ms. Sandra Coleman
General Library, Reference Department
University of New Mexico
Albuquerque, N.M. 87131
- Ms. Ingrid Vollnhofer
New Mexico State Library
Box 1629
Santa Fe, N.M. 87503
Waste Isolation Pilot Plant

## NEW YORK

- Mr. Ralph W. Schmidt
Oswego County Office Building
46 East Bridge Street
Oswego, N.Y. 13126
Nine Mile Point Nuclear Plant
Sterling Nuclear Plant
FitzPatrick Nuclear Plant
- Mrs. June Rogoff
Rochester Public Library
Business & Social Science Division
115 South Avenue
Rochester, N.Y. 14604
Ginna Nuclear Plant
- Mr. Oliver Swift
White Plains Public Library
100 Martine Avenue
White Plains, N.Y. 10601
Indian Point Nuclear Plant
- Shoreham-Wading River Public Library
Route 25A
Shoreham, N.Y. 11786
Shoreham Nuclear Plant
- Mrs. E. Overton
Riverhead Free Library
330 Court Street
Riverhead, N.Y. 11901
Jamesport Nuclear Plant
- Mrs. Dorothy Augustine
Catskill Public Library
One Franklin Street
Catskill, N.Y. 12414
Greene County Nuclear Plant
- Mr. Stanley Zukowzki
Buffalo & Erie County Public Library
Lafayette Square
Buffalo, N.Y. 14203
- Ms. Marsha Russell
Town of Concord Public Library
23 North Buffalo Street
Springville, N.Y. 14141
NFS Fuel Reprocessing Plant and $UF_6$ Facility
- Mr. Sol Becker
Public Health Library
New York City Department of Health
125 Worth Street
New York, N.Y. 10013
Columbia University Research Reactor
- Mr. Harold Ettelt
Columbia-Greene Community College
P.O. Box 100
Hudson, N.Y. 12534
Greene County Nuclear Plant

## NORTH CAROLINA

- Mrs. Ruth Osborne
Public Library of Charlotte & Mecklenburg County
310 North Tryon Street
Charlotte, N.C. 28202
McGuire Nuclear Plant
- Mr. Roy Dicks
Wake County Public Library
104 Feyetteville Street
Raliegh, N.C. 27601
Shearon Harris Nuclear Plant
- Mr. David G. Ferguson
Davie County Public Library
416 North Main Street
P.O. Box 158
Mocksville, N.C. 27028
Perkins Nuclear Plant
- Southport-Brunswick County Library
109 West Moore Street
Southport, N.C. 28461
Brunswick Nuclear Plant
- Mrs. Charlotte Ellis
Franklin County Library
1026 Justice Street
Louisburg, N.C. 27549
Gulf Youngsville Fuel Fabrication Facility

## OHIO

- Mrs. Betty Waltman
  Perry Public Library
  3753 Main Street
  Perry, Ohio 44081
  Perry Nuclear Plant
- Clermont County Library
  Third and Broadway Streets
  Batavia, Ohio 45103
  Zimmer Nuclear Plant
- Mr. Donald Fought
  Ida Rupp Public Library
  310 Madison Street
  Port Clinton, Ohio 43452
  Davis-Besse Nuclear Plant
- Mrs. Esther Schedley
  Berlin Township Public Library
  Four East Main Street
  Berlin Heights, Ohio 44814
  Erie Nuclear Plant

## OKLAHOMA

- Mr. Craig Buthod
  Tulsa City-County Library
  400 Civic Center
  Tulsa, Okla. 74102
  Black Fox Nuclear Plant
- Mrs. O.J. Grosclaude
  Sallisaw City Library
  111 North Elm
  Sallisaw, Okla. 74955
  Sequoyah $UF_6$ Facility
- Ms. Hazel Nicholson
  Guthrie Public Library
  402 East Oklahoma Street
  Guthrie, Okla. 73044
  Cimarron Pu Fabrication Plant
  and Uranium Fuel Facility

## OREGON

- Mr. H. B. Allen
  City Hall, Records Office
  Arlington, Ore. 97812
  Pebble Springs Nuclear Plant
- Mr. Zimmer
  Columbia County Courthouse
  Law Library Circuit Court Room
  St. Helens, Ore. 97501
  Trojan Nuclear Plant

## PENNSYLVANIA

- Reference Department
  Osterhout Free Library
  71 South Franklin Street
  Wilkes-Barre, Pa. 18701
  Susquehanna Nuclear Plant
- Mrs. Margaret Atwood
  York College of Pennsylvania
  Country Club Road
  York, Pa. 17405
  Three Mile Island Nuclear Plant
- Mr. John Geschwindt
  Government Publications Section
  State Library of Pennsylvania
  Education Building
  Commonwealth and Walnut Street
  Harrisburg, Pa. 17126
  Peach Bottom Nuclear Plant
  Three Mile Island Nuclear Plant
  Fulton Nuclear Plant
- Mrs. Gordon Bauerle
  Pottstown Public Library
  500 High Street
  Pottstown, Pa. 19464
  Limerick Nuclear Plant
- Apollo Memorial Library
  219 North Pennsylvania Avenue
  Apollo, Pa. 15613
  Apollo $UF_6$ and Pu Facilities
- Mr. Anthony Martin
  Carnegie Library of Pittsburgh
  4400 Forbes Avenue
  Pittsburgh, Pa. 15213
  Cheswick Fuel Development
  Laboratories
- Mr. F.E. Virostek
  B.F. Jones Memorial Library
  663 Franklin Avenue
  Aliquippa, Pa. 15001
  Beaver Valley Nuclear Plant
  Shippingport Light Water Breeder
  Reactor

## PUERTO RICO

- Mrs. Rosario Cabrera
  Public Library, City Hall
  Jose de Diego Avenue
  P.O. Box 1086
  Arecibo, P.R. 00612
- Mrs. Amalia Ruiz De Porras
  Etien Totti Public Library
  College of Engineers, Architects
  & Surveyors
  Urb Roosevelt Development
  Hato Rey, P.R. 00918
  North Coast Nuclear Plant

## RHODE ISLAND

- Mrs. Ann Crawford
  Cross Mill Public Library
  Old Post Road
  Charlestown, R.I. 02831
- Mr. Thomas Reynolds
  University of Rhode Island
  University Library
  Government Publications Office
  Kingston, R.I. 02881
  New England Nuclear Plant

## SOUTH CAROLINA

- Joe E. Garcia
  York County Library
  325 South Oakland Avenue
  Rock Hill, S.C. 29730
  Catawba Nuclear Plant
- Reference Department
  Richland County Public Library
  1400 Sumter Street
  Columbia, S.C. 29201
  Summer Nuclear Plant
- Miss Louise Marcum
  Oconee County Library
  501 W. Southbroad
  Walhalla, S.C. 29691
  Oconee Nuclear Plant
- Mrs. Allene Reep
  Hartsville Memorial Library
  Home and Fifth Avenues
  Hartsville, S.C. 29550
  H.B. Robinson Nuclear Plant
- Mr. David Eden
  Cherokee County Library
  300 East Rutledge Avenue
  Gaffney, S.C. 29340
  Cherokee Nuclear Plant
- Mr. Fred Bodiford
  County Office Builidng
  Room 105
  P.O. Box 443
  Barnwell, S.C. 29812
  Barnwell Fuel Plant
  $UF_6$ Facility
  Barnwell Fuel Storage Station
- Mr. Carl Stone
  Anderson County Library
  202 East Greenville Street
  Anderson, S.C. 29621
  Recycle Fuel Plant
- Mrs. Ellen Jenkins
  Barnwell County Library
  Hagood Avenue
  Barnwell, S.C. 29812
  Chem-Nuclear Plant

## TENNESSEE

- Miss Kendall J. Cram
  Tennessee State Library and Archives
  403 Seventh Avenue, North
  Nashville, Tenn. 37219
  Hartsville Nuclear Plant
- Ms. Dorothy Dismuke
  Oak Ridge Public Library
  Civic Center
  Oak Ridge, Tenn. 37830
- Mrs. Patricia Rugg
  Lawson McGhee Public Library
  500 West Church Street
  Knoxville, Tenn. 37902
  Clinch River Breeder Plant
  Exxon Nuclear Fuel Recovery
  Center
  Fuel Fabrication Facility
- Mr. Wally Keasler
  Chattanooga-Hamilton County
  Bicentennial Library

