Verification of the Coupled 3-D Neutronics and Thermal-Hydraulic Code SKETCH-INS/TRAC-P

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Abstract
In order to analyze the complex transients with coupled interactions between core behavior and plant dynamics, the three-dimensional neutronics code SKETCH-INS was coupled with the thermal-hydraulic code TRAC-P. The capability SKETCH-INS code was verified against the 3-D transient benchmark problem. The capability of the coupled code SKETCH-INS/TRAC-P was verified against the NEACRP 3-D LWR core transient benchmark and OECD MSLB benchmark problems. The results of analyses were in reasonable agreement with the reference and different codes results of benchmarks. This paper provides the outline of the coupled code SKETCH-INS/TRAC-P and the results of benchmarks.

KEYWORDS: Coupled code, 3-D neutronics, Thermal-Hydraulic, SKETCH-INS, TRAC-P, Benchmark

1. Introduction

Traditionally, a point kinetics model has been employed for the safety assessment of system-transients. This approach requires the additional conservatism by applying static power distributions. The conservative core power distribution is applied in the thermal-hydraulic analysis to include uncertainties due to the fixed spatial power distribution during the transient. The detailed dynamic thermal-hydraulic (T-H) system analysis together with coupled detailed three-dimensional (3-D) core kinetics will provide a more appropriate evaluation of the safety margins found in previous (licensing) simulations for which a point kinetics model was used. Benefits of this type of coupled T-H/neutronics simulation are seen in evaluating the actual safety margins, improving the design/operating conditions and establishing a basis for advancing the technology. In order to improve the T-H/neutronics simulation capability of complex transients, the 3-D neutronics code SKETCH-INS was coupled with the thermal-hydraulic code TRAC-P. The capability of SKETCH-INS code was verified against the 3-D transient benchmark problem. The capability of the coupled code SKETCH-INS/TRAC-P was verified against the NEACRP 3-D LWR core transient benchmark and OECD MSLB benchmark problems. This paper provides the outline of SKETCH-INS/TRAC-P code and the results of benchmarks.

2. Outline of SKETCH-INS/TRAC-P Code

The three-dimensional neutronics code SKETCH-INS was coupled with the thermal-hydraulic code TRAC-P. The SKETCH-INS code is a three-dimensional neutron kinetics code based on the modern nodal method. [1] The SKETCH-INS code is a modification of the SKETCH-N code which was originally developed in ex-JAERI. [2],[3] The assembly discontinuity factors (ADFs) and pin power reconstruction model have been incorporated into the SKETCH-INS code in JNES. The nonlinear iteration scheme based on the polynomial and analytical nodal method is used to solve the diffusion equations. The time integration of neutron kinetics is performed by the fully implicit scheme.

The TRAC-P code is a best-estimate system transient analysis code, which has a three-dimensional thermal-hydraulic analysis capability. The code solves the general transient two-phase coolant conditions in one, two, or three dimensions using a realistic six-equation for two-fluid with finite difference model. [4]
The coupling and data transfer between the codes is organized using the message-passing library Parallel Virtual Machine (PVM).

3. Results of Benchmarks

3.1 Three-Dimensional LMW Benchmark

The Langenbuch-Maurer-Werner (LMW) benchmark problem is an operational transient in a LWR model. [5] Fig.1 shows the geometry of the model reactor. The core contains two types of fuel and two groups of control rod. The macro cross sections and other data defining the transient are given. The transient is initiated by withdrawing the control rod group 1 from t=0 to t=26.6 s and another control rod group 2 is inserted from t=7.5 to t=47.5 s. The SKETCH-INS calculation was performed using a quarter-core geometry. A neutronics spatial mesh was 10x10x10 cm (4 nodes per fuel assembly in radial plane and 20 axial layers including reflector part). The SKETCH-INS result was compared with the reference results obtained by the 3-D coarse mesh neutronics code CUBBOX with a mesh size of 20x20x20. Fig.2 shows the comparison of results for time-dependent average power density and local power densities. The SKETCH-INS results agreed well with the reference results.

Figure 1: Geometry of three-dimensional model reactor.

