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Potential of thorium based fuel cycles to constrain plutonium and reduce long lived waste toxicity

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1995–2001

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POTENTIAL OF THORIUM BASED FUEL CYCLES TO CONSTRAIN PLUTONIUM AND REDUCE LONG LIVED WASTE TOXICITY

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FOREWORD

An important function of the International Atomic Energy Agency is to "foster the exchange of scientific and technical information" and to "encourage and assist research on, and development and practical application of, atomic energy for peaceful uses throughout the world". For innovative advanced nuclear reactor concepts, IAEA Member States in many cases find it attractive to co-operate internationally in technology development. The IAEA's fast reactor and hybrid systems technology development activities, which are conducted within its nuclear power programme, encourage international co-operation through technical information exchange and collaborative research. As regards the latter, co-ordinated research projects (CRPs) are tools that are effectively used in the implementation of the IAEA's activities, both to promote exchange of scientific and technical information, and to pursue collaborative research and development tasks. Apart from allowing the efforts to be shared on an international basis and benefitting from the joint experience and expertise of researchers from the participating institutes, CRPs foster international team building.

From 1995 to 2001, the IAEA initiated a CRP on Potential of Thorium Based Fuel Cycles to Constrain Plutonium and to Reduce Long Term Waste Toxicity. The Member States involved in the CRP were: China, Germany, India, Israel, Japan, Republic of Korea, Netherlands, Russian Federation and the United States of America.

The research programme was divided into three stages: (1) benchmark calculations, (2) optimization of the incineration of plutonium in various reactor types, and (3) assessment of the resulting impact on the waste radio toxicity.

The results of all three stages were presented at international conferences, specifically, ICENES 98, ICENES 2000, and PHYSOR 2002 respectively.

The present report was prepared with the assistance of H.J. Rütten, Research Center Jülich (FZJ, Germany). The IAEA officer responsible for this publication was A. Stanculescu of the Division of Nuclear Power.

EDITORIAL NOTE

In preparing this publication for press, staff of the IAEA have made up the pages from the original manuscript(s). The views expressed do not necessarily reflect those of the IAEA, the governments of the nominating Member States or the nominating organizations.

Throughout the text names of Member States are retained as they were when the text was compiled.

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1. INTRODUCTION

Large stockpiles of civil plutonium have accumulated in the world from the different countries' nuclear power programs. There is a serious public and political concern in the world about misuse of this plutonium and about accidental release of highly radiotoxic material into the environment. It therefore becomes necessary to keep the plutonium under strong security. One alternative for the management of plutonium is to incinerate it in reactors. But if the plutonium is fueled in reactors in the form of uranium/plutonium mixed oxide (MOX), second-generation plutonium is produced. A possible solution to this problem is to incinerate plutonium in combination with thorium. The thorium cycle produces ^{233}U which, from a non-proliferation point of view, is preferable to plutonium for two reasons. Firstly, it is contaminated with ^{232}U, which decays to give highly active daughter products. This would make handling and diversion difficult. Secondly, in case this is not sufficient deterrent, the ^{233}U could be denatured by adding some U^{238} to the thorium. The quantity of ^{238}U could be fine-tuned so as to be sufficient to denature the ^{233}U, but not so much as to produce a significant quantity of plutonium. The thorium option not only produces electricity, but also replaces the plutonium with denatured ^{233}U, which can be used in other reactors at a later date. All this can be done in existing reactors.

In the framework of IAEA activities on the use of thorium as nuclear fuel, a report on the performance of the thorium cycle, entitled A Fresh Look at the Thorium Fuel Cycle was drafted in 1991 and distributed as Working Material. IAEA-TECDOC-1155, entitled Thorium Based Fuel Options for the Generation of Electricity: Developments in the 1990s, was published as a follow-up action.

Co-ordinated Research Projects (CRPs) are tools that are effectively used by the IAEA to promote exchange of scientific and technical information and assist advanced nuclear power reactor technology research and development. CRPs allow the sharing of efforts on an international basis, benefiting from the experience and expertise of researchers from the participating institutes, and fostering international team building.

At the Consultants Meeting on Important Consideration on the Status of Thorium held in Vienna from 29 November to 1 December 1994, participants recommended the IAEA to organize a CRP on thorium-based fuel cycle issue. In 1995, the IAEA approved the topic for the CRP: Potential of Thorium based Fuel Cycles to Constrain Plutonium and to Reduce Long term Waste Toxicity. The scope of this CRP was discussed and agreed upon by the participants of the Consultants Meeting on Thorium based Fuel Cycles, held from 6 to 9 June 1995 at the IAEA in Vienna. The participating countries in the CRP were: China, Germany, India, Israel, Japan, Republic of Korea, Netherlands, Russian Federation and the United States of America.

This CRP examined the different fuel cycle options in which plutonium can be recycled with thorium to incinerate the burner. The potential of the thorium-matrix has been examined through computer simulations. Each participant has chosen his own cycle, and the different cycles were compared through certain predefined parameters (e.g., annual reduction of plutonium stockpiles). The toxicity accumulation and the transmutation potential of thorium-based cycles for current, advanced and innovative nuclear power reactors were investigated. As a final outcome, the CRP as next step would suggest to concentrate on the practical demonstration of plutonium-thorium incineration in a reactor in one of the member countries.

2. SYNTHESIS OF THE CO-ORDINATED RESEARCH PROJECT'S TASKS AND RESULTS

2.1. COMPARISON OF METHODS AND BASIC NUCLEAR DATA

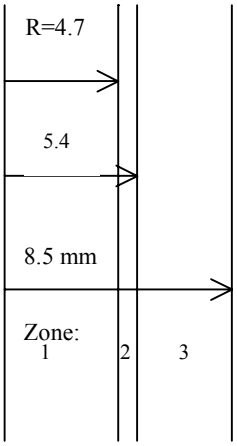
2.1.1. Cell-burnup calculations

In order to establish a comparison of the effect of different methods and databases applied in the countries participating in the CRP, benchmark calculations had to be performed before the start of the actual fuel cycles studies. For the first plutonium incineration benchmark calculations, the PWR-type reactor has been chosen because it is the reactor type that has the largest share in the current production of nuclear energy. The following topic was selected for the IAEA benchmark 1:

"Calculation of the isotopic composition, cross-sections and fluxes for a typical PWR-cell loaded with (Pu-Th)O_2 - fuel, as a function of the fuel burnup."

2.1.1.1. Definition of the fuel cell and tasks

The geometry of the reference fuel cell is displayed in Fig. 2.1. Table 2.1 gives the description of the material composition in terms of nuclide concentrations for the different cell zones.



Infinite lattice

Average power:	P = 211 W/cm
Average temperature of the fuel:	T_{fuel} = 1023 K
Average temperature of the water:	T_{mod} = 583 K

FIG. 2.1. Layout of the reference fuel cell.

TABLE 2.1. INITIAL NUCLIDE DENSITIES IN THE CELL (atoms/cm^3)

	average in cell	zone 1	zone 2	zone 3
Th-232	6.45E+21	2.11E+22		
Pu-238	2.97E+18	9.72E+18		
Pu-239	1.83E+20	5.99E+20		
Pu-240	7.10E+19	2.32E+20		
Pu-241	2.35E+19	7.69E+19		
Pu-242	1.46E+19	4.78E+19		
Cr	1.99E+20		8.14E+19	3.20E+20
Mn	1.26E+19			2.11E+19
Fe	5.20E+20		1.60E+20	8.46E+20
Ni	2.24E+20			3.76E+20
Zr	4.27E+21		4.37E+22	
C	1.60E+18			2.68E+18
H	2.86E+22			4.80E+22
O	2.78E+22	4.41E+22		2.40E+22

The task to be performed for this benchmark exercise was defined as follows:

Calculate the fuel burnup at constant power (211 W/cm) as a function of time, not using any neutron poison for reactivity control. For a burnup of 0, 30, 40, and 60 MWd per kg of heavy metal report the following items:

(1) Neutron multiplication (k_{eff});
(2) Total neutron flux;
(3) Average energy per fission;
(4) Residual amount of plutonium;
(5) Fraction of fissile plutonium;
(6) Amount of generated minor actinides;
(7) Average, (1-group, for the comparison) microscopic cross-sections for absorption, and fission for the heavy metal isotopes from ^{232}Th through ^{244}Cm.

2.1.1.2. Benchmark results

The comparison of the results achieved by the participants is displayed in Figs 2.2 - 2.8 and in Tables 2.2 and 2.3. The results show some deviations, e.g., in the calculated cell reactivity (ranging from $\Delta\rho \approx 2\%$ initially to $\Delta\rho \approx 5\%$ at the end of burnup) and in the average effective energy per fission of the respective mixture of fissionable isotopes (discrepancy up to 4%). The results for the incineration rate of the plutonium isotopes, and for the buildup of minor actinides out of plutonium, as well as the ^{233}U buildup from ^{232}Th are in a good agreement.

Based on these results, the participants of the CRP came to the conclusion that:

— generally, the different methods and databases are comparable to the degree, needed to permit sharing of the research for different reactor types among different groups of countries;

— however, a second benchmark should be performed for the special heterogeneity of a PWR-lattice.

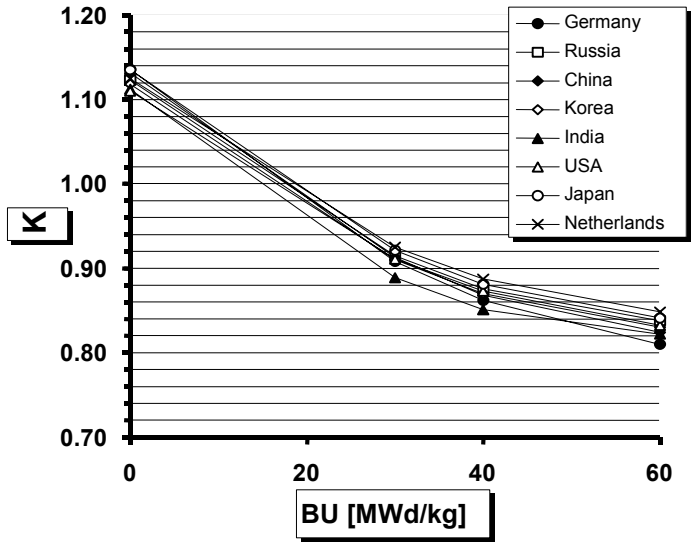


FIG. 2.2. Neutron multiplication vs. heavy metal burnup.

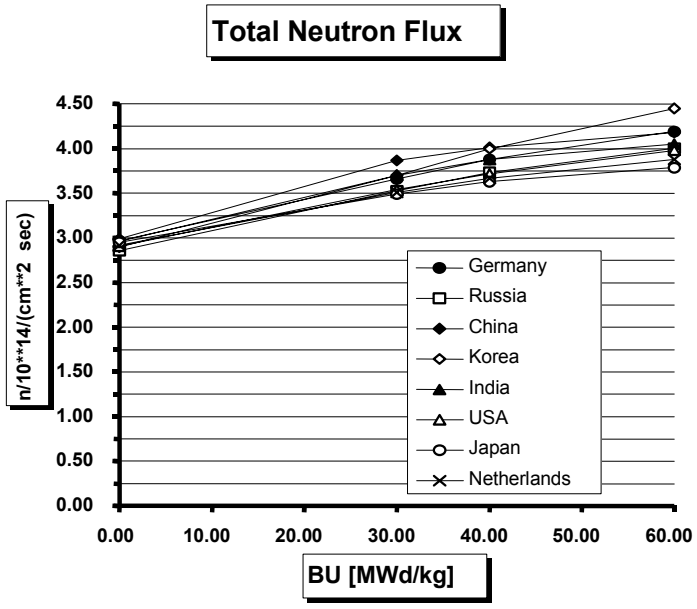


FIG. 2.3. Total neutron flux vs. heavy metal burnup.

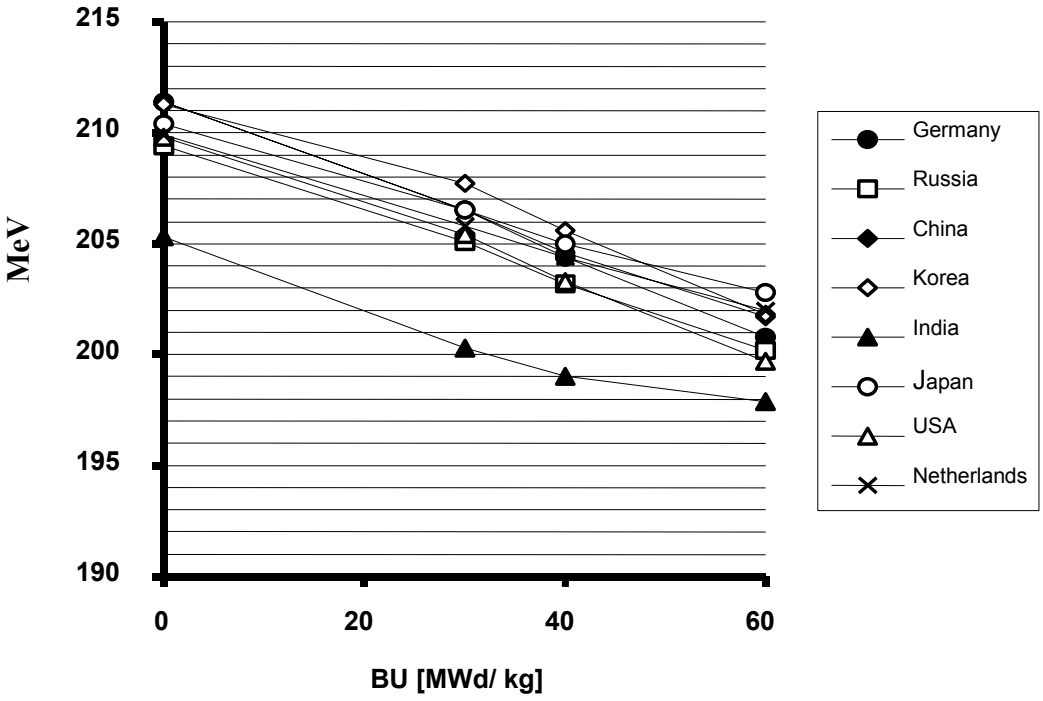


FIG. 2.4. Average energy per fission vs. heavy metal burnup.

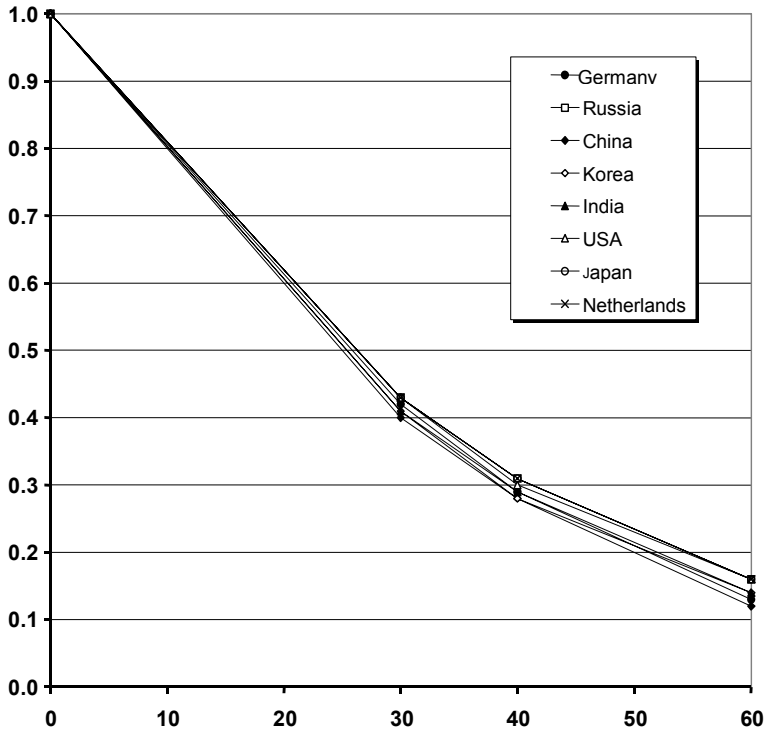


FIG. 2.5. ($Pu/Pu_{initial}$) vs. heavy metal burnup (MWd/t).

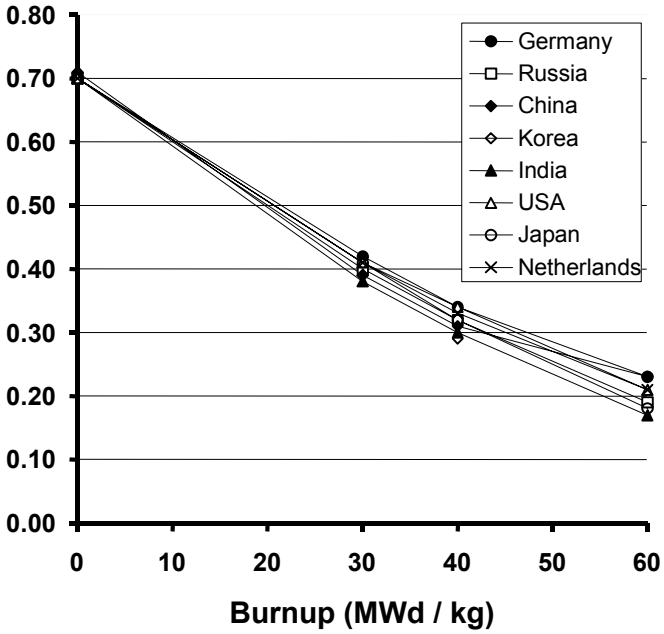


FIG. 2.6. (Pu-fiss/Pu-total) vs. heavy metal burnup.

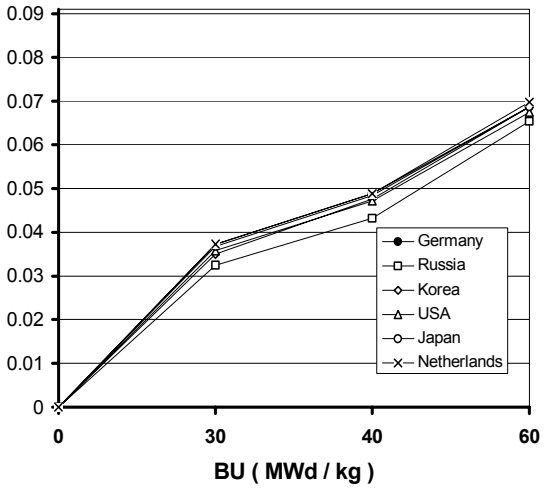


FIG. 2.7. Minor actinides/initial plutonium.

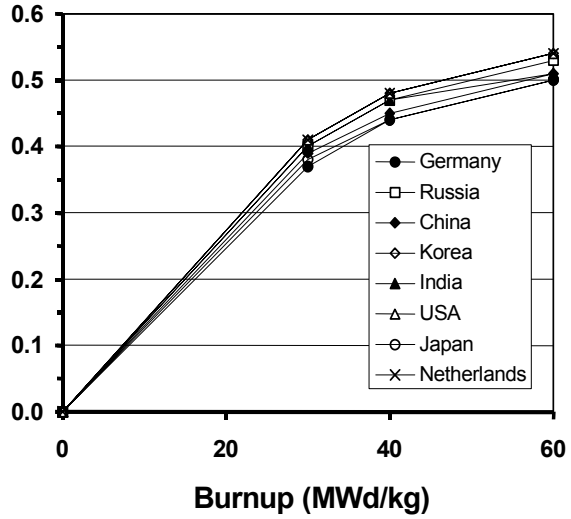


FIG. 2.8. ^{233}U bred from Th/initial fissile plutonium.

TABLE 2.2. CROSS-SECTIONS AT BURNUP = 0 MWd/kg

		Germany	Russia	China	Korea	India	USA	Japan	NL
Th-232									
	fission	0.0249	0.0260	0.0247	0.0281	0.0259	0.0276	0.029	0.0264
	absorption	0.7889	0.8579	0.7986	0.8557	0.8664	0.858	0.817	0.849
Pa-233									
	fission	0.1631	0.3879		0.175	0.3836		0.128	0.1743
	absorption	24.95	28.11		24.55	27.7		24.97	25.11
U-233									
	fission	35.45	36.77		34.29	35.7		34.93	35.16
	absorption	40.6	42.05		39.37	41.3		40.03	40.29
U-234									
	fission	0.531	0.4961		0.5607			0.549	0.565
	absorption	22.46	21.97		22.66			20.9	22.22
U-235									
	fission	22.43	22.5		21.51	22.6		21.65	22.12
	absorption	29.07	28.81		27.55	29.3		27.73	28.32
U-236									
	fission	0.3466	0.2119		0.3528			0.352	0.357
	absorption	10.85	9.669		10.11			10.12	10.6
U-238									
	fission	0.11	0.1062					0.114	0.114
	absorption	8.219	8.200					7.785	7.923
Np-237									
	fission	0.528	0.5607		0.5656			0.571	0.564
	absorption	27.92	25.56		27.14			27.21	27.7
Np-239									
	fission	0.621	0.6383		0.6578			0.659	0.664
	absorption	15.28	15.03		15.9			14.55	14.74
Pu-238									
	fission	1.981	2.171		1.963	2.04	2.037	2.038	2.05
	absorption	17.94	17.60		17.3	17.5	17.05	16.83	17.3
Pu-239									
	fission	44.63	44.98	44.31	43.23	44.1	44.05	43.89	44.09
	absorption	69.74	69.90	69.00	66.98	68.5	69.06	68.11	68.4
Pu-240									
	fission	0.630	0.5847	0.6224	0.6209	0.5459	0.5659	0.618	0.6593
	absorption	49.72	50.90	42.70	49.62	48.3	48.38	50.24	49.57
Pu-241									
	fission	55.03	55.32	56.27	52.89	53.6	54.18	52.29	54.27
	absorption	72.97	73.04	74.56	69.74	74.16	71.84	69.68	71.69
Pu-242									
	fission	0.4502	0.4448	0.4455	0.4754		0.4988	0.475	0.4979
	absorption	33.91	19.84	35.49	19.14	27.2	29.68	22.81	23.63
Am-241									
	fission	0.8569	0.8580		0.9062	0.923		0.881	0.9617
	absorption	62.47	63.68		60.6	65.09		60.24	64.72
Am-242m									
	fission	289.3	312.0		277.31			264.0	287.1
	absorption	346.0	382.9		331.8			314.9	352.1
Am-243									
	fission	0.4479	0.4638		0.5038			0.485	
	absorption	49.67	49.51		48.77			50.04	
Cm-242									
	fission	0.4503	1.295		0.4705			1.312	1.0818
	absorption	5.699	5.326		4.222			5.056	4.952
Cm-243									
	fission	72.51	74.18		70.51			58.34	61.33
	absorption	80.97	82.82		78.77			66.72	71.8
Cm-244									
	fission	0.9772	0.8405		1.082			0.872	1.041
	absorption	18.29	19.6		23.55			19.90	18.01

TABLE 2.3. CROSS-SECTIONS AT BURNUP = 60 MWd/kg

		Germany	Russia	China	Korea	India	USA	Japan	NL
Th-232									
	fission	0.0228	0.0240	0.0226	0.0246	0.0237	0.0254	0.025	0.0234
	absorption	1.0893	1.141	1.085	1.295	1.170	1.1425	1.163	1.132
Pa-233									
	fission	0.1492	0.3568	0.1482	0.1512	0.3500	0.3669	0.113	0.155
	absorption	21.78	23.85	21.78	22.96	24.3	23.6	22.21	21.34
U-233									
	fission	57.42	57.56	57.16	67.68	58.9	56.35	60.79	55.4
	absorption	64.40	64.23	64.17	75.68	66.2	63.41	68.06	62.12
U-234									
	fission	0.5056	0.4561	0.5025	0.5093		0.468	0.483	0.528
	absorption	23.56	16.74	24.29	24.44		16.73	19.66	19.1
U-235									
	fission	47.45	46.79	47.27	57.63	48.6	47.14	50.66	45.8
	absorption	58.36	57.03	58.23	69.62	59.7	57.5	61.56	55.91
U-236									
	fission	0,3086	0.1951	0.3116	0.2818		0.203	0.309	0.309
	absorption	9.252	8.289	9.627	6.829		8.342	8.82	8.64
U-238									
	fission	0.1006	0.0792				0.1071	0.10	0.101
	absorption	7.595	7.227				7.207	7.241	7.234
Np-237									
	fission	0.4830	0.5160	0.4796	0.483		0.5083	0.505	0.509
	absorption	34.69	31.66	34.07	39.19		34.59	35.48	34.19
Np-239									
	fission	0.5667	0.5864		0.5581		0.5981	0.583	0.6001
	absorption	15.69	15.08		19.68		15.03	15.25	15.01
Pu-238									
	fission	2.468	2.635		2.655	2.58	2.538	2.615	2.525
	absorption	38.75	36.92		47.22	38.9	36.96	39.61	36.36
Pu-239									
	fission	116.7	116.9	113.3	146.3	120.9	116.5	129.3	114.01
	absorption	182.8	183.0	177.0	228.3	189.0	184.1	202.3	178.1
Pu-240									
	fission	0.5927	0.5572	0.5836	0.5611	0.5117	0.5322	0.566	0.6163
	absorption	126.3	135.5	97.98	172.6	123.6	121.0	143.27	126.9
Pu-241									
	fission	126.3	124.9	125.3	153.7	124.6	124.7	133.6	121.8
	absorption	168.5	166.9	167.1	205.6	175.3	166.4	180.0	162.8
Pu-242									
	fission	0.4115	0.4089	0.4087	0.4060		1.437	0.419	0.446
	absorption	29.49	16.75	31.91	14.00	21.70	79.95	17.24	17.84
Am-241									
	fission	1.090	1.118		1.269	1.283	1.285	1.204	1.256
	absorption	111.8	116.3		136.5	123.9	119.6	126.4	116.01
Am-242m									
	fission	718.7	784.7		888.5			745.8	706.01
	absorption	864.0	967.6		1069			892.2	870.5
Am-243									
	fission	0.4094	0.4262		0.4363		0.4268	0.431	0.423
	absorption	42.02	41.98		46.39		43.09	44.22	41.73
Cm-242									
	fission	0.5550	14.18		0.6125			1.457	1.184
	absorption	6.2300	5.893		5.26			5.837	5.597
Cm-243									
	fission	98.66	97.21		112.3			88.11	73.7
	absorption	109.0	107.4		123.8			102.28	87.47
Cm-244									
	fission	0.9174	0.8148		0.9747			0.829	0.964
	absorption	16.98	17.06		21.67			18.56	15.39

2.1.2. Lattice calculations for LWR

While for a Pebble-bed HTR, having a nearly homogeneous core structure, a neutronics calculation for the entire core is regarded to be the adequate step following the cell calculation, an additional inter-comparison of the heterogeneous lattice calculation methodology appeared to be useful in case of the PWR. Thus, for the PWR part of the CRP, a second benchmark was established. Five countries participated in this benchmark: India, Israel, Japan, Republic of Korea, and Russian Federation. The benchmark was designed to compare assembly-level calculation methods, by defining a 2-D lattice simulating a typical PWR fuel assembly.

2.1.2.1. Benchmark definition and tasks

General:

17x17 array of the fuel rods, including 25 water hole positions.
No guide tubes material. No assembly casing.
No buckling. Quarter assembly symmetry.
Burnup calculations with constant specific power of 37.7 MW/t (initial heavy metal).

Geometry:

Outer dimensions, cm:	22.662×22.662
Cell pitch, cm:	1.33306
Fuel pellet radius, cm:	0.4127
Cladding thickness, cm:	0.0617
Equiv. cell radius, cm:	0.7521

Material compositions (atoms/barn x cm):

Fuel:

5% PuO_2 + 95% ThO_2. Temperature: 900 K.

Th-232	2.0592E-2
Pu-238	2.2900E-5
Pu-239	7.4780E-4
Pu-240	2.9030E-4
Pu-241	1.5340E-4
Pu-242	5.0100E-5
O-16	4.3710E-2

Cladding:

Natural Zr. Temperature: 600 K.

Zr-nat.	4.3241E-2

Moderator:

Light water, with 500 ppm natural boron. Temperature: 573 K.

H-1	4.7708E-2
O-16	2.3854E-2
B-10	3.9518E-6
B-11	1.5906E-5

Results for comparison:

(1) Criticality as a function of burnup. Burnup range from 0 to 60 GWd/t;
(2) Fuel composition as a function of burnup (major actinides and fission products);
(3) Local pin-by-pin power distribution;
(4) Moderator temperature coefficient for 0 and 60 GWd/t;
(5) Doppler coefficient for 0 and 60 GWd/t;
(6) Soluble boron worth for 0 and 60 GWd/t.

2.1.2.2. Benchmark results

Infinite multiplication factor:

The results are summarized in Table 2.4 and Fig. 2.9.

TABLE 2.4. k_{inf} AS A FUNCTION OF BURNUP

Burnup, GWd/T	Russian Federation	Japan	Republic of Korea	India	Israel
0	1.189	1.1987	1.1864	1.2076	1.1956
0.5	1.1569	1.1670	1.1551	1.1736	1.1643
20	1.0298	1.0521	1.0303	1.0372	1.0290
40	0.9147	0.9527	0.9167	0.9104	0.9119
60	0.8315	0.8657	0.8310	0.8294	0.8314

Significant discrepancies are found between k_{inf} values: ~ 2.3% ΔK at BOL and ~ 3.5% ΔK at EOL. It is also noted that there is no clear burnup dependency of the discrepancies.

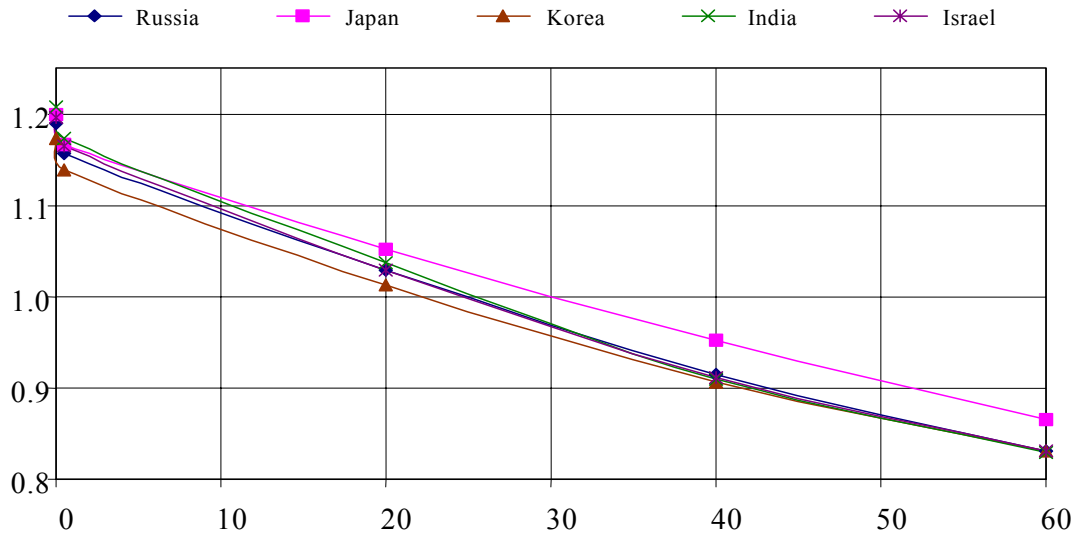


Fig. 2.9. k_{inf} as a function of burnup.

Fuel composition:

The results obtained for the composition of the actinide fuel are listed in Table 2.5 as a function of the heavy metal burnup. Note that there is:

— a good agreement for the ^{238}Pu concentration;

— a reasonable agreement for the even plutonium isotopes concentrations;

— significant discrepancies are found for the odd plutonium isotopes;

— a reasonable agreement in the case of the ^{232}Th and ^{233}U concentrations.

TABLE 2.5. FUEL COMPOSITION (ACTINIDES) AS A FUNCTION OF BURNUP

Number Density (atoms/barn×cm)					
Burnup, GWd/T	Russia	Japan	Rep. of Korea	India	Israel
Th-232					
0.0	2.059×10^{-2}	2.059×10^{-2}	2.059×10^{-2}	2.059×10^{-2}	2.059×10^{-2}
0.5	2.059×10^{-2}	2.059×10^{-2}	2.059×10^{-2}	-	-
20.0	2.037×10^{-2}	2.036×10^{-2}	2.037×10^{-2}	2.036×10^{-2}	2.037×10^{-2}
40.0	2.011×10^{-2}	2.008×10^{-2}	2.011×10^{-2}	-	2.010×10^{-2}
60.0	1.977×10^{-2}	1.975×10^{-2}	1.978×10^{-2}	1.970×10^{-2}	1.977×10^{-2}
Pu-238					
0.0	2.290×10^{-5}	2.290×10^{-5}	2.290×10^{-5}	2.290×10^{-5}	2.290×10^{-5}
0.5	2.279×10^{-5}	2.279×10^{-5}	2.279×10^{-5}	-	-
20.0	1.940×10^{-5}	1.952×10^{-5}	1.928×10^{-5}	1.829×10^{-5}	1.937×10^{-5}
40.0	1.834×10^{-5}	1.879×10^{-5}	1.798×10^{-5}	-	1.793×10^{-5}
60.0	1.687×10^{-5}	1.816×10^{-5}	1.636×10^{-5}	7.488×10^{-6}	1.611×10^{-5}
Pu-239					
0.0	7.478×10^{-4}	7.478×10^{-4}	7.478×10^{-4}	7.478×10^{-4}	7.478×10^{-4}
0.5	7.348×10^{-4}	7.351×10^{-4}	7.349×10^{-4}	-	-
20.0	3.174×10^{-4}	3.270×10^{-4}	3.175×10^{-4}	2.993×10^{-4}	3.147×10^{-4}
40.0	0.810×10^{-4}	0.961×10^{-4}	0.820×10^{-4}	-	0.773×10^{-4}
60.0	0.118×10^{-4}	0.170×10^{-4}	0.121×10^{-4}	0.479×10^{-4}	0.105×10^{-4}
Pu-240					
0.0	2.903×10^{-4}	2.903×10^{-4}	2.903×10^{-4}	2.903×10^{-4}	2.903×10^{-4}
0.5	2.911×10^{-4}	2.909×10^{-4}	2.911×10^{-4}	-	-
20.0	2.826×10^{-4}	2.678×10^{-4}	2.820×10^{-4}	2.846×10^{-4}	2.853×10^{-4}
40.0	1.981×10^{-4}	1.845×10^{-4}	1.991×10^{-4}	-	2.014×10^{-4}
60.0	0.809×10^{-4}	0.839×10^{-4}	0.874×10^{-4}	0.670×10^{-4}	0.846×10^{-4}
Pu-241					
0.0	1.534×10^{-4}	1.534×10^{-4}	1.534×10^{-4}	1.534×10^{-4}	1.534×10^{-4}
0.5	1.540×10^{-4}	1.543×10^{-4}	1.541×10^{-4}	-	-
20.0	1.591×10^{-4}	1.703×10^{-4}	1.605×10^{-4}	1.545×10^{-4}	1.578×10^{-4}
40.0	1.233×10^{-4}	1.360×10^{-4}	1.231×10^{-4}	-	1.214×10^{-4}
60.0	0.650×10^{-4}	0.741×10^{-4}	0.641×10^{-4}	0.539×10^{-4}	0.639×10^{-4}
Pu-242					
0.0	0.5010×10^{-4}	0.5010×10^{-4}	0.5010×10^{-4}	0.5010×10^{-4}	0.5010×10^{-4}
0.5	0.5050×10^{-4}	0.5043×10^{-4}	0.5051×10^{-4}	-	-
20.0	0.7088×10^{-4}	0.6813×10^{-4}	0.7248×10^{-4}	0.7203×10^{-4}	0.7020×10^{-4}
40.0	0.9877×10^{-4}	0.9245×10^{-4}	1.0380×10^{-4}	-	0.9832×10^{-4}
60.0	1.1890×10^{-4}	1.1030×10^{-4}	1.2880×10^{-4}	1.1624×10^{-4}	1.1940×10^{-4}
U-233					
0.0	-	-	-	-	-
0.5	0.7319×10^{-6}	0.7918×10^{-6}	0.7378×10^{-6}	-	-
20.0	1.5150×10^{-4}	1.5996×10^{-4}	1.5350×10^{-4}	1.5960×10^{-4}	1.5330×10^{-4}
40.0	2.6120×10^{-4}	2.7492×10^{-4}	2.6400×10^{-4}	-	2.6750×10^{-4}
60.0	3.1350×10^{-4}	3.3109×10^{-4}	3.1600×10^{-4}	3.1910×10^{-4}	3.2350×10^{-4}
U-234					
0.0	-	-	-	-	-
0.5	0.2361×10^{-7}	0.2522×10^{-7}	0.1556×10^{-7}	-	-
20.0	0.8565×10^{-4}	0.9714×10^{-5}	0.8025×10^{-5}	0.9627×10^{-5}	0.7913×10^{-5}
40.0	2.6680×10^{-4}	2.8855×10^{-5}	2.5200×10^{-5}	-	2.5290×10^{-5}
60.0	5.3200×10^{-4}	5.4315×10^{-5}	4.9070×10^{-5}	6.1950×10^{-5}	5.0450×10^{-5}

Pin-by-pin power distribution:

The local power distributions are shown in Figs 2.10a and 2.10b for the BOL and 60 GWd/t burnup points, respectively. A very good agreement for the BOL is indicated, with less than 2% relative power differences. The "hot rod" is identified with almost identical power of 1.124. A divergence of the local power values with burnup resulted in 5–10% differences, which may be partially attributed to different fissile (odd numbers) plutonium isotope concentrations.

W								
1.071 1.068 1.071 1.075	1.025 1.019 1.015 1.015			xxxx xxxx xxxx xxxx	Russia Japan Korea Israel			
1.059 1.068 1.071 1.076	1.014 1.020 1.015 1.016	1.015 1.021 1.017 1.018						
W	1.068 1.068 1.070 1.075	1.072 1.071 1.073 1.078	W					
1.063 1.066 1.070 1.074	1.016 1.018 1.015 1.015	1.009 1.023 1.019 1.020	1.084 1.084 1.087 1.091	1.080 1.068 1.077 1.074				
1.072 1.061 1.066 1.069	1.013 1.014 1.011 1.011	1.005 1.019 1.016 1.017	1.08 1.085 1.088 1.092	1.122 1.107 1.126 1.124	W			
W	1.053 1.051 1.056 1.060	1.058 1.054 1.061 1.064	W	1.099 1.081 1.100 1.100	1.049 1.045 1.053 1.055	0.963 0.965 0.964 0.964		
1.051 1.021 1.031 1.034	0.979 0.977 0.978 0.980	0.98 0.978 0.980 0.981	1.033 1.024 1.034 1.036	0.973 0.976 0.976 0.976	0.922 0.941 0.928 0.931	0.910 0.915 0.902 0.904	0.886 0.899 0.884 0.884	
0.925 0.929 0.926 0.926	0.924 0.925 0.923 0.922	0.921 0.925 0.922 0.921	0.930 0.929 0.924 0.923	0.922 0.921 0.914 0.914	0.907 0.910 0.898 0.897	0.889 0.901 0.886 0.885	0.889 0.896 0.878 0.876	0.876 0.898 0.876 0.872

0 GWd / t

FIG. 2.10a. Relative power distribution.

W								
0.951 1.037 1.025 1.054	0.926 1.027 1.015 1.027			xxxx xxxx xxxx xxxx	Russia Japan Korea Israel			**60 GWd / t**
0.944 1.037 1.025 1.055	0.931 1.027 1.015 1.027	0.939 1.027 1.015 1.028						
W	0.955 1.037 1.024 1.054	0.96 1.038 1.025 1.056	W					
0.959 1.034 1.023 1.051	0.952 1.024 1.013 1.025	0.950 1.027 1.014 1.028	0.986 1.042 1.027 1.064	0.975 1.039 1.022 1.057				
0.979 1.028 1.019 1.046	0.957 1.018 1.010 1.019	0.967 1.021 1.011 1.023	1.001 1.038 1.024 1.060	0.922 1.041 1.026 1.074	W			
W	0.984 1.017 1.013 1.034	0.991 1.019 1.013 1.037	W	1.024 1.024 1.015 1.052	1.003 1.007 1.004 1.022	1.014 0.978 0.983 0.969		
1.026 0.997 1.003 1.013	1.009 0.989 0.994 0.988	1.022 0.989 0.994 0.989	1.032 0.997 1.001 1.012	1.030 0.985 0.989 0.981	1.051 0.971 0.978 0.952	1.029 0.957 0.968 0.930	1.027 0.946 0.960 0.912	
1.054 0.966 0.985 0.955	1.045 0.965 0.983 0.952	1.049 0.965 0.983 0.952	1.055 0.965 0.982 0.952	1.044 0.961 0.977 0.942	1.036 0.954 0.970 0.927	1.024 0.948 0.962 0.914	1.017 0.944 0.956 0.903	1.013 0.945 0.953 0.897

FIG. 2.10b. Relative power distribution.

Temperature coefficients and boron worth values are presented in Table 2.6.

TABLE 2.6. TEMPERATURE COEFFICIENTS AND BORON WORTH ($\times 10^{-4}$)

	0 GWd/T			60 GWd/t		
	MTC[a]	DC[b]	BW[c]	MTC	DC	BW
Russian Federation	-3.500	-0.280	-0.380	-1.5	-0.360	-1.100
Japan	-2.696	-0.283	-0.341	-0.969	-0.378	-0.864
Republic of Korea	-3.200	-0.311	-0.408	-1.289	-0.397	-1.125
Israel	-3.333	-0.292	-0.400	-1.142	-0.477	-1.119

$MTC = \frac{\Delta K}{K_1 * K_2 * \Delta T_m}$ - Moderator temperature coefficient, T_m – moderator temperature.

$DC = \frac{\Delta K}{K_1 * K_2 * \Delta T_f}$ - Doppler coefficient, T_f – fuel temperature.

$BW = \frac{\Delta K}{K_1 * K_2 * \Delta C}$ - Soluble boron worth, C – boron concentration in ppm.

Note 1: All temperature coefficients are negative, the burnup dependence, i.e., plutonium depletion effect is correct.

Note 2: All BOL values show reasonable agreement, the divergence of the EOL values may be attributed to different plutonium concentrations.

2.2. EVALUATION OF THE POTENTIAL OF LWRs, HTRs, HWRs, AND MSRs FOR PLUTONIUM INCINERATION

2.2.1. Incentives

The aim of the research during the second stage of the CRP was to find fuelling strategies, which – on the basis of proven reactor technology are suitable to incinerate plutonium most effectively on the one hand, and to minimize the amount of plutonium to be disposed, on the other hand. Only plutonium of the first generation, namely typical LWR-plutonium, and weapons plutonium were regarded within scope of this CRP.

Four types of reactors were investigated in view of their potential to burn plutonium, each by one group of countries. Israel, Republic of Korea, Russian Federation, and the USA have done research on LWRs; China, the Netherlands and Germany have studied plutonium burning in Modular HTRs; India studied the respective potential of the PHWR, and Japan that of the MSR.

Two characteristic values – aiming at two different optimization goals — may describe the effectiveness of plutonium incineration in the different reactors:

(1) The amount of plutonium, which is burned per unit of produced electric energy. Maximization of this characteristic optimizes the reduction rate of existing plutonium stockpiles.
(2) The relation between the amount of plutonium, which is burned during the lifetime of the fuel elements, and the amount of plutonium, which is residual in the unloaded fuel. Maximization of this characteristic minimizes the plutonium quantity, which either has to be finally disposed or has to be re-fabricated a second time.

Some initial remarks have to be made concerning the admissibility of a comparison and the assessment of the data presented by the countries participating in this CRP.

Each country did the research on its favorite reactor concept, using its own methods and computer codes as well as its specific database. Although, as already mentioned, benchmark calculations for a PWR-cell and for a PWR-lattice were performed during the first stage of this CRP, the numerical simulation of a complete reactor and of his fuel cycle is a much more complex matter, and the results may well be influenced by the degree of detail of the

numerical modeling. Also, the restrictions of the reactor fuelling in view of safety aspects (power peaking, temperature coefficients, transient behavior) may have been observed by the various groups up to different degrees. Nevertheless, the information resulting from the comparison presented in this report seems to be well appropriate and sufficiently reliable to show the potential of the different reactor concepts incinerating plutonium in an effective manner, and to help the suggesting a demonstration of plutonium burning in a reactor in one of the member countries.

By the time the data presented here was compiled the Netherlands had not yet performed calculations for an entire reactor, but for an HTR-fuel cell only. Thus, general effects and operational constraints resulting from, e.g., neutron leakage, local power peaking, and requirements of reactor control may not be observed. Furthermore, a fuel cell containing only plutonium and no thorium was investigated. Therefore, the data is excluded from the summary tables in order to avoid confusion. Results of this research have been published elsewhere [1].

2.2.2. Results

Tables 2.7 and 2.8 show the results of the research work, which has been done based on LWR plutonium and on weapons grade plutonium, respectively. Blank spaces indicate that results have not yet been evaluated so far. The thermal efficiency 0.33 for water cooled reactors and a value 0.4 for HTR and for MSR has been assumed when compiling these tables. Two countries (Russian Federation and Germany) investigated 2 different alternative fuelling strategies each, to be used for their favored reactor concept:

Russian Federation studied the WWER-reactor applying:

a) partial (1/3 of the core) loading with $PuO_2 – ThO_2$ fuel ("Partial Pu-Inv.");
b) loading PuO_2-ThO_2 in the entire core ("Full Pu-Inv.").

Germany in its study made use of the capability of the coated particle fuel of the HTR for a very high heavy metal burnup in order to minimize the amount of residual plutonium in the discharged fuel elements (core "minimal residual plutonium"). The possibility to increase the incineration rate by a reduction of the burnup is demonstrated by the case "increased inciner." (for "LWR-Pu" only, Table 2.7).

The amount of LWR plutonium (Table 2.7), which is burned per unit of produced energy in the different reactor concepts generally decreases with increasing heavy metal burnup of the fuel. In view of the minimization of the residual plutonium in the discharged fuel elements, however, a distinct advantage of the high burnup is obvious. This is indicated by the ratio between the amount of plutonium, which is burned until the fuel element is discharged, and the residual amount, which either has to be disposed or has to be re-fabricated a second time (ratio Pu-burned/Pu-discharged).

Incinerating weapons grade plutonium (Table 2.8) is generally more effective, especially if a high burnup is applied as for PHWR (India) and HTR (China and Germany). The practically complete absence of plutonium isotopes higher than ^{240}Pu in the fresh fuel shifts the occurrence of increased parasitic neutron absorption by ^{242}Pu and by minor actinides to a high burnup of this fuel. Thus, the plutonium can be burned to a high degree, achieving a high yearly destruction rate at the same time. The ratio between burnt and residual plutonium can go up to a value of 5.9 compared to 4.2 in the case of LWR plutonium.

Figure 2.11a and b expressively elucidates these relations. The incineration rate of plutonium (ordinate) is given in units of 200 kg/GW_{el}a, which is approximately the amount of plutonium produced by current LWRs. In other words, this number indicates, how many LWRs units could have their spent fuel plutonium incinerated by one unit of the considered plutonium burner.

