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Relative Power Density Distribution Calculations of the Kori Unit 1 Pressurized Water Reactor with Full-Scope Explicit Modeling of Monte Carlo Simulation

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Abstract

Relative power density distributions of the Kori Unit 1 pressurized water reactor are calculated by Monte Carlo modeling with the MCNP code. The Kori Unit 1 core is modeled on a three-dimensional representation of the one-eighth of the reactor in-vessel component with reflective boundaries at 0 and 45 degrees. The axial core model is based on half core symmetry and is divided into four axial segments. Fission reaction density in each rod is calculated by following 100 cycles with 5,000 test neutrons in each cycle after starting with a localized neutron source and ten noncontributing settle cycles. Relative assembly power distributions are calculated from fission reaction densities of rods in assembly. After 100 cycle calculations, the system converges to a k value of $1.00039 \geq 0.00084$. Relative assembly power distribution is nearly the same with that of the Kori Unit 1 FSAR. Applicability of the full-scope Monte Carlo simulation in the power distribution calculation is examined by the relative root mean square error of 2.159%.

1. Introduction

An accurate estimate of the neutron fluence at the reactor pressure vessel is necessary to ensure the integrity over the designed lifetime and to support analyses for a potential plant-life extension. The pressure vessel fluence calculation needs a detail and careful modeling because of the uncertainties from geometry, source terms, cross sections, analysis methods, and so on. Conventional calculation method is mainly the use of two-dimensional deterministic codes such as discrete ordinates transport code, *e.g.*, DORT.[1] In order to reduce the uncertainties associated with geometric modeling, some Monte Carlo models have been attempted to determine the neutron fluence at the pressure vessel by previous workers.[2,3]

In earlier study, it showed that the constant power distribution in the outer assemblies causes an error of 15 to 25% in the pressure vessel fluence calculation.[4] The pin-by-pin source terms, however, were not considered in the previous Monte Carlo calculations using MCNP code.[2,3,5] The homogenized assembly models with volume averaged cross section and approximated source terms at the outer assemblies were used for the calculations. The major advantages of the Monte Carlo method were not applied in their MCNP models because the pin-by-pin source terms were not calculated by using the MCNP code.

In the conventional computations[4,6] including the previous works,[2,3] the source terms in the reactor vessel fluence calculations have been obtained from the experimental data or calculated by coupling of the fuel assembly and core burn-up calculation codes which are usually provided by vendors. In the conventional source term calculations, the boundary condition at the reflective regions having outer assemblies, baffles, and coolant are very important because it may affect the power distributions in the outer assemblies. However, it is not easy to know the boundary conditions because current and cross section of the reflective regions should be separately evaluated by another transport code. The Monte Carlo method, however, does not need any conditions because all particles are directly simulated over the boundaries.

In order to calculate the pin-by-pin fission source using the Monte Carlo method, all reactor components including fuel rods should be modeled on a full-scope three-dimensional reactor model. The modeling is not simple because of its vast size and complication.

In this work, the criticality and fission reaction density are determined by Monte Carlo calculations. The pin-by-pin power distributions are calculated from the fission reaction density in each rod during the criticality calculation. The relative assembly power distributions are obtained from the pin-by-pin power distributions of the full-scope explicit three-dimensional MCNP modeling. The pin-by-pin power distributions can be used in the neutron fluence calculations of the pressure vessel in future works without any assumptions, such as homogenized assemblies, uniform source distributions, and energy group cross sections.

2. The Kori Unit 1 Reactor Core

The Kori Unit 1 is the first commercial nuclear power plant in Korea since 1978. The design specifications of the Kori Unit 1 core are listed in Table 1. [7] The reactor core consists of 121 fuel assemblies as shown in Figure 1. Each fuel assembly contains a 14×14 rod array composed of 179 fuel rods, 16