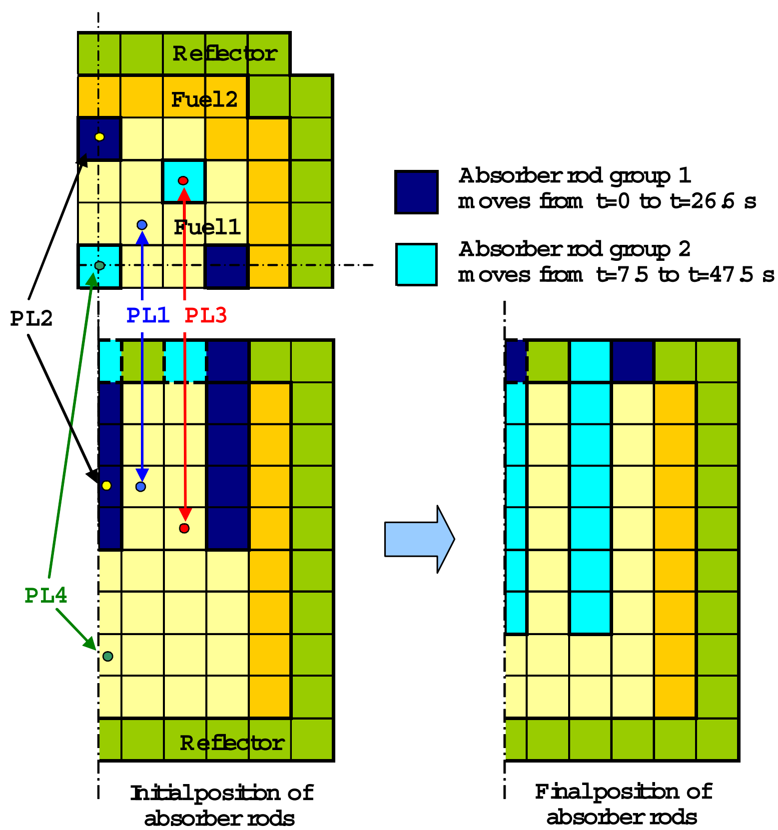
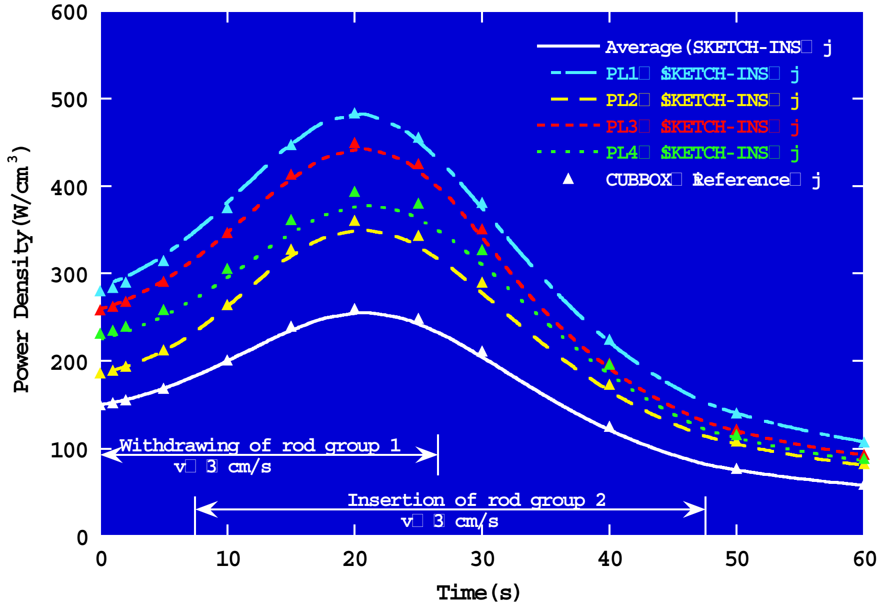


Figure 2: Average and local power density versus time.



3.2 NEACRP 3-D LWR Core Transient Benchmark

The NEACRP 3-D LWR core transient benchmark is a PWR rod ejection problem. [6] The benchmark aimed at assessing the discrepancies between three-dimensional codes for transient calculation in LWR cores. The PWR core model is derived from the real reactor geometry and operation data. The transient is initiated by a rapid ejection of a control rod at the hot zero power of the beginning of cycle 1 for the case A1. The benchmark calculation for the case A1 was performed by the coupled code SKETCH-INS/TRAC-P. The result was compared with the solution obtained by the PARCS code as a reference result. [7]

3.2.1 Analytical Model and Calculation Conditions

For the neutronics model, two neutron energy groups and six groups of delayed neutron precursors are used. The reactor core configuration of the case A1 is shown in Fig.3. Within the core geometry 11 different compositions and the corresponding sets of cross sections are defined. The macro cross sections are given as functions of the boron concentration, fuel temperature, moderator temperature and moderator density. The SKETCH-INS calculation was performed using a quarter-core geometry for the case A1. A neutronics spatial mesh was defined with 4 nodes per fuel assembly in radial plane and 30 axial layers including reflectors in the SKETCH-INS model.

In the TRAC-P code, the VESSEL component is used to simulate the reactor. The spatial mesh in r-θplane contains 21 nodes for the case A1. The axial spatial mesh has 26 layers. The reactor boundary conditions are given using the FILL component on the bottom and the BREAK component on the top, which specify the mass flow rate and the reactor pressure respectively.

