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PNC's Results on the Metal-Fueled Fast Reactor Benchmarks

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Abstract

WPPR (Working Party on Physics of Plutonium Recycling) has been organized in Nuclear Science Committee of OECD/NEA since November 1992. More than ten advanced countries (France, United Kingdom, Germany, Russia, United States, Canada, Japan, etc.) participate in this working party. An aim of WPPR is to clarify some physical issues related to the technology for recycle of plutonium. To evaluate different scenarios for the use of plutonium, international benchmarks were developed for various types of reactors (MOX-fueled fast reactor, metal-fueled fast reactor, PWR and advanced converter). Among these, we contributed to the metal-fueled fast reactor benchmarks. In this report, our calculated results are summarized with all the information required. Each result is listed independently in a table according to the sequence indicated in the benchmark proposal, NEA/NSC/DOC(93)24.

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WPPR金属燃料高速炉ベンチマーク計算結果

大木繁夫*、山本敏久**

要　　旨

WPPR（プルトニウムリサイクルの物理に関するワーキングパーティー）は 1992年11月、OECD/ NEA のNuclear Science Committee の中に設けられた。参加国はフランス、イギリス、ドイツ、ロシア、カナダ、アメリカ、カナダ、日本など先進10ヵ国以上に及んでいる。WPPRの目的はプルトニウムリサイクルの技術に関するいくつかの物理的課題を明らかにすることにある。活動の一環として、プルトニウム利用の異なるシナリオを評価するために、様々なタイプの原子炉（MOX燃料高速炉、金属燃料高速炉、PWR、新型転換炉）についての国際ベンチマークがおこなわれている。本報告書は我々の参加した金属燃料高速炉ベンチマークの計算結果をまとめたものである。それぞれの結果はベンチマーク提案書（NEA/NSC/DOC(93)24）に示された順序にしたがって、表形式にまとめてある。

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1. Introduction

In this report, results for the Metal-Fueled Fast Reactor Benchmarks are given with all the information required. Each result is listed independently in a table according to the sequence indicated in the benchmark proposal, NEA/NSC/DOC(93)24. In sections 2 and 3, calculated results for the Startup Core Benchmark and the Once-Through Core Benchmark are described respectively.

2. Result for the Metal-Fueled Burner Startup Core Benchmark

2.1 Basic Data

1) Nuclear data

A JENDL-2[1] based 70-group cross section file JFS-3-J2 was used for all the nuclides except ^{246}Cm, which data were taken from JENDL-3[2]. The following are other conditions:

• Average fuel temperature was set to 850K. For other compositions besides fuel, temperature was fixed to 650K.

• Self-shielding effects were evaluated with a preliminary MOL composition in order to take into account the composition change due to burnup.

2) Broad group energy boundaries (18-group)

———See Table 2-1

3) Broad group cross section preparation

An RZ 70-group diffusion calculation was performed for the geometry described in Fig. 2-1 with a preliminary MOL composition, and average flux was stored for each region to get region-wise 18-group microscopic cross sections.

2.2 BOL Neutron Balance

1) Spatial representation

An RZ model was used. Dimensions and mesh sizes are given in Fig. 2-1. For the burnup compositions, the core region of the driver fuel was divided into 13 subregions totally, that is, radially four rings, axially three layers and an extra layer of the core center ring additionally.

2) Neutron balance solution algorithm

The classical CITATION code[3] was used, thus description of the solution algorithm is omitted.

3) BOL eigenvalue and convergence criterion

———See Table 2-2

As the mesh sizes in burnup calculation ($R \simeq 3cm, Z = 5cm$) were insufficient for this purpose, a mesh correction factor was applied to get the mesh-effect-free eigenvalue. See Appendix A for the procedure.

4) Broad group flux spectrum at core center

———See Table 2-3

The position of the core center region is marked in Fig. 2-1.

5) k_∞ using central flux spectrum and composition

———See Table 2-4

6) Core leakage / Core absorption

———See Table 2-5

7) Model leakage / Model absorption

———See Table 2-6

8) Core capture fractions

———See Table 2-7

9) Energy-averaged cross sections collapsed using central fluxes

———See Table 2-8

2.3 Depletion Methodology

1) Description of the burnup chain representation

Fig. 2-2 shows the burnup chain. Nuclides with dotted squares were not treated in burnup process and considered to decay immediately after production. Spontaneous decay was treated for all the nuclides of which half lives are shorter than 10^6 years. Decay constants from Ref(4) were used and these are listed in Table 2-9.

2) Flux normalization

Since the CITATION code cannot treat heat emission due to capture reactions, fission energy of each heavy nuclide was corrected to take this effect into account. This method, however, cannot apply to non-fissionable nuclides and as a result, heat production due to structural materials were neglected. Fission energy and capture energy were taken from the recommended values by Sher(5) and ENDF/B-Ⅳ, respectively. The fission energy emission in the form of anti-neutrino was excluded. For six nuclides of which fission energies were not available in Ref(5), ORIGEN2(6) recommendation was used. The correction factors applied here were evaluated using a preliminary MOL composition and are shown in Table 2-10.

3) Burnup numerical solution process

The burnup cycle was divided into 5 burnup steps of 62.05EFPD and flux was recalculated and renormalized to reactor power at the beginning of each step. Table 2-11 gives k_{eff} and flux amplitude at each burnup step. Heavy nuclide mass in the thirteen fuel subregions were treated and stored independently through burnup calculation.

4) Fission product representation

At present, JFS-3-J2 has four lumped fission product cross sections from typical fissiles; ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu. Substitution for other nuclides was necessary and they were grouped as described in Table 2-12.

2.4 BOL to EOL Transition and EOL Neutron Balance

1) Mass increments by isotope

———See Table 2-12

2) EOL eigenvalue and convergence criterion

———See Table 2-13

The same correction factor as in BOL was used to obtain the mesh-effect-free eigenvalue.

3) Burnup swing

———See Table 2-14

4) TRU breeding ratio

———See Table 2-15

5) EOL neutron spectrum at core center

———See Table 2-16

3. Result for the Metal-Fueled Once-Through Burner Core Benchmark

In this once-through core benchmark, a preliminary MOEC concentration was used to evaluate self-shielding effects, region-wise fluxes for group collapsing and a correction factor for mesh-effect-free eigenvalues. The preliminary MOEC concentration was obtained by depletion calculation with the effective cross sections using the fuel compositions at the mid of the first cycle. Except for this, the same methodology as the previous benchmark was used for the neutron balance and depletion calculations.

3.1 BOEC Neutron Balance

1) Fuel management representation

The depletion calculation was performed until the core compositions settled down to a state of equilibrium. Consequently we regarded the reactor reached to equilibrium at the fifth cycle. The compositions of assemblies were treated separately according to the core loading batch they belong, although the flux calculation was done with the smeared fuel composition on the subregion basis. The 1/3 core refueling was carried out at the beginning of each cycle. Decrease of the short-lived nuclides during the cooling interval between operating cycles, 54.75days, was also considered.

2) Fresh fuel enrichment

The TRU enrichment of fresh fuel assemblies was determined so that the eigenvalue approached towards unity at EOEC. The result of parametric survey is shown in Table 3-1. In this survey, TRU / HM mass ratio was changed keeping the fuel volume constant. Consequently, the suitable TRU / HM mass ratio was found to be 26.85%.

Note that we didn't include here any E/C bias correction based on critical experimental analysis.

3.2 BOEC to EOEC Transition and Mass Flow

Table 3-2 gives k_{eff} and flux amplitude at each burnup step of the equilibrium cycle.

1) Burnup swing

——— See Table 3-3

2) TRU breeding ratio

——— See Table 3-4

3) Mass increments by heavy metal isotope

——— See Tables 3-5 and 3-6

4) Safety parameters

——— See Tables 3-7 and 3-8

Safety parameters were calculated using the CITATION code[3]. The sodium void worth and the Doppler coefficient were obtained by 70-group exact perturbation. The 70-group effective cross sections were regenerated using BOEC and EOEC concentrations.

The Doppler coefficient was evaluated from the reactivity change due to fuel temperature rise from 850K to 1350K. And the following isotopes were taken into account: ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{243}Am and ^{244}Cm. Other heavy metal isotopes have no f-tables in JFS-3-J2 file at present.

The delayed neutron fraction β_{eff} was calculated using Tuttle's yield data[7] and the spectrum data recommended by Saphier et al.[8]

5) Radioactivity and decay

In order to evaluate radioactivity and toxicity, the mass variation of discharged heavy metal isotopes was calculated by the ORIGEN2 code[6], on the condition that the reactor was shut down at EOEC.

Decay heat was calculated with regard to 1/3-core discharged assemblies burnt through three cycles. Since it is difficult to evaluate an isotopic distribution of fission products by the CITATION code, we made a substitutional use of the ORIGEN2 code for this burnup and decay calculations with following fission product yield data adaptions:

Isotope	^{235}U	^{238}U	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am
Applied FP yield data	^{235}U	^{238}U	^{238}U	^{239}Pu	^{241}Pu	^{241}Pu	^{241}Pu	^{241}Pu

6) Curie increments

——— See Table 3-9

7) Toxicity hazard increments

——— See Table 3-10

References

(1) T. Nakagawa, Summary of JENDL-2 General Purpose File, NEANDC(J)99/AU, JAERI-M 84-103,(1984).

