

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 neu-

tron 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, °F	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, °F	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, °F	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 Material	Stainless steel 304
Boron Loading(wt. % B ₂ O ₃ in Glass Rod)	12.5
Baffle	
Thickness, cm	2.8575
Material	Stainless Steel

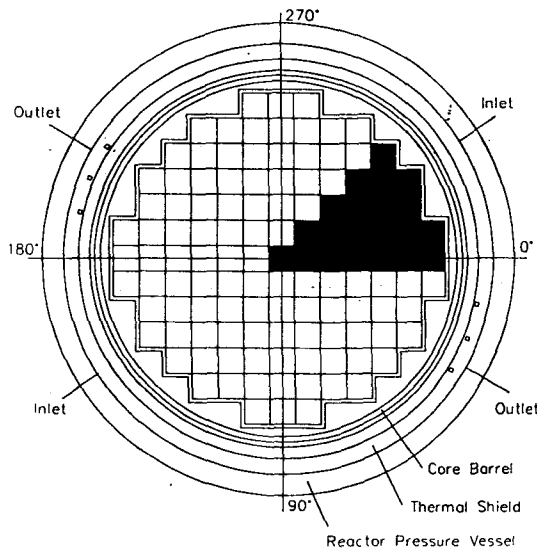


Fig. 1. Cross Sectional View of Kori Unit 1 Reactor

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.

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

Table 2. Material Compositions of the Kori Unit 1 Structure

Structure	Composition	Number Density(#/cm-barn)
$\text{UO}_2(2.10\text{w/o})$	U^{235}	4.84371×10^{-4}
	U^{238}	2.22957×10^{-2}
	O	4.55524×10^{-2}
$\text{UO}_2(2.83\text{w/o})$	U^{235}	6.52748×10^{-4}
	U^{238}	2.21294×10^{-2}
	O	4.55524×10^{-2}
$\text{UO}_2(3.20\text{w/o})$	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}

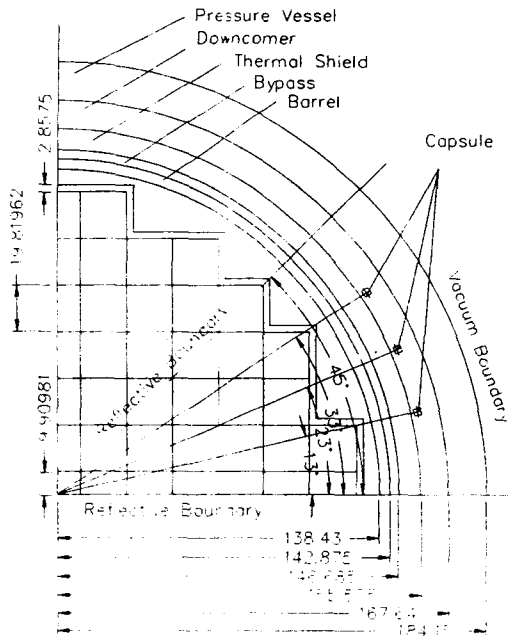


Fig. 2. The Cross Sectional View of Kori Unit 1 MCNP Model

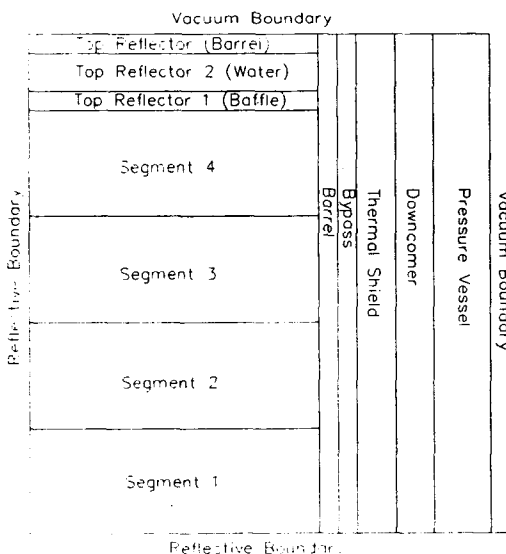
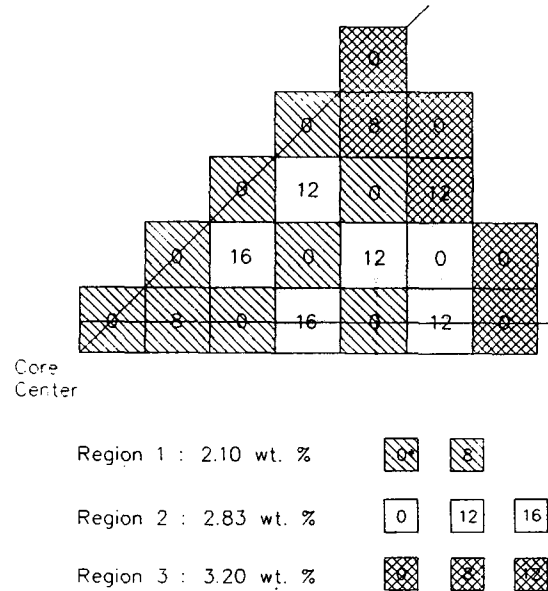


