RELAP5-3D Model for the KURSK 1 NPP[1]

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Abstract

A RELAP5-3D model has been developed for the RBMK Kursk 1 NPP. The primary features of this model are the three-dimensional neutronics model of the reactor core and the user subroutine that generates the neutron cross-sections. This subroutine contains the same neutron cross-section libraries that are presently being used in the Russian STEPAN/KOBRA calculations. The implementation of the neutronics mesh into the reactor core is described along with the results done to validate the accuracy of this representation.

General Description

A RELAP5-3D model has been developed to perform reactor safety analysis simulations for the Kursk 1 NPP. The model consists of an input file that is read and interpreted by the RELAP5-3D code and a set of Fortran subroutines that serve to calculate neutronics cross-sections for the various compositions of fuel channels, non-fuel channels, control rods, and reflector regions. The input file includes representations of the reactor core region and main circulating circuit, the main steam and feedwater systems, and the emergency core coolant systems. It also contains the reactor trip system and control rod logic, controls for the various pumps and valves of the system, and a 3-dimensional core neutronics representation. The reactor core region and main circulating circuit are divided into two halves, named the “Accident Side” and “Non-Accident Side”. The core is further divided into quadrants, with three azimuthal regions per quadrant. Additionally, the Accident Side includes a distinct representation of the channels of a single group distribution header, divided into three azimuthal regions plus two single channels representing high and low power regions, respectively. The core representation itself consists of the fluid channels, their associated fuel bundles, and the graphite matrix, and is divided into 10 axial regions. The representations of the main circulating circuits, main steam and feedwater systems, and emergency core coolant system were incorporated from a RELAP5/MOD3.2 model previously developed by RRC-KI. In the main circulating circuit these include the steam drums, downcomers, suction headers, main circulation pumps, pressure tubes, group distribution headers, and associated connecting piping. In the main steam and feedwater systems these include the main steam lines, Steam Relief Valves, the BRU-B and BRU-K valves, and main feedwater supply system. In the Emergency core coolant system these include the emergency feedwater pumps and accumulators and associated piping.

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This model has been developed jointly by the Russian Research Center "Kurchatov Institute" (RRC-KI) and the Idaho National Engineering and Environmental Laboratory (INEEL).

Neutron Kinetics Model

The reactor power generated in the heat structures representing the fuel and the graphite is calculated using the RELAP5-3D 3-dimensional neutron kinetics model. The reactor core can be represented as an array of 0.25 x 0.25 m graphite columns, each with a cylindrical hole in the center that has a pressure tube containing the fuel channel. Surrounding the core are four rows of graphite reflector blocks. The reactor map is roughly circular in cross-section, as shown in Figure 1.

The neutronics mesh consists of an array of nodes with dimensions 56 x 56 x 12 in cartesian geometry that includes the core and the surrounding reflector region. The mesh represents each rectangular 0.25 x 0.25 m graphite column as an individual stack of 12 nodes with a total height of 8 m. The active fuel is a 7-m-high region containing the consisting of 10 nodes of height 0.7 m each. The reflector regions above and below the core are each represented using a single node of height 0.5 m. The final mesh size is an array 56 x 56 square x 12 high. Of the total reactor power, 95% is generated in the fuel, 4.5% is generated in the graphite column, and 0.5% is generated in the reflector region.

User Subroutine

The neutronics calculation gets the neutron cross-sections for each node from the user subroutine interface, a feature that has been recently installed in RELAP5-3D. At the beginning of the neutronics calculation for each transient time step, information is passed to the top-level user subroutine from the thermal-hydraulic model and the rod control system for the region of the core corresponding to the neutronics node being evaluated. The top-level subroutine determines the appropriate type of cell (fuel, non-fuel, control rod, or reflector) based on the composition map, and calls the corresponding subroutine from the two-group macro cross section library. This library is the same as is used to calculate the neutron cross-sections for the RBMK simulation using the STEPAN code, and has been provided to INEEL by RRC-KI. The library subroutines return the diffusion, absorption, fission, and scattering cross-sections for the two neutron groups. The top level subroutine then transfers the cross-sections to the NESTLE kinetics solver. The process is repeated for each node of the kinetics mesh.