Table 1. Specifications of the Kori Unit 1 Reactor Core

Core	
Core Average Active Fuel Height, cm	365.76
Average Temperature, ℉	573.0
Fuel Assembly	
Number of Fuel Assemblies	121
Region 1	41
Region 2 and 3	40
Rod Array	14×14
Rods per Assembly	179
Rod Pitch, cm	1.141224
Fuel Rod	
Number	21,659
Outside Diameter, cm	1.07188
Diameter Gap, cm	0.01905
Clad Thickness, cm	0.061722
Clad Material	Zircaloy-4
Average Clad Temperature, ℉	634.7
Fuel pellets	
Material	UO_2 Sintered
Density(Percent of Theoretical)	95
Fuel Enrichment, wt. %	
Region 1	2.10
Region 2	2.83
Region 3	3.30
Diameter, cm	0.929386
Length, cm	1.524
Average Temperature, ℉	1399.3
Rod Cluster Control Assemblies	
Neutron Absorber	Ag-In-Cd
Composition	80% 15% 5%
Diameter, cm	0.99187
Density, g/cm^3	10.15463698
Clad Material	Stainless Steel 304
Clad Thickness, cm	0.047752
Burnable Poison Rods	
Number	624
Material	Borosilicate Glass
Outside Diameter, cm	1.09474
Inner Tube, O.D., cm	0.60071
Clad Material	Stainless Steel 304
Inner Tube Meterial	Stainless steel 304
Boron Loading(wt. % B_2O_3 in Glass Rod)	12.5
Baffle	
Thickness, cm	2.8575
Material	Stainless Steel

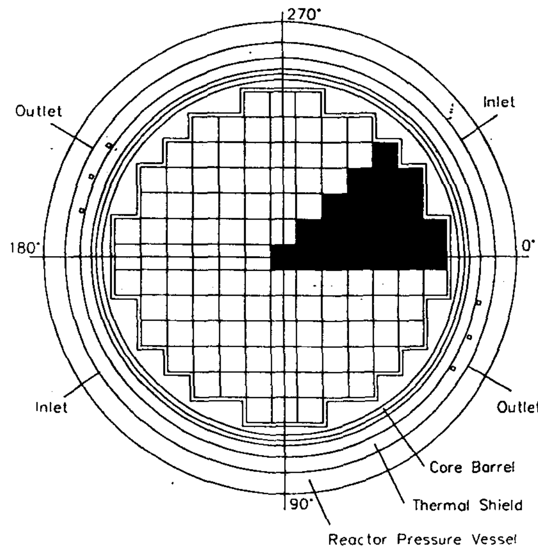

Fig. 1. Cross Sectional View of Kori Unit 1 Reactor

rod cluster control (RCC) guide thimbles, and an incore instrumentation thimble. The fuel rods are constructed by Zircaloy-4 cylindrical tubes containing slightly enriched UO_2 fuel pellets. All material compositions of the Kori Unit 1 structure are summarized in Table 2.

Table 2. Material Compositions of the Kori Unit 1 Structure

Structure	Composition	Number Density(#/cm-barn)
UO_2(2.10w/o)	U^{235}	4.84371×10^{-4}
	U^{238}	2.22957×10^{-2}
	O	4.55524×10^{-2}
UO_2(2.83w/o)	U^{235}	6.52748×10^{-4}
	U^{238}	2.21294×10^{-2}
	O	4.55524×10^{-2}
UO_2(3.20w/o)	U^{235}	7.38089×10^{-4}
	U^{238}	2.20452×10^{-2}
	O	4.55524×10^{-2}
Stainless Steel Type 304	C	3.971×10^{-4}
	Cu	1.727×10^{-4}
	Si	6.284×10^{-4}
	Mo	2.387×10^{-4}
	Ni	4.956×10^{-4}
	Mn	1.320×10^{-3}
	Cr	7.339×10^{-5}
	Fe	8.244×10^{-2}
Steel (Pressure Vessel)	C	8.736×10^{-4}
	Cu	5.254×10^{-5}
	Si	4.415×10^{-4}
	Mo	2.834×10^{-4}
	Ni	5.930×10^{-4}
	Mn	5.469×10^{-4}
	Cr	2.935×10^{-4}
	Fe	8.296×10^{-2}
Zircaloy-4	C	4.016×10^{-5}
	O	2.379×10^{-4}
	Al	1.135×10^{-5}
	Cr	9.690×10^{-5}
	Fe	1.619×10^{-4}
	Zr	4.310×10^{-2}
Pyrex	B^{10}	9.5436×10^{-4}
	B^{11}	3.8656×10^{-3}
	Al	4.7678×10^{-4}
	Na	2.2885×10^{-3}
	O	4.3561×10^{-2}
	Si	1.7891×10^{-2}
Moderator	H	4.8483×10^{-2}
	O	2.4242×10^{-2}
	B^{10}	1.2222×10^{-5}
	B^{11}	4.1406×10^{-2}

3. MCNP Model Development

The Kori Unit 1 core geometry model is a three-dimensional representation of the one-eighth of the reactor in-vessel component with reflective angular boundaries at 0 and 45 degrees. The computational models constructed for the Kori Unit 1 are shown in Figures 2 and 3. The axial model shown in Figure 3 is based on half core symmetry. The half core is divided into four axial segments for detailed axial power distribution calculations. Vacuum boundary condition is prescribed on the outside of the reactor vessel. Top reflector is assumed by three layers of baffle, coolant, and barrel as like the side reflector region.

