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《Original》

A Three-Dimensional Simulation of Kori-1 Core by Nodal Method

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Abstract

The KINS (KAERI-Improved Nodal Simulation) program, a three-dimensional nodal simulation code for pressurized water reactors, has been developed and benchmarked against the first cycle of the Kori-1 reactor. The KINS program is based on the computational model used in FLARE code and has been modified to represent the PWR characteristics more explicitly. The critical boron concentration and three-dimensional power distribution at the beginning of life hot zero power have been calculated and compared with the operating data. A three-dimensional depletion calculation at the intervals of 1000 MWD/MTU burnup steps has been performed. As the result of comparison, our calculation is shown to be in excellent agreement with the operating data. It is displayed that, incorporated with the computing time, the KINS program is an effective and powerful tool for PWR core management.

요 약

가압경수로심의 3차원적 simulation 코드인 KINS를 개발하여 고리1호기 제1주기에 대한 benchmark 계산을 수행하였다. KINS는 FLARE에서 사용하고 있는 모델을 기초로 하여 가압경수로심 해석에 보다 유용하게 쓸 수 있도록 발전시킨 것이다. 제1주기초에서는 hot zero power 상태에서의 임계붕소농도, 핵연료집합체별 출력분포, 노심평균축방향 출력분포 등을 계산하여 실측자료와 비교하였다. 아울러 연소도 1000MWD/MTU 단위로 연소계산을 수행하여 여기서 산출된 임계붕소농도와 핵연료집합체별 출력분포를 실측자료와 비교하였다. 계산결과는 실측자료와 매우 훌륭하게 일치하고 있으므로 KINS 가 가압경수로의 노심관리에 아주 경제적이며 유효한 도구가될 것임을 보여주는 것이라고 생각된다.

I. Introduction

The economic and safe operation of a nuclear power reactor requires accurate simulation of the three-dimensional behavior of the actual core. The finite-difference representation of the multi-group diffusion equation is the most straightforward approach to the solution of space-time problems. However, since this pproach usually requires a large computer storage and long computing time for multi-dimonsional problems, a variety of methods have been developed so far in an attempt to obtain an economical, with good accuracy, spatial approximation.

The nodal method for reactor analysis was first employed in the FLARE code[1]. It has

been used successfully for numerous operating reactors since then, including the application for the Big Rock Point[1] and Yankee[2] reactors. Because the FLARE code was developed initially for the boiling water reactor simulation, the modified versions of the FLARE, which can be used more easily for the pressurized water reactor simulation, have been developed by several companies for their own use, such as FLAME-3[3] of Babcock and Wilcox, NUSIM-3[4] of NUS, and TRILUX[5] of Belgo-Nucleaire.

The KINS program[6], developed by the Korea Atomic Energy Research Institute for three-dimensional simulation of PWR core, also utilizes the FLARE model. The computational model is based on a modified one-group diffusion theory, in which only infinite multiplication factor (K_∞) and neutron migration area (M^2) of each region are involved instead of group-dependent neutron cross-sections. The leakage of neutrons from each region is determined from the transport kernel which is calculated from the neutron migration area and the linear dimensions of the region, and the leakage of neutrons at the core-reflector interface is accounted for by albedos which effectively replace the reflectors.

The neutron source at each node is calculated as a function of infinite multiplication factor at that node, the source at the six neighboring points, and a transport kernel. The migration area which is used for the calculation of transport kernel is calculated in turn at each node based on a fit to the moderator-coolant temperature. The k_∞ at each node is calculated, resulting in the space-dependent effects of control rod positions, moderator-coolant temperature, equilibrium or time-dependent xenon, Doppler coefficient of reactivity, soluble poison concentration, and the fuel burnup. The major modifications taken place so far on the KINS program are as follows:

1. Nuclear properties for each fuel type are input by fits of infinite multiplication factor and neutron migration area to moderator-coolant temperature;

2. Relations representing reactivity changes due to Doppler effect, xenon effect and soluble boron concentrations are modified to give more realistic results;

3. In the Westinghouse PWR's the burnable poison rods are generally removed from the core upon the end of the first cycle. When the burnable poison rods are removed from an assembly, it becomes identical to the assembly with the same enrichment but no burnable poison rods, except for chronological effects. An entry is provided to account for this effect;

4. Minimum DNBR calculation has been carried out using W-3 correlations[7].

The KINS program has been benchmarked against cycle 1 of the Kori-1 reactor which is now in its second cycle.