The control rod subroutines of the two-group macro cross-section libraries include indigenous logic for modeling the positions of the poison, water column and displacer regions of the various types of control rods in RBMK. The input to these models includes specification of the type of control rod , its insertion direction, and the position of the poison tip from the edge of the fueled region, and the axial position of the top and bottom of the neutronics node. The subroutines then calculate the fractions of poison, water, and displacer within the node and return the appropriate cross-sections.

In addition to the thermal-hydraulic and control rod position information, the two-group macro cross-section library requires the composition, the local burnup, and the relative power of the node. The composition map is specified based on the core loading pattern presented in "Full Withdrawal of a Single CR"[1], and is input to the kinetics model in tabular form. The burnup distribution is also specified in Reference 1 and is input as independently varying radial and axial distributions, also in tabular form. The relative node power is used to calculate the value of equilibrium xenon in the node. Relative power is calculated from the following equation:

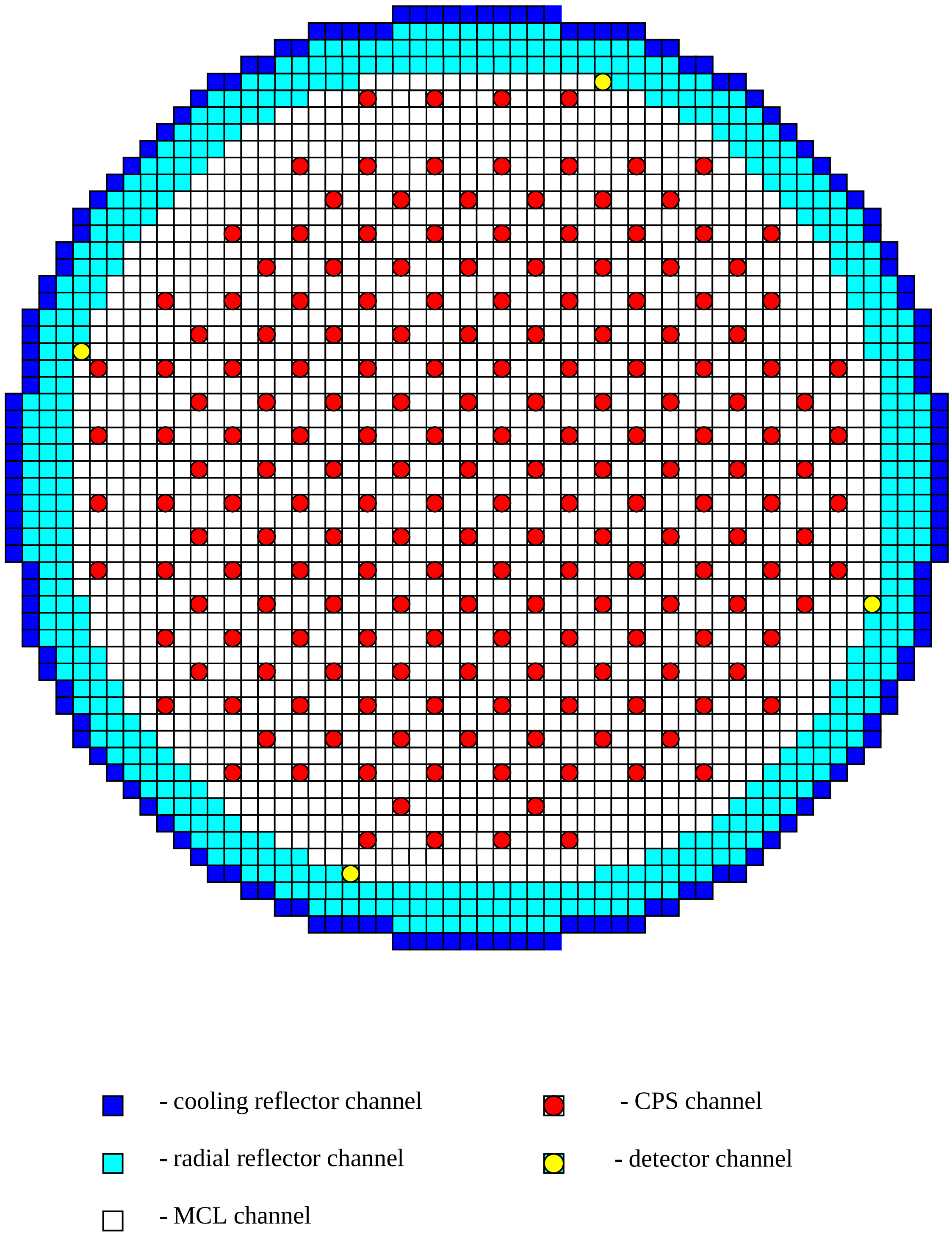

Figure 1. Full-scale reactor map

pow = (phi(1) * sigf1p(xyz,1) + phi(2) * sigf2p(xyz,2)) * G * K * V * N / R

where

pow is the relative node power
phi(i) is the neutron flux for group i in the current mesh position
sigf1p(xyz,i) is the macroscopic fission cross-section for group i for the current mesh position (this value is saved from the previous time step)
G = 200 Mev/fission
K = 1.6021917e-13 J/fission
N = 1661 fuel assemblies
V = fuel assembly volume (25*25*700 = 437500 cm^3)
R = rated power (3200 MW)

The variable "pow" is then passed to the cross-section library, where it is used to account for the local effect of xenon on the cross-sections. The equation is cast using neutron flux and fission cross section variables instead of using node power directly because the neutron fluxes are passed to the top-level subroutine, whereas node power is not. The fission cross-section from the previous time step is saved for use in the relative power calculation for the current time step. At the end of the steady-state initialization, the final values for relative power are saved and entered into the neutronics input in tabular form. These (constant) values are then used for the transient calculation. Thus, during the calculation of transient response, the magnitude of xenon in the node is assumed to be "frozen" at the equilibrium value corresponding to the relative power at the steady-state condition.

