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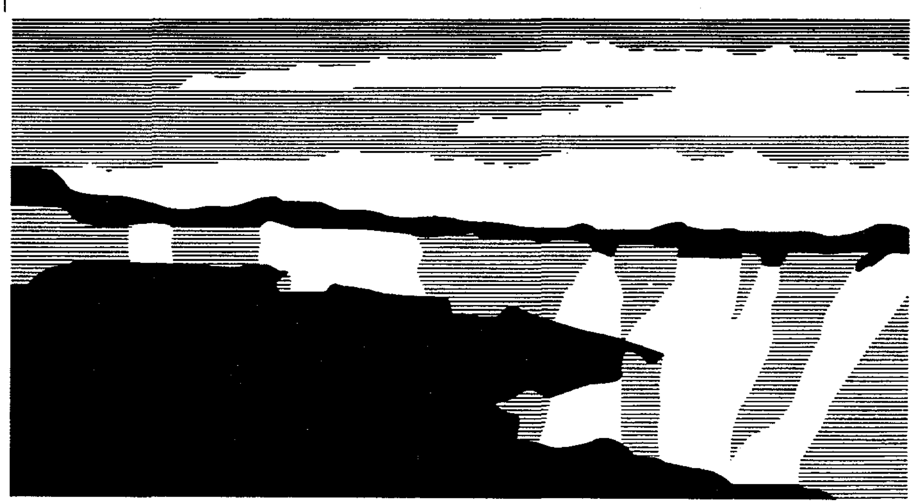

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A WIMS-NESTLE REACTOR PHYSICS MODEL FOR AN RBMK REACTOR

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ABSTRACT

This work describes the static neutronic calculations made for a three-dimensional model of an RBMK (Russian) reactor. Future work will involve the use of this neutronic model and a thermal-hydraulic model in coupled calculations. The lattice code, WIMS-D, was used to obtain the cross sections for the static neutronic calculations. The static reactor neutronic calculations were made with NESTLE, a three-dimensional nodal diffusion code. The methods used to establish an RBMK reactor model for use in these codes are discussed, and the cross sections calculated are given.

INTRODUCTION

A project to develop a coupled neutronic-thermal-hydraulic code system has begun at Los Alamos National Laboratory. The code system will couple the three-dimensional neutronic code NESTLE[1] with the thermal-hydraulic code TRAC.[2] The first reactor selected for analysis was the Russian RBMK. In anticipation of these studies, a reactor model was developed. Cross sections that are a function of moderator density, temperature, and burnup were calculated using the lattice physics code WIMS-D.[3] Static neutronic calculations were made using these cross sections. This paper describes the RBMK model developed and the static neutronic calculations made. Lists of the cross sections calculated are presented for verification or independent analysis.

REACTOR MODEL

This section describes the RBMK reactor model. This model was used for the lattice physics calculations with WIMS-D and subsequent neutronic calculations using NESTLE. First, a detailed model of each of the reactor components, i.e., fuel assembly, control rod assembly, and safety rod assembly, was developed. This detailed model of each component, individually, was used as input to the lattice physics transport code, and an eigenvalue calculation of an infinite array of the component was made. The calculations were made with a multigroup neutron cross-section library, and the resulting neutron flux was used as a weighting function to collapse the cross sections for the component into a homogenized two-group cross-section set.

This process is repeated for each component. This results in a library of homogenized two-group cross sections. Using these cross sections, a reactor model or partial reactor model may be built for input to the NESTLE code. Individual components are described next.

RBMK Assemblies

Three different assembly types were modeled: fuel cell, control rod, and control rod follower. The various models for each assembly used in the calculations are described in the following sections.