1001 Broad Street
Chattanooga, Tenn. 37402
Sequoyah Nuclear Plant
Watts Bar Nuclear Plant

- Mr. T. Cal Hendrix
Kingsport Public Library
Broad and New Streets
Kingsport, Tenn. 37660
Phipps Bend Nuclear Plant
- Mr. H.E. Zittel
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, Tenn. 37830
Tyrone Nuclear Plant

## TEXAS

- Mrs. Tim Whitworth
Somervell County Public Library
On The Square
P.O. Box 1417
Glen Rose, Tex. 76043
Comanche Peak Nuclear Plant
- Newton County Library
P.O. Box 657
Newton, Tex. 77034
Blue Hills Nuclear Plant
- Matagorda County Courthouse
Matagorda County Law Library
P.O. Box 487
Bay City, Tex. 77414
South Texas Nuclear Plant
- Mrs. Kroesche
Sealy Public Library
201 Atchison Street
Sealy, Tex. 77474
Allens Creek Nuclear Plant

## VERMONT

- Mrs. June Bryant
Brooks Memorial Library
224 Main Street
Brattleboro, Vt. 05301
Vermont Yankee Nuclear Plant

## VIRGINIA

- Ms. Sandra Peterson
Swem Library
College of William & Mary
Williamsburg, Va. 23185
Surry Nuclear Plant
- Ms. Manrique
Board of Supervisors
Louisa County Courthouse
P.O. Box 27
Louisa, Va. 23093
- Mr. Gregory Johnson
Alderman Library
Manuscripts Department
University of Virginia
Charlottesville, Va. 22901
North Anna Nuclear Plant

## WASHINGTON

- Ms. D. E. Roberts
Richland Public Library
Swift and Northgate Streets
Richland, Wash. 99352
WPPSS 1, 2 and 4 Nuclear Plants
Exxon Fuel Plant
- Mrs. D. Stendal
Sedro Wooley Library
802 Ball Avenue
Sedro Wooley, Wash. 98294
Skagit Nuclear Plant
- Ms. Selma Nielsen
W. H. Abel Memorial Library
125 Main Street South
Montesano, Wash. 98563
WPPSS 3 and 5 Nuclear Plants

## WISCONSIN

- Mrs. Jane Radloff
LaCrosse Public Library
800 Main Street
LaCrosse, Wis. 54601
LaCrosse BWR Nuclear Plant
- Mr. Arthur M. Fish
Document Department, Library
University of Wisconsin
Stevens Point
Stevens Point, Wis. 54481
Point Beach Nuclear Plant
Wood Nuclear Plant
- Mead Public Library
710 North Eighth Street
Sheboygan, Wis. 53081
- Madison Public Library
Business and Science Division
201 West Mifflen Street
Madison, Wis. 53703
Haven Nuclear Plant
- Ms. Sue Grossheuch
Kewaunee Public Library
822 Juneau Street
Kewaunee, Wis. 54216
Kewaunee Nuclear Plant
- Mr. John Jax
University of Wisconsin
Stout Library
Menomonie, Wis. 54751
- Mr. Robert Fetvedt
University Library
University of Wisconsin—Eau Claire
Park and Garfield Avenues
Eau Claire, Wis. 54710
- Mrs. Robert Goodrich
Durand Free Library
315 Second Avenue, West
Durand, Wis. 54736
Tyrone Nuclear Plant

## WYOMING

- Mrs. Carroll Highfill
Converse County Library
Douglas, Wyo. 82633
Highland Uranium Mill
- Mrs. Margaret Baker
Carbon County Public Library
Courthouse
Rawlins, Wyo. 82301
Shirley Basin Uranium Mill

# Appendix 4

# Regulations and Amendments—Fiscal Year 1979

The regulations of the Nuclear Regulatory Commission are contained in Title 10, Chapter 1, of the Code of Federal Regulations. Effective and proposed regulations concerning licensed activities, and certain policy statements relating thereto, which were published in the *Federal Register* during fiscal year 1979, are described briefly below.

## REGULATIONS AND AMENDMENTS PUT INTO EFFECT

### Distribution of Applications and Environmental Statements to Local Officials—Parts 2 and 51

On October 6, 1978, amendments to Parts 2 and 51 were published, effective November 6, 1978, to provide for notice to the chief executives of the appropriate alternative municipalities or counties which have been identified in the application or environmental report as alternative sites for nuclear facilities or activities.

### Amendments Regarding Basic Component—Part 21

On October 19, 1978, amendments to Part 21 were published, effective immediately, to limit the types of items that are within the scope of the NRC rule for reporting defects and noncompliance.

### Revocation of Certain Reporting Requirements—Part 50

On October 25, 1978, amendments to Part 50 were published, effective immediately, to revoke two reporting requirements on antitrust information and to revoke a requirement for retention of 25 copies of the antitrust report during the antitrust review.

### Standards for Combustible Gas Control Systems—Part 50

On October 27, 1978, amendments to Part 50 were published, effective November 27, 1978, to clarify the NRC's position on its general design criterion regarding the containment design basis and to provide a new section specifying the standards for combustible gas control systems.

### Miscellaneous Amendments—Parts 20, 21, 40, and 73

On November 9, 1978, amendments to Parts 20, 21, 40, and 73, were published, effective immediately, to change the telephone number for the NRC's Inspection and Enforcement Regional Office I and to exempt the general licensee under §40.25 from the requirements of the NRC's regulation "Notices, Instructions, and Reports to Workers; Inspection."

### Radiation Surveys of Therapy Patients—Part 35

On November 28, 1978, an amendment to Part 35 was published, effective December 28, 1978, to require licensees authorized to treat patients with temporary implants incorporating radioactive material to confirm the removal of the implants at the end of the treatment by (1) a source count and (2) a radiation survey of the patient.

### Codes and Standards for Nuclear Power Plants—Part 50

On November 30, 1978, amendments to Part 50 were published, effective immediately, to clarify certain ambiguities to avoid misinterpretations of provisions which deal with requirements for in service inspection of nuclear power plants.

### Calibration of Teletherapy Units—Part 35

On January 8, 1979, amendments to Part 35 were published, effective July 9, 1979, to require medical licensees to (1) calibrate each teletherapy unit annually and (2) perform monthly spot checks on those calibrations.

### Miscellaneous Amendments—Parts 20, 21, and 73

On January 12, 1979, amendments to Parts 20, 21, and 73 were published, effective immediately, which change the telephone number for the Commission's Inspection and Enforcement Regional Office V.

### Clarification of Computation of Time Provisions—Part 2

On January 22, 1979, amendments to Part 2 were published, effective immediately, to clarify the requirements for computing the time prescribed for replying to a petition for leave to intervene.

### Change in License Conditions for Certain Medical Licenses—Part 35

On February 20, 1979, an amendment to Part 35 was published, effective March 22, 1979, (a) to permit physicians greater latitude, when they use certain low risk diagnostic radiopharmaceuticals, by no longer designating authorized clinical procedures and (b) by deleting from several licensing groups certain chemical forms not approved by FDA.

**Requirements for the Physical Protection of Nuclear Power Plants—Part 73**

On February 28, 1979, an amendment to Part 73 was published, effective immediately, to change from February 23, 1979 to August 1, 1979, the implementation date when pat-down searches of regular employees at nuclear power plants, the two man rule procedures, and compartmentalization have to be implemented for protection against insider sabotage.

**Waiver or Reduction of Fees for Searching and Reproduction of Records—Part 9**

On March 16, 1979, an amendment to Part 9 was published, effective April 16, 1979 to add a new section "Waiver or Reduction of Fees."

**Rules of General Applicability to Domestic Licensing of Byproduct Material, Domestic Licensing of Source Material and Domestic Licensing of Special Nuclear Material—Parts 30, 40 and 70**

On March 22, 1979, an amendment to Parts 30, 40, and 70 was published, effective June 5, 1979, to require licensees to notify the Commission when they decide to permanently discontinue all activities involving materials authorized under a license. This will allow NRC to terminate the license in an orderly and timely manner.

**Financial Protection Requirements and Indemnity Agreements; Miscellaneous Amendments—Part 140**

On April 6, 1979, amendments to Part 140 were published, effective May 1, 1979, to increase the level of the primary layer of financial protection required of certain indemnified licensees.

