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TECHNICAL REPORT

Benchmark Problem Suite for Reactor Physics Study of LWR Next Generation Fuels

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This paper proposes a benchmark problem suite for studying the physics of next-generation fuels of light water reactors. The target discharge burnup of the next-generation fuel was set to 70 GWd/t considering the increasing trend in discharge burnup of light water reactor fuels. The UO_2 and MOX fuels are included in the benchmark specifications. The benchmark problem consists of three different geometries: fuel pin cell, PWR fuel assembly and BWR fuel assembly. In the pin cell problem, detailed nuclear characteristics such as burnup dependence of nuclide-wise reactivity were included in the required calculation results to facilitate the study of reactor physics. In the assembly benchmark problems, important parameters for in-core fuel management such as local peaking factors and reactivity coefficients were included in the required results. The benchmark problems provide comprehensive test problems for next-generation light water reactor fuels with extended high burnup. Furthermore, since the pin cell, the PWR assembly and the BWR assembly problems are independent, analyses of the entire benchmark suite is not necessary: *e.g.*, the set of pin cell and PWR fuel assembly problems will be suitable for those in charge of PWR in-core fuel management, and the set of pin cell and BWR fuel assembly problems for those in charge of BWR in-core fuel management.

KEYWORDS: light water cooled reactors, next-generation fuel design, benchmarks, fuel pin cell, PWR type reactors, BWR type reactors, fuel assemblies, UO_2 fuel, MOX fuel, lattice physics code, burnup, reactivity coefficient, uranium dioxide, MOX, nuclear fuels

I. Introduction

The working party on Reactor Physics for LWR Next-Generation Fuels in the Research Committee on Reactor Physics, which is organized by the Japan Atomic Energy Research Institute, was started in 1999 to conduct research associated with the reactor physics of advanced light water reactor fuels.

Through discussions in the working party, it was clarified that the prediction accuracy of nuclear design tools for light water reactor fuels, which will be adopted in the near future, was considered to be satisfactory. Therefore, the working party concluded that the current motivation to design new benchmark problems should be the verification of prediction capability in nuclear design for extended high burnup regions, in which no actual verification data such as critical experiments or core tracking exists. Consequently, the benchmark problems presented in this report do not take into account the current design limitations (or envelopes) of light water reactor fuels, *e.g.* the limitation of ^{235}U enrichment (5 wt%), maximum burnup or mechanical integrity.

The average discharge burnup was assumed to be 70 GWd/t throughout the benchmark problems. The discharge burnup was set based on the current design of high burnup fuels both for PWR and BWR, whose burnup limitations are 55 GWd/t.

The UO_2 and MOX fuels were both adopted in the benchmark problems.

Three different geometries are included in the benchmark problems: the fuel pin cell, the PWR fuel assembly and the BWR fuel assembly. In the simple pin cell geometry, basic nuclear characteristics such as nuclide-wise reactivity were chosen as required calculation results so that necessary information will be available for physical interpretations of calculation results from the benchmark problems. On the other hand, the PWR and the BWR fuel assembly geometry models represent more realistic configurations, and important nuclear characteristics for commercial reactors, *e.g.* local power peaking factor, are included in the required calculation results.

The above three configurations constitute a comprehensive benchmark problem suite for extended high burnup fuels of light water reactors. However, the dependency among the three problems is minimized, *i.e.* each configuration provides an independent problem. Therefore, a complete analysis of these problems is not necessary; ones can choose any single problem or any combination of them for thier analysis.

II. Specification of Fuel Pin Cell Problem

1. UO_2 Fuel Pin

(1) General Description

A UO_2 fuel pin cell is the same fuel cell as the 17×17 type PWR fuel assembly has. The ^{235}U enrichment has been set to ensure mean discharge burnup of up to 70 GWd/t for 21 effective full power months operations using the three batch loading strategy. The adopted ^{235}U enrichment is 6.5 wt%,

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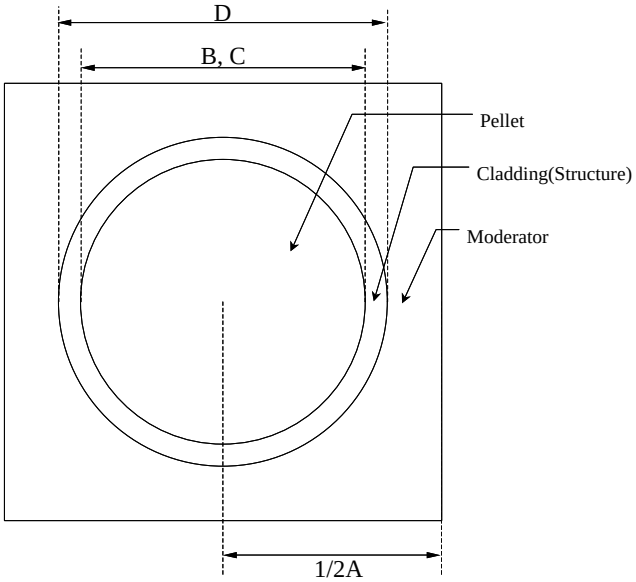
which exceeds the current limitation of light water reactor fuels (5 wt%).

(2) Geometrical Configuration

The geometrical description and the schematic view of the pin cell geometry are given in **Fig. 1**.



Abbreviations	Description	cm
A	Fuel rod pitch	1.265
B	Pellet outer diameter	0.824
C	Cladding inner diameter	0.824
D	Cladding outer diameter	0.952

Remarks:
Thickness of cladding:0.064cm
Gap between pellet and cladding:0.0cm(No gap assumed.)

Note 1: Geometrical configuration is the same as that in Refs. 1) and 2).
Note 2: Square lattice geometry
Note 3: Reflective boundary condition in radial direction and infinite axial length (*i.e.* zero axial bucking) are assumed.

Fig. 1 Schematic view of the fuel pin cell

(3) Isotopic Compositions

Atomic number densities and fuel rod specifications are tabulated in **Table 1** for UO_2 fuel. Those for structural and moderator materials are tabulated in **Table 2**. Isotopic composition of Zr-nat. given in **Table 3** is recommended to be used if Zr-nat. is not available in the cross section library.

(4) Temperature

Temperature in each region is shown in **Table 4**.

(5) Power Density

The power density is 37.0 W/gU (or 179 W/cm, 111.9 W/cm^3).