Fuel Assembly

The fuel is a water-cooled, graphite-moderated, 2.4% enriched uranium dioxide assembly. It is a square with 25-cm sides. For the WIMS-D calculation, the square is cylindricized. The outer boundary has a 14.105-cm radius. There are 18 fuel rods in 2 rings in a coolant channel. Six of the fuel rods are equally spaced at a radius of 1.6 cm, and the remaining 12 are equally spaced at a radius of 3.09 cm. The fuel rods have a 0.75-cm radius and a 0.102-cm-thick zircaloy clad. The coolant channel is composed of a pressure tube surrounding the fuel rods. The pressure tube is 0.4 cm thick, has an outside radius of 4.4 cm, and is composed of zircaloy. The fuel assembly is supported by an a 0.75-cm radius zircaloy rod in the center of the cell and thus in the center of the pressure tube. The coolant channel is contained in a graphite matrix, with an outside radius, as noted, of 14.105 cm. The base case densities and temperatures of the fuel cell are given in Table I.

TABLE I

RBMK Fuel Assembly Base Temperatures and Densities

Component and Material	Temperature, K	Density, g/cm^3
Fuel—uranium dioxide	873.0	9.69
Clad—zircaloy	588.0	5.997
Support—zircaloy	588.0	5.997
Coolant—water	557.0	0.516
Pressure tube—zircaloy, 97.5 wt %, alloyed with niobium	570.0	6.49
Moderator—graphite	873.0	1.656

Control Rod

The control rod assembly also is a square with 25-cm sides. It also was cylindricized for the WIMS-D calculation. Its geometry consists of a series of concentric circles containing coolant, boron carbide, aluminum, and zircaloy contained in a graphite matrix. Significant burnup of the poison was not expected to occur; thus, only one set of cross sections was calculated. A small ring (0.005 cm thick) of graphite containing 1.6e-06 g/cm^3 of ^{235}U was placed on the outside of the control rod in the lattice calculation to give a quasi-fission spectrum weighting to the control rod assembly. Table II gives the materials, dimensions, temperatures and densities of the control rod components.

Control Rod Follower

The control rod follower assembly is also a square with 25-cm sides. It was also cylindricized for the WIMS-D calculation. Its geometry consists of a series of concentric circles containing coolant, aluminum, and zircaloy contained in a graphite matrix. Only one set of cross sections was calculated. A small ring (0.005 cm thick) of graphite containing 1.6e-06 g/cm^3 of ^{235}U was placed on the outside of the follower in the lattice calculation to give a quasi-fission spectrum weighting to the assembly. Table III gives the materials, dimensions, temperatures, and densities of the control rod follower components.

TABLE II

RBMK Control Rod Model

Material	Outside radius, cm	Temperature, K	Density, g/cm^3
Water	2.5	343.0	1.0
Aluminum	2.7	343.0	2.7
Gas Gap	2.875	343.0	0.0
Boron Carbide	3.25	343.0	1.65
Gas Gap	3.3	343.0	0.0
Aluminum	3.5	343.0	2.7
Water	4.1	343.0	1.0
Zircaloy—97.5 wt % alloyed with niobium	4.4	343.0	6.49
Graphite	14.105	343.0	1.548

TABLE III

RBMK Control Rod Follower Model

Material	Outside radius, cm	Temperature, K	Density, g/cm^3
Graphite	3.45	343.	1.548
Aluminum	3.7	343.	2.7
Water	4.1	343.	1.0
Zircaloy— 97.5 wt % alloyed with niobium	4.4	343.	6.49
Graphite	14.105	343.	1.548

LATTICE PHYSICS CALCULATIONS

The lattice calculations were made with the multi-group WIMS-D collision probability transport code. WIMS-D is a two-dimensional code and may be used with complicated geometry. Eigenvalue calculations were made for each assembly, and the resulting multigroup neutron flux was used as a weighting function to obtain cell-averaged homogenized two-group cross sections for input for reactor core calculations. The code has the capability to determine the effects due to burnup, temperature, and density changes.

The code was imported and made operational on a SUN workstation at Los Alamos National Laboratory. A series of example problems was run to ensure that the code was operating correctly. In addition, after cross-section sets were developed using WIMS-D, the two-energy-group fuel assembly cross sections were used as input to an Sn transport code, ONEDANT,[4] and as input to NESTLE to determine if each of these two codes would return the same eigenvalue as WIMS-D. All three codes gave the same eigenvalue, 1.3496, for an infinite lattice at beginning of life (BOL), base case, clean core.