**General License Requirements For Any Person Who Possesses Formula Quantitites of Strategic Special Nuclear Material (SSNM) In Transit Subject to Certain Requirements—Part 170**

On May 8, 1979, amendments to Part 70 were published, effective June 7, 1979, to remove the exemptions to licensing for carriers and other persons who possess or control formula quantities of strategic special nuclear material for the purpose of transport, or storage incident of transport.

**Control of Radiation Exposure to Transient Workers—Parts 19 and 20**

On June 6, 1979, amendments to Parts 19 and 20 were published, effective August 20, 1979, which would require NRC licensees to control the total occupational radiation dose of individuals who work in NRC-licensed activities.

**Physical Protection of Irradiated Reactor Fuel in Transit—Part 73**

On June 15, 1979, amendments to Part 73 were published, effective July 16, 1979, which would establish requirements for protection of spent fuel in transit.

**Conduct of Employees Ownership of Stocks, Bonds, and Other Security Interests by NRC Employees—Part 0**

On July 17, 1979, amendments to Part 0 were published, effective immediately, in which the NRC revised its regulations governing the ownership of stocks, bonds, and other security interests.

**Safeguards Requirements for Special Nuclear Material of Moderate and Low Strategic Significance—Parts 70, 73, and 150**

On July 24, 1979, amendments to Parts 70, 73, and 150 were published, effective November 21, 1979, to require physical protection measures to detect theft of special nuclear material of moderate and low strategic significance.

**Rules of Practice; Selected Nuclear Power Plant Construction Permit Applications—Part 2**

On August 15, 1979, amendments to Part 2 were published, effective immediately, to permit the use of new staff procedures on a trial basis for the systems and site safety portions of selected nuclear power plant construction permit applications.

**Requirements for the Physical Protection of Nuclear Power Plants—Part 73**

On August 15, 1979, amendment to Part 73 was published, effective immediately, to change the date of the pat-down searches of regular employees at nuclear power plants from August 1, 1979 to November 1, 1979.

**Licensing of Production and Utilization Facilities; and Access for Resident Inspection—Parts 50 and 70**

On August 16, 1979, amendments to Parts 50 and 70 were published, effective September 17, 1979, to require power reactor licensees, construction permit holders, and selected fuel facility licensees to provide (1) on site, rent-free, exclusive use of office space and (2) immediate licensee facility access to Commission inspection personnel.

**Uranium Mill Tailings Licensing—Parts 40 and 150**

On August 24, 1979, amendments to Part 40 and 150 were published, effective immediately, to conform to the requirements of the Uranium Mill Tailings Radiation Control Act of 1978 and to the standards set forth in the draft Generic Environmental Impact Statement on Uranium Milling.

**Addition of Veterinarians to the In Vitro General License—Parts 31 and 32**

On August 28, 1979, amendments to Parts 31 and 32 were published, effective September 27, 1979, to add veterinarians to the groups already authorized to use byproduct material under general license for clinical or laboratory testing done outside the body.

**Licenses for Radiography and Radiography Safety Requirements for Radiographic Operations; Amendments of Radiography Regulations—Part 34**

On August 30, 1979, amendments to Part 34 were published, effective March 3, 1980, to require several changes intended to improve radiography safety and to formalize as regulations current licensing practices.

**Amendments to Appendix A—Requests for Declassification Review—Part 9**

On August 30, 1979 amendments to Part 9 were published effective immediately, to substitute references to E.O. 12065 and its implementing directive (Information Security Oversight office Directive No. 1) for the replaced E.O. 11652 and its implementing National Security Directive.

## REGULATIONS AND AMENDMENTS PROPOSED

**Storage of Spent Fuel in an Independent Spent Fuel Storage Installation—Part 72**

On October 6, 1978, proposed amendments to Part 72 were published for comment to add a new part which specifies procedures and requirements for issuance of licenses to store spent fuel in an independent spent fuel storage installation.

**Management and Disposal of Low-Level Wastes by Shallow Land Burial and Alternative Disposal Methods—Part 61**

On October 25, 1978, an advance notice of proposed rulemaking was published for comment to add a new Part 61 which specifies a regulatory program for management of low-level radioactive wastes.

**Licensing Procedures for Geologic Repositories for High-Level Radioactive Wastes—Proposed General Statement of Policy**

On November 17, 1978, a proposed general statement of policy was published for comment regarding establishment of procedures for licensing geologic high-level waste repositories to be constructed and operated by the U.S. Department of Energy.

**Burial of Small Quantities of Radionuclides—Part 20**

On December 4, 1978, proposed amendments to Part 20 were published for comment to require NRC licensees to obtain Commission approval prior to burial of small quantities of radionuclides.

**Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Plants—Part 50**

On December 6, 1978, an advance notice of proposed amendments to Part 50 was published for comment to change certain technical as well as nontechnical requirements within the existing emergency core cooling system rule.

**Codes and Standards for Nuclear Power Plants—Part 50**

On December 18, 1978, proposed amendments to Part 50 were published for comment to incorporate by reference new addenda of a national code that provides rules for the construction of nuclear power plant components.

**Indemnification of Spent Reactor Fuel Stored at a Reactor Site Different Than the One Where It was Generated—Part 140**

On January 8, 1979, the Commission published for comment a notice stating its decision to exercise its discretionary statutory authority under the Price-Anderson Act and extend Government indemnity to spent reactor fuel stored at a reactor site different than the one where it was generated in two specific situations. The Commission invited public comment on this decision and on the general policy question raised by such an extension.

**Codes and Standards for Nuclear Power Plants—Part 50**

On January 18, 1978, Proposed amendments to Part 50 were published for comment to incorporate by reference, with modifications, a new edition and addenda of the national code that specifies the requirements for the in-service inspection of nuclear power plant components.

**Notices, Instruction, and Reports to Workers: Inspection Standards for Protection Against Radiation—Parts 19 and 20**

On February 20, 1979, proposed amendments to Parts 19 and 20 were published for comment which would eliminate the accumulated dose averaging formula, 5(N-18), and the associated Form NRC-4 Exposure History, and impose annual dose-limiting standards while retaining quarterly standards.

**Rules of Practice—Part 2**

On March 7, 1979, a proposed amendment was published for comment to amend the rules dealing with ex-parte communications and the separation of adjudicatory and non-adjudicatory functions so that those rules will be in accord with Government in the Sunshine Act, Pub. L. 94-409.

**Domestic Licensing of Production and Utilization Facilities—Part 50**

On March 28, 1979, a proposed amendment to Part 50 was published for comment specifying fracture toughness and material surveillance program requirements for nuclear reactor to permit greater flexibility in meeting certain of these requirements, and to simplify others by substituting references to National Standards that have already been incorporated by reference into the NRC's Regulations.

**Financial Protection Requirements and Indemnity Agreements: Miscellaneous Amendments—Part 140**

On April 6, 1979, a proposed amendment to Part 140 was published for comment to amend regulations relating to the financial protection and indemnity required of licensees and to implement legislation that modified and extended for ten years (to August 1, 1987) the present Price-Anderson legislation.

**Human Uses of Byproduct Material; Changes in License Conditions for Certain Medical Licenses—Part 35**

On April 9, 1979, a proposed amendment to Part 35 was published for comment making it a requirement to appoint a radiation safety committee that will focus on radiation safety.

**Public Records—Part 9**

On April 17, 1979, an advance notice of proposed rule making was published for comment which would revise the Commission's regulations, "Public Records," implementing the Freedom of Information Act and E.O. 12044 which provides that regulations be written in "plain English.'.

**Rules of Practice—Part 2**

On April 18, 1979, proposed amendments to Part 35 were published for comment to make procedural changes to permit the use of new staff procedures on a trial basis for the system and site safety portions of selected nuclear power plants construction permit applications.

**Nondiscrimination in Federally Assisted Commission Programs; Application to the Handicapped—Part 4**

On May 8, 1979, a proposed amendment was published for comment to implement the requirements of Section 504 of the Rehabilitation Act of 1973. The amendment would make it unlawful for any recipient of Federal financial assistance to discriminate against a qualified handicapped person, on the basis of handicap, in employment or the receipt of services.