The fuel loading pattern for cycle 1 is modeled as follows. It is assumed that there is no fission products

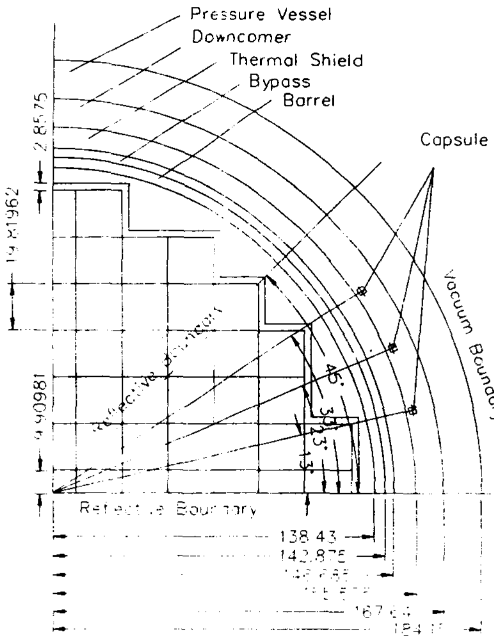

Fig. 2. The Cross Sectional View of Kori Unit 1 MCNP Model

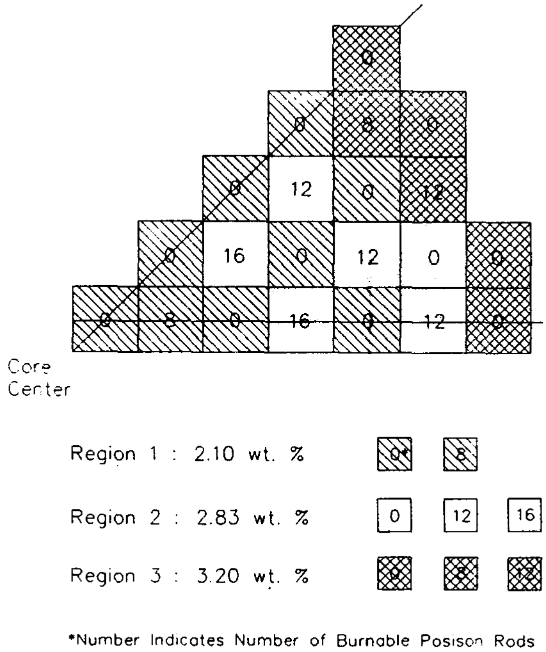

Fig. 4. Fuel Loading Arrangement of the Kori Unit 1 for Cycle 1

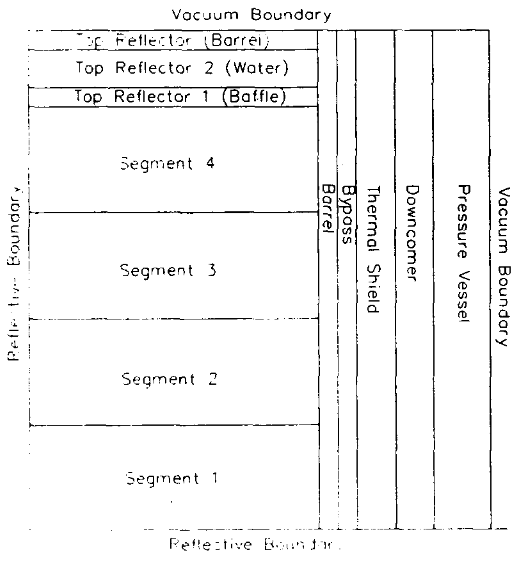

Fig. 3. The Axial View of Kori Unit 1 MCNP Geometry Model

from the core depletion because of fresh fuel. The fuel loading pattern used in this work is shown in Figure 4. This loading pattern consists of three regions with different ^{235}U enrichments. The enrichments for cycle 1 are 2.10, 2.83, and 3.20 weight percent in regions 1, 2, and 3, respectively. The fuel loading pattern is modeled with all rods out (ARO), no xenon, and 1278 ppm of boron concentration near the beginning of life (BOL).

The fuel element is explicitly modeled, *i.e.*, fuel, gap, clad, and coolant, to eliminate any homogenization effects. A cross sectional drawing of a rod is shown in Figure 5A. Borosilicate in the form of Pyrex glass containing 12.5% B_2O_3 is used as a burnable poison. A cross sectional drawing of a burnable poison rod inside a Zircaloy guide thimble is shown in Figure 5B. Burnable poison is also modeled explicitly.