WIMS-D uses a 69-energy-group cross-section set as input. The cross sections used in the calculations were obtained from Pacific Northwest Laboratory.[5] They are an enhanced set of cross sections that were developed for use with the code. Group one of the two-group set has energy boundaries of 10.0 MeV and 0.625 eV. The lower-energy cutoff for group two is 1.0E-05 eV.

Parametric Effects

As noted above, the cross sections vary when reactor parameters such as temperature and density change. Because of power distributions, one set of assembly cross sections is not valid for the entire core. It also was noted above that in the long term, cross sections change because of fuel burnup. Because of power distributions, the change in cross sections due to burnup must also be reflected in the core model.

To account for changes due to temperature and density, the following method was used. A base case at a particular burnup step was modeled. Then one parameter such as coolant density was changed and an eigenvalue calculation made. Two-group cross sections with the density changed were obtained. Again the base model for this particular burnup step was input and then another parameter, such as temperature, was changed. Two-group cross sections with the temperature changed were obtained. This process was repeated until cross-section sets for each burnup step used were obtained for the following conditions: the base cases had a fuel temperature of 873 K, a coolant temperature of 557 K, and a coolant density of 0.516 g/cm^3. The off-base conditions were a high fuel temperature of 2300 K, a low coolant temperature of 300 K, a high coolant density of 1.0 g/cm^3, and a low coolant density of 0.04 g/cm^3. Four cross-section sets were developed for the fuel assembly, one cross-section set for the control rod and one set for the control rod follower. The fuel assembly macroscopic cross sections at xenon equilibrium are for BOL for 6000, 12,000, and 18,000 MW-days/ton burnup. Tables IV through VII gives the fuel assembly two-group macroscopic cross section for the base and off-base conditions of each burnup step. Table VIII gives the macroscopic cross sections for the control rod and control rod follower assemblies.

TABLE IV

Macroscopic Cross Sections for Fuel Assemblies at Beginning of Life and Equilibrium Xenon

Macroscopic Cross Section Type or Neutrons per fission, nu by Energy Group (gp)	Base Case Macroscopic Cross Sections, cm^{-1}	Low Coolant Temperature Macroscopic Cross Sections, cm^{-1}	High Coolant Density Macroscopic Cross Sections, cm^{-1}	Low Coolant Density Macroscopic Cross Sections, cm^{-1}	High Fuel Temperature Macroscopic Cross Sections, cm^{-1}
transport, gp 1	2.9503E-01	2.9500E-01	3.0008E-01	2.9113E-01	2.9485E-01
transport, gp 2	3.9828E-01	3.9920E-01	4.1998E-01	3.7766E-01	3.9818E-01
total, gp 1	3.2414E-01	3.2411E-01	3.3047E-01	3.1830E-01	3.2400E-01
total, gp 2	3.9987E-01	4.0132E-01	4.2383E-01	3.7833E-01	3.9975E-01
capture, gp 1	8.6662E-04	8.6656E-04	9.0155E-04	7.6838E-04	9.3584E-04
capture, gp 2	1.5619E-03	1.6636E-03	1.8097E-03	1.2607E-03	1.5436E-03
fission, gp 1	2.6568E-04	2.6520E-04	2.7204E-04	2.5358E-04	2.6362E-04
fission, gp 2	2.6042E-03	2.8270E-03	2.7840E-03	2.2548E-03	2.5653E-03
nu, gp 1	2.5532E+00	2.5536E+00	2.5638E+00	2.5407E+00	2.5546E+00
nu, gp 2	2.4308E+00	2.4306E+00	2.4307E+00	2.4309E+00	2.4308E+00
scattering, gp 2 to 1	1.1451E-03	1.1715E-03	1.0582E-03	1.2727E-03	1.1744E-03
scattering, gp 1 to 1	3.1803E-01	3.1806E-01	3.2326E-01	3.1334E-01	3.1785E-01
scattering, gp 2 to 2	3.9456E-01	3.9565E-01	4.1818E-01	3.7354E-01	3.9447E-01
scattering, gp 1 to 2	4.9713E-03	4.9210E-03	6.0318E-03	3.9403E-03	4.9520E-03