In the initial condition, a reactor is critical and a value of the boron concentration is calculated. The main calculation conditions used in the benchmark are shown in Table 1.

Figure 3: Geometry of three-dimensional model reactor.

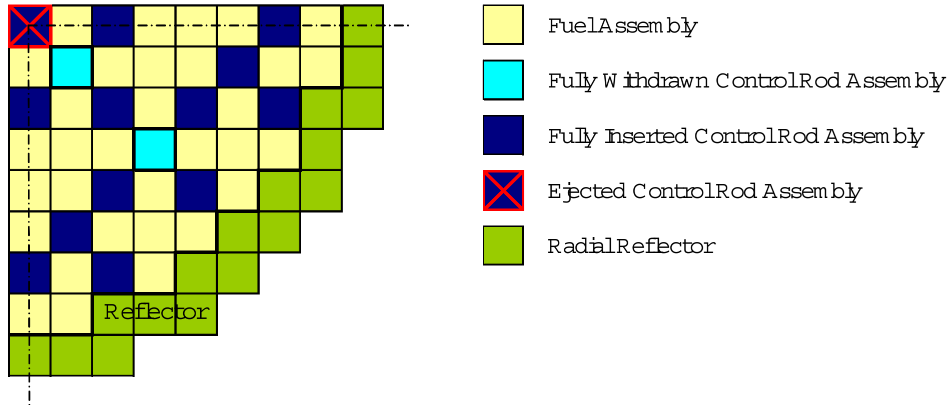


Table 1: Main calculation conditions for PWR rod ejection benchmark at HZP.

Parameters	Value
Rod ejection time (s)	0.1
Delayed neutron fraction (%)	0.76
Ratio of direct moderator heating (%)	1.9
Radial power distribution inside the fuel	flat profile

3.2.2 Results of Calculations

The steady-state result of the case A1 and a comparison with the PARCS solution is presented in Table 2. Fig.4 shows a comparison for the assembly power distribution at the HZP with the PARCS solution. The core averaged axial power distribution is compared in Fig.5. The critical boron concentration of the SKETCH-INS/TRAC-P is in good agreement with the PARCS solution. The maximum difference in the assembly power is 4.2 % and the RMS of differences is 2.0 %. The core averaged axial power distribution is in good agreement with the PARCS solution. The steady-state result of the SKETCH-INS/TRAC-P is in good agreement with the PARCS solution.

The transient result and a comparison with the PARCS solution is also presented in Table 2. Fig.6 shows a comparison for the total reactor power with the PARCS solution, together with the average fuel temperature. Fig.7 shows a comparison for the total reactivity with the PARCS solution, together with the average fuel and coolant temperatures. The inserted reactivity is in good agreement with the PARCS solution. However, the average fuel temperature during the transient is underestimated in the SKETCH-INS/TRAC-P. This difference may be the result of different code models in the heat transfer correlations. This fuel temperature difference causes the difference in the negative Doppler reactivity feedback, and hence the SKETCH-INS/TRAC-P slightly overestimates the peak power. Also, the power response after reaching the peak power is different due to the difference in the negative Doppler and moderator reactivity feedback, which is caused by the difference in the fuel and coolant temperatures.
A comparison results show that the coupled SKETCH-INS/TRAC-P code system can reasonably predict the most important parameters in the rod ejection analysis; the time and value of the peak power.

Table 2: SKETCH-INS/TRAC-P result of PWR rod ejection benchmark and a comparison with PARCS solution.

Parameter		SKETCH-INS/ TRAC-P	PARCS
Steady-State Results	Critical Boron Concentration (ppm)	560	561
	Maximum Difference in Assembly Power between SKETCH-INS/TRAC-P and PARCS	-4.21	
	RMS of Differences in Assembly Power between SKETCH-INS/TRAC-P and PARCS	2.00	
Transient Results	Inserted Reactivity (%Δk/k)	0.816	0.819
	Time to the Power Peak (s)	0.558	0.540
	Power at the Peak (-)	1.297	1.262

Figure 4: Assembly power distribution at the HZP.

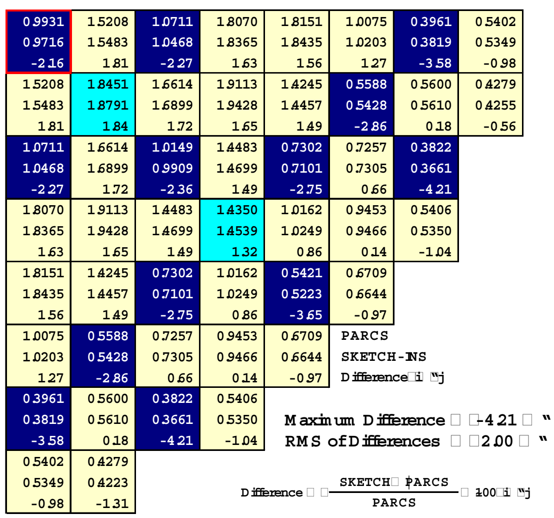


Figure 5: Core averaged axial power distribution.

