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Analysis of TMI-2 Benchmark Problem Using MAAP4.03 Code

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1. Introduction

Since the Three Mile Island Unit 2 (TMI-2) reactor accident, there have been extensive research activities to develop accident management strategies to mitigate the consequences from core-melting severe accidents [1]. As part of the TMI-2 analysis benchmark exercise sponsored by the Organization for Economic Cooperation and Development Nuclear Energy Agency (OECD/NEA), several member countries continue to improve their system analysis codes using the TMI-2 data [2]. The Republic of Korea joined this benchmark exercise in November 2005. Seoul National University has analyzed the TMI-2 accident as well as the currently proposed alternative scenario along with a sensitivity study using the Modular Accident Analysis Program Version 4.03 (MAAP4.03) code

2. Code Description

The MAAP4.03 primary system nodalization for tracking these quantities in a Babcock &Wilcox type plant is shown in Fig. 1. The reactor consists of four volumes: core, downcomer, upper plenum, and reactor dome. All loops except one, the broken loop, are lumped together, and the broken loop is treated separately. The broken loop refers to the loop that can contain a primary system break. The user selects whether the pressurizer is in the broken or unbroken loop. The A-loop is taken as the broken loop for analysis of the TMI-2 accident [3].

The reactor vessel is nodalized in the form of heat sinks and control volumes. For the core region the number of radial rings and axial rows is specified. A radial peaking factor and volume fraction are fixed for each ring, and an axial peaking factor is assigned for each row. Seven rings and thirteen rows are used to nodalize the TMI-2 core for simulation of the accident.

Mass and energy rates of change for core materials are calculated for each core node. Steam and hydrogen are assumed to flow along the uncovered and unblocked flow channels, and the mass flow rates and enthalpies in each channel are determined by accounting for the generation and consumption at each axial level. The core water pool is treated as a lumped mass and energy volume.

Figure 2 shows the containment nodalization and the flow paths used to track materials in the containment model. The cavity refers to the volume below the reactor vessel, the lower compartment to the volume below the operating deck and inside the crane wall, the annular compartment to the volume outside the crane wall below the operating deck, and last, the upper compartment to the volume above the operating deck. Within each of these volumes, the code tracks the thermofluid characteristics of steam, air, hydrogen, noncondensable gases and fission products. Within the lower compartment and the reactor cavity, corium and water are also accounted for.

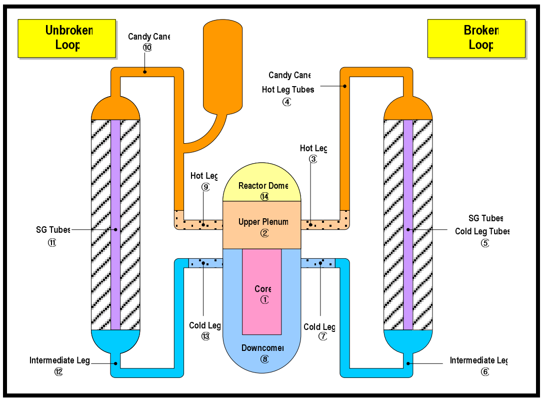

Figure 1. Primary system nodalization

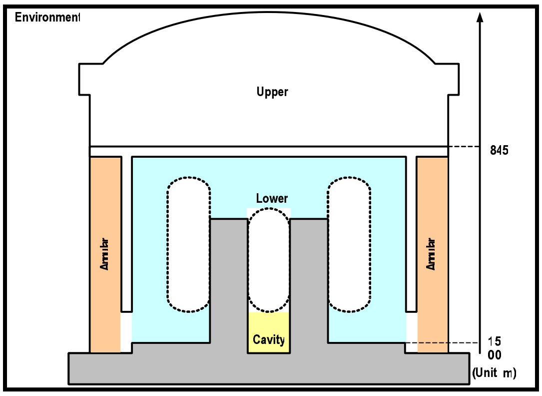

Figure 2. Containment nodalization

3. Results and Discussion

An accurate prediction of the TMI-2 transient relies on the proper definition of boundary conditions and plant characteristics. Having demonstrated the boundary conditions for the primary system, plant data for the primary system can be compared against the code results. The following data can be used for comparison with predictions: primary system pressure, pressurizer water level, broken steam generator pressure, water level. From the TMI-2 accident simulation point of view, the primary system pressure turns out to be a key parameter for comparison. The data provide with a continuing measure of the energy balance on the core, primary system and the two steam generators. As such, the pressure reflects the correctness of the boundary conditions as well as the adequacy of a code's thermal hydraulic models.

