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Nuclear Characteristics of a New(PWR-PHWR) Fuel Cycle

Jae Woong Song and Chang Hyun Chung

Seoul National University
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PWR-PHWR 핵연료 주기의 핵적 특성

송 재 웅 · 정 창 현

서울대학교
(1985. 5. 12 접수)

Abstracts

The fissile content of PWR spent fuel is higher than that of natural uranium which is normal fuel for CANDU type reactor. Investigated are the concepts of PWR spent fuel utilization in CANDU type reactor to diversify uranium resource and partially to solve storage problems of PWR spent fuel being gradually accumulated. Nuclear characteristics of uranium-plutonium mixed oxide fuel loaded in CANDU type reactor are analysed using the WIMS/D computer code. In this study, analyses are solely carried out upon the current CANDU type reactor design without changingany reactivity control devices.

요 약

가압경수로에서 나오는 사용후 핵연료의 fissile 양은 CANDU형 원자로에 쓰는 천연우라늄의 농축도 보다 높다. 따라서 핵연료 활용을 다양화하고 점차 누적되고 있는 가압경수로의 사용후 핵연료의 저장문제를 부분적으로 나마 해결하기 위하여, 가압경수로의 사용후 핵연료를 CANDU 형 원자로에 사용하는 방안을 검토 하였다. 가압경수로에서 나온 사용후 핵연료에서 가공되는 혼합핵연료(Mixed Oxide Fuel)를 CANDU형 원자로에 장전하였을 경우, WIMS/D 코드를 이용하여 핵적특성을 분석하였다 그리고 본 분석에서는 현 CANDU형 원자로의 반응도 조절장치를 변경시키지 않고 혼합핵연료를 CANDU형 원자로에 사용할 수있는 방안만 조사하였다.

1. Introduction

Nuclear power reactors under construction and/or operation are 8 PWRs and 1 CANDU type reactor in Korea. As is shown in Table.1, one nuclear power reactor will be put into operation every year after 1984. Therefore, spent fuel from PWRs may give rise to some probl ems on storage.

Considering at-reactor (AR) storage capacities currently designed or planned,[1] Korea Nuclear Unit (KNU) 1 will lose full core reserve capa city by 1991. Storage capacities of both KNU 1 and KNU 2 will be in short in around 1995 even after transshipment of spent fuel from

Table 1. Nuclear Power Reactors in Korea

KNU*	Site	Capacity MWe)	Reactor Type	Operation	Remarks
KNU#1	Kori	587	PWR	1978, 4	Operation
KNU#2	Kori	650	PWR	1983, 12	Operation
KNU#3	Wolsung	679	CANDU	1983, 4	Operation
KNU#5	Kori	950	PWR	1984, 9	Construction
KNU#6	Kori	950	PWR	1985, 9	Construction
KNU#7	Kyema	950	PWR	1986, 3	Construction
KNU#8	Kyema	950	PWR	1987, 3	Construction
KNU#9	Buku	950	PWR	1988, 3	Construction
KNU#10	Buku	950	PWR	1989, 3	Construction

*Korea Nuclear Unit

Table 2. AR Storage Capacity in Nuclear Power Reactor

KNU	MSC[1] MTHM[4]	Spent Fuel[2] (MTHM)[4]	Storage Limit Year[3]	Site
KNU# 1	253	18	1991	Kori
KNU# 2	414	18	2003	Kori
KNU# 5	324	23.4	1995	Kori
KNU# 6	324	23.4	1996	Kori
KNU# 7	324	23.4	1966	Kyema
KNU# 8	324	23.4	1997	Kyema
KNU# 9	212	23.4	1995	Buku
KNU#10	212	23.4	1996	Buku
KNU# 3	10 Year +1 Core	480[5]	1992	Wolsung

1) Maximum Storage Capacity; Not Considered Full Core Reserve(FCR)
2) Per Year
3) Considered FCR
4) 1 Assembly $\simeq$0.45 MTHM
5) Number of Bundle

KNU 1 to KNU 2 is carried out. The other reactors will also lose their full core reserve capacities in the middle of the 1990's (cf. Table.2.).

Taking into account PWRs and CANDU-PHWR which are on operation or under construction in Korea, PWR-PHWR new fuel cycle may be one of solutions to utilize fissionable materials in PWR spent fuels and conceptionally extend spent fuel storage. The principal modes in this fuel cycle have been studied into two basic categories: reconfiguration and refabrication[2].

