Adaption of the PARCS Code for Core Design Audit Analyses

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1. Introduction

PARCS [1] is a three-dimensional neutronic code capable of performing both the eigenvalue calculation for steady-state reactor analysis and the kinetics calculation based on the transient fixed source problem. The eigenvalue calculation also includes quasi-static core depletion analyses. PARCS has implemented variety of features and has been qualified as a regulatory audit code in conjunction with other NRC thermal-hydraulic codes such as TRACE or RELAP5.

In this study, as an adaptation effort for audit applications, PARCS is applied for an audit analysis of a reload core design. The lattice physics code HELIOS [2] is used for cross section generation.

2. Characteristics of the Reference Core

Yonggwang Nuclear Power Plant Unit 5 Cycle 8 [3] has been selected for the audit calculation. The Yonggwang Nuclear Power Plant Unit 5 is a typical OPR1000 reactor having 177 fuel assemblies and the reload core has a low leakage loading pattern where 8 fresh fuel assemblies out of 64 are loaded in the outer most peripheries.

The fuel assemblies are classified into 5 fuel types, which have different numbers and locations of low enriched fuel rods and gadolinia-bearing rods. Top and bottom 6 inches of each rod are cutback with axial blank regions of 2.0 w/o enrichment uranium.

3. Calculation Methodology

3.1 Cross Section Representation

MARS is provided with burnup dependent cross section files for given fuel types. These files are called PMAXS files and are created from HELIOS outputs by an interface code GenPMAXS [4]. The cross sections in PMAXS are represented in the macroscopic form supplemented by the microscopic absorption cross sections of the Xenon and Samarium in case of absorption cross section as follows:

$$\Sigma^{Abs} = \Sigma^{Abso} + N_{Xe}\sigma_{Xe}^{Abs} + N_{Sm}\sigma_{Sm}^{Abs} \quad , \tag{1}$$

where $\Sigma^{abs,o}$ is Xenon-and-Samarium-free base cross section, and other notations have the usual standard meanings.

The cross sections depend on several state variables. These variables are split into two groups: instantaneous variables such as control rod fraction, coolant density, soluble boron concentration, fuel temperature, and coolant temperature; and history variables such as burnup and control rod history.

The cross section data is tabulated in a tree-leave structure comprising the cross sections at the reference branch and the partial derivatives at other branches. The resultant cross section at each node is represented by

$$\Sigma = \Sigma^{Ref} + \sum_k \left(\frac{\partial \Sigma}{\partial x_k}\right)(x_k - x_k^{Ref}), \tag{2}$$

where the superscript *Ref* refers to the reference state on which the depletion proceeds and x_k refers to each of the state variables except burnup.

3.2 Cross Section Files Generation

PMAXS files are generated for each of fuel types and for the axial blanket. Depletion calculation is performed by HELIOS at the reference state. Additional branch calculations are performed for rodded state, different coolant densities, soluble boron concentrations, and fuel temperatures in the order of priority. History variable dependencies are not modeled in this study. HELIOS calculation is performed on 1/8 fuel assembly, and the assembly discontinuity factor (ADF) and the corner discontinuity factors (CDF) are obtained for pin power calculation. For the reflector regions, the equivalent cross section generated for Yonggwang Nuclear Power Plant Unit 3 is used [5].

3.3 PARCS Input Modeling

Radially, each fuel assembly having 2x2 neutronic nodes is assigned by one thermal hydraulic node. Reflector nodes having the size of fuel assembly are placed in the periphery of the core and the zero flux boundary condition is imposed on the outer boundary of the reflector nodes.

Number of axial neutronic nodes is 28, including top and bottom reflector nodes. Neutronic nodes are grouped into 22 thermal hydraulic nodes so that the axial size is made similar to the radial size of fuel assembly. Outer boundary of the top and bottom reflectors is also imposed with the zero flux boundary condition.

The source expansion nodal method (SENM) is chosen for nodal kernel [6] and the pin power calculation is performed using the group form factor (GFF) provided by HELIOS.

As-built nodal burnup data for the cycle is used in the history file for the BOC burnup distribution. Other history data are not taken into consideration.

4. Calculation Results

4.1 Core Power Results

Fig.1 shows comparison results of the assembly power distribution along the centerline of the core at BOC Hot Full Power (HFP) condition and Table I summarizes the power distribution calculation results, showing a reasonable agreements.

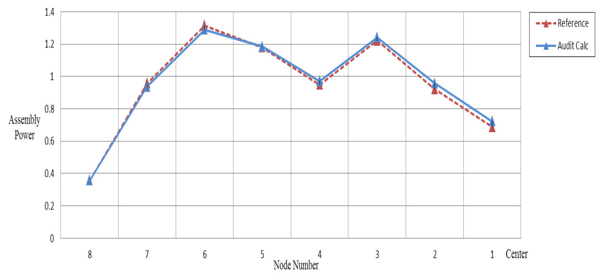

Fig. 1. Assembly power distribution at BOC, HFP

Table I: Power distribution summary

	Reference	Audit Calc.	Error (%)
Assembly Power rms Error	--	--	2.15 %
Max. Assemby Power (Location)	1.319 (3, 6)	1.3229 (3, 6)	+2.35 %
Max. Fr	1.545	1.6645	+ 7.7 %

4.2 Cycle Depletion Results

The cycle depletion calculation is performed with the burnup step of 30 EFPD. Fig.2 shows the boron letdown results from the quasi-static core depletion.

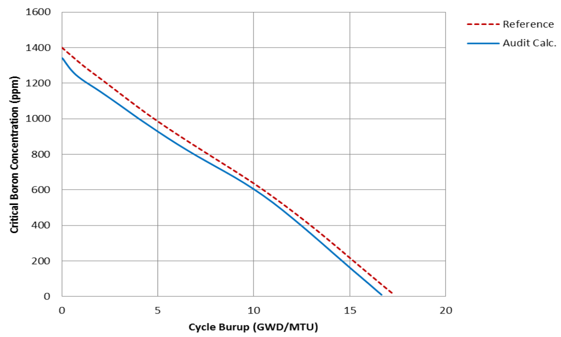

Fig. 2. Critical Boron Concentration over Cycle Burnup at HFP, ARO, Eq. Xenon

4.3 Control Rod Worths

Table II shows the rod worths calculation results at HFP, equilibrium Xenon condition. Rod worths have been calculated only for 4-fingered banks. A reasonable agreement has been found.

Table II: Control Rod Worth Results

Cycle Burnup	Control Rod Bank	Rod Worth (pcm)		
		Referenc e	Audit Calc.	Error
BOC (0 EFPD)	5	406	436	+30
	5+4	879	918	+39
	5+4+3	1212	1264	+52
	5+4+3+2	1751	1854	+100
EOC (420 EFPD)	5	418	428	+10
	5+4	966	1000	+34
	5+4+3	1364	1449	+85
	5+4+3+2	1922	2203	+181

5. Conclusions

PARCS-HELIOS code system has been established as a core analysis tool. Calculation results have been compared on a wide spectrum of calculations such as power distribution, critical soluble boron concentration, and rod worth.

A reasonable agreement between the audit calculation and the reference results has been found. The errors could have been reduced further, if the BOC burnup distribution input were replaced with that used for the design report, history variables such as control history were considered, cross section tables were modeled more delicately, and the cross section for the reflectors were derived directly from the actual geometry and compositions.

This study showed that the PARCS-HELIOS code has a good potential for audit analysis and may be used for safety analysis in a coupled form with the thermal hydraulic system codes.

REFERENCES

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