2. MOX Fuel

(1) General Description

A MOX fuel pin cell is the same fuel cell as the 17×17

Table 1 Atomic number densities and fuel rod specifications of UO_2 fuel

	UO_2 fuel rod
^{235}U enrichment	6.5 wt%
UO_2 density	10.3 g/cc
Atomic number density (#/barn/cm)	
^{235}U	1.5122E-03
^{238}U	2.1477E-02
^{16}O	4.5945E-02

Note 1: UO_2 density is derived by smearing dish and chamfer of fuel pellet and by assuming 95% theoretical density (TD).
Note 2: Number density of ^{234}U is assumed to be zero for simplicity and only those of ^{235}U and ^{238}U are taken into account in this specification. The objective of this benchmark problem is the simulation of next generation fuel whose design is not finalized, hence the detailed treatment of the isotopic composition is not taken into account here.
Note 3: No thermal expansion is considered, *i.e.* the atomic number densities are assumed to be independent to the pellet temperature.

Table 2 Atomic number densities of structural and moderator materials

	Structural material	Moderator material			
		Cold Void fraction: 0%	Hot Void fraction: 0%	Hot Equivalent BWR void fraction: 40%	Hot Equivalent BWR void fraction: 70%
Density (g/cc)	6.53	0.996	0.660	0.549	0.409
Atomic number density (#/barn/cm)					
Zr-nat.	4.3107E-02	0.0	0.0	0.0	0.0
H_2O	0.0	3.3315E-02	2.2074E-02	1.8383E-02	1.3678E-02

Remarks Structural material: cladding, Boron concentration: 0 ppm

Note 1: Isotopic composition of the structural material is assumed to be Zr-nat. rather than Zircaloy for simplicity. From preliminary estimations, difference of pin cell multiplication factors between Zr-nat. and Zircaloy cladding is abcut 0.1%dk/k. Therefore, selection of the structural material has no significant impact on the objective of the benchmark problem.
Note 2: No thermal expansion is assumed, *i.e.* atomic number density of structural material is independent to its temperature.
Note 3: The boron concentration is assumed to be zero since the pin cell benchmark problem is designed to be applicable both for PWR and BWR.
Note 4: Moderator state of void fraction 0% (Hot) simulates the PWR operating condition.
Note 5: Moderator states of equivalent BWR void fraction 40% and 70% (both Hot) simulate assembly average moderator density in BWR operating condition. These are the average values of in-channel moderator, whose void fraction is 40% or 70%, and gap water, water rod and/or water channel whose void fraction is 0%.
Note 6: Grid spacer is neglected for simplicity.
Note 7: Temperatures of hot and cold state are shown in Table 4.

Table 3 Atomic number densities recommended for isotope wise specification of Zr-nat.

Isotope	Atomic number density (#/barn/cm)
^{90}Zr	2.2200E-02
^{91}Zr	4.8280E-03
^{92}Zr	7.3713E-03
^{94}Zr	7.5006E-03
^{96}Zr	1.2070E-03

Note: Isotopic composition of Zr-nat. given in Table 3 is recommended to be used if Zr-nat. is not available in the cross section library.

Table 4 Temperature in each region

Region	Calculation condition	Temperature (K)
Pellet	Doppler	1,800
	Hot	900
	Cold	300
Structural and moderator materials	Doppler	600
	Hot	600
	Cold	300

Note: Burnup calculation is performed under the condition of Hot temperature and 0% void fraction. The temperatures for Cold and Doppler are used to derive reactivity due to variation of temperature.

Table 5 Plutonium composition

Isotopes	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am
Composition (wt%)	2.1	54.5	25.0	9.3	6.4	2.7

Note: The plutonium composition is taken from Refs. 1) and 2).

type PWR fuel assembly has. The Pu content has been set to ensure mean discharge burnup of up to 70 GWd/t for 21 effective full power months operations using three batch loading strategy. The adopted Pu fissile content is 11 wt%, which significantly exceeds the Pu fissile content of current design (∼6 wt%) of MOX fuels for PWR. The isotopic composition of the Pu is taken from Refs. 1) and 2).

Note that the fuel mechanical integrity was not taken into account in the above specifications to clarify the objective of the benchmark problem.

(2) Geometrical Configuration

The geometrical description and the schematic view of the pin cell geometry are given in Fig. 1.

(3) Isotopic Composition

Plutonium composition used in this problem is shown in **Table 5**. Atomic number densities and fuel rod specifications are tabulated in **Table 6** for MOX fuel. Those for structural and moderator materials are tabulated in Table 2.

Isotopic composition of Zr-nat. given in Table 3 is recommended to be used if Zr-nat. is not available in the cross section library.

(4) Temperature

Temperature in each region is shown in Table 4.

Table 6 Atomic number densities and fuel rod specifications of MOX fuel

	MOX fuel rod
MOX density	10.4 g/cc
^{235}U enrichment	0.2 wt%
Put content	17.2 wt%
Puf content	11.0 wt%
Atomic number density (#/barn/cm)	
^{235}U	3.8879E-05
^{238}U	1.9159E-02
^{238}Pu	8.3986E-05
^{239}Pu	2.1706E-03
^{240}Pu	9.9154E-04
^{241}Pu	3.6732E-04
^{242}Pu	2.5174E-04
^{241}Am	1.0664E-04
^{16}O	4.6330E-02

Note: Definition of Puf content is (^{239}Pu+^{241}Pu)/(^{235}U+^{238}U+^{238}Pu +^{239}Pu+^{240}Pu+^{241}Pu+^{242}Pu+^{241}Am).

Table 7 Output format for burnup dependency of infinite multiplication factor

Burnup (GWd/t)	Multiplication factor
0	
0.1	
5	
10	
15	
20	
30	
50	
70	

Note 1: The specified burnup points shown above are those where the results should be edited. They do not specify the burnup steps adopted in a lattice calculation code. Since choice of burnup steps depends on the lattice calculation code, each analyst should be responsible for the selection of burnup step in the calculation code used.

Note 2: To make detail comparison, submission of multiplication factors at all calculation points (burnup steps) is recommended.

(5) Power Density

The power density is 36.6 W/gHM (or 179 W/cm, 111.9 W/cm^3).

3. Conditions for Burnup Calculation

Temperature is the Hot condition shown in Sec. II-1 or II-2. Void fraction is assumed to be 0%. Zero Xenon concentration is assumed at 0 GWd/t and equilibrium Xenon concentration is assumed greater than or equal to 0.1 GWd/t.