2.2.3. Conclusions

— Incinerating LWR plutonium:
Water cooled reactors (LWR and HWR), having a relatively low heavy metal burnup (40-50 GWd/to) reach a large plutonium-incineration rate in the range of about 700-850 kg/GW_{el}a. On the other hand, this amount of incinerated plutonium is small compared to that remaining for disposal or reuse at the end of burnup (Pu burned/Pu discharged equals 0.8-1.7, which corresponds to a residual plutonium fraction in the range of 56 to 37%). Achieving a large heavy metal burnup (e.g., by HTR-fuel) results in a smaller amount of incinerated plutonium per unit of produced energy (500-650 kg/GW_{el}a), but distinctly reduces the fraction of residual plutonium down to 19%.

— Burning weapons grade plutonium:
The higher neutronic value of weapons grade plutonium and the strongly reduced build-up of minor actinides out of ^{242}Pu generally make weapons grade plutonium incineration more effective than for LWR plutonium. While the amount of incinerated weapons plutonium per unit of produced electric energy is comparable to the incineration rate of LWR plutonium, the quantity of residual plutonium can be strongly reduced in case of weapons grade plutonium. The ratio "Pu-burned/Pu-discharged" equals 1.5 to 2 (40 to 33% residual plutonium fraction) for LWRs, about 4 (20% residual plutonium fraction) for the HWR and up to 6 (14% residual plutonium fraction) in case of the HTR. Here, the advantage of reactors having high burnups, becomes obvious once more.

TABLE 2.7. BURNING LWR PLUTONIUM: MASS BALANCE: kg/GW_{el}a (FULL POWER)

	China	Germany (HTR)	India (PHWR)	Israel +USA (LWR)	Japan (MSR)	Republic of Korea (LWR)	Russian Federation (LWR)		
		minimal resid – plutonium	increased inciner.					Partial Pu-Inv.	Full Pu-Inv.
U-235/ U-233 charged	-	624/0	578/0				6/0	612 / 0	
Pu-charged	2521	615	929	1098	1419	1435	1708	519	1803
Pu-discharged	1576	119	288	405	614	435	875	401	953
Pu-burned	945	496	641	693	805	1000	833	117	850
Ratio Pu-burned/ Pu-discharged	0.6	4.2	2.2	1.7	1.3	2.3	0.95	0.29	0.89
U-233 produced		116	161	286		141	366	100	291
Average HM burnup (MWd/kg)		192	128	46		100	40	41	40

TABLE 2.8. BURNING WEAPONS GRADE PLUTONIUM: MASS BALANCE: kg/GW_{el}a (FULL POWER)

	China (HTR) (Case "11g")	Germany (HTR)	India (PHWR)	Israel + USA (LWR)	Japan (MSR)	Republic of Korea (LWR)	Russian Federation (LWR)	
							Partial Pu-Inv.	Full Pu-Inv.
U-235/ U-233 charged	-	188	-	-		7	612	-
Pu-charged	1097	820	725	1095	1425	1264	354	1220
Pu-discharged	212	118	141	361	521	507	266	462
Pu-burned	885	702	584	734	904	757	87	758
Ratio Pu-burned/ Pu-discharged	4.2	5.9	4.1	2.0	1.7	1.5	0.33	1.64
U-233 produced	207	151	204		233	397	100	294
Average HM burnup (MWd/kg)	103	128	70		100	40	41	41

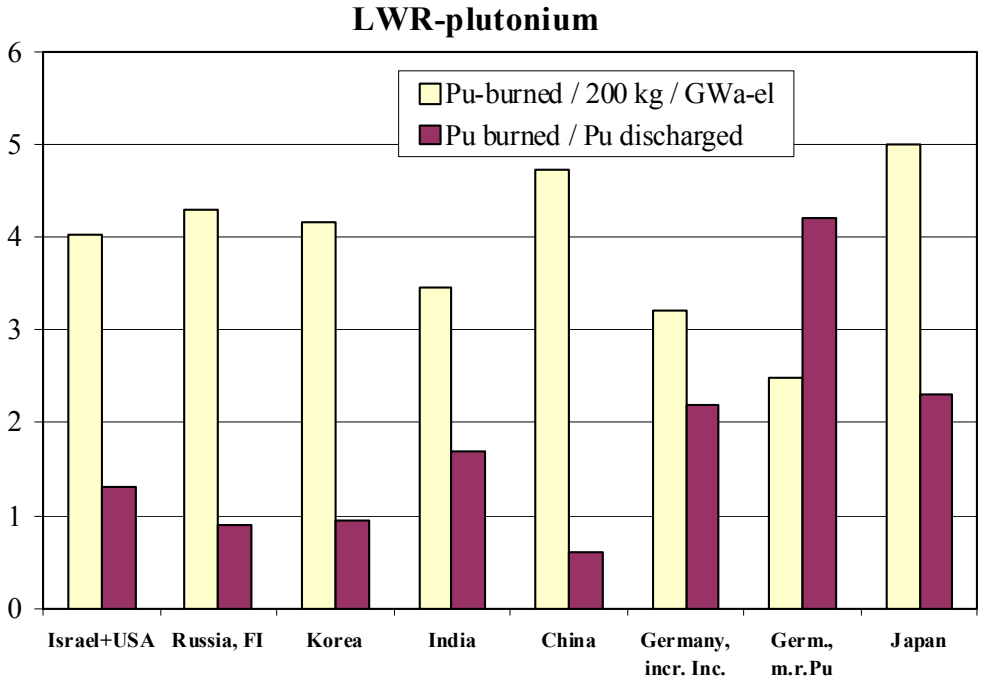


FIG. 2.11a. Burning LWR-grade plutonium.

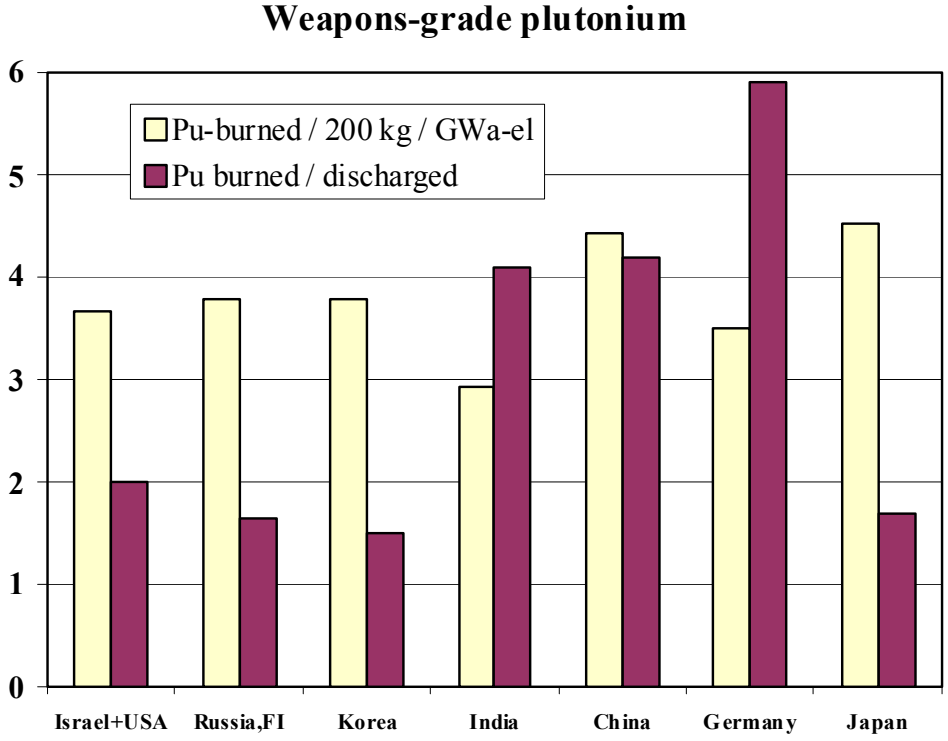


FIG. 2.11b. Burning weapons grade plutonium.

2.3. EFFECT OF PLUTONIUM INCINERATION ON THE TOXICITY OF DISPOSED NUCLEAR WASTE

2.3.1. Incentives and database

The research reported in Section 2.2 primarily aims at the minimization of the proliferation risk by minimizing the plutonium production and maximizing the plutonium incineration. The question still remains, whether and to which degree the incineration of plutonium furthermore is an appropriate tool to significantly reduce the hazard potential of the nuclear waste, which in the end remains for final disposal.

The most common procedure in order to assess the toxicity of nuclear materials is based on the recommendations of the *International Commission on Radiological Protection*, defining "Annual Limits on Intake" for the radio-toxic isotopes. Recent recommendations are given in Sv/Bq and are called "Dose Coefficients for Intake (DCI)". It was agreed to use the values according to *ICRP Publications 68 /*ICRP 1994*/ and 61 /*ICRP 1991*/* as the common database within the frame of this CRP. A comparison between the waste of uranium-fuelled LWRs (providing no reprocessing of the discharged fuel) on the one hand and the waste produced at a scenario applying plutonium burning reactors on the other hand helps to assess the related effect on the toxicity.

2.3.2. Toxicity benchmark

In order to assure that the computer codes used in the different countries have been correctly updated in the sense of the statements of Section 2.3.1, the first step of this evaluation was a benchmark with respect to the toxicity of the spent fuel resulting from one year operation of a $1GW_{el}$ reference-PWR. The composition of the unloaded heavy metal isotopes is defined in Table 2.9.

TABLE 2.9. DISCHARGE RATE OF HEAVY METAL ISOTOPES (kg/year) FOR A TYPICAL PWR*

U-234	4.51E 00
U-235	2.70E 02
U-236	1.07E 02
U-237	2.70E-01
U-238	2.69E 04
Np-237	1.11E 01
Np-239	2.27E 00
Pu-238	3.12E 00
Pu-239	1.43E 02
Pu-240	5.78E 01
Pu-241	3.11E 01
Pu-242	1.02E 01
Am-241	8.87E-01
Am-242m	1.66E-02
Am-242	2.12E-03
Am-243	1.77E 00
Am-244	6.41E-05
Cm-242	2.38E-01
Cm-243	5.17E-03
Cm-244	4.56E-01
Cm-245	1.66E-02
Σ	2.75E 04

* All data normalized to 1000 MW_{el} the electric power output and 300 full-power days.

The tasks to be commonly performed then were to evaluate:

— The ingestion hazard of the complete heavy metal waste;
— The inhalation hazard of the complete heavy metal waste;
— The ingestion hazard of the heavy metal waste remaining after separation of 99% of all plutonium isotopes;
— The inhalation hazard of the heavy metal waste remaining after separation of 99% of all plutonium isotopes.

Dose coefficients of intake (DCI) for the heavy metal isotopes and for the fission products, to be commonly used, are given in Table 2.10.

TABLE 2.10. DOSE COEFFICIENTS OF INTAKE (DCI)

Unit: Sv/Bq
NUCL: Isotope identification number = Z×10000 + W×10 + IS, with
Z: the atomic number
W: the atomic weight
IS: equal 0 or 1 for ground or metastable state, respectively
DCI-W: DCI value for water
DCI-A: DCI value for air

References:
ICRP Publication 68 (1994)
ICRP Publication 61 (1991), DCI calculated for reference dosis 20 mSv/a

NUCL	DCI-W	DCI-A	NUCL	DCI-W	DCI-A	NUCL	DCI-W	DCI-A
20040	0.E+00	0.E+00	882250	1.E-07	6.E-06	942360	9.E-08	2.E-05
812070	5.E-10	5.E-09	882260	3.E-07	2.E-05	942380	2.E-07	4.E-05
812080	5.E-10	5.E-09	882280	7.E-07	3.E-06	942390	3.E-07	5.E-05
812090	5.E-10	5.E-09	892250	2.E-08	8.E-06	942400	3.E-07	5.E-05
822060	0.E+00	0.E+00	892270	1.E-06	6.E-04	942410	5.E-09	9.E-07
822070	0.E+00	0.E+00	892280	4.E-10	3.E-08	942420	2.E-07	4.E-05
822080	0.E+00	0.E+00	902270	9.E-09	1.E-05	942430	9.E-11	1.E-10
822090	6.E-11	3.E-11	902280	7.E-08	4.E-05	942440	2.E-07	4.E-05
822100	7.E-07	1.E-06	902290	5.E-07	1.E-04	942450	7.E-10	7.E-10
822110	2.E-10	6.E-09	902300	2.E-07	4.E-05	952410	2.E-07	4.E-05
822120	6.E-09	3.E-08	902310	3.E-10	4.E-10	952421	2.E-07	4.E-05
822140	1.E-10	5.E-09	902320	2.E-07	4.E-05	952420	3.E-10	2.E-08
832090	0.E+00	0.E+00	902330	5.E-10	5.E-09	952430	2.E-07	4.E-05
832100	1.E-09	8.E-08	902340	3.E-09	7.E-09	952440	5.E-10	2.E-09
832110	9.E-10	3.E-08	912310	7.E-07	1.E-04	952450	6.E-11	8.E-11
832120	3.E-10	4.E-08	912320	7.E-10	1.E-08	962420	1.E-08	5.E-06
832130	2.E-10	4.E-08	912330	9.E-10	4.E-09	962430	2.E-07	3.E-05
832140	1.E-10	2.E-08	912341	5.E-10	5.E-09	962440	1.E-07	3.E-05
842100	2.E-07	3.E-06	912340	5.E-10	6.E-10	962450	2.E-07	4.E-05
842110	9.E-10	3.E-08	922320	3.E-07	4.E-05	962460	2.E-07	4.E-05
842120	9.E-10	3.E-08	922330	5.E-08	9.E-06	962470	2.E-07	4.E-05
842130	9.E-10	3.E-08	922340	5.E-08	9.E-06	962480	8.E-07	1.E-04
842140	9.E-10	3.E-08	922350	5.E-08	8.E-06	962490	3.E-11	5.E-11
842150	9.E-10	3.E-08	922360	5.E-08	8.E-06	962500	4.E-06	8.E-04
842160	9.E-10	3.E-08	922370	8.E-10	2.E-09	972490	1.E-09	2.E-07
842180	9.E-10	3.E-08	922380	4.E-08	7.E-06	972500	1.E-10	1.E-09
852170	9.E-10	3.E-08	922390	3.E-11	4.E-11	982490	4.E-07	7.E-05
862190	3.E-11	7.E-11	922400	1.E-09	8.E-10	982500	2.E-07	3.E-05
862200	3.E-11	7.E-11	932360	2.E-08	3.E-06	982510	4.E-07	7.E-05
862220	3.E-11	7.E-12	932370	1.E-07	2.E-05	982520	9.E-08	2.E-05
872210	9.E-10	3.E-08	932380	9.E-10	2.E-09	982530	1.E-09	1.E-06
872230	2.E-09	1.E-09	932390	8.E-10	1.E-09	982540	4.E-07	4.E-05
882230	1.E-07	7.E-06	932401	5.E-10	5.E-09	992530	6.E-09	3.E-06
882240	7.E-08	3.E-06	932400	8.E-11	1.E-10			

On this basis the benchmark definition was:

— Calculate the total toxicity of the given heavy metals and of their daughter products for a decay period ranging from 10^0 years through 10^6 years;

— Exclude the bulk of plutonium and its daughter products from the calculation by a reduction of all initial plutonium isotopes by the factor 0.01. Calculate the remaining toxicity.

The results of the benchmark are displayed in Figs 2.12 and 2.13. The ingestion hazard and the inhalation hazard, respectively, resulting from the given initial isotope mixture, is plotted as a function of the decay time. The agreement between the participants is quite satisfying except that Israel+USA generally evaluate somewhat higher toxicity values for a decay time of about 10^6 years and longer. However – as can be seen from the figures – this fact does not appear to be important for the assessment of plutonium separation, because the impact of plutonium and its daughter nuclides on the toxicity of the waste tends to vanish at such a decay time, anyway.

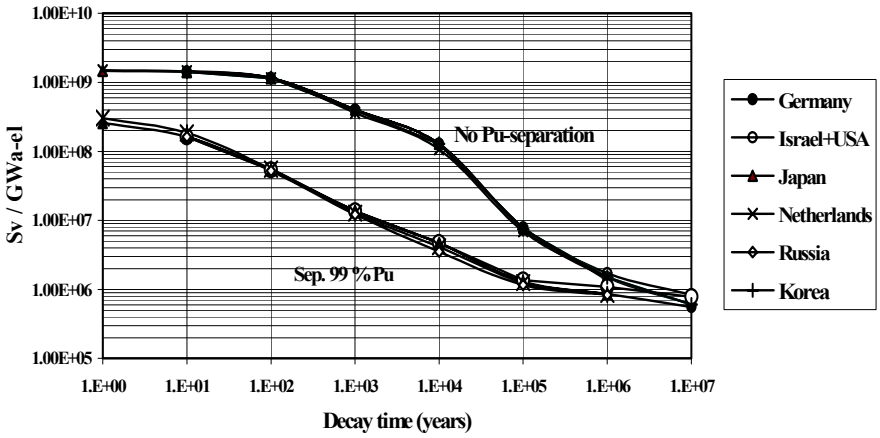


FIG. 2.12. Benchmark: Ingestion hazard of the heavy metal isotopes.

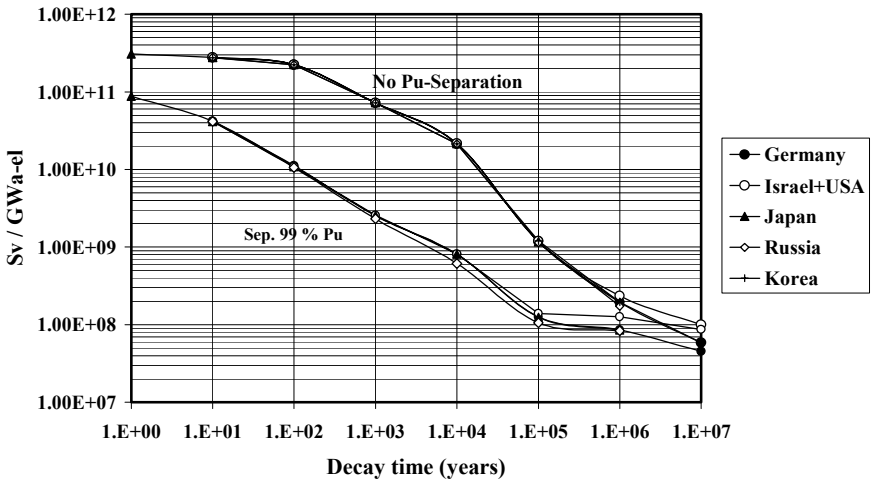


FIG. 2.13. Benchmark: Inhalation hazard of the heavy metal isotopes.

2.3.3. Possible reduction of the waste radio-toxicity

In the realistic scenario of the worldwide population of nuclear reactors we are faced with an increasing number of reactors producing plutonium at an increasing yearly amount. Once a decision will be made, to build plutonium incinerators, the yearly rate of incinerated plutonium would have to increase with time, too. In order not only to compensate future production of plutonium but to also reduce current stockpiles, the number of incinerators would have to increase as rapidly as possible until an equilibrium between plutonium production in uranium-fuelled reactors, on the one hand, and plutonium burning in thorium/plutonium-fuelled reactors, on the other hand, could be reached. Switching over from uranium to thorium in as many existing reactors and as soon as possible could limit the necessary number of plutonium incinerators.

Future scenarios are hard to evaluate. Therefore, we restricted ourselves in the CRP to a relatively simple model, comparing two alternative reactor scenarios:

Scenario 1:

Assume a given reactor population which we name "conventional reactor"(e.g., typical PWRs). These are all operated by use of uranium fuel, and their waste is disposed without separation of any isotopes. These reactors produce a certain, yearly amount of radio-toxicity, which we measure in $Sv/GW_{el}a$.

Scenario 2:

Alternatively, we assume a nuclear scenario of the same size (in GW), where we have an equilibrium combination of "conventional", uranium fuelled reactors and of a certain fraction of "plutonium incinerators" using thorium based fuel. The principle mass flow of the fuel is illustrated in Fig. 2.14.

The bulk waste of "conventional" reactors is disposed and it produces a certain, yearly amount of radio-toxicity (Tox. (a)). 99% of the unloaded plutonium, however, are separated to be used as feed fissile material (or part of it, respectively) of the "plutonium incinerator". The spent fuel of these burners is finally disposed, representing another yearly amount of radio-toxicity, Tox. (b). We add the amounts of toxicity disposed from both the "producers" and the "burners", normalized to 1 $GW_{el}a$.

The sharing of the power production between the two reactor types depends on the amount of plutonium, which the respective incinerator (according to each country's proposal) is able to load each year:

PDR: the plutonium discharge rate of one "conventional reactor, commonly assumed to be equal 245 $kg/GW_{el}a$ in this study;

PCR: the plutonium charge rate of the regarded plutonium burner;

X: at equilibrium, number, the power share of plutonium producers;

Y: at equilibrium, the power share of plutonium incinerators, Y=1-X.

Then it follows (Fig. 2.14):

$$X = \frac{PCR}{PCR + PDR}$$

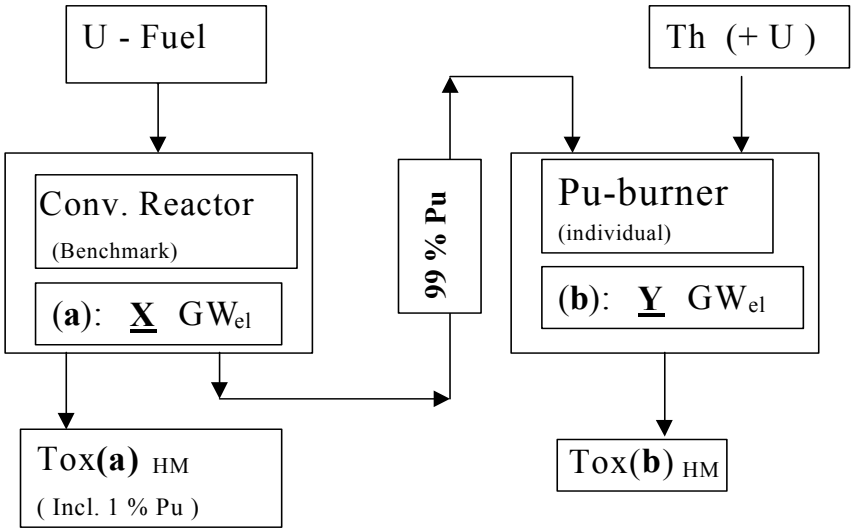


FIG. 2.14. Symbiosis of plutonium producers and plutonium burners.

The comparison of the radio-toxicity produced by this combined system of reactors (Tox.(a) + Tox.(b)) to the toxicity produced by scenario 1 is an indication of the potential of plutonium incineration with the help of thorium fuelled reactors, not only in view of the proliferation concern, but also in order to reduce the long term waste radio-toxicity.

2.3.4. Results and conclusions

Figures 2.15 and 2.16 elucidate the principal impact of plutonium incineration on the amount and on the time dependence of the waste radio-toxicity. Here, the German results (HTR, increased incineration) have been taken as an example; all other contributions show similar characteristics.

A certain reduction of the radio-toxicity occurs at a decay time of the waste between some 10^2 and $5x10^4$ years as the consequence of the strongly reduced amount of plutonium and of its daughter nuclides in the disposed waste. However, during the first two decades, the radio-toxicity of the heavy metal fuel is even increased due to the highly toxic minor actinides, which are produced by neutron captures in ^{242}Pu (since plutonium is being recycled). This is true for the ingestion hazard (Fig. 2.15), as well as for the inhalation hazard (Fig. 2.16).

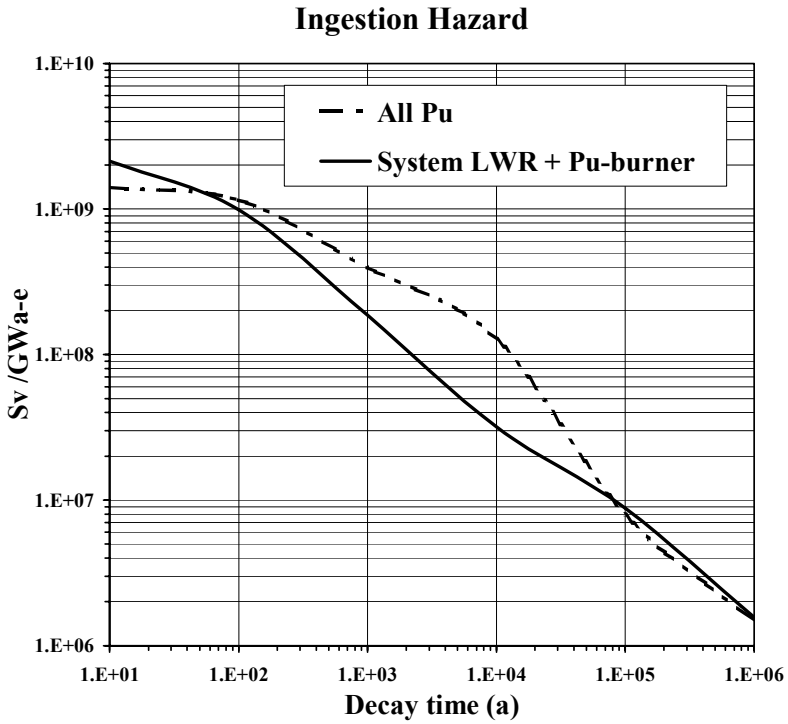


FIG. 2.15. Comparison: ingestion hazard of heavy metal waste with and without plutonium incineration (example, German burner variant).

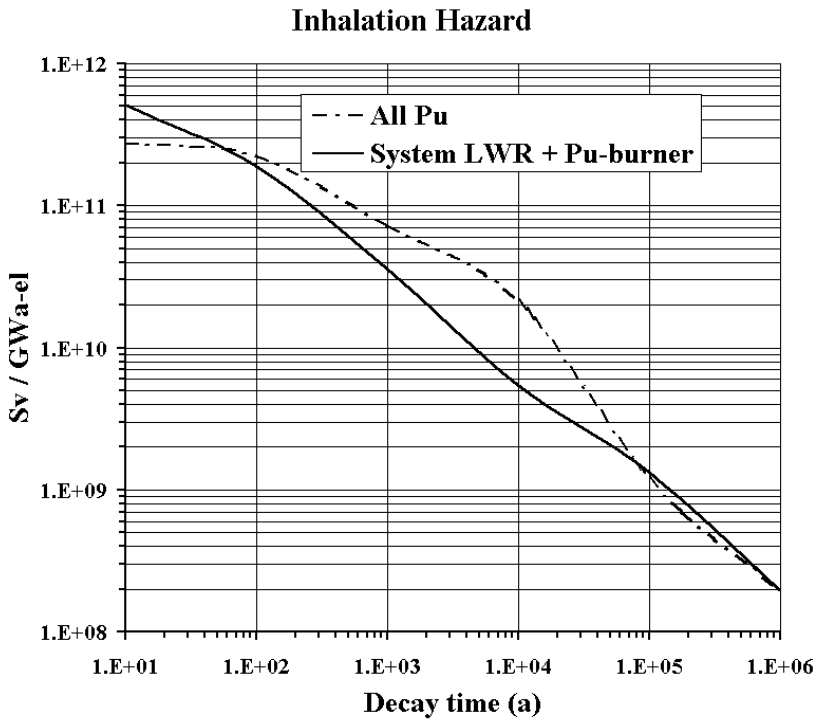


FIG. 2.16. Comparison: inhalation hazard of heavy metal waste with and without plutonium incineration (example, German burner variant).

Figures 2.17 and 2.18 give a comparison of the investigation results, which have been worked out by the participating countries, making use of their individual, favored reactor concept for plutonium incineration. A "Decontamination-Factor (DF)" is plotted versus the decay time. The factor DF is equal to the radio-toxicity of the heavy metal waste, which appears after the (partial) incineration of the LWR plutonium in the various burner concepts, divided by the radio-toxicity resulting from direct disposal of the heavy metal waste. A cubic fit of the calculated values has been used as a guide to the eyes.

With the exception of the common results of Israel and USA, the results of all participants show the same principal effects:

— For the first decades after disposal, the radio-toxicity of the waste is increased (up to a factor ≈ 2). It is obvious, that the initial increase of the radio-toxicity is the more distinct, the higher the heavy metal burnup of the plutonium incinerator (particularly HTR and MSR) is.

— The radio-toxicity is also increased at decay times larger than 10^5 years (up to a factor ≈ 2).

— It is decreased for the period between about 10^2 years and about $0.5–1.0 \times 10^5$ years by at maximum the factor 2 to 4. Here, the high burnup achieved in the HTR causes by far the strongest reduction.

A reduction of the waste radio-toxicity by an order of magnitude or more seems not to be achievable by any of the considered concepts.

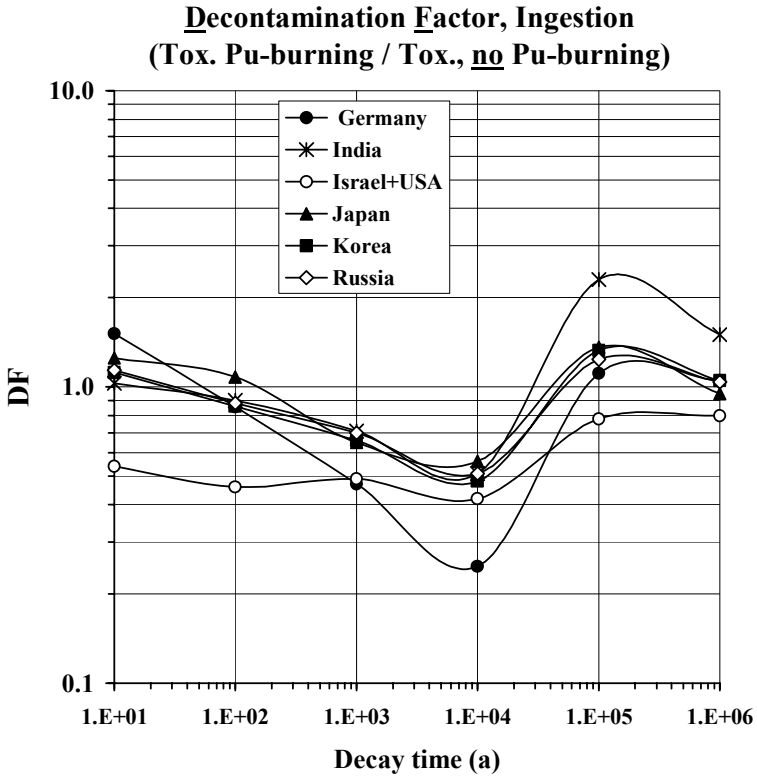


FIG. 2.17. Relative change of the ingestion hazard of heavy metal waste achieved by plutonium incineration in the various burner concepts.

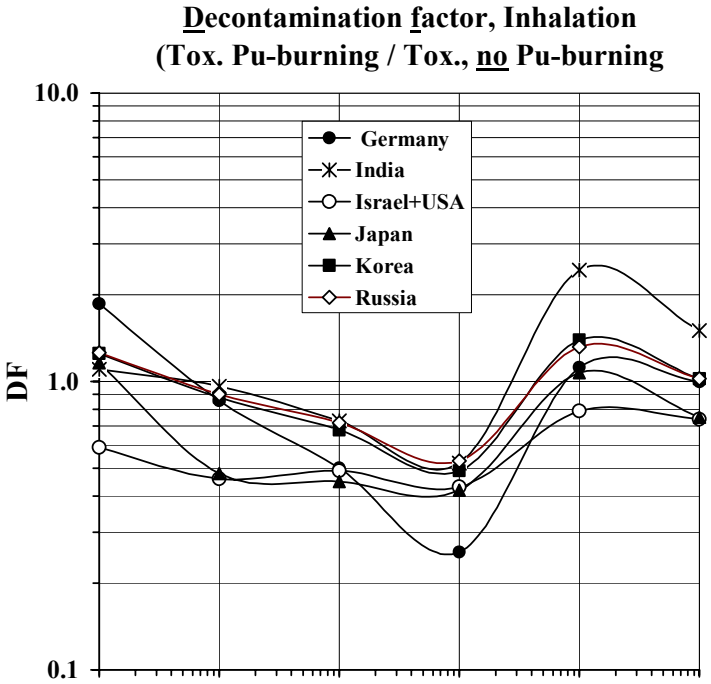


FIG. 2.18. Relative change of the inhalation hazard of heavy metal waste achieve by plutonium incineration in the various burner concepts.

2.4. CONCLUSIONS

In the course of the IAEA CRP, participation from different Member States performed three benchmark tasks for different reactor concepts. Their incentive was the comparison of the various methods and nuclear database. The assessment of thorium fuelled thermal reactors in view of their potential for the incineration of plutonium and for a possible, combined reduction of the waste radio-toxicity has been performed. The agreement of the benchmark results generally was very satisfying. The results obtained are very satisfactory and deemed to constitute a sufficiently reliable basis for overall conclusions on the potential of thorium-based cycles to constrain plutonium and to reduce the **long germ(? term)** potential radiotoxic hazard of the waste.

The incineration of two kinds of plutonium has been considered during the CRP:

— reactor grade plutonium of first generation, i.e., plutonium typically discharged from current reactors;

— Weapons grade plutonium.

Plutonium utilization may be looked at from the point of view of two main optimization goals: Firstly, to achieve a large incineration rate in relation to the amount of produced electric energy, and, secondly, to minimize the amount of plutonium, which is still residual in the discharged fuel elements after their use in a plutonium incinerating reactor. For each of the two plutonium types, the overall conclusions to be drown with regard to both optimization goals can be summarized as follows:

— LWR plutonium:
Water cooled reactors (LWR and HWR), characterized by a relatively low heavy metal burnup, attain a large plutonium incineration rate. On the other hand, the amount of incinerated plutonium is low compared to the remaining plutonium inventory in the spent fuel (to be disposed off or recycled). Achieving a large heavy metal burnup (e.g., in HTR-fuel) results in a smaller amount of incinerated plutonium per unit of produced energy, but distinctly reduces the fraction of residual plutonium. Typically, a plutonium incinerator burns approximately 2.5 to 4 times the amount of plutonium, which is produced by an LWR of the same power.

— Weapons grade plutonium:
The higher neutronic value of weapons grade plutonium, and the strongly reduced buildup of minor actinides starting from 242Pu, generally makes its incineration more effective than in the case of LWR plutonium. While the amount of weapons grade plutonium, which is burned per unit of produced electric energy, is comparable to the incineration rate of LWR plutonium, the quantity of residual plutonium can be strongly reduced in the case of weapons grade plutonium. In view of the minimization of the amount of plutonium remaining for final disposal, there is a clear advantage for reactors having an especially high heavy metal burnup.

Generally, there is a remarkable potential to effectively constrain the production of plutonium and to reduce existing plutonium stockpiles by implementing the thorium fuel cycle in a large number of current reactors.

This path offers a promising near-future plutonium. However, plutonium incineration in thermal reactors turns out to be less effective from the point of view of the reduction of the long term radio-toxicity of the nuclear waste. A reduction by an order of magnitude or more of the potential long term radiotoxic hazard of the waste seems not to be achievable by any of the considered plutonium incinerating thermal reactors. Most of the calculations performed for LWR plutonium indicate that the waste radio-toxicity will be decreased by not more than a factor of 2 to 4, and only for the period between approximately long times after disposal. The waste radio-toxicity is even increased during the first decades and for extremely long times after disposal.

REFERENCES TO SECTION 2

[1] CHERNOCK, W., HORTON, K.E., Status of liquid metal cooled fast reactor development in the USA, IAEA-TECDOC-791, Vienna (1995).

[2] MICHELBACHER, J.A., HENSLEE, S.P., MCDERMOTT, M., ROSENBERG, K.E., PRICE, J.R., WELLS, P.B., The sodium process facility at Argonne National Laboratory-West, paper presented in the Consultancy on Technical options for the decommissioning of the BN-350 LMFR, 23–27 February 1998, Obninsk, Russian Federation.

[3] BENDE, E.E., Unit-Cell Calculations for Plutonium fuelled HTR Pebbles, ECN-R-98-023 (1998).

[4] ICRP 1994, Dose Coefficients for Intake of Radionuclides by Workers, Replacement of ICRP Publication 61, ICRP Publication 68, Annals of the ICRP 24, No 4 (1994).

[5] ICRP 1991, Annual Limits on Intake of Radionuclides by Workers based on the 1990 Recommendations, ICRP Publication 61, Annals of the ICRP 22, No 4 (1991).

3. INDIVIDUAL CONTRIBUTIONS OF THE VARIOUS COUNTRIES

3.1. CHINA

3.1.1. Study of thorium fuel cycles burning weapons grade and civil grade plutonium in the Module-HTR

3.1.1.1. Introduction

Since development work on the 200 MW-MODULE pebble bed reactors began in 1979 in Germany, the HTR module is considered as one kind of the advanced nuclear reactors with completely passive safety properties [1, 2]. At any accident the release of radioactivity in HTR Module is prohibited, without putting technical safety equipment into operation.

The coating of the coated particle embedded in the fuel elements does not permit any radioactivity, radioactive gaseous or metallic fission products, to escape from intact fuel particles up to a temperature of 1600ºC.

Residual heat can be removed from the core even under extreme accidental conditions by means of passive heat transfer processes based on natural laws, such as heat conduction and radiation. HTR Module has negative reactivity temperature coefficient. Therefore the core temperature raise can offset reactivity increase as a result of reactivity accidents.

The spherical fuel elements are used for the HTR Module. Because fuel elements are able to receive a very great variety of fuel cycles, that ball permits really a wide flexibility in the conception of the reactor.

Thorium based fuel cycle in the HTR Module would produce a small amount of toxic fuel waste or long lived radiotoxic waste. In order to reduce plutonium stockpiles, Pu (239 and 241) is used for thorium fuel cycle as fissile material in HTR Module.

3.1.1.2. HTR Module and calculation

Main design data are of the HTR-Module listed in Table 3.1. Power density 3 MW/m^3 and reactor dimensions have been harmonized to provide sufficiently high passive removal of the decay heat at loss of coolant, thus keeping the fuel temperature below 1600ºC. The weight of heavy metal and enrichment in the sphere are optimized for burning as much plutonium as possible, and keeping negative temperature coefficient.

Under normal operation the temperature for spectrum calculation is listed Table 3.2. The VSOP code [3] is used for calculation of HTR Module. The reactor is divided into eleven spectrum zones. The pebble bed is divided into five spectrum zones.

The thorium absorption cross-sections of resolved and unresolved resonances are generated by ZUT-DGL code basing on resonance data.

TABLE 3.1.1. MAIN DESIGN DATA OF THE HTR MODULE

Reactor core	
Thermal power, MW	200
Power density, MW/m^3	3
Core height/diameter, m	9.43/3.0
Heating of helium, ºC	250→700
Helium pressure, bar	60
Helium mass flow rate, kg/s	85.4
Fuel element	
Diameter of pebble, cm	6
Diameter of fuel zone, cm	5
Density of graphite in the matrix and outer shell, g/cm^3	1.75
Volumetric filling faction of elements	0.61
Number of passes of spheres through the core	10
Coated particle	
Radius of the kernel, cm	0.025
Fuel composition	PuO_2-ThO_2
Density of the kernel, g/cm^3	10.5
Isotopic composition of weapons grade plutonium, %	Pu-239/Pu-240 = 94/6
Isotopic composition of civil grade plutonium, %	Pu-239/Pu-240/Pu-241/Pu-242 = 62.63/24.24/8.08/5.05
Coating layers	C/C/SIC/C
Density, g/cm^3	1.05/1.90/3.18/1.90
Thickness, cm	0.009/0.004/0.0035/0.0035

TABLE 3.1.2. TEMPERATURE OF SPECTRUM ZONES

Zone/cm		Temperature/ºC
0<R<150	305<Z<493.6	Fuel: 340.90 moderator: 331.21
0<R<150	493.6<Z<682.2+	Fuel: 456.41 moderator: 442.44
0<R<150	688.2<Z<870.8	Fuel: 569.58 moderator: 554.95
0<R<150	870.8<Z<1059.4	Fuel: 658.30 moderator: 645.94
0<R<150	1059.4<Z<1248	Fuel: 711.62 moderator: 703.37
0<R<250	0<Z<255	192.28
0<R<150	255<Z<305	264.71
150<R<162	155<Z<1248	427.62
162<R<250	155<Z<1248	260.30
0<R<150	1248<Z<1518	696.24
0<R<150	1518<Z<1693	342.28
150<R<250	1248<Z<1693	342.28

3.1.1.3. Calculation results

The equilibrium core is calculated for the case of the different heavy metal loading and enrichment.

3.1.1.3.1. Burning weapons grade plutonium

The pebble bed reactor operates by continuous loading and discharging of fuel elements. Therefore a little excess reactivity is required in normal operation.

In order that the reactor has negative moderator temperature coefficient, heavy metal loading in sphere is increased gradually and enrichment is adjusted so that k_{eff} value of core is around 1.01. The results for heavy metal weight in sphere 7, 9, 11, and 13 g are given in Tables 3.1.3 and 3.1.4.

Table 3.1.3 shows that as heavy metal weight in sphere increases the enrichment should increase for retaining a close k_{eff} value, the conversion ratio increases; moderator temperature coefficient changes from positive to negative.

TABLE 3.1.3. MAIN PERFORMANCE FOR DIFFERENT HEAVY METAL LOADING

Heavy metal loading	g/ball	7	9	11	13
Enrichment	%	9.6	10	11	12
Burnup	MWd/T_{HM}	100003	100140	100201	100173
Fuel element residence time	Days	1258	1618	1977	2337
Conversion ratio		0.513	0.529	0.542	0.560
Power peaking max./average		4.15	3.54	3.09	2.85
Max. power per ball	KW/ball	2.31	1.97	1.72	1.59
Core leakage	%	9.93	8.95	8.27	7.89
Neutron flux	$E+14/(cm^2 \times s)$				
Average thermal flux	(<1.86ev)	0.4068	0.2662	0.1720	0.1247
Average fast flux	(>0.1Mev)	0.2384	0.2358	0.2340	0.2353
Average total flux		1.1896	1.0336	0.9260	0.8748
Temperature coefficient	($\Delta k/k/^oC$)				
Fuel	(10^{-5})	-1.82	-2.25	-2.62	-2.99
Moderator	(10^{-5})	5.10	1.73	-0.876	-1.99
Reflector	(10^{-6})	2.79	2.36	2.00	1.79

TABLE 3.1.4. INVENTORY, SUPPLY AND DISCHARGE OF MAIN ISOTOPE

Heavy metal loading	g/ball	7	9	11	13
Inventory	(kg/GWth)				
U-233		137.19	192.99	253.96	322.10
Pu-239		232.32	383.72	686.99	1077.73
Pu-241		101.48	158.62	243.05	327.11
Th-232		11050.10	14119.23	17030.09	19838.69
Pu-240		133.80	162.28	203.70	243.68
Pu-242		45.64	52.89	53.71	53.98
Supply-Discharge	(kg/GWDth)				
U-233		0-0.1890	0-0.2065	0-0.2244	0-0.2405
Pu-239		0.9873–0.0055	1.0278–0.0156	1.1298–0.0533	1.2314–0.1128
Pu-241		0-0.0447	0-0.0655	0-0.1089	0-0.1441
Th-232		8.9632–8.6098	8.9150–8.5488	8.8032–8.4311	8.6889–8.3002
Pu-240		0.0633–0.0249	0.0659–0.0258	0.0724–0.0364	0.0789–0.0454
Pu-242		0-0.0663	0-0.0621	0-0.0557	0-0.0493
Consumption of Pu-239	(kg/GWDth)	0.9818	1.0122	1.0765	1.1186

Table 3.1.4 shows that with increment of heavy metal loading per sphere weight of U-233 and Pu-239 in equilibrium code increases, weight of U-233 and Pu-239 in discharged spheres increases, consumption of Pu-239 increases.

3.1.1.3.2. Burning civil grade plutonium

The computational results of heavy metal loading 7 g sphere for enrichment 10, 11, and 12% are given in Tables 3.1.5 and 3.1.6.

TABLE 3.1.5. MAIN PERFORMANCE FOR DIFFERENT ENRICHMENT WITH CIVIL GRADE PLUTONIUM

Enrichment	%	10	11	12
k_{eff}		1.00027	1.00528	1.00707
Burnup	MWd/T_{HM}	100116	100117	100095
Fuel element residence time	days	1258	1258	1258
Conversion ratio		0.599	0.598	0.598
Power peaking max./average		2.52	2.34	2.22
Max. power per ball	KW/ball	1.40	1.30	1.23
Core leakage	%	8.66	8.38	8.17
Neutron flux				
Average thermal flux	$E+14/(cm^2s)$	0.2532	0.2146	0.1880
Average fast flux	(<1.86ev)	0.2425	0.2411	0.2405
Average total flux	(>0.1M eV)	1.0462	1.0015	0.9716
Temperature coefficient	($\Delta k/k/^oC$)			
Fuel	(10^{-5})	-1.62	-1.58	-1.54
Moderator	(10^{-5})	-0.598	-1.63	-2.31
Reflector	(10^{-6})	2.28	2.08	1.95

TABLE 3.1.6. INVENTORY, SUPPLY AND DISCHARGE OF MAIN ISOTOPE WITH CIVIL-GRADE PLUTONIUM

Enrichment	%	10	11	12
Inventory	(kg/GWth)			
U-233		127.32	123.31	119.94
Pu-239		354.46	458.24	564.94
Pu-241		262.92	306.76	345.70
Th-232		10576.15	10404.79	10230.77
Pu-240		281.23	323.56	364.03
Pu-242		141.13	143.75	147.34
Supply-discharge	(kg/GWDth)			
U-233		0-0.1871	0-0.1839	0-0.1807
Pu-239		0.9098–0.0419	0.9993–0.0804	1.0895–0.1288
Pu-241		0.1182–0.1534	0.1300–0.2054	0.1417–0.2529
Th-232		8.5523–8.2621	8.4067–8.1352	8.2615–8.0037
Pu-240		0.3533–0.0854	0.3884–0.1165	0.4235–0.1471
Pu-242		0.0742–0.1431	0.0816–0.1449	0.0890–0.1464
Consumption	(kg/GWDth)			
Pu-239		0.8670	0.9189	0.9607
Pu-241		–0.0352	–0.0754	–0.1112

Because weight of ^{240}Pu and ^{241}Pu in reactor with civil grade plutonium is more than that with weapons grade plutonium, its physical performance is different. Table 3.1.5 shows that the moderator temperature coefficient is negative, with increment of enrichment its absolute value increases. Table 3.1.6 shows that with increment of enrichment weight of ^{239}Pu and ^{241}Pu in equilibrium core increase, weight of ^{239}Pu and ^{241}Pu in discharged spheres increase, consumption of ^{239}Pu increases.

3.1.2. Physics studies of energy production and plutonium burning in pebble-bed type high temperature gas cooled module reactor (HTMR)

3.1.2.1. Introduction

Physics studies were done for two HTMRs, one of which is a current energy-producing reactor (CER) with 200 MW of thermal power used to burn uranium, the other is a Plutonium Burning reactor (PBR) with the same power of that of the CER. For PBR, the compositions of the isotopes of plutonium in the fresh fuel element are the same as those in the discharged fuel of CER.

3.1.2.2. Calculation model of HTMR

For CER and PBR, most designed parameters are the same. The differences consist only in the choices of fissile and fertile materials for the two cases.