In the reconfiguration option, PWR spent fuel rods would be removed from PWR fuel assembly and incorporated directly into a PHWR fuel element. In order to accept PWR type fuel elements in PHWR, this option would require potentially deleterious modifications to either PWR or PHWR designs.

Refabrication option is generally to remove fuel pellets from PWR spent fuel elements and then refabricate into new PHWR bundles. But all operations, because of the inherent radiation hazard, must be performed remotely. The substantial decay heat source in fuel pellets and fuel powders will also give problems since a coolable geometry must be maintained at all times.

Resultantly, since reconfiguration concept requires new reactor design of PWR or PHWR but refabrication concept requires well known reprocessing technology, refabrication seems to be more feasible. Therefore, nuclear characteristics are investigated for utilizing the uranium-plutonium mixed oxide (MOX) fuel refabricated from PWR spent fuel in current CANDU-PHWR system.

II. Computer Code Usage

WIMS/D code[3], a general code for reactor lattice cell calculations on a wide range of reactor systems, is used to perform the lattice calculation for PWR spent fuel in CANDU 600MWe reactor.

The basic library has been complied with 14 fast groups, 13 resonance groups and 42 thermal groups. The 'Spectrox' collision theory[4] procedure gives accurate spectrum computations in the 69 groups of the library for the principal regions of the lattice using a simplified geometric representation of complicated lattice cells. The computed spectra are then used for the condensation of cross-sections to the number of groups selected for solution of the transport equation in detailed geometry.

III. Numerical Models

Ⅲ. 1. Lattice

A CANDU 600 MWe reactor core consists of 380 channels loaded with 4560 fuel bundles, 12 bundles per channel along the axial direction. The fuel channels are arranged on a square lattice pitch of 28.575cm. In order to control nuclear reactions, there are various types of reactivity control devices derived from the top of the reactor structure.

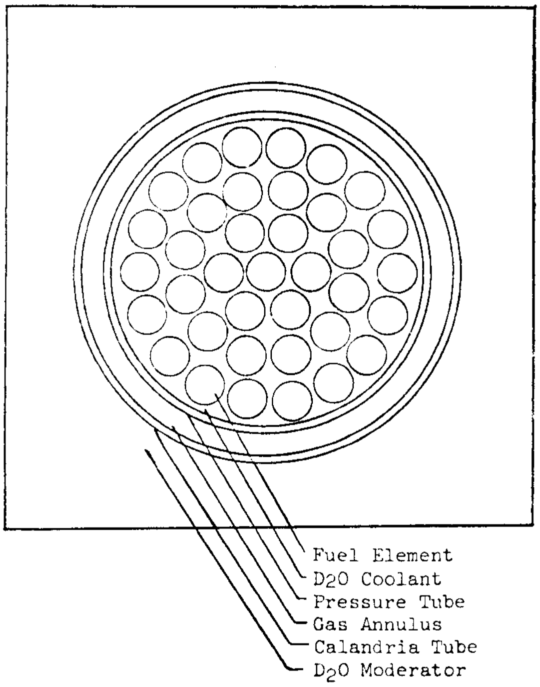

Fig. 1. Calculational Model

As shown in Fig.1, the lattice for the CANDU 600 MWe reactor consists of 37 fuel rods, D_2O coolant and moderator, pressure and calandria tubes. Fuel rods and D_2O coolant are enclosed in zirconium-niobium pressure tubes which form part of the heat transport circuit, and D_2O moderator is seperated by the calandria tube. Gas annulus thus is located between pressure tube ant calandria tube in order to minimize the heat to be transferred from coolant to moderator.

Although in principle it is possible to perform transport theory calculations in all 69 energy groups in WIMS/D code, lattice parameters are computed for 4 energy groups since neutrons are expected to be well thermalized in the lattice.