Core Thermal-hydraulic Nodalization

The axial hydrodynamic nodalization of the core was one-for-one with the kinetics mesh, and consists of 10 axial fluid volumes per hydrodynamic region, each with height of 0.7 m, for a total active fuel length of 7.0 m. The core planar model was divided azimuthally into quadrants, as shown in Figure 2. The flow channels of each quadrant were further divided radially into three geometric rings. Additionally, the Northeast quadrant has a single group distribution header modeled separately, with five flow paths that model each of the three geometric regions and two additional individual channels (one near the center and one near the edge of the fuel). Therefore, there are a total of 17 flow channels through the core.

Sensitivity studies were performed on the radial nodalization. The purpose was to determine whether a different method for combining the individual pressure channels into groups would result in a flatter radial power profile. The proposed method was to group the channels by burnup. Burnup was chosen as the parameter for grouping channels into regions because it was assumed that, in general, the lower burnup regions would correlate to regions of higher power, and therefore, channel void distribution, which is the most important feedback variable, would correspond more closely to that in the actual reactor channels. This did not turn out to be the case, however, as the burnup did not correlate closely to channel power. A more logical choice for grouping channels would have been channel relative power. This method was not used because it would have required some iteration to establish the channel grouping. This iterative procedure would have involved significant effort because it would be necessary to balance the channel inlet flows following each iteration. The iteration would be required because the power in individual channels may change as the nodalization is changed due to differences in inlet flow rate. Also, differences in initial control rod position from case to case would affect the azimuthal power distribution, thus resulting in differently nodalized models for different transient scenarios. Therefore, attempts to group channels into regions according to channel power were not pursued.