For the two reactors, the geometric parameters are the same (Section 3.1.1, Fig. 3.1.1). The main data are according to Section 3.1.1, Table 3.1.1. The heavy metal load of the fuel elements is different for CER (7 g) and for PBR (7.33 g). The total power and average power density for both reactors are the same. Under normal operation the temperature for spectrum calculation is listed in Section 3.1.1, Table 3.1.2. The code VSOP [3] were used for the calculations. The reactor was divided into eleven spectrum zones among which five spectrum zones were used for the core region.

The thorium absorption cross-sections of resolved and unresolved resonance were generated by code ZUT-DGL [3].

3.1.2.3. Calculation results

For the CER, ^{235}U was chosen as fissile material and ^{238}U as fertile material. For the PBR, ^{232}Th was chosen as fertile material because thorium based fuel cycle has less HLW compared with uranium/plutonium fuel cycle and can not produce extra plutonium. Because the PBR is used to burn the isotopes of plutonium from the discharged fuel of CER, we did the two designs in this way, firstly the CER, secondly the PBR.

TABLE 3.1.7. PARAMETERS OF FUEL PARTICLE

	CER	PBR
Radius of the kernel, cm	0.025	
Fuel composition	UO2	(^{233}U-Pu-Th)O_2
Density of the kernel, g/cm^3	10.5	10.5
Weight composition of fissile material	^{235}U	$^{233}U/^{239}Pu/^{241}Pu$ = 0.261/0.538/0.20
Weight fractions fissile/fertile	$^{235}U/^{238}U$ = 0.077/0.923	($^{233}U+^{239}Pu+^{241}Pu$)/ ($^{232}Th+^{240}Pu+^{242}Pu$) = 0.172/0.828
Weight compositions of fertile materials	^{232}Th	$^{232}Th/^{240}Pu/^{242}Pu$ = 0.889/0.077/0.034
Coating layers		C/C/SIC/C
Density, g/cm^3		1.05/1.90/3.18/1.90
Thickness, cm		0.009/0.004/0.0035/0.0035

For the convenience of comparison, the data for both reactors are listed in the same table, e.g., Table 3.1.7. This does not mean that the data, e.g. the weight compositions of fissile materials of PBR can be defined at the same time of the definition of those of CER. In fact, the compositions were defined according to those in the discharged fuel of CER listed in Table 3.1.8.

The compositions of the isotopes of plutonium of CER discharged fuel in Table 3.1.4 show that the weight fraction of fissile materials including ^{239}Pu and ^{241}Pu is only 0.581 that is too low to be used as fresh fuel for PBR. Therefore we used plutonium produced by CER plus ^{233}U as fuel of PBR. The detailed data are listed in Table 3.1.3.

We defined Cn as the ratio of plutonium consumption of PBR to the plutonium production of CER, i.e.,:

Cn = consumption of plutonium in PBR / production of plutonium in PBR.

From Table 3.1.8, Cn = 1.036/0.153 = 6.77.

Tables 3.1.9 and 3.1.10, respectively, show the data of fractional fission of heave metals and main characteristics of CER and PBR at equilibrium state.

According to Table 3.1.8, for PBR, the relative burnup in discharged fuel for plutonium isotopes is 37.5% and that for ^{233}U is only 5.81%. The reason why the consumption of ^{233}U is so low is that the conversion ratio of PBR has a high value of 0.62 (Table 3.1.10). From Table 3.1.9 we can see that the fission fraction of plutonium isotopes is 80.8% and about 4 times as much as that of ^{233}U.

In order to maintain the PBR at criticality, the fuel enrichment of PBR is higher than that of CER. This is because that there are ^{240}Pu and ^{242}Pu in PBR, which have larger neutron absorptive micro-cross-sections than ^{238}U and also leads to larger magnitude of conversion ratio for PBR.

TABLE 3.1.8. INVENTORY, SUPPLY AND DISCHARGE OF MAIN ISOTOPES

reactor		CER	PBR
Inventory	(kg/GW_{th})		
^{232}Th			9542.38
^{233}Pa			8.06
^{233}U			564.44
^{234}U			17.12
^{235}U		426.56	1.37
^{236}U		89.54	
^{238}U		11568.96	
^{239}Pu		57.53	764.53
^{240}Pu		29.66	617.63
^{241}Pu		13.91	578.32
^{242}Pu		6.54	353.96
^{237}Np		5.27	
^{243}Am			56.42
Total heavy metal	(kg/GWd_{th})	12197.97	12504.23
Supply-discharge			
^{232}Th			9.285–9.069
^{233}U			0.568–0.535
^{235}U		0.981–0.139	0.000–0.004
^{236}U		0.000–0.130	
^{238}U		11.747–11.253	
^{239}Pu		0.000–0.065	1.168–0.428
^{240}Pu		0.000–0.045	0.807–0.413
^{241}Pu		0.000–0.024	0.437–0.551
^{242}Pu		0.000–0.019	0.351–0.335
Production/consumption	(kg/GWd_{th})	Production	Consumption
Of plutonium		0.153	1.036

TABLE 3.1.9. THE FRACTIONAL FISSION OF HEAVY METALS, %

Heavy metal	CER	PBR
^{232}Th		0.05
^{233}U		19.17
^{235}U	63.99	0.03
^{236}U	0.02	
^{238}U	0.32	
^{239}Pu	28.67	45.63
^{240}Pu	0.01	0.19
^{241}Pu	6.99	34.87
^{242}Pu		0.06

TABLE 3.1.10. MAIN CHARACTERISTICS OF CER AND PBR

reactor		CER	PBR
Fuel enrichment	%	7.7	17.2
Average burnup of discharged fuel	MWd/t_{HM}	80000	80000
Fuel element residence time		1007	1040
Conversion ratio		0.45	0.62
Power peaking max./average		2.66	1.84
Max. power per ball	kW	1.46	1.02
Neutron flux	E+14/(cm^2×s)		
Average thermal flux	(<1.86eV)	0.70	0.12
Average total flux		1.42	0.88

REFERENCES TO SECTION 3.1

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[2] HAAS, K.A., RÜTTEN, H.J., TEUCHERT, E., Rechnerische Darstellung des HTR-Modul-Reaktors, Jül-2618, Forschungszentrum Jülich (1992).

[3] RÜTTEN, H.J., TEUCHERT, E., V.S.O.P.('94) Computer Code System for reactor Physics and Fuel Cycle Simulation, Jül-2897, Forschungszentrum Jülich (1994).

3.2. GERMANY

3.2.1. Introduction

Large amounts of civil plutonium have accumulated in the world in burnt fuel from the operation of nuclear power reactors. Until the year 2000 they will amount to about 1400 mg [1, 2]. Moreover, there are big quantities of military plutonium, which could possibly find a peaceful use by being the source of civil production of electric energy. There seems to be a serious public and political concern about its radiotoxicity [3], and about misuse of this material. If plutonium is recycled in the form of U-Pu-MOX-fuel, separable second generation plutonium is produced. This can be avoided, if plutonium is burned in combination with thorium instead of uranium as fertile material.

Basic research on burning plutonium in a pebble-bed HTR has been reported by M. Khorochev [4]. Based on the results of these investigations, more detailed work had to be carried out since then in order to work out a "preliminary reference strategy" for burning plutonium in a pebble-bed HTR. In parallel, methods and databases have continuously been improved in view of a more precise prediction especially of the generation and depletion of the minor actinides. The aim of the research was to find a fuelling strategy for:

— burning as much plutonium as possible per unit of produced energy;

— minimizing the residual amount of plutonium in the fuel for long term waste disposal.

Only plutonium of the first generation – typical LWR plutonium and weapons grade plutonium has been regarded for the time being.

The second concern of this study then is to evaluate, if and to which degree the incineration of plutonium in the Modular HTR is capable to reduce the entire toxicity of the heavy metal waste, which is to be finally disposed.

3.2.2. Optimization of plutonium incineration in the modular HTR

3.2.2.1. Design of the reactor and of the fuel element

The high temperature pebble-bed reactor (HTR), which has been chosen as the basis for this study, is a close approximation of the HTR-MODUL-reactor, as it was designed in Germany by the INTERATOM company [5, 6]. The small size of this reactor was required in order to avoid a release of fission products from the fuel elements by an inherent restriction of the fuel temperature to less than 1600°C even in case of a complete loss of active cooling. The schematic of the reactor is illustrated in Fig. 3.2.1. The reactor core consists of a statistical filling of about 360 000 pebble-shaped fuel elements, which are circulated about ten times through the core, until they reach their final burnup. It is surrounded by the graphite reflectors, which contain the control devices and the ducts for the supply of helium. In this the cooling gas was heated up in the reactor core from 250 to 750°C.

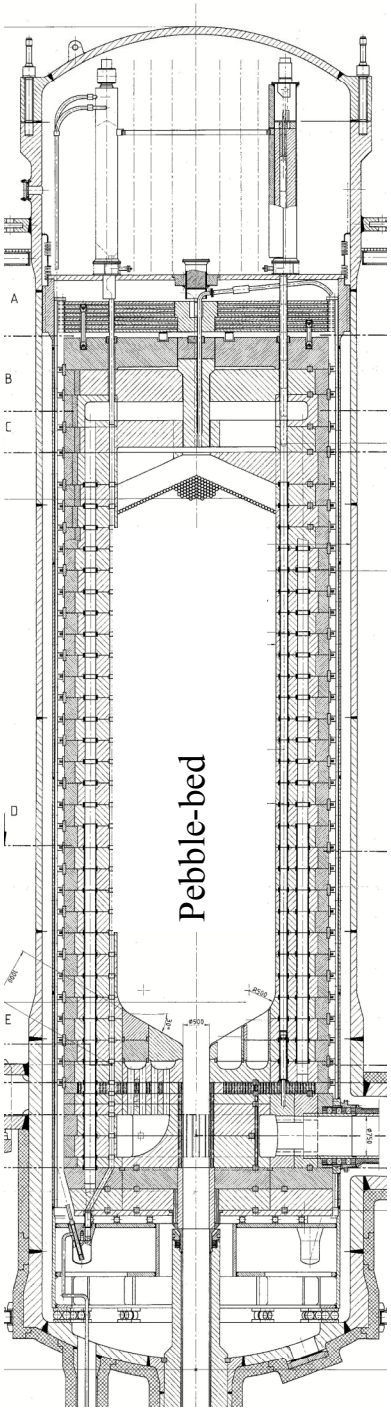

FIG. 3.2.1. Schematic of the HTR-MODUL-200 reactor layout.

On the basis of this reactor concept, development work has been going on since a couple of years for a pebble-bed Modular reactor with a direct helium cycle in South Africa.

Its conceptual design was presented at a Technical Committee Meeting of the International Atomic Energy Agency on High Temperature Gas Cooled Technology Development [7]. According to the requirements of the Brayton Cycle, the cooling gas (helium) has to enter the reactor core with the temperature 550–600°C and to be heated up to 900°C. The larger temperature range between operation and shutdown conditions compared to a steam cycle should be taken into account assessing the general performance of an HTR in view of its possible application for the incineration of plutonium. As the core geometry of the above cited reactor concept has not been finally fixed yet on the one hand and in detail is of minor importance for the effects of isotope burnup on the other hand, we based our research on plutonium burning on the core geometry according to Fig. 3.2.1 in combination with the temperature conditions required for the helium turbine cycle (Table 3.2.1).

TABLE 3.2.1. SPECIFICATION OF THE HTR–MODUL CORE FOR PLUTONIUM BURNING

Thermal power	200 MW
Core height	9.43 m
Core diameter	3.0 m
Average power density	3.0 MW/m^3
Number of fuel elements/m^3	5394
Helium temperature	
Core inlet	600°C
Core outlet	900°C
Helium pressure, Core inlet	60 bar
Fuel element specifications	
Diameter of pebbles	6 cm
Diameter of fuel zone	5 cm
Density of graphite	1.75 g/cm^3
Specification of the coated particles:	
Coating layers	C / C / SiC / C
Density	1.05/1.90/3.18/1.90 g cm^{-3}
Fuel / Diameter of kernel / Maximum burnup:	
PuO_2	0.24 mm 800 MWd/kg
$(U\text{-}Th)O_2$	0.50 mm 120 MWd/kg
Assumed isotope composition of Pu:	
Fraction of	$^{238}Pu/^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu$
(a) LWR grade	0.01 / 0.62 / 0.24 / 0.08 / 0.05
(b) Weapons grade	0. / 0.94 / 0.06 / 0. / 0.

3.2.2.2. Methods and computer codes

The numerical investigations within this study have been performed by means of V.S.O.P(97) [8]. This computer code system has been developed for the comprehensive numerical simulation of the physics of thermal reactors. It includes processing of cross-sections, the setup of the reactor and of the fuel element, repeated neutron spectrum evaluation, neutron diffusion calculation in two or three dimensions, fuel burnup, fuel shuffling, reactor control and thermal hydraulics. The involved cross section libraries have been derived from the evaluated nuclear data files ENDF/B-V and JEF-1. The code can simulate the reactor operation from the initial core towards the equilibrium core.

3.2.2.3. Fuelling strategy

The HTR has some unique features, which make this system especially suitable to burn plutonium on the basis of thorium-fuel:

The fuel is loaded in the form of coated particles, which are embedded in the graphite matrix of the fuel elements. By this way, an exceptionally high heavy metal burnup can be achieved.

The different fuel materials can be loaded into different types of coated particles. In a pebble-bed HTR the different fuel particles – plutonium on the one hand and thorium and uranium on the other hand – can also be loaded into different fuel elements. As they are continuously loaded and disloaded, these elements can even have different numbers of passes through the core – if desired – until they will reach their final burnup. On-line burnup measurements and the discrimination of LEU- vs. thorium fuel, respectively, have been successfully practiced for many years during the AVR reactor in Jülich (FRG) [9–11], it was provided for the THTR-plant (FRG) [12] and for the MODUL-200 HTR plant [13].

The simultaneous use of different coated particle variants was originally developed for the high temperature reactors using prismatic fuel elements, as they were applied in the Fort. St. Vrain plant. Here, the fissile material was used in the form of small, so called “feed” particles reaching a burnup of about 90% of its initial amount. Larger particles — “breed” particles — were made of ThO_2

In the present study the pebble-bed core is fueled by use of a two-pebble-concept (Table 3.2.2).

One type of pebbles (PU-FE) contains PuO_2–coated particles with a diameter of 0.24 mm having a total of 3 g plutonium per pebble. The second pebble type (U/Th-FE) contains 20 g (HEU-Th)O_2 in the form of larger coated particles (diameter 0.5 mm). On the one hand the addition of uranium to the thorium is necessary to sustain criticality — depending on the desired burnup of the fuel —, and, on the other hand, in order to achieve a prompt temperature increase of the resonance absorber, thorium, in case of an increase of the neutron flux, thus causing a prompt negative reactivity feedback. The uranium is highly enriched (93%) in order to minimize the buildup of plutonium. While the uranium/thorium-load of the fuel elements for the THRU-plant was 11g, values between 16 g and 20 g were provided in the frame of the German HUT-project (*H*ochtemperaturreactor mit *H*elium*t*urbine) and the reference maximum burnup was 120 GWd/t [14]. Elements containing up to 30 g heavy metal were developed within the same project [15]. A restriction to 20 g, however, should limit the fraction of coated particles, which will be damaged during the fabrication process of the elements, to 10^{-4} [16]. As in particular the use of high burnup plutonium particles cannot be regarded as proven technology, an irradiation program will be required to:

— demonstrate that a burnup equal about 80% “fissions per initial metal atom” (FIMA) can be achieved for the plutonium (feed) particles without an inadmissible failure rate of the fuel coating;

— qualify uranium/thorium mixed oxide fuel elements loaded with 20 g heavy metal;

— investigate the fission product retention of both the fuel element variants at a temperature level, which might occur in a loss-of-coolant accident.

A strategy for burning plutonium can be optimized in view of two principal objectives. Today’s main goal probably should be to reduce the separated amounts of civil and the stockpiles of military plutonium as soon as possible. This – in other words – means to maximize the amount of plutonium depleted in nuclear reactors per unit of produced energy, which is equivalent to maximizing of the fractional power production by plutonium in the reactors. Another important aspect, however, comes up with a view to intermediate storage of burned fuel and to final disposal of fuel without plutonium separation, as well as with respect to the non-proliferation aspect. From these points of view the minimization of residual plutonium in the discharged fuel elements should be the main goal of the fuelling strategy. Here, the high burnup, which is achievable in case of HTR fuel elements, is a feature of particular importance. The positive features of this fuelling strategy, of course, imply the need to handle highly enriched uranium and pure PuO_2.

3.2.2.4. Results

3.2.2.4.1. Mass balance of the fuel

Table 3.2.2 shows a comparison of two fuelling strategies for the incineration of spent LWR plutonium in the considered HTR core. The first strategy is designed to achieve a high plutonium burning ratio, the second one to achieve an especially small amount of residual plutonium in the discharged fuel elements. Table 3.2.3 displays the corresponding mass balance of the plutonium and of the fissile uranium.

Both cases apply two kinds of fuel elements, as it has been described above. About half the reactor power is produced by fissions of the plutonium. The charged plutonium is depleted by 81% (Table 3.2.3) and about 500 kg plutonium is incinerated per GWa of produced electrical energy, assuming the efficiency 0.4 for the HTR power plant. In case a) the burnup period of the fuel elements is reduced from 11 down to 7.3 years of full power, and thus the average burnup of the fuel is lowered to a standard operation value of the German AVR reactor. In consequence the amount of plutonium burned per $GW_{el}a$ increases by 30%. On the other hand, the residual plutonium of the discharged fuel also increases from 19 to 31% of the initial amount. Balancing these facts the authors come to the conclusion, that in view of the main incentive for plutonium-burning [(case a) in Table 3.2.2.)] (high plutonium-burning ratio) should be the reference strategy, representing an adequate compromise between the different optimization aspects. The requirement of uranium is similar in both cases.

TABLE 3.2.2. FUELLING STRATEGY OF THE HTR-MODUL REACTOR FOR BURNING LWR PLUTONIUM

	a) High Pu-burning ratio		b) Low residual Pu in disch. fuel	
	Pu-FE (50%)	U/Th-FE (50%)	Pu-FE (50%)	U/Th-FE (50%)
Pu	3 g	---	3 g	---
Th	---	18 g	---	16.7 g
U (HEU)	---	2 g	---	3.3 g
In core time	7.3 F. P. Years		11.0 F. P. Years	
Fractional power production	65%	35%	52%	48%
HM-burnup	595 MWd/kg	58 MWd/kg	700 MWd/kg	116 MWd/kg
Average	128 MWd/kg		192 MWd/kg	

TABLE 3.2.3. MASS BALANCES FOR THE HTR-MODUL BURNING LWR PLUTONIUM

	a) High Pu-burning ratio			b) Low residual Pu in disch. fuel		
	Pu-FE (50%)		U/Th-FE (50%)	Pu-FE (50%)		U/Th-FE (50%)
Pu-charged	929	$kg/GW_{el}a$	----	615	$kg/GW_{el}a$	---
Pu-discharged	265	$kg/GW_{el}a$	23	93	$kg/Gw_{el}a$	26
Pu-burned	664	$kg/GW_{el}a$	-23	522	$kg/GW_{el}a$	-26
	Σ 641			Σ 496		
Pu-burned/ Pu-discharged	2.2			4.2		
^{235}U charged	------	$kg/GW_{el}a$	578	-------	$kg/GW_{el}a$	624
^{235}U discharged	------	$kg/GW_{el}a$	251	-------	$kg/GW_{el}a$	171
^{233}U produced	------	$kg/GW_{el}a$	161	-------	$kg/GW_{el}a$	116

TABLE 3.2.4. COMPARISON: BURNING LWR/WEAPONS-PLUTONIUM

	a) LWR plutonium		b) Weapons-plutonium	
	Pu-FE (50%)	U/Th-FE (50%)	Pu-FE (50%)	U/Th-FE (50%)
Pu	3 g	---	3 g	---
Th	----	18 g	---	19.3
U (HEU)	---	2 g	---	0.7
In core time	7.3 F. P. Years		8.2 F. P. Years	
Fract. Power	65%	35%	89%	11%
HM-burnup	595 (MWd/kg)	58	793 (MWd/kg)	46
Average	128 MWd/kg		143 MDW/kg	
Pu-charged	929 kg/GW_{el}a	----	820 kg/GW_{el}a	---
Pu-discharged	265 kg/GW_{el}a	23	106 kg/Gw_{el}a	12
Pu-burned	664 kg/GW_{el}a	-23	714 kg/GW_{el}a	-12
	Σ 641		Σ 702	
Pu-burned/ Pu-discharged	2.2		5.9	
^{233}U produced	161 kg/GW_{el}a		151 kg/GW_{el}a	
^{233}U prod./ Pu_{fiss} charged	0.31		0.21	
^{235}U charged	578 kg/GW_{el}a		188 kg/GW_{el}a	
U_3O_8-requirement*	148 Mg/GW_{el}a		48 Mg/GW_{el}a	

* 0.25 % Tails enrichment

TABLE 3.2.5. ISOTOPIC FRACTION OF URANIUM AND PLUTONIUM IN BURNT PLUTONIUM-FUEL ELEMENTS AND IN BURNT HEU/TH-FUEL ELEMENTS

	LWR plutonium		Weapons-plutonium	
	Pu-FE	U/Th-FE	PU-FE	U/Th-FE
U3/U4/U5/U6/U8 (%)	------	30/ 3/48/13/ 6	--------	58/ 6/22/10/ 4
Pu8/Pu9/Pu0/Pu1/Pu2 (%)	0/13/20/38/29	93/ 7/ 0/ 0/ 0	0/ 8/13/38/41	97/ 3/ 0/ 0/ 0

A more significant difference is caused by the two neutronic quality levels of reactor plutonium on the one hand and of weapons plutonium on the other hand. Using the latter fuel drastically improves the neutron economy of the reactor. The comparison of the fuel mass flow data (Table 3.2.4) shows that in case of weapons plutonium the uranium loading of the thorium-based fuel elements can be decreased by about 2/3, while the average burnup of the

total unloaded fuel even slightly increases. About 90% of the reactor power is produced by plutonium fissions and the initial plutonium is incinerated by 86%, while this is only 69% in case of LWR plutonium. The yearly rate of burned plutonium is larger by 10% compared to the use of LWR plutonium.

An outline of the isotopic composition of the discharged plutonium and of the discharged uranium is given in Table 3.2.5 for both the incineration of LWR plutonium and of weapons-plutonium. The bred U^{233} and the remaining amount of ^{235}U are denatured by ^{234}U – produced by neutron capture of ^{233}U -, by ^{236}U –produced by neutron capture of ^{235}U – and by ^{238}U. The fraction of ^{233}U is 30 and 58%, respectively. So, in view of the proliferation aspect, the question obviously is not, whether to accept the production of U^{233}, but whether and where — in the sense of the non-proliferation treaty — to accept the handling of HEU in order to incinerate the plutonium.

3.2.2.4.2. Temperature coefficients

Parametric study on the temperature coefficients of a HTR for plutonium burning showed the need for a relatively large plutonium load of the fuel elements, favourably about 3 g of plutonium [4]. The result is a "hard" thermal neutron spectrum, which favours the parasitic absorption of neutrons in the resonance of the ^{240}Pu– absorption cross section at the energy 1 eve. Its increase with the moderator temperature dominates some others — partly contrary — spectral effects. Thus, the value of the moderator coefficient is strongly influenced by the fraction of ^{240}Pu in the fuel. From this point of view, PWR-plutonium is more advantageous compared to weapons-plutonium. It is important to state that these given facts may not be generalized, but are sensitive to the degree of neutron moderation in the core and to the configuration of the reflector areas.

TABLE 3.2.6. TEMPERATURE COEFFICIENTS ($\Delta k_{ef}/\Delta T$) AT OPERATING TEMPERATURE (GAS HEATING 600–900°C, DESIGN FOR HIGH PLUTONIUM–BURNING RATIO)

	LWR plutonium	Weapons grade plutonium
Doppler coefficient, K	-1.71×10^{-5}	-1.80×10^{-5}
Moderator coefficient, K	-4.46×10^{-5}	-2.76×10^{-5}
Total temp. coefficient, K	-6.17×10^{-5}	-4.56×10^{-5}

The temperature coefficients of the regarded reactor are listed in Table 3.2.6. They were calculated for full-power operation of the core, heating the cooling gas from 600 up to 900°C. Increasing the operation temperature of the fuel and of the moderator, respectively, at each local position by the same value simultaneously, results in the reactivity change described by the given coefficients. All temperature coefficients are negative and sufficiently large in the absolute value; the use of weapons grade plutonium results in the less negative one.

For safe control of the reactor over the complete temperature range, which is passed through during load changes and during startup and shutdown of the reactor, the knowledge of the temperature coefficient in a range between room temperature and the operation temperature of the reactor is necessary.

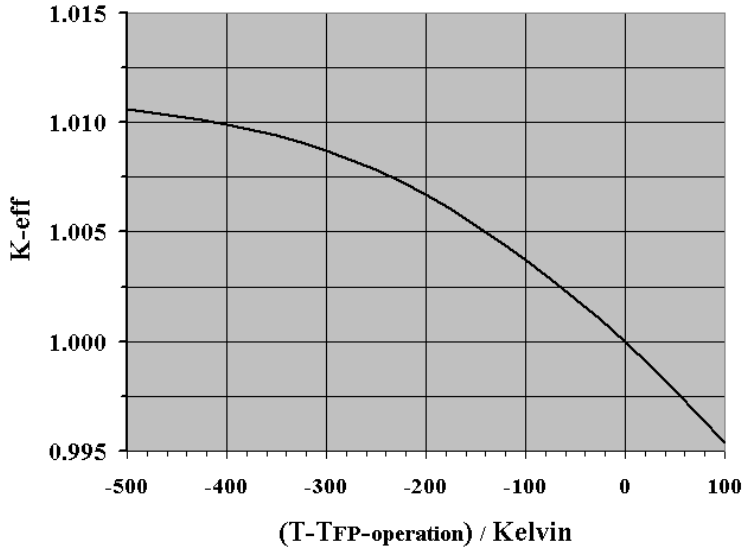

FIG. 3.2.2. Neutron multiplication as a function of the deviation of the fuel - + moderator – temperatures T from normal full-power operation values $T_{FP\text{-}operation}$ (weapons grade plutonium).

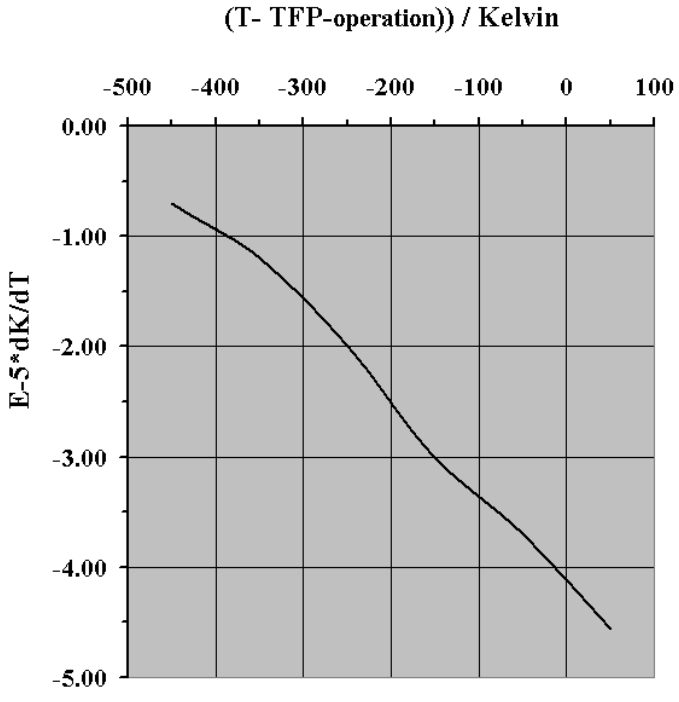

FIG. 3.2.3. Temperature coefficient as a function of the deviation of the fuel - + moderator – temperatures T from normal full-power operation values $T_{FP\text{-}operation}$ (weapons grade plutonium).

For the more sensitive core (weapons grade plutonium, Table 3.2.6) the reactivity of the reactor was calculated in many steps of temperature decrease in parallel at each local position, until the lowest local temperature value equals room temperature. It can be seen from Figs 3.2.2 and 3.2.3 that the temperature coefficient becomes continuously smaller in its absolute value vs. decreasing temperature, but it stays negative in the whole temperature range.

3.2.3. Effect of plutonium incineration on the long lived waste toxicity

The most common procedure in order to assess the toxicity of nuclear materials is based on the recommendations of the *International Commission on Radiological Protection*, defining "Annual Limits on Intake" for the radio-toxic isotopes. Recent recommendations are given in Sv/Bq and are called "Dose Coefficients for Intake (DCI)". This study uses the respective values according to *ICRP Publications 68* [3] *and 61* [17]. A comparison between the long lived waste of uranium-fuelled LWRs (providing no reprocessing of the discharged fuel) on the one hand and the waste produced at a scenario applying plutonium burning HTRs on the other hand helps to assess the related effect on the toxicity. For this purpose, a relatively simple model has been applied, comparing two alternative reactor scenarios:

Scenario 1:

Assume a given population of which we name "conventional reactors" (e.g., typical PWRs). These are all operated by use of uranium fuel, and their waste is disposed without separation of any isotopes. These reactors produce a certain, yearly amount of toxicity, which we measure in $Sv/GW_{el}a$.

Scenario 2:

Alternatively, we assume a nuclear scenario of the same size (in GW), where we have an equilibrium combination of "conventional", uranium-fuelled reactors and of a certain fraction of "HTR-Pu-burners". The principle mass flow of the fuel is illustrated in Fig. 3.2.4.

The bulk waste of "conventional" reactors is disposed and it produces a certain, yearly amount of toxicity (Tox. (a)). 99% of the disloaded plutonium, however, are separated to be used as feed fissile material (or part of it, respectively) of the HTR-plutonium burner. The spent fuel of these HTRs is finally disposed, representing another yearly amount of toxicity, Tox. (b). We add the amounts of toxicity disposed from both the "producers" and the "HTRs", normalized to 1 $GW_{el}a$.

We assume the plutonium discharge rate of one PWR to be equal 245 kg plutonium/$G_{el}a$ (Section 2.3.2, Table 2.3.9) and the HER-burner to load 929 kg plutonium/$G_{el}a$ (Section 2.2.2, Table 2.3.7). Then the sharing of an equilibrium number of plutonium producers in the total power would be 20% and that of the Lars would be 80%.

A comparison of the toxicity produced by this combined LWR-HTR-system (Tox.(a)+Tox.(b)) to the toxicity produced by scenario 1 has then been performed. The result is given in Figs 3.2.5 and 3.2.6 for the ingestion hazard and for the inhalation hazard, respectively. The toxicity of the nuclear waste is measured in Sv and it is normalized to 1 GWa of electric energy produced either by LWRs or by the combined reactor system, using HTR- plutonium burners.

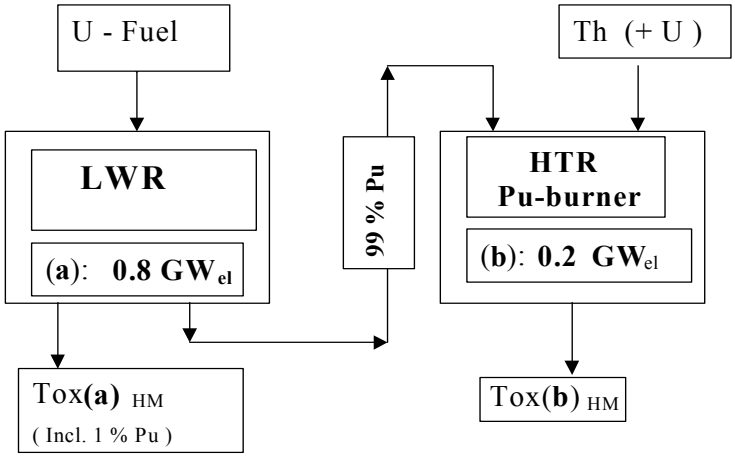

FIG. 3.2.4. Symbiosis of plutonium producers and HTR – plutonium burners.

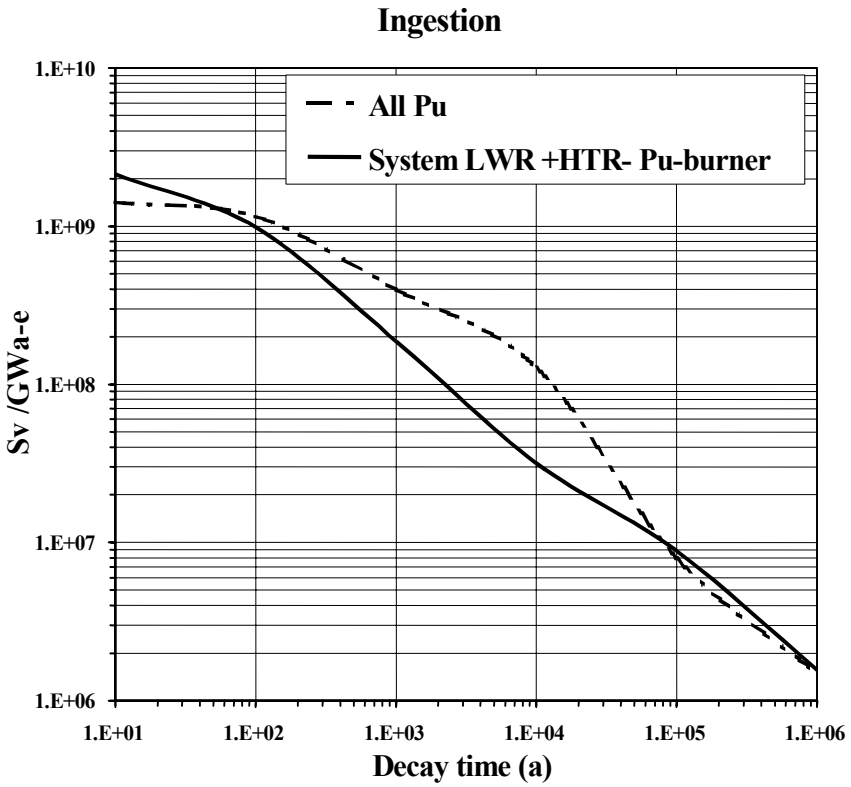

FIG. 3.2.5. Effect of plutonium incineration on the ingestion hazard of the heavy metal waste.

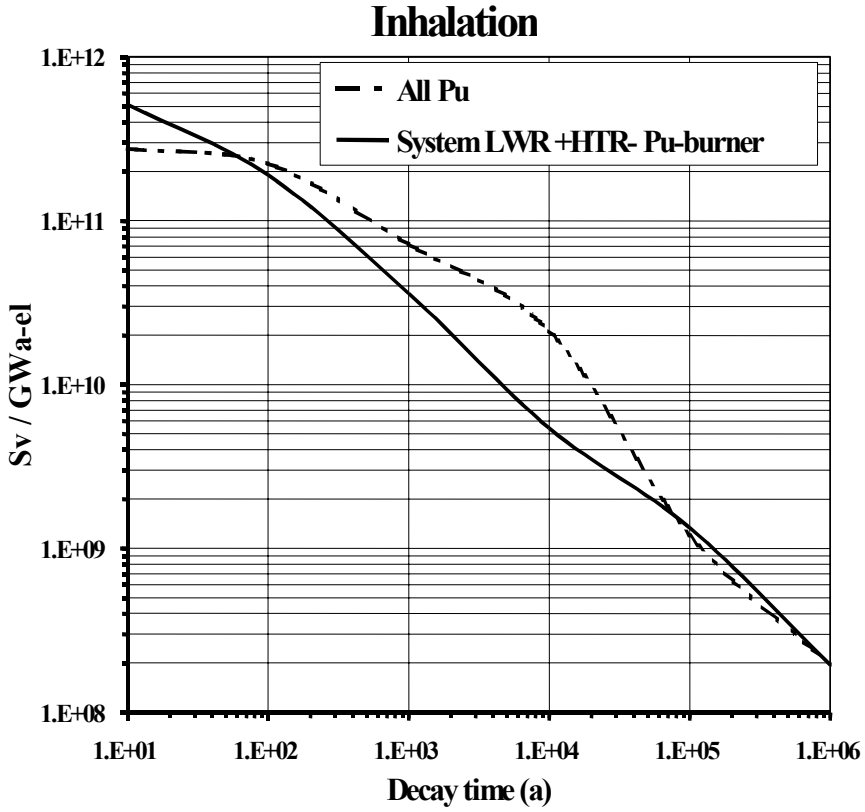

FIG. 3.2.6. Effect of plutonium incineration on the inhalation hazard of the heavy metal waste.

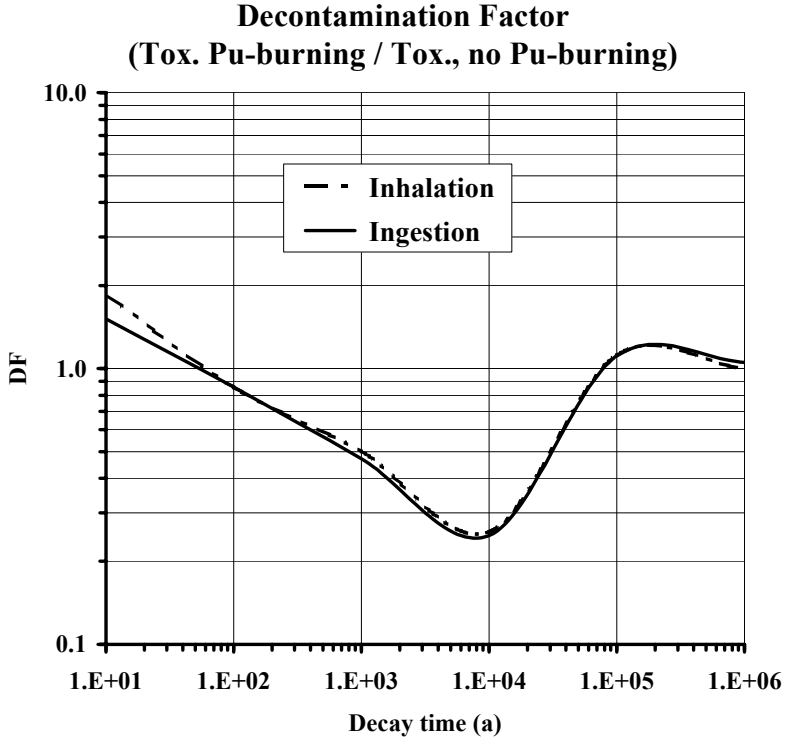

FIG. 3.2.7. Reduction of the toxicity of the heavy metal waste by plutonium incineration.

Figures 3.2.5 and 3.2.6 give a comparison of the heavy metal toxicity resulting from the two scenarios for ingestion and for inhalation, respectively. A certain reduction of the toxicity appears at a time ranging from 10^2 to 10^5 years after the end of fuel irradiation, while the toxicity is increased for a decay time smaller than 10^2 and longer than 10^5 years. The latter effect is caused by the increased buildup of minor actinides mainly out of ^{242}Pu.

Figure 3.2.7 illustrates this effect clearer in quantity. For both the ingestion and the inhalation hazard it shows a decontamination factor, i.e., the relative change of the toxicity induced by the incineration of the plutonium. The maximum reduction of the hazard potential appears at 10^4 years of isotope decay, and it amounts to about the factor 0.25. On the other hand, the toxicity is almost doubled in the beginning and also slightly increased at a decay time larger than 10^5 years.

3.2.4. Summary and conclusions

Within the scope of global efforts to reduce the large stockpiles of plutonium by making use of this fissile material for the production of electrical energy, one goal is to avoid the production of second-generation plutonium. This objective should lead from today's use of U-Pu-MOX-fuel to burning plutonium in combination with thorium as fertile material. The coated particle fuel of HTRs allows an incineration fraction of the initially loaded plutonium by about 90% for weapons plutonium and by about 70% for typical LWR plutonium during the lifetime of the fuel elements.

In view of a more rapid reduction of the existing plutonium stockpiles the burnup of the fuel may be lowered in order to increase the quantity of plutonium, which is incinerated per unit of produced electrical energy. A reduction of the average discharge burnup of the fuel from 190 MWd/kg to about 130 MWd/kg, e.g., increases the incineration rate for LWR plutonium from about 560 to about 640 kg per $GW_{el}a$ on the one hand, but on the other hand increases the fraction of residual plutonium from roughly 20 to 30% of the initial amount.

Even more effective is the incineration of weapons plutonium. Compared to the use of LWR plutonium, its destruction rate is higher by about 10% and the amount of residual plutonium is only 14% of the initial amount even in case of reduced burnup.

The temperature coefficients of the reactor – both the Doppler and the moderator coefficient — are sufficiently negative over the whole applied temperature range of reactor operation as well in case of LWR plutonium as in case of weapons-plutonium. However, the moderator coefficient in its absolute value is smaller by 40% for weapons grade plutonium compared to the use of LWR plutonium.

A desirably strong reduction of the hazard potential of the heavy metal waste — by an order of magnitude or more — is not achieved by the incineration of the plutonium. Dependent on the duration of disposal the hazard reduction factor amounts to 1.2 through 0.25. Thus, the proliferation concern turns out to be the main incentive for plutonium incineration rather than the aspect of the long term hazard potential, which is combined with its disposal.

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3.3. INDIA

3.3.1. Introduction

The objective of this CRP is to look for thorium based fuel cycles that will reduce the quantity of plutonium in the world and also create less long lived actinide wastes. This means examining different fuel cycles in which plutonium can be recycled with thorium to burn this plutonium or replace the plutonium with materials that are less unacceptable to the public. Thorium cycles are feasible in all types of reactors from LWRs to ADS. All reactors types have their own special advantages, but the major characteristic of thorium as the fuel of future comes from its superior fuel utilization. From this perspective, it turns out that the best system (barring the source driven system) is the molten salt breeder reactor, with the heavy water reactors coming in the second best. The molten salt technology is not yet commercially established, so it can be said that the present strategy would be to put the accent on thorium cycles in heavy water reactors. There are many advantages of using thorium as carrier for plutonium. The ^{233}U obtained from ^{232}Th could be used as future energy resource. The ^{233}U would be safeguarded in the spent fuel, with all the proliferation-resistant features. Moreover, the radiation fields caused due the presence of ^{232}U and its daughter products provide a high degree of self-protection and render ^{233}U unattractive as a weapons material. The option of using plutonium-thorium cycle in heavy water reactors has the highest merit in terms of plutonium destruction and has the highest energy yield because of good neutron economy even without recycling the ^{233}U. Recycling the ^{233}U would increase the energy yield many fold. To achieve these objectives, the CRP has been divided into 3 stages. The first stage is the code verification, second stage is the evaluation of the potential of plutonium incineration in different reactor systems and we have chosen the 200 MW(e) PHWR for this purpose. The third stage is the assessment of the effect of plutonium incineration on waste toxicity.

3.3.2. Benchmarks

The purpose of this stage is to verify the calculational methodology for establishing a consistent basis and to check whether the accuracy range of the reported results is adequate for the comparison of different conceptual thorium based cycles.

As a beginning, the pin level calculations were verified. The various parameters like, multiplication factor, total neutron flux, average energy per fission, isotope densities and cross-sections as a function of burnup were compared for a (Th,Pu)MOX fuelled light water reactor pin.

In the later stage, these parameters were compared for an LWR fuel assembly consisting of 17×17 fuel box lattice with the water gaps. The calculations required were assembly and core criticality curves, fuel composition, pin-by-pin power distribution and various temperature coefficients and boron worth.

3.3.2.1. Cell burnup calculations

For the pin level verification, the benchmark given was a LWR cell made of three coaxial cylindrical regions-fuel pellet, clad and the coolant moderator. Their dimensions, isotopic composition and other relevant data is as given in Section 2.1.1. Our analysis was done using the WIMSD/4 code with 69 groups WIMS library. The methodology adopted was a heterogeneous infinite lattice cell calculation followed by a homogenous leakage calculation.

The detailed results are given in Section 2.1.1. We give here the results of two of the important parameters i.e. the integral quantity k_{inf} and a differential quantity, the one group microscopic cross-sections of ^{232}Th, ^{233}U and ^{239}Pu (Tables 3.3.1 and 3.3.2).

3.3.2.2. Lattice calculations for LWR

The benchmark lattice consisted of an LWR fuel assembly consisting of 17×17 array of fuel rods, including 25 "water hole" positions. Burnup calculations were done with the given constant specific power of 37.7 w/g. The lattice calculations were done using the WIMS 69-group lattice code as before. Water holes were distributed among the adjacent pin cells. So the lattice parameters were calculated for three types of cells i.e., fuel cell sharing no water cell, sharing one water cell and sharing two water cells. These parameters were used in a diffusion theory code to perform the supercell calculations to obtain k_{eff} and pin power distributions, assuming zero current boundary conditions.

Assembly criticality is as shown in the Table 3.3.3 and that of fuel composition of the actinides in Table 3.3.4.

3.3.3. Evaluation of the potential of HWRs for plutonium incineration

For this purpose, it was agreed that each CRP participant will choose a particular reactor system. India has chosen a typical 200 MW(e) Indian PHWR (RAPS). The details of this reactor system are given below. This is a pressure tube type reactor that is moderated by heavy water and cooled by pressurized heavy water. The pressure tubes are 306 in number, are made of zircalloy and are arranged horizontally in a square lattice of pitch 22.86 cm inside a large cylindrical vessel referred to as calandria vessel. A description of this reactor is given in Table 3.3.5. No drastic modifications are envisaged for the introduction of thorium cycles into this reactor core.

The thorium/plutonium open cycle with direct disposal of the entire spent fuel has been chosen. In this cycle, we take 5.0% plutonium in thorium. Two varieties of plutonium were used in these studies. One was the standard LWR plutonium whose isotopic composition was supplied by IAEA. The other one was weapon grade plutonium. The two compositions are given in Table 3.3.6. The thorium/plutonium cycle with LWR plutonium is referred to as reactor grade plutonium (RG), and with weapons grade plutonium (WG).

The fuel is in the form of short fuel bundles of 49.53 cm length stacked along the length of the pressure tube. There are 10 such bundles in the active portion of the core, with one bundle each on either side outside the core. The fuel bundle is a 19-rod cluster. Each rod has a zircalloy tube as canning, containing a number of short ThO_2 fuel pellets with a small air gap filled with a mixture of helium and argon. A description of the fuel bundle is given in Table 3.3.7.

Lattice level calculations have been performed using the WIMS-D/4 code that gives the actinide composition as a function of burnup apart from k_{inf} and k_{eff}. The time average burnup calculations were performed using the TAQUIL code for the optimization of burnup and power distribution. The analysis details are given in Section 3.3.5.

3.3.3.1. Results of calculations

Assembly criticality curves for reactor and weapons grade plutonium are plotted in Fig. 3.3.1. Core criticality curves are shown in Fig. 3.3.2. Typical bundle power distributions at the core axial mid-plane at 0 and 600 FPD's, selected local power peaking factors as well as fuel and cladding temperatures in the hottest bundle in the core for some selected cases are given in Section 3.8.4. They are within the operating limits.

Table 3.3.8 gives the amount of initial and discharged total plutonium masses. One can note that about 65% of the initial reactor grade plutonium is burnt in this core while 86% of the initial weapons grade plutonium is burnt in the same core. The initial reactor grade plutonium that is loaded into the burner is equivalent to nearly 4.5 times the discharged plutonium from a single 1 GW(e) LWR.

