Core Nodalization Effects in the Main Steam Line Break Analysis Using the MARS/PARCS Coupled Code

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1. Introduction

The thermal-hydraulic (TH) system code MARS has been developed to have an option for coupling with the three-dimensional (3D) kinetics code MASTER for detailed core transient calculations [1,2] and validated through the OECD main steam line break (MSLB) benchmark problem [3]. In the MARS/MASTER code system, the MASTER code is provided in a Dynamic Link Library (DLL) and may be replaced with other kinetics codes with equivalent capabilities. In this study, the PARCS code [4] is modified for coupling with the MARS code to make its variety of features available to the users and core modeling effects are investigated.

2. Code Coupling and Verification

2.1 Data Communication

MARS is allowed to call a DLL of 3D kinetics code and remains unchanged for coupling with PARCS. The transmitted data from MARS to PARCS includes coolant density and temperature, and fuel temperature of each node in the reactor core region. The data returned from PARCS at each call includes the total core power and the relative nodal power density of each node. PARCS calculates control rod axial positions when the trip signal is received from the MARS which models the trip logics of the simulated plant.

Since the TH nodes in MARS and the neutronic nodes in PARCS do not exactly match and generally the former is coarser than the latter, a mapping process between the two node structures is required with a pre-specified mapping table input.

2.2 Description of the MSLB event

For verification of the MARS/PARCS coupling, the so-called second scenario is used to test better the prediction of a return to power after the reactor trip. The rupture of one steam line is assumed to occur upstream of the cross-connect. The feedwater regulating valve in the broken steam generator is assumed to fail in the open position. Following reactor trip, the steam line turbine stop valves are assumed to slam shut. The reactor trip is caused by the over-power or the low pressure setpoints with appropriate time delays.

2.3 Verification Results of the Code Coupling

For verifying the code coupling, this study uses the same system noding structure with MARS 1D module shown in Fig.1 as adopted in Ref. 1.

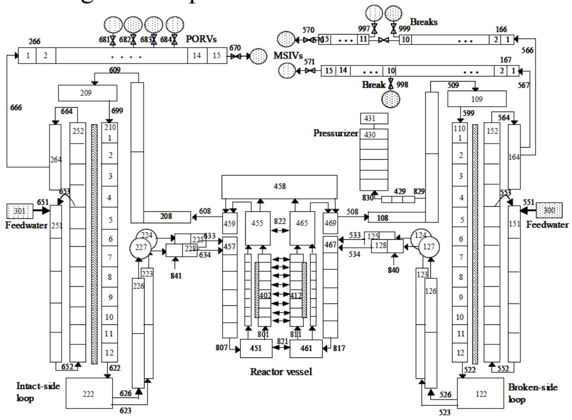


Fig.1. Plant system nodalization

The reactor vessel is modeled with two pipe components for the core region and one additional pipe for the core bypass.

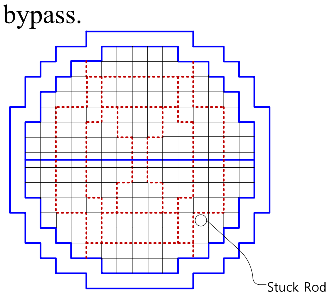


Fig. 2. Core model layout (2CH-18HS)

The core in Fig. 2 shows radially two TH channels contoured by the blue solid lines, with the surrounding peripheral bypass region. The lower channel in the figure corresponds to the broken side with the stuck rod location indicated, while the upper channel corresponds to the intact side. Each channel contains 9 fuel regions contoured by red dotted lines modeled by heat structures (HS) representing 7.5, 10, or 12 fuel assembly squares (Denoted as Case 2CH-18HS).

Figs. 3-5 show the transient calculation results of core power, cold leg temperature, and normalized power peaking factor calculated by MARS/PARCS compared against those by MARS/MASTER.

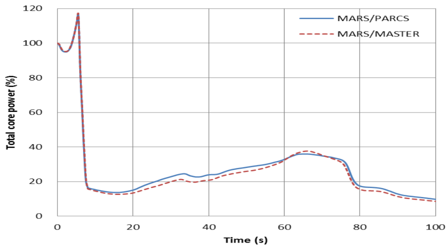


Fig. 3. Total core power behavior

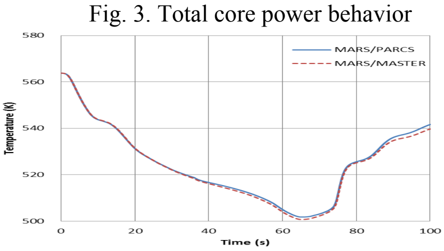


Fig. 4. Cold leg temperature (Broken side)