4. Required Results

(1) Burnup Dependency of Infinite Multiplication Factor

Output format is shown in **Table 7**. The results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup.

(2) Burnup Dependency of Major Isotopic Composition

Output format is shown in **Table 8**. The following results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup;

- Heavy nuclides: ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu–^{242}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm–^{246}Cm
- Fission products: ^{95}Mo, ^{99}Tc, ^{103}Rh, ^{133}Cs, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{152}Sm, ^{143}Nd, ^{145}Nd, ^{153}Eu, ^{155}Gd

Table 8 Output format for burnup dependency of major isotopic composition

	Burnup (GWd/t)								
	0	0.1	5	10	15	20	30	50	70
^{235}U									
^{236}U									
^{238}U									
^{237}Np									
^{238}Pu									
^{239}Pu									
^{240}Pu									
^{241}Pu									
^{242}Pu									
^{241}Am									
^{242m}Am									
^{243}Am									
^{242}Cm									
^{243}Cm									
^{244}Cm									
^{245}Cm									
^{246}Cm									
^{95}Mo									
^{99}Tc									
^{103}Rh									
^{133}Cs									
^{147}Sm									
^{149}Sm									
^{150}Sm									
^{152}Sm									
^{143}Nd									
^{145}Nd									
^{153}Eu									
^{155}Gd									

Note: Isotopes are selected based on the discussion in Ref. 3).

Results should be given as the volume averaged number density inside a fuel cell.

(3) Burnup Dependency of Neutron Production Cross Sections

Output format is shown in **Table 9**. The following results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup;

- One group macroscopic cross sections summing all nuclides
- One group microscopic cross sections for ^{235}U, ^{238}U, ^{238}Pu-^{242}Pu, ^{241}Am, ^{243}Am, ^{242}Cm, ^{244}Cm.

Results should be given as the average value of fuel cell.

(4) Burnup Dependency of Neutron Absorption Cross Sections

Output format is shown in **Table 10**. The following results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup;

- One group macroscopic cross sections summing all nuclides
- One group microscopic cross sections for ^{235}U, ^{238}U, ^{238}Pu-^{242}Pu, ^{241}Am, ^{243}Am, ^{242}Cm, ^{244}Cm
- One group macroscopic cross sections summing other fissile and fertile nuclides (Other 1)
- One group macroscopic cross sections summing fission products and ^{16}O in the pellet (Other 2)
- One group macroscopic cross sections of Cladding (Zr)
- One group macroscopic cross sections of Moderator (H_2O).

Results should be given as the average value of fuel cell.

(5) Definitions of Quantities in Required Results in Tables 9 and 10

$$\bar{N}(i) = \frac{\int_V d\boldsymbol{r}\, N_i(\boldsymbol{r})}{V} : \text{Average number density of nuclide } i \text{ inside a fuel cell}$$

Table 9 Output format for burnup dependency of neutron production cross section (average of fuel cell)

	Burnup (GWd/t)								
	0	0.1	5	10	15	20	30	50	70
$\nu\Sigma_f$ (Sum of all nuclides)									
$\nu\sigma_f$ (^{235}U)									
$\nu\sigma_f$ (^{238}U)									
$\nu\sigma_f$ (^{238}Pu)									
$\nu\sigma_f$ (^{239}Pu)									
$\nu\sigma_f$ (^{240}Pu)									
$\nu\sigma_f$ (^{241}Pu)									
$\nu\sigma_f$ (^{242}Pu)									
$\nu\sigma_f$ (^{241}Am)									
$\nu\sigma_f$ (^{243}Am)									
$\nu\sigma_f$ (^{242}Cm)									
$\nu\sigma_f$ (^{244}Cm)									

Note: $\nu\Sigma_f$: One group macroscopic cross sections (1/cm)
$\nu\sigma_f$: One group microscopic cross sections (barn)

Table 10 Output format for burnup dependency of neutron absorption cross section (average of fuel cell)

	Burnup (GWd/t)								
	0	0.1	5	10	15	20	30	50	70
Σ_a (Sum of all nuclides)									
σ_a (^{235}U)									
σ_a (^{238}U)									
σ_a (^{238}Pu)									
σ_a (^{239}Pu)									
σ_a (^{240}Pu)									
σ_a (^{241}Pu)									
σ_a (^{242}Pu)									
σ_a (^{241}Am)									
σ_a (^{243}Am)									
σ_a (^{242}Cm)									
σ_a (^{244}Cm)									
Σ_a (Other 1)									
Σ_a (Other 2)									
Σ_a (Zr)									
Σ_a (H_2O)									

Note 1: Σ_a: One group macroscopic cross section (1/cm)
σ_a: One group microscopic cross section (barn)
Note 2: Other 1: Sum of other fissile and fertile nuclides
Other 2: Sum of ^{16}O and fission products

$$\bar{\phi} = \frac{\int dE \int_V d\boldsymbol{r}\phi(\boldsymbol{r},\ E)}{V} : \text{Average one group neutron flux inside a fuel cell}$$

$$\bar{\sigma}_x(i) = \frac{\int dE \int_V d\boldsymbol{r} N_i(\boldsymbol{r})\sigma_{x,i}(\boldsymbol{r},\ E)\phi(\boldsymbol{r},\ E)}{\bar{N}_i\bar{\phi}V} :$$

Average one group microscopic cross section of nuclide i, reaction x inside a fuel cell

$$\bar{\Sigma}_x = \frac{\sum_i \int dE \int_V d\boldsymbol{r} N_i(\boldsymbol{r})\sigma_{x,i}(\boldsymbol{r},\ E)\phi(\boldsymbol{r},\ E)}{\bar{\phi}V} :$$

Average one group macroscopic cross section of reaction x inside a fuel cell.

Note that abbreviations in Tables 9 and 10 omit upper bars of $\bar{\sigma}_x(i)$ and $\bar{\Sigma}_x$.

For further study, breakdown of reactivity difference can be derived from the equation below:

$$\frac{\delta K_\infty}{K_\infty} = \frac{\delta P}{P} \quad \frac{\delta A}{A} = \frac{\delta \nu\Sigma_f}{\nu\Sigma_f} \quad \frac{\delta\Sigma_a}{\Sigma_a}$$

$$= \frac{\sum_i (\delta\bar{N}(i)\overline{\nu\sigma}_f(i) + \bar{N}(i)\delta\overline{\nu\sigma}_f(i))}{\nu\Sigma_f}$$

$$\frac{\sum_i (\delta\bar{N}(i)\bar{\sigma}_a(i) + \bar{N}(i)\delta\bar{\sigma}_a(i))}{\Sigma_a},$$

where P: Total production rate
A: Total absorption rate
δ: Variation of quantities.