3.3.4. Assessment of the effect of plutonium incineration on waste toxicity

3.3.4.1. Benchmark calculation for typical LWR fuel

This benchmark aimed at standardizing the calculational methods of different participants in predicting the plutonium toxicities as a function of time, starting from the same actinide vector for a conventional LWR discharge fuel. First the activity of the various actinides was calculated using ORIGEN code with the initial composition (mass) as given for the discharged LWR fuel. These activities were later converted to toxicities by multiplying the DCI values (supplied by IAEA) of the corresponding actinides. Then they were summed up at each chosen time intervals for the decay time of 10 years up to 10^6 years to give the total toxicity of the spent fuel. Table 3.3.9 gives the total toxicity of the actinides in water and air with 100% plutonium and 99% plutonium-removed.

3.3.4.2. Assessment of the effect of plutonium burning on the waste toxicity

In this part of the study, the efficiency of the plutonium burner (with reactor grade plutonium) is compared to that of the conventional LWR. First the individual masses, activities and the toxicities were calculated in one ton of fuel using ORIGEN code. The initial composition of the fuel is given in Table 3.3.10. Then the discharged masses (at 46 000 MWD/t for reactor-grade plutonium) and activities obtained were normalized to 1 GW(e). These activities were used to calculate the individual toxicities and the total toxicity at specified times from 10 years to 10^6 years after discharge. The individual toxicities are given in Table 3.3.11. For comparing the toxicity of the plutonium burner with that of the conventional LWR, a scheme was worked out at the last RCM in Taejon. The calculational methodology proposed is as follows:

— Let the initial charge of plutonium for the plutonium burner be X kg, and the plutonium discharged from the conventional LWR be 245 kg, per GW(e) per year.

— If the fraction of toxicity of the plutonium burner is T, then it is estimated from: X/245 = (1-T)/T.

— This T% of the plutonium burner (toxicity.a) is added to the (1-T)% of the conventional LWR toxicity with 99% plutonium removed (toxicity.b).

— This total is called toxicity2 that is compared to the toxicity of conventional LWR with 100% plutonium-toxicity1.

Following the above method, we get T = 18% taking the plutonium charged per year as 1098 kg. Thus to get toxicity2, 18% toxicity of the plutonium burner is added to 82% toxicity of conventional LWR with 99% plutonium removed.

The results of total toxicities from the evaluations are given in Table 3.3.12a for ingestion (water) and in Table 3.3.12b for inhalation (air). In the last column of these tables the decontamination factor (DF) is given which is the ratio of Toxcity2 and Toxicity1 for LWRs given in Table 3.3.9 without plutonium separation.

The results show that the DF is less than unity and decreases up to 10 000 years, but goes beyond 1 to as high as 2.3 (water) or 2.5 (air). This sudden jump at large times show that at first look the burner concept is not effective in reducing plutonium toxicity as a means of burning plutonium. The sudden increase beyond 10 000 years is not due to plutonium per se but due to the daughter isotopes of U-233 and U-234. Almost equivalent quantities of U-233 are also produced from the decay chains from Pu-241 and Am-241 that are produced from plutonium isotopes during burnup. But since the intermediate isotope Np-237, has a large half-life (2.11×10^6 years), the U-233 produced in the chain do not appear in these time periods of interest). The U-233 chain contributes to nearly 83% and U-234 chain nearly 12% to the total activity at 10^5 years. As can be seen from Tables 3.3.11a and 3.3.11b, the major contribution to toxicity at 10^5 years is due to Th-229, which is nearly 53 and 67% in water and air, respectively. The decay of U-233 with half-life of about 1.6×10^5 years to Th-229 dictates the toxicity values.

We noticed that the U-233 mass at discharge evaluated by ORIGEN with the same specific power as employed in WIMS are about 30% larger than those evaluated by the WIMS-D4 lattice code. For the actinide masses of discharged fuel as estimated from WIMS, we performed only the decay calculation with ORIGEN. This will be also consistent with the exercise in LWR benchmark analysis (Section 3.3.4.1.) where masses were supplied and one performed only the decay calculations. These results are given in Tables 3.3.13a and 3.3.13b. The DF factors are much lower. However the toxicity values at initial times would not be correct, since decay of actinides beyond Am-241 have a major contribution to toxicity in these periods. This limitation is due to our present version of WIMS library (of 1986 vintage) that does not contain actinides beyond Am-241, which was the reason why we did the analysis with ORIGEN.

Our present ORIGEN-2 code uses the cross-sections generated from PHWR spectrum, which may not be the correct set for enriched thorium systems, where the spectrum is harder.

One of our immediate efforts would to update both the WIMS and ORGEN libraries.

TABLE 3.3.1. k_{inf} OF PIN CELL

Burnup (GWd/T)	0.0	30.0	40.0	60.0
k_{inf}	1.112	0.889	0.851	0.822

TABLE 3.3.2a.. CROSS-SECTIONS AT BURNUP = 0.0 MWd/kg & 60.0 MWd/kg

	X-sections (barns) at burnup 0.0 MWD/kg	X-sections (barns) at burnup 60.0 MWD/kg
	Th-232	
Fission	0.0259	0.0237
Absorption	0.8664	1.170
	U-233	
Fission	35.7	58.9
Absorption	41.3	66.2
	Pu-239	
Fission	44.1	120.9
Absorption	68.5	189.0

TABLE 3.3.3. k_{inf} OF THE LATTICE

Burnup (GWd/t)	k_{inf}
0.0	1.1852
0.5	1.1735
20.0	1.0372
40.0	0.9104
60.0	0.8294

TABLE 3.3.4. FUEL COMPOSITION (ACTINIDES) AS A FUNCTION OF BURNUP

Burnup (MWD/t)	Number density (atom/barn×cm)	
	Th-232	**Pu-239**
0.0	2.059E-2	7.478E-4
20.0	2.036E-2	2.993E-4
60.0	1.970E-2	0.479E-4
	U-233	**Pu-240**
0.0	-	2.903E-4
20.0	1.596E-4	2.846E-4
60.0	3.191E-4	0.670E-4
	U-234	**Pu-241**
0.0	-	1.534E-4
20.0	0.9627E-5	1.545E-4
60.0	6.195E-5	0.539E-4
	Pu-238	**Pu-242**
0.0	2.29E-5	0.5010E-4
20.0	1.829E-5	0.7203E-4
60.0	7.488E-6	1.1624E-4

TABLE 3.3.5. DESCRIPTION OF PHWR REACTOR

Number of fuel channels	306
Lattice pitch, cm	22.86
Calandria inner radius, cm	299.8
Calandria length, cm	500.0
Number of bundles per channel inside the active portion of the core	10
Extrapolated core radius, cm	303.3
Extrapolated length, cm	508.5
Number of absorber rods (xenon override)	4
Number of regulating rods (for reactor regulation)	2
Total thermal power to coolant, MWth	655
Maximum channel power, MW	2.9
Maximum bundle power, KW	440
Maximum coolant outlet temperature, °C	297
Coolant inlet temperature, °C	249
Average fuel temperature, °C	625
Average coolant temperature, °C	271
Specific power, KW/kg	19.2

TABLE 3.3.6. PLUTONIUM COMPOSITION

Isotope	reactor grade plutonium initial %	reactor grade plutonium at discharge %	Weapons grade plutonium initial %	Weapons grade plutonium at discharge %
Pu-238	1.0	0.9	-	-
Pu-239	62.0	16.3	94.0	29.7
Pu-240	24.0	45.0	6.0	35.3
Pu-241	8.0	12.7	-	14.4
Pu-242	5.0	25.0	-	20.5

TABLE 3.3.7. DESCRIPTION OF THE 19 ROD ELEMENT FUEL ASSEMBLY

Fuel material (normal)		Nat UO_2
Fuel material (CRP)		ThO_2 (5% Pu)
Number of fuel rods in a fuel bundle		19
Sheath material		Zr-4
Diameter of the fuel rod, mm		14.4
Outer diameter of the sheath, mm		15.2
Clad thickness, mm		0.38
Diameter of the first ring (6 pins), cm		3.3
Diameter of the second ring (12 pins), cm		6.36
Bundle mass (ThO_2), kg		12.0
Pressure tube (zircalloy)	ID, cm	8.26
	OD, cm	9.1
Air gap thickness, mm		8.5
Calandria tube (zircalloy)	ID, cm	10.8
	OD, cm	11.1

TABLE 3.3.8. MASS BALANCES [kg/Gwe year)]

	reactor grade plutonium	Weapons grade plutonium
U^{233} charged	-	-
U^{235} charged	-	-
Pu charged	1098	725
Pu-discharged	405	141
Pu-burned	693	584
Ratio Pu-burned/Pu-charged	0.63	0.81
^{233}U produced	286	204
Average Mwd/kg	46	70

TABLE 3.3.9. TOXICITY OF CONVENTIONAL LWR

Time (years)	Ingestion		Inhalation	
	100% Pu (toxicity1)	99% Pu-removed	100% Pu (toxicity1)	99% Pu-removed
10^1	1.42E+9	1.60E+8	2.78E+11	4.11E+10
10^2	1.16E+9	5.27E+7	2.24E+11	1.07E+10
10^3	4.02E+8	1.36E+7	7.27E+10	2.56E+9
10^4	1.28E+8	4.68E+6	2.14E+10	8.04E+8
10^5	7.56E+6	1.14E+6	1.11E+9	8.86E+7
10^6	1.25E+6	7.19E+5	1.44E+8	5.26E+7

TABLE 3.3.10. ISOTOPIC COMPOSITION IN 1 t OF FUEL (INITIAL) OF THE PLUTONIUM BURNER

Isotope	Quantity (kg)
Th-232	834.847
Pu-238	0.439
Pu-239	27.24
Pu-240	10.55
Pu-241	3.52
Pu-241	2.2

TABLE 3.3.11a. TOXICITY OF REACTOR GRADE PLUTONIUM IN WATER (Sv/GWe)

Time (years)	0	10^1	10^2	10^3	10^4	10^5	10^6
Tl-209	0.567E+00	0.187E+01	0.148E+02	0.138E+03	0.913E+03	0.105E+04	0.565E+02
Pb-209	0.315E+01	0.104E+02	0.822E+02	0.766E+03	0.507E+04	0.585E+04	0.314E+03
Pb-210	0.101E+01	0.212E+01	0.181E+03	0.260E+05	0.102E+07	0.783E+07	0.133E+07
Pb-212	0.198E+06	0.301E+06	0.129E+06	0.572E+03	0.550E+03	0.550E+03	0.550E+03
Pb-214	0.268E-03	0.137E-02	0.446E-01	0.371E+01	0.146E+03	0.112E+04	0.190E+03
Bi-210	0.144E-02	0.303E-02	0.258E+00	0.371E+02	0.146E+04	0.112E+05	0.190E+04
Bi-212	0.991E+04	0.150E+05	0.645E+04	0.286E+02	0.275E+02	0.275E+02	0.275E+02
Bi-213	0.105E+02	0.346E+02	0.274E+03	0.255E+04	0.169E+05	0.195E+05	0.105E+04
Bi-214	0.268E-03	0.137E-02	0.446E-01	0.371E+01	0.146E+03	0.112E+04	0.190E+03
Po-210	0.301E+00	0.606E+00	0.517E+02	0.742E+04	0.292E+06	0.224E+07	0.379E+06
Po-213	0.462E+02	0.152E+03	0.121E+04	0.112E+05	0.744E+05	0.858E+05	0.461E+04
Po-214	0.241E-02	0.123E-01	0.401E+00	0.334E+02	0.131E+04	0.101E+05	0.171E+04
Po-216	0.297E+05	0.451E+05	0.193E+05	0.858E+02	0.825E+02	0.825E+02	0.825E+02
Po-218	0.241E-02	0.123E-01	0.401E+00	0.334E+02	0.131E+04	0.101E+05	0.171E+04
At-217	0.473E+02	0.156E+03	0.123E+04	0.115E+05	0.760E+05	0.877E+05	0.471E+04
Rn-220	0.991E+03	0.150E+04	0.645E+03	0.286E+01	0.275E+01	0.275E+01	0.275E+01
Rn-222	0.804E-04	0.410E-03	0.134E-01	0.111E+01	0.438E+02	0.336E+03	0.569E+02
Fr-221	0.473E+02	0.156E+03	0.123E+04	0.115E+05	0.760E+05	0.877E+05	0.471E+04
Ra-224	0.231E+07	0.351E+07	0.150E+07	0.667E+04	0.642E+04	0.642E+04	0.642E+04
Ra-225	0.525E+04	0.173E+05	0.137E+06	0.128E+07	0.845E+07	0.974E+07	0.523E+06
Ra-226	0.804E+00	0.410E+01	0.134E+03	0.111E+05	0.438E+06	0.336E+07	0.569E+06
Ra-228	0.287E+05	0.502E+05	0.642E+05	0.642E+05	0.642E+05	0.642E+05	0.642E+05
Ac-225	0.105E+04	0.346E+04	0.274E+05	0.255E+06	0.169E+07	0.195E+07	0.105E+06
Ac-228	0.164E+02	0.287E+02	0.367E+02	0.367E+02	0.367E+02	0.367E+02	0.367E+02
Th-228	0.231E+07	0.351E+07	0.150E+07	0.667E+04	0.642E+04	0.642E+04	0.642E+04
Th-229	0.263E+05	0.864E+05	0.685E+06	0.639E+07	0.422E+08	0.487E+08	0.262E+07
Th-230	0.419E+03	0.712E+03	0.384E+04	0.394E+05	0.375E+06	0.223E+07	0.379E+06
Th-232	0.183E+05	0.183E+05	0.183E+05	0.183E+05	0.183E+05	0.183E+05	0.183E+05
Pa-233	0.148E+07	0.217E+02	0.485E+03	0.312E+04	0.392E+04	0.381E+04	0.285E+04
U-233	0.708E+07	0.708E+07	0.708E+07	0.705E+07	0.679E+07	0.465E+07	0.257E+06
U-234	0.898E+06	0.912E+06	0.101E+07	0.110E+07	0.107E+07	0.831E+06	0.649E+05
U-236	0.358E+02	0.586E+02	0.288E+03	0.248E+04	0.159E+05	0.243E+05	0.237E+05
Np-237	0.360E+03	0.242E+04	0.539E+05	0.346E+06	0.436E+06	0.423E+06	0.316E+06
Np-239	0.119E+06	0.119E+06	0.118E+06	0.109E+06	0.467E+05	0.996E+01	0.298E-03
Pu-238	0.223E+10	0.213E+10	0.105E+10	0.894E+06	0.133E-12	0.000E+00	0.000E+00
Pu-239	0.104E+09	0.104E+09	0.104E+09	0.103E+09	0.853E+08	0.696E+07	0.112E+00
Pu-240	0.512E+09	0.515E+09	0.517E+09	0.470E+09	0.181E+09	0.130E+05	0.763E+00
Pu-241	0.269E+10	0.174E+10	0.229E+08	0.113E+04	0.543E+03	0.352E+00	0.000E+00
Pu-242	0.284E+07	0.284E+07	0.284E+07	0.283E+07	0.278E+07	0.237E+07	0.473E+06
Am-241	0.744E+09	0.198E+10	0.376E+10	0.896E+09	0.222E+05	0.148E+02	0.000E+00
Am-243	0.299E+08	0.298E+08	0.296E+08	0.272E+08	0.117E+08	0.249E+04	0.744E-01
Cm-242	0.580E+09	0.170E+06	0.112E+06	0.186E+04	0.280E-14	0.000E+00	0.000E+00
Cm-244	0.124E+10	0.882E+09	0.281E+08	0.308E-07	0.000E+00	0.000E+00	0.000E+00
Total	0.815E+10	0.741E+10	0.553E+10	0.152E+10	0.344E+09	0.918E+08	0.715E+07

TABLE 3.3.11b. TOXICITY OF REACTOR GRADE PLUTONIUM IN AIR (Sv/GWe)

Time (years)	0	10^1	10^2	10^3	10^4	10^5	10^6
Tl-209	0.567E+01	0.187E+02	0.148E+03	0.138E+04	0.913E+04	0.105E+05	0.565E+03
Pb-209	0.158E+01	0.518E+01	0.411E+02	0.383E+03	0.253E+04	0.292E+04	0.157E+03
Pb-210	0.144E+01	0.303E+01	0.258E+03	0.371E+05	0.146E+07	0.112E+08	0.190E+07
Pb-212	0.991E+06	0.150E+07	0.645E+06	0.286E+04	0.275E+04	0.275E+04	0.275E+04
Pb-214	0.134E-01	0.684E-01	0.223E+01	0.185E+03	0.730E+04	0.559E+05	0.948E+04
Bi-210	0.116E+00	0.242E+00	0.207E+02	0.297E+04	0.117E+06	0.895E+06	0.152E+06
Bi-212	0.132E+07	0.200E+07	0.860E+06	0.381E+04	0.367E+04	0.367E+04	0.367E+04
Bi-213	0.210E+04	0.691E+04	0.548E+05	0.511E+06	0.338E+07	0.390E+07	0.209E+06
Bi-214	0.536E-01	0.273E+00	0.891E+01	0.742E+03	0.292E+05	0.224E+06	0.379E+05
Po-210	0.451E+01	0.909E+01	0.775E+03	0.111E+06	0.438E+07	0.336E+08	0.569E+07
Po-213	0.154E+04	0.507E+04	0.402E+05	0.375E+06	0.248E+07	0.286E+07	0.154E+06
Po-214	0.803E-01	0.410E+00	0.134E+02	0.111E+04	0.438E+05	0.336E+06	0.569E+05
Po-216	0.991E+06	0.150E+07	0.645E+06	0.286E+04	0.275E+04	0.275E+04	0.275E+04
Po-218	0.804E-01	0.410E+00	0.134E+02	0.111E+04	0.438E+05	0.336E+06	0.569E+05
At-217	0.158E+04	0.518E+04	0.411E+05	0.383E+06	0.253E+07	0.292E+07	0.157E+06
Rn-220	0.231E+04	0.351E+04	0.150E+04	0.667E+01	0.642E+01	0.642E+01	0.642E+01
Rn-222	0.188E-04	0.957E-04	0.312E-02	0.260E+00	0.102E+02	0.783E+02	0.133E+02
Fr-221	0.158E+04	0.518E+04	0.411E+05	0.383E+06	0.253E+07	0.292E+07	0.157E+06
Ra-224	0.991E+08	0.150E+09	0.645E+08	0.286E+06	0.275E+06	0.275E+06	0.275E+06
Ra-225	0.315E+06	0.104E+07	0.822E+07	0.766E+08	0.507E+09	0.585E+09	0.314E+08
Ra-226	0.536E+02	0.273E+03	0.891E+04	0.742E+06	0.292E+08	0.224E+09	0.379E+08
Ra-228	0.123E+06	0.215E+06	0.275E+06	0.275E+06	0.275E+06	0.275E+06	0.275E+06
Ac-225	0.420E+06	0.138E+07	0.110E+08	0.102E+09	0.676E+09	0.779E+09	0.418E+08
Ac-228	0.123E+04	0.215E+04	0.275E+04	0.275E+04	0.275E+04	0.275E+04	0.275E+04
Th-228	0.132E+10	0.200E+10	0.860E+09	0.381E+07	0.367E+07	0.367E+07	0.367E+07
Th-229	0.525E+07	0.173E+08	0.137E+09	0.128E+10	0.845E+10	0.974E+10	0.523E+09
Th-230	0.838E+05	0.142E+06	0.768E+06	0.788E+07	0.749E+08	0.447E+09	0.759E+08
Th-232	0.367E+07	0.367E+07	0.367E+07	0.367E+07	0.367E+07	0.367E+07	0.367E+07
Pa-233	0.660E+07	0.966E+02	0.216E+04	0.139E+05	0.174E+05	0.169E+05	0.127E+05
U-233	0.127E+10	0.127E+10	0.127E+10	0.127E+10	0.122E+10	0.837E+09	0.463E+08
U-234	0.162E+09	0.164E+09	0.182E+09	0.198E+09	0.193E+09	0.150E+09	0.117E+08
U-236	0.573E+04	0.938E+04	0.462E+05	0.397E+06	0.255E+07	0.389E+07	0.378E+07
Np-237	0.721E+05	0.483E+06	0.108E+08	0.693E+08	0.872E+08	0.847E+08	0.633E+08
Np-239	0.149E+06	0.149E+06	0.148E+06	0.136E+06	0.584E+05	0.124E+02	0.372E-03
Pu-238	0.446E+12	0.426E+12	0.210E+12	0.179E+09	0.266E-10	0.000E+00	0.000E+00
Pu-239	0.174E+11	0.174E+11	0.173E+11	0.171E+11	0.142E+11	0.116E+10	0.186E+02
Pu-240	0.854E+11	0.858E+11	0.861E+11	0.783E+11	0.302E+11	0.216E+07	0.127E+03
Pu-241	0.483E+12	0.313E+12	0.412E+10	0.203E+06	0.977E+05	0.634E+02	0.000E+00
Pu-242	0.567E+09	0.567E+09	0.567E+09	0.566E+09	0.557E+09	0.474E+09	0.946E+08
Am-241	0.149E+12	0.396E+12	0.752E+12	0.179E+12	0.444E+07	0.297E+04	0.000E+00
Am-243	0.597E+10	0.597E+10	0.592E+10	0.544E+10	0.233E+10	0.498E+06	0.149E+02
Cm-242	0.290E+12	0.850E+08	0.562E+08	0.928E+06	0.140E-11	0.000E+00	0.000E+00
Cm-244	0.373E+12	0.265E+12	0.844E+10	0.925E-05	0.000E+00	0.000E+00	0.000E+00
Total	0.185E+13	0.151E+13	0.109E+13	0.284E+12	0.585E+11	0.146E+11	0.946E+09

TABLE 3.3.12a. TOXICITY (WATER) OF PLUTONIUM-BURNER AGAINST CONVENTIONAL LWR (USING ORIGEN MASSES & ORIGEN DECAY)

Time (years)	Toxicity of the Pu-burner without Pu-separation	Toxicity.a = 0.82 of LWR with Pu-separation	Toxicity.b = 0.18 of the Pu-burner without Pu-separation	Toxicity2 = Toxicity.a + Toxicity.b	DF
10	7.41E+9	1.31E+8	1.33E+9	1.46E+9	1.03
10^2	5.53E+9	4.32^{E}+7	9.95E+8	1.04E+9	0.89
10^3	1.52E+9	1.11^{E}+7	2.74E+8	2.85E+8	0.71
10^4	3.44E+8	3.84^{E}+6	6.19E+7	6.57E+7	0.51
10^5	9.18E+7	9.35^{E}+5	1.65E+7	1.74E+7	2.3
10^6	7.15E+6	5.89^{E}+5	1.29E+6	1.88E+6	1.5

TABLE 3.3.12b. TOXICITY (AIR) OF PLUTONIUM-BURNER AGAINST CONVENTIONAL LWR (USING ORIGEN MASSES & ORIGEN DECAY)

Time (years)	Toxicity of the Pu-burner without Pu-separation	Toxicity.a = 0.82 of LWR with Pu-separation	Toxicity.b = 0.18 of the Pu-burner without Pu-separation	Toxicity2 = Toxicity.a + Toxicity.b	DF
10	1.51E+12	3.37E+10	2.72E+11	3.06E+11	1.10
10^2	1.09^{E}+12	8.77E+9	1.96E+11	2.05E+11	0.92
10^3	2.84^{E}+11	2.09E+9	5.11E+10	5.32E+10	0.73
10^4	5.85^{E}+10	6.59E+8	1.05E+10	1.12E+10	0.52
10^5	1.46^{E}+10	7.27E+7	2.63E+9	2.70E+9	2.43
10^6	9.46^{E}+8	4.31E+7	1.71E+8	2.14E+8	1.49

TABLE 3.3.13A. TOXICITY (WATER) OF PLUTONIUM-BURNER AGAINST CONVENTIONAL LWR (USING WIMS MASSES & ORIGEN DECAY)

Time (years)	Toxicity of the Pu-burner without Pu-separation	Toxicity.a = 0.82 of LWR with Pu-separation	Toxicity.b = 0.18 of the Pu-burner without Pu-separation	Toxicity2 = Toxicity.a + Toxicity.b	DF
10	3.62E+9	1.31E+8	6.52E+8	7.83E+8	0.55
10^2	2.72^{E}+9	4.32E+7	4.89E+8	5.32E+8	0.45
10^3	7.79^{E}+8	1.11E+7	1.40E+8	1.51E+8	0.38
10^4	2.41^{E}+8	3.84E+6	4.34E+7	4.69E+7	0.37
10^5	6.66^{E}+7	9.35E+5	1.19E+7	1.28E+7	1.69
10^6	4.26^{E}+6	5.89E+5	7.67E+6	1.36E+6	1.09

TABLE 3.3.13b. TOXICITY (AIR) OF PLUTONIUM-BURNER AGAINST CONVENTIONAL LWR (USING WIMS MASSES & ORIGEN DECAY)

Time (years)	Toxicity of the Pu-burner without Pu-separation	Toxicity.a = 0.82 of LWR with Pu-separation	Toxicity.b = 0.18 of the Pu-burner without Pu-separation	Toxicity2 = Toxicity.a + Toxicity.b	DF
10	6.95E+11	3.37E+10	1.25E+11	1.59E+11	0.57
10^2	5.26^E+11	8.77E+9	2.47E+11	1.04E+11	0.46
10^3	1.41^E+11	2.09E+9	2.54E+10	2.75E+10	0.38
10^4	4.10^E+10	6.59E+8	7.38E+9	8.04E+9	0.38
10^5	1.08^E+10	7.27E+7	1.95E+9	2.02E+9	1.82
10^6	5.47^E+8	4.31E+7	9.85E+7	1.42E+8	0.99

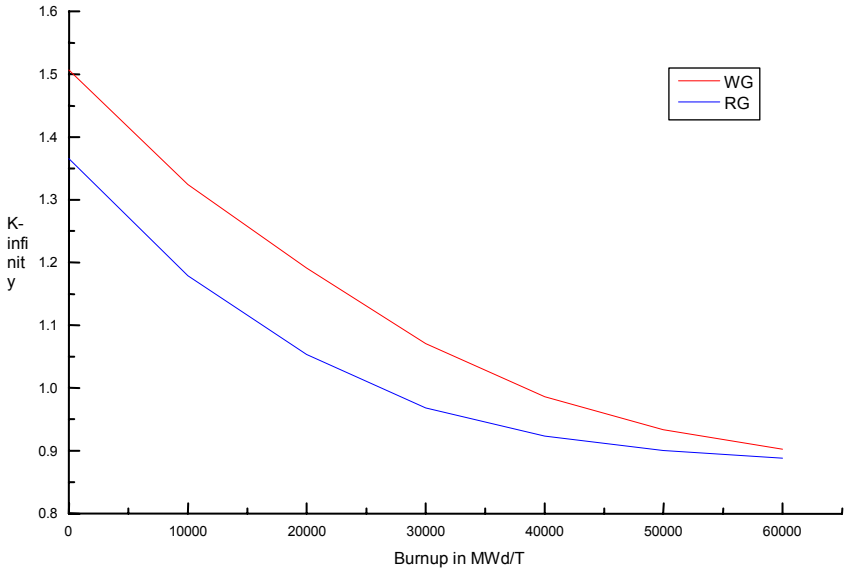

FIG. 3.3.1. Assembly criticality curves.

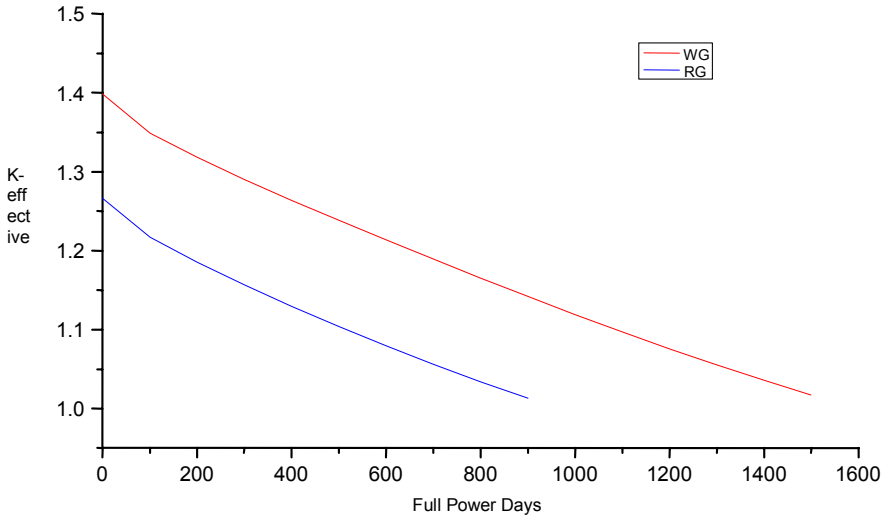

FIG. 3.3.2. Assembly criticality curves.

3.3.5. Details of reactor physics calculations for plutonium burner (PHWR)

Lattice level calculations have been performed using WIMS-D/4, a 69-group neutronics code that gives the actinide composition as a function of burnup apart from k_{inf} and k_{eff}, as well as two group parameters as a function of burnup. The time average code TAQUIL, takes these burnup dependent cross-sections and estimates the equilibrium core for the nominal adjuster configuration, with multi-bundle shift options.

In TAQUIL, two-bundle shift scheme has been used and the maximum channel power and bundle power obtained are 2.9 MW(th) and 476 kWth. While the channel power satisfies the present operational limits of our 200 MW(e) PHWR, the bundle power is too high in time average simulation. The bundle power peak appears in the fresh bundle which goes into the second string position in the core. The maximum burnup before refueling is about 10 000 MWD/t at this position. Adding a suitable burnable poison to the fresh bundles, which will vanish around the above burnup, this could be controlled in principle. The design power limits of a thorium bundle also will be higher than natural uranium bundles, due to better thermal conductivity and stability properties of thorium fuel. The equilibrium channel power distribution is given in Table 3.3.14. The requirement of adjuster rod reactivity values has also been verified. The estimated core average exit burnup for reactor grade plutonium and weapons grade plutonium are 46 000 and 70 000 MWD/t, respectively. Based on these exit burnup the average daily refueling requirements are about 1.39 bundles for reactor grade plutonium fuel and about 0.92 for weapons grade plutonium fuel. The spent fuel compositions at this estimated discharge burnup and fueling rates were used as the starting point after normalizing them to 1 GW(e) assuming 300 FPDS/year of operation. This reactor produces a fission power of 690 MW(th) of which 655 MW(th) is delivered to the coolant and 200 MW(e) is generated. However the results have been normalized to 1 GW(e) as specified in the CRP. This amounts to about 22 t for reactor grade plutonium and about 14.5 t for weapons grade plutonium, of heavy metal requirement per year. The corresponding plutonium throughput requirements for 5% enrichment are 1098 and 725 kg, respectively.

However, the core follow-up calculations were also done using a two-group diffusion theory finite difference code ASPECT for the initial core. In the initial core flux flattening was introduced through loading of plain thorium bundles in the 40 selected channels. The k_{eff} as a function of core burnup i.e., the core criticality curve and the power distributions at 0 & 600 FPD's were obtained from code ASPECT. The core simulations were done only up to the start of refuelling. The transition to equilibrium from start of refuelling was not simulated explicitly.

Table 3.3.6 gives a breakdown of the plutonium isotopes in the loaded and disloaded charges, respectively, of the plutonium burner. Typical bundle power distributions at the core axial mid-plane at 0 and 600 FPD's are given in Tables 3.3.15 and 3.3.16. Selected local power peaking factors are given in Tables 3.3.17 and 3.3.18. Fuel and cladding temperatures in the hottest bundle in the core for some selected cases are given in Tables 3.3.19 and 3.3.20, which are within the operating limits.

TABLE 3.3.14. TIME AVERAGE EQUILIBRIUM POWER DISTRIBUTION (REACTOR GRADE PLUTONIUM) CHANNEL POWER DISTRIBUTION

[(Average exit burnup 45940 MWD/t; Max. channel power: 2.88 MWth; Max. bundle power: 475 kWth (channel powers in Kwth)]

	1	2	3	4	5	6	7	8	9	10
A	1448	1392	1326							
B	1800	1740	1627	1492	1353					
C	2142	2083	1962	1771	1599	1483	1280			
D	2406	2353	2243	2056	1891	1743	1521	1312		
E	2520	2479	2453	2288	2142	2011	1796	1566		
F	2651	2614	2539	2461	2340	2231	2027	1752	1500	
G	2753	2718	2647	2585	2482	2394	2206	1935	1645	
H	2825	2789	2717	2659	2564	2493	2327	2066	1730	1412
J	2867	2829	2751	2627	2535	2470	2391	2148	1813	1458
K	2880	2841	2760	2631	2539	2481	2416	2182	1852	1499
L	2867	2827	2747	2619	2527	2469	2405	2172	1844	1494
M	2826	2789	2713	2590	2500	2437	2359	2119	1795	1492
N	2758	2723	2654	2598	2506	2437	2275	2023	1749	
O	2659	2626	2559	2502	2403	2319	2138	1873	1596	
P	2531	2497	2428	2357	2244	2142	1946	1681	1436	
Q	2374	2337	2316	2165	2032	1912	1713	1489		
R	2231	2185	2088	1919	1771	1645	1484	1240		
S	1953	1902	1797	1625	1478	1425				
T	1669	1617	1510	1344	1222					

TABLE 3.3.15. POWER DISTRIBUTION IN MIDPLANE – REACTOR GRADE PLUTONIUM

(*Bundle power distribution in kW for fresh core)*

```
                        2 175 202 207 190 161
                    1   1 170 218 245 248 227 184   2   1
              138   2 138   2 222 270 296 297 274 228   2 150 156 152
          2 175 187 192 217 272 315 336 334 312 271 222 202 200 185   2
        225 233 242 237 253 302 340 356 351 330 295 259 246 250 239 230
    263 265 281 282 266   3 310 345 355 346 329   3 277 273 287 286 270 267
    289 301 312 306 287 283 310 339 347   4 332 315 288 284 308 318 308 295
280 298 322 324 310 297 295   4 335 352 353 348 327 294   3 320 339 333 306 286
281 311 333 328   4 299 304 313 333 355 361 353 330   3 301 333 353 348 320 288
287 317 339 333 307 292 301 314   4 348 354 343 324 300 307 338 356 352 324 293
284 313 338 335 306   3 293 316 334 351 350   4 313   3 296 329 349 346 319 288
285 301 325 327 303   3 284 309 336 358 359 341 316 284   3 309 331 327 303 287
    290 300 309 297 272 273   3 327 354 359 344 314 274 259 285 302 297 289
    249 259 272 271 255 261 288 319 341 344 326 293 249   3 250 260 252 246
    202 201 216 222 216 232 266 296 314 313 291 256 219 201 208 206 195 198
          2 150 159   2 187 225 254 269 265 241   2 174 158 154 143   2
          1   1   1 112 136 169 197 211 207 185 152 123 106   1   1   1
                76   1  85   1 131 144 142 124   1   1   1   1
                     1   1   1   1   1   1   1   1
```

TABLE 3.3.15. POWER DISTRIBUTION IN MIDPLANE – REACTOR GRADE PLUTONIUM

(Bundle power distribution in kW 600 days)

						78	194	209	212	202	185							
				60	72	195	221	234	236	226	204	82	69					
		179	89	180	104	228	248	259	259	251	232	111	188	193	186			
	101	201	213	213	226	252	267	274	275	268	254	230	218	218	206	104		
	222	231	240	237	246	267	278	284	284	279	269	247	240	242	232	222		
229	236	251	257	252	180	276	285	289	290	286	206	259	252	258	251	235	229	
238	250	263	268	261	266	283	289	294	230	291	282	264	261	267	262	249	237	
239	258	271	277	271	276	219	293	294	296	292	285	276	196	275	270	258	238	225
244	264	276	210	282	284	292	296	296	296	294	290	212	282	279	274	263	243	224
247	266	277	283	285	287	294	234	300	298	297	291	288	282	281	277	266	246	227
247	266	277	283	210	288	293	299	299	300	235	295	214	286	282	277	266	246	227
244	263	275	281	207	287	294	297	299	299	298	291	287	205	281	275	263	243	232
246	257	270	275	273	278	220	298	299	299	295	288	274	272	273	268	256	245	
231	242	258	265	260	266	286	293	297	296	292	282	264	172	263	256	241	230	
211	218	235	245	244	254	276	286	291	290	284	273	250	242	242	232	216	210	
	92	200	213	122	233	258	272	278	277	269	163	228	213	208	195	89		
	56	66	74	180	199	227	245	253	251	239	219	188	169	65	58	52		
			146	45	153	80	200	211	209	194	70	45	33	35				
				23	28	42	57	66	65	53	36	20	16					

TABLE 3.3.16. POWER DISTRIBUTION IN MIDPLANE – WEAPONS GRADE PLUTONIUM
(Bundle power distribution in KW for fresh core)

						1	174	201	206	189	160							
				1	1	170	218	246	249	227	184	1	1					
		136	1	138	2	223	272	298	299	275	229	2	150	155	150			
	2	174	187	192	218	274	317	338	337	314	273	224	202	199	183	2		
	222	232	242	238	254	304	342	358	353	332	297	260	247	250	238	226		
256	262	279	282	266	3	311	346	357	347	330	3	278	274	287	284	266	261	
283	297	310	305	287	284	310	339	348	4	332	316	289	284	307	316	304	289	
292	318	322	308	297	294	3	334	352	353	348	327	293	3	319	337	330	300	278
305	329	324	3	298	303	312	331	354	360	352	329	3	299	331	350	344	314	280
311	335	330	304	290	300	312	4	346	353	342	323	298	305	336	353	348	319	285
307	333	331	303	3	291	314	332	349	348	4	310	3	294	326	346	341	313	280
294	320	323	300	3	282	306	334	356	357	339	314	282	3	306	327	323	297	278
295	305	294	270	271	3	325	351	357	342	312	272	256	282	298	293	282		
254	269	268	253	259	285	316	339	342	324	291	247	2	247	256	248	240		
197	213	220	215	230	264	293	311	310	289	254	217	200	206	203	191	192		
1	147	156	2	185	223	252	267	263	239	2	173	157	152	141	1			
1	1	1	110	134	168	195	209	205	183	150	122	105	1	1	1			
		74	1	84	1	129	142	140	123	1	1	1	1					
				1	1	1	1	1	1	1								

TABLE 3.3.16. POWER DISTRIBUTION IN MIDPLANE – WEAPONS GRADE PLUTONIUM

(Bundle power distribution in KW 600 days)

```
                                68 192 209 212 201 182
                            52  62 192 221 236 237 227 202  71  60
                       173  76 174  90 226 249 260 261 252 231  96 183 189 182
                    88 198 210 211 224 252 268 275 275 269 254 228 217 217 203  91
                   221 231 240 237 245 266 277 282 282 277 267 247 240 243 233 222
            231 237 253 258 252 156 272 281 284 285 280 177 257 252 259 253 237 230
            240 253 265 268 259 261 276 282 286 195 283 276 260 259 267 264 252 240
228 242 261 271 274 267 270 185 284 285 287 283 278 268 168 273 271 261 241 227
227 247 266 276 181 276 277 282 286 287 287 285 281 179 275 276 275 266 247 226
231 251 269 276 278 277 279 284 196 289 288 287 281 278 276 279 278 268 250 230
230 250 269 277 279 178 280 284 289 289 290 197 284 180 278 279 278 268 249 229
235 247 266 276 279 176 279 285 289 292 291 290 283 278 175 278 276 266 246 235
            249 260 272 275 270 273 188 291 293 294 291 283 269 267 271 270 258 248
            232 244 260 266 259 265 282 290 294 294 289 279 261 149 261 257 243 231
            210 217 235 245 243 253 275 286 290 289 282 270 248 240 241 231 215 208
                 80 195 209 106 231 257 272 279 277 267 141 225 210 204 190  77
                 48  56  63 174 194 224 243 252 250 236 214 183 163  55  49  44
                        137  38 145  69 195 206 204 188  59  38  28  29
                             19  23  35  48  56  55  45  30  17  13
```

TABLE 3.3.17. POWER PEAKING FACTORS – REACTOR GRADE PLUTONIUM

Burnup (MWD/t)	Ring: 1	Ring: 2	Ring: 2
0.0^{+}	0.49282	0.91676	1.08389
1000.0	0.51030	0.94761	1.06701
2000.0	0.52682	0.97611	1.05138
3000.0	0.54356	1.00350	1.03628
4000.0	0.56065	1.03012	1.02155
5000.0	0.57802	1.05584	1.00724
10000.0	0.66476	1.16276	0.94656
20000.0	0.81192	1.22673	0.90231
30000.0	0.90864	1.14804	0.93360
40000.0	0.95907	1.04187	0.98247
50000.0	0.95575	0.95873	1.02432
60000.0	0.91027	0.90978	1.05259

TABLE 3.3.18. POWER PEAKING FACTORS – WEAPONS GRADE PLUTONIUM

Burnup MWD/t	Ring: 1	Ring: 2	Ring: 2
0.0^{+}	0.45124	0.91362	1.08892
1000.0	0.46607	0.94457	1.07221
2000.0	0.47990	0.97408	1.05630
3000.0	0.49409	1.00340	1.04046
4000.0	0.50878	1.03275	1.02456
5000.0	0.52391	1.06197	1.00869
10000.0	0.60422	1.19874	0.93361
20000.0	0.76313	1.35345	0.84302
30000.0	0.89719	1.32900	0.84407
40000.0	1.01563	1.22327	0.88706
50000.0	1.10378	1.10116	0.94077
60000.0	1.12480	0.99509	0.99205

TABLE 3.3.19. MAXIMUM CLAD & FUEL TEMPERATURES – REACTOR GRADE PLUTONIUM

Days	Bundle	Clad	Fuel temperature
0.0	384	389	1487
100.0	361	382	1415
200.0	356	381	1399
300.0	340	376	1348
400.0	323	371	1293
500.0	307	366	1245
600.0	295	362	1204
700.0	289	361	1188
800.0	285	359	1173
900.0	281	358	1162

TABLE 3.3.20. MAXIMUM CLAD & FUEL TEMPERATURES — WEAPONS GRADE PLUTONIUM

Days	Bundle	Clad	Fuel temperature
0.0	385	390	1490
100.0	360	382	1409
200.0	352	380	1385
300.0	338	375	1340
400.0	321	370	1287
500.0	305	366	1237
600.0	292	362	1195
700.0	285	360	1175
800.0	280	358	1157
900.0	275	357	1142
1000.0	271	355	1130
1100.0	268	354	1120
1200.0	266	354	1113
1300.0	272	356	1134
1400.0	286	360	1175
1450.0	292	362	1196

3.4.1. Introduction

The design objective of the. present design is to use thorium based fuel for an efficient incineration of the excess plutonium Two plutonium compositions were considered: the weapon grade weapons grade plutonium and the reactor grade plutonium. A heterogeneous, seed-blanket (SBU) fuel assembly design was adopted [1, 2]. The main design approach is to use plutonium as a seed fuel providing neutrons to a subcritical blanket loaded mainly with thorium.

The seed fuel consists of Pu/Zr metal alloy and the blanket fuel consists of Th-Pu-U mixed oxide. The blanket plutonium provides a fissile component, while natural uranium part is added to denature (dilute) the U-233 builtup in thorium.

The efficiency of incinerating the excess weapons grade and reactor grade plutonium and reactor grade plutonium stockpiles by utilization of the mixed oxide fuel (MOX) is significantly reduced by the production of the "new" or the second-generation plutonium. For the MOX fuel based on the reactor grade plutonium and natural uranium the residual plutonium amounts to 60–70% of the initial plutonium load. Thus, using the MOX fuel is equivalent to a transformation of the pure weapons grade plutonium or the reactor grade plutonium into reactor grade plutonium contained within the discharged fuel. Replacing uranium by thorium as a (fertile) matrix material for plutonium incinerating cycle is investigated in this work as an alternative to the MOX fuel cycle.

A well-known design problem associated with the heavy plutonium loading required in the plutonium incinerating cycles is the reactivity control problem. The higher thermal absorption cross-section of plutonium, as compared with uranium, causes reduction of the reactivity worth of all LWR control mechanisms: control rods (CR), burnable poisons (BP's) and soluble poison. This leads to a reduction of the reactivity worth of the standard PWR control system by approximately a factor of two. Several solutions were proposed and investigated, such as using enriched boron, Gd, or even additional CR's to compensate this effect.

An alternative approach is provided by a heterogeneous, SBU fuel assembly geometry. The SBU geometry allows separate lattice optimization for the seed and blanket parts. Thus, the seed region is well moderated (Vm/Vf = 3.5) while the blanket region lattice is similar to that of a standard PWR (Vm/Vf = 1.7). In the present design the CR's and BP's are concentrated mainly in the seed region with a high moderator content leading to an increased reactivity worth of all control mechanisms based on thermal absorption materials.

The fuel management scheme, reflecting the heterogeneous fuel assembly design, is based on two separate material flows for the seed and blanket fuel parts. The seed part of the core (consisting of all seed sub-assemblies) is managed in three batches, each residing in core 300 full power days (FPD's). Thus, the seed in-core residence time is 900 FPD's. The blanket is managed as a single batch residing for 6 seed cycles, i.e., 1800 days. This fuel management scheme is designed to assure an efficient utilization of thorium, in terms of natural uranium savings. In addition, a 3-batch seed reload scheme was chosen to provide an "optimal" balance between two different performance parameters: the plutonium incineration rate and the residual plutonium content in the discharged fuel. The first one should be maximized and the second one should be minimized.

The thorium based fuel cycle proposed and investigated in this work was designated for a standard PWR core, similar to Westinghouse and/or EPR design. Two plutonium composition cases were considered: a weapons grade plutonium composition case and a reactor grade plutonium composition case. The main core and lattice parameters for both cases were kept identical, the fuel composition being the main difference between the two cases.

3.4.1.1. Weapons grade plutonium case

The main design parameters for the weapons grade plutonium case are summarized in Table 3.4.1.

TABLE 3.4.1. CORE DESIGN PARAMETERS

Power output (MWth)	= 3 400
Number of fuel assemblies (SBU's)	= 193
Average power density (w/cc)	= 104
Total coolant flow (kg/s)	= 19 480
Seed design parameters:	
Assembly Volume fraction (%)	= 40.1
Composition	7.0 weight % weapons grade Pu + 93.0% weight % zircalloy
Number of fuel rods	= 96
Number of guide tubes	= 24 (+ one central)
Moderator to fuel volume ration	= 3.535
Lattice (cell positions)	= 11x11
Cell Geometry: fuel pellet radius (cm)	= 0.310
Clad outside radius (cm)	= 0.350 (no gap)
Lattice pitch (cm)	= 1.205
Average fuel temperature (^{o}C)	= 470.0
Average cladding temperature (^{o}C)	= 340.0
Average moderator temperature (^{o}C)	= 306.0
Average specific power (Mw/t)	= 186.0
Blanket design parameters:	
Assembly volume fraction (%)	= 59.9
Composition:	= 0.8% weapons grade Pu oxide + 8.2% Natural U oxide + 91.0% Th oxide
Cell geometry: Fuel pellet radius (cm)	= 0.4095
Clad outside radius (cm)	= 0.475
Lattice pitch (cm)	= 1.258
Average fuel temperature (^{o}C)	= 750.0
Average cladding temperature (^{o}C)	= 340.0
Average moderator temperature (^{o}C)	= 306.0
Number of fuel rods	= 168
Number of guide tubes	= 0
Moderator to fuel volume ratio	= 1.659
Average specific power (Mw/t)	= 30.0

3.4.1.1.1. Results of calculations (equilibrium cycle)

A full simulation of the proposed cycle involves calculations of a complete blanket lifetime, which is equivalent to 6 seed reload cycles. In this work this full simulation is approximated by a calculation of the “equilibrium” cycle assuming that its performance parameters are representative of a complete simulation, i.e., 6 seed cycles.