**Study of Nuclear Power Plant Construction During Adjudication; Request for Public Comments—Part 2**

On June 13, 1979, a notice of proposed rule making was published for comments seeking the views of the public on the Commission's immediate effectiveness rule (10 CFR 2.764) which provides that a construction permit can be issued on the basis of an initial decision of an Atomic Safety and Licensing Board even though that decision is subject to further review within the Commission.

**Testing of Radioisotope Generators—Parts 30 and 35**

On June 6, 1979, a proposed amendment was published for comment to require licensees to test radiopharmaceuticals for a contaminant called molybdenum-99.

**Access to and Protection of National Security Information and Restricted Data—Parts 25 and 95**

On July 2, 1979, a proposed amendment was published for comment governing access to and protection of National Security Information and Restrictive Data. When former Atomic Energy Commission regulations were reissued in March 1975 by the Nuclear Regulatory Commission, rules governing access to and protection of the National Security Information and Restrictive Data were not included.

**Safeguards on Nuclear Material; Implementation of US/IAEA Agreement—Parts 40, 50, 70, 75, 150 and 170**

On May 1979, proposed amendments were published for comment to enable the United States to Implement the US/IAEA Safeguards Agreement, with respect to licensed activities, as soon as the Agreement enters into force.

**Adequacy and Acceptance of Emergency Planning Around Nuclear Facilities—Part 50**

On July 17, 1979, an advance notice of proposed rulemaking was published for comment to establish as conditions of power reactor operation increased emergency readiness for public protection in the vicinity of nuclear power reactors on the part of both the licensee and local and state authorities.

**Privacy Act Regulations; Proposed Exemptions—Part 9**

On August 16, 1979, a proposed amendment was published for comment to exempt from certain requirements of the Privacy Act portions of a proposed new system of records "Special Inquiry File."

**Packaging of Radioactive Material for Transportation and Transportation of Radioactive Material Under Certain Conditions; Comparability With IAEA Regulations—Part 71**

On August 17, 1979, a proposed amendment was published for comment revising the regulations for the transportation of radioactive material to make them compatible with those of the IAEA and thus with those of most major nuclear nations of the world.

**Criteria Relating to Uranium Mill Tailings and Construction of Major Plants—Parts 30, 40, 70, 150 and 170**

On August 24, 1979, proposed amendments were published for comment to conform to the requirements of the Uranium Mill Tailings Radiation Control Act of 1978 and to the standards set forth in the draft generic environmental impact statement on uranium milling.

**Privacy Act Regulations; Notices of Proposed Exemptions—Part 9**

On August 28, 1979, an amendment was published for comment which proposed exemptions as provided under the Privacy Act of 1974. The proposed amendment of the Commission's regulations "Public Records" would exempt from certain requirements of the Privacy Act portions of a proposed new system of records "Document Control System."

**Production and Utilization Facility Licensees; Emergency Planning—Part 50 and 70**

On September 19, 1979, proposed amendments were published for comment which would require that all production and utilization facility licensees shall, as a condition of their license, submit emergency plans for NRC review and approval and maintain the emergency plans up to date. The proposed amendments would also require certain special nuclear material facility licensees (for processing and fuel fabrication, scrap recovery or conversion of uranium hexafluoride) to maintain the emergency plans up to date.

# Appendix 5

# Regulatory Guides - Fiscal Year 1979

Regulatory guides describe and make available to the public methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations and, in some cases, describe techniques used by the staff in evaluating specific problems or postulated accidents. Guides also may provide guidance to applicants concerning information needed by the staff in its review of applications for permits and licenses.

Comments and suggestions for improvements in guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. In its continuing effort to provide for increased public participation in the regulatory process, the NRC instituted a new procedure for developing and issuing guides during this fiscal year. Guides are now being issued for public comment in draft form before the guides have received complete staff review and before an official NRC staff position has been established.

Regulatory guides may also be withdrawn when they are superseded by the Commission's regulations, when equivalent recommendations have been incorporated in applicable approved codes and standards, or when changes in methods and techniques have made them obsolete.

When guides are issued, revised, or withdrawn, notices are placed in the *Federal Register* and public announcements made.

In order to reduce the burden on the taxpayer, the NRC has made arrangements with the U.S. Government Printing Office to become a consigned sales agent for certain NRC publications, Effective November 1, 1979, regulatory guides are being included in this sales program. Draft guides, which are issued for public comment, will continue to receive free distribution. Active guides will be sold on a subscription or individual copy basis. Licensees of the NRC will receive, at no cost, pertinent draft and active regulatory guides as they are issued.

The following guides were issued or revised (or withdrawn as noted) during the period October 1, 1978 to September 30, 1979.

**Division 1—Power Reactor Guides**

1.7 Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Revision 2)

1.9 Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (Revision 1)

1.28 Quality Assurance Program Requirements (Design and Construction) (Revision 2)

1.70 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants—LWR Edition (Revision 3)

1.72 Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin (Revision 2)

1.84 Design and Fabrication Code Case Acceptability-ASME Section III Division 1 (Revisions 14 and 15)

1.85 Materials Code Case Acceptability-ASME Section III Division 1 (Revisions 14 and 15)

1.104 WITHDRAWN. Overhead Crane Handling Systems for Nuclear Power Plants

1.125 Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Revision 1)

1.128 Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Revision 1)

1.130 Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Component Supports (Revision 1)

1.132 Site Investigations for Foundations for Nuclear Power Plants (Revision 1)

1.134 Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses (Revision 1)

1.136 Material for Concrete Containment (Revision 1)

1.144 Auditing of Quality Assurance Programs for Nuclear Power Plants

1.145 Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

**Division 2—Research and Test Reactor Guides**

2.6 Emergency Planning for Research Reactors

### Division 3—Fuels and Materials Facilities Guides

3.11.1 Operational Inspection and Surveillance of Embankment Retention Systems for Uranium Mill Tailings

3.34 Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Uranium Fuel Fabrication Plant (Revision 1)

3.35 Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Plutonium Processing and Fuel Fabrication Plant (Revision 1)

3.36 WITHDRAWN. Nondestructive Examination of Tubular Products for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants

3.42 Emergency Planning for Fuel Cycle Facilities and Plants Licensed Under 10 CFR Parts 50 and 70 (Revision 1)

3.43 Nuclear Criticality Safety in the Storage of Fissile Materials (Revision 1)

3.44 Standard Format and Content for the Safety Analysis Report to be Included in a License Application for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (Water-Basin Type)

### Division 4—Environmental and Siting Guides

4.15 Quality Assurance for Radiological Monitoring Programs (Normal Operations)—Effluent Streams and the Environment (Revision 1)

### Division 5—Materials and Plant Protection Guides

5.2 WITHDRAWN. Classification of Unirradiated Plutonium and Uranium Scrap

5.58 Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements

### Division 6—Product Guides

6.8 Identification Plaque for Irretrievable Well-Logging Sources

### Division 7—Transportation Guides

7.9 Standard Format and Content of Part 71 Applications for Approval of Packaging of Type B, Large Quantity, and Fissile Radioactive Material

### Division 8—Occupational Health Guides

8.19 Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants-Design Stage Man-Rem Estimates (Revision 1)

8.20 Applications of Bioassay for I-125 and I-131 (Revision 1)

8.23 Radiation Safety Surveys at Medical Institutions

8.24 Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication

### Division 9—Antitrust and Financial Review Guides

None

### Division 10—General Guides

10.7 Guide for the Preparation of Applications for Licenses for Laboratory and Industrial Use of Small Quantities of Byproduct Material (Revision 1)

10.8 Guide for the Preparation of Applications for Medical Programs

### Draft Guides

EM 805-5 Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants

FP 811-4 Safety-Related Permanent Dewatering Systems for Nuclear Power Plants

MP 711-4 Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance

OH 706-4 Guide for Preparation of Applications for the Use of Gamma Irradiators

OH 714-4 Applications of Bioassay for Fission and Activation Products

OH 717-4 Radiation Protection Training for Light-Water-Cooled Nuclear Power Plant Personnel

OH 804-4 Audible-Alarm Dosimeters

RH 802-4 Calculational Models for Estimating Radiation Doses to Man from Airborne Radioactive Materials Resulting from Uranium Milling Operations

RS 705-4 Lightning Protection for Nuclear Power Plants

RS 809-5 Qualification Test for Cable Penetration Fire Stops for Use in Nuclear Power Plants

RS 810-5 Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

SC 521-4 LWR Core Reloads: Guidance on Applications for Amendments to Operating Licenses and on Refueling and Startup Tests