Ⅱ. Geometry Description

The Kori-1 reactor core design parameters for the first cycle are specified in Table 1.

Table 1. Kori-1 Core Design Parameters*

Assembly Geometry and Composition			
	Region1	Region2	Region3
Enrichment(w/o U-235)	2.122	2.835	3.199
No. of Fuel Pins	179	179	179
Fuel Pin Pitch (in.)	0.556	0.556	0.556
Fuel Stack Height (in.)	144	144	144
Spacer Grid Material	Inconel-718	Inconel-718	Inconel-718
Rod Fill Gas	He	He	He
Gas Pressure(psia)	450	450	450

UO_2 Density Fraction	0.9431	0.9427	0.9464

Fuel Rod Geometry and Composition

Fuel Material	Sintered UO_2
Cladding Material	Zircaloy-4
Pellet O.D. (in.)	0.3659
Cladding I.D. (in.)	0.3735
Cladding O.D. (in.)	0.422

RCC Guide Tube Geometry and Composition

Guide Tube Material	Zircaloy-4
Guide Tube I.D. (in.)	0.505
Guide Tube O.D. (in.)	0.539

Burnable Poison Rod Geometry and Composition

Cladd Material	SS-304
Rod Material	Borosilicate Glass
Inner Clad I.D. (in.)	0.2235
Inner Clad O.D. (in.)	0.2365
Glass I.D. (in.)	0.245
Glass O.D. (in.)	0.385
Outer Clad I.D. (in.)	0.3835
Outer Clad O.D. (in.)	0.431

* All Dimensions are cold.

In most respects, the Kori-1 core design parameters are typical for a PWR of the same capacity. The fuel assembly has a 14 ×14 array structure including 16 rod cluster control (RCC) thimbles and an instrumentation thimble. The initial fuel is loaded in the normal checkerboard-type interspersal formation, and its loading is octantsymmetric. There are eight distinct fuel assembly types according to the fuel enrichment and the number of burnable poison rods present in the assembly as depicted in Figure 1. Although the fuel loading is octantsymmetric, the calculation was performed assuming quarter core symmetry so that the resulted three-dimensional burnup distributions in the fuel assemblies may be used for the succeeding cycles.

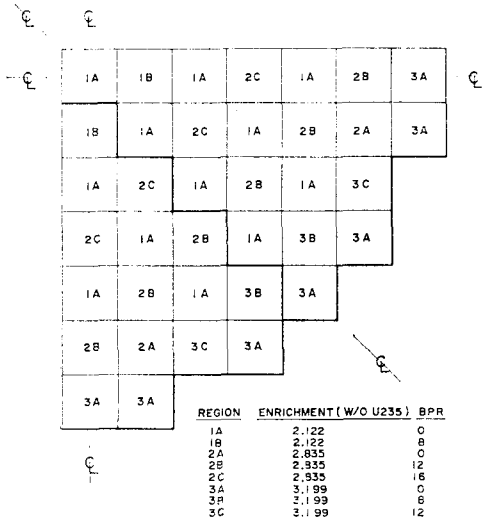

Fig. 1. Kori-1 Cycle Loading Pattern

Ⅲ. Benchmark Calculation Procedure

The schematic diagram for benchmark calculation procedure is presented in Figure 2. THERMO[8] code is used to calculate the temperatures of fuel pellet and the cladding for use in the lattice cell program NUMICE-

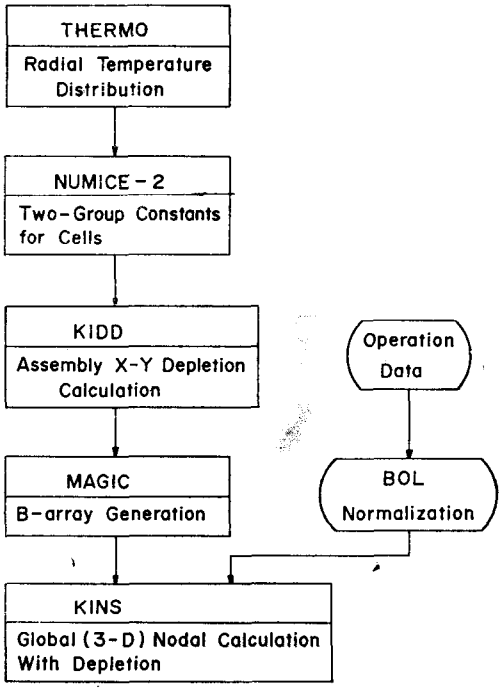

Fig. 2. Calculation Flow Diagran

2[9]. The physical model of THERMO to compute temperature distribution in a fuel rod is very simple, because only the heat transfer in radial direction is assumed herein.