Fig. 3. The Axial View of Kori Unit 1 MCNP Geometry Model

from the core depletion because of fresh fuel. The fuel loading pattern used in this work is shown in



*Number Indicates Number of Burnable Poison Rods

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

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 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,

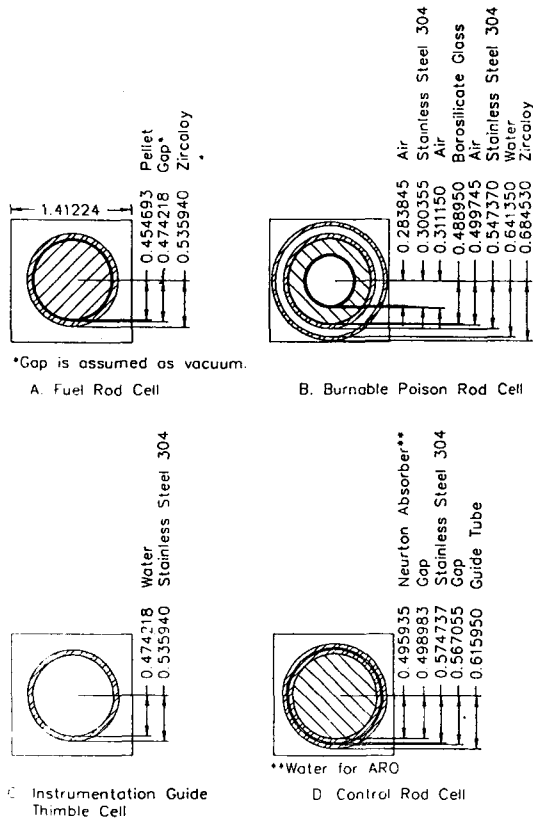


Fig. 5. Various Rods in MCNP Modeling

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.

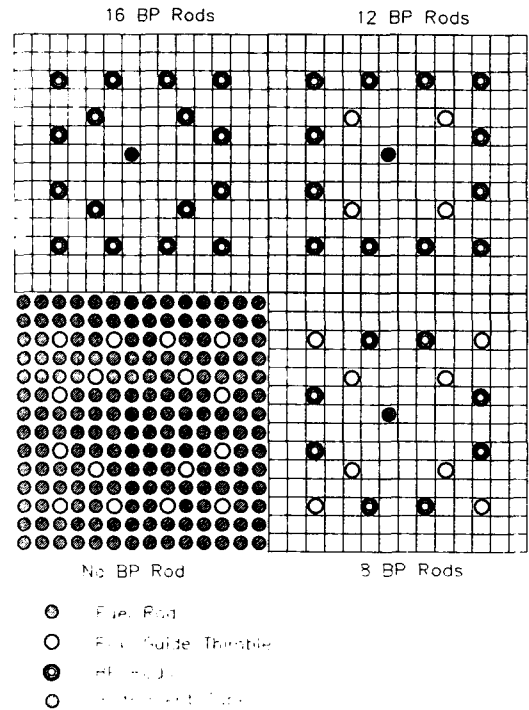


Fig. 6. Burnable Poison Rods, Rod Cluster Control Guide Thimbles, and Instrumentation Thimble within An Assembly

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 4eV accu-

rately. 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/cm²·sec·MeV,

N_{235} = atomic number density of ²³⁵U,
atoms/cm³,

N_{238} = atomic number density of ²³⁸U,
atoms/cm³,

$\sigma_f^{235}(\vec{r}, E)$ = microscopic fission cross section of ²³⁵U,
cm²/atom,

$\sigma_f^{238}(\vec{r}, E)$ = microscopic fission cross section of ²³⁸U,
cm²/atom,

and V = tally cell volume, cm³.