SC 704-5 Functional Specification for Safety-Related Valve Assemblies in Nuclear Power Plants

SC 705-4 Ultrasonic Testing of Reactor Vessel Welds During Inservice Examination

SC 721-4 Inservice Inspection Code Case Acceptability-ASME Section XI Division 1

1.8 Proposed Revision 2—Personnel Selection and Training

1.33 Proposed Revision 3—Quality Assurance Program Requirements (Operation)

1.35 Proposed Revision 3—Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments

1.35.1 Determining Prestressing Forces for Inspection of Prestressed Concrete Containments

1.58 Proposed Revision 1—Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel

1.94 Proposed Revision 2—Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils and Foundations During the Construction Phase of Nuclear Power Plants

1.131 Proposed Revision 1—Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants

5.7 Proposed Revision 1—Entry/Exit Control to Protected Areas, Vital Areas, and Material Access Areas

5.14 Proposed Revision 1—Use of Observation (Visual Surveillance) Techniques in Material Access Areas

5.44 Proposed Revision 2—Perimeter Intrusion Alarm Systems

5.57 Proposed Revision 1—Shipping and Receiving Control of Strategic Special Nuclear Material

8.8 Proposed Revision 4—Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable (ALARA)

# Appendix 6

# Nuclear Electric Generating Units in Operation, Under Construction or Planned

(As of September 30, 1979)

The following listing includes 192 nuclear power reactor electrical generating units which were in operation, under NRC review for construction permits, and ordered or announced by utilities in the United States at the end of September 1979, representing a total capacity of approximately 187,000 MWe. TYPE is indicated by: BWR—boiling water reactor, PWR—pressurized water reactor, HTGR—high temperature gas-cooled reactor, and LMFBR—liquid metal cooled fast breeder reactor. STATUS is indicated by: OL—has operating license, CP—has construction permit, UR—under review for construction permit, A/O—announced or ordered by the utility but application for construction not yet docketed by the NRC for review. The dates for operation are either actual or those scheduled by the utilities (N/S—not yet scheduled).

This listing includes 20 fewer units than a year ago, reflecting cancellations of plans for future facilities. In addition, delays in planned completion dates have been indicated during fiscal year 1979 for 47 other units. The reasons cited for delays and cancellations include (1) lower demand for electricity, (2) financial problems, (3) construction delays, (4) concerns for reactor safety, and (5) regulatory delays.

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **ALABAMA** | | | | | | |
| Decatur | Browns Ferry Nuclear Power Plant Unit 1 | 1,065 | BWR | OL 1973 | Tennessee Valley Authority | 1974 |
| Decatur | Browns Ferry Nuclear Power Plant Unit 2 | 1,065 | BWR | OL 1974 | Tennessee Valley Authority | 1975 |
| Decatur | Browns Ferry Nuclear Power Plant Unit 3 | 1,065 | BWR | OL 1976 | Tennessee Valley Authority | 1977 |
| Dothan | Joseph M. Farley Nuclear Plant Unit 1 | 829 | BWR | OL 1977 | Alabama Power Co. | 1978 |
| Dothan | Joseph M. Farley Nuclear Plant Unit 2 | 829 | PWR | CP 1972 | Alabama Power Co. | 1980 |
| Scottsboro | Bellefonte Nuclear Plant Unit 1 | 1,235 | PWR | CP 1974 | Tennessee Valley Authority | 1981 |
| Scottsboro | Bellefonte Nuclear Plant Unit 2 | 1,235 | PWR | CP 1974 | Tennessee Valley Authority | 1981 |

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **ARIZONA** | | | | | | |
| Winterburg | Palo Verde Nuclear Generating Station Unit 1 | 1,270 | PWR | CP 1976 | Arizona Public Service Co. | 1982 |
| Winterburg | Palo Verde Nuclear Generating Station Unit 2 | 1,270 | PWR | CP 1976 | Arizona Public Service Co. | 1984 |
| Winterburg | Palo Verde Nuclear Generating Station Unit 3 | 1,270 | PWR | CP 1976 | Arizona Public Service Co. | 1986 |
| **ARKANSAS** | | | | | | |
| Russelville | Arkansas Nuclear One Unit 1 | 850 | PWR | OL 1974 | Arkansas Power & Light Co. | 1974 |
| Russelville | Arkansas Nuclear One Unit 2 | 912 | PWR | OL 1978 | Arkansas Power & Light Co. | 1978 |
| **CALIFORNIA** | | | | | | |
| Eureka | Humboldt Bay Power Plant Unit 3 | 65 | BWR | OL 1962 | Pacific Gas & Electric Co. | 1963 |
| San Clemente | San Onofre Nuclear Generating Station Unit 1 | 436 | PWR | OL 1967 | So. Calif. Ed. & San Diego Gas & Electric Co. | 1968 |
| San Clemente | San Onofre Nuclear Generating Station Unit 2 | 1,140 | PWR | CP 1973 | So. Calif. Ed. & San Diego Gas & Electirc Co. | 1980 |
| San Clemente | San Onofre Nuclear Generating Station Unit 3 | 1,140 | PWR | CP 1973 | So. Calif. Ed. & San Diego Gas & Electric Co. | 1981 |
| Diablo Canyon | Diablo Canyon Nuclear Power Plant Unit 1 | 1,084 | PWR | CP 1968 | Pacific Gas & Electric Co. | 1980 |
| Diablo Canyon | Diablo Canyon Nuclear Power Plant Unit 2 | 1,106 | PWR | CP 1970 | Pacific Gas & Elec. Co. | 1979 |
| Clay Station | Rancho Seco Nuclear Generating Station Unit 1 | 917 | PWR | OL 1974 | Sacramento Municipal Utility District | 1975 |
| * | Stanislaus Unit 1 | 1,200 | BWR | A/O | Pacific Gas & Elec. Co. | Indef. |
| * | Stanislaus Unit 2 | 1,200 | BWR | A/O | Pacific Gas & Elec. Co. | Indef. |
| **COLORADO** | | | | | | |
| Platteville | Fort St. Vrain Nuclear Generating Station | 330 | HTGR | OL 1973 | Public Service Co. of Colorado | 1978 |
| **CONNECTICUT** | | | | | | |
| Haddam Neck | Haddam Neck Generating Station | 575 | PWR | OL 1967 | Conn. Yankee Atomic Power Co. | 1968 |
| Waterford | Millstone Nuclear Power Station Unit 1 | 660 | BWR | OL 1970 | Northeast Nuclear Energy Co. | 1971 |

* Site not selected.

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **CONNECTICUT—Continued** | | | | | | |
| Waterford | Millstone Nuclear Power Station Unit 2 | 830 | PWR | OL 1975 | Northeast Nuclear Energy Co. | 1975 |
| Waterford | Millstone Nuclear Power Station Unit 3 | 1,159 | PWR | CP 1974 | Northeast Nuclear Energy Co. | 1985 |
| **FLORIDA** | | | | | | |
| Florida City | Turkey Point Station Unit 3 | 693 | PWR | OL 1972 | Florida Power & Light Co. | 1972 |
| Florida City | Turkey Point Station Unit 4 | 693 | PWR | OL 1973 | Florida Power & Light Co. | 1973 |
| Red Level | Crystal River Plant Unit 3 | 825 | PWR | OL 1977 | Florida Power Corp. | 1977 |
| Ft. Pierce | St. Lucie Plant Unit 1 | 802 | PWR | OL 1976 | Florida Power & Light Co. | 1976 |
| Ft. Pierce | St. Lucie Plant Unit 2 | 842 | PWR | CP 1977 | Florida Power & Light Co. | 1983 |
| **GEORGIA** | | | | | | |
| Baxley | Edwin I. Hatch Plant Unit 1 | 786 | BWR | OL 1974 | Georgia Power Co. | 1975 |
| Baxley | Edwin I. Hatch Plant Unit 2 | 795 | BWR | OL 1978 | Georgia Power Co. | 1978 |
| Waynesboro | Alvin W. Vogtle, Jr. Plant Unit 1 | 1,100 | PWR | CP 1974 | Georgia Power Co. | 1984 |
| Waynesboro | Alvin W. Vogtle, Jr. Plant Unit 2 | 1,100 | PWR | CP 1974 | Georgia Power Co. | 1985 |
| **ILLINOIS** | | | | | | |
| Morris | Dresden Nuclear Power Station Unit 1 | 200 | BWR | OL 1959 | Commonwealth Edison Co. | 1960 |
| Morris | Dresden Nuclear Power Station Unit 2 | 794 | BWR | OL 1969 | Commonwealth Edison Co. | 1970 |
| Morris | Dresden Nuclear Power Station Unit 3 | 794 | BWR | OL 1971 | Commonwealth Edison Co. | 1971 |
| Zion | Zion Nuclear Plant Unit 1 | 1,040 | PWR | OL 1973 | Commonwealth Edison Co. | 1973 |
| Zion | Zion Nuclear Plant Unit 2 | 1,040 | PWR | OL 1973 | Commonwealth Edison Co. | 1974 |
| Cordova | Quad-Cities Station Unit 1 | 789 | BWR | OL 1972 | Comm. Ed. Co.-Iowa-Ill. Gas & Elec. Co. | 1973 |
| Cordova | Quad-Cities Station Unit 2 | 789 | BWR | OL 1972 | Comm. Ed. Co.-Iowa-Ill. Gas & Elec. Co. | 1973 |
| Seneca | LaSalle County Nuclear Station Unit 1 | 1,078 | BWR | CP 1973 | Commonwealth Edison Co. | 1979 |
| Seneca | LaSalle County Nuclear Station Unit 2 | 1,078 | BWR | CP 1973 | Commonwealth Edison Co. | 1980 |