The neutron spectrum calculation in the NUMICE-2 is based on the MUFT[10] -SOFOCATE[11] model as modified by Arnold[12] and again by Strawbridge[13]. The isotopics at the beginning of life and the burnup-dependent neutron cross-sections for fuel cells of each enrichments are obtained from pure cell edit of NUMICE-2 supercell calculations. The supercell must represent four separate regions: the fuel pellet, the cladding (including pellet-to-clad gap), the moderator (including homogenized spacer grid material), and an extra region to account for those lattice cells within the fuel assembly which do not contain fuel. The NUMICE-2 calculations for non-fuel cells are also performed to generate the neutron cross-sections in those regions.

The KIDD program[14], which were previously developed by the Korea Atomic Energy Research Institute for reactor analysis, is a diffusion-depletion code. It can handle up to five neutron energy groups and solve the diffusion equation up to three-dimensions. The two-dimensional KIDD assembly calculations with one mesh per cell side supply the data for the determination of many of the constants in the so-called B array of the nodal code, KINS. Since the KINS program typically uses one node per assembly at each axial position, the B-constants must represent averages over the assembly as a whole.

Consequently, KIDD calculations must be performed for each distinct assembly type in the core. For a PWR, each unique assembly type can be specified by the fuel enrichment, the number of burnable poison rods present in the assembly and, optionally, the presence of RCC. Since a PWR is generally operated with all control rods fully withdrawn position, the fuel type was specified by the enrichment and the number of burnable poison rods. Because Kori-1 has eight distinct fuel assembly types in the core as shown in Figure 1, the two-dimensional KIDD calculations were performed for these assembly types.

The entries in the B-array describe the behavior of certain parameters over the range of conditions that the fuel assembly will encounter during its residence in the core. These parameters are divided into two categories: moderator temperature-dependent parameters and fuel burnup-dependent parameters. The B-constants for temperature-dependent parameters were extracted from the two-dimensional KIDD BOL calculations at three moderator temperatures which span the range of in-core temperatures. The moderator temperatures of inlet (541.2°F), core average (576.9°F) and a value near saturation temperature were chosen in these calculations. The temperatures of all components except the moderator were given as their core average BOL hot full power (HFP) values. An additional BOL calculation, based on a uniform assembly temperature equal to the BOL hot zero power (HZP) moderator temperature, provides data for the calculation of the Doppler defect in KINS program. The B-constants for fuel burnup-dependent parameters were extracted from a two-dimensional KIDD depletion calculation with all component temperatures at their HFP core average values.

The MAGIC program[15], basically a polynomial regression program, was actually used to calculate the entries in the B-cons-

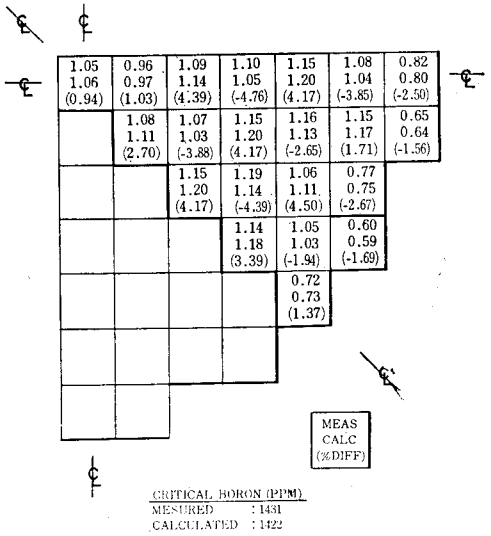

Fig.3. Assemblywise Power Distribution at BOC1, HZP Condition

tants. The B-constants are calculated from the least-square fitting of an overdetermined set of equations, and they appear in the printout.

The kernel parameters, g_v and g_h, as well as the albedos in the KINS program, are generic to the various reactor types. The horizontal kernel parameter, g_h, and albedos were determined through the comparison between two-dimensional KINS quarter core calculation and experimental data. These parameters were adjusted until a two-dimensional KINS calculation has given approximately the same assemblywise power distribution as the experimental data. The vertical albedos and g_v were also adjusted to account for leakage from the top and the bottom of the core.