Table 3. Basic Lattice Model

Zone	Radius(cm)	Volume(cm^3)	Material
1	.60770E+00	.11602E+01	Fuel
2	.65405E+00	.18373E+00	Cladding
3	.81274E+00	.73128E+00	D_2O Coolant
4	.16960E+01	.69611E+01	Fuel
5	.17965E+01	.11024E+01	Cladding
6	.21503E+01	.43876E+01	D_2O Coolant
7	.30092E+01	.13922E+02	Fuel
8	.31237E+01	.22047E+01	Cladding
9	.35427E+01	.87753E+01	D_2O Coolant
10	.43815E+01	.20883E+02	Fuel
11	.45001E+01	.33071E+01	Cladding
12	.49437E+01	.13163E+02	D_2O Coolant
13	.51689E+01	.71536E+01	D_2O Coolant
14	.56032E+01	.14697E+02	Pressure Tube
15	.64478E+01	.31976E+02	Gas Annulus
16	.65875E+01	.57209E+01	Calandria Tube
17	.73152E+01	.31782E+02	D_2O Moderator
18	.79767E+01	.31782E+02	D_2O Moderator
19	.85875E+01	.31782E+02	D_2O Moderator
20	.96912E+01	.63382E+02	D_2O Moderator
21	.10682E+02	.63382E+02	D_2O Moderator
22	.11588E+02	.63382E+02	D_2O Moderator
23	.12872E+02	.98677E+02	D_2O Moderator
24	.14039E+02	.98677E+02	D_2O Moderator
25	.15116E+02	.98677E+02	D_2O Moderator
26	.16122E+02	.98677E+02	D_2O Moderator

Fast (10000~9.118 Kev) and resonance (9118 ~4 ev) regions respectively are condensed to one energy group. And thermal energy region is condensed to two groups (4.0-0.625 ev, 0.625-0.0 ev) because the thermal cut-off energy is comparatively high, i.e. 4ev.

The typical lattice of a CANDU 600 MWe reactor is divided into 26 zones. Each fuel ring is divided into 3 zones (pellet, cladding and coolant) and coolant zone is seperated between pressure tube and outer ring. Pressure tube, gas annulus, and calandria tube are also divided into one zone. Moderator region with leakage effect[2] is divided into ten zones. Tabulated in Table 3 are individual areas and materials for the numerical study.

III.2. Burnups and Reactivities

The data for lattice calculation is taken from the design manual of the CANDU 600 MWe reactor[6], and is summarized in Table 4. When uranium-plutonium mixed oxide (MOX) fuel is loaded into the lattice, changes of lattice parameters and reactivity are investigated and the utilization is finally analysed. Table 5 shows the isotopic weight fraction when 3.2w/o enriched fresh fuel is discharged from PWR at burnup of 33000 MWD/MTU. The fissile content is, therefore, taken to be 1.55w/o enriched equivalent throughout numerical calculations. Reactivity changes of 1.55w/o MOX, 1.2w/o enriched uranium and natural uranium fuels are shown as a function of burnup in Fig. 2. Average discharge burnup of 1.55w/o MOX fuel obtained from Fig. 2 and (A-3) eguation in Appendix is 26000 MWD/MTU, about four times higher than that of natural uranium. And its initial reactivity is 248mk, 5.5times larger than that of natural uranium. But in some stndies[7-8], an nitial enrichment of about 1.2 w/o appears optimal both from the standpoint of fuel utilization and economic performance. The studies also indicated that it would be

Table 4. Lattice Parameters

Fuel Rod	Pellet Diameter, cm	1.2154
	Pellet Density, gm/cc	10.45
	Fuel Temperature, °C	687
	Clad Diameter, cm	1.3081
	Clad Thickness, mm	0.419
	Clad Material	Zr-4
	Fuel Rod No. per Bundle	37
Coolant	Material	D_2O
	Density, gm/cc	0.804055
	Average Temperature, °C	290
Moderator	Material	D_2O
	Density, gm/cc	1.085575
	Temperature, °C	68
Pressure Tube	Material	Zr-2.5%
	Outer Diameter, cm	11.2064
	Thickness, mm	4.343
Calandria Tube	Material	Zr-2
	Outer Diameter, cm	13.175
	Thickness, mm	1.397
Reactor	Lattice Pitch, cm	28.575
	Power, MW/Te	25.43
	Radial Buckling, m^{-2}	0.23
	Axial Buckling, m^{-2}	0.53