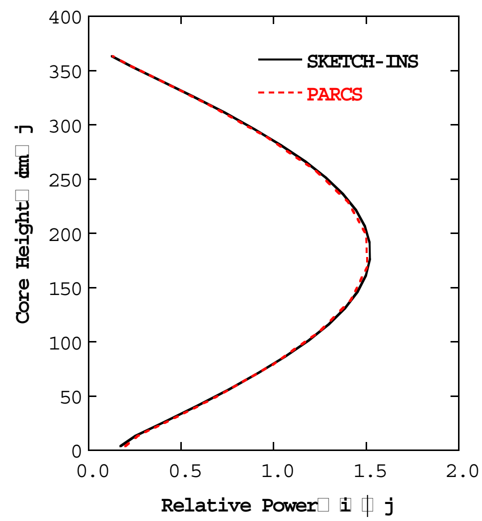


Figure 6: Reactor power and fuel temperature time history

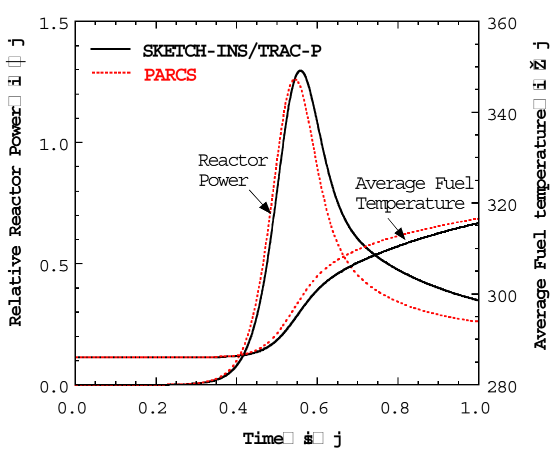
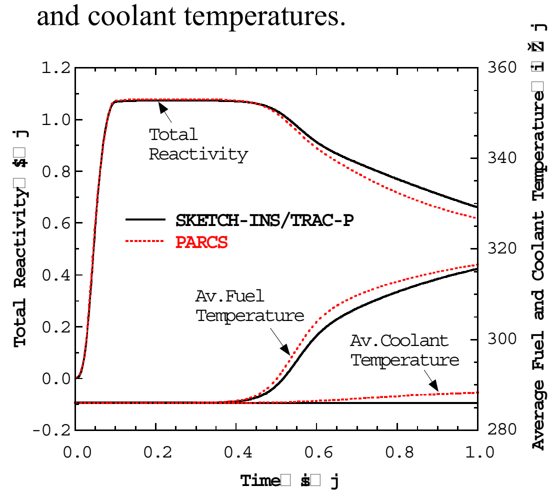


Figure 7: Total reactivity, average fuel and coolant temperatures.



3.3 OECD MSLB Benchmark

A Main Steam Line Break (MSLB) accident in PWR is characterized by the asymmetric behavior in the core due to the incomplete loop flow mixing before entering the core. The PWR MSLB Benchmark for the Three Mile Island Unit 1 (TMI-1) was performed to verify the capability of coupled code to analyze complex transients with coupled core-plant interactions and fully test the thermal-hydraulic coupling. [8] In this benchmark problem, two scenarios were specified. The first scenario is the best estimate case, in which the tripped rod worth is the best estimate value. The second scenario is the conservative case, which is employed in the current licensing analysis. In the second scenario, the tripped rod worth was conservatively reduced by about 30%. For two scenarios, the two different cross section libraries are provided in the benchmark. The PWR MSLB benchmark was analyzed by the SKETCH-INS/TRAC-P code using both the best estimate and conservative (non-realistic) cross section libraries. As a part of the analysis, the influence of loop flow mixing in the vessel was studied by changing the amount of vessel mixing.

3.3.1 Analytical Model

The TMI-1 core contains 177 (15×15 type) fuel assemblies. Twenty-nine assembly types are contained in the core. The radial SKETCH-INS nodalization model is shown in Fig.8, together with the radial distribution of these assembly average exposures. The axial noding consists of 26 axial nodes, including bottom and top reflectors.

The thermal-hydraulic plant model used in this study was provided by the Pennsylvania State University (PSU). [9] The provided model, Fig.9, contains approximately 600 hydrodynamic cells. All loop components, such as the hot legs, cold legs, steam generators, circulating pumps, and pressurizer, are modeled in detail by TRAC components. In the TRAC model, the pressure vessel is subdivided into 14 axial levels, 5 radial rings, and 6 azimuthal sectors, as shown in Fig.10.