Ⅳ. Results and Discussions

The KINS benchmark calculations against the first cycle of the Kori-1 reactor were performed in two steps. To begin with, a comparison between KINS quarter core

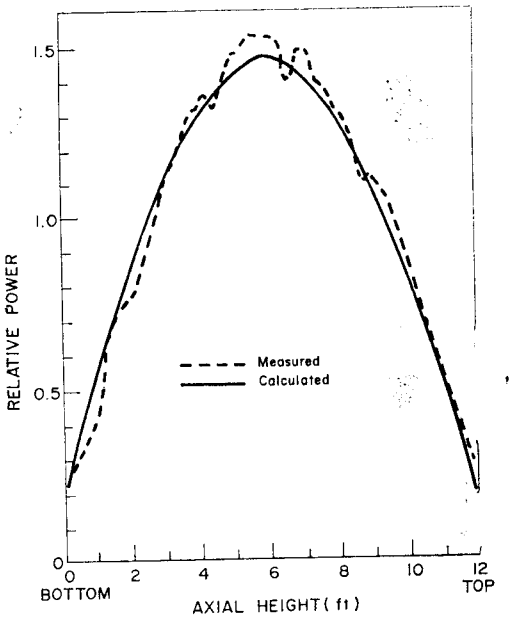

Fig. 4. Kori-1 Axial Power Distribution at BOC1, HZP Condition

calculation result and the operating data was made at the beginning of life and hot zero power condition. Since a large number of experiments are conducted under this condition, the operating data are the most accurate and the best known. The critica

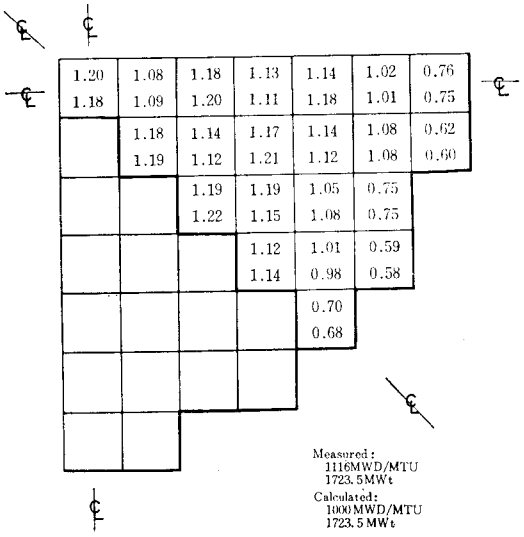

Fig. 5. Kori-1 Cycle 1 Assemblywise Power Distribution at 1000 MWD/MTU

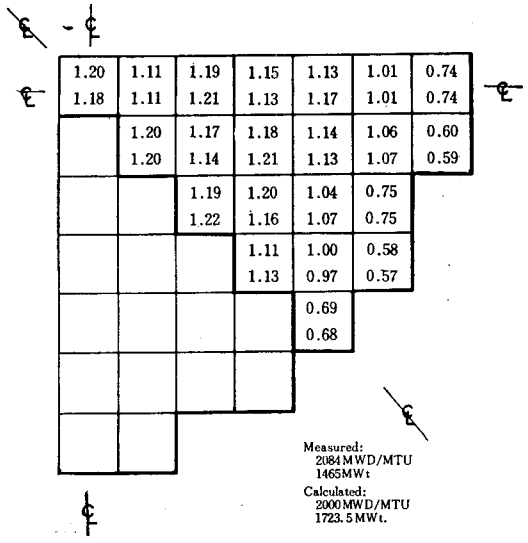

Fig. 6. Kori-1 Cycle 1 Measured and Calculated Assemblywise Power Distribution at 2000 MWD/MTU

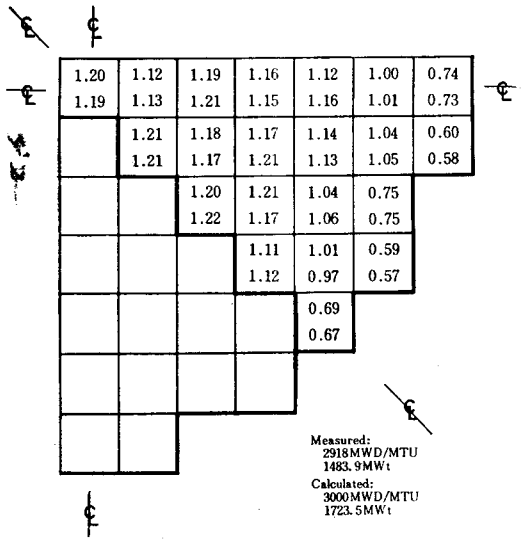

Fig. 7. Kori-1 Cycle 1 Measured and Calculated Assemblywise Power Distribution at 3000 MWA/MTU