Eight patterns of the fuel assembly (Figure 4) are modeled by means of the number of burnable poisons and the different enrichments. Four different models of the fuel assembly patterns with 0, 8, 12,

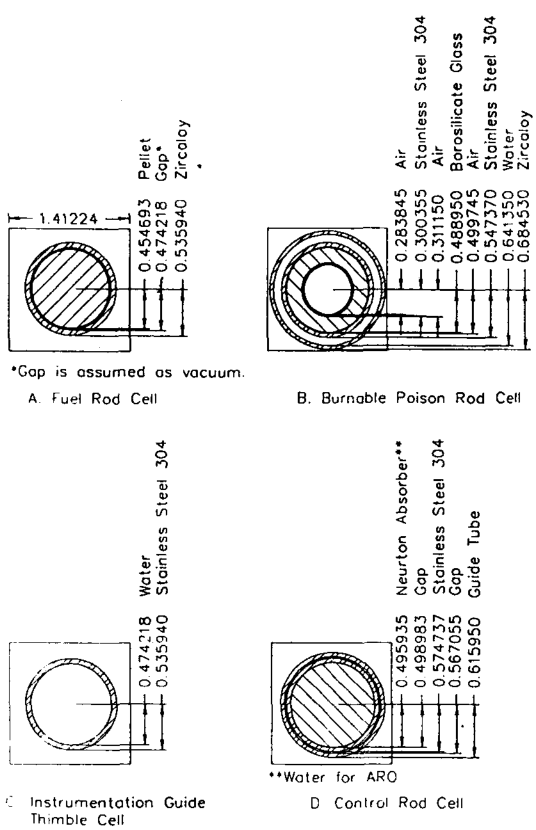

Fig. 5. Various Rods in MCNP Modeling

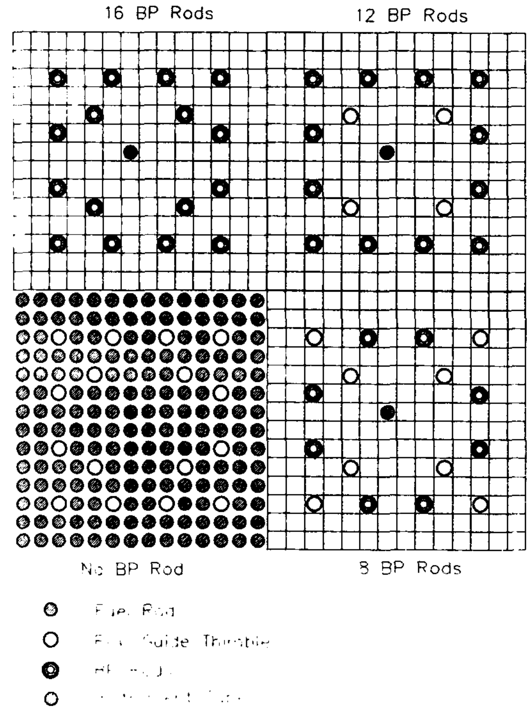

Fig. 6. Burnable Poison Rods, Rod Cluster Control Guide Thimbles, and Instrumentation Thimble within An Assembly

and 16 burnable poisons are shown in Figure 6.

The in-core instrumentation consists of 39 fuel assembly outlet thermo-couples and 36 movable neutron detectors. The in-core instrumentation systems are not equipped in this modeling for simplicity and are filled with water instead. The cross sectional drawing of the in-core instrumentation tube model is shown in Figure 5C. All RCC guide thimbles are filled with water because the modeling is a case of all rods out. The cross sectional drawing of a RCC guide thimble model is shown in Figure 5D. The fuel rods are supported at intervals along their length by Inconel-718 grid assemblies which maintain the lateral spacing between the rods. These grids are not considered for simplicity.

4. Cross Section Data

In addition to geometry and material composition, preparations of cross section data are also important for the Monte Carlo calculations. The new cross section library, called HYUXS, was generated at the operating core temperature by NJOY[8] and ENDF/B-VI.[9]. The HYUXS are to be compared with the existing libraries of the MCNP, *i.e.*, BMCCS, RMCCS, and ENDF5P, which were generated from ENDF/B-V or older ENDF/B versions at the room temperature.[5] The slow neutron scattering cross section data was evaluated at 600 *K*. This thermal scattering data library, TMCCS, is essential to model the neutron interactions at energies below 4*eV* accurately. The temperature of the moderator and coolant were assumed to be 573.0°F which is an average temperature for the core. All of the libraries in the this work are continuous energy cross sections.