(2) T. Nakagawa, Evaluation of Neutron Nuclear Data for Curium Isotopes, NEANDC(J)152/AU, JAERI-M 90-101,(1990).

(3) T. B. Fowler, D.R. Vondy and G.W. Cunningham, Nuclear Reactor Core Analysis Code: CITATION, ORNL-TM-2496, Rev. 2,(1970).

(4) Katz, Seaborg and Morss (editors), The Chemistry of the Actinide Elements Volume 1, 2, Chapman and Hall (1986)

(5) R. Sher, Fission Energy Release for 16 Fissioning Nuclides, Proc. Specialists' Mtg. Nuclear Data Evaluation and Procedures, Upton, New York, 1980,BNL-NCS-51363.

(6) A. G. Croff, A User's Manual for the ORIGEN2 Computer Code, ORNL-TM-7175,(1980).

(7) R. J. Tuttle, Review of Delayed Neutron Yields in Nuclear Fission, Proc. Consultants' Mtg. Delayed Neutron Properties, Vienna, Austria, March 26-30, 1979, INDC-NDS-107/G ± Special, p.29, International Atomic Energy Agency (1979).

(8) D.Saphier, D.Ilberg, S.Shalev, and S. Yiftah, Nucl. Sci. Eng., 62, 660 (1977).

Table 2-1 Broad 18-group energy boundaries

Group #/18	Upper Energy Boundary(eV)		
1	1.000E+7	10	4.087E+4
2	6.065E+6	11	1.931E+4
3	3.679E+6	12	9.119E+3
4	2.231E+6	13	4.307E+3
5	1.353E+6	14	2.035E+3
6	8.209E+5	15	9.611E+2
7	3.877E+5	16	4.540E+2
8	1.832E+5	17	2.144E+2
9	8.652E+4	18	1.013E+2

Table 2-2 BOL eigenvalue
((Startup Core))

Mesh width	Eigenvalue	Convergence Criterion*
R:≃3cm Z:=5cm	1.09775	0.00001
R:≃0cm** Z:≃0cm	1.09095	--------

*Common to both eigenvalue and flux
**See Appendix A for detail

Table 2-3 Broad group flux spectrum at core center at BOL
((Startup Core))

Group #/18	Group Flux	Normalized to Unity
1	1.317E+13	3.98E−03
2	5.497E+13	1.66E−02
3	1.219E+14	3.68E−02
4	1.848E+14	5.58E−02
5	2.536E+14	7.66E−02
6	6.459E+14	1.95E−01
7	7.431E+14	2.24E−01
8	5.611E+14	1.69E−01
9	3.527E+14	1.06E−01
10	2.116E+14	6.39E−02
11	1.096E+14	3.31E−02
12	3.549E+13	1.07E−02
13	9.815E+12	2.96E−03
14	1.175E+13	3.55E−03
15	2.149E+12	6.49E−04
16	2.506E+11	7.57E−05
17	3.031E+10	9.15E−06
18	4.316E+09	1.30E−06
Total	3.312E+15	1.000

Table 2-4 k_∞ at core center at BOL ((Startup Core))

	Fission Production*	Absorption*	k_∞ = FP/A
Group sum over core center region	2.9857E-2	1.6749E-2	1.7826

*Arbitrary unit

Table 2-5 Core leakage and absorption at BOL ((Startup Core))

	Leakage (events/sec)	Absorption (events/sec)
Group sum over Core	4.787E+19	8.179E+19

Table 2-6 Model leakage and absorption at BOL ((Startup Core))

	Leakage (events/sec)	Absorption (events/sec)
Group sum over Model	3.333E+18	1.263E+20

Table 2-7 Core capture fractions at BOL ((Startup Core))

	capture (events/sec)	Sum	Fractions (Cap / Total abs)
Heavy metal			
U-235	7.150E+16		
U-236	0.0		
U-238	1.786E+19		
Np-237	1.777E+18		
Pu-236	2.217E+12		
Pu-238	1.893E+17		
Pu-239	4.781E+18		
Pu-240	2.226E+18		
Pu-241	1.498E+18	3.073E+19	0.3757
Pu-242	3.656E+17		
Am-241	1.068E+18		
Am-242m	1.124E+15		
Am-243	8.246E+17		
Cm-242	1.104E+14		
Cm-243	3.434E+14		
Cm-244	6.509E+16		
Cm-245	2.142E+15		
Cm-246	4.122E+14		
Structure			
Zr	6.950E+17		
Fe	1.679E+18		
Cr	2.323E+17	2.780E+18	0.03399
Mo	1.256E+17		
Ni	3.611E+16		
Mn-55	1.164E+16		
Coolant			
Na-23	9.867E+16	9.867E+16	0.001206
Total	3.361E+19	3.361E+19	0.4109

Table 2-8 Energy-averaged cross sections of TRU isotopes at core center ((Startup Core))

	σ_f (barn)	$\nu\sigma_f$ (barn)	σ_c (barn)
Np-237	4.326E−1	1.192E+0	9.342E−1
Pu-236	8.234E−1	2.490E+0	5.623E−1
Pu-238	1.205E+0	3.653E+0	5.437E−1
Pu-239	1.682E+0	4.994E+0	2.624E−1
Pu-240	4.826E−1	1.451E+0	3.171E−1
Pu-241	2.008E+0	6.027E+0	3.276E−1
Pu-242	3.569E−1	1.086E+0	2.679E−1
Am-241	3.993E−1	1.404E+0	1.232E+0
Am-242m	2.591E+0	8.680E+0	2.988E−1
Am-243	3.224E−1	1.140E+0	9.501E−1
Cm-242	6.349E−1	2.382E+0	3.214E−1
Cm-243	2.783E+0	9.798E+0	1.239E−1
Cm-244	5.487E−1	1.927E+0	3.336E−1
Cm-245	2.232E+0	8.746E+0	1.212E−1
Cm-246	3.552E−1	1.261E+0	1.852E−1

Table 2-9 Decay constants

Isotope Before Decay	Decay Type	Isotope After Decay	λ(sec-1)
U-235	----	----	----
U-236	----	----	----
U-238	----	----	----
Np-237	----	----	----
Pu-236	α	Out of chain	7.709E-9
Pu-238	α	Out of chain	2.505E-10
Pu-239	α	U-235	9.116E-13
Pu-240	α	U-236	3.349E-12
Pu-241	β	Am-241	1.526E-9
Pu-242	α	U-238	5.841E-14
Am-241	α	Np-237	5.080E-11
Am-242m	IT	Pu-242*	1.559E-10
Am-243	α	Pu-239**	2.978E-12
Cm-242	α	Pu-238	4.924E-8
Cm-243	α	Pu-239	7.712E-10
Cm-244	α	Pu-240	1.214E-9
Cm-245	α	Pu-241	2.586E-12
Cm-246	α	Pu-242	4.647E-12

*via Am-242(EC),**via Np-239(β)

Table 2-10 Fission energy of each isotope

Isotope Before Decay	Fission energy (MeV / fission)	Correction Factor for (n,γ)	Corrected Fission Energy(MeV)
U-235	193.72	1.009	195.4
U-236	191.62	1.088	208.5
U-238	194.81	1.123	218.7
Np-237	193.70	1.077	208.5
Pu-236	N/A	----	209.5*
Pu-238	197.21	N/A	197.2
Pu-239	199.92	1.007	201.3
Pu-240	197.79	1.025	202.8
Pu-241	201.98	1.005	203.0
Pu-242	200.88	1.027	206.3
Am-241	201.02	1.100	221.1
Am-242m	N/A	----	215.5*
Am-243	201.02	1.070	215.2
Cm-242	N/A	----	219.4*
Cm-243	N/A	----	219.8*
Cm-244	205.61	1.034	212.6
Cm-245	N/A	----	220.5*
Cm-246	N/A	----	220.9*

*taken from ORIGEN2

Table 2-11 K_{eff} and flux amplitude at each time step ((Startup Core))

	Effective Full Power Days	k_{eff} (without mesh-correction)	Maximum Power Density (W / cc)	Flux amplitude at core center ($n/cm^2/sec$)
BOL	0.0	1.09775	560.49	3.312E+15
Step1	62.05	1.08583	556.32	3.352E+15
Step2	124.10	1.07406	552.33	3.393E+15
Step3	186.15	1.06245	548.48	3.434E+15
Step4	248.20	1.05099	544.80	3.477E+15
Step5 =EOL	310.25	1.03967	541.27	3.520E+15

Table 2-12 Mass increment by isotope (whole core)
((Startup Core))

Isotope	Mass at BOL (kg)	Mass at EOL (kg)	Mass Increment (kg)
U-235	2.5426E+1	2.1797E+1	-3.6286E+0
U-236	0.0	7.9084E−1	7.9084E−1
U-238	1.2850E+4	1.2594E+4	-2.5680E+2
Np-237	2.3899E+2	2.1539E+2	-2.3601E+1
Pu-236	4.9363E−4	3.5629E−3	3.0693E−3
Pu-238	4.4891E+1	5.9496E+1	1.4605E+1
Pu-239	2.2675E+3	2.1172E+3	-1.5031E+2
Pu-240	8.9201E+2	8.8672E+2	-5.2900E+0
Pu-241	6.0318E+2	4.9902E+2	-1.0415E+2
Pu-242	1.7537E+2	1.8317E+2	7.8000E+0
Am-241	1.1298E+2	1.1947E+2	6.4920E+0
Am-242m	5.0172E−1	2.5790E+0	2.0773E+0
Am-243	1.1257E+2	1.0519E+2	-7.3760E+0
Cm-242	4.3978E−2	4.5424E+0	4.4984E+0
Cm-243	3.5670E−1	3.4677E−1	-9.9330E−3
Cm-244	2.5156E+1	3.1211E+1	6.0547E+0
Cm-245	2.3245E+0	2.6714E+0	3.4688E−1
Cm-246	2.9000E−1	3.0323E−1	1.3235E−2
Cm-247	0.0	4.7291E−3	4.7291E−3
Fission Products	Substitution :		
U-235	U-236	2.9595E+0	2.9595E+0
U-238	Np-237	6.5387E+1	6.5387E+1
Pu-239	Pu236~240	3.3647E+2	3.3647E+2
Pu-241	Others	1.0429E+2	1.0429E+2

Table 2-13 EOL eigenvalue ((Startup Core))

Mesh width	Eigenvalue	Convergence Criterion*
R:≃3cm Z:=5cm	1.03967	0.00001
R:≃0cm** Z:≃0cm	1.03288	--------

*Common to both eigenvalue and flux
**See Appendix A for detail

Table 2-14 Burnup swing
((Startup Core))

Mesh width	Burnup Swing (%dk/kk')
R:≃3cm Z:=5cm	5.088

Table 2-15 TRU breeding ratio ((Startup Core))

	TRU Mass at BOL(kg)	TRU Mass at EOL(kg)	Breeding Ratio
Sum	4.4761E+3	4.2273E+3	0.94442

Table 2-16 Broad group flux spectrum at core center at EOL
((Startup Core))

Group #/18	Group Flux	Normalized to Unity
1	1.356E+13	3.85E−03
2	5.658E+13	1.61E−02
3	1.259E+14	3.58E−02
4	1.923E+14	5.46E−02
5	2.652E+14	7.53E−02
6	6.782E+14	1.93E−01
7	7.880E+14	2.24E−01
8	6.014E+14	1.71E−01
9	3.815E+14	1.08E−01
10	2.306E+14	6.55E−02
11	1.206E+14	3.43E−02
12	3.933E+13	1.12E−02
13	1.100E+13	3.12E−03
14	1.338E+13	3.80E−03
15	2.494E+12	7.09E−04
16	2.957E+11	8.40E−05
17	3.636E+10	1.03E−05
18	5.047E+09	1.43E−06
Total	3.520E+15	1.000

Table 3-1 Determination of the fresh fuel enrichment ((Once-Through Burner Core))

TRU / HM mass ratio	EOEC eigenvalue*
27.0%	1.00307
26.85%	1.00009
26.0%	0.98239

* mesh-effect-free (see Appendix A for detail)

Table 3-2 K_{eff} and flux amplitude at each time step ((Once-Through Burner Core))

	Effective Full Power Days	k_{eff} (without mesh-correction)	Maximum Power Density (W / cc)	Flux amplitude at core center ($n/cm^2/sec$)
BOEC	0.0	1.06259	543.87	3.421E+15
Step1	62.05	1.05113	540.25	3.463E+15
Step2	124.10	1.03980	536.80	3.507E+15
Step3	186.15	1.02862	533.49	3.551E+15
Step4	248.20	1.01758	530.33	3.597E+15
Step5 =EOEC	310.25	1.00669	527.31	3.643E+15

Table 3-3 Burnup swing ((Once-Through Burner Core))

Mesh width	k_{eff} at BOEC	k_{eff} at EOEC	Burnup Swing (%dk/kk')
R: $\simeq$ 3cm Z: = 5cm	1.06259	1.00669	5.227
R: $\simeq$ 0cm* Z: $\simeq$ 0cm	1.05600	1.00009	---------

* See Appendix A for detail

Table 3-4 TRU breeding ratio ((Once-Through Burner Core))

	TRU Mass at BOEC(kg)	TRU Mass at EOEC(kg)	Breeding Ratio
Sum	4.3925E+3	4.1459E+3	0.94386

Table 3-5 Mass increment by isotope (whole core)
((Once-Through Burner Core))

Isotope	Fresh Fuel Mass (kg)	Mass at BOEC (kg)	Mass at EOEC (kg)	Mass increment* (kg)
U-235	2.5071E+1	2.1534E+1	1.8338E+1	-3.1956E+0
U-236	0.0	7.8124E−1	1.4594E+0	6.7818E−1
U-238	1.2671E+4	1.2408E+4	1.2148E+4	-2.5950E+2
Np-237	2.5163E+2	2.2645E+2	2.0316E+2	-2.3288E+1
Pu-236	5.2189E−4	3.3087E−3	5.4899E−3	2.1812E−3
Pu-238	4.7063E+1	6.3128E+1	7.7268E+1	1.4140E+1
Pu-239	2.3672E+3	2.2017E+3	2.0507E+3	-1.5101E+2
Pu-240	9.2729E+2	9.2089E+2	9.1245E+2	-8.4440E+0
Pu-241	6.2441E+2	5.1821E+2	4.3003E+2	-8.8183E+1
Pu-242	1.8080E+2	1.8819E+2	1.9365E+2	5.4590E+0
Am-241	1.1696E+2	1.2465E+2	1.2602E+2	1.3670E+0
Am-242m	5.1723E−1	2.6611E+0	4.5086E+0	1.8475E+0
Am-243	1.1556E+2	1.0803E+2	1.0115E+2	-6.8810E+0
Cm-242	4.5339E−2	3.4447E+0	5.9307E+0	2.4861E+0
Cm-243	3.6626E−1	3.7626E−1	4.1737E−1	4.1116E−2
Cm-244	2.5722E+1	3.1701E+1	3.7081E+1	5.3799E+0
Cm-245	2.3672E+0	2.7740E+0	3.2283E+0	4.5433E−1
Cm-246	2.9403E−1	3.0979E−1	3.2841E−1	1.8623E−2
Cm-247	0.0	5.0565E−3	1.0178E−2	5.1211E−3
Sum				
TRU	4.6602E+3	4.3925E+3	4.1459E+3	-2.4661E+2
HM	1.7356E+4	1.6822E+4	1.6314E+4	-5.0862E+2
FP				
U-235	0.0	2.8929E+0	5.5097E+0	2.6168E+0
U-238	0.0	6.6404E+1	1.3147E+2	6.5062E+1
Pu-239	0.0	3.5796E+2	7.0239E+2	3.4443E+2
Pu-241	0.0	1.0663E+2	2.0363E+2	9.7001E+1

* Mass increment = EOEC mass − BOEC mass

Table 3-6 Mass increment of TRU isotopes (whole core)
((Once-Through Burner Core))

	TRU increment (EOEC − BOEC)
in units of kg	-2.4661E+2
in units of kg / MW_{th}days	-5.0467E−4

Table 3-7 Safety parameters
((Once-Through Burner Core))

	BOEC	EOEC
β_{eff}	3.504E−3	3.460E−3
Doppler Coefficient (%Tdk/kk'/dT) (%Tdk/dT)	 -0.156 -0.175	 -0.167 -0.168
Sodium Void Worth (%dk/kk') Core Core+Plenum	 0.674 -2.177	 0.884 -2.133

Table 3-8 Decay heat level of the discharged fuel (1/3-core, 140 driver assemblies) ((Once-Through Burner Core))

[Watts]

	0 =EOEC	1 hr	1 month	1 year	10 y	10^2 y	10^3 y	10^4 y
Heavy metal	1.602E+6	1.071E+6	2.919E+5	1.163E+5	5.480E+4	3.013E+4	7.503E+3	1.955E+3
FP	2.682E+7	5.418E+6	6.496E+5	1.363E+5	1.013E+4	1.120E+3	4.912E−1	4.623E−1
Total	2.842E+7	6.489E+6	9.415E+5	2.526E+5	6.493E+4	3.125E+4	7.504E+3	1.956E+3

Table 3-9 Curie increments for each isotope (whole core) ((Once-Through Burner Core))

(1) Radioactivity

[Curies]

	Fresh Fuel	0 =BOEC	1 year	10 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
U-235	5.422E−2	4.657E−2	3.969E−2	4.081E−2	5.210E−2	1.637E−1	1.151E+0	4.386E+0	4.646E+0
U-236	0.0	5.056E−2	9.539E−2	1.511E−1	7.197E−1	6.159E+0	3.956E+1	6.032E+1	5.874E+1
U-238	4.262E+0	4.173E+0	4.087E+0	4.087E+0	4.087E+0	4.087E+0	4.087E+0	4.097E+0	4.141E+0
Np-237	1.775E+2	1.597E+2	1.433E+2	1.453E+2	1.913E+2	4.492E+2	5.286E+2	5.143E+2	3.843E+2
Pu-236	2.774E+2	1.759E+3	2.814E+3	3.155E+2	9.909E−8	0.0	0.0	0.0	0.0
Pu-238	8.061E+5	1.081E+6	1.343E+6	1.327E+6	6.660E+5	9.255E+2	1.358E−15	0.0	0.0
Pu-239	1.469E+5	1.367E+5	1.273E+5	1.272E+5	1.270E+5	1.243E+5	9.863E+4	7.641E+3	9.019E−4
Pu-240	2.106E+5	2.091E+5	2.072E+5	2.094E+5	2.131E+5	1.938E+5	7.464E+4	5.355E+0	0.0
Pu-241	6.434E+7	5.340E+7	4.399E+7	2.852E+7	3.753E+5	5.119E+2	2.457E+2	1.594E−1	0.0
Pu-242	7.102E+2	7.392E+2	7.608E+2	7.608E+2	7.616E+2	7.624E+2	7.510E+2	6.395E+2	1.275E+2
Am-241	4.012E+5	4.276E+5	4.429E+5	9.475E+5	1.658E+6	3.948E+5	2.457E+2	1.678E−1	0.0
Am-242m	5.422E+3	2.790E+4	4.724E+4	4.533E+4	3.008E+4	4.964E+2	7.458E−16	0.0	0.0
Am-243	2.305E+4	2.155E+4	2.016E+4	2.016E+4	1.998E+4	1.837E+4	7.888E+3	1.683E+0	9.036E−4
Cm-242	1.501E+5	1.141E+7	1.557E+7	3.464E+4	2.298E+4	3.791E+2	5.715E−16	0.0	0.0
Cm-243	1.892E+4	1.943E+4	2.148E+4	1.726E+4	1.933E+3	6.027E−7	0.0	0.0	0.0
Cm-244	2.083E+6	2.567E+6	2.985E+6	2.116E+6	6.750E+4	7.397E−11	0.0	0.0	0.0
Cm-245	4.066E+2	4.765E+2	5.545E+2	5.541E+2	5.500E+2	5.112E+2	2.453E+2	1.529E−1	0.0
Cm-246	9.039E+1	9.523E+1	1.010E+2	1.008E+2	9.951E+1	8.721E+1	2.333E+1	4.377E−5	0.0
Cm-247	0.0	4.694E−4	9.450E−4	9.450E−4	9.450E−4	9.450E−4	9.441E−4	9.404E−4	9.037E−4

Table 3-9 (continued)

(2) Radioactivity change from BOEC

[Curies]

	1 year	10 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
U-235	-6.886E−3	-5.761E−3	5.528E−3	1.171E−1	1.104E+0	4.340E+0	4.599E+0
U-236	4.483E−2	1.006E−1	6.691E−1	6.109E+0	3.951E+1	6.027E+1	5.869E+1
U-238	-8.661E−2	-8.661E−2	-8.661E−2	-8.661E−2	-8.661E−2	-7.652E−2	-3.279E−2
Np-237	-1.640E+1	-1.435E+1	3.156E+1	2.895E+2	3.689E+2	3.546E+2	2.246E+2
Pu-236	1.055E+3	-1.443E+3	-1.759E+3	-1.759E+3	-1.759E+3	-1.759E+3	-1.759E+3
Pu-238	2.614E+5	2.457E+5	-4.153E+5	-1.080E+6	-1.081E+6	-1.081E+6	-1.081E+6
Pu-239	-9.354E+3	-9.416E+3	-9.664E+3	-1.240E+4	-3.803E+4	-1.290E+5	-1.367E+5
Pu-240	-1.882E+3	3.204E+2	3.954E+3	-1.528E+4	-1.345E+5	-2.091E+5	-2.091E+5
Pu-241	-9.409E+6	-2.488E+7	-5.302E+7	-5.340E+7	-5.340E+7	-5.340E+7	-5.340E+7
Pu-242	2.166E+1	2.166E+1	2.244E+1	2.323E+1	1.184E+1	-9.971E+1	-6.116E+2
Am-241	1.526E+4	5.199E+5	1.230E+6	-3.276E+4	-4.273E+5	-4.276E+5	-4.276E+5
Am-242m	1.934E+4	1.743E+4	2.180E+3	-2.740E+4	-2.790E+4	-2.790E+4	-2.790E+4
Am-243	-1.382E+3	-1.382E+3	-1.562E+3	-3.179E+3	-1.366E+4	-2.154E+4	-2.155E+4
Cm-242	4.163E+6	-1.137E+7	-1.138E+7	-1.141E+7	-1.141E+7	-1.141E+7	-1.141E+7
Cm-243	2.047E+3	-2.177E+3	-1.750E+4	-1.943E+4	-1.943E+4	-1.943E+4	-1.943E+4
Cm-244	4.185E+5	-4.511E+5	-2.499E+6	-2.567E+6	-2.567E+6	-2.567E+6	-2.567E+6
Cm-245	7.798E+1	7.764E+1	7.352E+1	3.470E+1	-2.312E+2	-4.763E+2	-4.765E+2
Cm-246	5.722E+0	5.599E+0	4.277E+0	-8.019E+0	-7.190E+1	-9.523E+1	-9.523E+1
Cm-247	4.756E−4	4.756E−4	4.756E−4	4.756E−4	4.747E−4	4.710E−4	4.343E−4
TRU total [Ci / $MW_{th}d$]	-9.292E+0	-7.353E+1	-1.353E+2	-1.403E+2	-1.414E+2	-1.418E+2	-1.418E+2

Table 3-10 Toxicity increments for each isotope (whole core) ((Once-Through Burner Core))

(1) Toxicity

[Cancer Deaths]

	Fresh Fuel	0 =BOEC	1 year	10 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
U-235	3.920E−1	3.367E−1	2.869E−1	2.951E−1	3.767E−1	1.183E+0	8.319E+0	3.171E+1	3.359E+1
U-236	0.0	3.792E−1	7.155E−1	1.133E+0	5.398E+0	4.619E+1	2.967E+2	4.524E+2	4.405E+2
U-238	2.971E+1	2.909E+1	2.848E+1	2.848E+1	2.848E+1	2.848E+1	2.848E+1	2.855E+1	2.886E+1
Np-237	3.499E+4	3.149E+4	2.826E+4	2.866E+4	3.772E+4	8.859E+4	1.042E+5	1.014E+5	7.578E+4
Pu-236									
Pu-238	1.984E+8	2.661E+8	3.304E+8	3.266E+8	1.639E+8	2.278E+5	3.343E−13	0.0	0.0
Pu-239	3.931E+7	3.656E+7	3.406E+7	3.404E+7	3.397E+7	3.324E+7	2.638E+7	2.044E+6	2.413E−1
Pu-240	5.633E+7	5.594E+7	5.544E+7	5.603E+7	5.700E+7	5.185E+7	1.997E+7	1.432E+3	0.0
Pu-241									
Pu-242	1.900E+5	1.977E+5	2.035E+5	2.035E+5	2.037E+5	2.039E+5	2.009E+5	1.711E+5	3.412E+4
Am-241	1.095E+8	1.167E+8	1.209E+8	2.586E+8	4.523E+8	1.077E+8	6.706E+4	4.580E+1	0.0
Am-242m	1.451E+6	7.463E+6	1.264E+7	1.213E+7	8.046E+6	1.328E+5	1.995E−13	0.0	0.0
Am-243	6.289E+6	5.880E+6	5.502E+6	5.502E+6	5.453E+6	5.012E+6	2.153E+6	4.594E+2	2.466E−1
Cm-242	1.036E+6	7.870E+7	1.074E+8	2.390E+5	1.585E+5	2.616E+3	3.943E−15	0.0	0.0
Cm-243	3.725E+6	3.826E+6	4.229E+6	3.398E+6	3.806E+5	1.187E−4	0.0	0.0	0.0
Cm-244	3.395E+8	4.184E+8	4.866E+8	3.449E+8	1.100E+7	1.206E−8	0.0	0.0	0.0
Cm-245	1.155E+5	1.353E+5	1.575E+5	1.574E+5	1.562E+5	1.452E+5	6.966E+4	4.521E+1	0.0
Cm-246	2.567E+4	2.705E+4	2.867E+4	2.864E+4	2.826E+4	2.477E+4	6.625E+3	1.243E−2	0.0
Cm-247									

Table 3-10 (continued)

(2) Toxicity change from BOEC

[Cancer Deaths]

	1 year	10 y	10^2 y	10^3 y	10^4 y	10^5 y	10^6 y
U-235	-4.979E-2	-4.166E-2	3.997E-2	8.467E-1	7.982E+0	3.137E+1	3.325E+1
U-236	3.363E-1	7.542E-1	5.018E+0	4.582E+1	2.963E+2	4.520E+2	4.402E+2
U-238	-6.037E-1	-6.037E-1	-6.037E-1	-6.037E-1	-6.037E-1	-5.333E-1	-2.286E-1
Np-237	-3.233E+3	-2.830E+3	6.223E+3	5.710E+4	7.274E+4	6.993E+4	4.429E+4
Pu-236							
Pu-238	6.434E+7	6.046E+7	-1.022E+8	-2.659E+8	-2.661E+8	-2.661E+8	-2.661E+8
Pu-239	-2.502E+6	-2.519E+6	-2.585E+6	-3.316E+6	-1.017E+7	-3.451E+7	-3.656E+7
Pu-240	-5.035E+5	8.571E+4	1.058E+6	-4.088E+6	-3.597E+7	-5.594E+7	-5.594E+7
Pu-241							
Pu-242	5.794E+3	5.794E+3	6.004E+3	6.214E+3	3.167E+3	-2.667E+4	-1.636E+5
Am-241	4.165E+6	1.419E+8	3.356E+8	-8.941E+6	-1.166E+8	-1.167E+8	-1.167E+8
Am-242m	5.174E+6	4.663E+6	5.831E+5	-7.330E+6	-7.463E+6	-7.463E+6	-7.463E+6
Am-243	-3.772E+5	-3.772E+5	-4.262E+5	-8.676E+5	-3.727E+6	-5.879E+6	-5.880E+6
Cm-242	2.873E+7	-7.846E+7	-7.854E+7	-7.870E+7	-7.870E+7	-7.870E+7	-7.870E+7
Cm-243	4.031E+5	-4.287E+5	-3.446E+6	-3.826E+6	-3.826E+6	-3.826E+6	-3.826E+6
Cm-244	6.822E+7	-7.353E+7	-4.074E+8	-4.184E+8	-4.184E+8	-4.184E+8	-4.184E+8
Cm-245	2.215E+4	2.205E+4	2.088E+4	9.854E+3	-6.566E+4	-1.353E+5	-1.353E+5
Cm-246	1.625E+3	1.590E+3	1.215E+3	-2.277E+3	-2.042E+4	-2.705E+4	-2.705E+4
Cm-247							
TRU total [CD / $MW_{th}d$]	3.431E+2	1.060E+2	-5.265E+2	-1.619E+3	-1.926E+3	-2.021E+3	-2.026E+3

