Development of Neutronics Model for ShinKori Unit 1 Simulator

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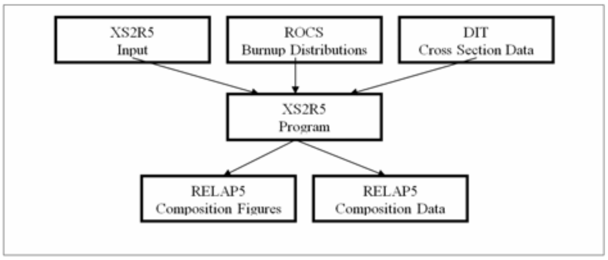
1. Introduction

ShinKori-Unit 1&2 is being built in the Kori site which will be operated at 2815 MWt of thermal core power. The purpose of this paper is to report on the performance of the developed neutronics model of ShinKori Unit 1&2. Also this report includes the convenient tool (XS2R5) for processing the large quantity of information received from the DIT/ROCS model and generating cross-sections. The neutronics model is based on the NESTLE code inserted to RELAP5/MOD3 thermal-hydraulics analysis code [1] which was funded as FY-93 LDRD Project 7201 and is running on the commercial simulator environment tool (the 3KeyMaster^{TM} of the WSC). As some examples for the verification of the developed neutronics model, some figures are provided. The output of the developed neutronics model is in accord with the Preliminary Safety Analysis Report (PSAR) [2] of the reference plant.

2. Cross Section Processing Utility

To facilitate development of the RELAP5 input data, a cross section processing utility was developed to process the large quantity of information received from the DIT/ROCS models into the form needed by Nodal Neutron Kinetics Module (NNKM). This utility performs specific functions. First, it takes the DIT/ROCS provided cross section data at each burnup and generates a set of cross sections and feedback coefficients for each lattice type. A lattice type is defined by the ^{235}U and burnable poison loading (i.e. IFBA, WABA, gadolinia). Second, the assignment of fuel compositions to kinetics model nodes is performed. For the ShinKori Unit 1 simulator project, actual design data for the core at the Beginning-of-Cycle 1 (BOC1), Middle-of-Cycle 1 (MOC1), and End-of-Cycle 1 (EOC1) were acquired from DIT/ROCS and used to determine the base cross section values and the feedback coefficients. These data consisted of the ShinKori Unit 1 Cycle 1 individual bundle lattice descriptions, the three-dimensional burnup distributions, and the cross section data for each individual lattice in the core. As was stated above, a cross section processing utility was developed to generate the RELAP5 NNKM input. Figure 1 illustrates the procedure for processing the RELAP5 NNKM cross section data.



[Fig.1] Dataflow within XS2R5

3. Neutronics Modeling

This section shows the technical descriptions applied to the RELAP NNKM.

3.1.Computation of Zone Average Properties

Each zone is subdivided into a number of regions and defines averages of volume and heat structure quantities for each region of a zone. The region average moderator temperature, average moderator density, the region average poison density and the region average structure temperature are computed using Eq.1, 2, 3 and 4 respectively except that the summations and weighting fractions are defined for a region in a zone rather than for the entire zone.

$$\bar{T}_{mik} = \frac{\sum_{j \in M_{ik}} T_{mj} W^{m}_{jik}}{\sum_{j \in M_{ik}} W^{m}_{jik}} \quad (1)$$

$$\bar{\rho}_{mik} = \frac{\sum_{j \in M_{ik}} \rho_{mj} W^{\rho}_{jik}}{\sum_{j \in M_{ik}} W^{\rho}_{jik}} \quad (2)$$

$$\bar{B}_{fik} = \frac{\sum_{j \in M_{ik}} B_j W^{B}_{jik}}{\sum_{j \in M_{ik}} W^{B}_{jik}} \quad (3)$$

$$\bar{T}_{sik} = \frac{\sum_{j \in S_{ik}} \bar{T}_{sj} W^{s}_{jik}}{\sum_{j \in S_{ik}} W^{s}_{jik}} \quad (4)$$

Where, the explanation of each notations is provided in reference [3].

3.2.Computation of Control Fractions

After the control rod positions are computed, the control fraction for each axial level is computed. The control fraction for each node N on axial level is given by

$$Cf_l = \frac{\sum_{r \in R_l} W_l^r Cf_l^r}{\sum_{r \in R_l} W_l^r}$$

where,

Cf_l = control fraction for node n,

Cf_l^r = control fraction for control rod group r on level l,

W_l^r = control rod weighting factor for control rod group r in node n, and,

R_l = list of rod groups to be used for node n.

3.3.Computation of the Neutron Cross Sections

Once the control fractions are computed on each axial level, the neutron cross sections for each level can be computed. Each cross section consists of two parts, a base cross section that only depends on the control fraction, and a variable part that depends upon changes in the thermal-hydraulic properties from those defined for the base state. The base portion of the cross section X in node L is computed from Eq.6.

$$\Sigma_{xl}^{base} = Cf_l\,(\Sigma_{xn}^{base})_{controlled} + (1 - Cf_l)\,(\Sigma_{xn}^{base})_{uncontrolled} \qquad (6)$$