Figure 8: Radial SKETCH-INS nodalization and assembly exposure distribution at EOC.

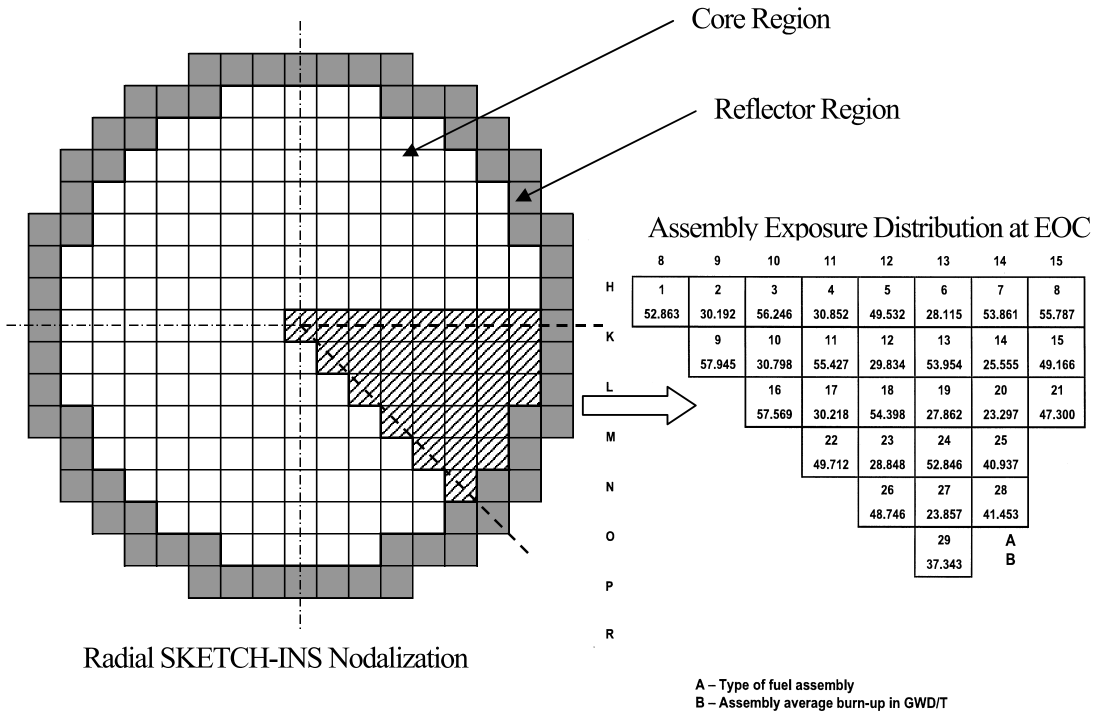


Figure 9: TMI-1 TRAC model. [9]