The equilibrium cycle for a 3-batch fuel management scheme is represented by a core which includes three seed fuel types — fresh, once-burned, and twice-burned, and a single blanket fuel type with an averaged burnup value of 900 FPD's.

A low-leakage reload pattern, typical for PWR's of the current generation, was adopted with once and twice burned seed subassemblies positioned at the core periphery.

The following table presents a summary of the main performance parameters of the weapons grade plutonium cycle analysis. The reactivity run-down curve demonstrates that the amount of fuel loaded is sufficient to sustain 300 Full Power Days (FPD's) of power operation with an excess reactivity of about 6%. This excess reactivity will be compensated by insertion of CR's, while the soluble poison system will be used for the reload operation, cold to hot reactivity shift and possibly Xe effect.

The second and the third columns show the power share of the seed and blanket parts of the fuel respectively. The values are given in total MWatts for a quarter of the core. It is shown that about 60% of the total power, averaged over the cycle, are produced in the seed and the remaining 40% are generated in blanket.

It should be noted, that for the plutonium incinerator design the seed-blanket power sharing impacts mainly the maximum local power density of the seed. This power density, in turn, defines the maximum local fuel temperature (fuel rod centerline), which is constrained by safety considerations. This temperature is presented in the last column, showing values consistent with thermal limits of a typical PWR plant.

TABLE 3.4.2. REACTIVITY, POWER SHARING, AND FUEL TEMPERATURE SUMMARY

Days	k_{eff}	Power (MWatts) seed	Power (MWatts) blanket	Max. fuel temperature (^{o}C)
0	1.06931	507.0	343.0	681.6
20	1.06460	503.4	346.6	648.7
100	1.04801	490.6	359.4	583.9
160	1.03633	481.2	368.8	572.3
200	1.02845	474.7	375.3	570.4
260	1.01613	464.6	385.4	570.6
300	1.00745	457.5	392.5	570.2
310	1.00522	455.6	394.3	571.1

The power density map showing the averaged values in units of w/cc in fuel is presented in Fig. 3.4.1. Drastically different values may be noted for the seed (upper value) and for the blanket (lower value) that is consistent with the chosen design approach, where the seed fuel is metallic alloy and the blanket fuel is oxide. Relatively low maximum fuel temperatures of the seed fuel (see last column in Table 3.4.2) illustrate this approach.

1240.3 247.6					
933.5 233.1	885.0 223.8				
896.5 222.3	941.1 204.4	869.1 218.2			
973.7 207.8	876.8 219.9	953.5 206.9	895.7 226.3		
921.8 227.9	974.8 211.6	916.3 231.0	939.0 238.4	956.8 242.1	
1030.0 219.9	937.6 235.8	1054.8 228.8	1152.5 249.9	1060.8 229.7	1146.6 234.3
979.5 242.5	1047.0 227.2	1146.6 248.5	1018.0 219.4	1053.7 208.2	895.5 170.0
899.7 211.1	1059.9 208.1	1032.7 202.9	875.6 166.1		

FIG. 3.4.1. Power distribution map – 1/8 core (subassembly averaged, w/cc in fuel).

The fuel cycle mass flow summary is summarized Table 3.4.3.

3.4.1.1.2. Discussion: Weapons grade plutonium case

Results of the thermal-hydraulic analysis of the hot-channel, i.e., SBU with the highest power density indicated that all major thermal constraints typical for a PWR were satisfied, e.g., fuel temperatures, and that relevant heat flux values were well below the critical heat flux (DNBR limit).

The results of the equilibrium cycle simulation presented above demonstrate the basic feasibility of the proposed design. The criticality rundown curve shows that amount of fissile material weapons grade plutonium is sufficient to sustain the 300 full power days and that the excess criticality is about 7%.

Additional data is provided for the thermal-hydraulic parameters. The power distribution between seed and blanket shows 0.6 power share for the seed and 0.4 for the blanket. This power distribution reflects the design objective of the proposed design: efficient plutonium incineration. Clearly, higher seed power share results in higher plutonium destruction rates. Increase of the seed power share will lead to increased power density and higher fuel temperatures correspondingly.

The results also indicate that the thermal limits of a standard PWR core are observed. The maximum fuel temperature (centerline) is somewhat above the design limit of 500^{o}C. It seems reasonable to suppose that a further optimization of the reload design and possible improvement in burnable poison loading patterns may lead to a further flattening of the power distribution, a reduction of an overall power peaking factor and subsequently a reduction of the maximum fuel temperature.

The summary of the cycle mass flow may be used to evaluate an overall fuel cycle performance. All main actinide isotopes are accounted for in Table 3.4.3, showing the annual (cycle) charge, the core inventories, and the cycle discharge. This table also shows an estimate

of the annual plutonium destruction rates and residual plutonium content in the discharged fuel stockpile.

The plutonium incineration rate for a complete cycle is estimated as 677 kg of weapons grade plutonium per year. This value accounts for 95 kg incinerated annually in the blanket and represents a value, which is equivalent to 383 kg of weapons grade plutonium incinerated in blanket during its six years of the in-core residence time.

The residual fraction of plutonium in the discharged fuel is 0.35 in seed and may be reduced significantly by shifting from a 3-batch to a 4-batch fuel management scheme. Clearly, this will lead to a corresponding reduction in the plutonium incineration rate by approximately 10%.

3.4.1.2. Reactor grade plutonium case

The second variant of the plutonium incineration cycle considered is based on reactor grade plutonium. The main core and assembly parameters are identical to those of the weapons grade plutonium design. In addition, the fuel management scheme, the load configurations, and the resulting power distributions are also almost identical to the weapons grade plutonium case. The main difference between the two design options is restricted to the fuel composition and subsequently the cycle mass flow balance.

TABLE 3.4.3. MASS FLOW SUMMARY (kg)

Core charge		Core inventory				Core discharge	
Material	Weight, kg	Material	Weight, kg	Material	Weight, kg	Material	Weight, kg
Seed (fresh)		Seed (fresh)		Seed (once)			
		Pu-238		Pu-238	0.05		
Pu-239	873.07	Pu-239	873.07	Pu-239	546.37		
Pu-240	55.73	Pu-240	55.73	Pu-240	125.09		
		Pu-241		Pu-241	32.50		
		Pu-242		Pu-242	2.15		
		Seed (once)		Seed (twice)			
		Pu-238	0.04	Pu-238	0.25		
		Pu-239	565.82	Pu-239	276.51		
		Pu-240	128.33	Pu-240	165.34		
		Pu-241	29.10	Pu-241	53.91		
		Pu-242	1.80	Pu-242	9.26		
		Seed twice		Seed out		Seed out	
		Pu-238	0.21	Pu-238	0.83	Pu-238	0.83
		Pu-239	278.72	Pu-239	91.54	Pu-239	91.54
		Pu-240	163.40	Pu-240	154.35	Pu-240	154.35
		Pu-241	52.95	Pu-241	57.67	Pu-241	57.67
		Pu-242	8.60	Pu-242	21.97	Pu-242	21.97
Initial load		Blanket					
Th-232	47484.0	Th-232	46098.16	Th-232	45629.82		
Pa-231		Pa-231	3.95	Pa-231	4.38		
U-232		U-232	2.62	U-232	3.73		
U-233		U-233	633.70	U-233	708.64		
U-234		U-234	81.19	U-234	115.20		
U-235	33.4	U-235	20.32	U-235	26.09		
U-238	4664.8	U-238	4376.22	U-238	4279.75		
Pu-238		Pu-238	1.33	Pu-238	2.10		
Pu-239	475.0	Pu-239	63.09	Pu-239	59.82		
Pu-240	30.3	Pu-240	42.91	Pu-240	29.95		
Pu-241		Pu-241	43.78	Pu-241	33.94		
Pu-242		Pu-242	35.12	Pu-242	40.78		
Summary		Total Pu, ncinerated (kg/y)		^{239}Pu, incinerated (kg/y)		Residual fraction	
Seed		602		778		0.35	
Blanket		75		70		0.11	
Total		677		848			

TABLE 3.4.4. FUEL COMPOSITION FOR THE REACTOR GRADE PLUTONIUM CASE

Seed composition: 9.0 weight % of RG Pu + 91.0 weight % of zircalloy.
Blanket composition: 1.0 weight % of RG PuO_2 + 8.5 weight % of Nat. UO_2 + 91.0 weight % of ThO_2.

The fuel composition was chosen to sustain a 300 FPD's inter-refueling interval for an equilibrium cycle, i.e., represented by the average blanket burnup and three seed fuel types: fresh, once-burned and twice-burned. The main performance parameters of the reactor grade plutonium case are summarized in Table 3.4.5. The excess reactivity, seed and blanket power sharing, and maximum centerline temperatures indicate, similarly to the weapons grade plutonium case, a basic compatibility of the proposed design with a PWR plant.

TABLE 3.4.5. REACTIVITY, POWER SHARING, AND FUEL TEMPERATURE SUMMARY

Days	k_{eff}	Power (MWatts) seed	Power (MWatts) blanket	Max. fuel temp. (ºC)
0	1.04162	505.0	345.0	665.9
20	1.03885	502.3	347.3	644.4
100	1.02691	491.7	358.3	590.0
160	1.01747	483.9	366.1	574.8
200	1.01091	478.7	371.3	571.3
260	1.00069	470.5	379.5	567.8
300	0.99362	464.8	385.2	570.2
310	0.99184	463.4	386.6	566.0

The seed (upper value) and blanket (lower value) power densities show reasonable values, which are consistent with the basic thermal limits of a PWR plant.

1135.8 242.9					
909.4 235.0	920.5 228.6				
949.2 231.0	968.9 214.4	928.6 229.3			
1008.0 219.4	935.6 231.0	986.0 218.2	952.7 236.5		
977.7 237.4	997.3 220.8	961.9 238.2	978.3 243.9	980.1 243.4	
1036.6 225.6	967.0 238.9	1037.8 229.5	1111.6 246.9	1024.0 226.1	1098.7 227.2
994.3 241.7	1019.4 225.3	1089.3 241.6	969.0 212.9	1005.4 201.3	853.4 164.4
1061.3 207.3	1019.6 203.0	983.2 195.8	831.3 160.1		

FIG. 3.4.2. Power distribution map – 1/8 core (subassembly averaged) (w/cc in fuel).

3.4.1.2.1. Discussion: reactor grade plutonium case

The results of the equilibrium cycle simulation presented above demonstrate the basic feasibility of the proposed design. The criticality rundown curve shows that the amount of the fissile material (reactor grade plutonium) is sufficient to sustain the 300 full power days inter-refueling interval and that the excess criticality is about 4%.

TABLE 3.4.6. MASS FLOW SUMMARY (kg)

Core charge		Core inventory				Core discharge	
Material	Weight, kg	Material	Weight, kg	Material	Weight, kg	Material	Weight, kg
Seed fresh		Seed fresh		Seed once			
Pu-238	17.90	Pu-238	17.90	Pu-238	15.94		
Pu-239	665.15	Pu-239	665.15	Pu-239	439.97		
Pu-240	251.97	Pu-240	251.97	Pu-240	261.13		
Pu-241	193.45	Pu-241	193.45	Pu-241	171.03		
Pu-242	65.68	Pu-242	65.68	Pu-242	77.67		
		Seed once		Seed twice			
		Pu-238	16.32	Pu-238	14.89		
		Pu-239	448.48	Pu-239	244.26		
		Pu-240	269.98	Pu-240	260.62		
		Pu-241	168.92	Pu-241	141.00		
		Pu-242	79.56	Pu-242	95.37		
		Seed twice		Seed out		Seed out	
		Pu-238	14.90	Pu-238	13.97	Pu-238	13.97
		Pu-239	244.37	Pu-239	98.33	Pu-239	98.33
		Pu-240	258.42	Pu-240	221.66	Pu-240	221.66
		Pu-241	138.87	Pu-241	106.86	Pu-241	106.86
		Pu-242	93.27	Pu-242	111.26	Pu-242	111.26
Initial load		Blanket					
Th-232	47226.1	Th-232	46172.69	Th-232	45413.86		
Pa-233		Pa-233	58.41	Pa-233	58.36		
U-232		U-232	2.58	U-232	3.75		
U-233		U-233	635.76	U-233	712.91		
U-234		U-234	78.13	U-234	111.28		
U-235	37.5	U-235	20.76	U-235	26.14		
U-238	4832.6	U-238	4537.40	U-238	4438.36		
Pu-238	12.5	Pu-238	6.38	Pu-238	6.32		
Pu-239	350.1	Pu-239	69.02	Pu-239	65.69		
Pu-240	131.3	Pu-240	51.47	Pu-240	35.20		
Pu-241	100.0	Pu-241	54.64	Pu-241	40.77		
Pu-242	37.5	Pu-242	78.57	Pu-242	79.18		
Summary		Total Pu incinerated (kg/y)		^{239}Pu, incinerated (kg/y)		Residual fraction	
Seed		642		567		0.46	
Blanket		101		71		0.16	
TOTAL		743		638			

Results of the thermal-hydraulic analysis of the hot-channel, i.e., SBU with the highest power density indicated that all major thermal constraints typical for a PWR were satisfied, e.g. fuel temperatures, and that relevant heat flux values were well below the critical heat flux (DNBR limit).

3.4.1.3. Reactivity control issues

A well-known problem of the reactivity control of a plutonium-loaded core is investigated and discussed in this section. The presence of plutonium leads to a reduction in the reactivity worth of various control mechanisms based on parasitic absorption of thermal neutrons. A series of assembly level calculations were carried out to evaluate the reactivity worth of the different control mechanisms and the moderator temperature coefficients. It should be noted that the values were generated on the assembly level representing “core averaged” values and, therefore, are applicable only for the comparison of different cycle options.

Several plutonium-based cycle options were considered and designated as follows:

PWR	A standard slightly enriched uranium fuel;
MOX – RG	A mixed oxide fuel (uranium-plutonium oxide, reactor grade plutonium);
MOX- WG	A mixed oxide fuel (uranium-plutonium oxide, weapons grade plutonium);
TMOX – RG	A thorium based homogeneous mixed oxide fuel (Th-Pu oxide reactor grade plutonium);
TMOX – WG	A thorium based homogeneous mixed oxide fuel (Th-Pu oxide weapons grade plutonium);
RTF – RG	Radkowsky thorium fuel SBU (reactor grade plutonium);
RTF – WG	Radkowsky thorium fuel SBU (weapons grade plutonium);

TABLE 3.4.7. SUMMARY OF REACTIVITY WORTH AND MTC VALUES

Fuel cycle option	MTC $\Delta\rho/^{o}C$	Soluble boron $\Delta\rho$/ppm bron	CR worth - $\Delta\rho$ (all rods inserted)
PWR	-3.19E-04	-6.50E-05	-0.3332
MOX-RG	-5.16E-04	-2.97E-05	-0.2157
MOX-WG	-3.47E-04	-3.21E-05	-0.2217
TMOX-RG	-5.07E-04	-3.05E-05	-0.2318
TMOX-WG	-2.81E-04	-2.97E-05	-0.223
RTF-RG	-2.79E-04	-5.60E-04	-0.2688
RTF-WG	2.52E-04	-5.82E-05	-0.2936

The results presented in Table 3.4.7 demonstrate the advantages of the RTF design with regard to the reactivity control issue. The moderator temperature coefficient (MTC) of a plutonium fuel homogeneously mixed with the matrix fuel, either uranium or thorium, shows much higher values in comparison with the reference PWR case. The immediate effect of such increase is the corresponding increase in cold-to-hot reactivity effect, and consequently an increase in the reactivity control requirements. In addition, a reduction of the reactivity worth of soluble boron and control rods for the RTF cases is 10–20%, while homogeneous cases show a reduction of more than 50%. Clearly, the reactivity control problem is alleviated by the heterogeneous (SBU) assembly geometry.

3.4.2. Toxicity calculations

3.4.2.1. RTF plutonium indicator

The results described below are for an RTF fuel cycle as an incinerator cycle for the burnout of weapon grade plutonium. The basic RTF core design is maintained, namely a seed-blanket core, keeping the same overall core dimensions as well as the SBU dimensions. The stock pile

hazards of this cycle are compared, for a power generation of 1000 MW(e), with those emanating from MOX cores, designed for the same purpose. The seed fuel is Pu/Zr metallic alloy, where the plutonium is weapon grade. One t of seed H.M. contains 940 kg ^{239}Pu and 60 kg Pu-240. The seed residence time in the core is 3 years, namely each year a third of all seeds, the thrice burned, are discharged into the stock pile. The blanket fuel is oxide; one t of blanket H.M. contains 900 kg Th-232, 90 kg U-238, 9 kg Pu-239, and about ½ kg of both U-235 and Pu-240. The core residence time of the blankets is 7 years, after which they are added to the stock pile. The reference MOX core operates on oxide fuel; one t H.M. in the MOX core contains 956 kg U-238, 35 kg Pu-239, 7 kg U-235, and 2 kg Pu-240.

3.4.2.2. Methodology

Three major codes are involved. Point burnup, followed by decay, is performed with ORIGEN. This simplistic calculation is improved, using results obtained from calculations with the ELCOS system codes BOXER and SILWER, for actual 3 dimensional setups of the cores under consideration. The improvement is in replacing, prior to the ORIGEN decay calculation, the ORIGEN derived density values of 33 discharged actinides, and 56 major fission products, with the more accurate ELCOS values. The toxicity estimates are based on the ICRP-68 library.

All calculations are performed first on the basis of 1 t of H.M. The results are then normalized to annual outputs into the stock pile. The annual tonnage discharges are 30 t from the MOX core, and an average of 8.5 t from the RTF core, divided up to 1.4 t from the seeds and 7.1 t from the blankets. More details can be found in Table 3.4.8.

3.4.2.3. Summary of toxic hazards

The stock pile inhalation and ingestion hazards are summarized in Tables 3.4.9 and 3.4.10 for the MOX core, and in Tables 3.4.11 and 3.4.12 for the RTF core. Comparative results, RTF vs. MOX, are to be found in Table 3.4.13 and Fig. 3.4.3, for the inhalation, and in Table 3.4.14 and Fig. 3.4.4, for the ingestion. Hazard comparisons of actinides vs. fission products are to be found in Table 3.4.15 and Fig. 3.4.5, for the MOX, and in Table 3.4.16 and Fig. 3.4.6, for the RTF.

3.4.2.4. Discussion

During the period of 10 years to 40 000 years in stock pile residence, the RTF and MOX hazards, as concerns both inhalation and ingestion, is practically the same. This is attributed to the domination of the plutonium and minor actinides in the accumulated hazard. Between 40 000 years and 1 000 000 years the RTF hazard is slightly (at most 60%) higher than the MOX hazard. This is due to the growth of Th-229, Pb-210, and Ra-226 in the thorium-based fuel of the blankets that is faster than in the uranium based fuel of the MOX. From 1 000 000 years and onward these isotopes lose importance in the RTF pile, attaining peak values in the MOX pile, with the result that the RTF becomes less hazardous. In the light of the intrinsic inaccuracies, the stock pile hazards of the RTF and MOX seem to be quite the same for the duration of 10 000 000 years. With regard to the hazards posted by actinides vs. the fission products hazards, Tables 3.4.15 and 3.4.16, or, graphically, Figs 3.4.5 and 3.4.6, show that, both in the RTF and in the MOX, they are comparable for the first few tens of years in stock pile life, upon which the FP hazards drop fast below those of the AC, to 4 orders of magnitude already at 1000 years in stock pile life.

TABLE 3.4.8. ANNUAL FUEL DISCHARGE WEIGHT, G/1 GWE

	Reference MOX	Seed	Blanket	RTF Total
Actinides	2.88E+07	5.46E+05	6.63E+06	7.18E+06
Fission products	1.14E+06	8.22E+05	4.66E+05	1.29E+06
Total	3.00E+07	1.37E+06	7.10E+06	8.47E+06

TABLE 3.4.9. REFERENCE MOX, RADIOACTIVE INHALATION TOXICITY, SV

Time (years)	10	10^2	10^3	10^4	10^5	10^6	10^7
Actinides	7.31E+11	6.77E+11	2.51E+11	7.09E+10	3.26E+09	4.56E+08	1.21E+08
Fission products	8.12E+09	8.45E+08	2.18E+05	2.00E+05	1.29E+05	4.34E+04	3.53E+03
Total	7.39E+11	6.78E+11	2.51E+11	7.09E+10	3.26E+09	4.56E+08	1.21E+08

TABLE 3.4.10. REFERENCE MOX, RADIOACTIVE INGESTION TOXICITY, SV

Time (years)	10	10^2	10^3	10^4	10^5	10^6	10^7
Actinides	3.83E+09	3.53E+09	1.39E+09	4.25E+08	1.97E+07	2.80E+06	9.44E+05
Fission products	2.17E+09	2.40E+08	1.25E+05	6.18E+04	2.17E+04	7.42E+03	3.24E+03
Total	5.99E+09	3.77E+09	1.39E+09	4.25E+08	1.97E+07	2.81E+06	9.47E+05

TABLE 3.4.11. RTF INCINERATOR, RADIOACTIVE INHALATION TOXICITY, SV

Time (years)	10	10^2	10^3	10^4	10^5	10^6	10^7
Actinides	8.71E+11	6.33E+11	2.21^{E}+11	5.65E+10	5.47E+09	4.42E+08	3.35E+07
Fission Products	1.20E+10	1.30E+09	2.49^{E}+05	2.28E+05	1.49E+05	5.54E+04	4.21E+03
Total	8.83E+11	6.34E+11	2.21^{E}+11	5.65E+10	5.47E+09	4.42E+08	3.35E+07

TABLE 3.4.12. RTF INCINERATOR, RADIOACTIVE INGESTION TOXICITY, SV

Time (years)	10	10^2	10^3	10^4	10^5	10^6	10^7
Actinides	4.23E+09	3.28E+09	1.22E+09	3.34E+08	3.36E+07	2.95E+06	1.71E+05
Fission Products	2.99E+09	3.31E+08	1.43E+05	6.99E+04	2.36E+04	8.66E+03	3.93E+03
Total	7.22E+09	3.61E+09	1.22E+09	3.34E+08	3.36E+07	2.96E+06	1.75E+05

TABLE 3.4.13. RADIOACTIVE INHALATION TOXICITY COMPARISON MOX VS. RTF, SV

Total actinides + fission products	Time (years)	10	10^2	10^3	10^4	10^5	10^6	10^7
	Reference MOX	7.39E+11	6.78E+11	2.51E+11	7.09E+10	3.26E+09	4.56E+08	1.21E+08
	RTF	8.83E+11	6.34E+11	2.21E+11	5.65E+10	5.47E+09	4.42E+08	3.35E+07

TABLE 3.4.14. RADIOACTIVE INGESTION TOXICITY COMPARISON MOX VS. RTF, SV

Total actinides + fission products	Time (years)	10	10^2	10^3	10^4	10^5	10^6	10^7
	Reference MOX	5.99E+09	3.77E+09	1.39E+09	4.25E+08	1.97E+07	2.81E+06	9.47E+05
	RTF	7.22E+09	3.61E+09	1.22E+09	3.34E+08	3.36E+07	2.96E+06	1.75E+05

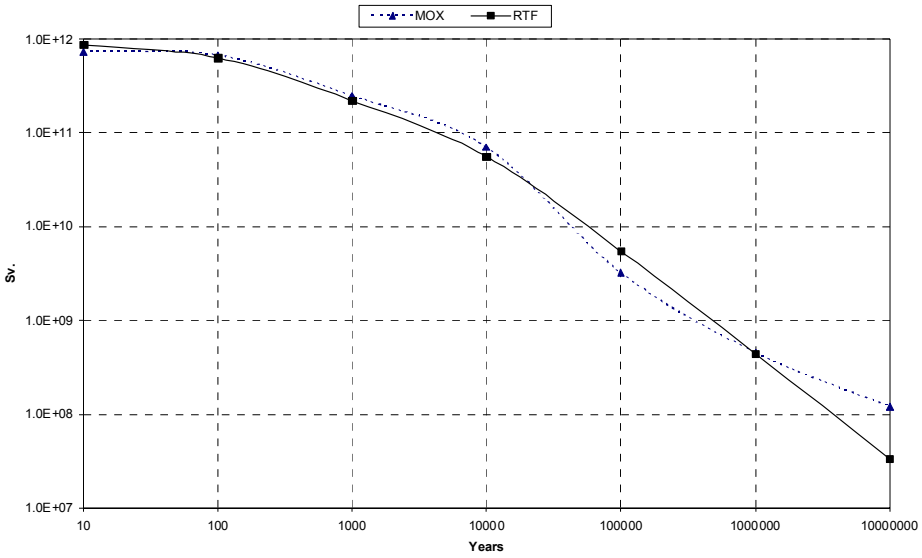

FIG. 3.4.3. Radioactive inhalation toxicity comparison MOX vs. RTF (Sv).

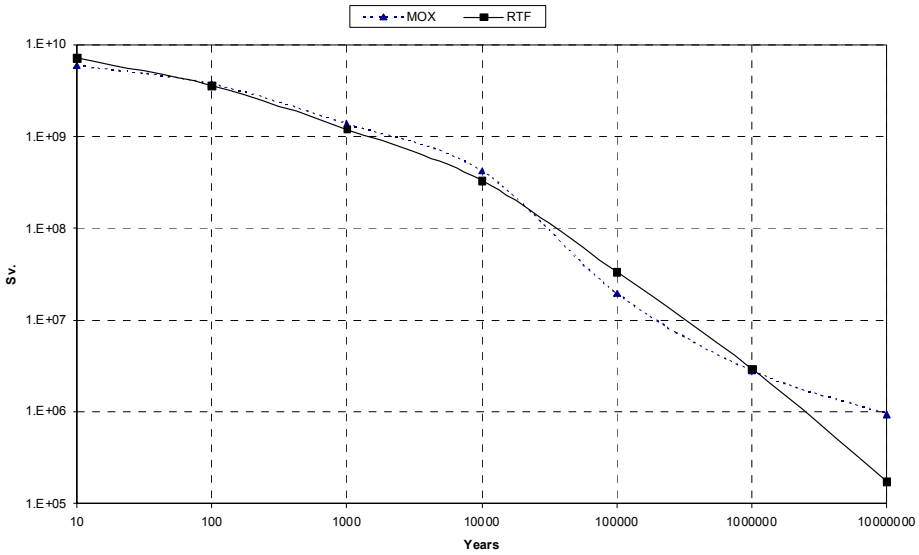

FIG. 3.4.4. Radioactive ingestion toxicity comparison MOX vs. RTF (Sv).

TABLE 3.4.15. REFERENCE MOX RADIOACTIVE INGESTION TOXICITY COMPARISON (ACTINIDES VS. FISSION PRODUCTS), SV

Time (years)	10	10^2	10^3	10^4	10^5	10^6	10^7
Actinides	3.83E+09	3.53E+09	1.39E+09	4.25E+08	1.97E+07	2.80E+06	9.44E+05
Fission Products	2.17E+09	2.40E+08	1.25^{E}+05	6.18E+04	2.17E+04	7.42E+03	3.24E+03

TABLE 3.4.16. RTF RADIOACTIVE INGESTION TOXICITY COMPARISON (ACTINIDES VS. FISSION PRODUCTS), SV

Time (years)	10	10^2	10^3	10^4	10^5	10^6	10^7
Actinides	4.23E+09	3.28E+09	1.22E+09	3.34E+08	3.36E+07	2.95E+06	1.71E+05
Fission Products	2.99E+09	3.31E+08	1.43E+05	6.99E+04	2.36E+04	8.66E+03	3.93E+03

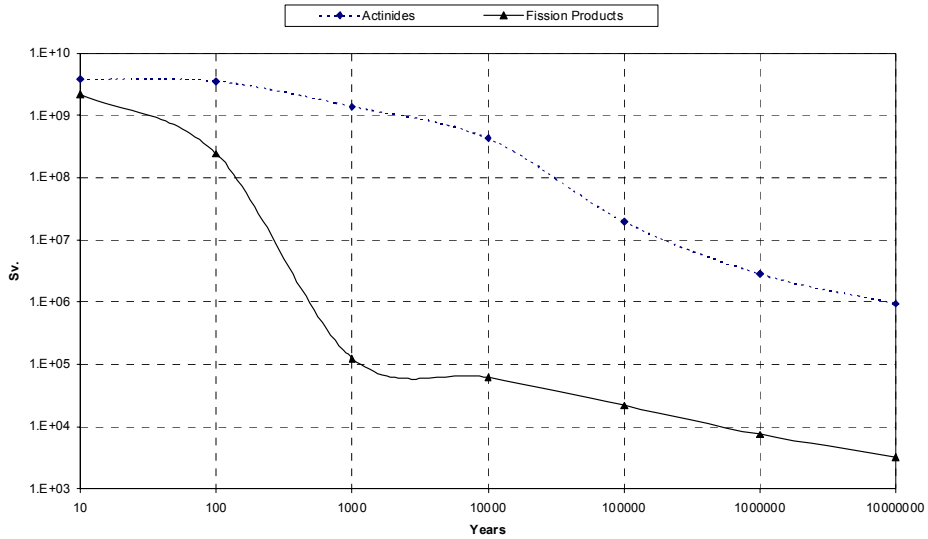

FIG. 3.4.5. Reference MOX radioactive ingestion toxicity comparison (Sv).

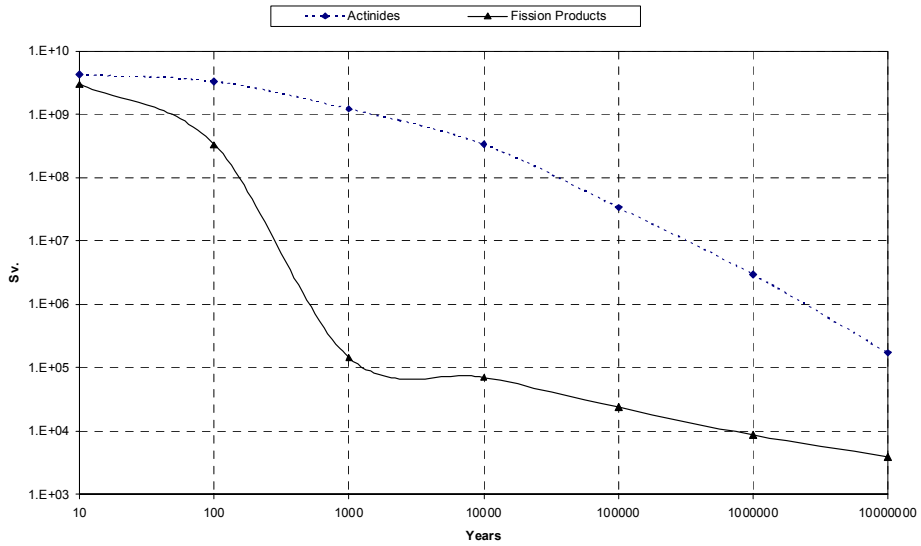

FIG. 3.4.6. RTF radioactive ingestion toxicity comparison (Sv).

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[1] RTPC Report, The Non-proliferative Light Water Thorium reactor, WWERT Preliminary Reference Design 1996 (unpublished).

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3.5. JAPAN

3.5.1. Introduction

It is considered that a molten salt reactor (MSR) based on ^{233}U-Th fuel cycle is one of the best reactor system from the standpoint of neutron economy and it is free from fuel fabrication, therefore, several studies have been made for incinerating minor actinides (MA) produced in a light water reactor (LWR) using MSR [1–3]. In 1995, IAEA organized a Co-ordinated Research Program (CRP) on Potential of Th-based Fuel Cycles to Constrain Pu and to Reduce Long Term Toxicities. The purpose of the study is to look for the best reactor system in reducing the large stockpile of plutonium both from a conventional reactor (reactor grade plutonium) and nuclear disarmament using the present and near-term reactor technology. We decided to study a MSR because of the reason stated above.

The first stage of the study was to make a benchmark calculation of the isotope composition, cross-sections and neutron fluxes for a typical PWR cell loaded with (Pu-Th)O_2 fuel as a function of fuel burnup until 60 MWD/kg and make comparisons of the effect of different computing system and data basis applied in each participants. The results are reported in [4] and the conclusion was the results of each participant agreed reasonably well. In the second stage, each participant is asked to analyze and to compare the main performance parameters of the Th-based fuel cycle options aiming to the objective of the CRP. A part of the study was reported in [5] and at the IAEA RCM (research co-ordinated meeting) at Taejon in October 1999, however, we recalculated the fuel burnup to meet the standard of 300 day operation in a year and made the calculation of toxicities accordingly.

3.5.2. Reactor model

The reactor model was taken based on FUJI model [6], which consists of graphite moderator columns with a circular fuel salt channel as shown in Fig. 3.5.1. The atomic number densities of both reactor grade plutonium and weapons grade plutonium are shown in Table 3.5.1. According to preliminary calculations for $V_F/V = 0.10$ (radius of the fuel zone r = 6.64 cm) where V_F is the fuel volume and V is the total volume of a cell, the fuel salt composition was determined as ^{7}LiF-BeF_2-ThF_4-PuF_3 = 72-16-11.8-0.2 mol%. Then to see the effect of V_F/V, V_F/V was changed to 0.05 (r = 6.64 cm) and 0.20 (r = 9.3946 cm) and the case that gives the same k-infinity (k_{inf}) as the case with $V_F/V = 0.1$ was looked for. Although for $V_F/V = 0.10$ and the fuel salt composition of ^{7}LiF-BeF_2-ThF_4-PuF_3=72-16-11.8-0.2 mol%, the burning of plutonium was most favorable, this case gave the positive temperature coefficient of $dk/dT = 3.2\times10^{-5}$ $\Delta k/k$ from 839 K to 977 K. On the other hand, for the case with $V_F/V = 0.2$, and the fuel salt composition of ^{7}LiF-BeF_2-ThF_4-PuF_3 = 72-16-11.4-0.6 mol%, the temperature coefficient was -6.5×10^{-6} $\Delta k/k$. Therefore, we selected the case $V_F/V = 0.2$ which gives k_{inf} = 1.156 for reactor grade plutonium. As for the case weapons-grade plutonium, the same V_F/V and the fuel salt composition of ^{7}Li-BeF_2-ThF_4-PuF_3 = 72-16-11.8-0.2 was selected which gives k_{inf} = 1.156 though the temperature coefficient was $+1.66\times10^{-5}$ $\Delta k/k$ because of the positive density coefficient (the fuel temperature coefficient was -6.45×10^{-6} $\Delta k/k$). It was assumed that the reactor consisted of 19 columns of the hexagonal graphite moderator 2 m high shown in Fig. 3.5.1. The effective radius of the core was 91.6 cm. The core was surrounded with a 5 cm thick fuel salt path and 100 cm thick graphite reflector in each direction as shown in Fig. 3.5.2. For this model, two dimensional (2-D) calculation for the fresh fuel core gave k_{eff} = 1.0219 and almost same k_{eff} for the case of ~ 100 MWD/kg of burnup. Therefore, it is enough to calculate the change in k_{inf}, since in the case of MSR, the flux distribution during the burnup should be

the same so long as the k_{eff} is kept ~ 1.00. It is assumed that the reactor power was 200 MWth and the fuel salt composition was same throughout the reactor because of the mixing of the fuel salt. For this model, the fuel salt volume inside the reactor vessel (core and fuel salt path surrounding the core) is $1.939\times10^6 cm^3$. It was also assumed that the fuel volume ratio inside the reactor vessel and outside of the vessel was 1:1. Therefore, the total fuel volume was $3.878\times10^6 cm^3$ and the average power density was $51.6 W/cm^3$. Since it is assumed that the reactor is operated 300 days annually, in the actual calculation, it was assumed that the reactor was operated 365 days with the power of $51.6\times(300/365) = 42.4\ W/cm^3$ of power density instead. Plutonium of the same isotopic compositions as the initial ones was continuously added to keep k_{eff} ~ 1.00. In the actual calculation, the fuel salt was added with small time interval so that the k_{inf} is kept between 1.158 and 1.130. The existence of ^{135}Xe was totally neglected. The average temperature of the core during the calculation was assumed to be 900K.

3.5.3 Calculation of fuel depletion

The burnup calculation was carried out with SWAT code system [7] and the 2D calculation was carried out with SRAC [8] code. As stated in the previous section, the fresh plutonium was fed continuously so that k_{inf} was kept between 1.158 and 1.13. The quantities and the time intervals were determined with trial and error. Actually, the calculation of reactor-grade plutonium until 10 years was carried out with the power density of 51.6 W/cm^3 up to 3000 days and normalized to 10 years so that the same MWD/kg was attained. This assumption should be appropriate except for ^{241}Pu whose half-life is ~ 13 years, and this was confirmed by the calculation of weapons grade plutonium at 10 years. Therefore, we adopted the method because of the computation economy. Also, the calculation for reactor grade plutonium after 15 years was carried out with the different computer system (with the same computing system) and small discrepancy existed in some isotopes, this effect was not serious.

The quantities of the added plutonium are shown in Table 3.5.2 for reactor grade plutonium and Table 3.5.3 for weapons grade plutonium. The values at 0 year mean the initially loaded plutonium. The changes in k_{eff} are shown in Figs 3.5.3 and 3.5.4 for reactor-grade plutonium and weapons grade plutonium, respectively. The minima for k_{inf} after ~ 15 years are a little less than 1.13, the reason of which is attributed to the use of the different computer system. The changes in atomic number densities are shown in Figs 3.5.5 and 3.6.6 for reactor grade plutonium and weapons grade plutonium, respectively. In the case of MSR, it is possible to continue the calculation indefinitely by adding certain amount of plutonium to maintain k_{eff} ~1, though in reality if we add too much plutonium, plutonium would not become soluble. Therefore we arbitrary determined to finish the calculation 10 year and 20 year of burnup. These values correspond to ~ 104 MWD/kg and ~ 208 MWD/kg for the initially loaded heavy metal (5.77 t). The depletion of thorium was not supplemented, though the amount of the depletion was ~ 11.5%.

The amount of plutonium at the end of each time interval, fissile percent of plutonium and the quantities of $^{233}Pa+^{233}U$ produced are also shown in Tables 3.5.2 and 3.5.3. The amount of the loaded and the burned plutonium at 10 years and 20 years are shown in these tables. In the case of reactor grade plutonium, the quantity of $^{233}Pa+^{233}U$ is going to saturate towards the end of 20-year period. The quantity of ^{239}Pu shows minimum at about 11 years, then it increases again. Other plutonium isotopes increase monotonically and the fissile percent of plutonium decreases though the rate is very slow at the end of 20 years. Since the fissile percent is ~ 30% and it includes a large amount of ^{241}Pu which decays with the half-life

of ~ 13 years, the quality of plutonium is very poor. However, the quantity of the burned plutonium is not large, 1.48 times of the initially loaded plutonium in 10 years and 3.43 times of that in 20 years.

In the case of weapons grade plutonium, the atomic number densities of all the plutonium isotopes monotonically increase and fissile percent of plutonium shows minimum at 9.43 years and then it increases again though the rate of increase is very slow. It is impossible to degrade the fissile percent less than ~ 60%, and if the decay of ^{241}Pu is taken into account, the fissile percent would become ~ 53%. Therefore, it is difficult to degrade the quality of weapons grade plutonium by burning in MSR only, but the discharged fuel could be used as the reactor grade plutonium. Though the increase of ^{233}Pa + ^{233}U shows the tendency of saturation towards the end of 20 year period, the quantity of the increase of plutonium is rather accelerated towards the end of 20 year period. This is because the spectrum hardening due to the increase of the absorption of actinide isotopes. From the standpoint of plutonium burning, 4.66 times of initially loaded plutonium can be burnt in 10 years and 9.26 times in 20 years. However, considering the fact that the minimum of fissile percent appears at about 10 years, it seems unnecessary to burn plutonium more than 10 years.

3.5.4. Calculation of toxicity

The main purpose of the present study is to see if it is possible to reduce the long term toxicity by burning plutonium in a reactor system. For this purpose, the toxicities from the discharged fuel were calculated using ORIGEN-2 [9] code for the decay and build up of isotopes and DCI (dose coefficient of intake) [10]. As the first step, to check the appropriateness of the calculation method and data, the change in toxicities of the discharged fuel from a LWR was calculated for both the dose from water and that from air. These values were obtained by multiplying the DCI values to the activities calculated by ORIGEN-2 code and given in Siebert (Sv). The calculation was conducted both for the discharged fuel in Table 3.5.4 and for the case where 99% of plutonium isotopes were removed from the values in Table 3.5.4. Table 3.5.5, and Figs 3.5.5 and 3.5.6 show the results of the calculation. Though the value at 0 year of the case of 99% plutonium is removed in Fig. 3.5.5 looks extraordinary, this is due to ^{239}Np. These results were compared at the IAEA's RCM in Taejon and it was confirmed that the method and the database are appropriate.

The same procedure was applied for the calculation of toxicities of discharged fuel after 10 and 20 years of burnup in a 200 MWth MSR. They are shown in Table 3.5.6 for reactor-grade plutonium and in Table 3.5.7 for weapons grade plutonium, respectively. Comparing the results of 10 and 20 years of burnup for reactor grade plutonium, it is found there is not much difference between 10 year and 20 year burnup except for the short period from discharge. The large difference in short period from discharge is mainly attributed to the increase in ^{242}Am and ^{244}Cm. The fact there are not much differences between the 10 year and 20 year burnup means there is some effect in reduction of the toxicity by burning plutonium longer in MSR. However, the quantity of reactor grade plutonium burned in a MSR is not large and the decrease in the toxicity is small.

In the case of weapons grade plutonium, the differences between the 10 and 20-year operation, respectively, are larger than the corresponding reactor grade plutonium cases. This is because in weapons grade plutonium the quantities of actinide isotopes rather increase rapidly near the end of 20 years of operation.

To make the comparison of the effectiveness of different fuel cycles, the following procedure was taken according to the agreement in the IAEA's RCM in Taejon.

Since the annual output of plutonium from a 1 GWe LWR is 245 kg as shown in Table 3.5.4, and after the reprocessing 99% of that quantity is fed to the plutonium burning system. In our MSR system, the consumption of reactor grade plutonium is 784.7 kg in 10 years and 1435.2 kg in 20 years, therefore, the annual requirement is 78.47 kg and 71.76 kg, respectively. Hence, to consume all the reprocessed plutonium produced in one 1 GWe LWR, 245×0.99/78.47 = 3.09 units of 200 MWth MSR are required for the 10 year burnup case and 245×0.99/71.76 = 3.38 units of the same MSR are required for the 20 year burnup case. Assuming the conversion efficiency of 40%, they produce 0.25 GW and 0.27 GW of electricity, respectively. Therefore, the doses from the combined fuel cycle can be calculated as (1-0.25)× (values in Table 3.5.5 of 99% of plutonium removed) + 3.09×(values in Table 3.5.6)/10 for case of the 10 year burnup case and (1-0.27)×(values in Table 3.5.5 for 99% of plutonium is removed) + 3.38× (values in Table 3.5.6)/20 for the 20 year burnup case. The results are shown in Table 3.5.8, and in Figs 3.5.7 and 3.5.8 together with the toxicities from the discharged fuel from LWR only. Except for the very long time (1.00×10^7 years from the fuel discharge) the dose from the combined system does not differ from that of LWR only. This is mainly due to the increased quantities of ^{242}Am and ^{244}Cm in the short period from the fuel discharge and ^{229}Th that is the decay product of ^{233}U for the long period from the fuel discharge. Although the case of 20-year burnup generally gives somewhat smaller doses for the longer period, the difference from the case of 10-year operation is very small. Thus, the reduction of toxicity by plutonium burning in MSR will not be expected even if we adopt the longer irradiation.

In the case of weapons grade plutonium, as it is difficult to consider the corresponding reactor system as in the case of reactor grade plutonium, we compared with the toxicities originated from the ^{239}Pu, ^{240}Pu and ^{232}Th which are loaded in the reactor as the fresh fuel as shown in Table 3.5.9. Because of the production of ^{241}Pu and ^{233}U which does not exist in the chain of the fresh fuel, the dose from the discharged fuel from MSR is very high until 100 years, and except for the points of 1.0×10^4 and 10×10^7 year the dose from both system are similar. The decrease at 1.0×10^4 years is attributed to the burnup of ^{239}Pu whose half-life is 2.39×10^4 year, however, due to the build up of ^{233}U and ^{229}Th, the doses from the both system again become similar. Therefore, the reduction of toxicity is not hopeful by burning weapons grade plutonium in MSR.

3.5.5. Conclusion

The burnup calculation up to 20 year was carried out for 200 MWth MSR loaded with both the reactor grade plutonium and the weapons grade plutonium. Also the toxicities from the discharged fuel at 10 years and 20-year burnup were calculated. It was shown that MSR with weapons grade plutonium is able to burn 4.66 times of the initially loaded plutonium in 10 years and 9.26 times of that in 20 years. The corresponding values for MSR with reactor grade plutonium are 1.43 times and 3.43 times, respectively. As for the quality of plutonium, the fissile percent of weapons grade plutonium cannot be less than 60%; on the other hand, the fissile percent becomes almost 30%. That toxicity from the discharged fuel is normalized to the toxicity/year from the combined fuel cycle that produces 1 GWe of electricity for reactor grade plutonium. It was found except for 1.0×10^7 years, the toxicity from only LWR is smaller than that of combined system.

TABLE 3.5.1. ATOMIC NUMBER DENSITIES AT THE BEGINNING OF CALCULATION (n/cm^3)

Nuclide	Reactor grade	Weapons grade
Th-233	3.649E-03	3.800E-03
Pu-238	1.908E-06	0
Pu-239	1.172E-04	5.920E-05
Pu-240	4.539E-05	3.780E-06
Pu-241	1.505E-05	0
Pu-242	9.531E-06	0
Li-7	2.260E-02	2.260E-02
Be-9	5.037E-03	5.037E-03
F-19	4.785E-02	4.785E-02
C-12	9.266E-02	9.266E-02

TABLE 3.5.2. MASSES OF PLUTONIUM AND U-233 (kg) (REACTOR-GRADE)

Burnup	Pu-added		Pu at the end of interval			Pa-233 +
(years)	total Pu	fissile Pu	total Pu	Fissile Pu	fissile %	U-233
0.32	291.7*	203.7*	279.5	189.5	0.678	6.4
0.6	29.2	20.3	297.1	196.6	0.662	12.5
0.91	29.2	20.3	312.5	197.7	0.633	64
1.26	23.3	16.3	315.2	192.2	0.610	29.1
1.62	23.3	16.3	314	186.3	0.593	38.1
2	20.4	14.3	312.6	177.7	0.568	46.8
2.4	20.4	14.3	311.8	169.5	0.544	55.2
2.71	20.4	14.3	311.4	162.3	0.521	62.9
2.98	20.4	14.3	314.8	156	0.496	69.9
3.29	20.4	14.3	318.2	153.7	0.483	75.4
3.66	20.4	14.3	318.8	151.7	0.476	80.4
4.09	20.4	14.3	319.4	147.3	0.461	86.1
4.54	20.4	14.3	320.9	143	0.446	91.4
5.02	20.4	14.3	322.2	140.1	0.435	96.3
5.54	20.4	14.3	325.4	137.1	0.421	100.8
5.98	20.4	14.3	328.6	136.1	0.414	104.6
6.46	20.4	14.3	332.3	135.1	0.407	108.3
6.94	20.4	14.3	335.8	134.8	0.401	111.6
7.41	20.4	14.3	339.4	134.4	0.396	114.7
7.87	20.4	14.3	343.1	134.1	0.391	117.7
8.26	20.4	14.3	332.9	134.1	0.403	120.5
8.79	20.4	14.3	347.9	133	0.382	123.3
9.32	20.4	14.3	350.9	132.1	0.376	126
9.71	20.4	14.3	352.5	130.9	0.371	130.7
10	20.4	14.3	351	130.6	0.372	133
Total up to 10 years						
Loaded Pu	784.7	548.5				
Burned Pu	433.7	417.9				
	initially loaded Pu *					

TABLE 3.5.2. (cont.)