5. Monte Carlo Calculations

All of the criticality calculations were performed with the KCODE option in the MCNP version 4A[5] The Monte Carlo discrete representation of a fission generation is a cycle. Cycles are made up of histories, where each history is the random walk of a simulated neutron from its birth as a source neutron to its death by escape, absorption, or fission. In criticality calculation of the MCNP, however, the neutron sources are started from the fission neutrons which are settled from seed neutrons in initial cycles. Initial cycles, called inactive cycles, serve to converge the fission source spatially. In this work, 84 neutrons were used as the seeds in the initial cycles. Once the source is converged, the subsequent cycles are called the active cycles. The criticality calculations were performed during the active cycles, 100 cycles in this study. The number of neutrons per cycle is 5000, and the total number of cycles is 110 for the eigenvalue and fission reaction density calculations.

The fission reaction density calculations were performed in all rods of the four segments. F4 tally and FM4 multiplier were used to calculate the fission reaction density during the active cycles. The F4 tally estimates the flux in volumes. The FM4 multiplier provides the fission reaction cross sections. Fission reaction density in each tally cell are calculated as follows.

Fission reaction density in a cell

$$= N_{235} \int_V \int_t \int_E \sigma_f^{235}(\vec{r}, E)\, \Phi(\vec{r}, E, t)\, dE\, dt \frac{dV}{V} + N_{238} \int_V \int_t \int_E \sigma_f^{238}(\vec{r}, E)\, \Phi(\vec{r}, E, t)\, dE\, dt \frac{dV}{V} \quad (1)$$

where,

$\Phi(\vec{r}, E, t)$ = particle flux, neutron/cm^2·sec·MeV,

N_{235} = atomic number density of ^{235}U, atoms/cm^3,

N_{238} = atomic number density of ^{238}U, atoms/cm^3,

$\sigma_f^{235}(\vec{r}, E)$ = microscopic fission cross section of ^{235}U, cm^2/atom,

$\sigma_f^{238}(\vec{r}, E)$ = microscopic fission cross section of ^{238}U, cm^2/atom,

and V = tally cell volume, cm^3.

The approximate run time for both the eigenvalue and fission reaction density calculations is about 5, 700 CPU minutes on the HP 9000/770 workstation.

6. Computational Results and Discussions

The fission reaction density calculations in each pin in four segments were carried out, after running the KCODE calculation of the MCNP. About 500, 000 initial source neutrons with Watt fission spectrum were used to obtain a pin-by-pin fission density in the full core assemblies.

Two calculations were carried out with different cross section libraries. The first calculation, called ROOM, was performed with the MCNP library of the

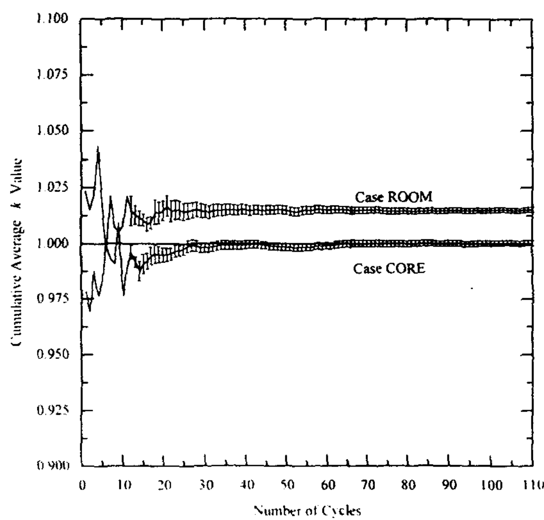

Fig. 7. Cummulative Average k Values of the Kori Unit 1 Core in KCODE Mode

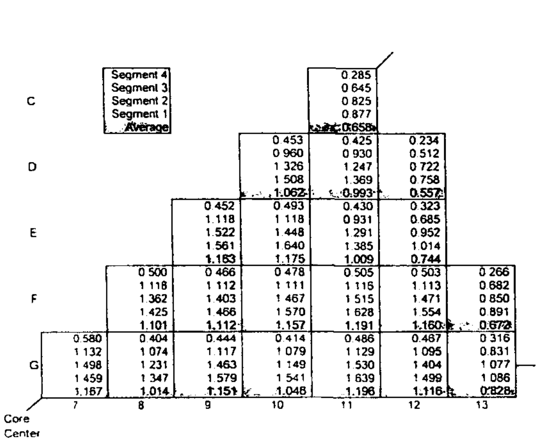

Fig. 8. **Relative Assembly Power Density Distribution of 4 Segments in the ROOM Case(Near Beginning of Life Unrodded Core, Hot Full Power, No Xenon)**

