Development of MELCOR Modeling Input for SBO Analysis of APR1400 NPP

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1. Introduction

After the Fukushima Dai-ichi accident, realistic analysis for a Station Blackout (SBO) Accident has been stood out as a verification of the effectiveness of accident management strategies. In this study, an improved for severe accident analysis model using MELCOR program[1] was developed based on the previous study[2] in an attempt to analyze the behavior of the reactor coolant system (RCS) during a long-term SBO accident in the APR1400 NPP. The advanced MELCOR SBO model was compared and reviewed with a previous MELCOR model.

2. Methods and Results

The advanced SBO model has been prepared by reconstructing the previous MELCOR model in detail so that natural circulation and other possible phenomena occurring during a SBO can be simulated. The evaluation of the natural circulation with the new MELCOR model was conducted referring to the ShinKori 3&4 Final Safety Accident Report[3].

2.1 MELCOR Modeling

The detailed MELCOR model was built by referring to the method used in SOARCA (State-of-the-Art Reactor Consequence Analysis)[4] performed by the USNRC which has focused on providing a realistic evaluation of accident progression, source term, and offsite consequences for the Surry Nuclear Power Station.

2.1.1 Primary System Modeling

The SOARCA model provided an insight of the detailed primary system nodalization to figure out thermal hydraulic phenomena that may occur in the primary system under the accident condition. The followings were main considerations in the development of the advanced model.

- The hot leg was divided into two axialy separate pipes which consist of a upper pipe and a lower pipe to simulate a counter-current flow in the hot leg.
- In connection with the counter-current flow model, the primary side of the steam generator was divided into two separate flow paths. Each path was connected with one of the subdivided hot legs.
- The reactor vessel was subdivided into 59 control volumes to simulate local phenomena and temperature gradients.
- Other components such as the pressurizer and safety injection tank including the connection pipe were modeled in detail by comprising of more number of nodes.

Figure 1 shows the RCS primary system nodalization of the previous model that consists of few number of control volumes. Figure 2 illustrates the nodal configuration for the primary system, which is comprised of much more control volumes and flow paths so that natural circulation flow paths may be modeled during a severe accident process. Table 1 shows the comparison of number of nodes for the previous model and the developed model.

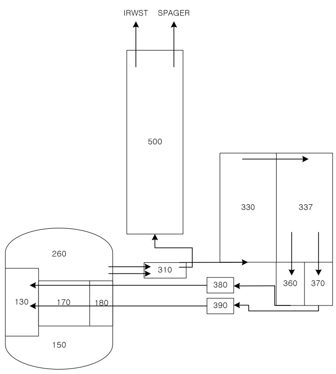

Figure 1. RCS Primary System Nodalization of the Previous Model

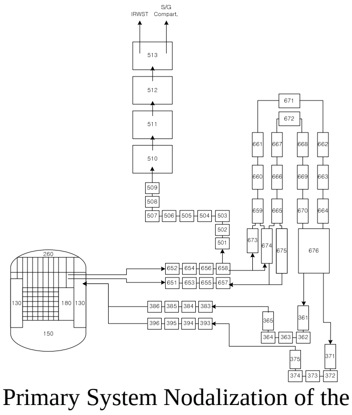

Figure 2. RCS Primary System Nodalization of the Developed Model

Table 1. Number of Control Volume in MELCOR models

	Previous model (EA)	Developed model(EA)
Core	1	35
Upper Plenum	1	24
Pressurizer	1	4
Pressurizer Surge line		9
Hot leg (A)	1	8
S/G Inlet Plenum (A)		3
S/G Inner Tube (A)	1	7
S/G Outer Tube (A)	1	7
S/G Outlet Plenum (A)	2	1
Loop Seal (A)		10
Cold leg (A)	2	8
S/G 2nd (A)	1	5
S/G 2nd Downcomer (A)	1	5

2.1.2 Modeling of Steam Generator Secondary System

The Steam Generator secondary volume was modeled to have ten nodes as shown in Fig. 2. The elevations of the nodes were equalized with those of the corresponding primary-side node. The nodes are connected as a closed loop so that the secondary system model can simulate flow circulation inside Steam Generator.

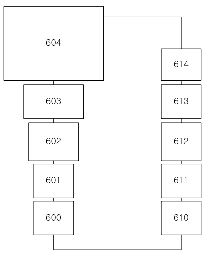

Figure 2. S/G Secondary System Nodalization

2.2 Accident Scenario

A steady-state analysis was performed for 1000 seconds at the full power operation condition. Then the SBO accident was simulated with a Reactor Coolant Pump (RCP) trip, main feedwater pump trip, reactor trip and startup of auxiliary feedwater supply.

3. Preliminary Analysis

3.1 SBO Accident Analysis

Figure 3 shows a comparison of the coolant temperatures between the previous model and the currently developed model. The previous model indicated only one temperature change since the core was modeled as one control volume, whereas the developed one evaluated the temperature differences at different locations.

The RCS pressure was compared for two MELCOR models as shown in Fig. 4. After one hour the developed model predicted lower pressure than the previous model because of the different amount of heat transfer through the steam generator.

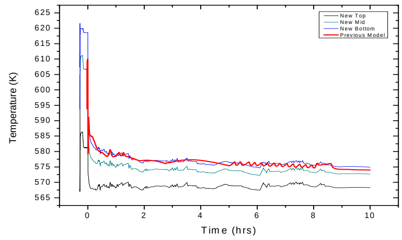

Figure 3. Coolant Temperature in core

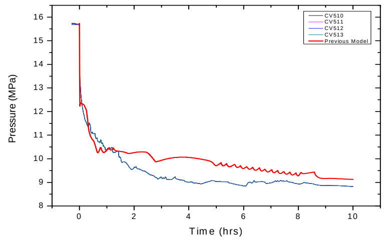

Figure 4. Pressurizer pressure

3.2 Evaluation of Natural Circulation

In the SBO accident, the natural circulation was estimated to be 4% of the steady-state flow rate in the full power operation as shown in Fig. 5, following the trend of the natural circulation reported in the ShinKori 3&4 FSAR. However, the developed model estimated one percent higher the FSAR prediction.

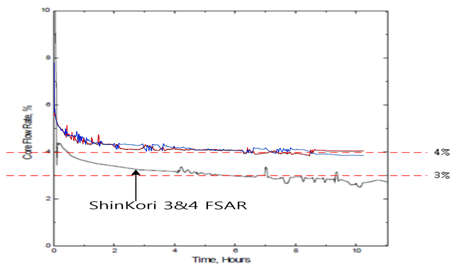

Figure 5. Comparison of Natural Circulation Flows

4. Conclusions

The developed MELCOR model made a reasonable agreement with the previous model and the FSAR evaluation. In the future the model will be further advanced by reviewing and modifying major input parameters in order to simulate realistic thermal-hydraulic phenomena during the SBO accident.

REFERENCES

[1] MELCOR Computer Code Manuals, Version 1.8.6
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[3] ShinKori 3&4 Final Safety Analysis Report
[4] NUREG/CR-7110, Vol.2 State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis