CALCULATION METHODOLOGY for REACTOR VESSEL NEUTRON FLUX and FLUENCE

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Abstract

This report contains the methodology used by Mitsubishi Heavy Industries (MHI) to determine the typical fast neutron fluence in the reactor vessel, the uncertainty in the evaluation, and the results of fast neutron fluence for the US-APWR.

The methodology of neutron fluence evaluation for the US-APWR reactor vessel is based on the guidance provided in Regulatory Guide 1.190,“Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence”.

Table of Contents

List of Tables ii

List of Figures iii

List of Acronyms iv

1.0 INTRODUCTION 1

2.0 NEUTRON TRANSPORT CALCULATION 2

2.1 NEUTRON TRANSPORT CALCULATION METHODS 2

2.2 CROSS SECTION LIBRARY 3

2.3 GEOMETRY MODELING 3

2.4 CORE NEUTRON SOURCE AND FISSION SPECTRA 5

2.5 OTHER CONDITIONS 6

2.6 UNCERTAINTY AND BIAS 7

2.7 RESULTS OF NEUTRON TRANSPORT CALCULATIONS 7

2.7.1 HORIZONTAL CROSS SECTION 7

2.7.2 VERTICAL CROSS SECTION 8

3.0 REFERENCES 9

APPENDIX A STUDY OF UNCERTAINTY AND BIAS A-1

List of Tables

Table 1 Energy Group Structure of 47 Neutron Groups (Energy Group Structure of BUGLE96) 10

Table 2 Selected general data and dimensions of US-APWR reactor 11

Table 3 Number Density of Each Structure Component 12

Table 4 Fission Spectra of 47 Neutron Groups 15

Table 5 Core Neutron Source and Fission Spectra 16

Table 6 Neutron Flux Inside RV 17

Table 7 Fast Neutron Fluence Inside RV and on Welded Lines 17

Table A- 1 Analysis input condition for DORT calculations A-9

Table A- 2 Comparison to Measurement and Calculational Benchmark A-10

Table A- 3 Input parameters evaluated in the analytical sensitivity studies A-11

Table A- 4 Evaluated analytic uncertainties(E>1MeV) A-12

List of Figures

Figure 1 Horizontal cross section of US-APWR reactor 18

Figure 2 Schematic sketch of the axial geometry 19

Figure 3 Core Center Cross Section ((r, θ) Geometry) 20

Figure 4 Core Center Cross Section ((x, y) Geometry) 21

Figure 5 Core Vertical Cross Section ((r, z) Geometry) 22

Figure 6 Cycle-Averaged Relative Power Distribution for Fuel Assemblies at Core Horizontal Cross Section 23

Figure 7 Cycle-Averaged Relative Power Distribution for Fuel Assembly, in Core Vertical Direction 24

Figure 8 Distribution of Neutron Flux in Azimuthal Direction at Inner Surface in RV ((r, θ) Geometry) 25

Figure 9 Distribution of Neutron Flux in Azimuthal Direction at RV 1/4 Thickness((r, θ)Geometry) 26

Figure 10 Relative Axial Distribution of Fast Neutron Flux (E>1 MeV) at Inner Surface in RV (Normalized as Maximum Value Set at “1”) 27

Figure A- 1 Core Center Cross Section ((r, θ) Geometry) of H.B.Robinson-2 ..... A-13

Figure A- 2 Core Vertical Cross Section ((r, z) Geometry) of H.B.Robinson-2 ..... A-14

Figure A- 3 Cycle-Averaged Relative Power Distribution for Fuel Assemblies at Core Horizontal Cross Section of H.B.Robinson-2 ..... A-15

Figure A- 4 Cycle-Averaged Relative Power Distribution for Fuel Assembly, in Core Vertical Direction of H.B.Robinson-2 ..... A-16

List of Acronyms

The following list defines the acronyms used in this document.

DCD	Design Control Document
EFPY	Effective Full Power Years
ENDF	Evaluated Nuclear Data File
EOL	End-of-Life
LWR	Light Water Reactor
MHI	Mitsubishi Heavy Industries
ORNL	Oak Ridge National Laboratory
RG	Regulatory Guide
RV	Reactor Vessel
SUS	Steel Use Stainless

1.0 INTRODUCTION

Regulatory Guide(RG) 1.190 “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence” [1] provides the guidance for the application and qualification of a methodology for determining the neutron fluence experienced by materials in the beltline region of Light Water Reactor(LWR) reactor vessel(RV). The calculation methodology for RV neutron fluence of the US-APWR is according to RG1.190. The neutron fluence evaluation must include the evaluation of the maximum exposure at the inner surface in the RV, and also, the radial, axial, and azimuthal direction distribution in the RV. The evaluation of neutron fluence in the RV is carried out by the plant specific neutron transport calculation which is qualified using the benchmark experiment. In the evaluation of qualification, the measured date is used to assess the accuracy of transport calculation.

The purpose of this report is to describe the methodology used by Mitsubishi Heavy Industries (MHI) to determine the typical fast neutron fluence in the RV, the uncertainty in the evaluation, and the results of fast neutron fluence for US-APWR.

The typical neutron fluence in the RV is evaluated from the neutron transport calculation for LWR geometry, considering the uncertainty obtained from the measured value in benchmark problem.

The qualification of the neutron fluence evaluation for PWR geometry is carried out in the following two evaluations of uncertainty;

1) Comparison of calculations with surveillance capsule measurements from the H.B.Robinson power reactor benchmark experiment[2].