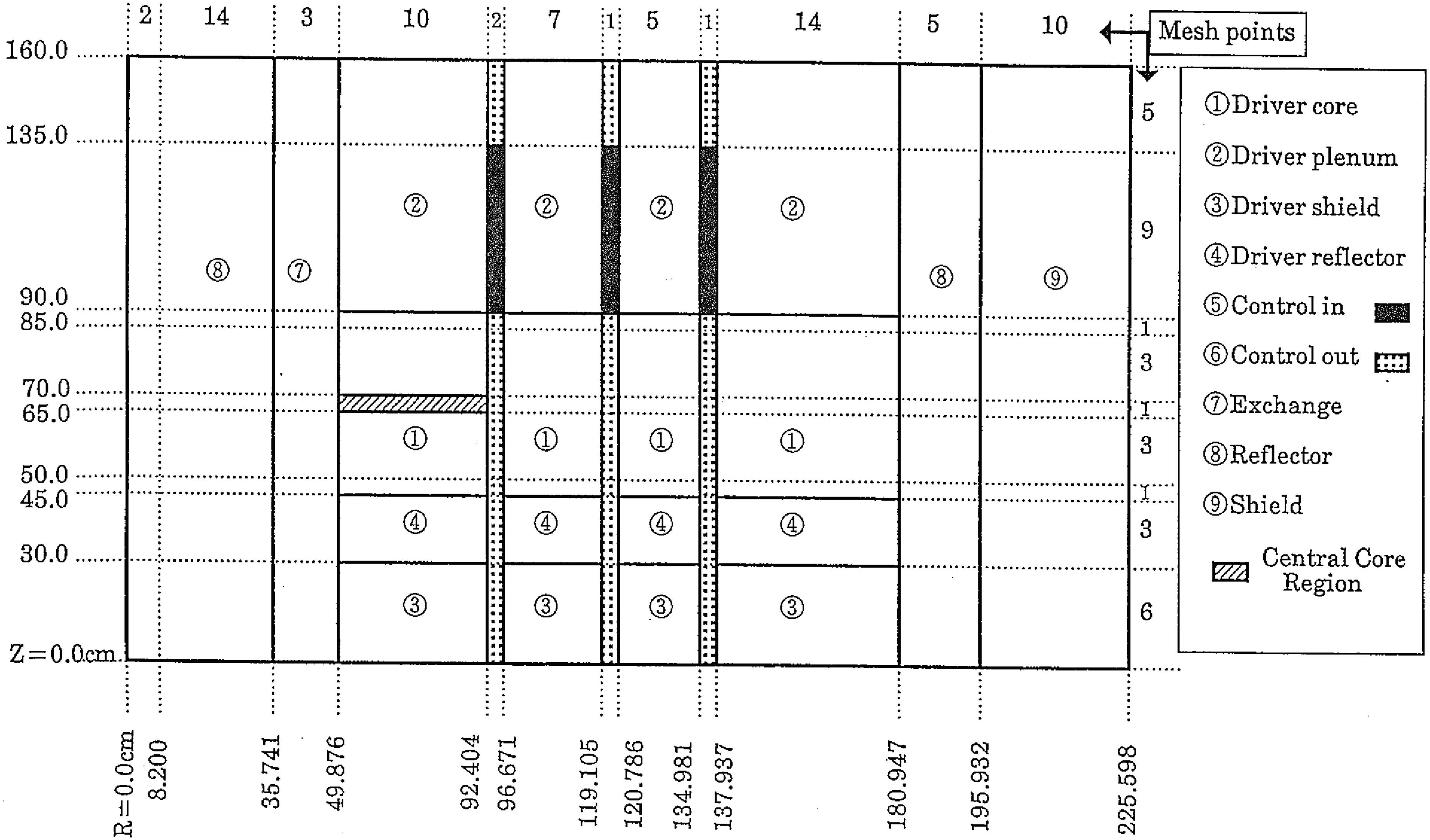


Fig. 2-1 RZ model for the Breeding Ratio ≃0.5 Core

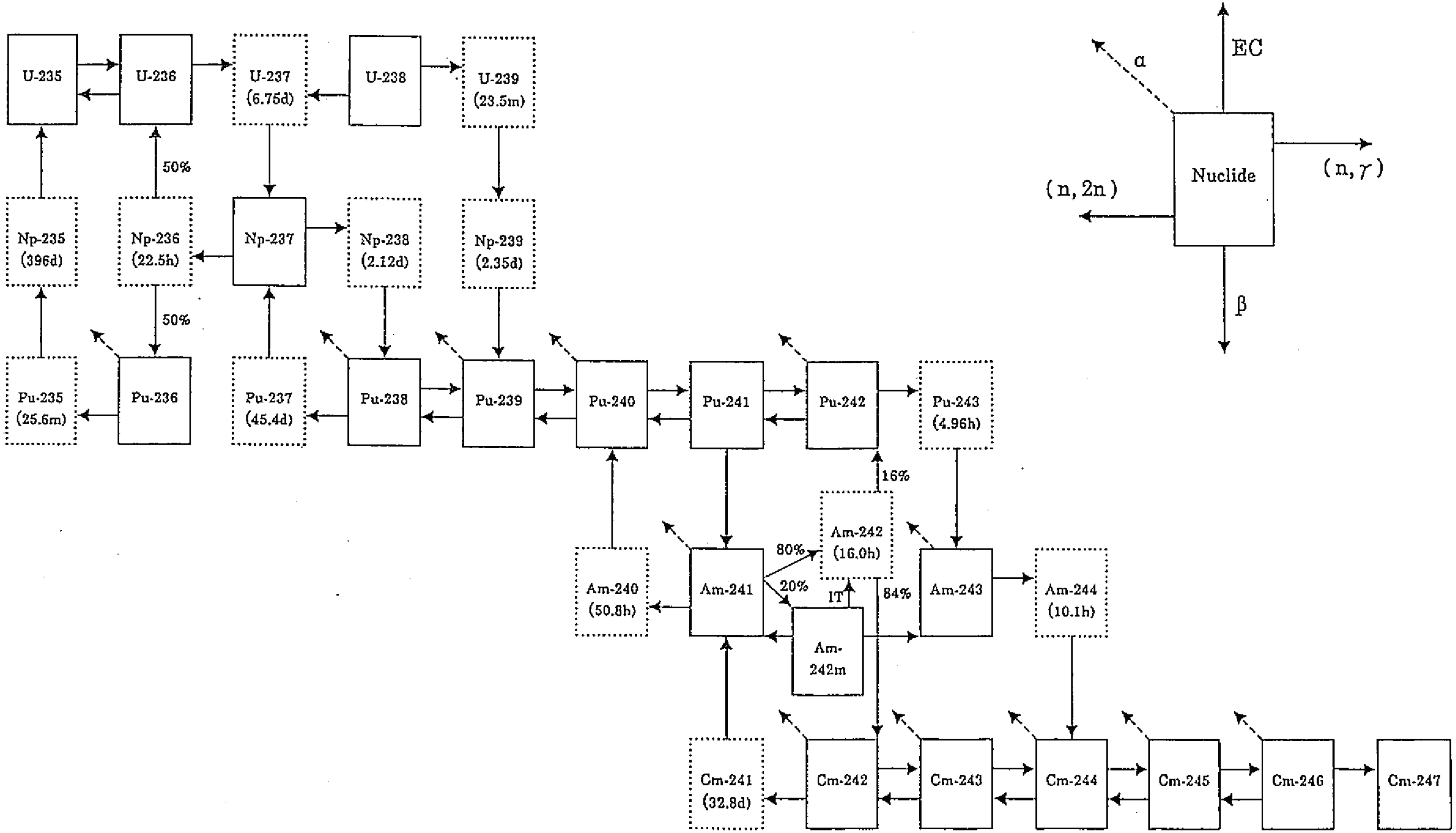


Fig. 2-2 Burnup chain model

Appendix A Mesh correction to k_{eff}

Error due to mesh effect in an RZ diffusion calculation is formulated as follows:

$$\Delta k = k(N_R \to \infty, Nz \to \infty) - k(N_R, N_z)$$
$$= \Delta k_R + \Delta k_z = A/N_R{}^2 + B/N_z{}^2 \qquad (1)$$

where, N_R and N_z are numbers of meshes along R- and Z-axis, respectively.

Consider a case N_R is doubled, i.e., $R' = 1/2R$. The error will be

$$\Delta k' = k(N_R \to \infty, Nz \to \infty) - k(2N_R, N_z) = \Delta k'_R + \Delta k_z = A/4N_R{}^2 + B/N_z{}^2 \qquad (2)$$

Subtracting Eq.(2) from Eq.(1), Δk_R can be obtained as:

$$\Delta k_R = 4/3[k(2N_R, N_z) - k(N_R, N_z)] \qquad (3)$$

Same is true for the case when N_z is doubled and obtain Δk_z.

Tables A-1 and A-2 show the mesh corrections using Eq.(3). As mesh effect on burnup reactivity is small, same factor can be applied to both BOC and EOC k_{eff}.

Table A-1 Mesh correction to k_{eff}
for the Startup Core Benchmark

Mesh width	k_{eff} (MOL*)	Δk_{eff} R or Z
Reference R≃3cm, Z=5cm	1.07720	-----
R' = 1/2 R	1.07584	-0.00182
Z' = 1/2 Z	1.07348	-0.00497
Mesh correction	-----	-0.00680

* Preliminary composition

Table A-2 Mesh correction to k_{eff}
for the Once-Through Core Benchmark

Mesh width	k_{eff} (MOEC*)	Δk_{eff} R or Z
Reference R≃3cm, Z=5cm	1.03716	-----
R' = 1/2 R	1.03585	-0.00175
Z' = 1/2 Z	1.03353	-0.00484
Mesh correction	-----	-0.00659

* Preliminary composition

Appendix B

Specification for Metal-Fueled Fast Reactor Benchmarks

Extracted from NEA/NSC/DOC(93)24

§ 0 Specifications Which Are Common to All Benchmarks

Core Layouts and Dimensions

The metal-fueled fast reactor benchmark is based on 600 MWe (1575 MWth) core designs developed previously and reported in Refs. 1, 2, and 3. For the purpose of serving as a benchmark, the dimensions are idealized and are shown in Fig.4. *It is assumed that the neutronics modeling will employ Hex-Z spatial nodalization; if RZ is used, the benchmark participant should make the transformation of Fig.4 to an RZ representation and describe how this transformation was done.*

Feedstock Stream Composition

The composition of the TRU feedstock stream from LWR once through discharge fuel after pyroprocessing is given in Table 1. It is based on PWR UO_2 fuel discharged after 35000 MWd/tonne initial heavy metal average burnup, 3.17 years cooling followed by an assumed instantaneous pyroprocessing and fabrication (in which all TRU isotopes report to a common product stream -- making the Pu^{241}/Am^{241} ratio independent of the cooling interval subsequent to processing and dependent only on the 3.17 years total duration between LWR discharge and LMR fueling).

Base Temperatures for the Cross Section Generation

Driver	fuel	850K
	structure	750K
	coolant	700K
Non-fueled	structure	650K
	coolant	650K

The participant should indicate in his solution which isotopes are treated as resonance isotopes and in his display of kinetics parameters which isotopes are Doppler broadened.

Toxicity Factors

Radiotoxicity can be expressed in terms of induced cancer deaths over and above that due to natural causes. The related isotopic radiotoxicity factors per isotopic decay, f_i, are calculated using values provided in Ref. 5 by Cohen. These radiotoxicity factors assess the hazard (consequences given oral intake) for the actinide elements; "hazard" (as opposed to "risk") analyses assume complete exposure to the radioactive inventory and do not evaluate mitigating effects (e.g., shielding, barriers to release, etc.) In this paper, the fatal cancer dose measure developed by Cohen[5] is utilized to quantify the hazard; this measure is based on dose exposure data from the ICRP publications[6] and cancer risk data from the BEIR reports.

Table 3 lists the radiotoxicity factors per unit isotopic decay (in curies) which were generated as indicated above. The values for non-TRU isotopes are provided for information only.

Table 1. LWR Transuranic Isotopics

Isotopic values are the weight fraction of the individual isotope in the total transuranic mass

Isotope	LWR +3.17 y Cool
Np^{237}	5.40-2
Pu^{236}	1.12-7
Pu^{238}	1.01-2
Pu^{239}	0.508
Pu^{240}	0.199
Pu^{241}	0.134
Pu^{242}	3.88-2
Am^{241}	2.51-2
Am^{242m}	1.11-4
Am^{243}	2.48-2
Cm^{242}	9.73-6
Cm^{243}	7.86-5
Cm^{244}	5.52-3
Cm^{245}	5.08-4
Cm^{246}	6.31-5
MA/fiss. Pu	0.172
MA/Pu	0.124
Np^{237}/MA	0.490
Am^{241}/MA	0.228
Am^{243}/MA	0.225
Np-chain	0.213

MA = sum of minor actinides; fiss. Pu = Pu^{239} + Pu^{241}; Np-chain = Np^{237} + Am^{241} + Pu^{241}

Table 2. Fuel Cycle Assumptions

Reactor Segment of Cycle

Cycle Length	365 days
Capacity Factor	85%
Power Rating	1575 MW_{th}
Core Driver Refueling	⅓ per cycle
Blanket Refueling	¼ per cycle

The 365 day cycle represents the actual refueling interval; thus, a 365 day cycle at an 85% capacity factor implies 310 Effective Full Power Days in the annual operating cycle.

Table 3. Actinide Isotopic Toxicity Factors

	Isotope	Toxicity Factor CD/Ci
TRU		
	Ac^{227}	1.1850E+03
	Th^{229}	1.2730E+02
	Th^{230}	1.9100E+01
	Pa^{231}	3.7200E+02
	U^{234}	7.5900E+00
	U^{235}	7.2300E+00
	U^{236}	7.5000E+00
	U^{238}	6.9700E+00
	Np^{237}	1.9720E+02
	Pu^{238}	2.4610E+02
	Pu^{239}	2.6750E+02
	Pu^{240}	2.6750E+02
	Pu^{242}	2.6750E+02
	Am^{241}	2.7290E+02
	Am^{242m}	2.6750E+02
	Am^{243}	2.7290E+02
	Cm^{242}	6.9000E+00
	Cm^{243}	1.9690E+02
	Cm^{244}	1.6300E+02
	Cm^{245}	2.8400E+02
	Cm^{246}	2.8400E+02
Other		
	Pb^{210}	4.5500E+02
	Ra^{223}	1.5600E+01
	Ra^{226}	3.6300E+01
	Sr^{90}	1.6700E+01
	Y^{90}	6.0000E-01
	Zr^{93}	9.5000E-02
	Tc^{99}	1.7200E-01
	I^{129}	6.4800E+01
	Cs^{135}	8.4000E-01
	Cs^{137}	5.7700E+00
	C^{14}	2.0000E-01
	Ni^{59}	8.0000E-02
	Ni^{63}	3.0000E-02
	Sn^{126}	1.7000E+00