(6) Multiplication Factors of Branch Calculations for Void Fraction and/or Temperature Change

Output format is shown in **Table 11**. The results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup. Specified calculation points (*i.e.* branching point) are as follows:

- Cold, 0% void
- Hot, 40, 70% void
- Doppler, 0% void.

Table 11 Output format for multiplication factor of branch calculations for void fraction and/or temperature

	Burnup (GWd/t)								
	0	0.1	5	10	15	20	30	50	70
Hot, 0% void									
Hot, 40% void									
Hot, 70% void									
Doppler, 0% void									
Cold, 0% void									

Note 1: The hot-cold reactivity swing, the void reactivity and the Doppler reactivity can be evaluated by comparing the above results and those of burnup calculations.
Note 2: The multiplication factor of Hot 0% void status in Table 11 is identical to that of the same condition in Table 7.

III. Specification of PWR Fuel Assembly Problem

1. UO_2 Fuel Assembly

(1) General Description

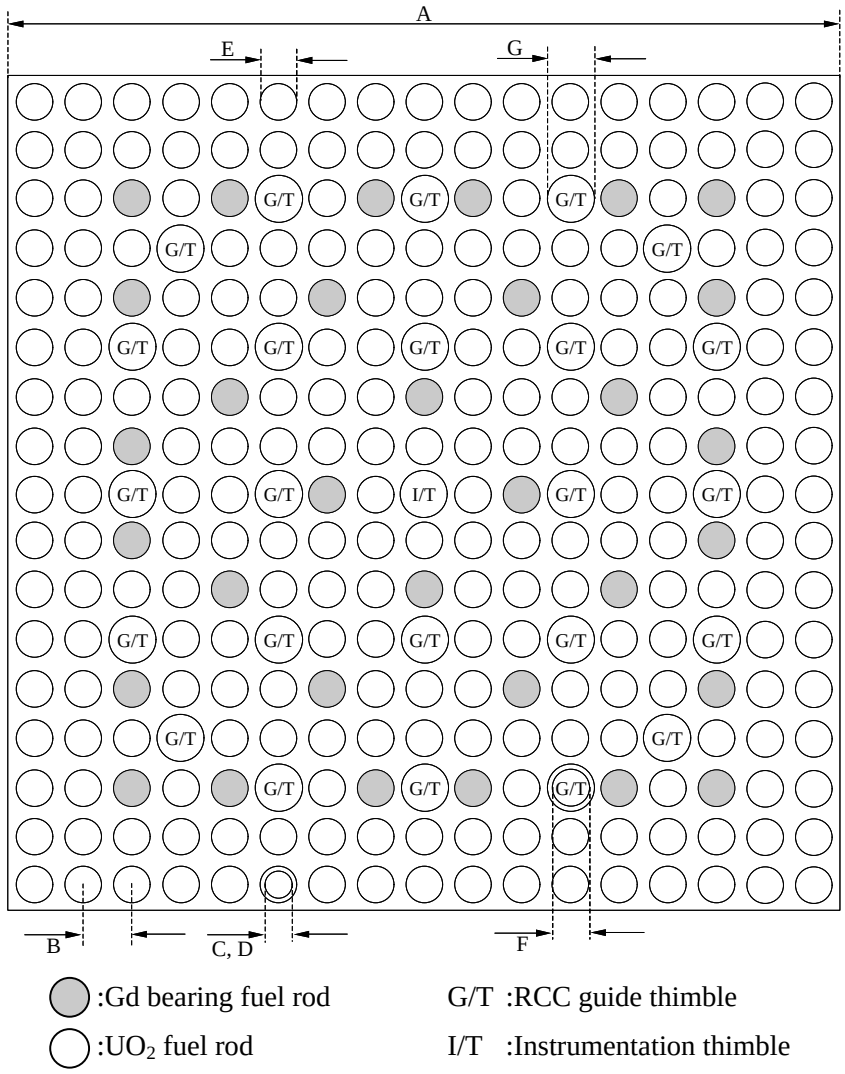
A PWR UO_2 fuel assembly is the same geometrical configuration as a 17×17 type PWR fuel design. The average ^{235}U enrichment is 6.2 wt% assuming 21 effective full power months operations using three batch loading strategy. The assembly is composed of UO_2 and UO_2–Gd_2O_3 (Gd) fuel rods.

(2) Geometrical Configuration

The geometrical description and the configuration of the assembly geometry are given in **Fig. 2**.

(3) Composition

Atomic number densities and fuel rod specifications of



Abbreviations	Description	cm
A	Assembly pitch	21.505
B	Fuel rod pitch	1.265
C	Pellet outer diameter	0.824
D	Cladding inner diameter	0.824
E	Cladding outer diameter	0.952
F	I/T, G/T inner diameter	1.140
G	I/T, G/T outer diameter	1.220

Remarks:
Thickness of cladding:0.064cm
Gap between pellet and cladding:0.0cm(No gap assumed).
Dimensions of UO_2 and Gd fuel rod is identical to that of pin cell problem.
I/T: Instrumentation thimble, G/T: RCC guide thimble
I/T, G/T thickness:0.040cm
Gap between assemblies:0.0cm(No gap assumed.)

Note 1: The above fuel assembly has larger number of Gd bearing fuel rods than that of current assembly design from the viewpoint of moderator temperature coefficient.

Note 2: The reflective (*i.e.* mirror) boundary condition in radial direction and infinite dimension in axial direction (*i.e.* zero axial buckling) are assumed.

Fig. 2 Geometrical configuration of PWR UO_2 assembly

UO_2 fuel and Gd bearing fuel (UO_2–Gd_2O_3) are tabulated in Table 1 and **Table 12**, respectively. Table 2 shows atomic number densities of structural and moderator materials. Note that atomic number density of instrumentation thimble (I/T) and RCC guide thimble (G/T) is the same as that of the cladding material.

Isotopic composition of Zr-nat. given in Table 3 is recommended to be used if Zr-nat. is not available in the cross section library.

(4) Temperature

Temperature in each region is shown in Table 4. Note that same fuel temperature is used both for UO_2 and Gd bearing fuel rod.

(5) Power Density

The power density is 37.5 W/gU (102.2 W/cm^3, 179 W/cm). Note that "[W/cm^3]" corresponds to the assembly average value.