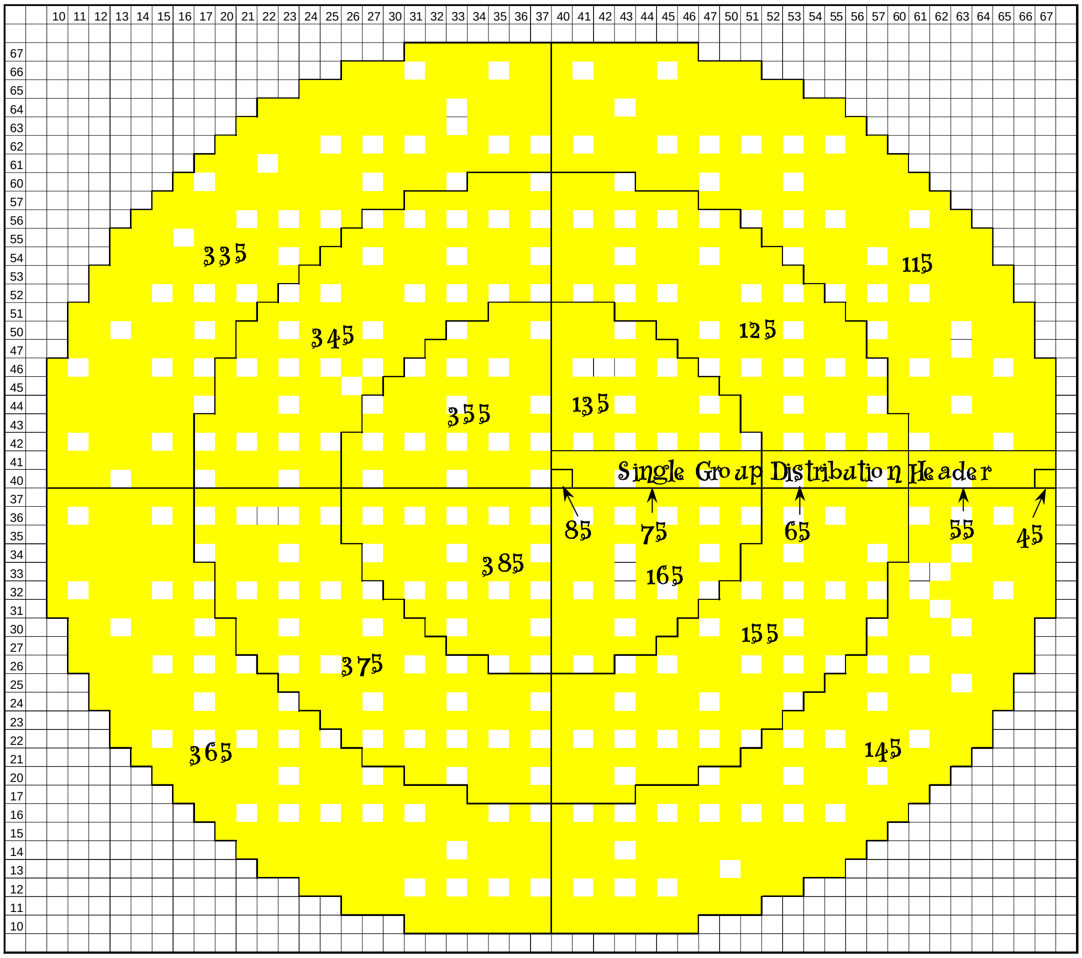

Figure 2. RELAP5-3D Radial Hydrodynamic Nodalization of Kursk 1 Reactor Core

Two sensitivity studies were performed. In the first case, the number of channels in each region was the same as in the geometrical model described above. Regions 135, 165, 355, and 385 contained the channels with lowest burnup, regions 125, 155, 345, and 375 were mid-burnup, and 115, 145, 335, and 365 had the highest burnup. In the single group distribution header region, 85 and 45 were the lowest and highest burnup channels, respectively, and regions 75, 65, and 55 were low-, mid-, and high-burnup regions, respectively.