Burnup (years)	Pu-added		Pu at the end of interval			Pa-233 + U-233
	total Pu	fissile Pu	total Pu	fissile Pu	fissile %	
10.33	20.4	14.3	355.1	128.1	0.361	134.9
10.66	17.5	12.2	355	126.3	0.356	136.8
10.99	20.4	14.3	353	126.7	0.359	138.5
11.32	20.4	14.3	362.3	126.6	0.349	140.1
11.64	20.4	14.3	364.7	127	0.348	141.7
11.97	20.4	14.3	368.5	127.3	0.345	143.2
12.3	20.4	14.3	370.7	127.6	0.344	144.5
12.63	20.4	14.3	374.5	128.1	0.342	145.8
12.96	20.4	14.3	376.7	128.6	0.341	147
13.29	20.4	14.3	380.7	129	0.339	148
13.62	20.4	14.3	384.6	129.7	0.337	149.2
13.95	20.4	14.3	386.8	130.1	0.336	150.2
14.27	20.4	14.3	390.8	130.7	0.334	151.1
14.6	20.4	14.3	393	131.4	0.334	152
14.93	20.4	14.3	396.9	132	0.333	152.9
15.26	20.4	14.3	400.7	132.4	0.330	153.7
15.86	20.4	14.3	399.5	129.5	0.324	154.5
16.14	20.4	14.3	401.3	129.7	0.323	155.8
16.41	20.4	14.3	404.6	129.7	0.321	155.8
16.68	20.4	14.3	406.6	130	0.320	155.8
16.96	20.4	14.3	409.9	130.1	0.317	157.3
17.23	20.4	14.3	411.8	130.4	0.317	157.3
17.51	20.4	14.3	415.2	130.6	0.315	157.3
17.78	20.4	14.3	417	130.7	0.313	158.8
18.05	20.4	14.3	420.6	131.2	0.312	158.8
18.33	20.4	14.3	422.4	131.5	0.311	158.7
18.6	20.4	14.3	424.7	131.7	0.310	158.7
18.88	20.4	14.3	427.7	132.1	0.309	160.2
19.15	20.4	14.3	429.4	132.4	0.308	160.2
19.42	20.4	14.3	432.9	132.9	0.307	160.2
19.7	20.4	14.3	434.8	133.2	0.306	160.2
Total up to 20 years						
Loaded Pu	1435.2	1002.2				
Burned Pu	1000.4	869.0				

TABLE 3.5.3a. MASSES OF PLUTONIUM AND U-233 (kg) (WEAPONS-GRADE)

Burnup	Pu-added		Pu at the end of interval			Pa-233 +
(years)	total Pu	fissile Pu	total Pu	fissile Pu	fissile %	U-233
0	97.06*	93.8*	92.01	85.03	0.924	2.31
0.08	16.99	15.97	104	94.16	0.905	5.8
0.2	16.5	15.51	113.04	99.48	0.880	11.12
0.38	17.47	16.42	120.43	102.46	0.851	17.95
0.6	17.96	16.88	125.99	103.14	0.819	26
0.86	18.44	17.33	130.46	102.78	0.788	34.41
1.15	18.93	17.79	134.42	101.79	0.757	42.97
1.45	19.89	18.70	138.89	101.53	0.731	51.24
1.79	20.38	19.16	143.03	100.99	0.706	59.36
2.15	21.35	20.07	147.38	100.87	0.684	67.05
2.51	21.83	20.52	152.36	101.63	0.667	74.31
2.89	22.32	20.98	157.5	102.7	0.652	80.98
3.28	22.81	21.44	163.12	104.36	0.640	87.21
3.66	22.81	21.44	168.9	106.32	0.629	93.01
4.05	23.29	21.89	175.42	109.04	0.622	98.23
4.44	23.3	21.90	181.77	111.75	0.615	103.15
4.85	23.78	22.35	188.14	114.47	0.608	107.93
5.25	24.27	22.81	195.1	117.93	0.604	112.28
5.65	24.75	23.27	202.48	121.7	0.601	116.33
6.06	24.75	23.27	209.83	125.61	0.599	120.24
6.47	25.24	23.73	217.64	129.83	0.597	123.85
6.89	25.72	24.18	225.74	134.35	0.595	127.32
7.31	26.21	24.64	233.98	133.86	0.572	130.79
7.74	26.69	25.09	242.81	143.98	0.593	133.97
8.16	27.18	25.55	251.46	148.95	0.592	137.02
8.58	27.18	25.55	260.11	153.92	0.592	139.91
9.01	27.66	26.00	269.18	159.34	0.592	142.65
9.43	27.67	26.01	278.1	164.6	0.592	145.41
9.85	27.66	26.00	287.65	170.32	0.592	147.85
Total up to 10 years						
Pu-loaded	643.03	604.45				
Pu-burned	355.38	434.13				
	* initially loaded Pu					

TABLE 3.5.3. (cont.)

Burnup (years)	Pu-added		Pu at the end of interval			Pa-233 + U-233
	total Pu	fissile Pu	total Pu	fissile Pu	fissile %	
10.28	27.67	26.01	297.21	176.19	0.593	151.05
10.7	27.66	26.00	306.62	181.76	0.593	152.48
11.13	28.15	26.46	316.49	176.19	0.557	155.39
11.55	28.14	26.45	325.46	193.05	0.593	156.81
11.97	28.14	26.45	336.23	199.82	0.594	159.71
12.4	28.15	26.46	345.5	205.24	0.594	161.15
12.82	28.63	26.91	356.12	212.01	0.595	164.05
13.25	28.63	26.91	365.39	217.73	0.596	165.5
13.67	28.63	26.91	376.02	224.05	0.596	166.94
14.09	28.63	26.91	385.44	229.47	0.595	169.84
14.52	28.63	26.91	396.22	236.24	0.596	171.27
14.94	28.63	26.91	405.34	241.51	0.596	172.72
15.37	29.11	27.36	416.11	248.28	0.597	174.16
15.79	29.12	27.37	427.05	255.05	0.597	175.6
16.21	29.12	27.37	436.32	260.05	0.596	177.05
16.64	29.12	27.37	446.65	267.25	0.598	179.95
17.06	29.11	27.36	458.34	274.02	0.598	181.4
17.47	29.11	27.36	467.16	279.28	0.598	182.85
17.89	29.11	27.36	477.47	286.06	0.599	184.29
18.3	29.12	27.37	489.33	292.83	0.598	185.74
18.72	29.12	27.37	499.52	299.6	0.600	187.16
19.13	29.11	27.36	508.33	304.87	0.600	188.63
19.55	29.12	27.37	520.03	311.64	0.599	190.08
19.96	29.11	27.36	530.37	318.4	0.600	190.06
Total up to 20 years						
Pu-loaded	1332.1	1252.17				
Pu-burned	801.73	933.77				

TABLE 3.5.4. DISCHARGE OF HEAVY METAL ISOTOPES (kg/year) FOR A TYPICAL PWR (NORMALIZED TO 1 GWE AND 300 FULL POWER DAYS)

U-234	4.51E+00
U-235	2.70E+02
U-236	1.07E+02
U-237	2.70E-01
U-238	2.69E+04
Np-237	1.11E+01
Np-239	2.27E+00
Pu-238	3.12E+00
Pu-239	1.43E+02
Pu-240	5.78E+01
Pu-241	3.11E+01
Pu-242	1.02E+02
Am-241	8.87E-01
Am-242m	1.66E-02
Am-242g	2.12E-03
Am-243	1.77E+00
Am-244	6.41E-05
Cm-242	2.38E-01
Cm-243	5.17E-03
Cm-244	4.56E-03
Cm-245	1.66E-02
Sum of Pu	2.45E+04
Σ	2.75E+04

TABLE 3.5.5. TOXICITIES/YEAR DUE TO DISCHARGED FUEL FROM A TYPICAL 1 GWe LWR

from water (Sv.)

year from discharge	0	1	10	1.00E+02	1.00E+03	1.00E+04	1.00E+05	1.00E+06	1.00E+07
discharged fuel	1.80E+10	1.48E+09	1.42E+09	1.16E+09	4.01E+08	1.28E+08	7.82E+06	1.48E+06	6.08E+05
99% of Pu removed	1.67E+10	2.59E+08	1.63E+08	5.37E+07	1.37E+07	4.71E+06	1.29E+06	8.59E+05	5.59E+05

from air (Sv.)

year from discharge	0	1	10	1.00E+02	1.00E+03	1.00E+04	1.00E+05	1.00E+06	1.00E+07
discharged fuel	4.39E+11	3.09E+11	2.79E+11	2.23E+11	7.26E+10	2.14E+10	1.17E+09	1.95E+08	5.91E+07
99% of Pu removed	2.16E+11	8.33E+10	4.17E+10	1.09E+10	2.56E+09	8.11E+08	1.25E+08	8.59E+07	4.59E+07

TABLE 3.5.6. TOXICITIES DUE TO DISCHARGED FUEL FROM 200 MWth MSR WITH REACTOR GRADE PLUTONIUM

from water (Sv.)

year from discharge	0	1	10	1.00E+02	1.00E+03	1.00E+04	1.00E+05	1.00E+06	1.00E+07
10 year burnup	1.05E+10	5.35E+09	4.58E+09	2.32E+09	8.04E+08	2.22E+08	3.14E+07	2.43E+06	1.02E+05
20 year burnup	1.95E+10	1.20E+10	9.41E+09	2.95E+09	9.95E+08	2.91E+08	4.31E+07	3.59E+06	1.17E+05

from air (Sv.)

year from discharge	0	1	10	1.00E+02	1.00E+03	1.00E+04	1.00E+05	1.00E+06	1.00E+07
10 year burnup	1.38E+12	1.33E+12	1.09E+12	4.54E+11	1.46E+11	3.76E+10	5.19E+09	3.62E+08	1.87E+07
20 year burnup	3.61E+12	3.27E+12	2.48E+12	5.90E+11	1.81E+11	4.95E+10	6.83E+09	4.93E+08	2.20E+07

TABLE 3.5.7. TOXICITIES DUE TO DISCHARGED FUEL FROM 200 MWth MSR WITH WEAPONS GRADE PLUTONIUM

from water (Sv.)

year from discharge	0	1	10	1.00E+02	1.00E+03	1.00E+04	1.00E+05	1.00E+06	1.00E+07
10 year burnup	8.27E+09	2.43E+09	2.13E+09	1.39E+09	5.54E+08	1.85E+08	3.47E+07	1.91E+06	8.36E+04
20 year burnup	1.25E+10	5.22E+09	4.45E+09	2.52E+09	1.02E+09	3.51E+08	5.09E+07	3.10E+06	1.35E+05

from air (Sv.)

year from discharge	0	1	10	1.00E+02	1.00E+03	1.00E+04	1.00E+05	1.00E+06	1.00E+07
10 year burnup	8.02E+11	5.86E+11	4.75E+11	2.69E+11	1.00E+11	3.13E+10	5.72E+09	2.65E+08	1.67E+07
20 year burnup	1.62E+12	1.29E+12	1.03E+12	4.86E+11	1.82E+11	5.92E+10	8.22E+09	4.19E+08	3.06E+07

TABLE 3.5.8. TOXICITIES/YEAR DUE TO DISCHARGED FUEL COMBINED CYCLE OF LWR AND MSR WITH REACTOR GRADE PLUTONIUM

from water (Sv)

year from discharge	0	1	10	100	1000	1.00E+04	1.00E+05	1.00E+06	1.00E+07
(a) 10 years MSR + LWR	1.58E+10	1.85E+09	1.53E+09	7.57E+08	2.59E+08	7.21E+07	1.07E+07	1.40E+06	4.51E+05
(b) 20 years MSR + LWR	1.55E+10	2.22E+09	1.71E+09	5.38E+08	1.78E+08	5.26E+07	8.23E+06	1.23E+06	4.29E+05
(c) LWR only	1.80E+10	1.48E+09	1.42E+09	1.16E+09	4.01E+08	1.28E+08	7.82E+06	1.48E+06	6.08E+05
(a) / (c)	0.88	1.25	1.08	0.65	0.65	0.56	1.37	0.95	0.74
(b) / (c)	0.86	1.50	1.20	0.46	0.44	0.41	1.05	0.83	0.71

from air (Sv)

year from discharge	0	1	10	100	1000	1.00E+04	1.00E+05	1.00E+06	1.00E+07
(a) 10 years MSR + LWR	5.88E+11	4.73E+11	3.68E+11	1.49E+11	4.70E+10	1.22E+10	1.70E+09	1.76E+08	4.02E+07
(b) 20 years MSR + LWR	7.69E+11	6.13E+11	4.50E+11	1.08E+11	3.25E+10	8.96E+09	1.25E+09	1.46E+08	3.72E+07
(c) LWR only	4.39E+11	3.09E+11	2.79E+11	2.23E+11	7.26E+10	2.14E+10	1.17E+09	1.95E+08	5.91E+07
(a) / (c)	1.34	1.53	1.32	0.67	0.65	0.57	1.45	0.90	0.68
(b) / (c)	1.75	1.98	1.61	0.48	0.45	0.42	1.07	0.75	0.63

TABLE 3.5.9. TOXICITIES FROM FRESH PLUTONIUM FUEL AND THOSE FROM DISCHARGED FUEL OF 200 MWth MSR WITH WEAPONS-GRADE PLUTONIUM

from water (Sv)

year from discharge	0	1	10	1.00E+02	1.00E+03	1.00E+04	1.00E+05	1.00E+06	1.00E+07
due to initially loaded Th-232	4.60E+03	6.23E+03	1.67E+04	2.41E+04	2.41E+04	2.41E+04	2.41E+04	2.41E+04	2.41E+04
due to initially loaded Pu	2.12E+08	2.12E+08	2.12E+08	2.10E+08	1.95E+08	9.95E+07	3.66E+06	2.03E+04	1.88E+04
due to added Pu to 10 years	5.15E+08	5.15E+08	5.15E+08	5.12E+08	4.93E+08	3.46E+08	2.35E+07	9.26E+04	9.11E+04
dose due to Pu (10 years) include Th series	7.27E+08	7.27E+08	7.27E+08	7.22E+08	6.88E+08	3.56E+08	2.72E+07	1.13E+05	1.10E+05
dose from discharged fuel	8.27E+09	2.43E+09	2.13E+09	1.39E+09	5.54E+08	1.85E+08	3.47E+07	1.91E+06	8.36E+04
due to added Pu to 20 years	1.13E+09	1.13E+09	1.13E+09	1.12E+09	1.08E+09	7.63E+08	5.21E+07	2.05E+05	2.01E+05
dose due to Pu (20 years) include Th series	1.34E+09	1.34E+09	1.34E+09	1.33E+09	1.28E+09	8.63E+08	5.58E+07	2.25E+05	2.20E+05
dose from discharged fuel	1.25E+10	5.22E+09	4.45E+09	2.52E+09	1.02E+09	3.51E+08	5.09E+07	3.10E+06	1.35E+05

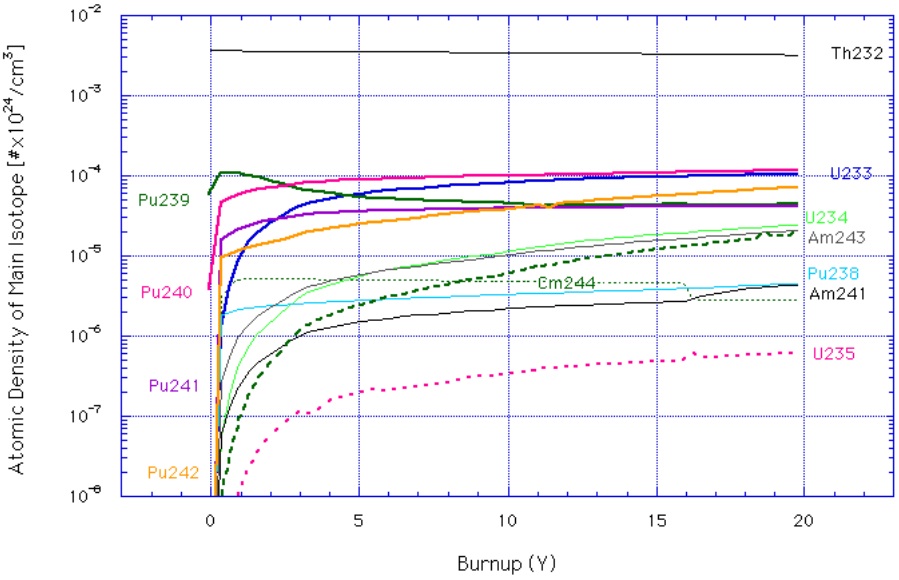

FIG. 3.5.5 Change of atomic density of main isotope with burnup time for MSRWith reactor grade plutonium.

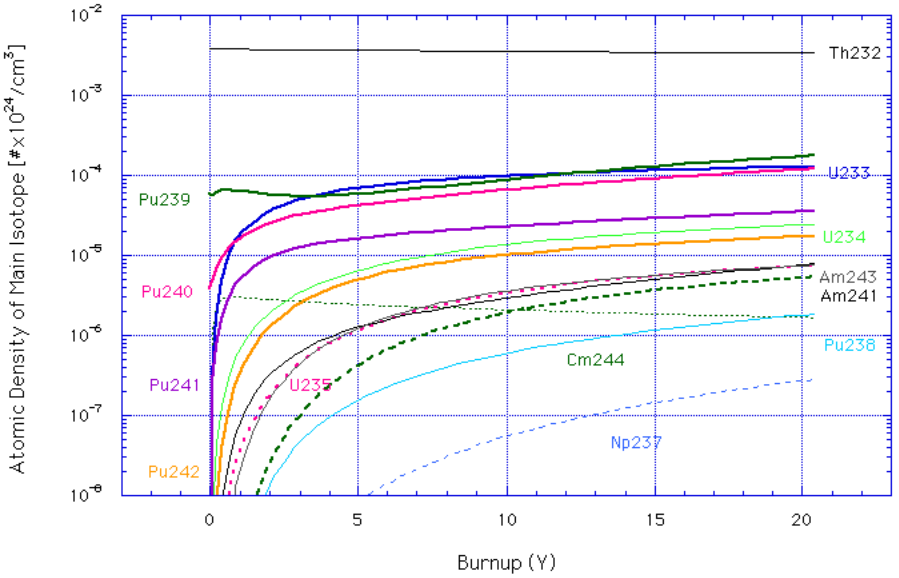

FIG. 3.5.6. Change of atomic density of main isotope with burnup time for MSR with weapons grade plutonium.

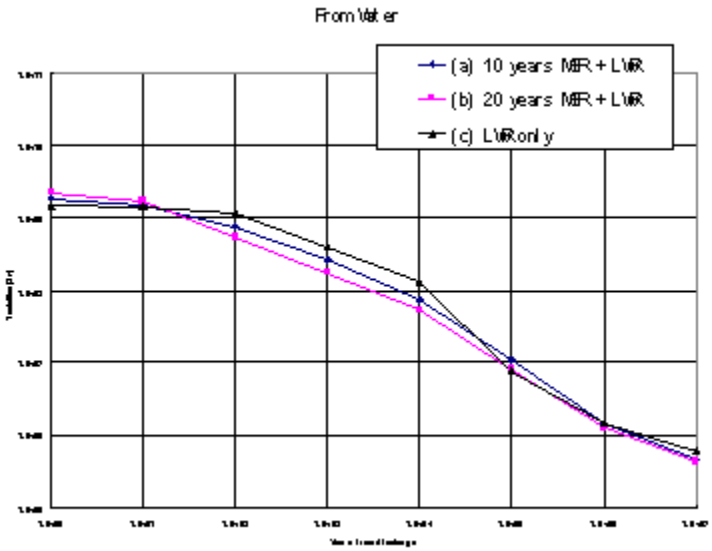

FIG. 3.5.7. Toxicities due to discharged fuel from combined fuel cycle of LWR and MSR with reactor grade plutonium (DVI-W).

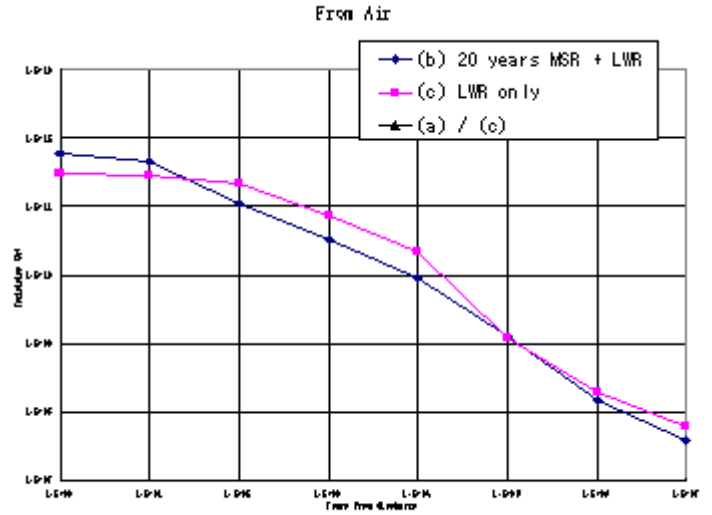

FIG. 3.5.8. Toxicities due to discharged fuel from combined fuel cycle of LWR and MSR with reactor grade plutonium (DVI-A).

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3.6. REPUBLIC OF KOREA

3.6.1. Potential of a thorium based fuel cycle for 900 MW(e) PWR core to incinerate plutonium

3.6.1.1. Introduction

During the second stage of CRP, Republic of Korea investigated the potential of thorium-based fuel to reduce the plutonium in PWR type reactor. A 900 MW(e) PWR currently operated in Republic of Korea was adopted as a reference plant in order to construct the conceptual core with ThO_2-PuO_2. The conceptual core with PuO_2-UO_2 (MOX) was also investigated for the comparison with thorium core. The conceptual cores were assumed to be fully loaded with thorium fuel or MOX fuel. Even though the fully loaded ThO_2-PuO_2 or MOX core concept needs to change the control rod and soluble boron systems to satisfy the current design limit and technical specification, any system design change to meet current design limit was not considered in this study.

In this study, reactor grade plutonium and weapon grade plutonium were considered. The changes in quantity and composition of plutonium isotopes due to fuel burnup were analyzed. The neutronic characteristics of conceptual cores such as power distribution, soluble boron concentration, reactivity parameters, control rod worth etc. were also calculated.

3.6.1.2. Design data for conceptual PWR core

The typical design data for Korean 900 MW(e) PWR were adopted for the conceptual plutonium cores and were summarized in Table 3.6.1. The reactor core is consisted of 157 fuel assemblies, which have 17×17 fuel array. The rated thermal power is 2775 MWth and the system pressure is 150 bars.

As for fuel material data, the typical plutonium composition of PWR spent fuel having burnup of 33 GWd/MtU is used for reactor grade plutonium. Isotopic composition of plutonium in reactor-grade ThO_2-PuO_2 and PuO_2-UO_2 (MOX) fuel is 1.8, 59.0, 23.0, 12.2, and 4.0w/o for ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu, respectively. The composition of weapon-grade plutonium isotopes is 0.0, 94.0, 6.0, 0.0, and 0.0w/o for ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu, respectively.

The plutonium contents of thorium and MOX fuel were determined so that conceptual cores have similar cycle length as uranium core currently being operated with longer than annual fuel cycle scheme. In this study, three types of fuel composition, the thorium and reactor-grade plutonium, the thorium and weapon-grade plutonium, and MOX fuel with reactor grade plutonium, were studied. The total plutonium contents of 7.5, 5.0 and 5.62 w/o were decided for the thorium fuel with reactor grade plutonium, the thorium fuel with weapon-grade plutonium, and MOX fuel with reactor grade plutonium, respectively. The isotopic number densities of each fuel rod are listed in Table 3.6.2.

TABLE 3.6.1. SUMMARY DATA FOR CONCEPTUAL PWR CORE

Core parameter:

Power rating: 2775 MW(t) (900 MW(e))
System pressure: 150 bars
Core average coolant temperature at hot full power: 309.9ºC
Inlet temperature: 291.7ºC
Average enthalpy rise: 35.7ºC
Number of fuel assembly: 157
Assembly pitch at hot state: 21.607 cm
Baffle thickness: 2.8575 cm stainless steel

Fuel assembly data:

Number of rods: 264 fuels, 24 guide tubes, 1 instrumentation tube in 17×17 array
Guide and instrumentation tube inner diameter (hot state); 11.418 mm
Guide and instrumentation tube outer diameter (hot state); 12.260 mm
Material for fuel cladding, guide and instrumentation tube: zircalloy-4
Fuel rod pitch (hot state): 12.66 mm

Fuel rod data (at hot state):

Pellet diameter: 8.05 mm
Active fuel length: 367.30 mm
Cladding inner diameter: 8.236 mm
Cladding outer diameter: 9.518 mm
Cladding material: zircalloy-4

TABLE 3.6.2. ISOTOPIC NUMBER DENSITY IN THORIUM AND MOX FUEL

Fuel type Isotope	Nuclide Number Density (10^{24} atoms/cm^3)		
	Thorium fuel		MOX fuel
	with reactor grade plutonium	with weapons grade plutonium	
^{232}Th	2.1102E-02	2.1670E-02	-
^{235}U	-	-	4.9367E-05
^{238}U	-	-	2.1615E-02
^{238}Pu	2.9936E-05	-	2.3082E-05
^{239}Pu	9.8125E-04	1.0444E-03	7.5657E-04
^{240}Pu	3.8252E-04	6.6420E-05	2.9494E-04
^{241}Pu	2.0291E-04	-	1.5645E-04
^{242}Pu	6.6525E-05	-	5.1293E-05
^{16}O	4.5530E-02	4.5562E-02	4.5893E-02

3.6.1.3. Description of analyses code system

HELIOS/MASTER [1] code system was used for neutronic analysis. HELIOS 1.4 [2] is two dimensional transport code that uses current coupling collision probability method for neutron transport calculation. HELIOS code with 34-neutron group library was used for generation of the group constants for thorium or MOX fuel assemblies.

MASTER [3], a nodal core simulator developed by KAERI, was used for the calculation of core physics with considering thermal hydraulic feedback effect. The original decay chain in MASTER did not include for thorium isotope and it's neighbor isotopes. Therefore the nuclide chain in MASTER code was extended to include ^{232}Th and associated nuclides such as ^{233}Pa, ^{233}U, and ^{234}U for thorium core analysis [4]. Figure 3.6.1 shows the extended nuclide chain in MASTER.

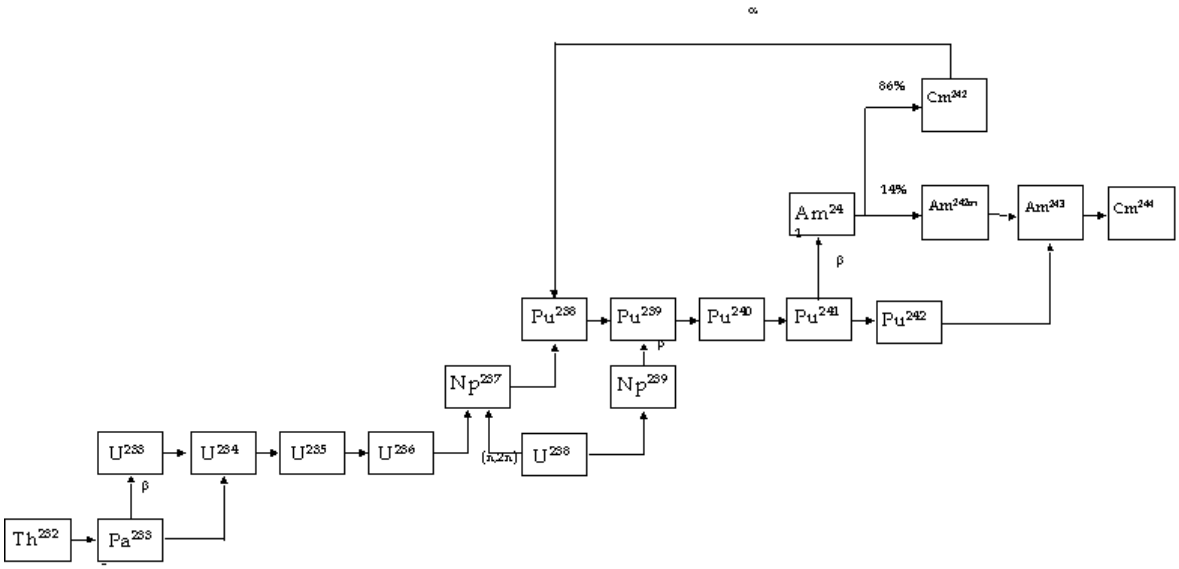

FIG. 3.6.1. The extended heavy nuclides chain in MASTER.

Since the isotopic inventories were calculated with MASTER code, the number of heavy nuclide was restricted within the number of nuclides in the nuclide decay chain shown in Fig. 3.6.1.

3.6.1.4. Fuel management scheme

As described in the previous section, the reference fuel cycle length of conceptual cores was longer than annual. Sixty-four fuel assemblies were discharged from and newly loaded into the reload core for each cycle. Some fresh fuel assemblies bear four or eight gadolinium rods as burnable poison rod to control excess core reactivity and core power distribution. The fuel cycle characteristics of thorium and MOX cores are summarized in Table 3.6.3.

The low-leakage loading strategy in which most of fresh fuel assemblies take inboard locations was applied. Figures 3.6.2 and 3.6.3 show the loading pattern of equilibrium core.

TABLE 3.6.3. FUEL CYCLE CHARACTERISTICS FOR THORIUM AND MOX CORES

Fuel cycle	Thorium core		MOX core
	with reactor grade plutonium	with weapons grade plutonium	
Core characteristics			
Number of fuel asseamblies in a core			
Thorium or MOX fuel assembly	157	157	157
Number of fresh fuel assemblies			
Without gadolinium	32	36	32
With 4 gadolinium	12	-	12
With 8 gadolinium	20	28	20
Fuel assemblies apecification			
Total plutonium content in fuel (w/o)	7.50	5.00	5.62
Fissile plutonium content in fuel (w/o)	5.34	4.70	4.00
Equilibrium cycle length (EFPD)	401	361	393
Fuel burnup (MWD/MtM)			
Batch burnup	40.48	36.40	38.37
Assembly maximum burnup	52.68	45.80	49.58

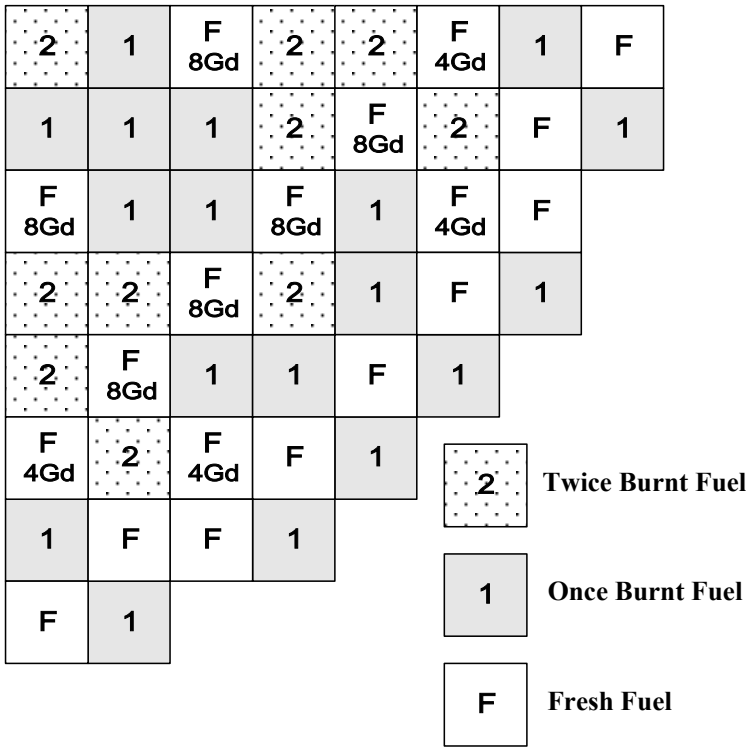

FIG. 3.6.2. Loading pattern for equilibrium core with thorium or MOX fuel with reactor grade plutonium.

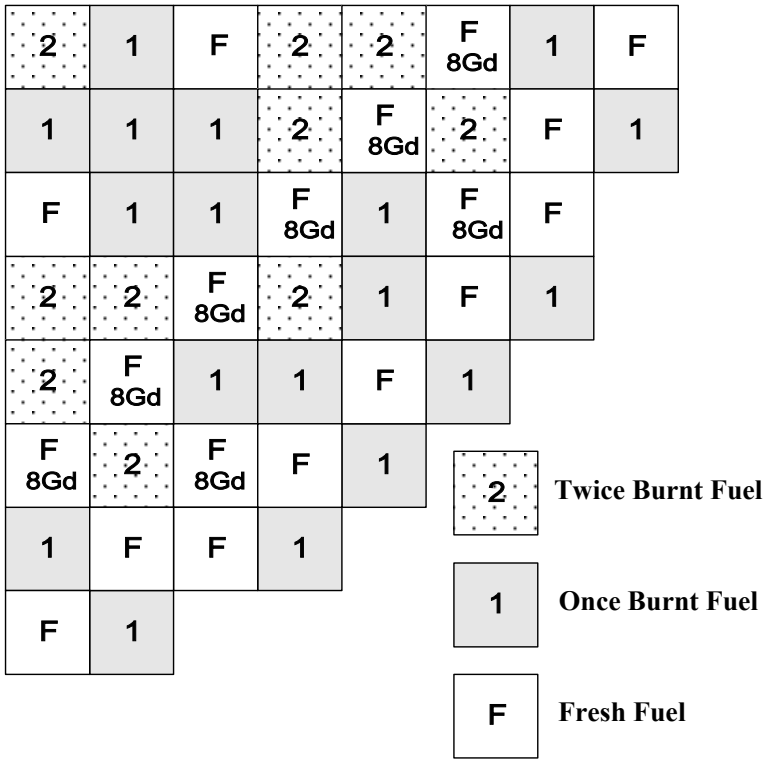

FIG. 3.6.3. Loading pattern for equilibrium core with thorium fuel with weapons grade plutonium.

3.6.1.5. Nuclear characteristics of conceptual cores

The infinite multiplication factors for two types of thorium fuel assemblies and MOX fuel assembly were calculated by HELIOS code. The soluble boron concentration was kept constant as 500 ppm for fuel burnup calculation. Figure 3.6.4 shows the criticality curves with burnup for thorium and MOX fuel assemblies.

The critical soluble boron concentrations for the equilibrium cores loaded with thorium cores and MOX core were shown in Fig. 3.6.5. In case of thorium core with weapon grade plutonium, the consumption of ^{239}Pu in is much larger than the conversion of fertile isotopes to fissile during core burnup, and the boron concentration was rapidly decreased as compared with the other conceptual core fuelled with reactor grade plutonium.

Assembly-wise radial power distributions for equilibrium cores were shown in Figs 3.6.6-3.6.8. The local power distribution and related local fuel and cladding temperature calculations were not performed.

Key core physics parameters such as soluble boron concentration, temperature coefficients, boron worth, and control rod worth were calculated with MASTER code and are listed in Table 3.6.4. The neutron spectrum of conceptual cores fuelled with plutonium is harder than that of uranium fuelled core. Since harder neutron spectrum enhances the neutron leakage from the core, the temperature coefficients of the conceptual cores are more negative than that of UO_2 core. Since boron is strong absorber for thermal neutron, boron worth is also strongly affected by neutron spectrum. The boron worth of conceptual cores are about half of nominal value of uranium fuelled core because of harder neutron spectrum. Control rod, which is also strong thermal neutron absorber, in the conceptual core, has less worth than in UO_2 core.

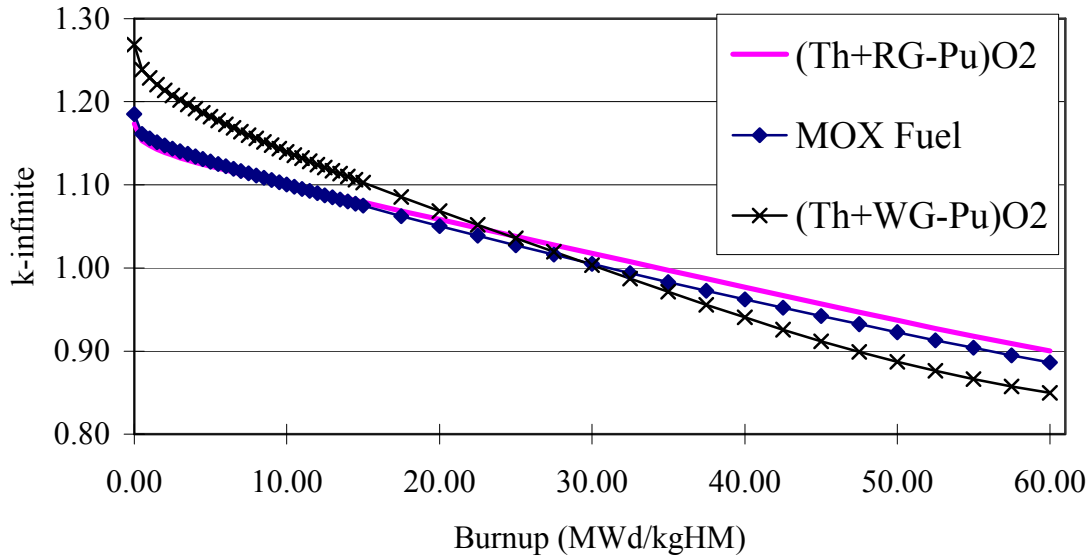

FIG. 3.6.4. Infinite multiplication factor with fuel burnup (constant 500 ppm of soluble boron).

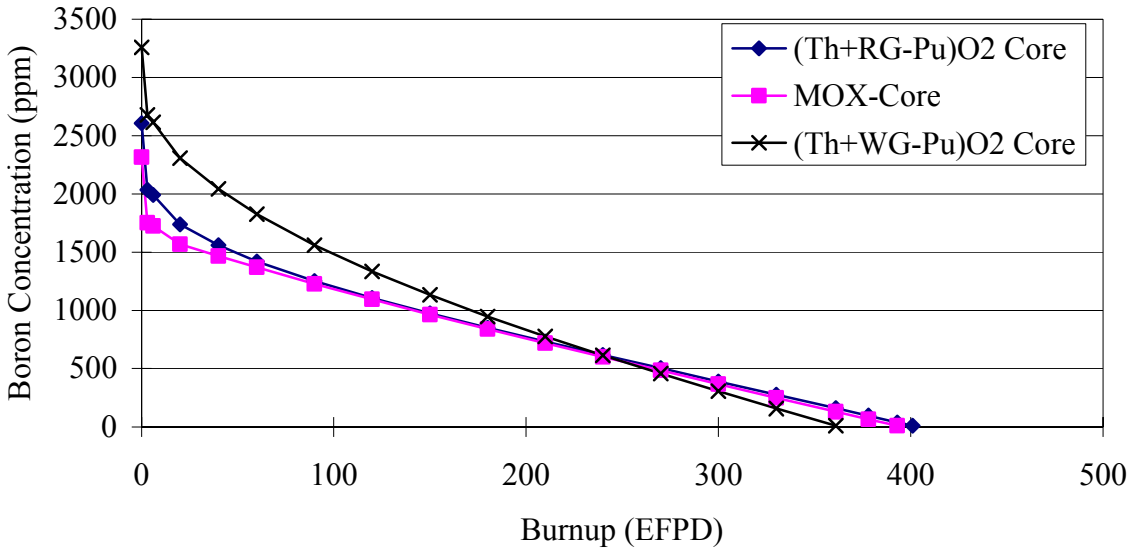

FIG. 3.6.5. Critical boron concentration for thorium and MOX cores with core burnup.

0.970 0.986 0.958	1.105 1.112 1.075	1.250 1.247 1.226	1.125 1.122 1.105	1.153 1.140 1.128	1.260 1.237 1.233	1.013 1.013 1.017	0.598 0.626 0.669
1.105 1.113 1.075	1.134 1.135 1.092	1.167 1.160 1.122	1.147 1.138 1.117	1.278 1.262 1.260	1.102 1.084 1.076	0.998 0.992 1.003	0.442 0.475 0.522
1.250 1.247 1.226	1.167 1.161 1.122	1.226 1.212 1.174	1.286 1.272 1.260	1.166 1.150 1.126	1.164 1.147 1.148	0.795 0.804 0.839	
1.125 1.122 1.105	1.146 1.138 1.117	1.281 1.268 1.256	1.097 1.090 1.069	1.069 1.068 1.053	0.958 0.959 0.970	0.464 0.494 0.538	
1.153 1.140 1.128	1.278 1.262 1.260	1.163 1.147 1.124	1.065 1.064 1.049	0.946 0.955 0.961	0.521 0.557 0.601		
1.261 1.238 1.234	1.103 1.085 1.077	1.163 1.146 1.147	0.956 0.958 0.969	0.520 0.556 0.600			
1.014 1.014 1.019	1.000 0.994 1.006	0.796 0.805 0.839	0.463 0.494 0.538			BOC MOC EOC	
0.597 0.625 0.668	0.443 0.476 0.522						

FIG. 3.6.6. Assembly-wise power distribution for the thorium fueled core with reactor grade plutonium.

0.686 0.843 0.881	0.883 1.019 1.027	1.184 1.236 1.211	0.967 1.023 1.029	1.012 1.018 1.025	1.329 1.249 1.231	1.133 1.068 1.045	0.732 0.729 0.755
0.883 1.019 1.027	0.896 1.022 1.025	0.981 1.066 1.060	1.003 1.044 1.046	1.292 1.256 1.247	1.102 1.051 1.039	1.182 1.091 1.068	0.502 0.523 0.564
1.184 1.235 1.210	0.981 1.066 1.060	1.082 1.135 1.120	1.271 1.266 1.252	1.127 1.112 1.094	1.295 1.209 1.188	0.956 0.903 0.914	
0.967 1.021 1.028	0.998 1.040 1.043	1.271 1.266 1.252	1.033 1.052 1.044	1.039 1.044 1.032	1.089 1.045 1.035	0.507 0.531 0.573	
1.012 1.018 1.024	1.288 1.252 1.243	1.127 1.112 1.094	1.040 1.045 1.033	1.032 1.025 1.017	0.548 0.591 0.636		
1.333 1.253 1.235	1.105 1.053 1.040	1.296 1.210 1.189	1.090 1.045 1.036	0.548 0.591 0.636			
1.135 1.071 1.048	1.189 1.096 1.073	0.958 0.904 0.915	0.507 0.531 0.573			BOC MOC EOC	
0.730 0.727 0.753	0.503 0.523 0.565						

FIG. 3.6.7. Assembly-wise power distribution for the thorium fueled core with weapons grade plutonium.

0.899 0.958 0.956	1.038 1.087 1.072	1.219 1.250 1.233	1.076 1.111 1.106	1.126 1.138 1.131	1.293 1.257 1.234	1.045 1.019 1.015	0.637 0.636 0.673
1.039 1.087 1.072	1.066 1.108 1.087	1.110 1.142 1.120	1.101 1.129 1.119	1.287 1.279 1.263	1.099 1.085 1.075	1.049 1.009 1.007	0.461 0.473 0.520
1.220 1.250 1.233	1.110 1.142 1.120	1.182 1.200 1.172	1.281 1.282 1.262	1.152 1.144 1.123	1.206 1.165 1.151	0.842 0.820 0.845	
1.076 1.112 1.107	1.101 1.129 1.120	1.276 1.278 1.258	1.066 1.075 1.064	1.059 1.053 1.043	0.997 0.969 0.972	0.479 0.490 0.536	
1.126 1.138 1.131	1.287 1.279 1.264	1.149 1.141 1.121	1.054 1.048 1.039	0.973 0.958 0.962	0.533 0.551 0.599		
1.295 1.258 1.235	1.101 1.086 1.076	1.206 1.164 1.150	0.995 0.968 0.971	0.532 0.550 0.598			
1.047 1.021 1.017	1.053 1.012 1.009	0.843 0.820 0.845	0.478 0.490 0.535			BOC MOC EOC	
0.635 0.635 0.672	0.462 0.474 0.520						

FIG. 3.6.8. Assembly-wise power distribution for MOX core.

TABLE 3.6.4. KEY CORE PHYSICS PARAMETER FOR THORIUM AND MOX CORE

Fuel cycle Core characteristics	Thorium core		MOX core
	with reactor grade plutonium	with weapons grade plutonium	
Boron concentration (ppm)			
To control at HZP, ARO, (k = 1.0)	3259	3704	2853
To control at HZP, ARI, (k = 1.0)	1405	1948	1141
To control at HFP, ARO, (k = 1.0)			
0 EFPD, No Xenon	2609	3258	2318
6 EFPD, Eq. Xenon	1992	2617	1724
Moderator temp. coefficient at HFP (pcm/^{o}C) at BOC/ EOC	-36.2/-67.2	-20.5/-62.5	-44.2/-79.8
Isothermal temp. coefficient at HZP (pcm/^{o}C) at BOC	-13.6	-2.5	-19.6
Fuel temp. coefficient at HFP (pcm/^{o}C) at BOC/ EOC	-3.74/-3.87	-3.50/-3.78	-3.04/3.20
Boron value at HFP (pcm/^{o}C) at BOC/ EOC	-3.05/-4.18	-3.63/-5.50	-3.50/-4.52
Total control rod value at HFP (pcm) at BOC/EOC	6618/7491	7290/7576	7048/7855

3.6.1.6. Change of heavy nuclide mass

The change in heavy nuclide mass for thorium and MOX fuel batches between beginning of irradiation and end of irradiation are listed in Table 3.6.5. The mass of ^{233}Pa that has a short half-life of 27-days was added to the mass of ^{233}U isotope.

Table 3.6.6. shows the mass change of heavy isotopes between BOC and EOC for the equilibrium core.

As noticed in Section 3.6.1.4, each conceptual core has different fuel cycle length. In order to compare the mass change under the same condition, the mass values in Tables 3.6.5 and 3.6.6. were adjusted to be equivalent to 1 GW-300 EFPD (Effective Full Power Day).