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

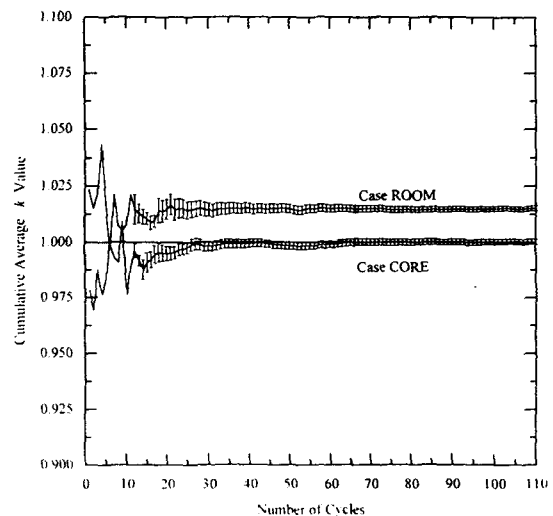


Fig. 7. Cumulative Average k Values of the Kori Unit 1 Core in KCODE Mode

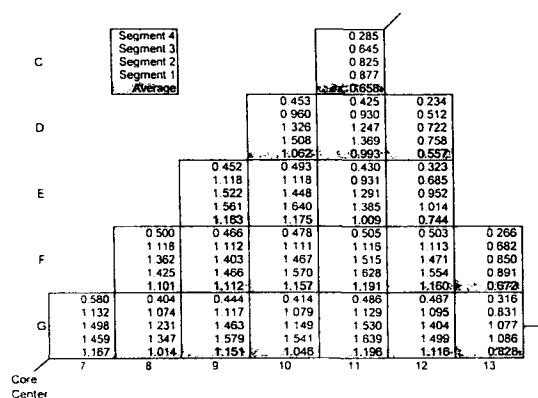


Fig. 8. Relative Assembly Power Density Distribution of 4 Segments in the ROOM Case(Near Beginning of Life Unrodded Core, Hot Full Power, No Xenon)

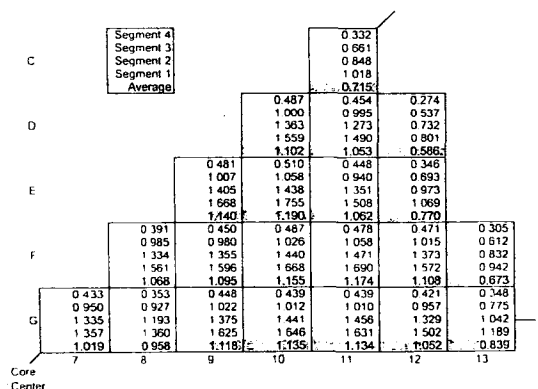


Fig. 9. Relative Assembly Power Density Distribution of 4 Segments in the CORE Case(Near Beginning of Life Unrodded Core, Hot Full Power, No Xenon)

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 $1.01556 \geq 0.00084$ and $1.00039 \geq 0.00084$, 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

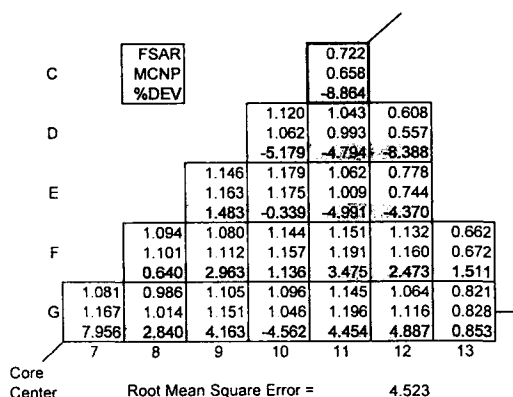


Fig. 10. Axially Averaged Power Density Distribution in the ROOM Case(Near Beginning of Life Unrodded Core, Hot Full Power, No Xenon)

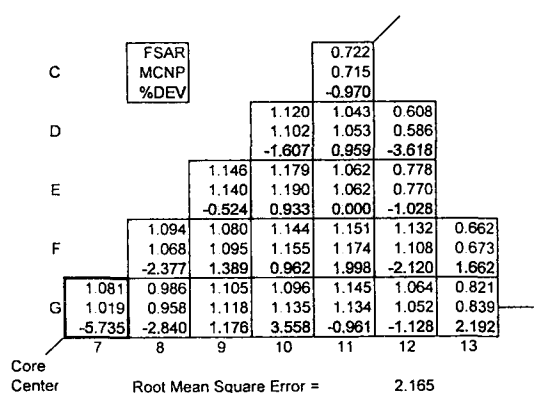


Fig. 11. Axially Averaged Power Density Distributions in the CORE Case(Near Beginning of Life, Unrodded Core, Hot Full Power, No Xenon)