The second case considered a finer radial nodalization, in which the number of channels per region was about 40. Again, channels were grouped into thermal-hydraulic regions according to burnup. This resulted in approximately 20 regions per side. The results of calculations with the two sensitivity cases showed no significant improvement in radial uniformity of the power profile. Therefore, the analysis cases to be done with this model will all be performed with the geometry-based nodalization scheme described above.

Control Rod Model

The 191 control rods in the plant are modeled with 87 control rods in the RELAP5-3D input deck. The rods are grouped by type, function, and initial insertion depth. Initial insertion depths were from Reference 1. Control systems were developed to model the movement of the control rods as described in the "RBMK 1000 Kursk 1 NPP Database"[2]. Control of the AR, LAR-BIK, compensating rods, and safety group rods is based on signals from ex-core neutron detectors. In the model, the steady state thermal neutron flux (i.e. the initial value calculated by the kinetics model) at the locations in the kinetics mesh corresponding to the detector locations are used as the basis for control rod movement. These steady-state values are assumed to be the desired flux values, and movement of the adjusting and compensating rods is based on the percent deviation of the transient flux from the steady-state value. The detectors are modeled as being at the core mid-plane (axially).

The adjusting, compensating, and safety group rods are assumed to move until the signal that initiated their movement is no longer valid. At that time, the rods stop moving and hold their current positions until another signal to insert or withdraw is received.

Validation Results

Several test calculations were performed to assess the validity of the power profile calculated by the nodal kinetics model. Three cases were performed with constant properties specified as inputs to the nodal kinetics model (i.e. feedback to the nodal kinetics based on thermal-hydraulic parameter changes was disabled).

Case 1 used
- uniform burnup (1100 MWD/FA)
- uniform graphite and fuel temperatures
- uniform water density for the channel and adjacent fuel channel

Case 2 used
- the actual radial burnup profile
- uniform graphite and fuel temperatures
- uniform water density for the channel and adjacent fuel channel

Case 3 used
- the actual radial burnup profile
- uniform graphite and fuel temperatures
- typical axial distribution of water density for the channel and adjacent fuel channel

The results were compared to results from the STEPAN code in Table 1 along with the results for the final model with feedback enabled and with steady-state conditions achieved. The eigenvalues calculated by RELAP5-3D for all cases are outside the normal range (the value should be less than Beta, which is 0.0059). The results, however, are fairly consistent with the STEPAN calculation for the cases with no feedback. For the final model with feedback enabled and steady-state conditions achieved, the eigenvalue is 0.0076, which is ~1.3 Beta. No comparison value from STEPAN is available for this case. Because of the relatively close agreement with STEPAN for the no-feedback cases, the eigenvalue calculated by the RELAP5-3D nodal kinetics is considered acceptable.

Table 1. Comparison of Kinetics Parameters for Kursk 1 NPP.

Case	Eigenvalue		Radial Nonuniformity		Axial Nonuniformity	
	RELAP5-3D	STEPAN	RELAP5-3D	STEPAN	RELAP5-3D	STEPAN
Burnup = 1100. Tgraphite = 700. K Tfuel = 700. K Gamma = 0.76 gm/cm^3 Gammaf = 0.76 gm/cm^3 Rel. Power for Xe = 0.7	1.0361	1.0300	7.77	5.00	1.59	1.60
Burnup = Real Tgraphite = 700. K Tfuel = 900. K Gamma = 0.76 gm/cm^3 Gammaf = 0.76 gm/cm^3 Rel. Power for Xe= 0.7	1.0080	1.0090	3.10	2.10	1.37	1.12
Burnup = Real Tgraphite = 700. K Tfuel = 900. K Gamma = Axial Distrib. Gammaf = Axial Distrib. Rel. Power for Xe = 0.7	1.0120	1.0095	2.46	1.97	1.25	1.18
Comparison to Plant Data	RELAP5-3D		RELAP5-3D	KURSK 1	RELAP5-3D	KURSK 1
Burnup = Real Thermal Hydraulic Model Feedback Enabled Relative Power for Xenon Calculated from Neutron Flux	1.0076		1.81	1.57	1.21	1.19