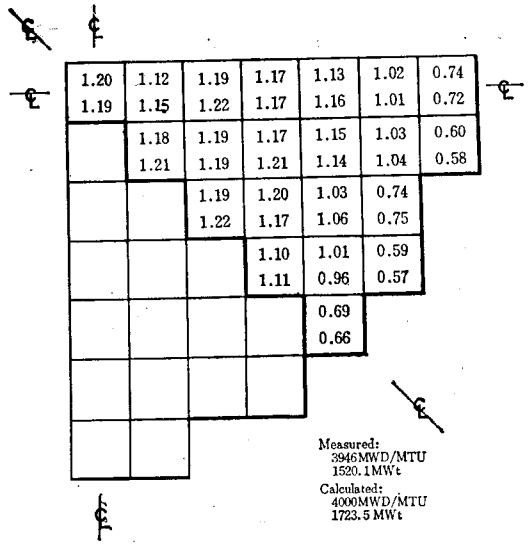

Fig. 8. Kori-1 Cycle 1 Measured and Calculated Assemblywise Power Distribution at 4000 MWD/MTU

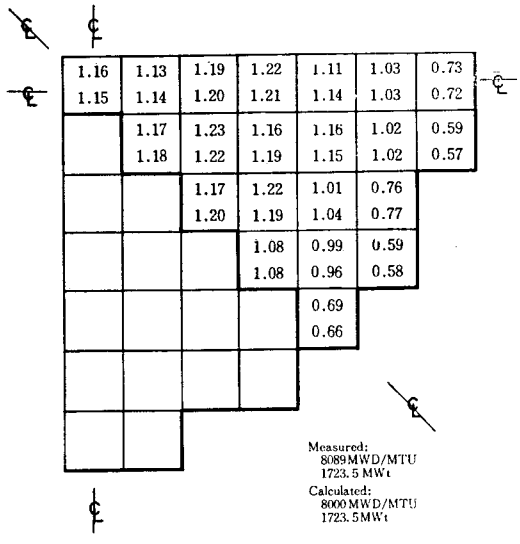

Fig. 9. Kori-1 Cycle 1 Measured and Calculated Assemblywise Power Distribution at 8000 MWD/MTU

boron concentration of 1422 ppm calculated by the KINS program displays an excellent agreement with the measured boron concentration of 1431 ppm.

Fgure 3. shows the axially averaged assemblyiwise power distribution under this condition. Relatively low power in the fuel assemblies which contain burnable poison rods calculated by the KINS program suggests that the burnable poison rod worth be

