OECD/NEA
Burnup Credit Criticality Benchmark
Phase IIIB

Nuclide Composition and Neutron Multiplication Factor of BWR Spent Fuel Assembly

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1 Introduction

We have studied an effect of a burnup profile on the criticality calculation through benchmarks Phase II. In these benchmarks, the type of fuels has been limited only to that for PWRs.

Considering the situation that BWRs are extensively used in some major countries, it is also important to evaluate an effect of the burnup profile of BWR fuels. It is, however, cumbersome to calculate the burnup profile of BWR on account of presence of void and inserted control rods, complicated design of the fuel and usage of gadolinium(Gd).

Special difficulties in evaluating burnup profile of BWR fuels come from the last one, because Gd is intensively used for suppressing the initial excess reactivity. Burnup of Gd results in an increase in reactivity during burnup of fuel for up to about 10 GWd/tHM.

This has given a good reason for Phase III exercises that treat burnup credit benchmarks for criticality calculations of burnup BWR fuels. Phase IIIA has intended to evaluate the neutron multiplication factor considering an axial burnup profile; Phase IIIB intends to demonstrate the evaluation tools of the nuclide composition are appropriate.

2 Geometry Specification

Figure 1 shows a horizontal cross section of a BWR fuel assembly of a typical 8 by 8 type. A large water rod is located in the center of the fuel assembly. The dimensions of the fuel assembly are shown in **Table 1**.

Table 1 Dimensions of Assembly, Fuel and Water Rods [cm]

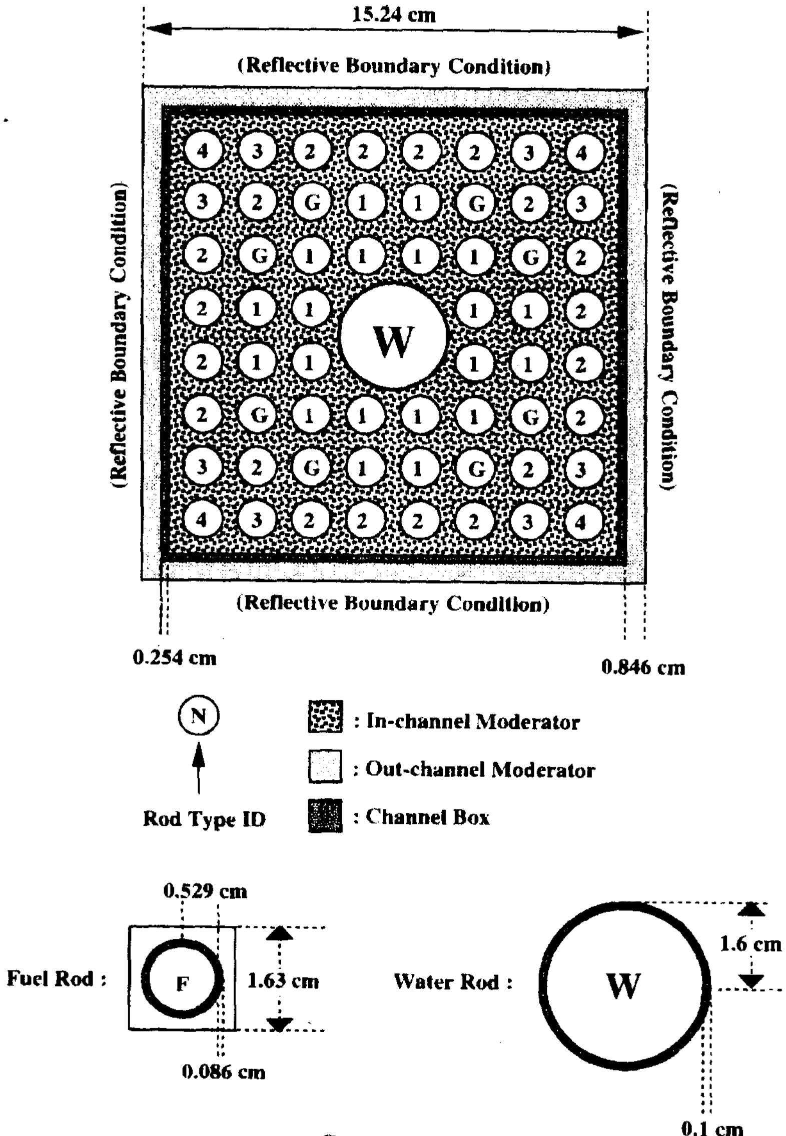
Assembly Pitch	15.24
Thickness of Channel Box	0.254
1/2 Thickness of Water Gap	0.846
Cell Pitch	1.63
Outer Radius of Fuel Rod	0.615
Inner Radius of Fuel Rod	0.529
Cladding Thickness of Fuel Rod	0.086
Outer Radius of Water Rod	1.6
Inner Radius of Water Rod	1.5
Cladding Thickness of Water Rod	0.1

3 Material Specification

The design of the fuel assembly of BWR is complicated compared with PWR assembly. As shown in **Figure 1**, it consists of five kinds of fuel rods, 4.9 , 3.6 , 3.0 and 2.3 wt % U-235-enriched UO_2 rods without Gd and 3.0 wt% U-235 - enriched UO_2 rods with 4.5 wt% Gd, and a water rod.

Table 2 shows nuclide densities of each fresh rod. In this table, the number densities shown are the values averaged over the entire inner rod region (pellet + gap). These of cladding and moderator are shown in **Tables 3** and **4**, respectively.

Fuel Rod Configuration



Geometry Data

Assembly Pitch	= 15.24 cm
Cell Pitch	= 1.63 cm
OuterRadius of Fuel Rod	= 0.615 cm
Inner Radius of Fuel Rod	= 0.529 cm
Cladding Thickness of Fuel Rod	= 0.086 cm
Outer Radius of Water Rod	= 1.6 cm
Inner Radius of Water Rod	= 1.5 cm
Cladding Thickness of Water Rod	= 0.1 cm
Channel Thickness	= 0.254 cm
1/2-Thickness of Water Gap	= 0.846 cm

Figure 1 Fuel Assembly of BWR

Table 2 Initial Isotopic Composition of Fuel and Water Rods(Temperature: Fuel rods at 900 K, water rod at 559 K)

Rod Type ID	Enrichment	Gadolinia	Isotope	Number Density($10^{24}/cm^3$)
1	4.9 wt%		U-234	1.0443E-5
			U-235	1.1284E-3
			U-236	6.9317E-6
			U-238	2.1606E-2
			O-16	4.5504E-2
2	3.6wt%		U-234	7.5720E-6
			U-235	8.2904E-4
			U-236	5.1701E-6
			U-238	2.1907E-2
			O-16	4.5497E-2
3	3.0 wt%		U-234	6.2468E-6
			U-235	6.9087E-4
			U-236	4.3570E-6
			U-238	2.2046E-2
			O-16	4.5494E-2
4	2.3 wt%		U-234	4.7008E-6
			U-235	5.2968E-4
			U-236	3.4083E-6
			U-238	2.2208E-2
			O-16	4.5491E-2
G	3.0 wt%	4.5 wt%	U-234	5.8824E-6
			U-235	6.5057E-4
			U-236	4.1028E-6
			U-238	2.0759E-2
			O-16	4.5095E-2
			Gd-154	3.2253E-5
			Gd-155	2.2141E-4
			Gd-156	3.0778E-4
			Gd-157	2.3576E-4
			Gd-158	3.7393E-4
			Gd-160	3.3200E-4
W			H-1	4.9316E-2
			O-16	2.4658E-2

Table 3 Cladding and Channel Box - Zircalloy-4(at 559K)

Isotope	Number Density ($10^{24}/cm^3$)
Cr	7.5891E-5
Fe	1.4838E-4
Zr	4.2982E-2

Table 4 Moderator - Water (at 559K)

Region	Void Fraction(%)	Isotope	Number Density ($10^{24}/cm^3$)
In-channel	0	H-1	4.9316E-2
		O-16	2.4658E-2
	40	H-1	3.0588E-2
		O-16	1.5294E-2
	70	H-1	1.6542E-2
		O-16	8.2712E-3
Out-channel	0	H-1	4.9316E-2
		O-16	2.4658E-2

4 Parameters of the Calculation

4.1 Case Names and Burnup Parameters

The nuclide composition and k_{inf} are requested to be evaluated for three void fraction. Specific power is 25.6 MW/tHM for all cases, fuel burnup is 40 GWd/tHM, and the cooling time after burnup is 0 and 5 years. The relation of the case name and burnup parameters is shown in **Table 5**. Note that the temperatures of fuel and other components of the assembly are assumed to remain in the hot condition even throughout the cooling period.

Table 5 Case Name

Case Name	Void Fraction [%]	Burnup* [GWd/tHM]	Specific Power [MW/tHM]	Cooling Time [year]
1	0	40	25.6	0, 5
2	40	40	25.6	0, 5
3	70	40	25.6	0, 5

* Assembly Value

4.2 Benchmarked Nuclides

Nuclides that should be benchmarked are 12 actinides and 20 FPs, as shown in **Table 6**.

Table 6 Benchmarked Nuclides

Actinide	U-234,235,236,238 Pu-238,239,240,241,242 Am-241,Am-243 Np-237
FP	Mo-95,Tc-99,Ru-101,Rh-103,Ag-109 Cs-133 Sm-147,149,150,151,152 Nd-143,Nd-145 Eu-153,Eu-155 Gd-155,156,157,158 Xe-131

5 Requested Data

For the 8 by 8 assembly, participants are requested to evaluate the following quantities.

1. Number densities of nuclides specified in **Table 6**, which are averaged over all fuel rods, in the unit of $10^{24}/\mathrm{cm}^3$
2. Burnup of each pin in the unit of GWd/tHM.
3. Neutron multiplication factors for the burnup of 0, 0.2 , 10 , 20, 30, 40 GWd/tHM.
4. Maximum neutron multiplication factor and the corresponding burnup, which should be evaluated by each participant.
5. Burnup of each fuel pin shown in **Figure 2**

These set of values are requested for 3 cases shown in **Table 5**.

6 Results and Media

Please forward the results ONLY via electrical mail to the following address, as floppy disks or paper writing materials may make some problems to treat.

nea-bc-phase3b@cyclone.tokai.jaeri.go.jp

E-mails to this address are forwarded to the coordinators. A recommended format of e-mailed data from participants is shown in **Table 7**. Line numbers from 6 through 41 should be repeated for cases 2 and 3. The separator of numerical data must be ',' without blank. If you have any missing results, you are requested to indicate them with the tag "NODATA" and to keep this format for our convenience.

Following those data, please describe your analysis environment. The description should include :

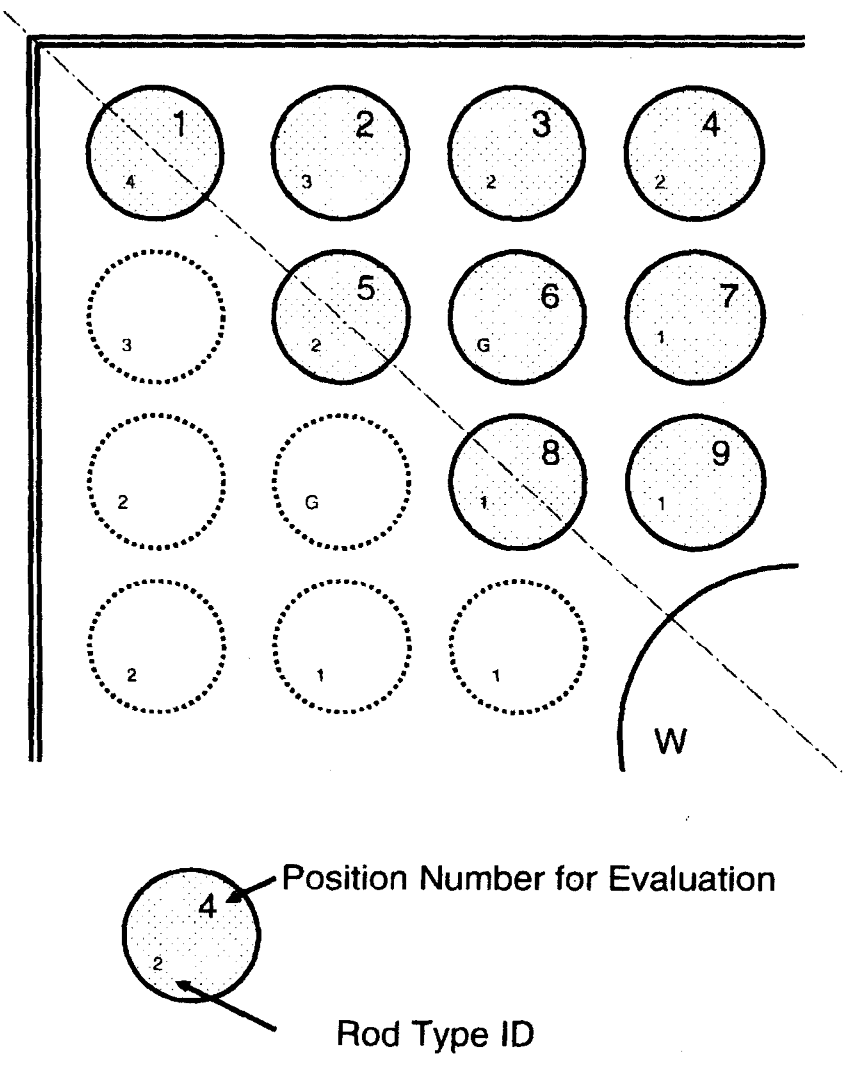


Figure 2 Position Number of Fuel Rod for Evaluation. Rod Type ID is shown in Figure 1

1. Institute and country
2. Participants
3. Neutron data library
4. Neutron data processing code or method
5. Number of neutron energy group
6. Description of your code system
7. Geometry modelings
8. Omitted nuclides, if any
9. Employed convergence limit or statistical errors for eigenvalue calculations
10. Other related information
11. References to your code system or library, if any

7 Schedule

End of March 1997	Deadline for participants to provide their results
End of May 1997	Deadline for coordinators to compile results into tables / figures

Table 7 Format of e-mailed Data

line number	data
1	Date
2	Institute
3	Contact person
4	E-mail address
5	Telefax number
6	Case name tag "CASE-1"
7	Nuclide density of U-234 for cooling time 0 and 5 years
8	Nuclide density of U-235 for cooling time 0 and 5 years
9	Nuclide density of U-236 for cooling time 0 and 5 years
10	Nuclide density of U-238 for cooling time 0 and 5 years
11	Nuclide density of Pu-238 for cooling time 0 and 5 years
12	Nuclide density of Pu-239 for cooling time 0 and 5 years
13	Nuclide density of Pu-240 for cooling time 0 and 5 years
14	Nuclide density of Pu-241 for cooling time 0 and 5 years
15	Nuclide density of Pu-242 for cooling time 0 and 5 years
16	Nuclide density of Am-241 for cooling time 0 and 5 years
17	Nuclide density of Am-243 for cooling time 0 and 5 years
18	Nuclide density of Np-237 for cooling time 0 and 5 years
19	Nuclide density of Mo-95 for cooling time 0 and 5 years
20	Nuclide density of Tc-99 for cooling time 0 and 5 years
21	Nuclide density of Ru-101 for cooling time 0 and 5 years
22	Nuclide density of Rh-103 for cooling time 0 and 5 years
23	Nuclide density of Ag-109 for cooling time 0 and 5 years
24	Nuclide density of Cs-133 for cooling time 0 and 5 years
25	Nuclide density of Sm-147 for cooling time 0 and 5 years
26	Nuclide density of Sm-149 for cooling time 0 and 5 years
27	Nuclide density of Sm-150 for cooling time 0 and 5 years
28	Nuclide density of Sm-151 for cooling time 0 and 5 years
29	Nuclide density of Sm-152 for cooling time 0 and 5 years
30	Nuclide density of Nd-143 for cooling time 0 and 5 years
31	Nuclide density of Nd-145 for cooling time 0 and 5 years
32	Nuclide density of Eu-153 for cooling time 0 and 5 years
33	Nuclide density of Eu-155 for cooling time 0 and 5 years
34	Nuclide density of Gd-155 for cooling time 0 and 5 years
35	Nuclide density of Gd-156 for cooling time 0 and 5 years
36	Nuclide density of Gd-157 for cooling time 0 and 5 years
37	Nuclide density of Gd-158 for cooling time 0 and 5 years
38	Nuclide density of Xe-131 for cooling time 0 and 5 years
39	k_{inf} for each burnup(0,0.2,10,20,30,40 [GWd/tHM])
40	Peak k_{inf} and corresponding burnup [GWd/tHM]
41	Burnup [GWd/tHM] of each pin position 1,2,....,8,9

Table 7 (Cont'd)

line number	data
42	Case name tag "CASE-2"
43 - 77	As the same items as 7-41
78	Case name tag "CASE-3"
79 - 113	As the same items as 7-41
114 -	Analysis environment (See **6 Results and Media**)