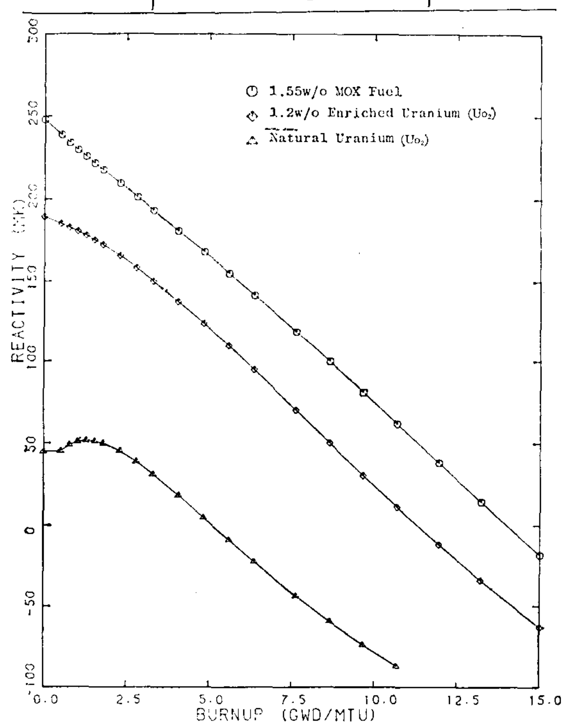

Fig. 2. Reactivity Changes-I

Table 5. Isotopic Weight Fraction of PWR Spent Fuel

Isotope	U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242
Compos. (%)	0.016	0.875	0.427	97.74	0.547	0.218	0.129	0.050

possible to retain the existing CANDU fuel design if the burnup is limited to 20000 MWD/MTU attainable from 1.2w/o fuel. Higher burnups than 20000 MWD/MTU in CANDU reactor would probably require modification of the fuel design to include internal fission gas plenums. Thus the initial reactivity (188mk) with 1.2w/o enriched uranium is considered to be controlled by the regulating and protection systems of the current CANDU reactor design. Taking this into account, some modifications to use 1.55w/o MOX fuel for the CANDU 600MWe reactor core must be introduced. The most ideal case is to reduce the reactivity within the controllable range and to extend discharge burnup to 20000 MWD/MTU, if possible. Therefore, described in the next section are some details about dilution of 1.55w/o MOX fuel, pelletradius changes and mixing of bnrnable poison materials, in order to effectively utilize 1.55w/o MOX fuel.

IV. Results and Discussions

IV. 1. Dilutions

Fig. 3 shows reactivity cha nges of 1.55w/o MOX, 1.2w/o enriched uranium, 1.2w/o MOX and 1.09w/o MOX fuels. Although fissile contents of 1.2w/o enriched uranium and 1.2w/o MOX fuels are the same, their slopes are different from one to another because of the difference in their absorption characteristics. The initial reactivity of 1.2w/o MOX fuel shown in Fig. 3 is higher than that of 1.2w/o enriched uranium fuel because MOX fuel initially includes fissile plutoniums whose absorption and fission cross sections are about 1.3~1.6 times higher compared to ^{235}U cross sections. But average discharge burnup is expected to be opposite when compared with initial reactivity. The problem in this case remains whether higher initial reactivity of 1.2w/o MOX fuel can be controlled by control devices of a current CANDU reactor.

Since the initial reactivity of 1.09w/o MOX fuel is the same as that of 1.2w/o enriched uranium fuel, any MOX fuel with fissile content below 1.09w/o may be used without having any changes or modifications of the current CANDU reactor design. Average discharge burnup of 1.09w/o MOX fuel roughly calculated from Fig. 3 and(A-3) eguation in Appendix is 17800 MWD/MTU, 68 percent of 1.55w/o MOX fuel.

Therefore, MOX fuel diluted to 1.09w/o equivalent can be used in the current CANDU reactor, but the option has a penalty of discharge burnup by about 32 percent compared to the discharge burnup of 1.55w/o MOX fuel.