TABLE V

Macroscopic Cross Sections for Fuel Assembles at 6000 MW-days/ton Burnup and Equilibrium Xenon

Macroscopic Cross Section Type or Neutrons per fission, nu by Energy Group (gp)	Base Case Macroscopic Cross Sections, cm^{-1}	Low Coolant Temperature Macroscopic Cross Sections, cm^{-1}	High Coolant Density Macroscopic Cross Sections, cm^{-1}	Low Coolant Density Macroscopic Cross Sections, cm^{-1}	High Fuel Temperature Macroscopic Cross Sections, cm^{-1}
transport, gp 1	2.9501E-01	2.9498E-01	3.0004E-01	2.9113E-01	2.9483E-01
transport, gp 2	3.9831E-01	3.9929E-01	4.1981E-01	3.7826E-01	3.9820E-01
total, gp 1	3.2413E-01	3.2411E-01	3.3045E-01	3.1833E-01	3.2400E-01
total, gp 2	3.9991E-01	4.0144E-01	4.2369E-01	3.7891E-01	3.9978E-01
capture, gp 1	9.4965E-04	9.4897E-04	9.8670E-04	8.4332E-04	1.0215E-03
capture, gp 2	1.8494E-03	1.9422E-03	2.0953E-03	1.5629E-03	1.8348E-03
fission, gp 1	2.2882E-04	2.2850E-04	2.3572E-04	2.1630E-04	2.2702E-04
fission, gp 2	2.4296E-03	2.5984E-03	2.5682E-03	2.1739E-03	2.4047E-03
nu, gp 1	2.5992E+00	2.5996E+00	2.6094E+00	2.5874E+00	2.6007E+00
nu, gp 2	2.5284E+00	2.5176E+00	2.5220E+00	2.5451E+00	2.5312E+00
scattering, gp 2 to 1	1.1660E-03	1.1821E-03	1.0701E-03	1.3179E-03	1.1970E-03
scattering, gp 1 to 1	3.1805E-01	3.1807E-01	3.2326E-01	3.1341E-01	3.1787E-01
scattering, gp 2 to 2	3.9447E-01	3.9572E-01	4.1796E-01	3.7386E-01	3.9435E-01
scattering, gp 1 to 2	4.9004E-03	4.8580E-03	5.9626E-03	3.8617E-03	4.8731E-03

TABLE VI

Macroscopic Cross Sections for Fuel Assembles at 12,000 MW-days/ton Burnup and Equilibrium Xenon

Macroscopic Cross Section Type or Neutrons per fission, nu by Energy Group (gp)	Base Case Macroscopic Cross Sections, cm^{-1}	Low Coolant Temperature Macroscopic Cross Sections, cm^{-1}	High Coolant Density Macroscopic Cross Sections, cm^{-1}	Low Coolant Density Macroscopic Cross Sections, cm^{-1}	High Fuel Temperature Macroscopic Cross Sections, cm^{-1}
transport, gp 1	2.9500E-01	2.9497E-01	3.0002E-01	2.9112E-01	2.9482E-01
transport, gp 2	3.9845E-01	3.9952E-01	4.2017E-01	3.7822E-01	3.9832E-01
total, gp 1	3.2413E-01	3.2411E-01	3.3043E-01	3.1835E-01	3.2400E-01
total, gp 2	4.0009E-01	4.0176E-01	4.2417E-01	3.7886E-01	3.9994E-01
capture, gp 1	1.0249E-03	1.0237E-03	1.0684E-03	9.0293E-04	1.1030E-03
capture, gp 2	2.0038E-03	2.1046E-03	2.2516E-03	1.7146E-03	1.9887E-03
fission, gp 1	1.9657E-04	1.9637E-04	2.0408E-04	1.8416E-04	1.9501E-04
fission, gp 2	2.1728E-03	2.3051E-03	2.2767E-03	1.9817E-03	2.1560E-03
nu, gp 1	2.6437E+00	2.6441E+00	2.6532E+00	2.6325E+00	2.6454E+00
nu, gp 2	2.5929E+00	2.5779E+00	2.5850E+00	2.6130E+00	2.5967E+00
scattering, gp 2 to 1	1.1502E-03	1.1617E-03	1.0530E-03	1.3069E-03	1.1821E-03
scattering, gp 1 to 1	3.1806E-01	3.1807E-01	3.2325E-01	3.1344E-01	3.1788E-01
scattering, gp 2 to 2	3.9476E-01	3.9619E-01	4.1859E-01	3.7386E-01	3.9461E-01
scattering, gp 1 to 2	4.8534E-03	4.8128E-03	5.9135E-03	3.8238E-03	4.8175E-03