2. MOX Fuel Assembly

(1) General Description

A PWR MOX fuel assembly is the same geometrical configuration as a 17×17 type PWR fuel design. The average Pu fissile content is 11 wt% assuming 21 effective full power months operations using three batch loading strategy. The assembly is composed of low, middle and high Pu content fuel rods.

(2) Geometrical Configuration

The geometrical description and the configuration of the assembly geometry are given in **Fig. 3**.

(3) Composition

Atomic number densities and fuel rod specifications of MOX fuel are tabulated in **Table 13**. Table 2 shows atomic number densities of structural and moderator materials. Note that atomic number densities of I/T and G/T are the same as that of the cladding material.

Isotopic composition of Zr-nat. given in Table 3 is recommended to be used if Zr-nat. is not available in the cross section library.

(4) Temperature

Temperature in each region is shown in Table 4.

(5) Power Density

The power density is 36.6 W/gHM (102.2 W/cm^3, 179 W/cm). Note that "[W/cm^3]" corresponds to assembly average value.

3. Conditions for Burnup Calculation

Temperature is the Hot condition shown in Sec. III-1 or III-2. Void fraction is assumed to be 0%. Zero Xenon concentration is assumed at 0 GWd/t and equilibrium Xenon concentration is assumed greater than or equal to 0.1 GWd/t.

4. Required Results

(1) Burnup Dependency of Multiplication Factor and Local Peaking Factor

Output format is shown in **Table 14**. The local peaking factor is defined as the maximum value of relative fission rate in the assembly. The results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup.

(2) Burnup Dependency of Fission Rate Distribution

Output format is shown in **Table 15**. The average value of fission rate of all fuel rods (*i.e.* 264 fuel rods) should be normalized to 1.0. The results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup.

(3) Multiplication Factor of Branch Calculations for Temperature

Output format is shown in **Table 16**. The results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup. Specified calculation points (*i.e.* branching point) are as follows:

- Cold, 0% void
- Doppler, 0% void.

IV. Specification of BWR Fuel Assembly Problem

1. UO_2 Fuel Assembly

(1) General Description

A BWR fuel assembly is the same geometrical configuration as a modern 9×9 BWR fuel design (STEP3 Type). The fissile contents ensure mean discharge burnups of up to 70 GWd/t for 18 months operation. Five types of the rod enrichments are considered: four types for UO_2 rods and one type for UO_2–Gd_2O_3 rods, and the assembly averaged enrichment is 5.5 wt%.

(2) Geometrical Configuration

The geometrical description and the geometrical configuration along with the rod enrichment distribution is depicted in **Fig. 4**.

(3) Isotopic Compositions

The atomic number densities and fuel rod specifications are tabulated in **Table 17** for UO_2 fuel rods and **Table 18** for Gd fuel rod. ^{234}U and ^{236}U are excluded in the analysis for simplicity.

The atomic number densities for structural and moderator materials are listed in **Table 19**.

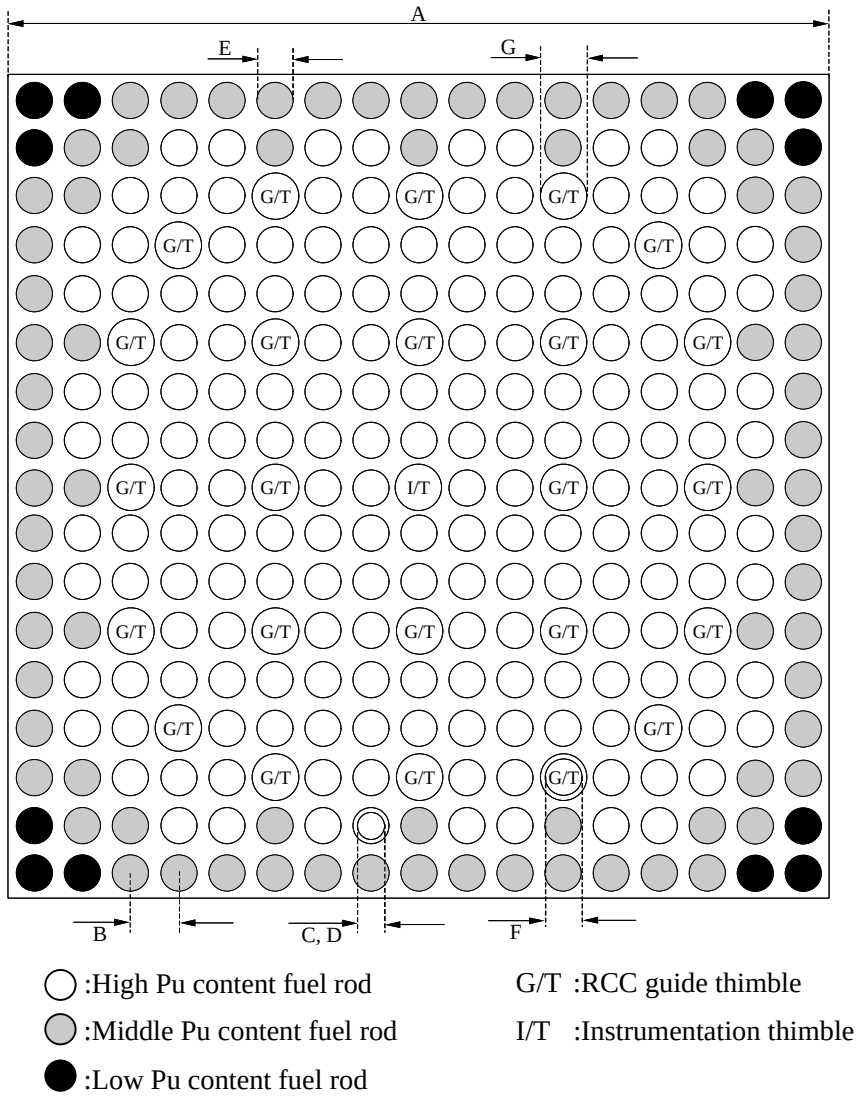
Isotopic composition of Zr-nat. given in Table 3 is recommended to be used if Zr-nat. is not available in the cross section library.

