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Power Density Distribution Calculation of a Pressurized Water Reactor with Fullscope Explicit Modeling by MCNP Code

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Abstract

Power density distribution and criticality of a pressurized water reactor are calculated with a Monte Carlo calculation using the MCNP code. The MCNP model is based on one-eighth core symmetry. Individual fuel assemblies are modeled with fullscope three dimensional description except grid spacer. The fuel rod is divided into eight axial segments. Core internals above and below the active fuel region is represented as coolant. After 400 cycle calculations, the system converges to a k value of 1.09151 ±0.00066. Fission reaction rate in each rod is also calculated to use as the source term in pressure vessel fluence calculation.

1. Introduction

Pressure vessel neutron fluence calculations are a key to the approval of plant life extension for pressurized water reactors. Conventional computation of pressure vessel neutron fluence employs spatial combination of two dimensional calculations using deterministic codes such as the discrete ordinates transport code. The neutron source in the conventional computation is calculated separately using depletion analysis codes and nodal diffusion codes. In the conventional calculations, the spatial distribution of the fission neutron source in each assembly is assumed as constant. Each fuel assembly is homogenized by volume fraction of the fuel pellet, cladding material, absorbing rods, and core coolant.

In earlier studies[1] the constant power distribution in the outer assemblies causes an error of 15 to 25% in the pressure vessel flux. Futhermore, the fast neutron flux of homogeneous model based on the volume fraction could be underestimated in core shielding calculations because the calculation method takes no account of the reaction rate conservation.[2]

The fullscope explicit three dimensional modeling is necessary to take account of the pin power distribution in the outer assemblies and heterogeneity of assemblies.

2. Computational Modeling

The data used for this study come from Kori Unit 1 (Kori-1), a pressurized water reactor. The Kori Unit 1 reactor core geometry model input into MCNP4A is a three dimensional representation of one-eighth of the reactor in-vessel components with reflective angular boundaries at 0 and 45 degees. Fuel rods, burnable poisons, instrumentation thimbles, baffles, barrel, thermal shield, and pressure vessel are modeled explicitly as like Figure 1. Core internals above and belows the active fuel region is represented as coolant.

Material compositions are largely drawn from specifications in FSAR(Final Safety Analysis Report) of Kori-1. Fuel loading pattern of cycle 1 is modeled with all rod out(ARO), no xenon, and 1278 ppm boron concentration(near the beginning of life).

3. Results

The pin-dependent power distribution for all assemblies is generated by running MCNP4A in a k(KCODE) calculation mode. An initial source of 20,000 neutrons with Watt fission spectrum makes pin by pin source distribution in the full core. After propagating for 400 cycles, the system converged to a k value of 1.09151 ± 0.00066. Figure 2 shows that the k value converges with cycle increasing. (The k value, which is calculated by the conventional discrete ordinates code, DORT, converges to less than 0.75.)

The fission rate is calculated in each rod with volume averaged fluence, F4 and fission cross section of U^{235} and U^{238}, FM4. Figure 3 shows that the assembly power distribution which is averaged from pin power distribution.

Acknowledgement

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References

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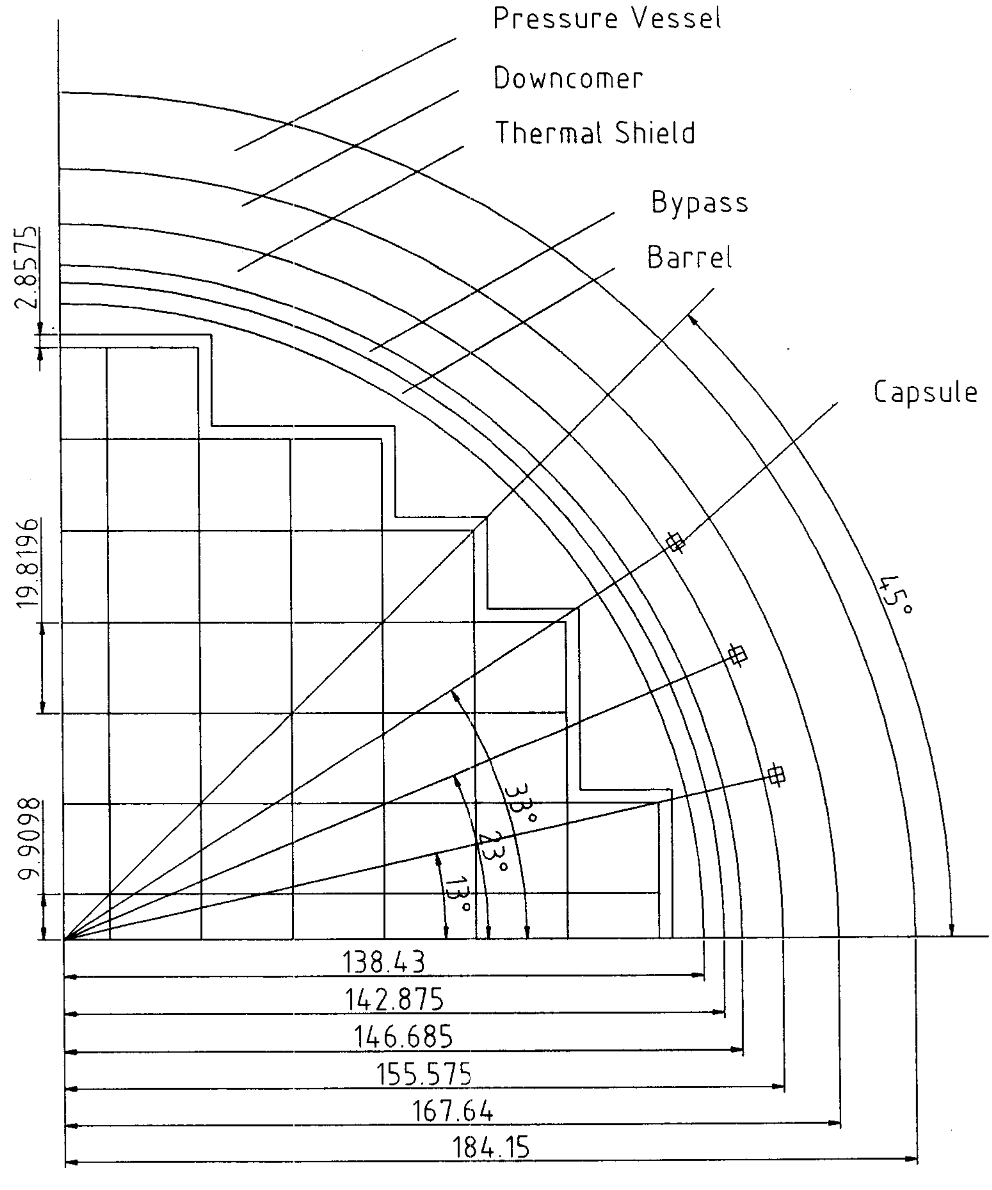