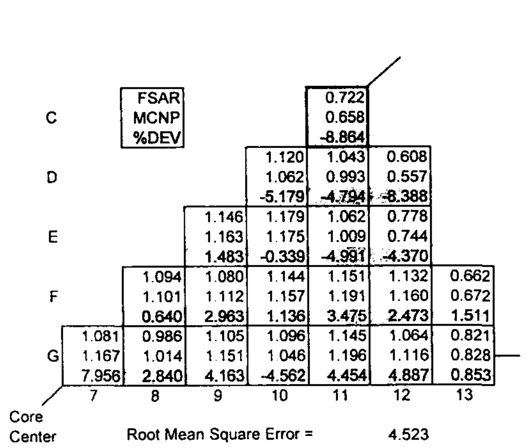

Fig. 10. **Axially Averaged Power Density Distribution in the ROOM Case(Near Beginning of Life Unrodded Core, Hot Full Power, No Xenon)**

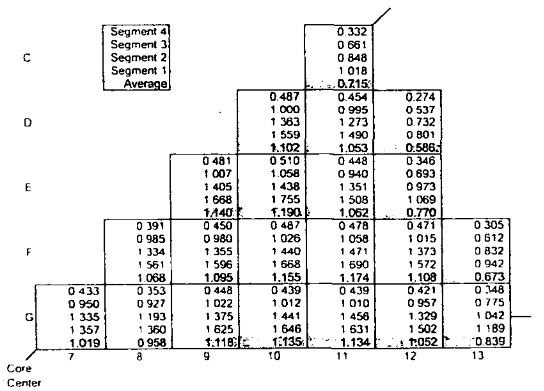

Fig. 9. **Relative Assembly Power Density Distribution of 4 Segments in the CORE Case(Near Beginning of Life Unrodded Core, Hot Full Power, No Xenon)**

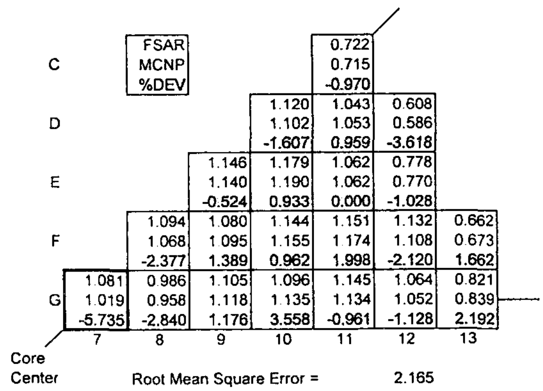

Fig. 11. **Axially Averaged Power Density Distributions in the CORE Case(Near Beginning of Life, Unrodded Core, Hot Full Power, No Xenon)**

room temperature. In the other calculation, called CORE, the MCNP library was replaced by the HYUX-S library for the core temperature.

The full core systems of the ROOM and CORE cases were converged to the *k* values of 1.01556≥0.00084 and 1.00039≥0.00084, respectively. Figure 7 shows the convergence of the *k* values with cycle increase. The calculated criticality of the CORE case is closer to the unity than that of the ROOM case. It shows that the HYUXS library can be used in the MCNP simulation of the Kori Unit 1 core.

The pin-by-pin power distributions were normalized by dividing the pin-by-pin fission reaction density to the total fission reaction density in the whole core. The normalized pin power distributions are the relative pin powers in each segment. The relative assembly powers were then calculated by averaging the relative pin powers in each assembly. Figures 8 and 9 are the ROOM and CORE cases, respectively.

The axially averaged pin powers were calculated

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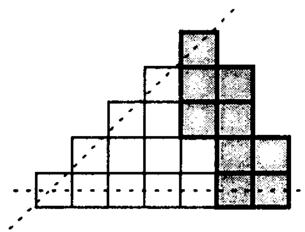