Table 12 Atomic number densities and fuel rod specifications of Gd bearing fuel rod

	Gd bearing fuel rod
Density	10.0 g/cc
^{235}U enrichment	4.0 wt%
Gd_2O_3 concentration	10.0 wt%
Atomic number density (#/barn/cm)	
^{235}U	8.1312E-04
^{238}U	1.9268E-02
^{154}Gd	7.1289E-05
^{155}Gd	4.8938E-04
^{156}Gd	6.8028E-04
^{157}Gd	5.2077E-04
^{158}Gd	8.2650E-04
^{160}Gd	7.2761E-04
^{16}O	4.5130E-02

Note 1: UO_2 density is derived by smearing dish and chamfer of fuel pellet and by assuming 95% theoretical density (TD).

Note 2: Gadolinium-152 is excluded in the analysis due to its negligible contribution to the final results.



Abbreviations	Description	cm
A	Assembly pitch	21.505
B	Fuel rod pitch	1.265
C	Pellet outer diameter	0.824
D	Cladding inner diameter	0.824
E	Cladding outer diameter	0.952
F	I/T, G/T inner diameter	1.140
G	I/T, G/T outer diameter	1.220

Remarks:
Thickness of cladding:0.064cm
Gap between pellet and cladding:0.0cm(No gap assumed).
Dimensions of MOX fuel rod is identical to that of pin cell problem.
I/T: Instrumentation thimble, G/T: RCC guide thimble
I/T, G/T thickness:0.040cm
Gap between assemblies:0.0cm(No gap assumed.)

Note: The reflective (*i.e.* mirror) boundary condition in radial direction and infinite dimension in axial direction (*i.e.* zero axial buckling) are assumed.

Fig. 3 Geometrical configuration of PWR MOX assembly

(4) Temperature

Temperature assigned to each region is listed in Table 4.

(5) Power Density

The power density is 25 W/gU (52.32 W/cm^3).

2. MOX Fuel Assembly

(1) General Description

A BWR MOX fuel assembly is the same geometrical configuration as the ATRIUM-10[4,5] type being a modern 10×10 design with large internal water structure, and the fissile contents ensure mean discharge burnups of up to 70 GWd/t and the cycle length of about 16–18 GWd/t. The BWR MOX assembly is composed of 77 MOX rods, five types of plutonium concentration with depleted Uranium, and 14 UO_2–Gd_2O_3 rods, one type of Gd_2O_3 concentration with enriched Uranium. The assembly averaged Pu concentration is 10.92%Put, 6.97%Puf. The isotopic composition of the Pu is taken from Refs. 1) and 2).

Table 13 Atomic number densities and fuel rod specifications of MOX fuel

	Low Pu content	Middle Pu content	High Pu content
MOX density (g/cc)	10.4	10.4	10.4
^{235}U enrichment (wt%)	0.2	0.2	0.2
Put concentration (wt%)	7.5	14.4	19.1
Puf concentration (wt%)	4.8	9.2	12.2
Atomic number density (#/barn/cm)			
^{235}U	4.3463E-05	4.0212E-05	3.8000E-05
^{238}U	2.1408E-02	1.9812E-02	1.8724E-02
^{238}Pu	3.6652E-05	7.0251E-05	9.3169E-05
^{239}Pu	9.4712E-04	1.8154E-03	2.4075E-03
^{240}Pu	4.3265E-04	8.2927E-04	1.0997E-03
^{241}Pu	1.6026E-04	3.0720E-04	4.0739E-04
^{242}Pu	1.0984E-04	2.1052E-04	2.7920E-04
^{241}Am	4.6536E-05	8.9200E-05	1.1828E-04
^{16}O	4.6358E-02	4.6338E-02	4.6325E-02

Note 1: Plutonium isotopic composition is the same with that used in MOX pin cell problem.
Note 2: MOX density is derived by smearing dish and chamfer of fuel pellet and by assuming 95% theoretical density (TD).

Table 14 Output format for burnup dependency of multiplication factor and local peaking factor

Burnup (GWd/t)	Multiplication factor	Local peaking factor
0		
0.1		
5		
10		
15		
20		
30		
50		
70		

Note 1: The specified burnup points shown above are those where the results should be edited. They do not specify the burnup steps adopted in a lattice calculation code. Since choice of burnup steps depends on the lattice calculation code, each analyst should be responsible for the selection of burnup step in the calculation code used.
Note 2: Definition of local peaking factor is the maximum value of relative fission rate.
Note 3: To enable detail comparison, submission of multiplication factors and local peaking factors at all calculation points (burnup steps) is recommended.

(2) Geometrical Configuration

The geometrical description, the geometrical configuration and the isotopic concentration distribution are depicted in **Fig. 5**.

(3) Isotopic Compositions

The isotopic composition of Pu is tabulated in Table 5. The atomic number densities and fuel rod specifications of MOX rods are tabulated in **Tables 20** and **21** for Gd fuel rod. ^{234}U and ^{236}U are excluded in the analysis for simplicity.

The atomic number densities for structural and moderator materials are listed in Table 19.

Table 15 Output format for fission rate distribution

Burnup (GWd/t)=0 0.1 5 10 15 20 30 50 70

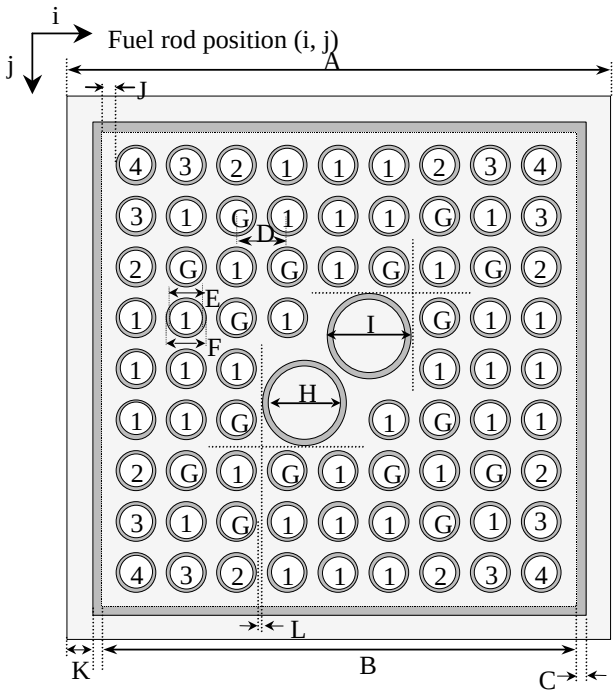
j ↓ \ i →	1	2	3	4	5	6	7	8	9
1									
2									
3									
4									
5									
6									
7									
8									
9									

Note 1: The position "(1, 1)" is the center of assembly and the position "(9, 9)" is the right-bottom (*i.e.* East-South) of assembly.
Note 2: Average value is normalized to be 1.0.