2) An analytical sensitivity study addressing the uncertainty resulting from the input parameters for neutron transport calculations used in the neutron fluence assessments.

In subsequent sections of this report, the methodology used by MHI to perform calculations of neutron fluence in RV for US-APWR, the uncertainties, and the results containing the uncertainty are described. The uncertainty and bias are shown in Appendix A.

2.0 NEUTRON TRANSPORT CALCULATION

As noted in Section 1 of this report, the typical fast neutron fluence of the RV is evaluated as absolute neutron transport calculation for LWR, considering the uncertainty that is estimated using the measurements obtained from the RV materials in the benchmark problems.

In this section, the neutron transport evaluation methodologies are discussed; and then, the necessary data, the uncertainty, required in RG1.190, and neutron fluence considering the uncertainty are presented.

2.1 NEUTRON TRANSPORT CALCULATION METHODS

The neutron fluence calculations are generally carried out using the two-dimensional discrete ordinates transport calculation method. The neutron transport calculations for US-APWR are accomplished using Discrete-Ordinates transport code DORT[3] and cross section library BUGLE-96[4] based on the Evaluated Nuclear Data File (ENDF/B-VI[5]). BUGLE-96 libraries provide a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water reactor application. In the DORT analysis, anisotropic scattering is treated with a P3 legendre expansion and the angular discretization is modeled with S8 angular quadrature.

Calculations are carried for (r, θ),(r,z) and (x,y) geometry.

For analysis, the actual geometries and core neutron source distributions of US-APWR are modeled.

The core neutron source distributions is based on the cycle-averaged power distributions of the 24 months equilibrium core considering special distributions and numbers of neutrons generated per fission.

The fundamental calculation equation is as follows;

$$\Omega \cdot \nabla \phi(r,\Omega,E) + \Sigma t(r,E) \cdot \phi(r,\Omega,E)$$
$$= \iint \Sigma s(r:\Omega' \to \Omega, E' \to E)\phi(r,\Omega',E')d\Omega' dE' + Q(r,\Omega,E)$$

where

$\phi(r,\Omega,E)$:angular flux (number of neutrons at a position r passing per unit time through a plane perpendicular to Ω per unit solid angle moving in the direction of a unit vector Ω)

$\Sigma t(r, E)$: total macro cross sections

$\Sigma s(r : \Omega' \rightarrow \Omega, E' \rightarrow E)$:macro scattering cross sections

$Q(r, \Omega, E)$:number of neutrons generated at fuel portion

[

]

2.2 CROSS SECTION LIBRARY

BUGLE96 is used as the cross section library. Developed by Oak Ridge National Laboratory (ORNL) in the USA on the basis of ENDF/B-VI, this library is an integrated cross section library consisting of neutron & gamma ray of 47 neutron groups and 20 gamma ray groups. Table 1 shows the energy group structure of these 47 neutron groups. [

]

2.3 GEOMETRY MODELING

In US-APWR actual geometry, the model includes the core, reactor internals, RV and the primary shield. In developing an analytical model of the reactor geometry, the nominal design

dimensions are normally used for geometry modeling. The coolant temperatures and coolant density in the reactor core and downcomer regions of the reactor are normally taken to be representative of full power operating conditions. [

] Selected general data and dimensions of US-APWR reactor are shown in Table 2. An octant of the horizontal cross-section of the reactor is shown in Figure 1, and axial geometry and dimensions are given in Figure 2.

The number density of each structure component is shown in Table 3.

(1) Horizontal cross section

[

]

(2) Vertical cross section

[

]

2.4 CORE NEUTRON SOURCE AND FISSION SPECTRA

The cycle-averaged power distribution of the 24 months equilibrium core[6] is used for the assessment of the RV neutron exposure. [

]

The cycle-averaged relative power distribution is shown in Figure 6 and the relative axial power distribution is shown in Figure 7.

The power distributions are calculated by the NRC approved code ANC[7], as described in Subsection 4.3.3.1 of the US-APWR DCD. The ANC can properly model the effect of the Neutron Reflector.

[

]

The core neutron source and fission spectra are summarized in Table 5.

2.5 OTHER CONDITIONS

Other conditions used for DORT calculation are shown as follows;

- Boundary condition

  (x,y)geometry

  Left, Bottom : reflection

  Right, Top : vacuum

  (r, θ)geometry

  Left, Top, Bottom : reflection

  Right : vacuum

  (r,z)geometry

  Left : reflection

  Right, Top, Bottom : vacuum

- point-wise inner iteration flux convergence criterion :0.001

2.6 UNCERTAINTY AND BIAS

In connection with the evaluation methodology of neutron flux, the R.G.1.190 defines that the effect of uncertainty (σ) and bias (B) shall be examined so as to correct the calculated value of neutron flux (φ_c) in accordance with the uncertainty and bias as indicated below and to set neutron flux (φ):

a) $\sigma \leq 20\%$: $\varphi = (1+B)\ \varphi_c$

b) $20\% < \sigma < 30\%$: $\varphi = (1+B+(\sigma - 20)/100)\ \varphi_c$

c) $\sigma \geq 30\%$: evaluation methodology inappropriate, review required

The uncertainty and bias for the current evaluation methodology are as follows and the basis is described in Appendix A.

uncertainty (σ) :20% or less

bias(B) : [ ]

Thus, the neutron flux and fluence inside the RV shall be evaluated considering the bias correction according to Item a) above to the neutron flux calculation results obtained by DORT calculation.

2.7 RESULTS OF NEUTRON TRANSPORT CALCULATIONS

The following sections summarize the neutron flux and the maximum fast neutron fluence obtained on the basis of the methodology and calculation conditions in previous section 2.1 to 2.6, where uncertainty and bias are taken into account.

2.7.1 HORIZONTAL CROSS SECTION

The neutron flux in the RV is calculated using (r, θ)geometry.

The maximum value in the azimuthal distribution appears at a 45 degree direction closest to the fuel assemblies, while the minimum value in the azimuthal distribution appears at a 0 degree direction.

The azimuthal distribution of neutron flux at inner surface in the RV is indicated in Figure 8. Similarly, the azimuthal distributions of neutron flux at 1/4 thickness in the RV is shown in Figure 9.

The maximum neutron flux at inner surface and 1/4 thickness in the RV is shown in Table 6. The maximum fast neutron fluence (E>1 MeV) inside the RV at the time of 60EFPY is shown in Table 7.

The neutron flux on the core boundary (boundary between the core and neutron reflector) in the radial direction is obtained using the (x, y) geometry.

The maximum value at the core boundary appears at the 45 degree direction.

The maximum neutron flux on the core boundary (boundary between the core and neutron reflector) are shown in Table 6.

2.7.2 VERTICAL CROSS SECTION

The axial distribution of neutron flux in the RV as well as neutron flux at the core center and on the core boundary (top and bottom side boundary) are calculated using (r, z) geometry.

The relative axial distribution (normalized as maximum value set at “1”) of neutron flux at inner surface in the RV is shown in Figure.10.

The maximum neutron flux in the core center and the neutron flux at the core boundary on center axis (top and bottom side boundary) are shown in Table 6. The maximum fast neutron fluence (E>1 MeV) on welded lines on the RV inner surface at the time of 60EFPY are shown in Table 7.

3.0 REFERENCES

1. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence", U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research, March 2001.

2. I.Remec and F.B.K.Kam, "H. B. Robinson-2 Pressure Vessel Benchmark", NUREG/CR6453 (ORNL/TM-13204), February 1998.

3. Oak Ridge National Laboratory,"DOORS 3.2: One-, Two- and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System", RSICC Computer Code Collection CCC-650.

4. Oak Ridge National Laboratory ,"BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications", RSICC Data Library Collection DLC=185.

5. P. F. Rose, ENDF/B-VI Summary Documentation, BNL-NCS-1741 (ENDF-201), 4th Edition

6. “US-APWR Fuel System Design Evaluation,” MUAP-07016-P, Appendix-A, February 2008

7. “Qualification of Nuclear Design Methodology using PARAGON/ANC,” MUAP-07019-P, December 2007

8. [
]

9. H.Soodak,Reactor Handbook Second Edition,Vol.III Part A,"Physics",1962

Table 1 Energy Group Structure of 47 Neutron Groups (Energy Group Structure of BUGLE96)

Neutron		
Group Number	Upper Energy (MeV)	Lower Energy (MeV)
1	1.7332E+01	1.4191E+01
2	1.4191E+01	1.2214E+01
3	1.2214E+01	1.0000E+01
4	1.0000E+01	8.6071E+00
5	8.6071E+00	7.4082E+00
6	7.4082E+00	6.0653E+00
7	6.0653E+00	4.9659E+00
8	4.9659E+00	3.6788E+00
9	3.6788E+00	3.0119E+00
10	3.0119E+00	2.7253E+00
11	2.7253E+00	2.4660E+00
12	2.4660E+00	2.3653E+00
13	2.3653E+00	2.3457E+00
14	2.3457E+00	2.2313E+00
15	2.2313E+00	1.9205E+00
16	1.9205E+00	1.6530E+00
17	1.6530E+00	1.3534E+00
18	1.3534E+00	1.0026E+00
19	1.0026E+00	8.2085E-01
20	8.2085E-01	7.4274E-01
21	7.4274E-01	6.0810E-01
22	6.0810E-01	4.9787E-01
23	4.9787E-01	3.6883E-01
24	3.6883E-01	2.9721E-01
25	2.9721E-01	1.8316E-01
26	1.8316E-01	1.1109E-01
27	1.1109E-01	6.7379E-02
28	6.7379E-02	4.0868E-02
29	4.0868E-02	3.1828E-02
30	3.1828E-02	2.6058E-02
31	2.6058E-02	2.4176E-02
32	2.4176E-02	2.1875E-02
33	2.1875E-02	1.5034E-02
34	1.5034E-02	7.1017E-03
35	7.1017E-03	3.3546E-03
36	3.3546E-03	1.5846E-03
37	1.5846E-03	4.5400E-04
38	4.5400E-04	2.1445E-04
39	2.1445E-04	1.0130E-04
40	1.0130E-04	3.7266E-05
41	3.7266E-05	1.0677E-05
42	1.0677E-05	5.0435E-06
43	5.0435E-06	1.8554E-06
44	1.8554E-06	8.7643E-07
45	8.7643E-07	4.1399E-07
46	4.1399E-07	1.0000E-07
47	1.0000E-07	1.0000E-11

Table 2 Selected general data and dimensions of US-APWR reactor

Reactor	
Number of loops	4
Thermal power	4451 MWt
Core	
Number of fuel assemblies	257
Fuel Element	
Type	17×17 array of fuel pins
Fuel pin per element	264
Horizontal cross section	rectangular
Height of fuel	420 cm
Neutron Reflector	
Dimensions	See Fig.1
Core Barrel	
Inner radius	[ ]
Thickness	[ ]
Pressure Vessel	
Cladding	
Inner radius	257.5 cm
Thickness	0.5 cm
Base metal	
Inner radius	258 cm
Thickness	26.5 cm
Total thickness (wall + cladding)	27 cm
Primary Shield	
Inner radius of cylinder surface	300.5 cm

Table 3 Number Density of Each Structure Component

Table 3 Number Density of Each Structure Component (Continued)

Table 3 Number Density of Each Structure Component (Continued)

Table 4 Fission Spectra of 47 Neutron Groups

Table 5 Core Neutron Source and Fission Spectra

Term	Contents
Core power	4451MWt
Core pattern	24 months equilibrium core load pattern[6]
Power distribution	See Figure 6 and 7
Axial peaking factor (used (r, θ), (x,y) geometry)	
Fission assignment rate	
Number of neutrons produced per fission[9]	
Energy released per fission	
Fission spectra of neutron	See Table 4

Table 6 Neutron Flux Inside RV

(Unit: n/(cm2•s))

	E > 1MeV	1MeV > E > 3.35keV	3.35keV > E > 0.414eV	E < 0.414eV
RV inner surface (maximum value: (r, θ)Geometry)	5.2E+9	1.1E+10	8.0E+9	1.1E+10
RV 1/4 thickness (maximum value) : (r, θ)Geometry)	2.5E+9	1.0E+10	3.3E+9	1.9E+8
Core center (maximum value: r,z Geometry))	1.0E+14	1.7E+14	1.2E+14	3.8E+13
Core outer boundary (maximum value: x,y Geometry)	4.6E+13	8.0E+13	5.7E+13	1.7E+13
Core top boundary (on center axis: r,z Geometry)	2.5E+13	3.9E+13	3.1E+13	2.4E+13
Core bottom boundary (on center axis: r,z Geometry)	2.7E+13	4.2E+13	3.3E+13	2.9E+13

Table 7 Fast Neutron Fluence Inside RV and on Welded Lines

(Unit: n/cm2)

	Fast Neutron fluence (E > 1MeV, 60EFPY)
RV inner surface (maximum value)	9.8E+18
RV 1/4 thickness (maximum value)	4.7E+18
Welded line (inner face) on bottom cylinder top (maximum value)	1.9E+18
Welded line (inner face) on bottom cylinder bottom (maximum value)	8.5E+18
Welded line (inner face) on bottom end cover top (maximum value)	< 1.0E+17 1)

1) As the bottom end cover top is located more than 150 cm below the bottom cylinder bottom, it is evident from Fig.8 that neutron fluence is well below 1.0 E+17 (n/c m^2).

Figure 1 Horizontal cross section of US-APWR reactor

Figure 2 Schematic sketch of the axial geometry

Figure 3 Core Center Cross Section ((r, θ) Geometry)

Figure 4 Core Center Cross Section ((x, y) Geometry)

Figure 5 Core Vertical Cross Section ((r, z) Geometry)

	1	2	3	4	5	6	7	8	9
1	0.70	0.85	1.08	1.02	1.22	1.08	1.10	1.20	0.58
2	0.85	0.91	0.95	1.28	1.07	1.27	1.27	1.19	0.59
3	1.08	0.95	1.29	1.27	1.05	1.33	1.08	1.08	0.55
4	1.02	1.28	1.27	1.05	1.28	1.26	0.99	1.10	0.52
5	1.22	1.07	1.05	1.28	1.30	0.99	1.20	1.06	0.41
6	1.08	1.27	1.33	1.26	0.99	0.90	1.02	0.60	
7	1.10	1.28	1.08	0.99	1.20	1.02	0.92	0.39	
8	1.20	1.19	1.08	1.10	1.06	0.60	0.39		
9	0.58	0.59	0.55	0.52	0.41				

Figure 6 Cycle-Averaged Relative Power Distribution for Fuel Assemblies at Core Horizontal Cross Section

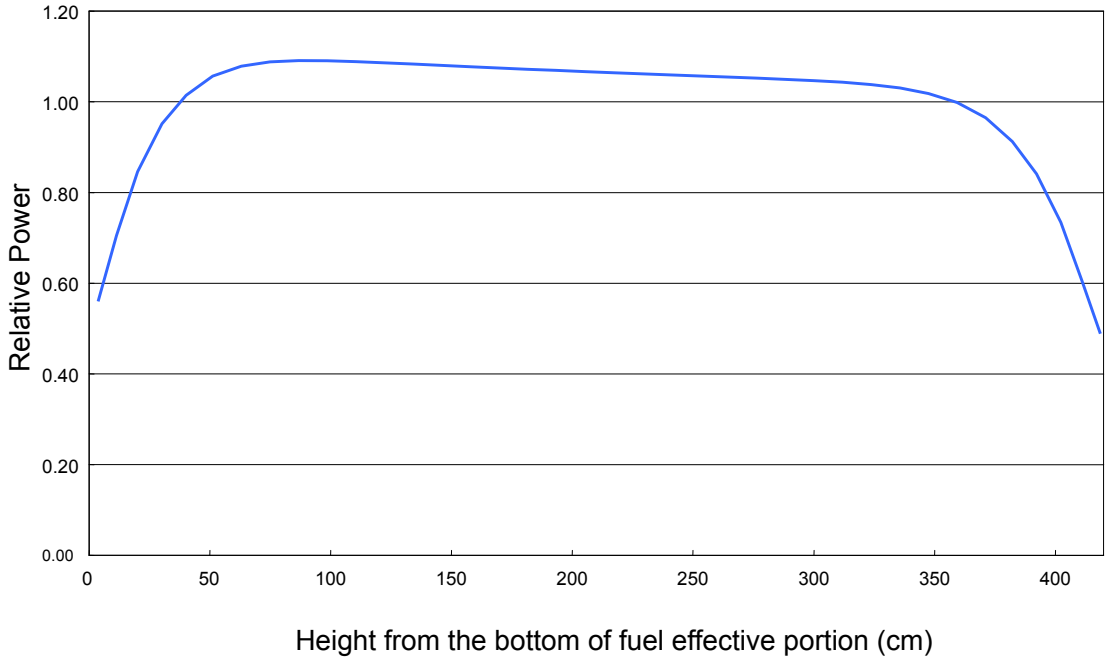

Figure 7 Cycle-Averaged Relative Power Distribution for Fuel Assembly, in Core Vertical Direction

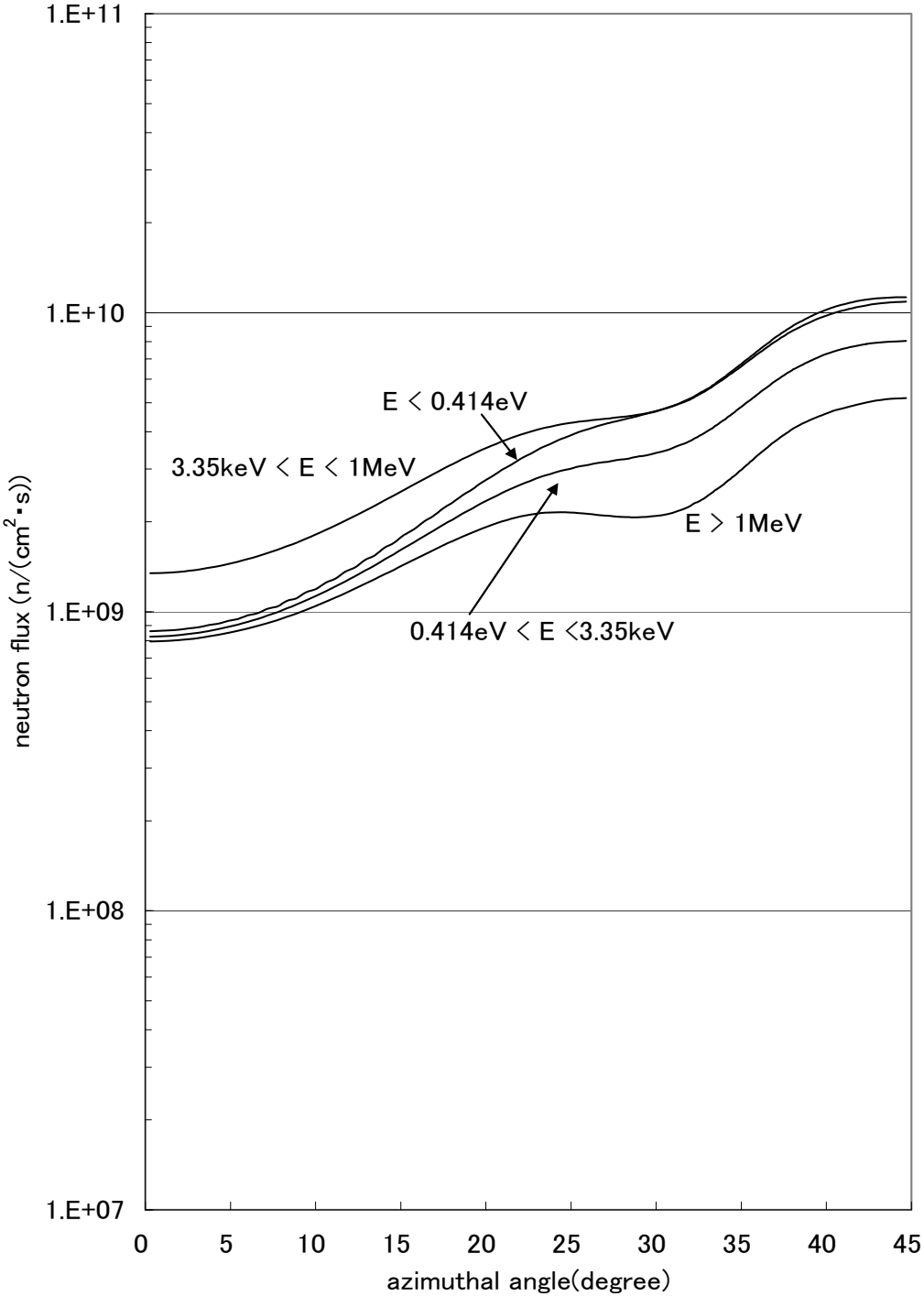

Figure 8 Distribution of Neutron Flux in Azimuthal Direction at Inner Surface in RV ((r, θ) Geometry)

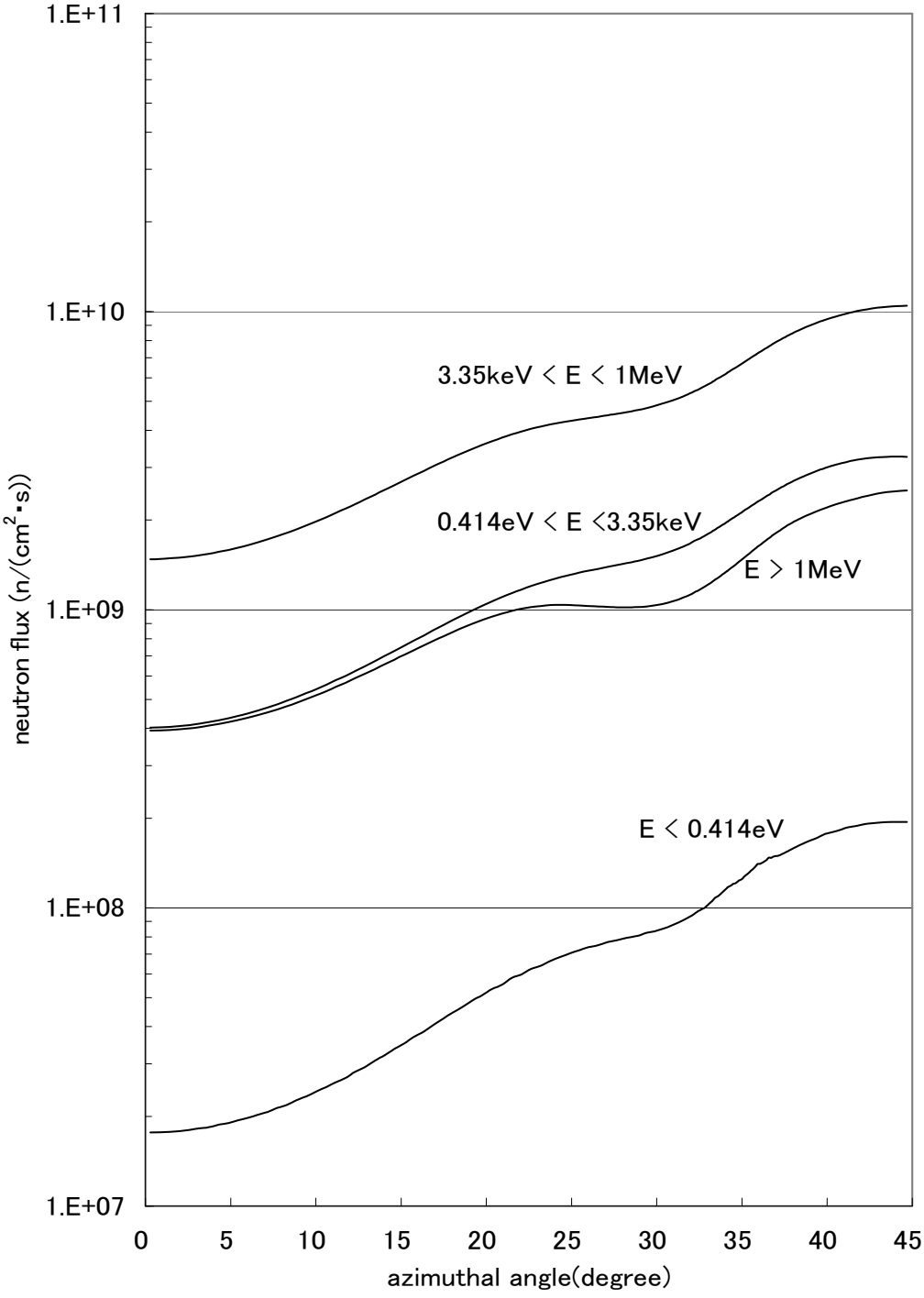

Figure 9 Distribution of Neutron Flux in Azimuthal Direction at RV 1/4 Thickness((r, θ)Geometry)

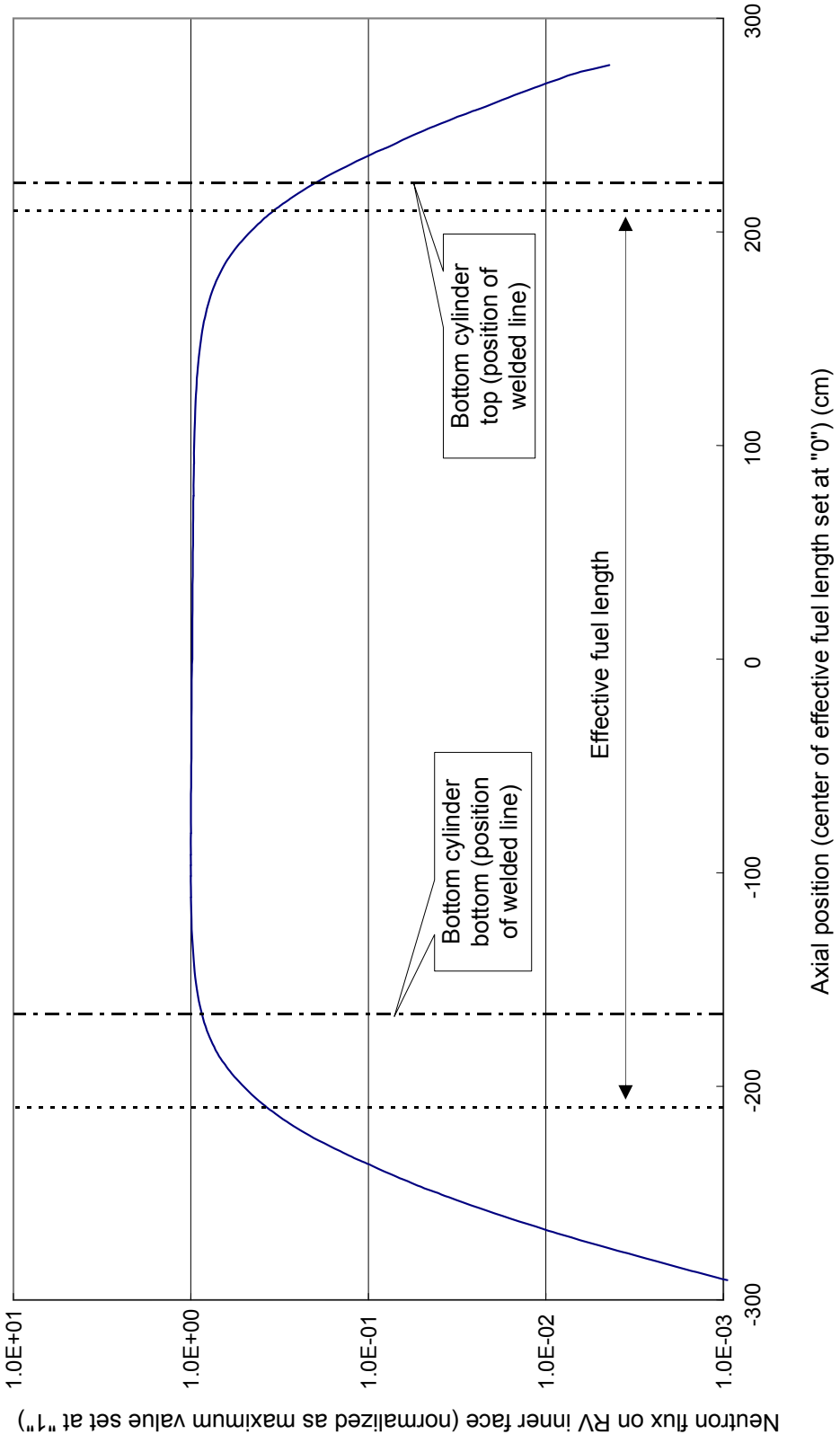

Figure 10 Relative Axial Distribution of Fast Neutron Flux (E>1 MeV) at Inner Surface in RV (Normalized as Maximum Value Set at "1")

APPENDIX A STUDY OF UNCERTAINTY AND BIAS

The benchmark analysis and the uncertainty analysis were done to qualify the MHI's methodology for Reactor Vessel (RV) Neutron Fluence calculation.

A.1 Comparisons measurements with calculational Benchmarks

A.1.1 Benchmark analysis

The benchmark analysis was done for the qualification of MHI methodology for RV neutron fluence calculations. The benchmark problem, analysed here, is H.B.Robinson-2 Pressure Vessel Benchmark (NUREG/CR-6453)[1].

The transport calculations were performed using the DORT computer code [2]. The calculation were done for (r, θ) geometry and (r,z) geometry. These geometries were shown in Figure A-1 and Figure A-2, respectively. In the (r, θ) geometry, the surveillance capsule was set in the geometry.

The BUGLE-96[3] cross section library based on ENDF/B-VI[4] is used in the calculations. [

]

The source distributions were made based on the power distribution averaged over the cycle-9 of H.B.Robinson. The cycle-averaged relative power distribution of each fuel assembly is shown in Figure A-3. [

]

The axial distribution was derived based on the axial power distribution averaged over the cycle-9 of H.B.Robinson. The relative axial power distribution is shown in Figure A-4. [

]

The major calculational conditions are summarized in Table A-1.

A.1.2 Results and discussion

The reaction rates calculated for the cycle average power distribution and core thermal power output of 2300MW, were given in Table A-2. In the table, the ratio (C/M) calculated to measured specific activities are also given.

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A.2 UNCERTAINTY ANALYSIS

An analytical sensitivity studies addressing the uncertainty resulting from the input parameters for neutron transport calculations used in the neutron fluence assessments were carried out for the parameters shown in Table A-3. The sensitivity studies are done for H.B.Robinson-2.

A.2.1 Modeling

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A.2.1.1 (r, θ) modeling

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A.2.1.2 Internal dimension

[

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A.2.1.3 Vessel inner radius

[

]

A.2.1.4 Coolant temperature

[

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A.2.2 Source Term

[

]

A.2.2.1 Peripheral assembly source strength

[

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A.2.2.2 Axial power distribution

[

]

A.2.2.3 Peripheral assembly burn-up (U/Pu ratio)

[

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A.2.2.4 Spatial distribution of the source

[

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A.2.3 Others

[
]

A.2.3.1 Convergence criteria

[
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A.2.3.2 Transport cross section

[

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A.2.3.3 Fission spectra

[

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A.2.3.4 Angular discretization

[

]

A.2.4 Results of sensitivity studies

[

]

The uncertainty in MHI's methodology is less than 20%.

APPENDIX A REFERENCES

1. I.Remec and F.B.K.Kam, "H. B. Robinson-2 Pressure Vessel Benchmark", NUREG/CR6453 (ORNL/TM-13204), February 1998.

2. Oak Ridge National Laboratory,"DOORS 3.2: One-, Two- and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System", RSICC Computer Code Collection CCC-650.

3. Oak Ridge National Laboratory,"BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications", RSICC Data Library Collection DLC=185.

4. P. F. Rose, ENDF/B-VI Summary Documentation, BNL-NCS-1741 (ENDF-201), 4th Edition

5. [RSIC Computer Code Collection CCC-276, "DOT 3.5, Two Dimensional Discrete Ordinates Radiation Transport Code", 1977.]