Figure 1. MCNP Modeling for Kori Unit 1

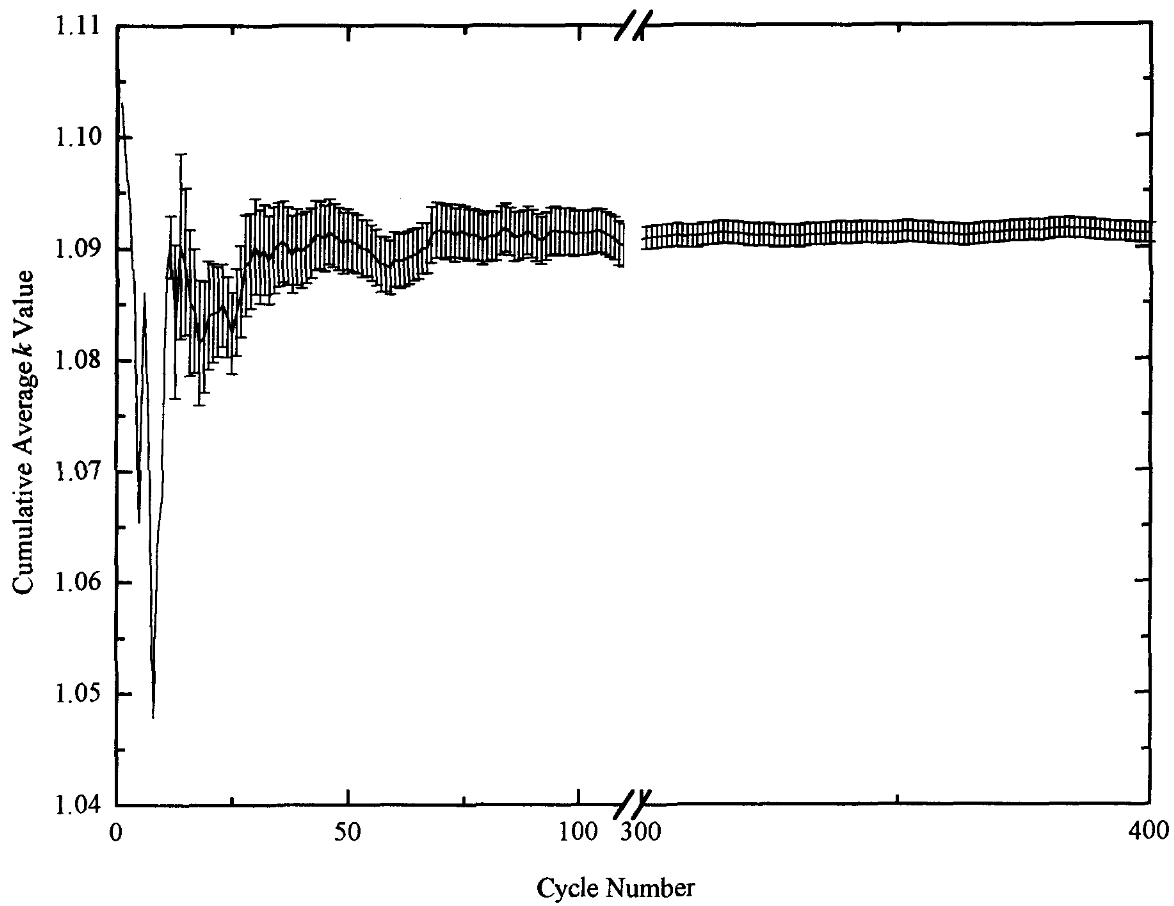

Figure 2. Cumulative Average k Values in KCODE Mode

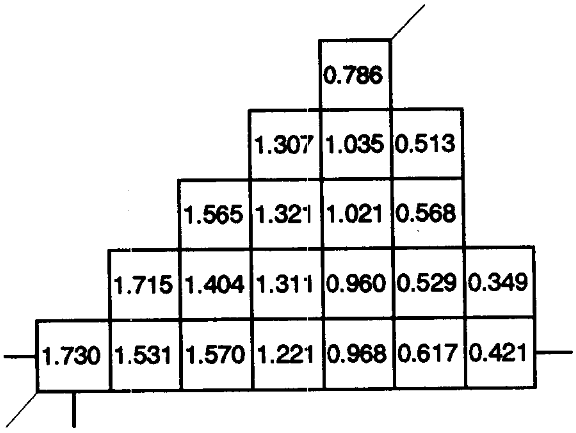

Figure 3. Normalized Power Density Distribution
Near Beginning of Life Unrodded Core, No Xenon