Table 16 Output format for multiplication factor of branch calculation for temperature

	Burnup (GWd/t)								
	0	0.1	5	10	15	20	30	50	70
Hot, 0% void									
Doppler, 0% void									
Cold, 0% void									

Note 1: The hot-cold reactivity swing and the Doppler reactivity can be evaluated by comparing the above results and those of burnup calculations.
Note 2: The multiplication factor of Hot 0% void status in Table 16 should be identical to that of the same condition in Table 14.

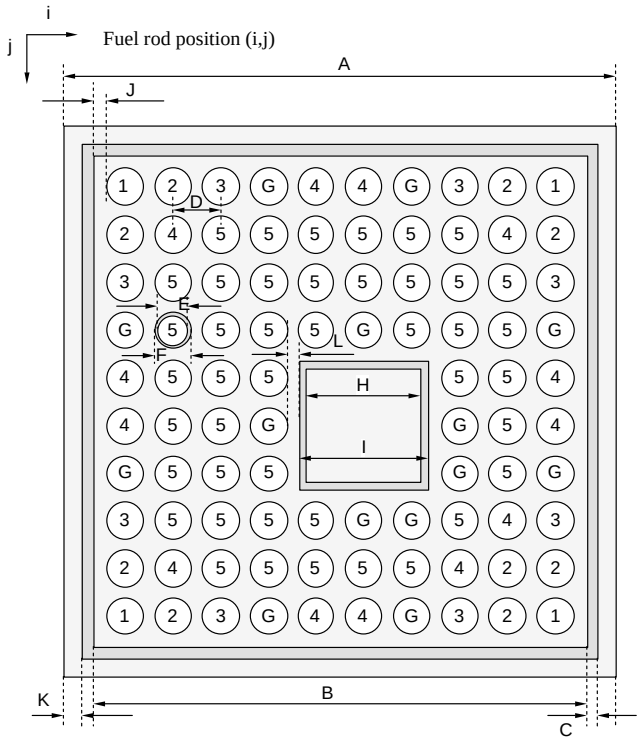


Abbreviations	Description	Dimension (cm)
A	Assembly pitch	15.24
B	C/B inside width	13.40
C	C/B thickness	0.250
D	Fuel rod pitch	1.440
E	Pellet diameter	0.980
F	Cladding outer diameter	1.120
H	W/R inner diameter	2.350
I	W/R outer diameter	2.490
J	Distance between C/B inside and cladding surface	0.380
K	1/2-thickness of water gap	0.670
L	Distance between W/R surface and cladding surface	0.160

Remarks:
C/B : channel box, W/R : water rod
Fuel cladding thickness=0.070 cm, Pellet cladding gap=0.0 (Where the pellet density is scaled in proportion to the smearing of pellet over inside cladding)
W/R tube thickness=0.070 cm, Symmetrical water gaps, Corner inside radius of C/B=0.0

Note 1: Uranium dioxide rods are denoted by integer numbers 1 to 4 corresponding to type-1 to type-4.
Note 2: Gadolinium bearing rods are denoted by G, corresponding to type-G.
Note 3: Control rod center is located at the upper left-hand corner. (Control rod is not considered in this calculation.)
Note 4: Reflective outer boundary condition, no leakage in axial direction.

Fig. 4 Geometrical configuration of BWR UO_2 fuel assembly



Abbreviations	Description	Dimension (cm)
A	Assembly pitch	15.24
B	C/B inside width	13.40
C	C/B thickness	0.250
D	Fuel rod pitch	1.295
E	Pellet diameter	0.884
F	Cladding outer diameter	1.005
H	W/C inner width	3.355
I	W/C outer width	3.500
J	Distance between C/B inside and cladding surface	0.370
K	1/2-thickness of water gap	0.670
L	Distance between W/C surface and cladding surface	0.3375

Remarks:
C/B: channel box, W/C: Water channel
Fuel cladding thickness: 0.0605 cm, Pellet-cladding gap =0.0 (Where the pellet density is scaled in proportion to the smearing of pellet over inside cladding)
W/C thickness: 0.0725 cm, Symmetrical water gaps, Corner inside radius of C/B =0.0

Note 1: MOX rods are denoted by integer numbers 1 to 5 corresponding to type-1 to type-5.
Note 2: Gadolinium bearing rods are denoted by G, corresponding to type-G.
Note 3: Control rod center is located at the upper left-hand corner.
Note 4: Reflective outer boundary condition, no leakage in axial direction.

Fig. 5 Geometrical configuration of BWR MOX fuel assembly

Table 17 Atomic number densities and fuel rod specifications for UO_2 fuels

	Type-1	Type-2	Type-3	Type-4
Number of rods	38	8	8	4
UO_2 density (g/cc)	10.1	10.1	10.1	10.1
^{235}U enrichment (wt%)	6.3	5.0	4.0	3.0
Atomic number density (#/barn/cm)				
^{235}U	1.4322E-03	1.1367E-03	9.0936E-04	6.8203E-04
^{238}U	2.1032E-02	2.1324E-02	2.1549E-02	2.1774E-02
^{16}O	4.4928E-02	4.4921E-02	4.4916E-02	4.4912E-02

Isotopic composition of Zr-nat. given in Table 3 is recommended to be used if Zr-nat. is not available in the cross section library.

Table 18 Atomic number densities and fuel rod specifications for Gd bearing fuels

	Type-G
Number of rods	16
Density	9.8 g/cc
^{235}U enrichment	5.0 wt%
Gd_2O_3 concentration	6.0 wt%
Atomic number density (#/barn/cm)	
^{235}U	1.0389E-03
^{238}U	1.9490E-02
^{154}Gd	4.1864E-05
^{155}Gd	2.8739E-04
^{156}Gd	3.9950E-04
^{157}Gd	3.0602E-04
^{158}Gd	4.8536E-04
^{160}Gd	4.3093E-04
^{16}O	4.3985E-02

Note: Gadolinium-152 is excluded in the analysis due to its negligible contribution to the final results.

(4) Temperature

Temperature assigned to each region is listed in Table 4.

(5) Power Density

The power density is 25 W/gHM (52.61 W/cm^3).