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **ILLINOIS—Continued** | | | | | | |
| Byron | Byron Station Unit 1 | 1,120 | PWR | CP 1975 | Commonwealth Edison Co. | 1982 |
| Byron | Byron Station Unit 2 | 1,120 | PWR | CP 1975 | Commonwealth Edison Co. | 1983 |
| Braidwood | Braidwood Unit 1 | 1,120 | PWR | CP 1975 | Commonwealth Edison Co. | 1982 |
| Braidwood | Braidwood Unit 2 | 1,120 | PWR | CP 1975 | Commonwealth Edison Co. | 1983 |
| Clinton | Clinton Nuclear Power Plant Unit 1 | 950 | BWR | CP 1976 | Illinois Power Co. | 1982 |
| Clinton | Clinton Nuclear Power Plant Unit 2 | 950 | BWR | CP 1976 | Illinois Power Co. | 1988 |
| Savannah | Carroll County Station Unit 1 | 1,150 | | A/O | Commonwealth Edison Co. | 1988 |
| Savannah | Carroll County Station Unit 2 | 1,150 | | A/O | Commonwealth Edison Co. | 1989 |
| **INDIANA** | | | | | | |
| Westchester Town | Bailly Generating Station | 660 | BWR | CP 1974 | Northern Indiana Public Service Co. | 1984 |
| Madison | Marble Hill Unit 1 | 1,130 | PWR | CP 1978 | Public Service of Indiana | 1982 |
| Madison | Marble Hill Unit 2 | 1,130 | PWR | CP 1978 | Public Service of Indiana | 1984 |
| **IOWA** | | | | | | |
| Pala | Duane Arnold Energy Center Unit 1 | 538 | BWR | OL 1974 | Iowa Elec. Light & Power Co. | 1975 |
| **KANSAS** | | | | | | |
| Burlington | Wolf Creek | 1,150 | PWR | CP 1977 | Kansas Gas & Elec. Co. | 1983 |
| **LOUISIANA** | | | | | | |
| Taft | Waterford Steam Electric Station | 1,165 | PWR | CP 1974 | Louisiana Power & Light Co. | 1981 |
| St. Francisville | River Bend Station Unit 1 | 934 | BWR | CP 1977 | Gulf States Utilities Co. | 1984 |
| St. Francisville | River Bend Station Unit 2 | 934 | BWR | CP 1977 | Gulf States Utilities Co. | N/S |
| **MAINE** | | | | | | |
| Wiscasset | Maine Yankee Atomic Power Plant | 790 | PWR | OL 1972 | Maine Yankee Atomic Power Co. | 1972 |
| **MARYLAND** | | | | | | |
| Lusby | Calvert Cliffs Nuclear Power Plant Unit 1 | 845 | PWR | OL 1974 | Baltimore Gas & Elec. Co. | 1975 |
| Lusby | Calvert Cliffs Nuclear Power Plant Unit 2 | 845 | PWR | OL 1976 | Baltimore Gas & Elec. Co. | 1977 |

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **MASSACHUSETTS** | | | | | | |
| Rowe | Yankee Nuclear Power Station | 175 | PWR | OL 1960 | Yankee Atomic Elec. Co. | 1961 |
| Plymount | Pilgrim Station Unit 1 | 655 | BWR | OL 1972 | Boston Edison Co. | 1972 |
| Plymouth | Pilgrim Station Unit 2 | 1,180 | PWR | UR | Boston Edison Co. | 1985 |
| Turners Falls | Montague Unit 1 | 1,150 | BWR | UR | Northeast Nuclear Energy Co. | N/S |
| Turners Falls | Montague Unit 2 | 1,150 | BWR | UR | Northeast Nuclear Energy Co. | N/A |
| **MICHIGAN** | | | | | | |
| Big Rock Point | Big Rock Point Nuclear Plant | 72 | BWR | OL 1962 | Consumers Power Co. | 1963 |
| South Haven | Palisades Nuclear Power Station | 805 | PWR | OL 1971 | Consumers Power Co. | 1971 |
| Lagoona Beach | Enrico Fermi Atomic Power Plant Unit 2 | 1,123 | BWR | CP 1972 | Detroit Power Co. | 1980 |
| Bridgman | Donald C. Cook Plant Unit 1 | 1,054 | PWR | OL 1974 | Indiana & Michigan Elec. Co. | 1975 |
| Bridgman | Donald C. Cook Plant Unit 2 | 1,100 | PWR | OL 1977 | Indiana & Michigan Elec. Co. | 1978 |
| Midland | Midland Nuclear Power Plant Unit 1 | 492 | PWR | CP 1972 | Consumers Power Co. | 1982 |
| Midland | Midland Nuclear Power Plant Unit 2 | 818 | PWR | CP 1972 | Consumers Power Co. | 1981 |
| St. Clair County | Greenwood Energy Center Unit 2 | 1,200 | PWR | UR | Detroit Edison Co. | N/S |
| St. Clair County | Greenwood Energy Center Unit 3 | 1,200 | PWR | UR | Detroit Edison Co. | N/S |
| **MINNESOTA** | | | | | | |
| Monticello | Monticello Nuclear Generating Plant | 545 | BWR | OL 1970 | Northern States Power Co. | 1971 |
| Red Wing | Prairie Island Nuclear Generating Plant Unit 1 | 530 | PWR | OL 1973 | Northern States Power Co. | 1973 |
| Red Wing | Prairie Island Nuclear Generating Plant Unit 2 | 530 | PWR | OL 1974 | Northern States Power Co. | 1974 |
| **MISSISSIPPI** | | | | | | |
| Port Gibson | Grand Gulf Nuclear Station Unit 1 | 1,250 | BWR | CP 1974 | Mississippi Power & Light Co. | 1982 |
| Port Gibson | Grand Gulf Nuclear Station Unit 2 | 1,250 | BWR | CP 1974 | Mississippi Power & Light Co. | 1984 |