The radial nonuniformity calculated by RELAP5-3D was somewhat higher than the STEPAN values for the three cases with no thermal-hydraulic feedback. For the final model, the RELAP5-3D model radial nonuniformity was 1.81, compared to the measured value of 1.57. Figure 3 is a comparison of total channel power radial distribution compared to data from Reference 1. The plots show channel power across the major axes of the core, in the East-West direction at Y = 37, and in the North-South direction at X = 36. Refer to Figure 2 for the X-Y coordinates. The data show that the channel power has a relatively constant mean value and spread (standard deviation) across the entire core. The RELAP5-calculated profile is slightly skewed, with a peak value of 2.65 MW near the North boundary at (X,Y) = (36,37) and a depression in the center region of the core. The reason for this difference is not clearly understood. A test of the NESTLE neutronics mesh solver was performed for this model. For this test, the composition was assumed to be uniform throughout the mesh (the entire mesh was set to the composition of 2.4% enriched fuel). The inputs of fluid and heat structure temperatures were set to constant, uniform values. The result was a symmetric, cosine-shaped power profile, which demonstrated that the NESTLE numerical solver is functioning properly.

Next, variations in the hydrodynamic nodalization were investigated. As noted above, the initial nodalization grouped the RELAP5-3D flow channels by radial geometric region, and that sensitivity studies were done with the channels grouped according to burnup. The two studies considered 1) regrouping of the original 17 channels and 2) an entire renodalization with the core divided into right and left halves, with each half grouped by burnup into approximately 20 groups of 44 or 45 channels each. Neither sensitivity study had a significant effect on the skewness of the radial profile. It is speculated that imprecise knowledge of some aspect of the burnup distribution, the cross-sections, or the material properties may be responsible for the increased nonuniformity. Additional exploration into the causes of the radial power shape nonuniformity may be in order.

It is known that the Russian KOBRA thermal-hydraulics code also calculates excessive radial nonuniformity, and that the Russians use a “renormalization procedure” in STEPAN to correct the problem. This renormalization consists of a set of correction factors, one for each node in the neutronics mesh, that adjusts the group-2 fission cross-section for the node. These data were obtained from RRC-KI and were implemented into the RELAP5-3D model. The renormalized values in the RELAP5-3D model showed no noticeable improvement in the radial flux profile. It is not surprising that renormalizing the RELAP5 model based on STEPAN factors did not work. Differences in node length (KOBRA uses 0.5-m nodes, the RELAP5-3D model uses 0.7-m nodes), thermal-hydraulic feedback, inlet flow distributions, and other effects would tend to alter the required corrections. Proper implementation of correction factors in RELAP5-3D would require using the correction methodology itself, and not the factors that correct for a different set of parameters.

It is also known that the GRS code, ATHLET, also suffers from high radial nonuniformity. At this point, the lack of agreement between the RELAP5-3D model and the data is attributed to inadequate knowledge of the nodal compositions. It is concluded that the lack of agreement in the RELAP5-3D model is no worse than the “uncorrected” KOBRA results, or the results produced by ATHLET.

The axial power distribution calculated by the RELAP5-3D model is compared with data from Reference 1 as shown in Figure 4. Note that the RELAP5-3D result agrees very closely with the Kursk data. The peak value calculated by RELAP5-3D is 1.21 compared to the measured peak value of 1.19.