63	56	53	53	51	48	46	37	32	29	31	25	24	22
64	65	71	63	58	52	42	42	52	44	40	29	30	24
77	73	0	79	71	0	64	65	0	57	49	0	29	25
77	73	81	82	76	66	64	71	64	48	56	49	40	31
86	76	93	94	0	71	71	68	67	0	48	57	44	29
86	83	0	81	79	82	79	73	70	67	64	0	52	32
103	96	96	94	82	89	0	80	73	68	71	65	42	37
87	82	101	92	90	84	82	0	79	71	64	64	42	46
97	102	0	99	95	88	84	89	82	71	66	0	52	48
105	115	108	99	0	95	90	82	79	0	76	71	58	51
109	110	117	104	99	99	92	94	81	94	82	79	63	53
100	105	0	117	108	0	101	96	0	93	81	0	71	53
109	98	105	110	115	102	82	96	83	76	73	73	65	56
117	109	100	109	105	97	87	103	86	86	77	77	64	63

131	123	109	95	105	89	89	86	88	82	79	88	72	63	61	58	54	44	49	39	38	31	30	27	28	24	18	16
130	129	122	115	96	84	91	94	81	94	92	86	81	67	62	66	60	56	48	45	47	41	39	33	36	24	21	19
121	129	0	125	94	0	92	85	0	93	88	0	85	73	68	71	0	64	62	0	48	46	0	40	42	0	25	21
136	129	128	111	116	97	105	98	93	98	95	87	85	87	75	65	75	73	66	60	56	46	48	50	41	35	27	26
131	112	109	113	0	116	102	105	102	0	90	77	79	86	85	77	74	77	0	66	54	55	55	0	40	32	34	33
136	119	0	109	118	117	100	106	99	96	92	0	81	85	86	85	0	79	70	68	69	61	51	52	42	0	36	27
131	122	109	108	117	123	123	118	115	103	94	76	84	98	88	87	83	79	80	73	70	68	66	56	44	55	38	26
127	108	101	114	111	109	116	0	111	98	92	87	91	91	93	85	82	79	77	69	75	0	63	51	51	52	37	26
133	108	0	118	122	113	111	121	108	103	86	0	90	82	99	93	0	85	78	70	69	70	68	63	54	0	40	31
140	130	127	124	0	124	110	110	116	0	104	91	85	100	90	97	96	96	0	78	71	66	71	0	57	46	39	39
138	137	136	122	114	116	114	118	105	113	113	115	94	102	92	89	91	87	89	76	71	72	64	64	59	47	41	39
138	125	0	124	127	0	109	107	0	110	115	0	105	89	95	105	0	98	93	0	86	81	0	63	60	0	42	35
157	138	131	133	132	127	114	115	102	109	111	125	115	108	101	86	93	92	96	88	78	75	66	52	59	53	46	29
138	144	145	145	146	142	127	129	128	115	117	116	120	121	94	96	85	94	81	73	75	69	67	60	59	48	42	34
110	97	107	103	109	100	99	102	101	96	89	81	91	87	99	94	82	82	77	74	67	82	71	61	58	49	41	32
110	98	112	110	113	117	114	108	104	99	90	94	87	74	103	92	77	76	73	72	81	78	62	63	48	46	42	37
108	107	0	122	116	0	115	116	0	105	111	0	96	90	115	98	0	90	75	0	66	63	0	59	57	0	45	42
104	110	114	119	114	114	119	112	116	118	103	95	96	85	116	104	98	89	78	81	73	69	67	68	55	50	52	40
111	111	118	122	0	115	118	116	106	0	115	105	94	98	116	108	85	84	0	95	77	68	70	0	58	48	53	44
110	119	0	126	112	112	116	106	107	102	113	0	105	90	126	96	0	87	90	94	84	76	68	69	53	0	41	42
122	108	124	120	118	114	111	121	106	110	113	114	103	93	116	105	83	82	100	94	87	86	80	78	65	47	51	46
103	116	125	122	117	118	116	0	124	111	115	103	97	95	121	109	91	100	101	97	92	0	84	83	65	60	48	55
105	115	0	131	128	116	110	123	124	119	114	0	96	88	113	99	0	88	105	90	90	85	87	86	64	0	52	39
111	124	123	125	0	118	114	124	127	0	117	110	103	98	123	110	103	101	0	95	93	92	88	0	71	61	58	42
110	119	127	126	120	129	131	120	123	125	115	112	99	98	125	108	97	98	107	99	85	82	83	76	70	64	55	48
109	114	0	118	111	0	123	127	0	126	118	0	105	109	126	113	0	101	96	0	92	93	0	79	72	0	58	51
120	122	122	115	115	116	105	107	119	120	113	117	100	100	145	121	105	92	101	87	102	97	90	83	73	69	74	65
103	107	106	109	112	103	116	116	111	114	109	106	112	98	137	133	103	101	101	121	111	100	96	97	93	81	78	75