TABLE 3.6.5. MASSES OF HEAVY NUCLIDE AT THE BEGINNING AND END OF IRRADIATION

Isotope	Mass (kg)					
	Thorium fuel				MOX fuel	
	with reactor grade plutonium		with weapons grade plutonium			
	BOI*	EOI**	BOI	EOI	BOI	EOI
Th-232	20980	20489.3	23932.1	23383.5	0.0	0.0
U-233	0.0	301.2	0.0	326.1	0.0	0.0
U-234	0.0	26.3	0.0	30.7	0.0	0.0
U-235	4.9	7.6	5.8	8.6	55.8	27.9
U-236	0.0	0.8	0.0	1.0	0.0	6.4
Np-237	0.0	0.1	0.0	0.1	0.0	2.7
U-238	267.0	259.7	319.8	311.8	22806.1	22198.3
Pu-238	30.6	26.5	0.0	2.8	24.1	21.4
Np-239	0.0	0.0	0.0	0.0	0.0	2.7
Pu-239	1005.0	237.5	1188.2	202.3	792.0	407.6
Pu-240	393.5	296.1	75.9	171.1	310.0	283.2
Pu-241	209.6	199.4	0.0	101.3	165.1	180.7
Am-241	0.0	15.4	0.0	4.7	0.0	12.4
Am-242	0.0	0.4	0.0	0.1	0.0	0.3
Cm-242	0.0	4.2	0.0	1.4	0.0	3.4
Pu-242	69.0	115.5	0.0	29.2	54.4	99.2
Am-243	0.0	22.9	0.0	5.6	0.0	18.2
Cm-244	0.0	11.9	0.0	1.9	0.0	10.4
Sum	22959.6	22014.8	25521.8	24582.2	24207.5	23274.8

*BOI: Beginning of Irradiation, **EOI: End of Irradiation

TABLE 3.6.6. MASSES OF HEAVY NUCLIDE AT THE BEGINNING AND END OF CYCLE IN EQUILIBRIUM CORE

Isotope	Mass (kg)					
	Thorium core				MOX core	
	with reactor grade plutonium		with weapons grade plutonium			
	BOC	EOC	BOC	EOC	BOC	EOC
Th-232	51253.7	50763.2	58452.2	57903.5	0.0	0.0
Pa-233	5.0	45.7	6.5	57.8	0.0	0.0
U-233	248.4	508.8	283.2	558.1	0.0	0.0
U-234	12.8	39.0	16.3	47.0	0.0	0.0
U-235	10.0	12.7	12.2	14.9	113.9	86.0
U-236	0.3	1.2	0.4	1.4	4.8	10.8
Np-237	0.0	0.1	0.0	0.1	2.1	4.7
U-238	535.5	528.2	646.6	638.7	55535.5	54927.5
Pu-238	70.7	66.5	0.4	1.8	55.6	52.6
Np-239	0.0	0.1	0.0	0.1	0.0	4.6
Pu-239	1807.7	1040.3	2004.9	1018.7	1618.2	1229.0
Pu-240	939.9	842.5	342.7	437.9	764.3	737.4
Pu-241	534.7	526.3	108.9	210.7	432.3	449.3
Am-241	20.0	32.5	4.1	7.7	15.5	25.6
Am-242	0.3	0.7	0.1	0.2	0.2	0.6
Cm-242	1.6	6.0	0.3	1.8	1.3	4.9
Pu-242	198.0	244.5	12.2	41.3	162.3	206.4
Am-243	17.9	40.8	1.6	7.2	14.0	32.1
Cm-244	5.1	17.0	0.3	2.2	4.2	14.7
Sum	55661.6	54716.1	61892.9	60951.1	58724.2	57786.2

3.6.1.7. Results and discussion

In order to investigate the potential of thorium based fuel for 900 MW(e) PWR to reduce the plutonium, the mass balance of plutonium isotope for thorium fuel was compared with that for MOX fuel and the results are shown in Table 3.6.7.

For the thorium fuel with reactor grade plutonium, the annual charged and discharged mass of plutonium are 1708 and 875 kg, respectively, which means 833 kg of plutonium is incinerated annually by 300 EFPD operation of one 1 000 MW(e) PWR. The incineration rate of plutonium for thorium core with weapon grade plutonium and MOX core are 757 and 351 kg per 1 GWe-300 EFPD, respectively. Therefore, thorium fuelled core can consume plutonium 2.2 or 2.4 times larger than MOX core. The fissile plutonium fraction change in thorium fuel is also twice or three times larger than in MOX fuel.

Based on these results, it is concluded that thorium fuelled PWR core has higher potential to reduce plutonium than MOX PWR core.

TABLE 3.6.7. PLUTONIUM MASS BALANCE

	Mass (kg)		
	Thorium core		MOX core
	with reactor grade plutonium	with weapons grade plutonium	
Plutonium charged	1708	1264	1346
Plutonium discharged	875	507	995
Plutonium burned	833	757	351
Fissile fraction for plutonium charged (%)	72	94	72
Fissile fraction for plutonium discharged (%)	51	60	61

3.6.2. Assessment of the effect of plutonium incineration on the long lived waste toxicity

3.6.2.1. Calculation procedure

In Section 3.6.2, the long lived waste toxicity of thorium based fuel cycle was evaluated. In order to do this, a combined system model with conventional UO_2 — and with $(Th+Pu)O_2$-fuelled reactor was applied. Since the plutonium produced from the conventional UO_2-fuelled PWR can be recycled into $(Th+Pu)O_2$ core or MOX core, the combined system is consisted of conventional UO_2 core as plutonium supplier and of $(Th+Pu)O_2$ core (or MOX core) as plutonium burner. For the comparison purpose, a conventional UO_2 reactor as a reference system and an UO_2+MOX combined system were also considered. So, the toxicity of the long lived waste from the following three scenarios were calculated and compared.

3.6.2.1.1. Scenario 1: Conventional UO_2 only system

A typical PWR fuelled with UO_2 is adopted as conventional UO_2 system. The waste from this system is assumed to be disposed without separation of any isotopes.

3.6.2.1.2. Scenario 2: Conventional UO_2 + $(Th+Pu)O_2$ (as plutonium burner) combined system

A combined system, which has the same size with a conventional UO_2 system, with certain fractions of UO_2 unit and of $(Th+Pu)O_2$ unit is considered. The plutonium of the spent fuel from UO_2 unit is separated and recycled into plutonium burner, $(Th+Pu)O_2$ unit, as illustrated in Fig. 3.6.9. The waste of this system is the heavy metal with plutonium separation from UO_2 unit and the spent fuel for $(Th+Pu)O_2$ unit.

3.6.2.1.3. Scenario 3: Conventional UO_2 + MOX (as plutonium burner) combined system

This system is the same one as Scenario 2 except that MOX unit is adopted as plutonium burner instead of $(Th+Pu)O_2$ unit in Scenario 2.

The plutonium discharge rate of one conventional UO_2-fuelled PWR is assumed to be 245 kg of plutonium per one GWa (300 EFPD). According to Section 3.6.1 calculation, the loading rates of plutonium are 1708 kg Pu/Gwa for one $(Th+Pu)O_2$ plutonium burner and 1346 kg Pu/Gwa for one MOX plutonium burner. Therefore, the number of the conventional UO_2 reactors required to supply the plutonium to one plutonium burner are 7.0 for a thorium/plutonium burner and 5.5 for a MOX plutonium burner.

The fractions of UO_2 unit and of plutonium burner unit in a combined system has to be decided to balance the plutonium between discharged from UO_2 unit and loaded into plutonium burner unit, and to have the same size with a conventional UO_2 system.

So, a combined system with conventional UO_2 and with $(Th+Pu)O_2$ (Scenario 2) is composed of 0.875 UO_2 units and 0.125 $(Th+Pu)O_2$ units, and a combined system with conventional UO_2 and with MOX (System 3) is composed of 0.8462 UO_2 units and 0.1538 MOX units.

3.6.2.2. Toxicity results for plutonium incineration systems

The results of toxicity calculation for each scenarios are given in Tables 3.6.9–3.6.10 for the radioactivity, the ingestion hazard, and the inhalation hazard, respectively. These are also illustrated in Figs 3.6.10–3.6.12.

For the near-term (~ 10^2 years) after discharge, Pu-238, Pu-241, Am-241, and Cm-244 dominate the toxicity. For this period, the toxicity of combined system is rather higher than that of conventional UO_2-fuelled PWR due to higher content of Cm-244. For the mid-term ($10^2 \sim 10^5$ years) after discharge, Pu-239, Pu-240, and Am-241 dominate the toxicity. For this period, the toxicity of combined system is lower than that of conventional UO_2-fuelled PWR due to the effect of plutonium incineration. For the long term ($10^5 \sim 10^6$ years) after discharge, Pu-239 and Th-229 are the major sources of the toxicity. For this period, the toxicity of combined system with $(Th+Pu)O_2$ unit is getting higher than that of conventional UO_2-fuelled PWR due to the decay effect of the daughter isotopes of U-233.

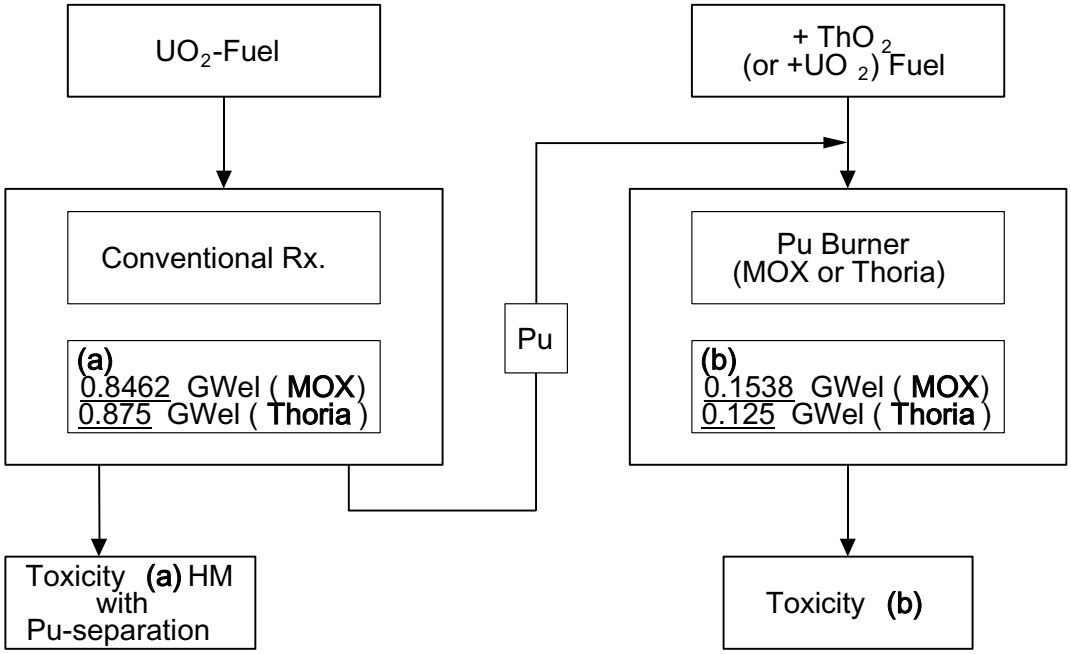

FIG. 3.6.9. Diagram of combined system.

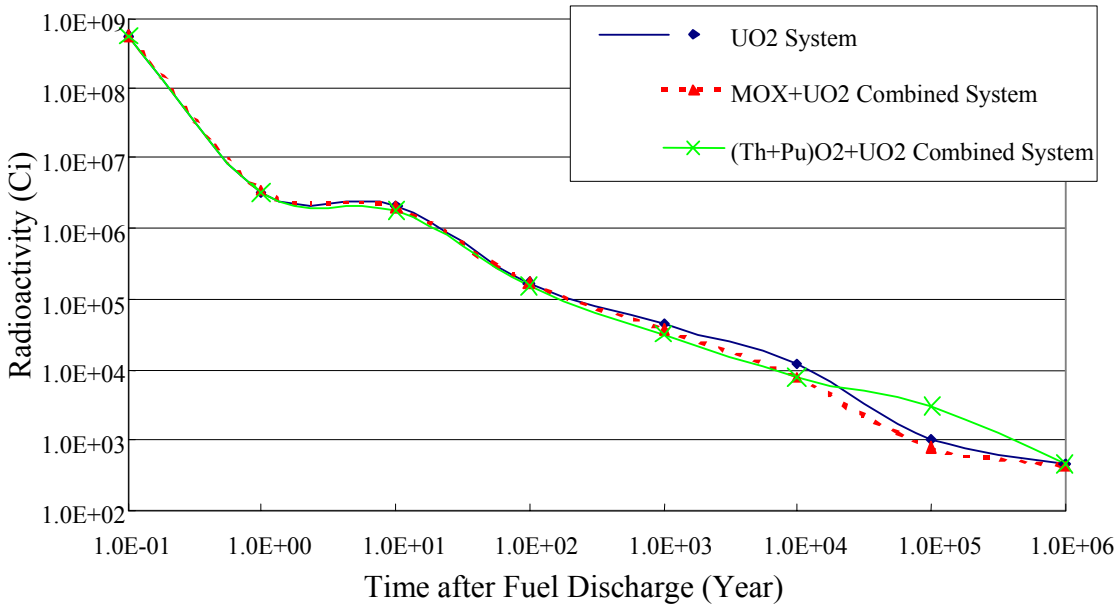

FIG. 3.6.10. Radioactivity of each scenario, (Ci).

TABLE 3.6.8. RADIOACTIVITY OF EACH SCENARIO (Ci)

Time (years)	0.1	1	10	100	1000	10 000	100 000	1 000 000
Scenario 1:	5.547E+08	3.346E+06	2.126E+06	1.721E+05	4.441E+04	1.193E+04	1.043E+03	4.458E+02
Scenario 2:	5.554E+08	3.232E+06	1.828E+06	1.538E+05	3.241E+04	7.818E+03	2.985E+03	4.616E+02
Scenario 3:	5.670E+08	3.516E+06	2.021E+06	1.676E+05	3.711E+04	7.757E+03	7.816E+02	4.260E+02

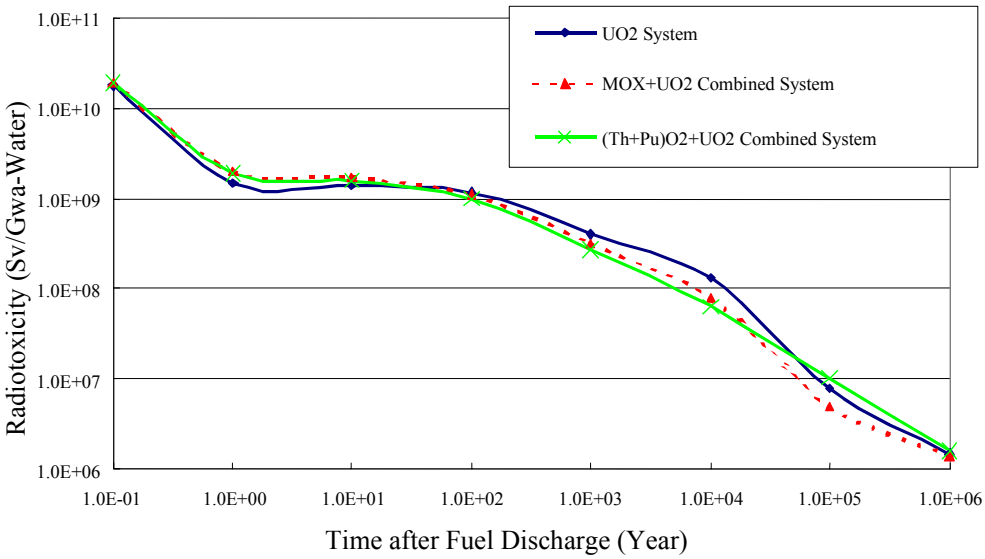

FIG. 3.6.11. Ingestion hazard of each scenario, (Sv/Gwa-water).

TABLE 3.6.9. INGESTION HAZARD OF EACH SCENARIO (SV/GWA-WATER)

Time (years)	0.1	1	10	100	1000	10 000	100 000	1 000 000
Scenario 1:	1.795E+10	1.487E+09	1.427E+09	1.160E+09	4.021E+08	1.286E+08	7.823E+06	1.488E+06
Scenario 2:	1.907E+10	1.913E+09	1.600E+09	9.999E+08	2.670E+08	6.220E+07	1.037E+07	1.570E+06
Scenario 3:	1.927E+10	2.030E+09	1.716E+09	1.101E+09	3.178E+08	7.937E+07	4.981E+06	1.440E+06

TABLE 3.6.10. INHALATION HAZARD OF EACH SCENARIO (SV/GWA-AIR)

Time (years)	0.1	1	10	100	1000	10 000	100 000	1 000 000
Scenario 1:	4.413E+11	3.096E+11	2.790E+11	2.240E+11	7.275E+10	2.146E+10	1.171E+09	1.952E+08
Scenario 2:	8.531E+11	4.808E+11	3.486E+11	1.971E+11	4.964E+10	1.053E+10	1.626E+09	1.994E+08
Scenario 3:	8.674E+11	5.044E+11	3.719E+11	2.160E+11	5.857E+10	1.331E+10	6.981E+08	1.848E+08

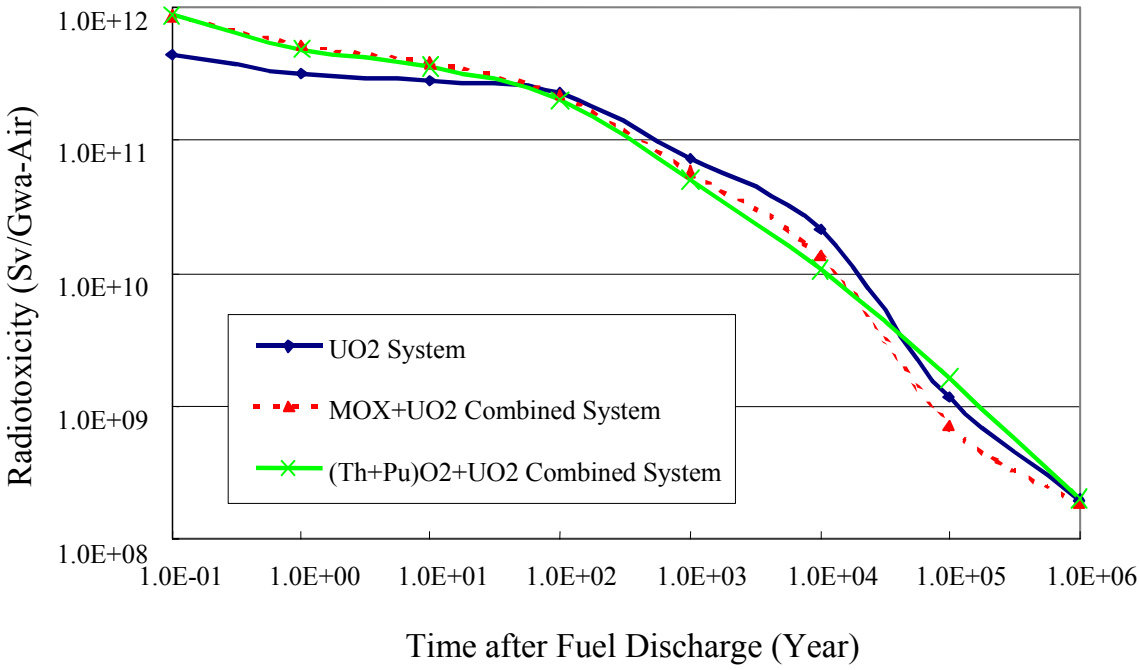

FIG. 3.6.12. Inhalation hazard of each scenario, (Sv/Gwa-air).

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[2] MICHELBACHER, J.A., HENSLEE, S.P., MCDERMOTT, M., ROSENBERG, K.E., PRICE, J.R., WELLS, P.B., The sodium process facility at Argonne National Laboratory-West, paper presented in the Consultancy on Technical options for the decommissioning of the BN-350 LMFR, 23–27 February 1998, Obninsk, Russian Federation.

[3] SONG, J.S., et al., Verification and Uncertainty Evaluation of HELIOS/MASTER Nuclear Design System, KAERI/TR–1310/99, Korea Atomic Energy Research Institute (1999).

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3.7.1. Calculations of the principal neutronics characteristics of the WWER-1000 reactor loaded with PuO_2–ThO_2 fuel based on weapons grade plutonium

3.7.1.1. Reactor design description

The simplest method to involve weapons grade plutonium and thorium in fuel cycle was considered — full or partial replacement of uranium fuel of WWER-1000 reactor with plutonium and thorium dioxides mixture. Any lattice optimization was not examined.

The study concerns 1000 MW(e) reactor with three-batch core management and one year cycles duration. Two reactor options: full and partial (1/3 of core) loading of PuO_2-ThO_2 fuel in WWER-1000 reactor were investigated.

The option with full inventory has the homogenous core with only one type of fuel assembly thus avoiding fuel zoning in fuel assemblies (FAs). The partial thorium core consists of two types of FAs: uranium dioxide fuelled FA (zoned FA with average enrichment 4.23%) and zoned FA with plutonium-thorium fuel (average content of plutonium is 4.8%). Zoned UO_2 fuelled FA is the improved FA of a standard WWER-1000 reactor. These FAs differ from those of standard reactor by the replacement of steel in guide tubes and spacer grids with zirconium and slight changes in guide tubes, central tube and absorber pin dimensions as well. As in the case of standard UO_2 fuel reactor, the first year FAs use boron burnable poison rods that are removed in reloading. Reactivity change when burning is controlled by dissolved boron.
Tables 3.7.1 and 3.7.2 present the basic characteristics of the core and fuel assemblies [1]. Weapons grade plutonium composition at the moment of reactor loading is taken as follows (in weight%) [2]:

^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am
0.02	93.94	5.81	0.18	0.03	0.02

The FA maps are presented in Figs 3.6.1 – 3.6.3. Figures 3.6.4 and 3.6.5 show the fuel assemblies arrangement and reloading patterns in reactor core (1/6 part). In reactor calculations, the operating group rods (FA № 4) are inserted at 71 cm core depth during the whole cycle.

TABLE 3.7.1. CORE CHARACTERISTICS

Thermal power, MW(t)	3000
Core size (hot state):	
Height, m	3.55
Effective diameter, m	3.16
Number of Fuel Assemblies (Fas)	163
Coolant:	
Pressure, Mpa	15.7
Coolant mass flow, m^3/h	84500
Inlet temperature, °C	287
Average enthalpy rise, K	30.3

TABLE 3.7.2. FUEL ASSEMBLY FEATURES

Geometry	hexagonal
Assembly pitch, mm	236
Dimension across flats, mm	234
Fuel rod	
Number	312
Fuel rod pitch, mm	12.75
Outer diameter, mm	9.1
Cladding	
Thickness, mm	0.69
Material	zircalloy
Fuel pellet	
Diameter/central hole diameter, mm	7.53/2.3 *)
Guide tube for control rods and burnable poison rods	
Number	18
Outer diameter/wall thickness, mm	13.58/0.85
Material	zircalloy
Central tube	
Outer diameter/wall thickness, mm	11.2/0.8
Material	zircalloy
Control rod	
Number	18
Outer diameter/cladding thickness, mm	8.2/0.50
Absorber/cladding material	B_4C/stainless steel
Absorber diameter, mm	7.2
Boron carbide density, g/cm^3	1.8
Burnable poison rod (BPR)	
Number	18
Outer diameter/cladding thickness, mm	9.1/0.69
Absorber/cladding material	CrB_2 +Al_2O_3/zircalloy
Boron density, g/cm^3	0.036
Space grids	
Number within the core	14
Material/mass, kg	zircalloy/0.53

*) For fuel pins with UO_2 fuel, central hole in fuel pins with ThO_2-PuO_2 was absent

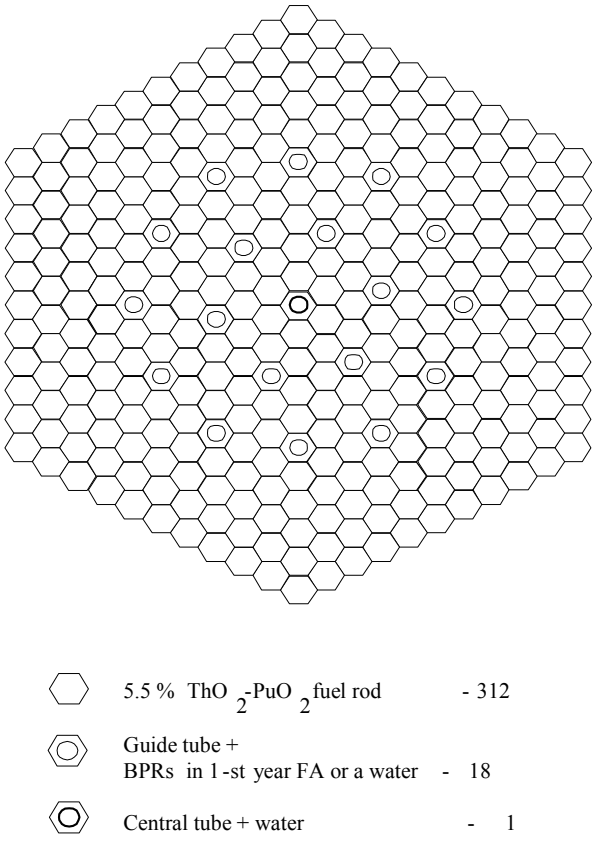


FIG. 3.7.1. Map of nonzoned ThO_2-PuO_2 assembly (full ThO_2-PuO_2 inventory).

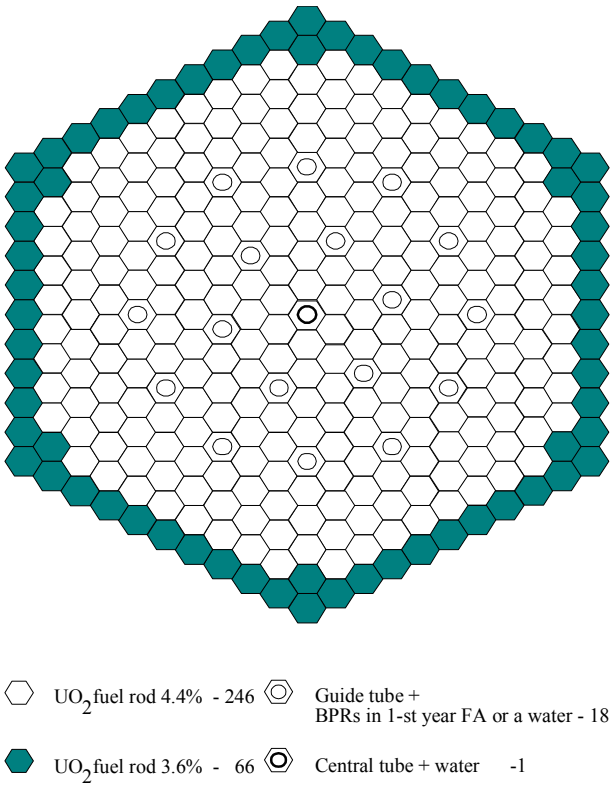


FIG. 3.7.2. Map of zoned UO_2FA (standard WWER-1000 FA).

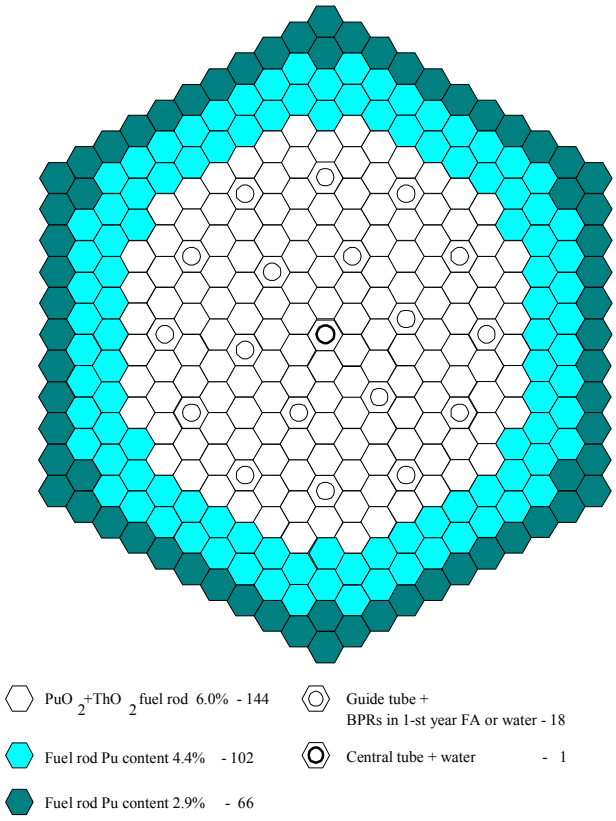


Fig. 3.7.3. Map of zoned fuel assembly with ThO_2-PuO_2 fuel.

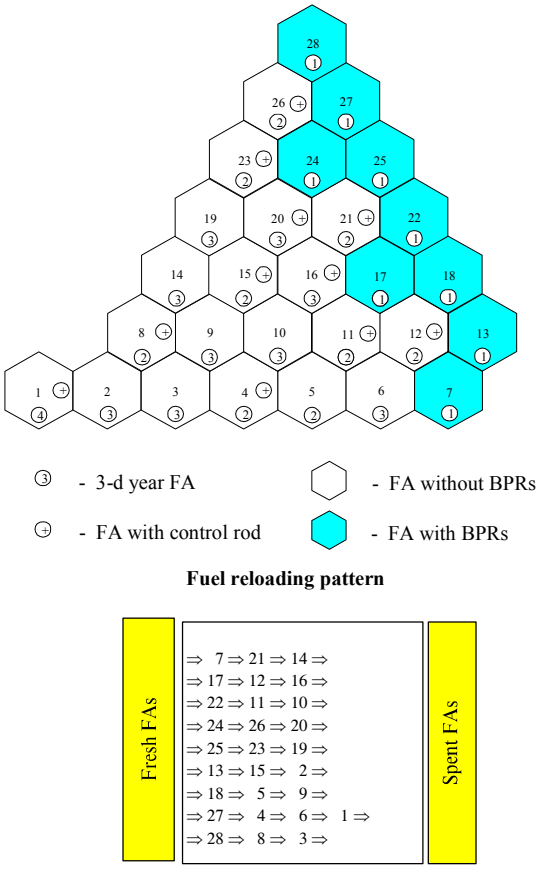
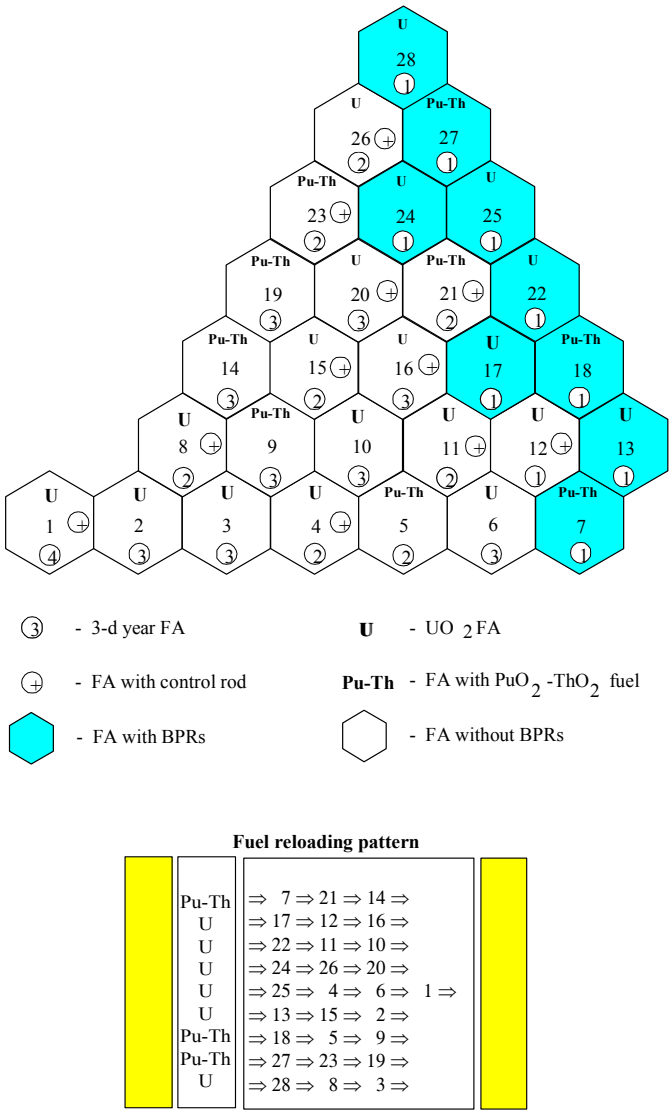


Fig. 3.7.4. FA arrangement in reactor with full load of PuO_2-ThO_2 fuel(1/6 part of core).



Fuel reloading pattern

Pu-Th	⇒ 7 ⇒ 21 ⇒ 14 ⇒
U	⇒ 17 ⇒ 12 ⇒ 16 ⇒
U	⇒ 22 ⇒ 11 ⇒ 10 ⇒
U	⇒ 24 ⇒ 26 ⇒ 20 ⇒
U	⇒ 25 ⇒ 4 ⇒ 6 ⇒ 1 ⇒
U	⇒ 13 ⇒ 15 ⇒ 2 ⇒
Pu-Th	⇒ 18 ⇒ 5 ⇒ 9 ⇒
Pu-Th	⇒ 27 ⇒ 23 ⇒ 19 ⇒
U	⇒ 28 ⇒ 8 ⇒ 3 ⇒

FIG. 3.7.5. FA arrangement in reactor with partial load ofPuO_2-ThO_2 fuel(1/6 part of core).

3.7.1.2. Codes used in neutronics calculations

The calculation of neutronics characteristics was carried out using WIMS-ABBN, PARSEC, TRIANG-PWR and CREDE codes.

Code WIMS-ABBN [3] is a modernized WIMS-D4 [4]. The modernization was done to include minor actinide chains which were absent in an original version, and to take account the first resonance of ^{242}Pu with E_0 = 2.65 eV self-shielding. At the same time, the principle nuclide constants were updated in the code library. The intercomparisons made with calculated results using other codes including plutonium-thorium benchmarks coordinated by the IAEA, as well as with the experimental benchmarks give grounds to hope that the modifications aforementioned allow to use the WIMS-ABBN for calculating thermal reactors with fuels of any nuclide composition, in particular, with plutonium/thorium fuel.

The FAs burnup calculations were performed using the WIMS-ABBN code, and the macroscopic cross-sections of FAs were obtained at various combinations of core parameters that describe reactor state (water density, temperatures of water and fuel, concentrations of

dissolved boron et al.). The results of calculations of macroscopic cross-sections were put in PARSEC code to determine the approximation coefficients. The calculation of the reactor was carried out by three dimensional diffusion code TRIANG-PWR using the approximated macroscopic cross-sections.

CREDE code was used to calculate nuclide concentrations that were absent in WIMS calculations and leave the neutron balance unaffected, but were important from the point of view of radiotoxicity, for example ^{232}U. In addition this code corrects some flows of WIMS-calculations of nuclide concentration evolution, for example, WIMS code cannot treat branching in the capture process and the reaction (n,2n) cannot be considered if the reaction (n, γ) has been included. The list of nuclides involved in calculations is significantly enlarged (from stable isotopes of Pb and Bi up to ^{245}Cm). Besides, CREDE calculates nuclide composition and radiotoxicity of spent fuel over a long term storage. The heavy nuclide chains, which are taken into account in CREDE code, are presented in Fig. 3.7.6.

3.7.1.3. Results of calculations

Table 3.7.3 shows FA criticality change when burning.

The k_{inf} calculations were made under the following conditions:

(a) fissile material content in fuel rods was assumed to be equal to average value, i.e., zoning in FAs were not taken into account;

(b) boric acid concentration was assumed to be constant and equal to average one over the cycle;

(c) interreloading intervals were not taken into account, i.e. concentrations of ^{233}Pa and ^{233}U corresponded to equilibrium state;

(d) boron burnable poison rods were not removed from FAs after first year. The influence of above listed items was taken into account just in reactor calculations.

TABLE 3.7.3. FA CRITICALITY CHANGE WHEN BURNING

reactor option with partial inventory of Pu-Th fuel				reactor option with full inventory of Pu-Th fuel		
		UO_2 FA	PuO_2-ThO_2 FA	PuO_2-ThO_2 FA		
Average enrichment, %		4.23	4.8	Average enrichment, %		5.5
Eff. days	Burnup (MWdays/t)	k_{inf}	k_{inf}	Eff. days	Burnup (MWdays/t)	k_{inf}
0.0	0.000	1.181112	1.167882	0.0	0.000	1.186928
2.0	91.680	1.144760	1.142707	2.0	89.692	1.163192
29.5	1353.655	1.142027	1.120663	29.9	1340.895	1.142558
59.1	2707.311	1.141636	1.106274	59.8	2681.791	1.128906
88.6	4060.966	1.139362	1.095775	89.7	4022.686	1.118968
118.1	5414.621	1.135733	1.087125	119.6	5363.582	1.110794
147.7	6768.276	1.131117	1.079400	149.5	6704.477	1.103557
177.2	8121.932	1.125799	1.072306	179.4	8045.373	1.096861
206.7	9475.587	1.119835	1.065626	209.3	9386.268	1.090581
236.2	10829.242	1.113286	1.059217	239.2	10727.163	1.084526
265.8	12182.897	1.106215	1.053013	269.1	12068.059	1.078586
295.3	13536.553	1.098690	1.046907	299.0	13408.954	1.072781
324.8	14890.208	1.090783	1.040849	328.9	14749.850	1.067052
354.4	16243.863	1.082568	1.034783	358.8	16090.745	1.061366
383.9	17597.518	1.074124	1.028679	388.7	17431.641	1.055695
413.4	18951.172	1.065496	1.022501	418.6	18772.535	1.050003
443.0	20304.826	1.056753	1.016228	448.5	20113.430	1.044274
472.5	21658.480	1.047941	1.009851	478.4	21454.324	1.038488
502.0	23012.135	1.039105	1.003397	508.3	22795.219	1.032660
531.5	24365.789	1.030282	0.996814	538.2	24136.113	1.026789
561.1	25719.443	1.021493	0.990132	568.1	25477.008	1.020827
590.6	27073.098	1.012758	0.983352	598.0	26817.902	1.014749
620.1	28426.752	1.004093	0.976478	627.9	28158.797	1.008558
649.7	29780.406	0.995510	0.969518	657.8	29499.691	1.002262
679.2	31134.061	0.987019	0.962485	687.7	30840.586	0.995883
708.7	32487.715	0.978627	0.955387	717.6	32181.480	0.989432
738.3	33841.371	0.970340	0.948229	747.5	33522.375	0.982912
767.8	35195.027	0.962162	0.941038	777.4	34863.270	0.976330
797.3	36548.684	0.954094	0.933832	807.3	36204.164	0.969688
826.8	37902.340	0.946136	0.926641	837.2	37545.059	0.962985
856.4	39255.996	0.938291	0.919495	867.1	38885.953	0.956238
885.9	40609.652	0.930566	0.912422	897.0	40226.848	0.949451
915.4	41963.309	0.922965	0.905464	926.9	41567.742	0.942651
945.0	43316.965	0.915492	0.898629	956.8	42908.637	0.935859
974.5	44670.621	0.908150	0.891939	986.7	44249.531	0.929091
1004.0	46024.277	0.900942	0.885415	1016.6	45590.426	0.922362
1033.6	47377.934	0.893871	0.879083	1046.5	46931.320	0.915686
1063.1	48731.590	0.886938	0.872963	1076.4	48272.215	0.909086
1092.6	50085.246	0.880149	0.867071	1106.3	49613.109	0.902586
1122.1	51438.902	0.873503	0.861421	1136.2	50954.004	0.896218
1151.7	52792.559	0.867004	0.856024	1166.1	52294.898	0.889985
1181.2	54146.215	0.860652	0.850883	1196.0	53635.793	0.883895

TABLE 3.7.4. CRITICAL BORON CONCENTRATION VERSUS CORE CYCLE BURNING AT FULL POWER *(all the results presented below were obtained in equilibrium cycle)*

reactor option with partial inventory of Pu-Th fuel		reactor option with full inventory of Pu-Th fuel	
Days	C_B, ppm	Days	C_B, ppm
0	1319.6 *)	0	2200.8 *)
50	1033.2	50	1536.9
100	807.7	100	1157.2
150	590.2	150	825.3
200	380.3	200	524.4
250	177.4	250	250.9
295.3	0.0	299.2	0.0

*) ^{233}Pa decay during a 30-days reloading interval was taken into account.

Figures 3.7.7 and 3.7.8 present assembly power peaking factors for BOC and EOC for the reactor options under consideration.

a) begin of cycle *)

.675

1.079 .863

1.174 1.275 .913

.942 .920 1.135 .915

.894 1.155 .926 1.288 .866

1.148 .915 .928 1.234 1.086 .676

.682 .904 .942 1.047 1.155 1.018 .974

b) end of cycle

.758

1.045 .851

1.077 1.292 1.031

.895 .939 1.135 1.036

.882 1.100 .958 1.317 .860

1.134 .903 .929 1.158 1.063 .763

.815 .984 .969 .998 1.044 .978 .926

*FIG. 3.7.7. Assembly power peaking factors at BOC and EOC in reactor with partial inventory of ThO_2-PuO_2 fuel. *) ^{233}Pa decay during a 30-days reloading interval was taken into account.*

a) begin of cycle *)

.719

1.121 .941

1.203 1.334 1.009

.917 .977 1.241 1.009

.819 1.040 .977 1.335 .941

.928 .829 .918 1.203 1.121 .719

.585 .754 .824 .963 1.145 1.056 1.021

b) end of cycle

.654

1.037 .819

1.159 1.166 .860

1.007 .996 1.116 .860

.992 1.136 .996 1.166 .819

1.150 1.000 1.008 1.159 1.037 .654

.863 1.013 1.033 1.084 1.163 1.029 .929

*FIG. 3.7.8. Assembly power peaking factors at BOC and EOC in reactor with full inventory of ThO_2-PuO_2 fuel. *) ^{233}Pa decay during a 30-days reloading interval was taken into account.*

Tables 3.7.5–3.7.7 present fuel cycle characteristics.

TABLE 3.7.5. FUEL LOAD CHARACTERISTICS

Characteristics	Partial inventory (1/3)		Full inventory
Assembly type	UO_2	PuO_2-ThO_2	PuO_2-ThO_2
Fuel weight, t h. m.	43.8	22.1	66.9
Total	65.9		66.9
Average initial content in fuel, %			
^{235}U	4.23	-	-
Pu	-	4.8	5.5
Cycle duration, eff. Days	295.3		299
Annual load of			
Heavy metals, t	14.45	7.38	22.16
^{235}U, kg	611.5	-	-
Pu + ^{241}Am, kg	-	354	1220

TABLE 3.7.6. ESTIMATES OF RESOURCES REQUIREMENTS PER 1 GWe/YEAR

Fuel	Partial inventory (1/3)	Full inventory
Natural uranium, t	114.007	-
Thorium, t	7.02	20.94
Plutonium, t	0.354	1.22
SWU's	148 000	-
Fuel fabrication, t h.m.	21.83	22.16

TABLE 3.7.7. ESTIMATES OF FUEL FLOW AT END OF CYCLE

Characteristics	Partial inventory (1/3)	Full inventory
Annually discharged fuel:		
Weight, t h.m.	21.83	22.16
Volume, m^3 of dioxides	2.78	2.78
Weight of Pu in discharged fuel, kg	266	462
Fissile isotopes in discharged Pu: ^{239}Pu, ^{241}Pu, total, %	44.6, 19.7, 64.3	36.3, 22.4, 58.7
Annual balance of Pu: unloading-loading, kg	-88	-758
^{233}U+^{233}Pa annual unloading, kg	100.1	293.7
^{232}U content in unloaded U, ppm	3862	3675
Minor actinides (^{231}Pa, Np, Am, Cm) annual unloading, kg	15.6	17.1
Average content in unloaded FAs, kg/t h.m.		
^{233}U+^{233}Pa	4.6	13.3
^{235}U	7.3	0.2
Pu	12.2	20.8
MA	0.71	0.77

TABLE 3.7.8. CHARACTERISTICS IMPORTANT FOR REACTOR SAFETY (BOC/EOC)

reactor state	MPUM state		Rated power	
characteristics	Partial inventory	Full inventory	Partial inventory	Full inventory
$\Delta\rho_{BA}$, %	14.0/6.6	13.4/6.3	8.1/0	7.8/0
$\frac{\partial\rho}{\partial t_F}$, $\frac{10^{-5}}{^{\circ}C}$	-3.8/-3.9	-4.5/-4.5	-2.6/-2.7	-3.0/-3.1
$\frac{\partial\rho}{\partial t_{H_2O}}$, $\frac{10^{-5}}{^{\circ}C}$	-4.0/-25.3	-7.2/-23.5	-21.6/-52.4	-26.1/-50.9
β_{eff}, 10^{-2}	-	-	0.52/0.49	0.28/0.32
Total control rod reactivity value, %	-	-	6.63/6.85	5.48/6.12
^{233}Pa decay effect (total), % T = 20°C, P = 0	0.40/0.80	1.06/2.00	0.38/0.71	0.92/1.64

Table 3.7.8. shows some characteristics that are important for reactor safety. The calculations were performed at minimum power under monitoring (MPUM) (T = 279 °C, P = 0) and rated power. For the beginning of cycle the values are calculated considering decay of ^{233}Pa and accumulation of ^{233}U over a 30-days reloading interval.

3.7.2. Calculations of the principal neutronics characteristics of the WWER-1000 reactor loaded with PuO_2–ThO_2 fuel based on reactor grade plutonium

The principle design parameters of reactor under consideration are the same as for burning weapons grade plutonium, except:

(1) In partial PuO_2-ThO_2 core average plutonium content in PuO_2-ThO_2 fuel assemblies (FA) is 7.0%.

(2) In full PuO_2-ThO_2 core plutonium content is 8.1%.

(3) Reactor grade plutonium composition at the moment of core loading is taken as follows (in weight%) [2]:

^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am
0.9	61.0	22.0	10.09	4.1	1.1

TABLE 3.7.9. FA k_{inf} CHANGE VERSUS BURNUP

	reactor option with partial inventory of Pu-Th fuel (1/3)		reactor option with full inventory of Pu-Th fuel	
Average enrichment, %	7.0		8.1	
Eff. days	Burnup (MWdays/t)	k_{inf}	Burnup (MWdays/t)	k_{inf}
0.0	0.000	1.155527	0.000	1.166996
1.0	44.764	1.134449	44.705	1.147836
2.0	89.529	1.133264	89.410	1.146799
29.3	1311.600	1.117534	1 309.859	1.132341
58.6	2623.200	1.107673	2 619.719	1.122961
87.9	3934.800	1.100699	3 929.578	1.116332
117.2	5246.399	1.094911	5 239.438	1.110904
146.5	6557.999	1.089597	6 549.297	1.106015
175.8	7869.599	1.084547	7 859.156	1.101370
205.1	9181.198	1.079669	9 169.016	1.096886
234.4	10492.798	1.074896	10 478.875	1.092499
263.7	11804.397	1.070201	11 788.734	1.088156
293.0	13115.997	1.065566	13 098.594	1.083815
322.3	14427.597	1.060981	14 408.453	1.079531
351.6	15739.196	1.056438	15 718.312	1.075294
380.9	17050.797	1.051929	17 028.172	1.071094
410.2	18362.396	1.047451	18 338.031	1.066928
439.5	19673.996	1.043000	19 647.891	1.062795
468.8	20985.596	1.038566	20 957.750	1.058692
498.1	22297.195	1.034144	22 267.609	1.054618
527.4	23608.795	1.029746	23 577.469	1.050568
556.7	24920.395	1.025380	24 887.328	1.046540
586.0	26231.994	1.021035	26 197.188	1.042568
615.3	27543.594	1.016692	27 507.047	1.038630
644.6	28855.193	1.012373	28 816.906	1.034700
673.9	30166.793	1.008078	30 126.766	1.030786
703.2	31478.393	1.003806	31 436.625	1.026883
732.5	32789.992	0.999561	32 746.484	1.022962
761.8	34101.594	0.995341	34 056.344	1.019060
791.1	35413.195	0.991152	35 366.203	1.015159
820.4	36724.797	0.986986	36 676.062	1.011281
849.7	38036.398	0.982846	37 985.922	1.007424
879.0	39348.000	0.978735	39 295.781	1.003585
908.3	40659.602	0.974663	40 605.641	0.999769

TABLE 3.7.9. (cont.)