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **MISSISSIPPI—Continued** | | | | | | |
| Yellow Creek | Yellow Creek Unit 1 | 1,285 | PWR | CP 1978 | Tennessee Valley Authority | 1985 |
| Yellow Creek | Yellow Creek Unit 2 | 1,285 | PWR | CP 1978 | Tennessee Valley Authority | 1991 |
| **MISSOURI** | | | | | | |
| Fulton | Callaway Plant Unit 1 | 1,150 | PWR | CP 1976 | Union Elec. Co. | 1982 |
| Fulton | Callaway Plant Unit 1 | 1,150 | PWR | CP 1976 | Union Elec. Co. | 1987 |
| **NEBRASKA** | | | | | | |
| Fort Calhoun | Fort Calhoun Station Unit 1 | 457 | PWR | OL 1973 | Omaha Public Power District | 1973 |
| Brownville | Cooper Nuclear Station | 778 | BWR | OL 1974 | Nebraska Public Power District | 1974 |
| **NEW HAMPSHIRE** | | | | | | |
| Seabrook | Seabrook Nuclear Station Unit 1 | 1,194 | PWR | CP 1976 | Public Service of N.H. | 1983 |
| Seabrook | Seabrook Nuclear Station Unit 2 | 1,194 | PWR | CP 1976 | Public Service of N.H. | 1985 |
| **NEW JERSEY** | | | | | | |
| Toms River | Oyster Creek Nuclear Power Plant Unit 1 | 650 | BWR | OL 1969 | Jersey Central Power & Light Co. | 1969 |
| Forked River | Forked River Generating Station Unit 1 | 1,070 | PWR | CP 1973 | Jersey Central Power & Light Co. | 1984 |
| Salem | Salem Nuclear Generating Station Unit 1 | 1,090 | PWR | OL 1976 | Public Service Elec. & Gas Co. | 1977 |
| Salem | Salem Nuclear Generating Station Unit 2 | 1,115 | PWR | CP 1968 | Public Service Elec. & Gas Co. | 1979 |
| Salem | Hope Creek Generating Station Unit 1 | 1,067 | BWR | CP 1974 | Public Service Elec. & Gas. Co. | 1984 |
| Salem | Hope Creek Generating Station Unit 2 | 1,067 | BWR | CP 1974 | Public Service Elec. & Gas Co. | 1986 |
| **NEW YORK** | | | | | | |
| Indian Point | Indian Point Station Unit 1 | 265 | PWR | OL 1962 | Consolidated Edison Co. | 1962 |
| Indian Point | Indian Point Station Unit 2 | 873 | PWR | OL 1971 | Consolidated Edison Co. | 1973 |
| Indian Point | Indian Point Station Unit 3 | 965 | PWR | OL 1975 | Consolidated Edison Co. | 1976 |
| Scriba | Nine Mile Point Nuclear Station Unit 1 | 610 | BWR | OL 1969 | Niagara Mohawk Power Co. | 1969 |

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **NEW YORK—Continued** | | | | | | |
| Scriba | Nine Mile Point Nuclear Station Unit 2 | 1,080 | BWR | CP 1974 | Niagara Mohawk Power Co. | 1984 |
| Ontario | R.E. Ginna Nuclear Power Plant Unit 1 | 490 | PWR | OL 1969 | Rochester Gas & Elec. Co. | 1970 |
| Brookhaven | Shoreham Nuclear Power Station | 854 | BWR | CP 1973 | Long Island Lighting Co. | 1980 |
| Scriba | James A. FitzPatrick Nuclear Power Plant | 821 | BWR | OL 1974 | Power Authority of State of N.Y. | 1975 |
| Long Island | Jamesport Unit 1 | 1,150 | PWR | CP 1979 | Long Island Lighting Co. | 1988 |
| Long Island | Jamesport Unit 2 | 1,150 | PWR | CP 1979 | Long Island Lighting Co. | 1990 |
| * | New Haven 1 | 1,250 | PWR | UR | N.Y. State Elec. & Gas Co. | Indef. |
| * | New Haven 2 | 1,250 | PWR | UR | N.Y. State Elec. & Gas Co. | Indef. |
| Sterling | Sterling Power Project Unit 1 | 1,150 | PWR | CP 1977 | Rochester Gas & Elec. Co. | 1988 |
| **NORTH CAROLINA** | | | | | | |
| Southport | Brunswick Steam Electric Plant Unit 2 | 821 | BWR | OL 1974 | Carolina Power & Light Co. | 1975 |
| Southport | Brunswick Steam Electric Plant Unit 1 | 821 | BWR | OL 1976 | Carolina Power & Light Co. | 1977 |
| Cowans Ford Dam | Wm. B. McGuire Nuclear Station Unit 1 | 1,180 | PWR | CP 1973 | Duke Power Co. | 1980 |
| Cowans Ford Dam | Wm. B. McGuire Nuclear Station Unit 2 | 1,180 | PWR | CP 1973 | Duke Power Co. | 1982 |
| Bonsal | Shearon Harris Plant Unit 1 | 915 | PWR | CP 1978 | Carolina Power & Light Co. | 1983 |
| Bonsal | Shearon Harris Plant Unit 2 | 915 | PWR | CP 1978 | Carolina Power & Light Co. | 1985 |
| Bonsal | Shearon Harris Plant Unit 3 | 915 | PWR | CP 1978 | Carolina Power & Light Co. | 1989 |
| Bonsal | Shearon Harris Plant Unit 4 | 915 | PWR | CP 1978 | Carolina Power & Light Co. | 1987 |
| Davie Co. | Perkins Nuclear Station Unit 1 | 1,280 | PWR | UR | Duke Power Co. | 1988 |
| Davie Co. | Perkins Nuclear Station Unit 2 | 1,280 | PWR | UR | Duke Power Co. | 1991 |
| Davie Co. | Perkins Nuclear Station Unit 3 | 1,280 | PWR | UR | Duke Power Co. | 1993 |
| **OHIO** | | | | | | |
| Oak Harbor | Davis-Besse Nuclear Power Station Unit 1 | 906 | PWR | OL 1977 | Toledo Edison-Cleveland Elec. Illum. Co. | 1977 |
| Oak Harbor | Davis-Besse Nuclear Power Station Unit 2 | 906 | PWR | UR ** | Toledo Edison-Cleveland Elec. Illum. Co. | 1988 |

* Site not selected.

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **OHIO—Continued** | | | | | | |
| Oak Harbor | Davis-Besse Nuclear Power Station Unit 3 | 906 | PWR | UR ** | Toledo Edison-Cleveland Elec. Illum. Co. | 1990 |
| Perry | Perry Nuclear Power Plant Unit 1 | 1,205 | BWR | CP 1977 | Cleveland Elec. Illum. Co. | 1982 |
| Perry | Perry Nuclear Power Plant Unit 2 | 1,205 | BWR | CP 1977 | Cleveland Elec. Illum Co. | 1984 |
| Moscow | Wm. H. Zimmer Nuclear Power Station Unit 1 | 810 | BWR | CP 1972 | Cincinnati Gas & Elec. Co. | 1979 |
| Berlin Hgts. | Erie Unit 1 | 1,260 | PWR | UR | Ohio Edison Co. | 1989 |
| Berlin Hgts. | Erie Unit 2 | 1,260 | PWR | UR | Ohio Edison Co. | 1991 |
| **OKLAHOMA** | | | | | | |
| Inola | Black Fox Unit 1 | 1,150 | BWR | UR ** | Public Service Co. of Oklahoma | 1983 |
| Inola | Black Fox Unit 2 | 1,150 | BWR | UR ** | Public Service Co. of Oklahoma | 1985 |
| **OREGON** | | | | | | |
| Prescott | Trojan Nuclear Plant Unit 1 | 1,130 | PWR | OL 1975 | Portland General Elec. Co. | 1976 |
| Arlington | Pebble Springs Unit 1 | 1,260 | PWR | UR | Portland General Elec. Co. | 1986 |
| Arlington | Pebble Springs Unit 2 | 1,260 | PWR | UR | Portland General Elec. Co. | 1989 |
| **PENNSYLVANIA** | | | | | | |
| Peach Bottom | Peach Bottom Atomic Power Station Unit 2 | 1,065 | BWR | OL 1973 | Philadelphia Elec. Co. | 1974 |
| Peach Bottom | Peach Bottom Atomic Power Station Unit 3 | 1,065 | BWR | OL 1974 | Philadelphia Elec. Co. | 1974 |
| Pottstown | Limerick Generating Station Unit 1 | 1,065 | BWR | CP 1974 | Philadephia Elec. Co. | 1983 |
| Pottstown | Limerick Generating Station Unit 2 | 1,065 | BWR | CP 1974 | Philadelphia Elec. Co. | 1985 |
| Shippingport | Shippingport Atomic Power Unit 1 | 90 | PWR | — [1] | Duquesne Light Co. & DOE | NA |
| Shippingport | Beaver Valley Power Station Unit 1 | 852 | PWR | OL 1976 | Duquesne Light Co. Ohio Edison Co. | 1976 |
| Shippingport | Beaver Valley Power Station Unit 2 | 852 | PWR | CP 1974 | Duquesne Light Co. Ohio Edison Co. | 1983 |

** Limited work authorization issued.

[1] Operable but OL not required.