3. Conditions for Burnup Calculation

Temperature is Hot condition, shown in Sec. IV-1 or IV-2. In-channel Void fraction is 40%. Void fractions of Gap water, inside of the Water rods and Water channel are 0%. Power density is shown in Sec. IV-1 or IV-2. Zero Xenon concentration is assumed at 0 GWd/t and equilibrium Xenon concentration is assumed greater than or equal to 0.1 GWd/t.

4. Required Results

(1) Infinite Multiplication Factor and Local Peaking Factor

Output format is shown in **Table 22**. The local peaking factor is defined as the maximum value of relative fission rate in the assembly. The results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup.

(2) Burnup Dependency of Fission Rate Distribution

Output format is shown in **Table 23**. The average value of fission rate of all fuel rods should be normalized to 1.0. The results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup.

Table 19 Atomic number densities for structural and moderator materials

	Structural material	Moderator material			
		Cold 0% Void	Hot 0% Void	Hot 40% Void	Hot 70% Void
Density (g/cc)	6.53	0.996	0.737	0.457	0.247
Atomic number density (#/barn/cm)					
Zr-nat.	4.3107E-02	0.0	0.0	0.0	0.0
H_2O	0.0	3.3315E-02	2.4658E-02	1.5294E-02	8.2712E-03

Note 1: Cladding, channel box, and water rod are treated as the structural material.
Note 2: Thermal expansion effect of the structural materials is neglected.
Note 3: Isotopic composition of the structural material is assumed to be Zr-nat. rather than Zircaloy, for simplicity.
Note 4: Void fraction points listed here are chosen for the typical values of the in-channel axial void distribution in hot BWR operating conditions.

Table 20 Atomic number densities and fuel rod specifications for MOX rods

	Type-1	Type-2	Type-3	Type-4	Type-5
Number of rods	4	9	8	13	43
Density (g/cc)	10.0	10.0	10.0	10.0	10.0
^{235}U enrichment (wt%)	0.2	0.2	0.2	0.2	0.2
Pu-t concentration (wt%)	4.0	6.0	10.0	12.0	16.0
Pu-f concentration (wt%)	2.6	3.8	6.4	7.7	10.2
Atomic number density (#/barn/cm)					
^{235}U	4.3208E-05	4.2347E-05	4.0621E-05	3.9755E-05	3.8019E-05
^{238}U	2.1288E-02	2.0864E-02	2.0014E-02	1.9587E-02	1.8732E-02
^{238}Pu	1.8680E-05	2.8046E-05	4.6831E-05	5.6249E-05	7.5139E-05
^{239}Pu	4.8276E-04	7.2482E-04	1.2103E-03	1.4537E-03	1.9419E-03
^{240}Pu	2.2053E-04	3.3110E-04	5.5286E-04	6.6405E-04	8.8705E-04
^{241}Pu	8.1694E-05	1.2266E-04	2.0481E-04	2.4600E-04	3.2861E-04
^{242}Pu	5.5987E-05	8.4059E-05	1.4036E-04	1.6859E-04	2.2520E-04
^{241}Am	2.3718E-05	3.5610E-05	5.9460E-05	7.1419E-05	9.5403E-05
^{16}O	4.4430E-02	4.4466E-02	4.4538E-02	4.4574E-02	4.4646E-02

(3) Multiplication Factor of Branch Calculations for Void Fraction and/or Temperature

Output format is shown in **Table 24**. The following results should be edited at 0, 0.1, 5, 10, 15, 20, 30, 50 and 70 GWd/t burnup;

- The instantaneous void change from Hot 40% void to Hot 0% void and from Hot 40% void to Hot 70% void
- The instantaneous temperature change from Hot 40% void to Cold 0% void and from Hot 40% void to Doppler 40% void.

Void fractions of gap water, inside of water rods and water channel are kept 0%.

Table 21 Atomic number densities and fuel rod specifications for Gd bearing rods

	Type-G
Number of rods	14
Density	9.8 g/cc
^{235}U enrichment	4.0 wt%
Gd_2O_3 content	3.5 wt%
Atomic number density (#/barn/cm)	
^{235}U	8.5735E-04
^{238}U	2.0316E-02
^{154}Gd	2.4930E-05
^{155}Gd	1.6925E-04
^{156}Gd	2.3409E-04
^{157}Gd	1.7897E-04
^{158}Gd	2.8407E-04
^{160}Gd	2.5228E-04
^{16}O	4.4063E-02

Note: Gadolinium-152 is excluded in the analysis due to its negligible contribution to the final results.

Table 22 Output format for burnup dependency of multiplication factor and local peaking factor

Burnup (GWd/t)	Multiplication factor	Local peaking factor
0		
0.1		
5		
10		
15		
20		
30		
50		
70		

Note 1: The specified burnup points shown above are those where the results should be edited. They do not specify the burnup steps adopted in a lattice calculation code. Since choice of burnup steps depends on the lattice calculation code, each analyst should be responsible for the selection of burnup step in the calculation code used.

Note 2: Definition of local peaking factor is the maximum value of relative fission rate.

Note 3: To enable detail comparison, submission of multiplication factors and local peaking factors at all calculation points (burnup steps) is recommended.

Table 23 Output format for relative fission rate distribution

UO_2 assembly

Exposure (GWd/t)=0 0.1 5 10 15 20 30 50 70

j ↓ \ i →	1	2	3	4	5	6	7	8	9
1									
2									
3									
4					W/R				
5									
6									
7									
8									
9									

MOX assembly

Exposure (GWd/t)=0 0.1 5 10 15 20 30 50 70

j ↓ \ i →	1	2	3	4	5	6	7	8	9	10
1										
2										
3										
4										
5					W/C					
6										
7										
8										
9										
10										

Note: Average value is normalized to be 1.0.

Table 24 Output format of the multiplication factor for instantaneous void and temperature change

Calculation condition	Exposure (GWd/t)								
	0	0.1	5	10	15	20	30	50	70
Hot, 0% Void									
Hot, 40% Void									
Hot, 70% Void									
Doppler, 40% Void									
Cold, 0% Void									

Note: The results for Hot, 40% void are identical to the ones in Table 22.

V. Summary

A benchmark problem suite for next-generation fuels of light water reactors for extended high burnup (approx. 70 GWd/t) was proposed. The benchmark suite consists of the fuel pin cell problem, the PWR and the BWR fuel assembly problems, and the UO_2 and MOX fuels are modeled on each configuration. The benchmark suite provides consistent and comprehensive tests for next-generation high burnup fuels of light water reactors, for which no actual measurements such as critical experiment or core tracking data exist. Therefore, the suite will be useful for further development of lattice calculation codes.

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