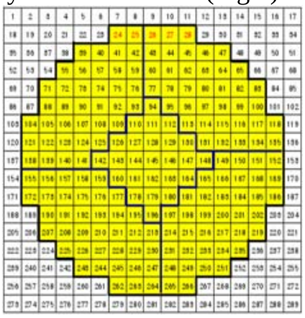
The variable portion of the cross section is defined for three control states, active controlled, driver controlled and uncontrolled states, and computed from,

$$\frac{\delta\Sigma_{xl}}{\Sigma_{xl}^{base}} = \sum_{i \in N_v} a_{xin}\Delta\bar{T}_{mik} + \sum_{i \in N_v} b_{xin}\Delta\bar{\rho}_{mik} + \sum_{i \in N_v} c_{xin}(\Delta\bar{\rho}_{mik})^2 + \sum_{i \in N_v} d_{xin}\Delta\bar{B}_{ik} + \sum_{i \in N_s} e_{xin}\Delta\bar{T}_{sik} \qquad (7)$$

$$\Sigma_l^x = C_{fl}^a\Sigma_{xc}^a\left(1+\frac{\delta\Sigma_{xc}^a}{\Sigma_{xc}^a}\right) + (1.-C_{fl}^a - C_{fl}^d)\Sigma_{xc}^u\left(1+\frac{\delta\Sigma_{xc}^u}{\Sigma_{xc}^u}\right) + C_{fl}^d\Sigma_{xn}^d\left(1+\frac{\delta\Sigma_{xc}^d}{\Sigma_{xc}^d}\right) \qquad (8)$$

3.4.Computation of the Neutron Cross Sections

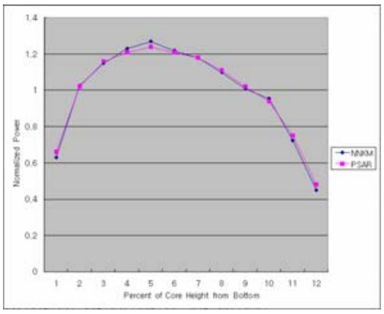
The developed neutronics model for the ShinKori Unit 1 simulator divided the total core into 12 axial slices and 177 radial meshes based on fuel assemblies, and 5 thermal-hydraulic volumes. (Fig.2)



[Fig.2] Nodalization diagram of the Core for NNKM

4. Verification Tests for Neutronics Model

Figure 3 shows the core average axial power distribution of PSAR[2] and NNKM and Figure 4 assembly-wise relative power for cycle 1 (ARO, HFP, Eq.Xe, 50 MWD/MTU) of PSAR and NNKM. In these figures, the results of the neutronics model correspond well with those of the PSAR of the ShinKori Unit 1 Cycle 1.



[Fig.3] Core Average Axial Power Distribution (HFP, ARO, Eq.Xe)

FUEL BOX TYPE NO. RELAP RPD PSAR RPD DIFF (%)	MAX -17.57	MIN -0.60	AVG 0.87		B 1 0.66 0.65 -1.54	B 2 0.95 0.91 -4.40
			B 4 0.56 0.62 10.48	C 5 1.08 1.03 -4.45	C1 6 1.01 1.09 7.34	C1 7 1.16 1.22 4.92
		C 9 0.50 0.75 33.42	C1 10 1.03 1.04 0.96	A 11 0.89 0.90 1.20	B2 12 1.19 1.21 1.65	A 13 0.96 0.93 -3.23
	B 15 0.55 0.62 11.29	C1 16 1.03 1.04 0.96	B1 17 1.15 1.19 3.36	B2 18 1.23 1.24 0.81	A 19 0.96 0.94 -2.13	C1 20 1.20 1.24 3.23
	C 22 0.99 1.03 3.88	A 23 0.89 0.90 1.21	B2 24 1.21 1.24 2.42	A 25 0.98 0.96 -2.08	B1 26 1.17 1.25 6.63	A 27 0.92 0.90 -2.22
B 29 0.56 0.65 13.85	C1 30 1.04 1.09 4.59	B2 31 1.20 1.21 0.83	A 32 0.98 0.94 -4.26	B1 33 1.17 1.25 6.64	A 34 0.92 0.89 -3.37	C1 35 1.28 1.15 -11.14
B 37 0.81 0.91 10.99	C1 38 1.23 1.22 -0.82	A 39 0.99 0.93 -6.45	C1 40 1.21 1.24 2.42	A 41 0.92 0.90 -2.22	C1 42 1.16 1.15 -0.87	A 43 0.77 0.76 -1.32
C 45 1.04 1.07 2.80	B1 46 1.23 1.24 0.81	A 47 0.95 0.94 -0.60	A 48 0.89 0.90 1.11	B1 49 1.22 1.19 -2.92	A 50 0.92 0.82 -12.20	A 51 0.76 0.74 -2.70

[Fig.4] Assembly-wise Relative Power (ARO, HFP, Eq.Xe, 50MWD/MTU)

6. Conclusions

To process a great volume of information from the DIT/ROCS for the NNKM, a cross section processing utility called XS2R5 is developed. With the tool, the neutronics model for the ShinKori Unit 1 simulator is developed. As some examples for the verification of the developed neutronics model, some figures are provided that show the comparison of the PSAR and NNKM for the axial power distribution and assembly-wise relative power. As a result, the developed model is proved to be consistent with the output of the PSAR of the ShinKori Unit1.

REFERENCES

[1] RELAP5-3D Code Manual, 2001. Vol. 1, Code Structure, System Models, and Solution Methods, The RELAP5-3D Code Development Team, INEEL-EXT-98-00834., Rev. 1.3a

[2] Korea Hydro & Nuclear Power Co., 2005, Preliminary Safety Analysis Report of ShinKori Unit 1&2, Chapter 4.

[3] Walter L.W., 1993. Software Design and Implementation Document – Three Dimensional Neutron Kinetics for RELAP5/MOD3, EGG-NRE-11021, Idaho National Engineering Laboratory.