Summary and Conclusions

A RELAP5-3D model has been developed for the RBMK Kursk 1 NPP. This model incorporates the thermal-hydraulic plant representation from the RELAP5/MOD3.2 model previously developed by RRC-KI. It includes a three-dimensional neutronics model of the core, dimensioned as an array of nodes with dimensions 56 x 56 x 12. The cross-sections supplied to the neutronics are generated in a user subroutine that contains the neutron cross-section libraries that are presently being used in the Russian STEPAN/KOBRA calculations. These libraries include cross-section formulas for the RBMK fuel, non-fuel, reflector, and control rod compositions. The hydrodynamic representation of the core includes four quadrants with three azimuthal sections each, plus a separate representation of a single group distribution header. The core hydrodynamic axial nodalization is one-for-one with the inside ten nodes of the kinetics mesh. The feedback capabilities provided by the thermal-hydraulic representation of the reactor core and graphite stack, and the main circulating loops and secondary steam system will permit

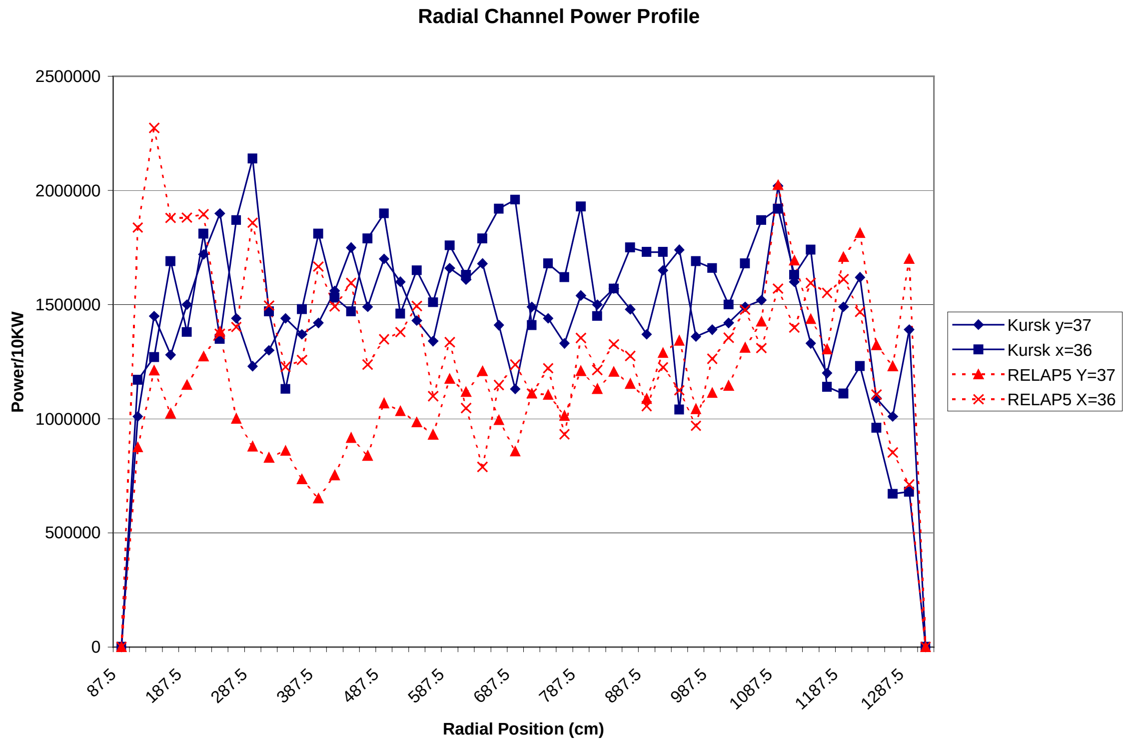