122	112	117	113	114	106	111	97	93	88	84	89	85	75	70	65	55	60	52	51	46	42	49	35	28	25	22	20
126	124	126	118	127	119	124	110	100	108	100	84	81	70	78	64	69	61	61	67	55	51	49	41	38	34	30	18
138	138	0	134	129	0	124	112	0	120	105	0	100	88	87	78	0	74	77	0	57	59	0	48	48	0	38	27
135	131	130	137	128	130	121	119	121	119	110	106	97	90	88	84	88	88	80	76	63	62	67	52	48	45	35	29
138	136	133	137	0	126	127	121	143	0	124	111	100	88	101	87	91	105	0	76	69	67	71	0	52	48	39	33
133	136	0	133	128	128	132	135	137	124	136	0	116	104	116	103	0	92	80	70	74	71	68	54	47	0	39	35
141	138	136	132	123	127	130	140	124	123	118	124	103	100	118	100	105	97	90	77	82	68	66	56	62	49	44	39
141	144	158	145	139	139	126	0	126	124	129	116	103	106	106	105	102	96	94	84	83	0	82	63	58	58	46	39
136	158	0	153	158	147	138	146	145	121	136	0	117	113	111	107	0	104	96	94	81	77	76	65	64	0	55	34
137	156	149	158	0	157	138	146	141	0	132	131	124	116	117	118	113	118	0	94	78	78	77	0	67	59	50	41
148	145	161	144	158	149	138	140	133	141	136	127	123	131	131	115	114	108	99	106	93	68	82	69	65	62	52	40
136	154	0	147	141	0	153	139	0	140	129	0	136	115	130	132	0	118	114	0	98	83	0	76	80	0	52	41
143	153	156	148	147	149	139	128	128	122	125	127	126	105	117	117	127	104	115	105	93	91	88	79	61	62	52	40
152	156	145	131	154	143	138	130	126	123	119	123	114	109	126	116	117	103	108	103	103	89	84	70	60	54	52	41
152	141	144	126	136	122	123	117	118	119	112	116	112	99	126	115	113	110	102	111	96	91	88	73	67	58	54	37
165	140	119	117	129	117	119	117	112	123	106	106	115	107	112	117	122	113	105	112	92	90	88	71	64	63	48	38
137	126	0	120	121	0	102	105	0	102	113	0	104	102	122	117	0	112	120	0	90	90	0	75	73	0	58	46
131	126	117	121	132	123	118	122	119	120	113	100	101	114	114	121	105	112	114	100	99	92	91	81	69	66	56	48
142	125	118	127	0	124	130	128	133	0	118	98	114	107	125	126	107	106	0	107	95	98	88	0	73	70	58	45
142	123	0	113	129	138	132	123	128	135	111	0	124	126	110	121	0	123	125	104	96	76	86	82	75	0	54	46
146	130	127	117	117	130	132	123	125	120	113	112	119	120	122	115	124	123	109	99	96	86	87	71	71	65	60	42

Fig. 12. Relative Pin Power Density of Outer Assemblies in Percent

from the pin powers of the same rod in four segments. The averaged relative assembly power was calculated from the averaged pin powers. Figures 10 and 11 show the averaged relative assembly power distributions together with that from FSAR[7] of the Kori Unit 1. Figures 10 and 11 are the ROOM and CORE cases, respectively.

Maximum relative error of the CORE case shows −5.735% and relative root mean square error gives 2.159%. Maximum relative error of the ROOM case, −8.864%, is larger than that of the CORE case. The root mean square error of the ROOM case gives 4.468%. Figure 12 shows the axially averaged pin power distributions of the outer assemblies, which can be used as a source term for the reactor vessel fluence calculation.

7. Conclusions

The full-scope Monte Carlo modeling of the Kori Unit 1 was established by using MCNP code. The modeling was confirmed by the criticality and the relative power distribution calculations. The *k* values approximated to unity in the MCNP criticality calculations. The MCNP criticality calculation with the core temperature library gives better *k* value than that with the room temperature library. The relative assembly power distributions with the core temperature library are also more accurate than that with the room temperature library. The root mean square errors of the relative assembly power distributions are less than 5% in the two cases, ROOM and CORE. The results of the criticality and the relative assembly power distribution show that the full-scope MCNP modeling of this study is well constructed.

A usage of the flux suppression fixture in support of vessel life extension is recently issued.[11] In such case, the pin-by-pin power distributions due to changes of the reflective regions have to be newly calculated for the pressure vessel fluence calculation. It is expected that the full-scope Monte Carlo simulation will be a useful tool in the power distribution calculation.

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