The calculated and TMI-2 standard reference primary system pressures are compared in Fig. 3. Generally good agreement with the reference is obtained during most of the simulated period. Particularly good agreement is observed from the start of the accident until 170 minutes, while relatively large deviations are noted thereafter.

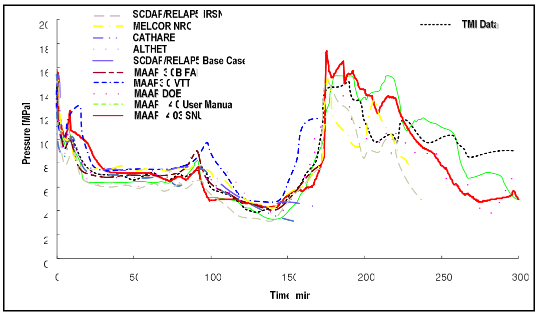

Figure 3. Primary System Pressure

The collapsed pressurizer level and the calculated level response are compared to the TMI-2 standard reference in Fig. 4. During the two phase discharge period through the PORV, the algorithm used in MAAP4.03 to calculate the pressurizer void fraction is iterative and produces oscillatory results. Therefore, the indicated level calculated for the period is fluctuating as well. However, as depicted in Fig. 5, this oscillatory behavior results in an average behavior that is in good agreement with the standard reference obtained throughout most of the transient.

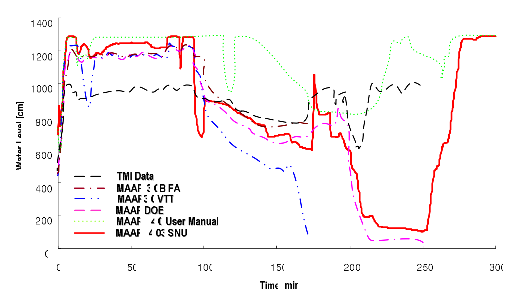

Figure 4. Pressurizer Level

The calculated broken steam generator pressure and TMI-2 standard reference are collected in Fig. 5. The secondary side pressures were approximately equal to the secondary side relief valve set point during most of the first 60 minutes of the accident. Thereafter, the atmospheric dump valves were opened and used to control the secondary pressure. This was simulated in MAAP4.03 by allowing the effective valve opening area for each steam generator to be changed at selected times between 60 and 174 minutes.

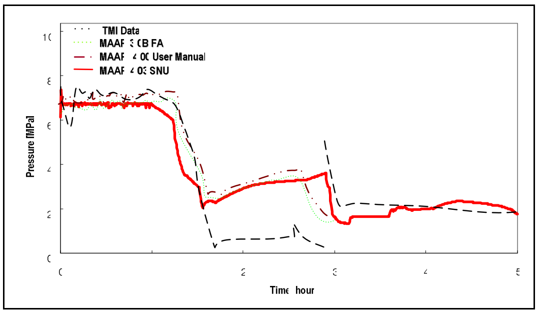

Figure 5. A-Loop Steam Generator Pressure

4. Conclusion

The present analysis showed general agreement with the standard reference of the TMI-2 accident. While the uncertainties in the boundary conditions render it difficult to draw unique quantitative conclusions regarding the core and the primary system behavior during a severe accident, understanding of the system trends and many other insights were gained from this analysis exercise. Many of the codes have difficulties in simulating the late phase of the TMI-2 accident.

ACKNOWLEDGMENTS

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