Figure 3. RELAP5-3D and Kursk 1 Radial Power Distributions

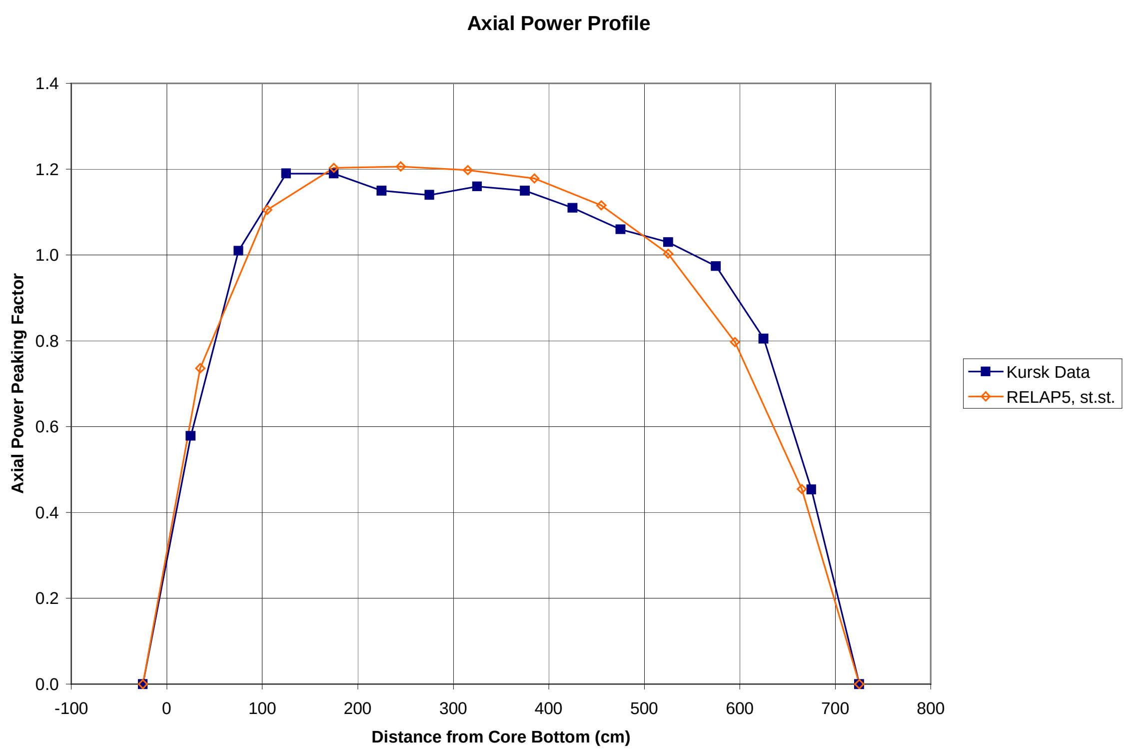

Figure 4. RELAP5-3D and Kursk 1 Axial Power Distributions.

accurate simulation of control rod withdrawal transients, anticipated transients without scram, and other accident scenarios for which the spatial distribution of power varies significantly during the course of the scenario. Although some improvements are indicated, the RELAP5-3D nodal kinetics model represents the Kursk 1 NPP power profile reasonably well. Improvements to this model should include further investigation of the reasons for a skewed radial power distribution.

REFERENCES

1 Russian Research Center "Kurchatov Institute", *Full Withdrawal of a Single Control Rod Transient, STEPAN/KOBRA Code Calculations*, Chapter 4.2.3 of Accident Analysis of and Training Programme for the RBMK 1000 Kursk 1 NPP, International Atomic Energy Agency Draft Report, Rev 0, 19 May 2000.

2 Russian Research Center "Kurchatov Institute", *Data Base for the RBMK 1000 Kursk 1 NPP*, Attachment 1 of Accident Analysis of and Training Programme for the RBMK 1000 Kursk 1 NPP, International Atomic Energy Agency Draft Report, Rev 0, 19 May 2000.