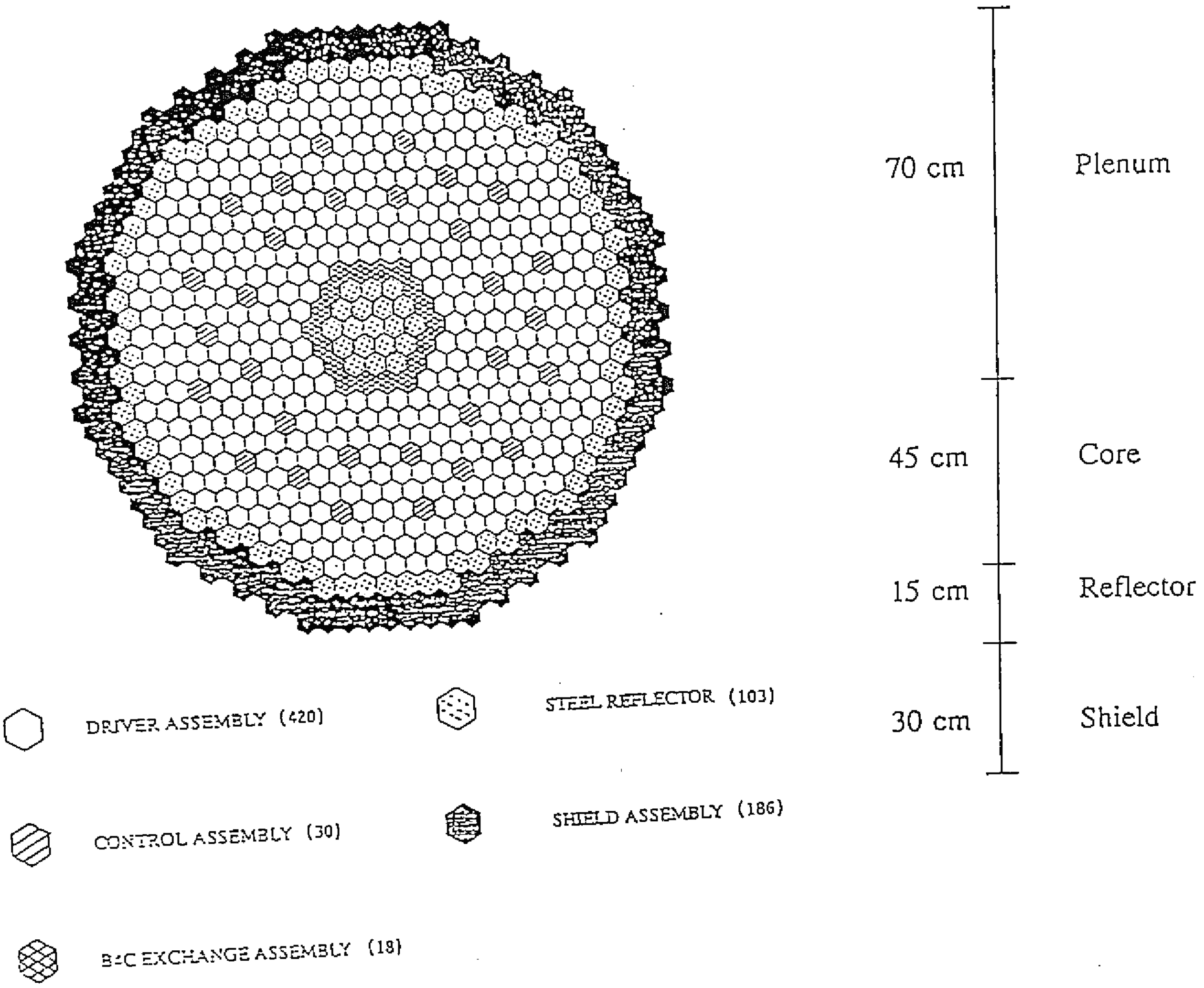
Cancer Deaths/Curie Upon Oral Ingestion

Table 4. MATERIAL COMPOSITION SPECIFICATIONS

(Number Densities in atoms/barn-cm)

Isotope	DRIVER Shield	DRIVER Reflector	DRIVER Core	DRIVER Plenum	CONTROL In	CONTROL Out	EXCHANGE	REFLECTOR	SHIELD
Na-23	7.447-3	7.447-3	7.637-3	1.678-2	8.865-3	2.080-2	6.075-3	3.546-3	3.546-3
Fe	1.179-2	4.821-2	1.790-2	1.790-2	1.538-2	4.865-3	1.406-2	6.088-2	1.516-2
Cr	1.761-3	7.201-3	2.674-3	2.674-3	2.297-3	7.265-4	2.100-3	9.092-3	2.263-3
Mo	7.952-5	3.252-4	1.207-4	1.207-4	1.038-4	3.281-5	9.485-5	4.106-4	1.022-4
Ni	6.499-5	2.658-4	9.868-5	9.868-5	8.479-5	2.681-5	7.751-5	3.356-4	8.354-5
Mn-55	2.777-5	1.136-4	4.217-5	4.217-5	3.624-5	1.146-5	3.313-5	1.434-4	3.570-5
B-10	9.278-3				2.783-2		8.017-3		9.495-3
B-11	3.758-2				3.092-3		3.247-2		3.845-2
C-12	1.171-2				7.731-3		1.012-2		1.199-2
Zr			3.189-3						
U-235			1.632-5						
U-238			8.144-3						
Np-237			1.521-4						
Pu-236			3.155-10						
Pu-238			2.845-5						
Pu-239			1.431-3						
Pu-240			5.606-4						
Pu-241			3.775-4						
Pu-242			1.093-4						
Am-241			7.071-5						
Am-242m			3.127-7						
Am-243			6.987-5						
Cm-242			2.741-8						
Cm-243			2.214-7						
Cm-244			1.555-5						
Cm-245			1.431-6						
Cm-246			1.778-7						

Fig. 4A. Geometry of Breeding Ratio = 0.5 Core



All assemblies have an axial height of 160 cm with a 15.617 cm lattice pitch and are arranged in a configuration with 1/6 core symmetry, as shown in the figure. Only nine distinct material zones are specified. In the driver assemblies, a 30 cm thick lower axial shield is below a 15 cm thick lower reflector zone which is adjacent to the 45 cm tall active core; there is a 70 cm plenum region above the active core. The absorber regions of the control assemblies are parked above the active core. All other assemblies have uniform axial compositions. The isotopic number densities of each non-driver, non-blanket assembly region are specified in Table 4. Table 4 contains the driver and blanket compositions for the first benchmark only.

From Ref. 4.

§ 1 Metal-Fueled Burner Startup Core Benchmark

Introduction and Goals

In this benchmark, the geometry is specified and the BOL composition is specified.

Then, a BOL neutron balance is computed and compared among participants with the goal to assess the degree of spread in neutronics predictions and the reasons (e.g., differing cross sections, leakage treatments, etc.) for the differences.

Then, a single depletion time step of specified duration and energy extraction is computed and both the End of Life (EOL) composition and the EOL neutron balance are compared among participants with the goal to assess the degree of spread in burnup predictions. The depletion step is done with a fission product representation and (artificially) without fission product buildup so as to assess the contribution to differences in EOL neutron balance which can be attributed to different fission product treatments among the participants. Note that for benchmark purposes the control rods are specified to (unphysically) remain fully withdrawn to the top of the fueled region.

This highly idealized benchmark is done preparatory to the subsequent benchmarks of Appendices B and C which are more relevant to the plutonium burning issues. For this idealized case, the intercomparison differences reduce to cross section and modeling effects alone when a specified geometry and BOL compositions are used. Alternately, in the subsequent benchmarks the BOEC composition itself is adjusted by each participant to achieve an EOEC eigenvalue of unity – and thus, the resulting BOEC composition will vary from participant to participant both because of differing eigenvalue, given a composition, and because of differing EOEC compositions, given a specified energy extraction per burn cycle.

Specification of the Model

Figure 4A prescribes the geometry of the core.

Table 4 prescribes the BOL composition by model region.[4]

Table 2 prescribes the burn cycle duration and energy extraction.

Basic Data Reporting

1) Identify the source of the basic nuclear data (e.g., ENDF/B-V) from which the cross sections are generated.

2) Show broad group energy boundaries (express in energy at top of group).

3) Provide a narrative synopsis of the process for broad group cross section preparation (e.g., state slowing down approximation, emission spectrum, choice of composition for collapse spectra, etc.)

BOL Neutron Balance Reporting

1) Provide a narrative synopsis of the spatial representation (e.g. Hex-Z nodal, or if RZ, show dimensions; mesh sizes, etc.)

2) Identification of neutron balance solution algorithm (e.g., code name, type: finite difference, nodal, etc.)