6. [ ]

7. [ ]

8. [ ]

9. [
]

Table A- 1 Analysis input condition for DORT calculations

term	contents	information
Benchmark problem	NUREG/CR-6453	H.B.Robinson-2
Computer code	DORT	DOORS-3.2[2]
Cross section library Activation cross section	BUGLE-96	[ ]
Material composition	Based on NUREG/CR-6453	NUREG/CR-6453[1]
Pℓ, Sn	[ ]	-
Geometry	(r, θ) : Figure A-1 (r, z) : Figure A-2	NUREG/CR-6453[1]
Source distribution	Radial distribution : Figure A-3 Axial distribution : Figure A-4	NUREG/CR-6453 Averaged over the cycle-9
Fission rate		
Fission spectra	Table4 of section 2.0	
Neutron production rate per fission		
Core thermal power	2300MWt	NUREG/CR-6453[1]
Boundary conditions	(r, θ) Left, Top, Bottom : reflection Right : vacuum (r, z) Left : reflection Right,Top,Bottom : vacuum	
Convergence criteria	≤0.001 (≤0.1%)	

Table A- 2 Comparison to Measurement and Calculational Benchmark

Table A- 3 Input parameters evaluated in the analytical sensitivity studies

INPUT PARAMETERS		
<MODELING>	<SOURCE TERM>	<OTHERS>
(r, θ) modeling	Peripheral assembly source strength	Convergence criteria
Internal dimension	Axial power distribution	Transport cross section
Vessel inner radius	Peripheral assembly burnup	Fission spectra
Coolant temperature	Spatial distribution of the source	Angular discretization

Table A- 4 Evaluated analytic uncertainties (E>1MeV)

		σ (%) in-vessel dosimeter location	σ (%) Inner Surface of RV
modeling	(r, θ)modeling		
	Internal dimension		
	Vessel inner radius		
	Coolant temperature		
Source term	Peripheral assembly source strength		
	Axial power distribution		
	Peripheral assembly burnup		
	Spatial distribution of the source		
others	Convergence criteria		
	Transport cross section		
	Fission spectra		
	Angular discretization		
TOTAL	= $\sqrt{\sum \sigma^2}$		

Figure A- 1 Core Center Cross Section ((r, θ) Geometry) of H.B.Robinson-2

Figure A- 2 Core Vertical Cross Section ((r, z) Geometry) of H.B.Robinson-2

	H	G	F	E	D	C	B	A
8	1.01	1.17	1.02	1.26	1.01	0.95	1.10	0.44
9	1.17	1.03	1.14	0.99	1.18	1.03	1.02	0.33
10	1.02	1.14	1.02	1.11	1.05	1.21	0.94	
11	1.27	1.01	1.11	1.01	1.16	1.12	0.72	
12	1.05	1.22	1.06	1.15	1.19	0.82		
13	0.98	1.06	1.23	1.12	0.81			
14	1.13	1.05	0.96	0.73				
15	0.44	0.33						

Figure A- 3 Cycle-Averaged Relative Power Distribution for Fuel Assemblies at Core Horizontal Cross Section of H.B.Robinson-2

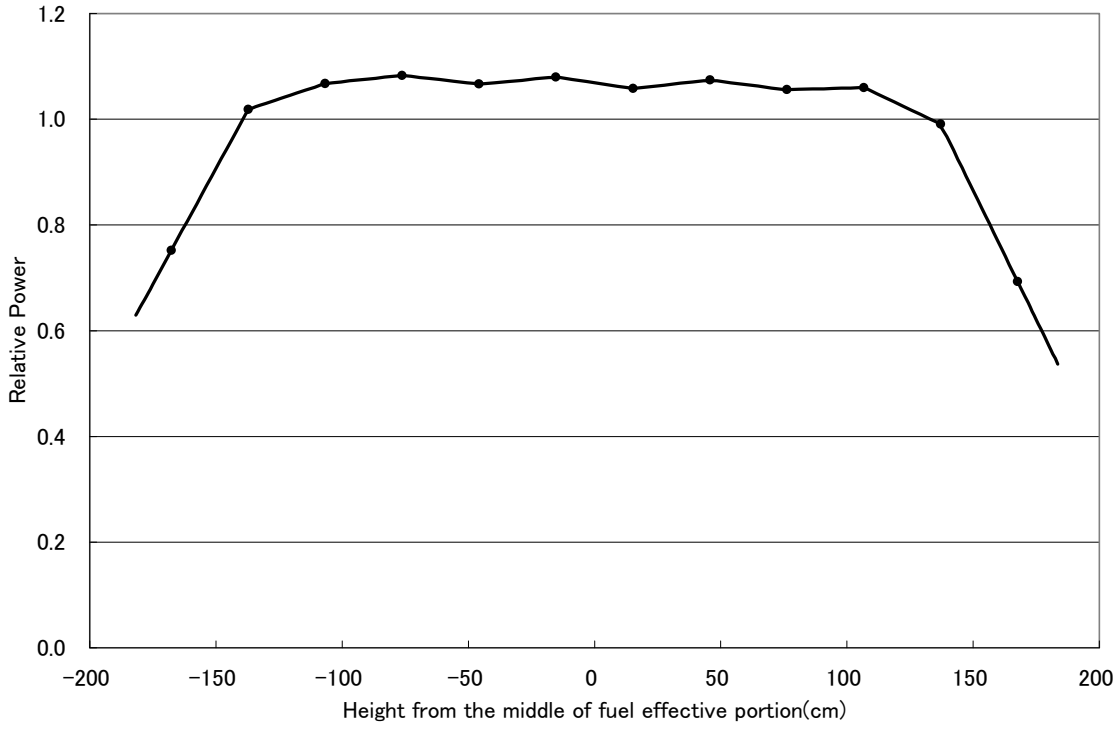

Figure A- 4 Cycle-Averaged Relative Power Distribution for Fuel Assembly, in Core Vertical Direction of H.B.Robinson-2