937.6	41971.203	0.970639	41 915.500	0.995979
966.9	43282.805	0.966668	43 225.359	0.992219
996.2	44594.406	0.962756	44 535.219	0.988488
1025.5	45906.008	0.958907	45 845.078	0.984786
1054.8	47217.609	0.955129	47 154.938	0.981117
1084.1	48529.211	0.951428	48 464.797	0.977473
1113.4	49840.812	0.947811	49 774.656	0.973869
1142.7	51152.414	0.944284	51 084.516	0.970309
1172.0	52464.016	0.940854	52 394.375	0.966796
1201.3	53775.617	0.937525	53 704.234	0.963336
1230.6	55087.219	0.934302	55 014.094	0.959932
1259.9	56398.820	0.931190	56 323.953	0.956588
1289.2	57710.422	0.928192	57 633.812	0.953309
1318.5	59022.023	0.925315	58 943.672	0.950099
1347.8	60333.625	0.922573	60 253.531	0.946961
1377.1	61645.227	0.919960	61 563.391	0.943900
1406.4	62956.828	0.917467	62 873.250	0.940919
1435.7	64268.430	0.915110	64 183.109	0.938023
1465.0	65580.031	0.912888	65 492.969	0.935212

TABLE 3.7.10. CRITICAL BORON CONCENTRATION VERSUS CORE CYCLE BURNING AT FULL POWER

reactor optionwith partial inventory of Pu-Th fuel		reactor optionwith full inventory of Pu-Th fuel	
Days	C_B, ppm	Days	C_B, ppm
0	1223.7	0	1678.3
50	976.1	50	1195.5
100	770.7	100	916.2
150	570.8	150	659.7
200	374.8	200	427.1
250	184.0	250	205.6
299.5	0.0	300	0.0

Tables 3.7.11–3.7.13 present fuel cycle characteristics.

TABLE 3.7.11. FUEL FLOW CHARACTERISTICS

Characteristics	Partial inventory (1/3)		Full inventory
Assembly type	UO_2	PuO_2-ThO_2	PuO_2-ThO_2
Fuel weight, t h. m.	43.8	22.2	
Total	66.0		67.1
Average initial content in fuel, %			
^{235}U	4.23	-	-
Pu	-	7.0	8.1
Cycle duration, eff. days	299.5		300
Annual load of			
Heavy metals, t	14.45	7.4	22.2
^{235}U, kg	611.5	-	-
Pu + ^{241}Am, kg	-	518.7	1803
Average burnup, Mwdays/kg HM	41.4	40.2	
	41.0		

TABLE 3.7.12. ESTIMATES OF RESOURCES REQUIREMENTS PER 1 GW(el)/YEAR

Fuel	Partial inventory (1/3)	Full inventory
Natural uranium, t	114 007	-
Thorium, t	6.9	20.3
Plutonium, t	0.519	1.803
SWU's	148 000	-
Fuel fabrication, t h.m.	21.85	22.2

TABLE 3.7.13. ESTIMATES OF FUEL FLOW AT END OF CYCLE

Characteristics	Partial inventory (1/3)	Full inventory
Annually discharged fuel:		
Weight, t h.m.	21.85	22.2
Volume, m^3 of dioxides	2.78	2.78
Weight of Pu in discharged fuel, kg	401	953
Fraction of ^{239} Pu in discharged Pu, %	36.6	29.5
Annual balance of Pu: unloading-loading, kg	-117	-850
^{233}U+^{233}Pa annual unloading, kg	99.6	291
^{232}U content in discharged U, ppm	3550	3322
Minor actinides (^{231}Pa, Np, Am, Cm): annual unloading, kg	32.4	67.0
Average content in unloaded FAs, kg/t h.m.		
^{233}U+^{233}Pa	4.5	13.1
^{235}U	7.2	0.2
Pu	18.3	42.9
MA	1.5	3.0

Radioactivity of annually discharged fuel (without any reprocessing) is shown in Table 3.7.6 (per 1 GW(el)/year) for EOC, 10 years after discharge, 100, 1000, 10000, 100000, and 1 million years.

TABLE 3.7.14. RADIOACTIVITY AND RADIOTOXICITY OF SPENT FUEL

Cooling time (years)	Partial (1/3) inventory			Full inventory		
	Bq	Sv for water	Sv for air	Bq	Sv for water	Sv for air
0	0.21E+20	0.24E+11	0.22E+13	0.17E+20	0.33E+11	0.61E+13
10	0.22E+18	0.46E+10	0.98E+12	0.57E+18	0.12E+11	0.26E+13
100	0.18E+17	0.31E+10	0.61E+12	0.48E+17	0.81E+10	0.16E+13
1 000	0.38E+16	0.87E+09	0.16E+12	0.99E+16	0.22E+10	0.42E+12
10 000	0.91E+15	0.22E+09	0.37E+11	0.23E+16	0.52E+09	0.89E+11
100 000	0.27E+15	0.27E+08	0.45E+10	0.74E+15	0.70E+08	0.12E+11
1 000 000	0.27E+14	0.25E+07	0.37E+09	0.62E+14	0.57E+07	0.87E+09

Table 3.7.15 shows some reactor safety characteristics. The calculations were performed at minimum power under monitoring (MPUM) (T = 279 °C, P = 0) and rated power. For the beginning of cycle the decay of ^{233}Pa and accumulation of ^{233}U in course of 30-days reloading interval were taken into account.

TABLE 3.7.15. SOME REACTOR SAFETY CHARACTERISTICS (BOC/EOC)

reactor state characteristic	MPUM		Rated power	
	Partial inventory	Full inventory	Partial inventory	Full inventory
$\Delta\rho_{BA}$,%	13.3/6.5	10.5/6.0	7.4/0	4.8/0
$\frac{\partial\rho}{\partial T_F}, \frac{10^{-5}}{C}$	-4.7/-4.7	-5.5/-5.5	-2.6/-2.7	-2.9/-2.9
$\frac{\partial\rho}{\partial T_{H_2O}}, \frac{10^{-5}}{C}$	-4.6/-25.0	-7.4/-21.5	-26.4/-57.3	-36.5/-56.2
β_{eff},10^{-2}	-	-	0.54/0.49	0.33/0.35
Total control rod reactivity worth, %	-	-	6.4/6.6	5.2/5.5

Table 3.7.16 shows the protactinium effect at rated power. The figures correspond to total effect and to 30-days loading interval effect as well.

TABLE 3.7.16. ^{233}Pa EFFECT VALUES, %

reactor state	Partial inventory		Full inventory	
	30-days effect	Total effect	30-days effect	Total effect
BOC	0.16	0.29	0.39	0.73
EOC	0.26	0.49	0.64	1.19

3.7.3. Assessment of the effect of plutonium burning on the waste toxicity

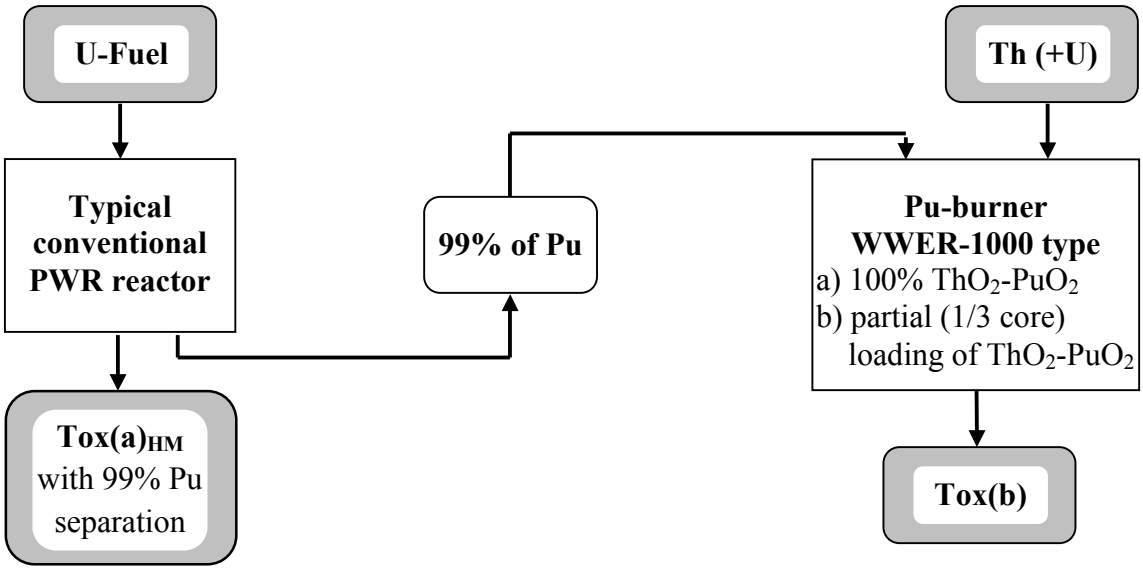
In the benchmark should be compared heavy metal radiotoxicity per 1 GWa(el) (where 1 year = 300 FPD) of spent fuel from:

1) typical conventional PWR reactor loaded with UO_2 fuel (case 1), and
2) in alternative plutonium burning in fuel cycle (case 2).

The calculation of radiotoxicity of a typical PWR (1000 MW(el), 300 FPD) case 1 was the subject of investigation at stage 3 of IAEA Benchmark, and the results were presented in Section 2.

The radiotoxicity of spent fuel in the fuel cycle shown in Fig. 3.7.9 – case 2 has to be determined as a sum of:

— heavy metal radiotoxicity of a typical conventional uranium PWR reactor after extracting 99% of plutonium isotopes from its spent fuel (Tox(a)); and



— radiotoxicity of spent fuel of reactor (Tox(b)) in which plutonium recycle is made (reactor-burner).

FIG. 3.7.9. Fuel cycle scheme, case 2.

Radiotoxicity has to be normalized per 1 GW_{el}a produced in the cycle. As this take place, a relationship between capacities of a typical uranium PWR and reactor-burners was defined based on plutonium mass balance, and total radiotoxicity of fuel cycle per 1 GW_{el}a (case 2) can be determined as:

$$R = \frac{R_1 + xR_2}{1 + x},$$

$$x = \frac{G_{Pu}^{D(1)}}{G_{Pu}^{C(2)}},$$

where

R_1 - radiotoxicity of discharged fuel of a typical PWR without of 99% of plutonium-Tox(a);

R_2 - radiotoxicity of discharged recycled fuel from reactor-burner - Tox(b);

$G_{Pu}^{D(1)}$ - annual plutonium discharge from a typical PWR (245 kg Pu/1 GW_{el}a as follows from Benchmark task for stage 3 (part 1));

$G_{Pu}^{C(2)}$ - annual plutonium charge for reactor-burners.

$G_{Pu}^{C(2)}$ - corresponds to individually chosen reactor-burners - for Russian Federation they are WWER-1000 type reactors with full or partial inventories of PuO_2-ThO_2 fuel: $G_{Pu}^{C(2)}$ - = 1803 kg for full inventory and $G_{Pu}^{C(2)}$ = 519 kg of plutonium for partial inventory.

Radiotoxicity computations were made on the basis of Dose Coefficients of Intake recommended by ICRP (ICRP publications, 1991, 1994). Only heavy atoms were taken into account.

The results are shown in Table 3.7.17 and Figs 3.7.10 and 3.7.11. From this figures it is obvious that reactor grade plutonium recycling does not tend to essential change in radiotoxicity: at initial stage of storage (tens of years) radiotoxicities are closely allied, then radiotoxicity when recycling appears to be slightly below radiotoxicity of open fuel cycle, between 10^5 and 10^6 years recycling causes increase in radiotoxicity in case of using plutonium-thorium burners, and at the end of given period they prove to be close again. By and large it can be noted that the use of thorium for decreasing in radiotoxicity does not give clear merits.

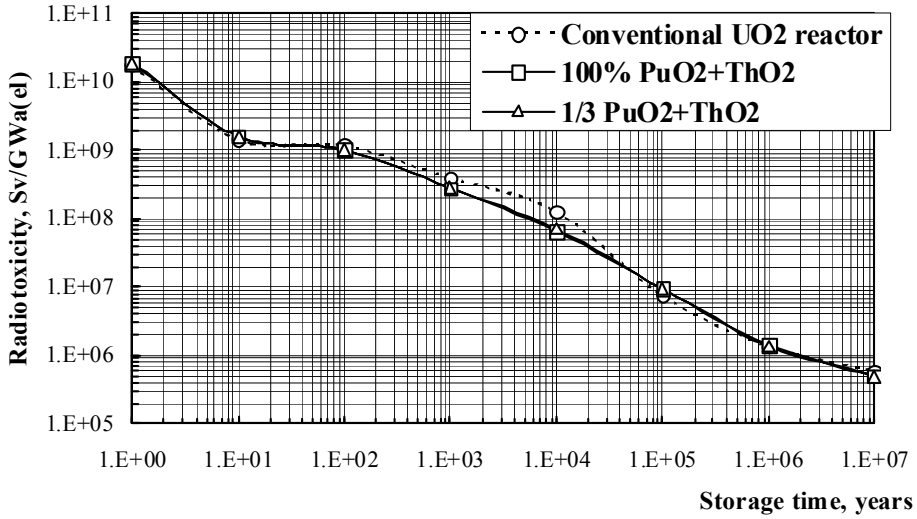


FIG. 3.7.10. Ingestion hazard of heavy metals. Conventional UO_2 reactor and one-through cycle of plutonium discharged from a typical PWR in PuO_2-ThO_2 reactors.

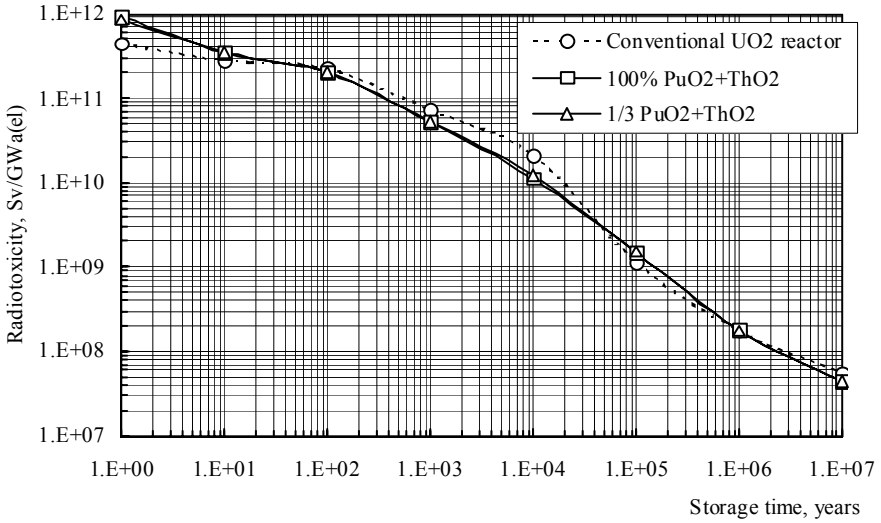


FIG. 3.7.11. Inhalation haggard of heavy metals. Conventional UO_2 reactor and one-through cycle of plutonium discharged from a typical PWR in PuO_2-ThO_2 reactors.

TABLE 3.7.17. RADIOTOXICITY OF HEAVY METAL FROM SPENT FUEL, $Sv/GW_{el}a$

Storage time (years)	Conventional UO_2 loaded reactor No Pu separation	Recycling 100% PuO_2-ThO_2 loaded burner	Pu in 1/3 PuO_2-ThO_2 loaded burner
Ingestion			
0	1.79E+10	1.87E+10	1.90E+10
10^1	1.43E+09	1.63E+09	1.60E+09
10^2	1.16E+09	1.02E+09	1.03E+09
10^3	3.99E+08	2.79E+08	2.88E+08
10^4	1.28E+08	6.56E+07	7.14E+07
10^5	7.63E+06	9.44E+06	9.51E+06
10^6	1.36E+06	1.42E+06	1.38E+06
10^7	5.88E+05	5.08E+05	4.89E+05
Inhalation			
0	4.41E+11	9.25E+11	8.59E+11
10^1	2.79E+11	3.52E+11	3.44E+11
10^2	2.24E+11	2.01E+11	2.03E+11
10^3	7.22E+10	5.18E+10	5.35E+10
10^4	2.13E+10	1.12E+10	1.22E+10
10^5	1.14E+09	1.50E+09	1.52E+09
10^6	1.74E+08	1.78E+08	1.76E+08
10^7	5.58E+07	4.47E+07	4.42E+07

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3.8. NETHERLANDS

3.8.1. Introduction

In the framework of the IAEA Co-ordinated Research Programme (CRP) on “Potential of the Thorium based Fuel Cycles to Constrain Plutonium to Reduce Long term Waste Toxicities”, a benchmark calculation on a simple pin cell geometry has been performed.

The purpose of the CRP is to assess advantages and disadvantages of thorium based fuel cycles with the aim to elaborate a selection of options and guidance for the IAEA's member states on the perspective fuel cycle for the next century. The first stage of the CRP is a benchmark calculation on a PWR pin cell geometry in order to intercompare the differences in cross section libraries and calculation methods of the participants. Of interest are the isotopic composition, cross-sections, fluxes and infinite multiplication factors. The specifications of the benchmark have been given in Section 2.1.1.

This document describes the NRG results obtained by the burnup code system OCTOPUS. This code system consists of different modules which take care of the spectrum and burnup calculations. The spectrum calculation was done by the SCALE-4.1 code system. For the burnup calculation ORIGEN-S was used.

Section 3.8.2 describes in detail the calculation with the code system OCTOPUS. In Section 3.8.3 the results of the calculations are presented in the form of pictures, tables and comments. Section 3.8.4. contains the requested numerical output.

3.8.2. Calculation method

The calculations have been done with the burnup and criticality code system OCTOPUS developed at NRG [1]. The code system interconnects all reactor codes available at NRG. OCTOPUS works with modules that are scripts that call interfacing codes and the underlying reactor codes. In the particular case of this benchmark OCTOPUS links the SCALE-4.1 code system with the burnup code ORIGEN-S [2]. SCALE-4.1 is a neutron transport code for pin-cells and assemblies. ORIGEN-S is a point-depletion code.

OCTOPUS uses cross-sections form the ECNAF data library, which contains activation and transmutation cross-sections for over 750 nuclides mainly based on the JEF2.2 data file [3–4]. SCALE-4.1 uses the EIJ2-XMAS library. This is a 172 group data library for reactor calculations with the XMAS group structure based on JEF2.2. In order to transfer cross-sections from one code to the other a generic format is used which can be converted to and from the formats used by the other codes. The format that is chosen is the AMPX weighted format (AMPX-W). All information needed for the coupled spectrum-burnup calculation is passed from one module to the other via a Binary Interface File (BIF).

One other important feature of the OCTOPUS code system is the (optional) use of the PSEUDO-module. This module can be used with the SCALE-spectrum module. It calculates for each burnable zone a fine-group cross-section set, which accounts for the neutron absorption and production of all actinides and fission products not explicitly included in the SPECTRUM-module. These cross-sections sets are added to the cross section library of the spectrum code used. For this benchmark a pseudo nuclide was included in the fuel region of the pin-cell. The connection between the modules and the BIF's is presented in Fig. 3.8.1.

3.8.3. Results of the benchmark calculation

The numerical output data as requested in the benchmark specifications are included in this document in Section 3.8.4. In this section a graphical presentation of the results is given. The figures will be commented qualitatively.

3.8.3.1. The infinite multiplication factor: K∞

The infinite multiplication factor is shown in Fig. 3.8.2. Each point in the curve of Fig. 3.8.2 corresponds with one spectrum calculation and subsequent burnup calculation. The time steps at the beginning are small because the xenon buildup, which influences the neutron flux spectrum, takes place on a relatively short time scale. The entire burnup sequence consists of 24 time steps.

3.8.3.2. Isotopic composition of actinides

In Fig. 3.8.3 the depletion of the plutonium isotopes is shown. After the exit burnup of 60 MWd/kgHM nearly all the Pu-239 has been burnt. The mass of Pu-241 first increases and after a while decreases. At the one hand the production of Pu-241 is due to capture in Pu-240 and at the other hand Pu-241 is depleted due to fissioning.

In Fig. 3.8.4 the densities of the protactinium and the uranium isotopes are shown. The buildup of the fissile nuclide U-233 is due to neutron capture in Th-232 and subsequent decay of Th-233 via Pa-233. Since the growth of U-233 becomes less during burnup, while the total flux is increasing as will be seen in the next section, one can conclude that fissioning of U-233 contributes more and more to the power.

The concentration of Pa-232 has reached its equilibrium value almost after 100 days. The level of the Pa-233 is increasing slightly during burnup due to increasing total neutron flux. U-234 is produced both by capture in U-233 and capture in Pa-233 and subsequent decay of Pa-234. A little U-235 is produced due to capture in U-234. The nuclide U-232 plays an important role in reprocessing since one of its decay products emits hard gammas. Uranium-232 is formed by (n,2n) reactions on U-233.

In Fig. 3.8.5 the concentrations of the minor actinides are plotted. Am-243 and Cm-244 are the most abundant minor actinides present in the fuel. Cm-244 is mainly produced by neutron capture in Am-243 and subsequent decay of Am-244. Americij-243 is produced by neutron capture of Pu-242 and subsequent decay of Pu-243. Cm-244 decays to Pu-240 with half-life of 18 years, while Am-243 decays to Pu-239 with the much longer half-life of 7370 years.

3.8.3.3. Total neutron flux

The total neutron flux is increasing during the burnup, because the macroscopic fission cross section decreases mainly due to depletion of the fissile nuclides. This is a direct consequence of the constant linear power assumed. In Fig. 3.8.2 is shown that the total neutron flux increases from 2.9×10^{14} to about 3.9×10^{14} $cm^{-2}s^{-1}$ during burnup.

3.8.3.4. Microscopic cross-sections

The absorption, fission and (n,2n) cross-sections at burnups of and 60 MWd/kgHM are presented in Section 3.8.4. The absorption cross section is the summation of the fission cross section and the neutron disappearance cross-section. The latter is defined as the sum of all cross-sections in which a neutron is not in the exit channel. In terms of ENDF/B MT-numbers this means MT = 18 plus MT = 102 through 114. Due to their high cross-sections and low

lying resonance peaks the nuclides Pu-239 and Pu-240 strongly determine the shape of the flux especially in the thermal region. This is shown in Fig. 3.8.6. If the concentrations of these nuclides change, the averaged microscopic cross-sections of all other nuclides change due to the different flux shape. One example of the influence of the changing flux spectrum is the change of the averaged microscopic cross section of Th-232. Since the concentration of Th-232 is virtually constant during burnup, which means that the self-shielding is constant too, a change of the microscopic cross section must be caused by the changing neutron flux spectrum. The averaged microscopic absorption cross-section of thorium increases from 0.85 barn at zero burnup to 1.13 barn at a burnup of 60 Mwd/kgHM. The effect of the decrease in self-shielding can be illustrated by the absorption cross-sections of Pu-239. At zero burnup $\sigma_a = 68$ barn and at a burnup of 60 Mwd/kgHM $\sigma_a = 178$ barn.

3.8.3.5. Average energy per fission

In Section 3.8.5 the values of the average energy per fission are shown. This includes energy generated due to the neutron captures of the nuclides in the fuel. Because of the change in composition of the fissile nuclides the average energy per fission changes. In the case of the plutonium/thorium fuel the average energy per fission decreases during burnup. The smooth transition from plutonium fissioning to U-233 fissioning causes the decrease in the average energy per fission during burnup. This is due to the fact that the fissile plutonium isotopes release about 200 MeV thermal energy per fission and U-233 only releases about 190 MeV thermal energy per fission.

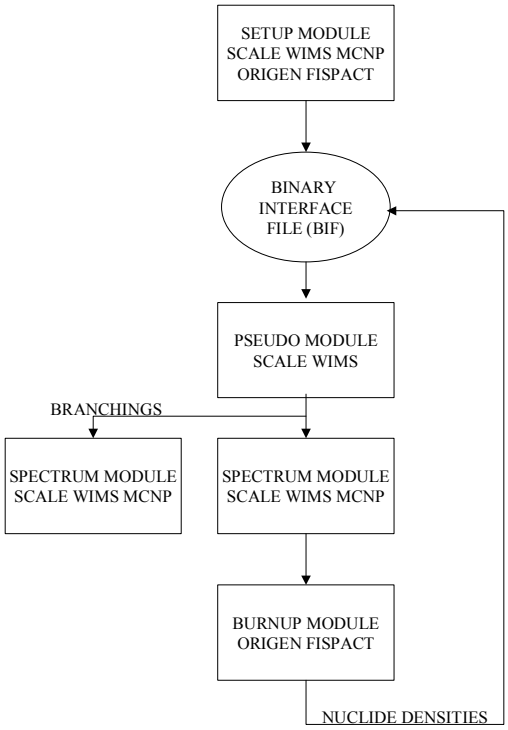

FIG. 3.8.1. The scheme of OCTOPUS burnup and criticality code system.

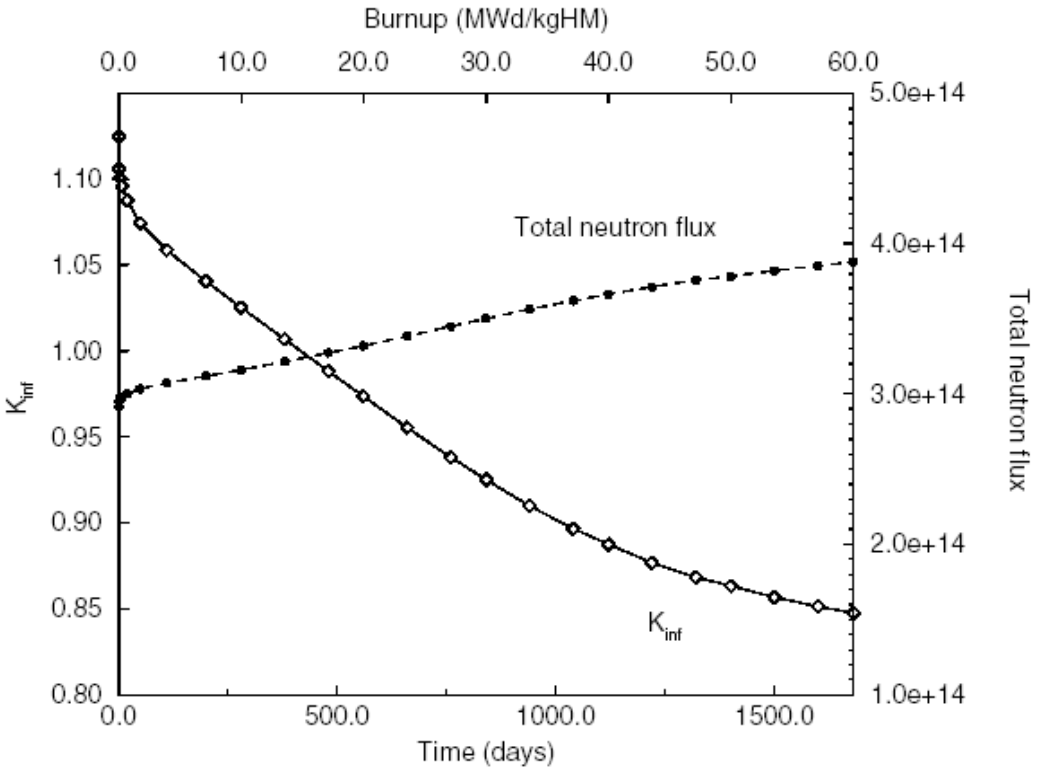

FIG. 3.8.2. k_{inf} and total neutron flux as function of burnup.

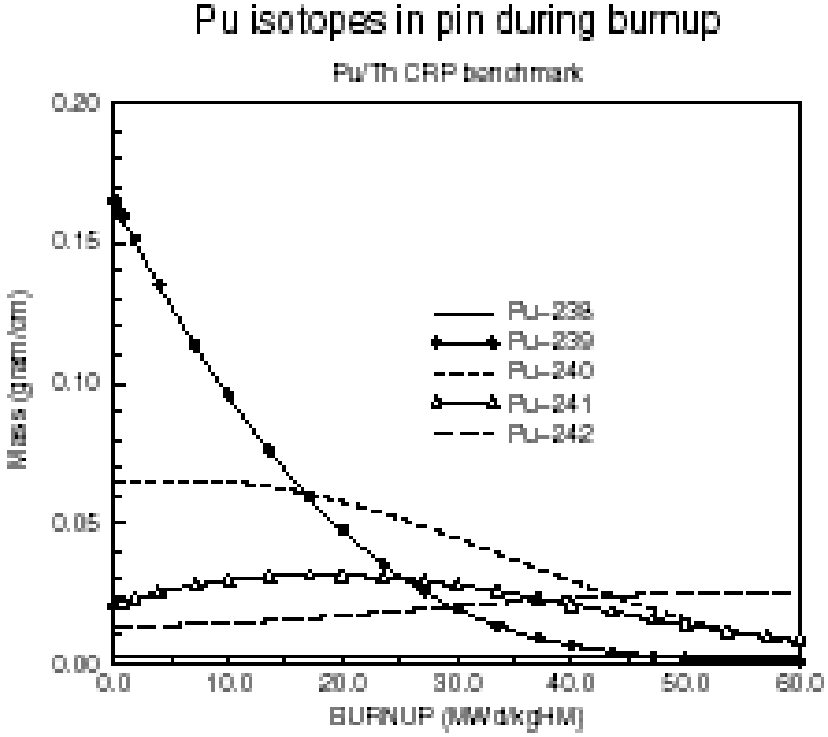

FIG. 3.8.3. Plutonium composition as function of burnup.

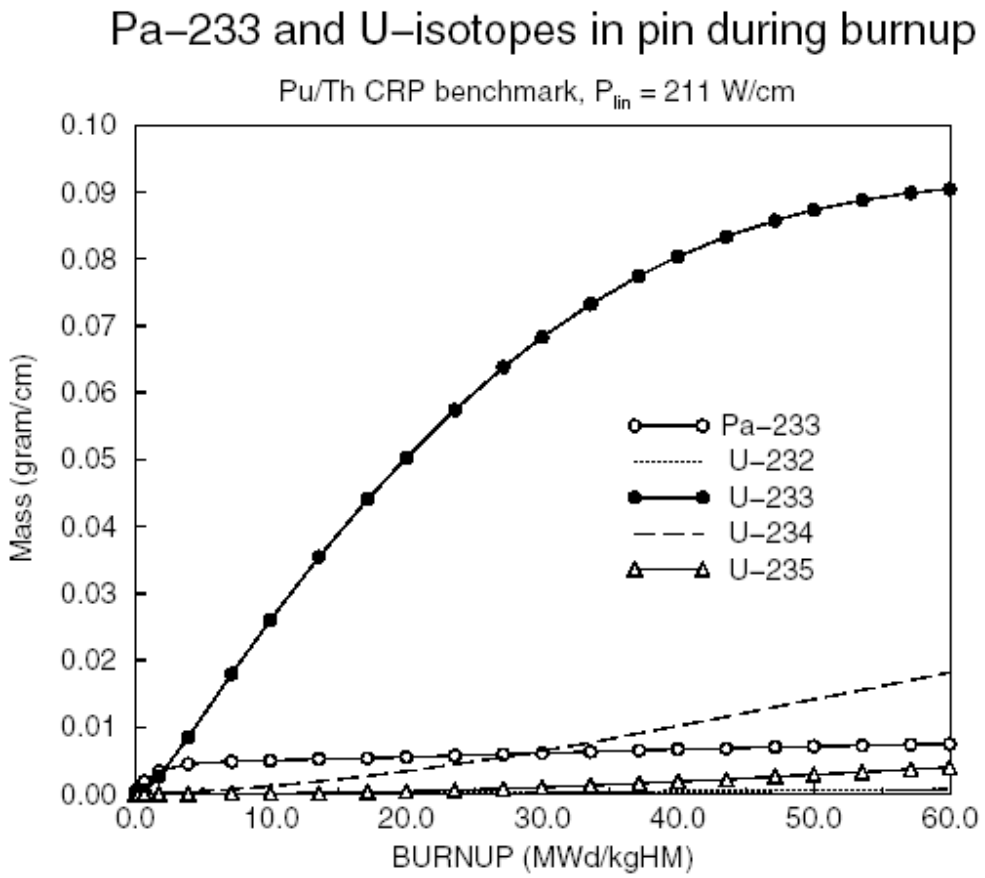

FIG. 3.8.4. Protactinium-233 and uranium as function of burnup.

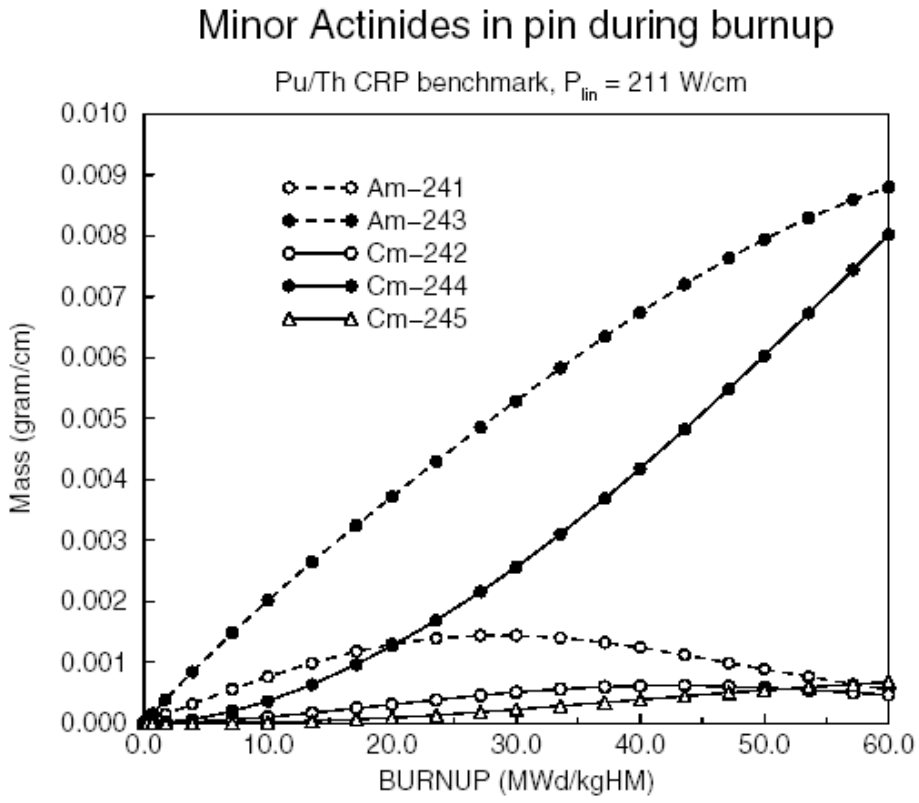

FIG. 3.8.5. The formation of minor actinides as function of burnup.

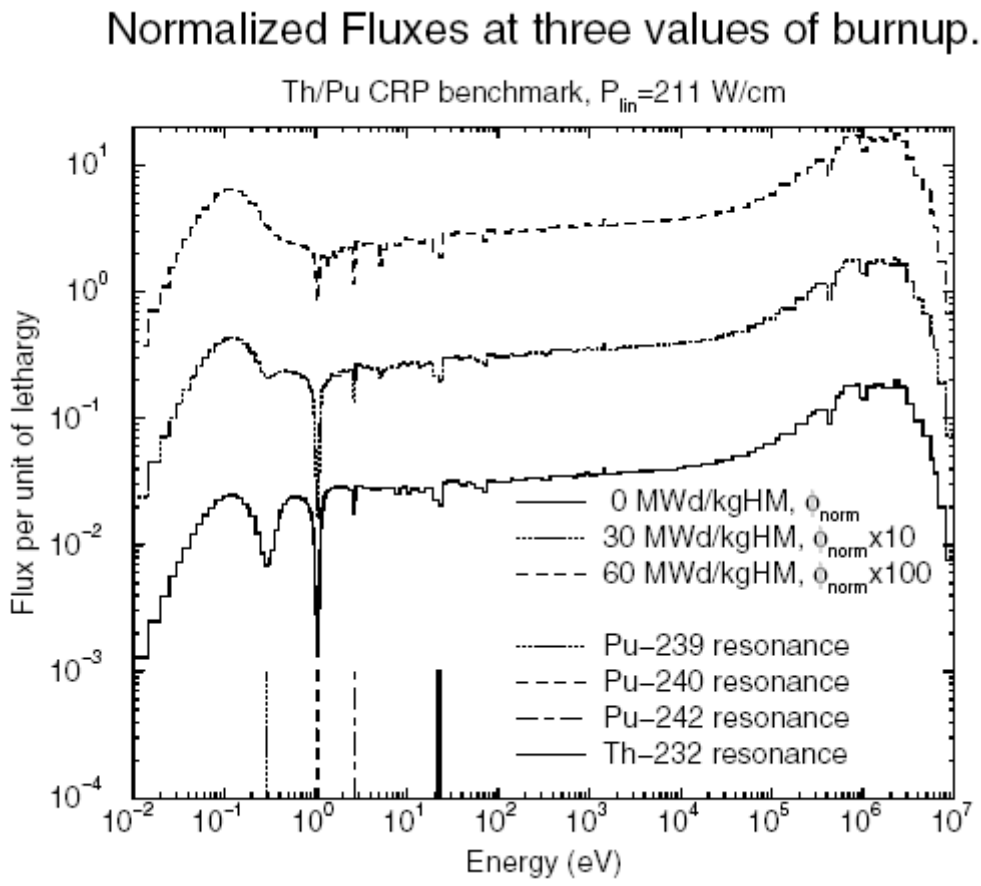

FIG. 3.8.6. Normalized flux at zero, half way and final burnup.

3.8.4. Numerical results of the benchmark

3.8.4.1. The infinite multiplication factor: K_{∞}

	0 MWd/kgHM	30 MWd/kgHM	40 MWd/kgHM	60 MWd/kgHM
k_{inf}	1.12479	0.925198	0.887499	0.847561

3.8.4.2. Isotopic composition of the actinides

nuclide concentrations 1/(barn x cm)

nuclide identifier = 10000xZ + 10xA + M,

where:
Z = the atomic number;
A = the atomic mass of the nuclide; and
M = the metastable state of the nuclide.
M = 0 is the groundstate, and M = 1 is the first metastable state of the nuclide.

TABLE 3.8.1. NUCLIDE CONCENTRATIONS [1/(cm^{2}s)]

	0 MWd/kgHM	30 MWd/kgHM	40 MWd/kgHM	60 MWd/kgHM
902300		1.99738E-08	2.66202E-08	3.93323E-08
902320	2.11000E-02	2.06456E-02	2.04631E-02	2.00579E-02
912310		2.06328E-06	2.31721E-06	2.37571E-06
912330		2.26210E-05	2.47680E-05	2.79479E-05
922320		9.23719E-07	1.49241E-06	2.54699E-06
922330		2.54433E-04	2.99339E-04	3.36896E-04
922340		2.39808E-05	3.75794E-05	6.72831E-05
922350		3.60476E-06	6.78927E-06	1.47382E-05
922360		2.35243E-07	5.90704E-07	2.25522E-06
922370		6.22406E-10	1.57728E-09	6.01748E-09
922380		3.31807E-10	6.91969E-10	3.06229E-09
932370		2.23173E-08	4.87085E-08	1.89977E-07
932380		6.13735E-11	1.45367E-10	6.38593E-10
932390		1.67812E-11	2.17215E-11	3.10352E-11
942360		3.96932E-13	4.98487E-13	7.14752E-13
942370		3.27285E-12	3.23564E-12	2.84032E-12
942380	9.72000E-06	8.17818E-06	8.54297E-06	8.18189E-06
942390	5.99000E-04	7.08363E-05	2.34028E-05	2.74119E-06
942400	2.32000E-04	1.61752E-04	1.06364E-04	2.41181E-05
942410	7.69000E-05	1.00379E-04	7.57607E-05	2.91171E-05
942420	4.78000E-05	7.31808E-05	8.38980E-05	8.91505E-05
952410		5.18667E-06	4.45826E-06	1.98443E-06
952420		1.17795E-08	1.18450E-08	6.45866E-09
952421		7.78570E-08	6.60042E-08	2.81144E-08
952430		1.88889E-05	2.40517E-05	3.14022E-05
962410		8.82669E-13	1.07916E-12	7.89001E-13
962420		1.80664E-06	2.18697E-06	1.65886E-06
962430		4.91356E-08	7.91052E-08	9.47679E-08
962440		9.09773E-06	1.48483E-05	2.85139E-05
962450		7.95648E-07	1.36685E-06	2.37964E-06
962460		7.83224E-08	2.26761E-07	8.70984E-07
962470		1.03602E-09	3.95592E-09	2.19781E-08
962480		5.66398E-11	3.23235E-10	3.44533E-09
972490		6.18007E-13	3.98013E-12	4.73533E-11

3.8.4.3. Total neutron flux

TABLE 3.8.2. TOTAL NEUTRON FLUX [1/(cm^{2}s)]

	0 MWd/kg$_{HM}$	30 MWd/kg$_{HM}$	40 MWd/kg$_{HM}$	60 MWd/kg$_{HM}$
Fuel	2.9131317E+14	3.5005688E+14	3.6624260E+14	3.8775357E+14
Clad	2.9255918E+14	3.5062147E+14	3.6663592E+14	3.8785461E+14
Moder	2.9300243E+14	3.5120984E+14	3.6725799E+14	3.8851187E+14

3.8.4.4. Microscopic cross-sections

microscopic cross-sections (barn) at 0 and 60 MWd/kgHM

nuclide identifier = 10000xZ + 10xA + M,

where:
Z = the atomic number;
A = the atomic mass of the nuclide; and
M = the metastable state of the nuclide.
M = 0 is the ground state and M = 1 is the first metastable state of the nuclide).

TABLE 3.8.4. MICROSCOPIC CROSS-SECTIONS

	0 MWd/kgHM			60 MWd/kgHM		
	SIGMA-ABS	SIGMA-FIS	SIGMA (N,2N)	SIGMA-ABS	NUCLIDE SIGMA-FIS	SIGMA (N,2N)
902300	2.39011E+01	7.02859E-02	6.30347E-0	1.95142E+01	6.27926E-02	5.56113E-03
902320	8.49193E-01	2.64410E-02	6.41957E-03	1.13212E+00	2.34336E-02	5.65989E-03
912310	3.43930E+01	4.26555E-01	4.24246E-03	5.14408E+01	3.85337E-01	3.74945E-03
912330	2.51083E+01	1.74255E-01	1.74513E-03	2.13351E+01	1.54669E-01	1.54123E-03
922320	2.56036E+01	1.49390E+01	3.11247E-03	2.82785E+01	1.65733E+01	2.77855E-03
922330	4.02880E+01	3.51551E+01	3.36468E-03	6.21158E+01	5.54029E+01	2.95478E-03
922340	2.22236E+01	5.65110E-01	6.16813E-04	1.90972E+01	5.27656E-01	5.50979E-04
922350	2.83248E+01	2.21161E+01	4.45086E-03	5.59127E+01	4.57966E+01	3.91303E-03
922360	1.05971E+01	3.56991E-01	3.15559E-03	8.63515E+00	3.08996E-01	2.77735E-03
922370	2.07333E+01	5.87340E-01	8.94844E-03	4.00685E+01	6.25727E-01	7.84089E-03
922380	7.92269E+00	1.13534E-01	4.64700E-03	7.23446E+00	1.00790E-01	4.09127E-03
932370	2.76985E+01	5.63459E-01	9.47485E-04	3.41883E+01	5.08507E-01	8.43972E-04
932380	8.25749E+01	7.48734E+01	5.65892E-03	1.73722E+02	1.57743E+02	4.97627E-03
932390	1.47395E+01	6.63625E-01	1.40592E-03	1.50134E+01	6.00058E-01	1.23564E-03
942360	2.68673E+01	1.39698E+01	1.20173E-03	3.84232E+01	1.96987E+01	1.07031E-03
942370	1.03081E+02	8.42578E+01	8.61504E-04	2.17628E+02	1.75111E+02	7.61526E-04
942380	1.73033E+01	2.04971E+00	3.15369E-04	3.63563E+01	2.52468E+00	2.80248E-04
942390	6.83990E+01	4.40855E+01	1.25423E-03	1.78121E+02	1.13973E+02	1.11049E-03
942400	4.95735E+01	6.59336E-01	1.52586E-03	1.26929E+02	6.16305E-01	1.34854E-03
942410	7.16909E+01	5.42661E+01	8.39213E-03	1.62759E+02	1.21833E+02	7.35902E-03
942420	2.36292E+01	4.97854E-01	2.57304E-03	1.78495E+01	4.46165E-01	2.26686E-03
952410	6.47173E+01	9.61650E-01	7.17547E-04	1.15965E+02	1.25616E+00	6.35437E-04
952420	2.33107E+02	8.95142E+01	1.65527E-03	5.53647E+02	1.83088E+02	1.45273E-03
952421	3.52077E+02	2.87143E+02	4.52328E-03	8.70527E+02	7.06021E+02	3.96576E-03
952430	5.06533E+01	4.74513E-01	1.95508E-03	4.17313E+01	4.23033E-01	1.71900E-03
962410	1.08542E+02	9.91149E+01	1.67822E-04	2.32468E+02	2.12145E+02	1.46885E-04
962420	4.95190E+00	1.01799E+00	4.31725E-04	5.59685E+00	1.18399E+00	3.88580E-04
962430	7.18032E+01	6.13304E+01	1.41869E-03	8.74651E+01	7.36996E+01	1.24944E-03
962440	1.80149E+01	1.04057E+00	1.78854E-03	1.53880E+01	9.63577E-01	1.57743E-03
962450	7.48131E+01	6.46135E+01	1.57724E-03	1.57262E+02	1.36000E+02	1.39709E-03
962460	3.74549E+00	6.77336E-01	1.97097E-03	3.33720E+00	6.04834E-01	1.74164E-03
962470	3.10501E+01	1.90076E+01	1.19287E-02	4.03327E+01	2.45006E+01	1.04651E-02
962480	8.29884E+00	8.32402E-01	2.49872E-03	7.60976E+00	7.52782E-01	2.19761E-03
972490	1.46732E+02	3.46302E-01	7.98240E-03	2.05614E+02	3.09741E-01	6.99766E-03

3.8.4.5. Average energy per fission

AVERAGE ENERGY PER FISSION (MeV per fission)

This value includes energy generated due to neutron captures of the nuclides in the fuel zone of the pin cell.

	0 MWd/kgHM	30 MWd/kgHM	40 MWd/kgHM	60 MWd/kgHM
Energy	207.891	205.775	204.411	202.009

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