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **PENNSYLVANIA—Continued** | | | | | | |
| Goldsboro | Three Mile Island Nuclear Station Unit 1 | 819 | PWR | OL 1974 | Metropolitan Edison Co. | 1974 |
| Goldsboro | Three Mile Island Nuclear Station Unit 2 | 906 | PWR | OL 1978 | Metropolitan Edison Co. | 1978 |
| Berwick | Susquehanna Steam Electric Station Unit 1 | 1,052 | BWR | CP 1973 | Pennsylvania Power & Light Co. | 1980 |
| Berwick | Susquehanna Steam Electric Station Unit 2 | 1,052 | BWR | CP 1973 | Pennsylvania Power & Light Co. | 1982 |
| **RHODE ISLAND** | | | | | | |
| No. Kingston | New England Unit 1 | 1,150 | PWR | UR | New England Power Co. | 1987 |
| No. Kingston | New England Unit 2 | 1,150 | PWR | UR | New England Power Co. | 1989 |
| **SOUTH CAROLINA** | | | | | | |
| Hartsville | H.B. Robinson S.E. Plant Unit 2 | 700 | PWR | OL 1970 | Carolina Power & Light Co. | 1971 |
| Seneca | Oconee Nuclear Station Unit 1 | 887 | PWR | OL 1973 | Duke Power Co. | 1973 |
| Seneca | Oconee Nuclear Station Unit 2 | 887 | PWR | OL 1973 | Duke Power Co. | 1974 |
| Seneca | Oconee Nuclear Station Unit 3 | 887 | PWR | OL 1974 | Duke Power Co. | 1974 |
| Broad River | Virgil C. Summer Nuclear Station Unit 1 | 900 | PWR | CP 1973 | So. Carolina Elec. & Gas Co. | 1980 |
| Lake Wylie | Catawba Nuclear Station Unit 1 | 1,145 | PWR | CP 1975 | Duke Power Co. | 1983 |
| Lake Wylie | Catawba Nuclear Station Unit 2 | 1,145 | PWR | CP 1975 | Duke Power Co. | 1985 |
| Cherokee County | Cherokee Nuclear Station Unit 1 | 1,280 | PWR | CP 1977 | Duke Power Co. | 1986 |
| Cherokee County | Cherokee Nuclear Station Unit 2 | 1,280 | PWR | CP 1977 | Duke Power Co. | 1988 |
| Cherokee County | Cherokee Nuclear Station Unit 3 | 1,280 | PWR | CP 1977 | Duke Power Co. | 1988 |
| **TENNESSEE** | | | | | | |
| Daisy | Sequoyah Nuclear Power Plant Unit 1 | 1,140 | PWR | CP 1970 | Tennessee Valley Authority | 1979 |
| Daisy | Sequoyah Nuclear Power Plant Unit 2 | 1,140 | PWR | CP 1970 | Tennessee Valley Authority | 1980 |
| Spring City | Watts Bar Nuclear Plant Unit 1 | 1,165 | PWR | CP 1973 | Tennessee Valley Authority | 1979 |

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **TENNESSEE—Continued** | | | | | | |
| Spring City | Watts Bar Nuclear Plant Unit 2 | 1,165 | PWR | CP 1973 | Tennessee Valley Authority | 1980 |
| Oak Ridge | Clinch River Breeder Reactor Plant | 350 | LMFBR | UR | U.S. Government | Indef. |
| Hartsville | TVA Plant A Unit 1 | 1,205 | BWR | CP 1977 | Tennessee Valley Authority | 1982 |
| Hartsville | TVA Plant A Unit 2 | 1,205 | BWR | CP 1977 | Tennessee Valley Authority | 1983 |
| Hartsville | TVA Plant B Unit 1 | 1,205 | BWR | CP 1977 | Tennessee Valley Authority | 1988 |
| Hartsville | TVA Plant B Unit 2 | 1,205 | BWR | CP 1977 | Tennessee Valley Authority | 1989 |
| Phipps Bend | Phipps Bend Unit 1 | 1,220 | BWR | CP 1978 | Tennessee Valley Authority | 1983 |
| Phipps Bend | Phipps Bend Unit 2 | 1,220 | BWR | CP 1978 | Tennessee Valley Authority | 1986 |
| **TEXAS** | | | | | | |
| Glen Rose | Comanche Peak Steam Electric Station Unit 1 | 1,150 | PWR | CP 1974 | Texas P&L, Dallas P&L, Texas Elec. Service | 1981 |
| Glen Rose | Comanche Peak Steam Electric Station Unit 2 | 1,150 | PWR | CP 1974 | Texas P&L, Dallas P&L, Texas Elec. Service | 1983 |
| Wallis | Allens Creek Unit 1 | 1,150 | BWR | UR | Houston Lighting & Power Co. | 1985 |
| Bay City | South Texas Nuclear Project Unit 1 | 1,250 | PWR | CP 1975 | Houston Lighting & Power Co. | 1983 |
| Bay City | South Texas Nuclear Project Unit 2 | 1,250 | PWR | CP 1975 | Houston Lighting & Power Co. | 1985 |
| **VERMONT** | | | | | | |
| Vernon | Vermont Yankee Generating Station | 514 | BWR | OL 1972 | Vermont Yankee Nuclear Power Corp. | 1972 |
| **VIRGINIA** | | | | | | |
| Gravel Neck | Surry Power Station Unit 1 | 822 | PWR | OL 1972 | Va. Electric & Power Co. | 1972 |
| Gravel Neck | Surry Power Station Unit 2 | 822 | PWR | OL 1973 | Va. Electric & Power Co. | 1973 |
| Mineral | North Anna Power Station Unit 1 | 907 | PWR | OL 1976 | Va. Electric & Power Co. | 1978 |
| Mineral | North Anna Power Station Unit 2 | 907 | PWR | CP 1971 | Va. Electric & Power Co. | 1979 |

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **VIRGINIA—Continued** | | | | | | |
| Mineral | North Anna Power Station Unit 3 | 907 | PWR | CP 1974 | Va. Electric & Power Co. | 1985 |
| Mineral | North Anna Power Station Unit 4 | 907 | PWR | CP 1974 | Va. Electric & Power Co. | 1986 |
| * | Central Virginia 1 | 1,150 | | A/O | American Electric Power Co. | 1990 |
| * | Central Virginia 2 | 1,150 | | A/O | American Electric Power Co. | 1990 |
| **WASHINGTON** | | | | | | |
| Richland | N-Reactor/WPPSS Steam | 850 | GR | —[1] | Wash. Public Power Supply System | |
| Richland | WPPSS No. 1 (Hanford) | 1,267 | PWR | CP 1975 | Wash. Public Power Supply System | 1983 |
| Richland | WPPSS No. 2 (Hanford) | 1,103 | BWR | CP 1973 | Wash. Public Power Supply System | 1981 |
| Satsop | WPPSS No. 3 | 1,242 | PWR | CP 1978 | Wash. Public Power Supply System | 1984 |
| Richland | WPPSS No. 4 | 1,267 | PWR | CP 1978 | Wash. Public Power Supply System | 1984 |
| Satsop | WPPSS No. 5 | 1,242 | PWR | CP 1978 | Wash. Public Power Supply System | 1985 |
| Sedro Wooley | Skagit Nuclear Power Project Unit 1 | 1,277 | BWR | UR | Puget Sound Power & Light Co. | 1985 |
| Sedro Wooley | Skagit Nuclear Power Project Unit 2 | 1,277 | BWR | UR | Puget Sound Power & Light Co. | 1987 |
| **WISCONSIN** | | | | | | |
| Genoa | Genoa Nuclear Generating Station (LaCrosse) | 50 | BWR | OL 1967 | Dairyland Power Coop. | 1969 |
| Two Creeks | Point Beach Nuclear Plant Unit 1 | 497 | PWR | OL 1970 | Wisconsion Michigan Power Co. | 1970 |
| Two Creeks | Point Beach Nuclear Plant Unit 2 | 497 | PWR | OL 1971 | Wisconsin Michigan Power Co. | 1972 |
| Carlton | Kewaunee Nuclear Power Plant | 535 | PWR | OL 1973 | Wisconsin Elec. Power Co. | 1974 |
| Ft. Atkinson | Haven Nuclear Plant | 900 | PWR | UR | Wisconsin Elec. Power Co. | 1987 |

* Site not selected.

[1] Operable but OL not required.

# Index

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