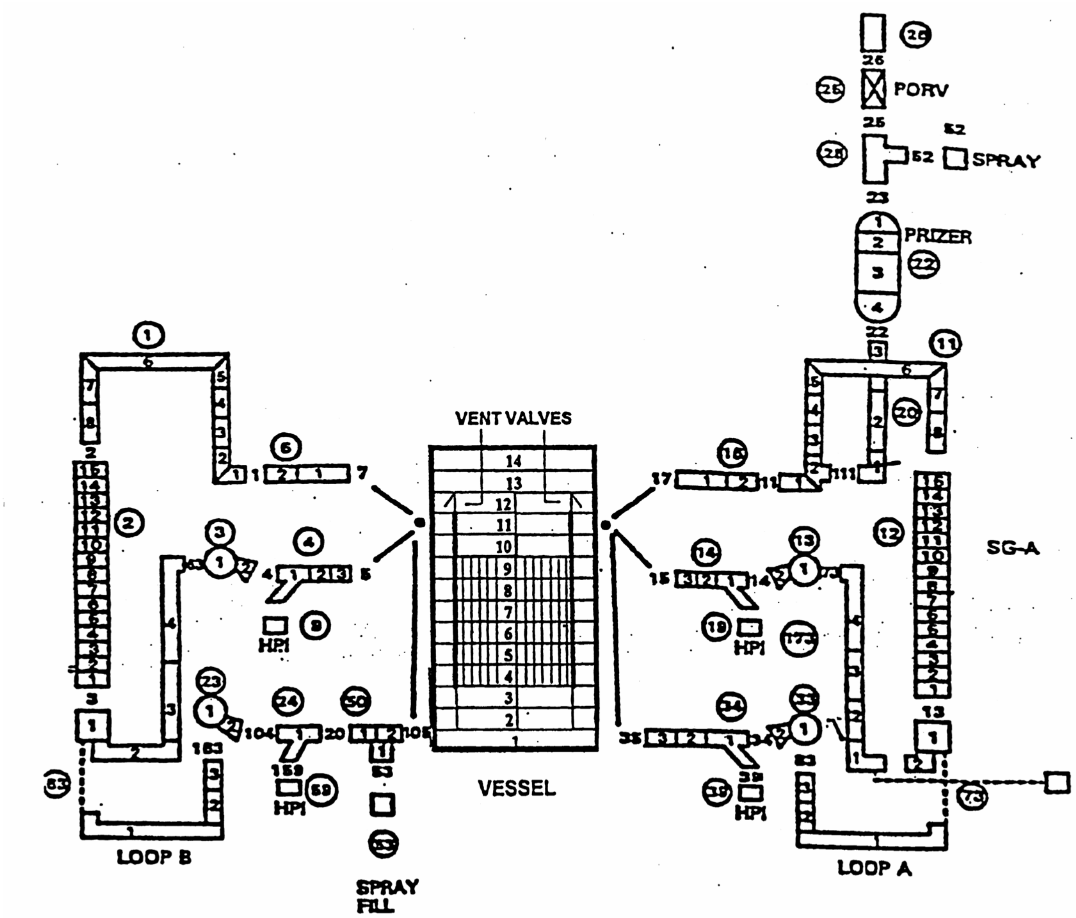
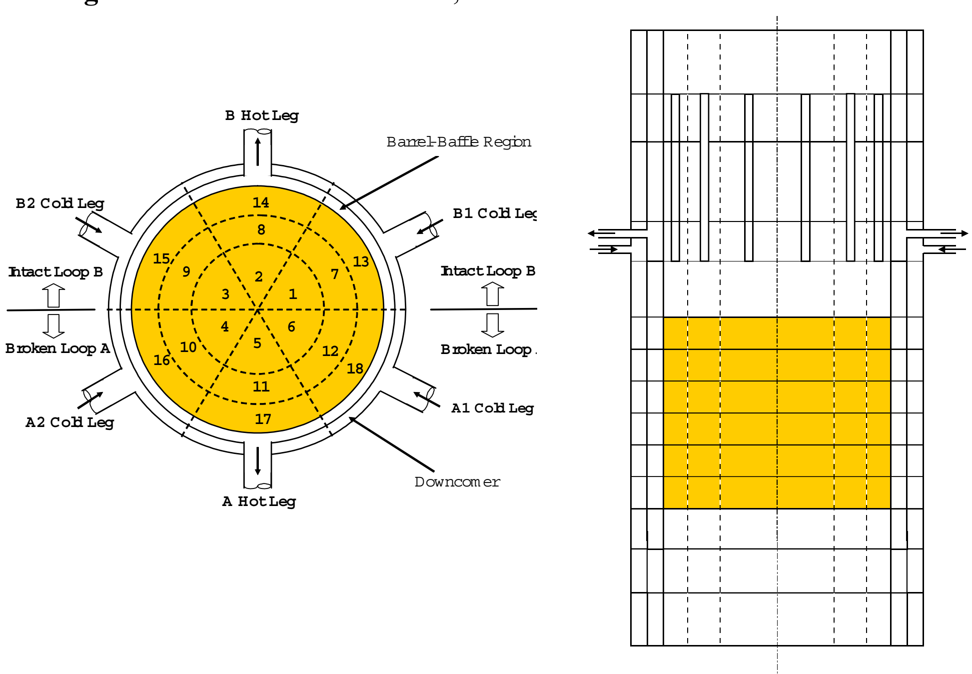


Figure 10: TMI-1 vessel radial, azimuthal and axial nodalization.



3.3.2 Loop Flow Mixing in the Vessel

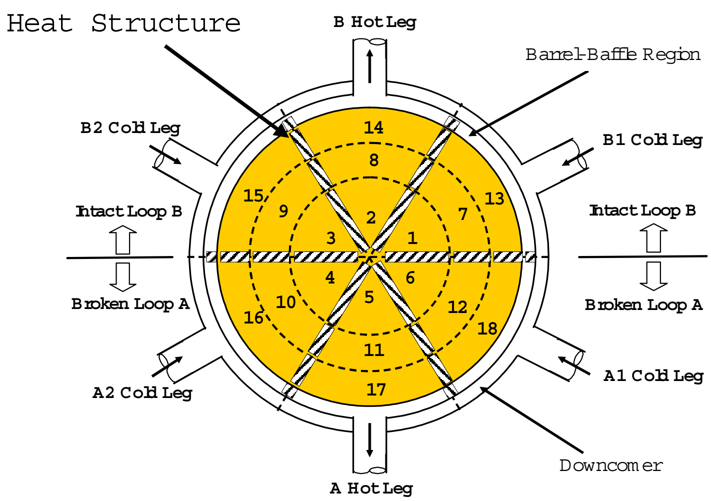
The amount of loop flow mixing in the vessel is defined as a ratio of the difference in hot-leg temperatures to the difference in cold-leg temperatures:

$$RATIO = [\, T_{hot}(intact) - T_{hot}(broken) \,] \,/\, [\, T_{cold}(intact) - T_{cold}(broken) \,]$$