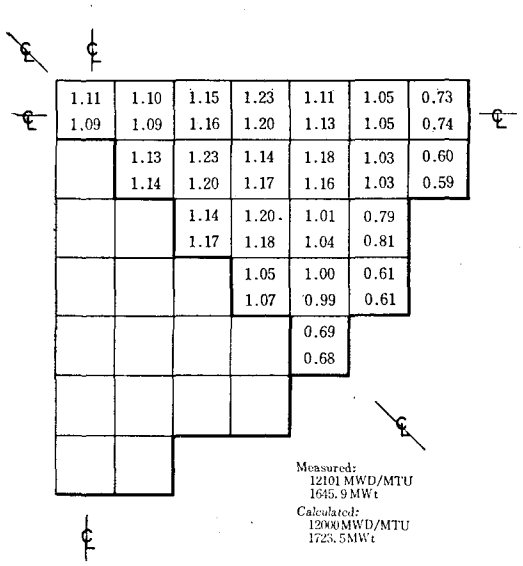

Fig. 10. Kori-1 Cyclel Assemlbywise Power Distribution at 12000 MWD/MTU

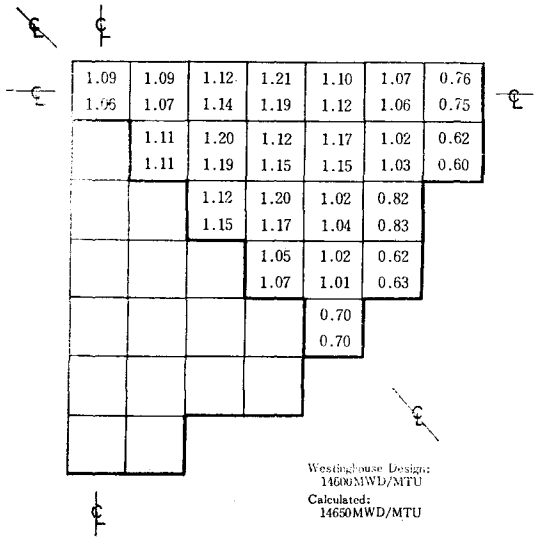

Fig. 11. Kori-1 Cycle 1 Assemblywise Power Distribution at End of Cycle 1

overestimated in the KIDD calculations, which in turn thrusts up the powers in neighboring fuel assemblies. Even so, considering the fact that the measured data itself has a certain range of uncertainty,