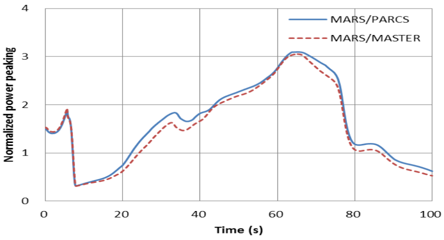


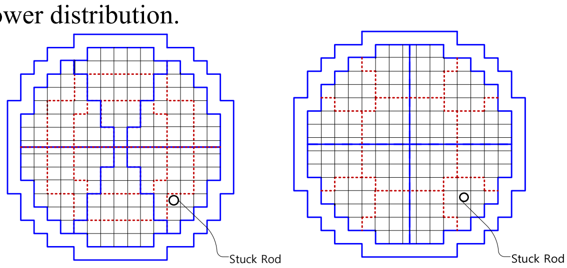
Fig. 5. Normalized power peaking factor

The compared calculation results are practically identical, considering that the kinetics modules are totally different. Therefore, code coupling of MARS and PARCS is decided to be successful.

3. Core Nodalization Effects

3.1 Core Model Descriptions

Several core models are developed to test the adaptability of the coupled code. Fig. 6 shows MARS 1D core models consisting of 6 TH channels and 18 heat structures (Case 6CH-18HS) and 4 TH channels and 16 heat structures (Case 4CH-16HS). Other core models not displayed here is a MARS 3D model [2] having 18 heat structures (Case 3D-18HS). The MARS 3D core model is presumed to give the most realistic power distribution.



< 6CH-18HS > < 4CH-16HS >

Fig. 6. Core model layouts

3.2 Calculation Results with Different Core Models

Since the total core power generally depends on the average TH condition in the core, it is not so sensitive to the core models as shown in Fig. 7. However, the core model causes a nontrivial impact on the power peaking factor as shown in Fig. 8.

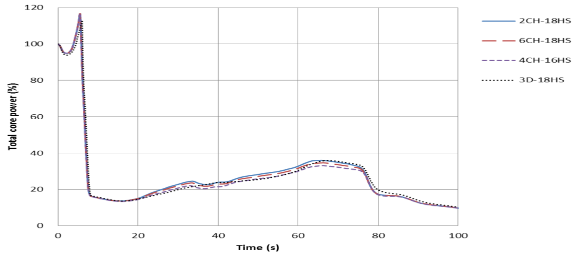


Fig. 7. Total core power variation

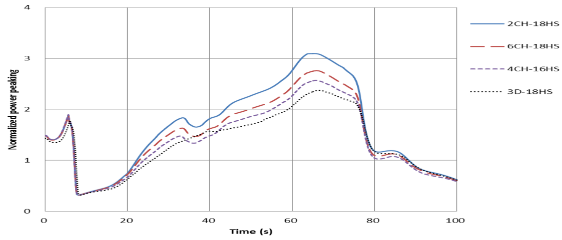


Fig. 8. Power peaking factor variation

The power peaking factor of Case 2CH-18HS is presumed conservative, since the TH feedback effects average out over the relatively large TH node. Case 4CH-16HS appears to give a reasonably realistic but still conservative peaking power when compared to Case 3D-18HS. In this case, the fuel assemblies around the stuck rod position with high power density at return-to-power are represented by one heat structure and, therefore, the fuel temperature is effectively reflected in the local reactivity through Doppler feedback. Also, the TH channels are reasonably sized. It is noted that the smaller channel size of Case 6CH-18HS is not effective.

4. Conclusions

MARS and PARCS codes have been successfully coupled and verified with the OECD MSLB benchmark problem. Various MARS 1D core models are tested with different node mapping. The TH channel model of quadrant core size appears reasonably efficient and the assemblies around the stuck rod position are suggested to be grouped and represented by a single heat structure.

REFERENCES

[1] “MARS Code Manual, Volume IV: Developmental Assessment Report,” KAERI/TR-3042/2005, Korea Atomic Energy Research Institute, pp.280-288, December 2009.

[2] J. -J. Jeong, et al., “Development and Verification of “System Thermalhydraulics - 3 Dimensional Reactor Kinetics” Coupled Calculation Capability using the MARS 1D Module and MASTER Code,” KAERI/TR-2232/2002, Korea Atomic Energy Research Institute, 2002.

[3] N. Todorova, et al., “Pressurized Water Reactor Main Steam Line Break (MSLB) Benchmark, Volume IV: Results of Phase III on Coupled Core-plant Transient Modeling,” NEA/NSC/DOC(2003)21, OECD, 2003.

[4] T. Downar, Y. Xu, and V. Seker, “PARCS v3.0: U.S. NRC Core Neutronics Simulator User Manual”, UM-NERS-09-0001, RES / U.S. NRC, 2012.