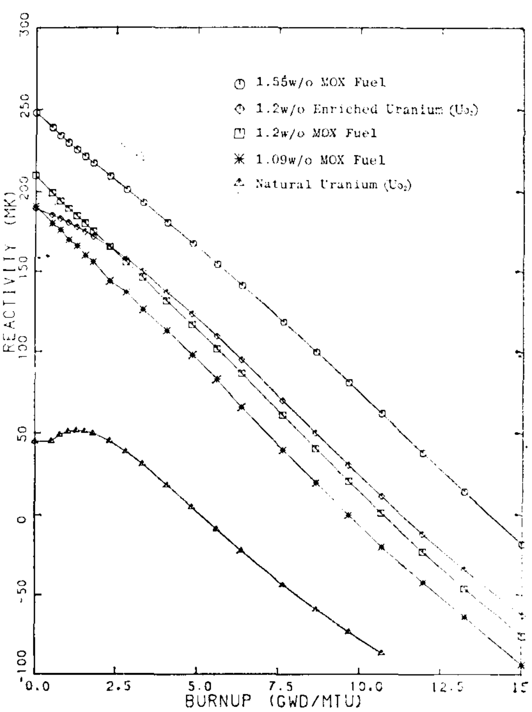

Fig. 3. Reactivity Changes-II

IV. 2. Pellet Radius Changes

In order to reduce the initial reactivity of 1.55w/o MOX fuel within the controllable range using the existing CANDU system, pellet radius change may be one of various options. In case that local linear power in a bundle and the initial reactivity of 1.55w/o MOX fuel have a sensitive effect on changes of pellet radius, the local power distribution may be flattened and also the initial reactivity reduced to the controllable value. Fig. 4 illustrates the relative initial reactivity of 1.55w/o MOX fuel for various pellet radii, and Table 6 shows local power distributions in a bundle at zero burnup which power peak is generally observed. The reference values in Fig. 4 and Table 6 are obtained from the pellet radius of the CANDU 600 MWe reactor. When pellet radius becomes larger, as shown in Fig. 4 and Table 6, the initial reactivity is lowered and the maximum local power is increased. They are caused by undermoderation and self-shielding effects based on the relative reduction of coolant volume. But when pellet radius becomes 1.1 times larger than that of the standard fuel pellet, the initial reactivity

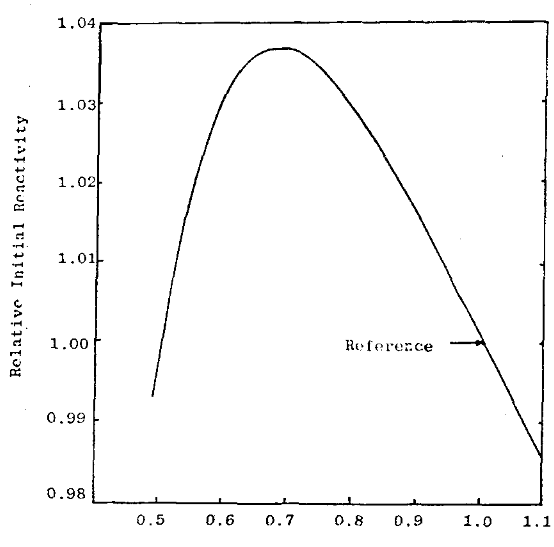

Fig. 4. Relative Initial Reactivities vs. Pellet Radius for 1.55w/o MOX Fuel.

Table 6. Local Power Distribution of 1.55 w/o MOX Fuel

Pellet Radius / Ring	Referece	1.1	0.9	0.7	0.5
Center	0.603	0.532	0.672	0.795	0.897
Inner	0.668	0.609	0.727	0.831	0.916
Intermediate	0.822	0.778	0.860	0.919	0.962
Outer	1.252	1.305	1.203	1.121	1.059

becomes lower than that of the reference values by about 1 percent and in this case neighboring fuel rods in a bundle may be come into contact one another.

On the other hand, the reduction of pellet radius, as shown in Fig. 4 and Table 6, produces opposite effects on the relative initial reactivity and the local power compared with the increase of pellet radius. And when pellet radius becomes 0.7 time smaller than that of the reference, the relative initial reactivity is increased by about 4 percent at the maximum, and the maximum local power is decreased by about 10 percent. But, for further smaller cases, the relative initial reactivities are gradually reduced because of reduction of fuel volume in the cell, and the maximum local power is continuously reduced because of overmoderation at the inner region of bundle and of decreasing selfshielding effects at the outer region of bundle. For example, when pellet radius becomes a half of the reference, the relative initial reactivity and the maximum local power are reduced by about 0.8% and 15% respectively.

Resultantly, changes of pellet radius flatten linear local powers in a bundle but cannot reduce the initial reactivity of 1.55w/o MOX fuel to the controllable value.

IV. 3. Boron Mixed Fuel

The concept using neutron absorbers like boron is investigated in order to lower the initial reactivity of 1.55w/o MOX fuel. The boron mixed fuel in this study is 1.55w/o MOX fuel