TABLE VII

Macroscopic Cross Sections for Fuel Assembles at 18,000 MW-days/ton Burnup and Equilibrium Xenon

Macroscopic Cross Section Type or Neutrons per fission, nu by Energy Group (gp)	Base Case Macroscopic Cross Sections, cm^{-1}	Low Coolant Temperature Macroscopic Cross Sections, cm^{-1}	High Coolant Density Macroscopic Cross Sections, cm^{-1}	Low Coolant Density Macroscopic Cross Sections, cm^{-1}	High Fuel Temperature Macroscopic Cross Sections, cm^{-1}
transport, gp 1	2.9497E-01	2.9494E-01	2.9999E-01	2.9110E-01	2.9479E-01
transport, gp 2	3.9862E-01	3.9980E-01	4.2073E-01	3.7799E-01	3.9848E-01
total, gp 1	3.2411E-01	3.2409E-01	3.3041E-01	3.1835E-01	3.2398E-01
total, gp 2	4.0032E-01	4.0215E-01	4.2489E-01	3.7862E-01	4.0015E-01
capture, gp 1	1.0800E-03	1.0785E-03	1.1284E-03	9.4666E-04	1.1620E-03
capture, gp 2	2.1200E-03	2.2359E-03	2.3716E-03	1.8229E-03	2.1030E-03
fission, gp 1	1.6969E-04	1.6958E-04	1.7723E-04	1.5744E-04	1.6847E-04
fission, gp 2	1.8927E-03	1.9917E-03	1.9666E-03	1.7572E-03	1.8830E-03
nu, gp 1	2.6890E+00	2.6893E+00	2.6979E+00	2.6788E+00	2.6906E+00
nu, gp 2	2.6517E+00	2.6350E+00	2.6437E+00	2.6721E+00	2.6559E+00
scattering, gp 2 to 1	1.1225E-03	1.1311E-03	1.0270E-03	1.2793E-03	1.1551E-03
scattering, gp 1 to 1	3.1803E-01	3.1805E-01	3.2322E-01	3.1344E-01	3.1786E-01
scattering, gp 2 to 2	3.9518E-01	3.9679E-01	4.1952E-01	3.7376E-01	3.9501E-01
scattering, gp 1 to 2	4.8303E-03	4.7895E-03	5.8895E-03	3.8075E-03	4.7896E-03

TABLE VIII

Macroscopic Cross Sections for the Control Rod and Control Rod Follower Assemblies

Macroscopic Cross Section Type by Energy Group (gp)	Control Rod Macroscopic Cross Sections, cm-1	Control Rod Follower Macroscopic Cross Sections, cm-1
transport, gp 1	2.6585E-01	2.7687E-01
transport, gp 2	4.1093E-01	3.9406E-01
total, gp 1	2.9525E-01	3.0223E-01
total, gp 2	4.1378E-01	3.9574E-01
capture, gp 1	2.1786E-03	5.2651E-05
capture, gp 2	6.7966E-03	6.5595E-04
scattering, gp 2 to 1	1.9606E-04	2.5235E-05
scattering, gp 1 to 1	2.8947E-01	2.9783E-01
scattering, gp 2 to 2	4.0679E-01	3.9506E-01
scattering, gp 1 to 2	3.6025E-03	4.3447E-03

NESTLE requires that the cross sections be input in the form of an equation. Using the macroscopic cross sections calculated, the coefficients for equations (linear in coolant density, quadratic in coolant density, linear in coolant temperature, and linear in square root of fuel temperature) were acquired by LU decomposition.