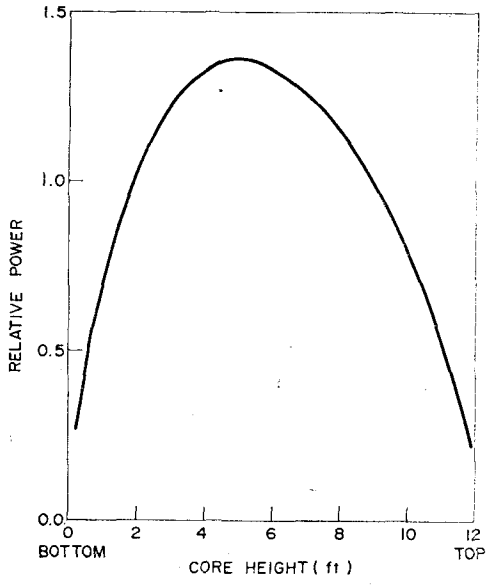

Fig. 12. Kori-1 Cycle 1 Axial Power Trace at 150 MWD/MTU

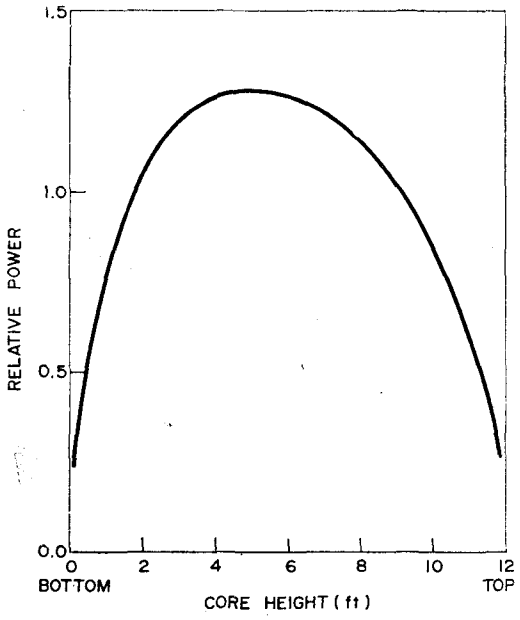

Fig. 13. Kori-1 Cycle 1 Axial Power Trace at 1000 MWD/MTU

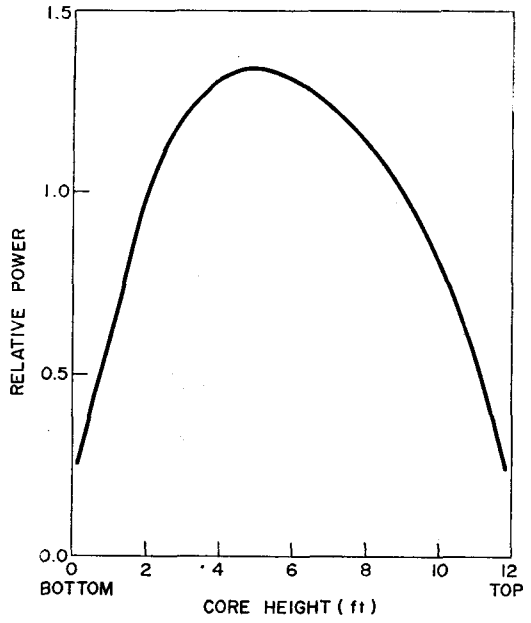

Fig. 14. Kori-1 Cycle 1 Axial Power Trace at 2000 MWD/MTU

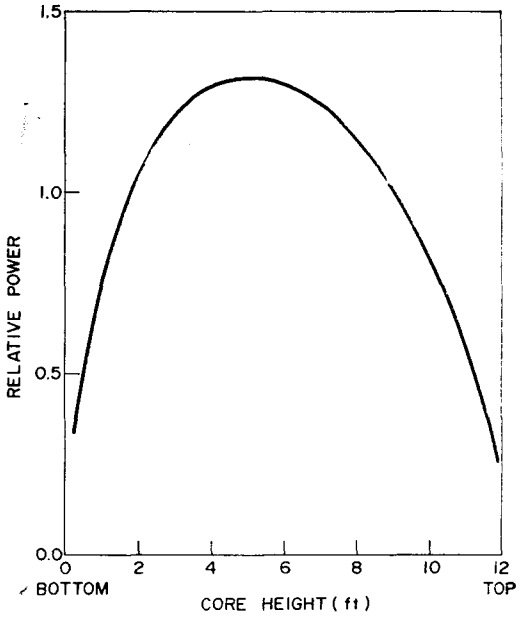

Fig. 16. Kori-1 Cycle 1 Axial Power Trace at 4000 MWD/MTU

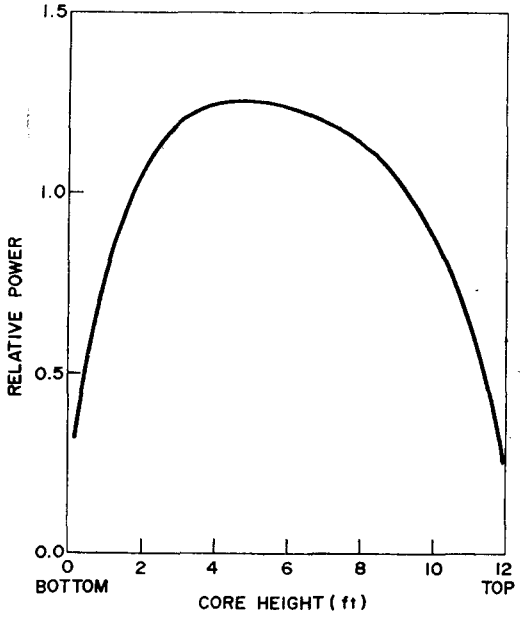

Fig. 15. Kori-1 Cycle 1 Axial Power Trace at 3000 MWD/MTU

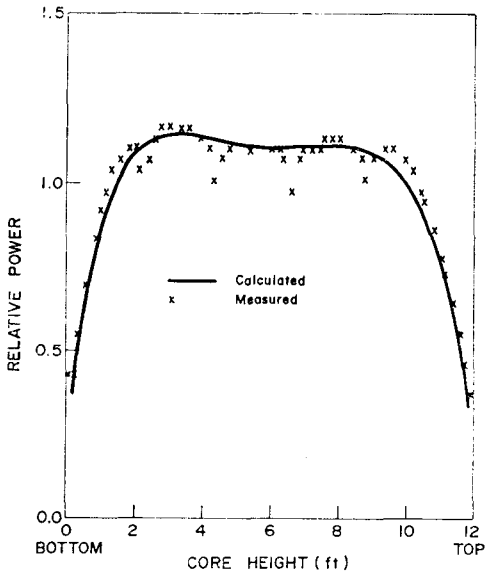

Fig. 17. Kori-1 Cycle 1 Axial Power Trace at 8000 MWD/MTU

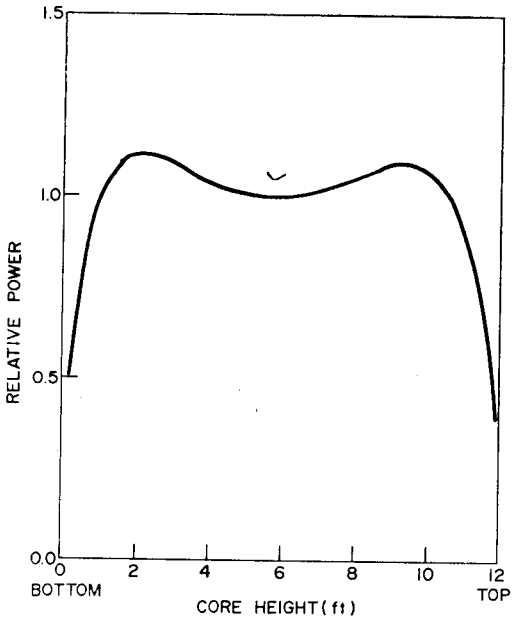

Fig. 18. Kori-1 Cycle 1 Axial Power Trace at 12000 MWD/MTU

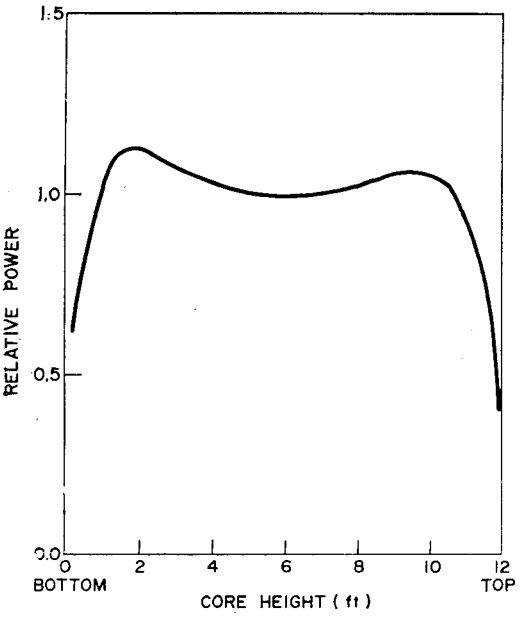

Fig. 19. Kori-1 Cycle 1 Axial Power Trace at 14650 MWD/MTU

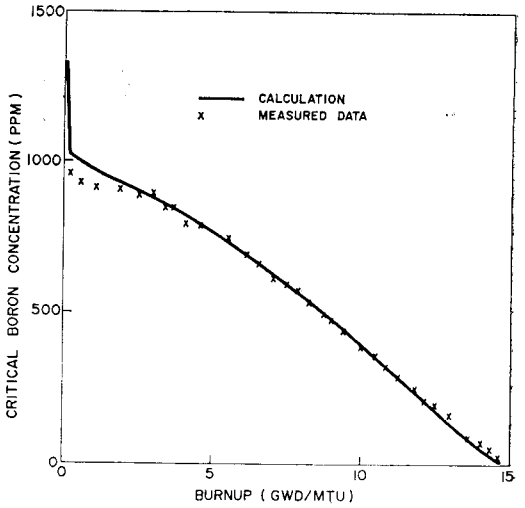

Fig. 20. Kori-1 Cycle 1 Boron Letdown Curve

the maximum deviation of 5% in power distribution may be considered as reasonably preferable. Figure 4 represents the core-averaged axial power distribution at the same condition. Although the KINS calculation cannot show the spacer grid effect explicitly, the core-averaged axial power distribution calculated by the KINS program proves to be in an excellent agreement with the operating data. This comparison verifies that the values chosen for the albedos and the horizontal and axial kernel parameters in the KINS program are appropriate.

Secondly, the KINS depletion calculation was made for the first cycle of Kori-1, and the results were compared with the operating data. Comparisons between the axially averaged assemblywise calculated and experimental relative powers for the first cycle of Kori-1 reactor are presented in Figures 5 through 11. Agreement falls within a few-per-cents throughout the cycle, and that agreement turns out to be generally improving with fuel burnup.

The core-averaged axial power traces calculated by the KINS program for the first

cycle of Kori-1 are presented in Figures 12 through 19. These curves exhibit the progressive axial flattening and double-hump peaking that conventionally occurs with burnup.

The experimental and calculated critical boron let-down curves for the first cycle are shown in Figure 20. Regarding the fact that the measured boron concentration has an uncertainty of ±50 ppm, the figure shows the excellent agreement with the operating data. This emphasizes that the KINS program is extremely useful for predicting the cycle length of PWR's.

V. Conclusions

The KINS program is a three-dimensional nodal simulation code for PWR's which has been developed and benchmarked against the first cycle of the Kori-1 reactor. The calculated results throughout the entire cycle are found to be well agreeable with the operating data with respect to power distributions and critical boron concentrations, such that the maximum errors fall within the range of 5% in the case of the former, while 4% for the latter, respectively. The computing time required for the three-dimensional KINS quarter core calculation for a single time step is reckoned up to be approximately 70 seconds by CDC-6400. The calculational accuracy, coupled with much less computing time, proves that the KINS program is an extremely economic and effective tool for PWR core analysis so as to be widely applicable for the practical calculation.

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