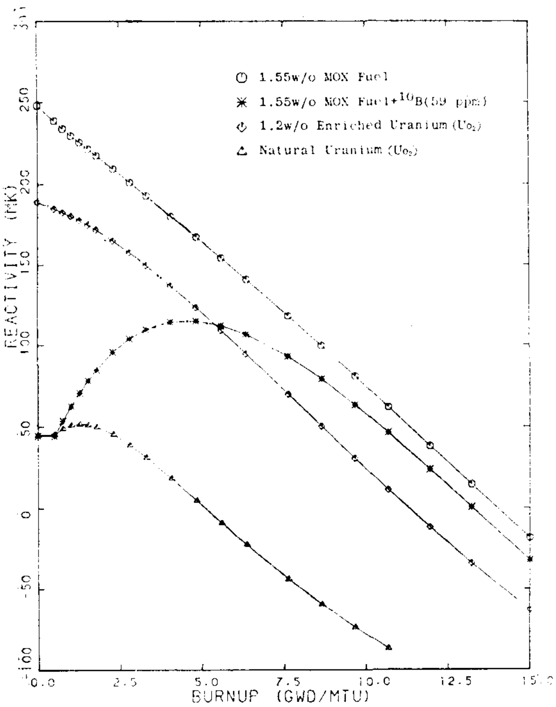

Fig. 5. Reactivity Changes of Boron Mixed Fuel

mixed with 59 ppm* of ^{10}B. The fuel, that is, is shown in the following.

「$(U+Pu)O_2+59$ ppm of ^{10}B→Boron Mixed Fuel」

Fig.5 shows reactivity behaviours of 1.2w/o enriched uranium, 1.55w/o MOX, natural uranium and boron mixed fuels. The others except boron mixed fuel are plotted in Fig.5 to compare with the behaviour of boron mixed fuel. Since ^{10}B absorbs more thermal neutrons than the other fissionable materials in the fuel, the active amount of ^{10}B comes to be promptly reduced with burnup. Therefore boron in the fuel certainly acts to reduce the initial reactivity and the boron mixed fuel approaches the discharge burnup to that of 1.55w/o MOX fuel. The maximum reactivity of boron mixed fuel is about 115mk and the average discharge burnup calculated from Fig.5 and (A-3) eguation in Appendix is about 23300 MWD/MTU.

* ppm: part of ^{10}B per million of $(U+Pu)O_2$ fuel

Although boron mixed fuel is reduced by 10 per cent in average discharge burnup compared with that of 1.55w/o MOX fuel, this option seems to be most effective without modifying the current CANDU reactor control and regulation systems.

V. Conclusions and Remarks

For PWR spent fuel utilization in the current CANDU 600 MWe reactors, some studies on nuclear characteristics have been carried out. The key problem is whether MOX fuel discharged from PWR is directly loaded in the CANDU reactor without system changes, or loaded with fuel composition and pellet radius changes.

Three options and their conclusions analyzed in this study are summarized as follows;

(1) MOX fuels diluted below 1.09w/o equivalent can be used in the current CANDU reactor but maximum average discharge burnup of them is barely about 68% of 1.55w/o MOX fuel.

(2) Changes of pellet radius reduce maximum local power in a bundle, but do not reduce the initial reactivity of 1.55w/o MOX fuel within the controllable range.

(3) Boron mixed fuel has a penalty of average discharge burnup by about 10%, but this option seems to be most effective.

Therefore, boron mixed fuel is most appropriate to utilize MOX fuel in current CANDU 600 reactors, but the difficulty in uniformly mixing boron with 1.55w/o MOX fuel is remained since the boron quantity(^{10}B 59 ppm) is infinitesimal compared with the total fissionable materials[9]. It is however, recommanded that static and dynamic analyses on nuclear physics and mechanics as thermodynamics should be carried out in order to practically use boron mixed fuel.

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Appendix

From the reactivity depletion characteristics of the PHWR cluster obtained with WIMS/D, the continuous maximum irradiation attainable was estimated as follows:

$$\text{Total neutron production} = \int \nu\Sigma_f\phi dt$$

$$\text{Total neutron removal} = \int (\Sigma_a + \text{Leakage})\phi t \qquad \text{(A-1)}$$

$$= \int \frac{1}{K_{\text{eff}}} \cdot \nu\Sigma_f\phi dt$$

Equating these integrals and assuming that $\nu\Sigma_f\phi$ is approximately constant (since power generation is constant in the calculations) we obtain

$$\frac{1}{T}\int_0^T \frac{1}{K_{\text{eff}}} dt = 1 \qquad \text{(A-2)}$$

where T is the time corresponding to the discharge exposure. Therefore, we obtain from (A-2) equation

$$\int_0^T \rho dt = 0 \qquad \text{(A-3)}$$

where $\rho(=\Delta K_{\text{eff}}/K_{\text{eff}})$ is a reactivity.