The coefficients were calculated for the transport, absorption, nu-fission, kappa-fission, up scatter, and down scatter cross section for each of the two groups. The coefficients for the equation for the number of neutrons per fission also were determined. The coefficients were obtained for each burnup condition used. The coefficients were input to the NESTLE code. A description of the NESTLE reactor model made using these cross sections follows.

NESTLE REACTOR MODEL

NESTLE is a neutron diffusion equation solver utilizing the nodal expansion method for eigenvalue problems in three dimensions. A reactor is described by creating a matrix of regular square prisms. The reactor model may be as simple as one prism, with reflecting boundary conditions simulating an infinite lattice, to as complex as a full core face that is layered so that axial changes as well as radial changes are possible. Each square prism contains a homogenized cross-section set. Thus it is possible to represent burnup axially as well as radially in the model. In the matrix, control rod cross sections may be appropriately placed in a column of prisms, and to represent those locations where the control rod has been withdrawn, the control rod follower cross sections are placed in the prisms below those prisms containing the homogenized control rod cross sections. The reactor core then may be modeled with the control rods in various states of withdrawal to produce eigenvalues of 1 and to flatten the flux. When desired, other core components such as safety rods may be added to the model. Other representations of the fuel control rods and moderator are also possible with NESTLE.

The reactor model developed for the RBMK used square prisms 0.25 x 0.25 x 0.50 m. In all cases, the prisms were placed in columns 14 prisms high for a reactor core height of 7 m.

Initial calculational models began with a 6 x 6 matrix of fuel and control rods representing the repeating structures found in an RBMK reactor. At this time, only reflecting boundary conditions were available for the inter-sides of the matrix, as opposed to the cyclic boundary conditions that should be used, because the model was not symmetric. This resulted in an unrealistic flux tilt toward the inter-sides. To overcome this problem, only four control rods were used in the 6 x 6 matrix for the initial calculations. This model, shown in Fig. 1, produced a very flat flux.

Numerous static calculations were made the 6 x 6 model with the four control rods. Various patterns of fuel cross sections with different burnup and control rod heights were utilized in parameter studies. For example, a model containing burnt fuel at the center of the core which decreasing in burnup toward the top and bottom of the core was analyzed. The control rods were partially removed toward the bottom of the core and were replaced with a control rod follower. This model results in a reasonable eigenvalue and a relatively flat flux, both axially and radially.

The 6 x 6 model was extended to a full quarter core model, and a large number of static eigenvalue calculations were made. Varying patterns of burned fuel and control rods with varying patterns of insertion depth were used in these parameter studies. Again, reasonable eigenvalues and power profiles were obtained in these static parameter studies.

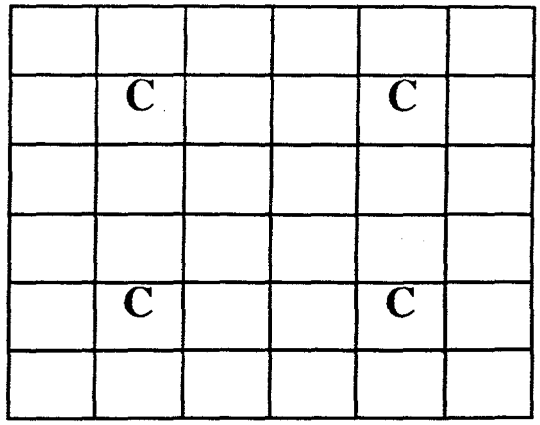

Fig. 1. Simplified RBMK reactor model. All cells contain fuel except those with C, which contain control rods or control rod followers.

CONCLUSION

A large number of static eigenvalue calculations were made for an RBMK reactor in which various patterns of burned fuel and control rods were used. These calculations were made to understand the core behavior for coupled thermal-hydraulic-neutronic calculations that were to be made at a later date.

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