A value of RATIO=0.0 implies perfect mixing while 1.0 implies no mixing. In the benchmark problem, a target value of RATIO=0.5 was chosen by following the test results performed on the Oconee plant. [10]

TRAC-P has a three-dimensional vessel fluid-dynamic capability with explicitly modeling radial cross flows between thermal-hydraulic cells. However, the code predicted the small loop flow mixing in the vessel compared to the target mixing value defined in the benchmark. To simulate this mixing in TRAC-P, the vessel model was modified to include heat structure components between the azimuthal sectors, as shown in Fig.11. The desired amount of loop flow mixing was obtained by exchanging the thermal energy between the sectors with heat exchangers.

Figure 11: Modified vessel model to simulate loop flow mixing within the vessel.

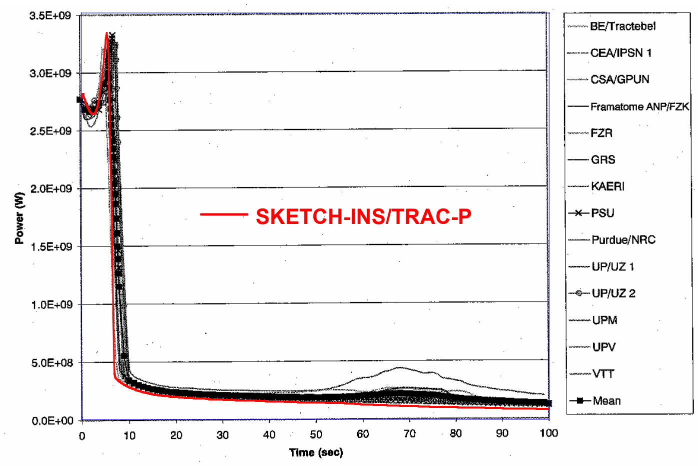


3.3.3 Results of Calculations

The result of the best estimate case (Scenario 1) for the reactor power response is shown in Fig.12, together with the different codes results. At the beginning of the transient, a rapid power rise occurs when the over-cooled primary coolant from the broken SG reaches the core. The power rise continues until the reactor trip. After the trip, a sharp decrease in power results from the negative reactivity inserted by the reactor scrams. In Scenario 1, no return to power and criticality is observed with coupled 3-D core/plant models. For this scenario, the SKETCH-INS/TRAC-P result is in good agreement with the different codes results.

The result of the conservative case (Scenario 2) for the reactor power response is shown in Fig.13, together with the sensitivity analysis result for the loop flow mixing within the vessel. A comparison with the different codes results is also shown in Fig.13. Fig.14 shows a comparison for the total reactivity behavior. In Scenario 2, return to power and criticality is observed with coupled 3-D core/plant models. The SKETCH-INS/TRAC-P result for the reactor power at the first power peak is in good agreement with the different codes results. However, the SKETCH-INS/TRAC-P predicted

Figure 12: Comparison of power response with different codes results for Scenario 1.



a smaller power and earlier time to the peak at the second power rise compared to the mean results of other codes. The discrepancy in the power behavior after the trip is due to the difference in the positive reactivity feedback inserted by the over-cooled primary coolant from the broken SG, as shown in Fig.14. Fig.15 shows a comparison for the coolant mass behavior of broken SG. Fig.16 shows a comparison for the cold leg temperature behavior of broken loop. The mass in the broken SG decreases throughout the transient until it eventually blows dry. The SKETCH-INS/TRAC-P predicted a rapid decrease of the mass and earlier termination of break flow from the broken SG compared to the mean results of other codes. This difference is caused by modeling differences in the break flow rate and the various other code correlations. This rapid decrease of the broken SG mass results in less overcooling of the primary coolant and earlier termination of its cooling capacity compared to the mean results of other codes. After blowing dry in the broken SG, the cold leg temperature of broken loop increases rapidly and then the power quickly decreases due to the negative moderator reactivity feedback. Time to the second power peak coincides with the broken SG blows dry, as shown in Figs.13 and 15. While the difference exists in the coolant mass behavior of the broken SG, the general behavior of the Scenario 2 was reasonably predicted by the SKETCH-INS/TRAC-P.

On the other hand, the sensitivity analysis result assuming no vessel mixing predicted a larger power rise, as shown in Fig.13. Thus the sensitivity analysis result indicated that the vessel flow mixing greatly influences to the moderator feedback and hence to the power response after the reactor trip.