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

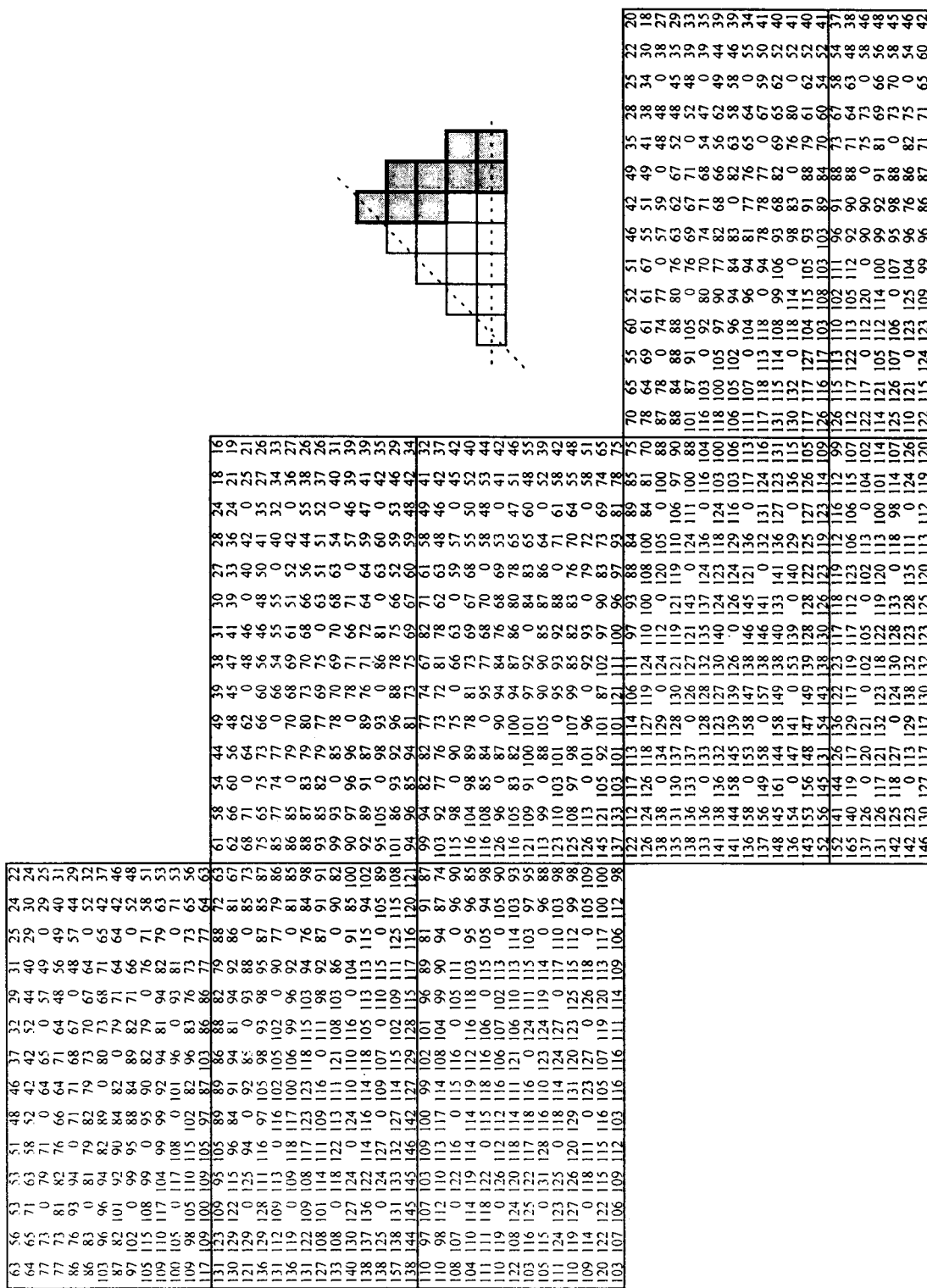


Fig. 12. Relative Pin Power Density of Outer Assemblies in Percent

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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