3) BOL eigenvalue and convergence criterion

4) Broad group flux spectrum at core center (specify whether group flux or flux per unit lethargy)

5) k_∞ using central (mesh or node) flux spectrum and core central (mesh or node) composition where

$$k_\infty = \frac{\text{group sum of fission production}}{\text{group sum of absorption}}$$

6) Core leakage/Core Absorption (i.e., for "core" exclude blankets & reflectors)

7) Model Leakage/Model Absorption (i.e., for "model" include all regions)

8) "Core" Capture fractions; where denominator is group and isotope sum of absorption and numerators are:

All Heavy Metal, All Structural, Coolant

9) Energy-averaged cross sections collapsed using central (mesh or node) fluxes*

$<\sigma_f>$, $<\nu\sigma_f>$, $<\sigma_c>$ by TRU isotope

Depletion Methodology Reporting

1) Description of the Burnup Chain Representation
   - diagram of isotopes considered
   - values of branching ratios, λ's, etc.

2) Provide a narrative description of how flux is normalized to prescribed power (e.g., fission only, fission + γ, etc.)

3) Provide a Narrative Synopsis of the Burnup Numerical Solution Process

e.g.,
   - macro fitted vs. exposure -- vis-a-vis number density solution of differential equations
   - one vs. multi energy groups in the depletion equations
   - number of time steps & flux shape re-solution (if any)
   - flux amplitude and time step renormalizations to constant power (if any)
   - time advance numerical method (e.g., Runga Kutta, etc.)

4) Provide a Narrative Synopsis of the Fission Product Representation

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* Where we ask for edits of flux spectrum at core center, we should have said "at the central radius of the fueled annulus" since the core has a central island of non-fuel assemblies.

BOL to EOL Transition and EOL Neutron Balance Reporting

1) Mass Increments by Isotope Occurring as a Result of the Burn Step
   - Sum over entire model of change in mass, δ(mass) by isotope
   - Sum over entire model of δ(mass) for the Fission Products

2) EOL eigenvalue and convergence criterion

3) Burnup Swing = $(k_{EOL} - k_{BOL})/(k_{BOL}\ k_{EOL})$

4) TRU Breeding Ratio =

$$\frac{\text{EOL TRU Mass Summed Over Isotopes for Whole Model}}{\text{BOL TRU Mass Summed Over Isotopes for Whole Model}}^{****}$$

5) EOL Neutron Spectrum at Core Center

****Note that U^{235} is excluded from this definition.

§ 2 Metal-Fueled Once-Through Burner Core Benchmark

Introduction and Goals

In this benchmark, the geometry is specified. Also given are a ⅓ core refueling pattern, a specified time and energy extraction per burn cycle, and a specified composition (isotopic mass fractions) of a TRU feedstream coming from LWR spent fuel processing. Then, a fresh-fuel enrichment (TRU mass/HM mass) is to be determined by each participant such that the EOEC reactor -- comprised of one cycle, two cycle, and three cycle burned fuel assemblies -- has an eigenvalue of 1.0 when all rods are withdrawn.

The edits of interest include

- the fresh fuel enrichment (TRU mass)/(Heavy Metal mass)
- the BOEC Safety parameters (defined later)
- the rate of consumption of the TRU feedstock expressed in
  - isotopic mass/MW_e year
  - Ci/MW_e year
  - Toxicity Hazard/MW_e year
  - Watts/MW_e year
- the rate of buildup of the LMR once-through spent fuel waste stream expressed in the same units

The goal of this benchmark is to discover the spread in results among participants and, for the relevant "issues" -- i.e., predictions of rate of reduction of LWR TRU and the buildup rate of LWR TRU and safety parameters -- to sort out their sensitivity to the diversity of basic data and methods in use among the participants.

Specification of the Model

The geometry is unchanged from the previous benchmark and is given in Fig. 4A. This again is the burner core with breeding ratio near 0.5. For all non-fuel region, the composition is given in Table 4.

The isotopic fractions of the TRU from LWR spent fuel processing -- which is to be used in fabricating fresh fuel assemblies is specified in Table 1.

The burn cycle duration and energy extraction are unchanged from the previous benchmark and are given in Table 2. Note that for benchmark purposes the control rods are specified to (unphysically) remain fully withdrawn to the top of the fueled region.

The TRU/HM enrichment of fresh fuel assemblies is to be determined by each participant such that at EOEC the eigenvalue of the core comprised of one-cycle, two-cycle, and three-cycle burned assemblies is 1.0 when all control rods are fully withdrawn to the top of the fueled region.

BOEC Neutron Balance Reporting

1) Narrative Synopsis of the Fuel Management Representation
   e.g., Discrete representation of composition of fresh, once-burned, and twice-burned assemblies vs. spatially smeared representations; fission product representation in partially burned assemblies; etc.

2) Fresh Fuel Enrichment = TRU/HM mass ratio.

BOEC to EOEC Transition and Mass Flow Reporting

1) Burnup Swing = $(k_{EOEC} - k_{BOEC})/k_{BOEC}\, k_{EOEC} \Big|_{\text{constant rod position}}$

2) TRU Breeding Ratio $= \dfrac{\text{EOEC TRU Mass Inventory}}{\text{BOEC TRU Mass Inventory}}$ *****

3) Mass Increments by Heavy Metal Isotope

a) Isotopic mass drawn from the LWR TRU for fabrication of the fresh fuel assemblies for each TRU isotope,

b) Sum over entire model of isotopic mass at BOEC for each TRU isotope

c) Sum over entire model of change in mass, δ(mass), due to burnup for each TRU isotope

d) Sum over TRU isotopes of the above (i.e., 3c) divided by the energy extraction (MW_{th} days) delivered during the burn cycle

4) Safety Parameters Reporting

a) β_{eff} (in units of $\Delta k/k$)

b) Fuel Doppler Coefficient i.e., of Heavy Metal isotopes (with a narrative synopsis of how the calculation is made and what isotopes are accounted for)

c) Sodium Void Worth

- of core (i.e., excluding blankets and reflectors)
- of core plus blanket/reflector regions above core

d) Burnup Swing of the Cycle -- (defined above under BOEC to EOEC transition)

(with rods at constant position)

e) Decay Heat Level for Decay Times of 1 hr, 1 month, 1 year, 10 y, 10^2 y, 10^3 y, 10^4 y.

- Total

*****Note that this definition excludes U^{235}

- Heavy Metal Component
- Fission Product Component

5) Radioactivity and Decay

a) Provide a narrative synopsis of the radioactivity chain representation used for long term out-of-core physics representations

- isotopes treated
- detailed chain representation specifically for the actinides showing all transitions and the values of all decay constants, branching ratios, etc.

b) Describe the numerical solution approach for the equations

c) Describe how the decay heat is computed

6) Curie Increments at the times of 1, 10, 10^2, 10^3, 10^4, 10^5, 10^6 years from the time of BOEC

a) Isotopic mass * λ for the TRU masses drawn from the LWR TRU for fabrication of the fresh fuel assemblies for each TRU isotope (expressed in Curies)

b) Sum over entire model of BOEC mass * λ for each TRU isotope

c) Sum over entire model of δ(mass) * λ by isotope for each TRU isotope

d) Sum over isotopes of the above (i.e., 6c) divided by the energy extraction (MW_{th} days) delivered during the burn cycle

7) Toxicity Hazard Increments at the times of 1, 10, 10^2, 10^3, 10^4, 10^5, 10^6 years from the time of BOEC

a) $\{$ (mass) $*$ (λ) $*$ (Toxicity Index)$\}$ by isotope for each TRU isotope drawn from the LWR spent fuel for fabrication of the fresh fuel assemblies expressed in long term cancer deaths via oral intake)

b) Sum over the entire model of BOEC {(mass) * (λ) * (Toxicity Index)} by isotope for each TRU isotope expressed in long term cancer deaths via oral intake)

c) Sum over entire model of

{δ (mass) * (λ) * (ToxicityIndex)} by isotope for each TRU isotope

d) Sum over isotopes of the above (i.e., 7c) divided by the energy extraction (MW_{th} days) delivered during the burn cycle.

References.

1. R. N. Hill, "LMR Design Concepts for Transuranic Management in Low Sodium Void Worth Cores," Proc. Intl. Conf. on Fast Reactor and Its Fuel Cycle, Kyoto, Japan, October 1991.

2. Y. I. Chang et al., "Core Concepts for Zero-Sodium-Void-Worth Core in Metal Fuelled Fast Reactor," Proc. Intl. Conf. on Fast Reactor and Its Fuel Cycle, Kyoto, Japan, October 1991.

3. Y. I. Chang et al., "Passive Safety Features of Low Sodium Void Worth Metal Fuelled Cores in a Bottom Supported Reactor Vessel," Proc. Intl. Conf. on Fast Reactor and Its Fuel Cycle, Kyoto, Japan, October 1991.

4. R. N. Hill, "Calculational Benchmark Comparisons for a Low Sodium Void Worth Actinide Burner Core Design," Proc. ANS Topical Meeting on advances in Reactor Physics, Charleston, SC (March 1992)

5. B. L. Cohen, "Effects of ICRP Publication 30 and the 1980 BEIR Report on Hazard Assessments of High-Level Waste," *Health Physics*, 2, 133-143 (February 1982).

6. International Commission on Radiological Protection 1979, "Limits for Intakes of Radionuclides by Workers," *ICPR Publication 30*, 3, No. 1-4.