Figure 13: Power response for Scenario 2.

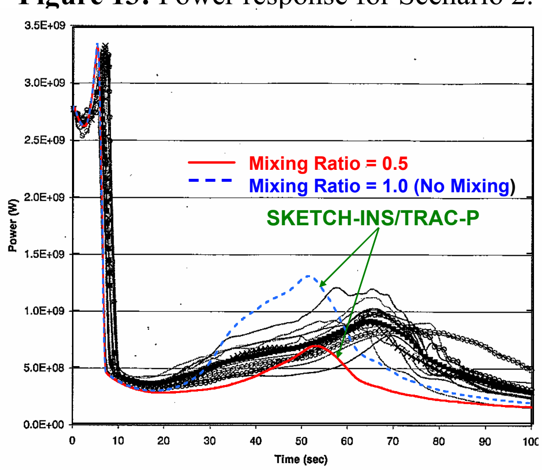


Figure 14: Total reactivity for Scenario 2.

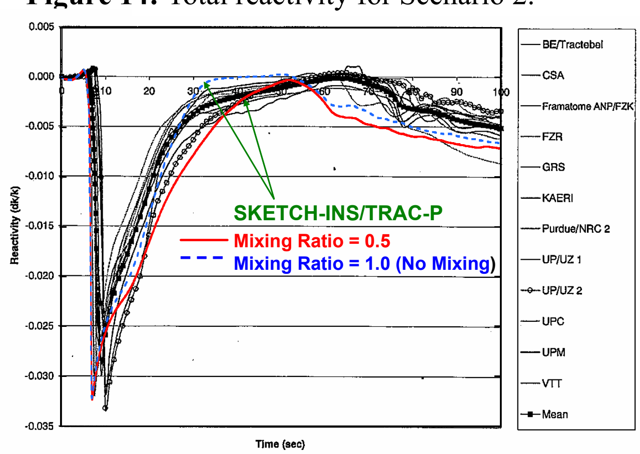


Figure 15: Broken SG coolant mass

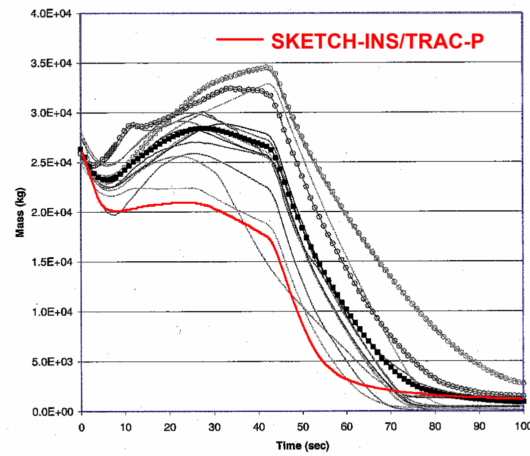
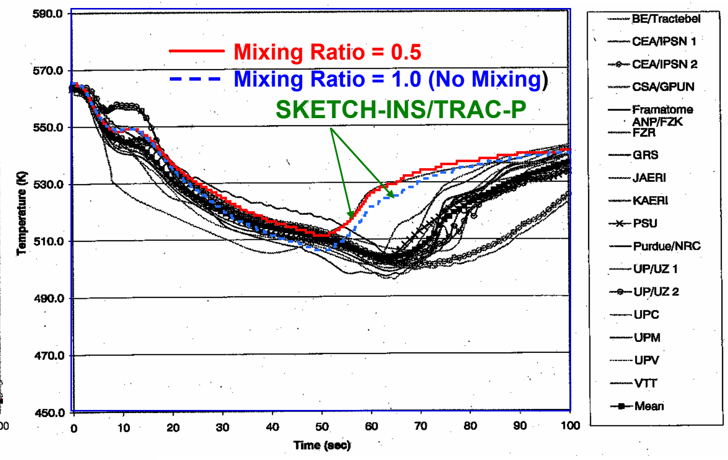


Figure 16: Broken loop cold leg temperature



4. Conclusion

The detailed dynamic thermal-hydraulic system analysis together with coupled detailed 3-D core kinetics will provide a more appropriate evaluation of the safety margins found in previous (licensing) simulations for which a point kinetics model was used. In order to improve the T-H/neutronics simulation capability of complex transients, the 3-D neutronics code SKETCH-INS was coupled with the thermal-hydraulic code TRAC-P. The capabilities of 3-D neutronics code SKETCH-INS and the coupled code SKETCH-INS/TRAC-P were verified against the benchmark problems. The results of analyses were in reasonable agreement with the reference and different codes results of benchmarks. The SKETCH-INS/TRAC-P code will be used to perform the detailed dynamic system analysis for evaluating the actual safety margins.

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