The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics, NURETH-14
Toronto, Ontario, Canada, September 25-30, 2011

NURETH14-122

RELAP5 ANALYSIS OF INTEGRAL TEST OF THE LOSS-OF-RHR EVENT DURING THE MID-LOOP OPERATION

M. Yamada[1], H. Nagumo[2] and I. Kinoshita[1]
[1] Institute of Nuclear Safety System, Inc., Fukui, Japan
[2] MHI Nuclear Engineering Co., Ltd., Kanagawa, Japan

Abstract

We are developing a statistical safety evaluation method using the RELAP5/MOD3.2 code for the loss-of-RHR (Residual Heat Removal) event during the mid-loop operation. To confirm the code prediction performance for gravity injection which is a one of the mitigation measures for this event, the Bethsy 6.9a test was analyzed using RELAP5/MOD3.2. In the analytical results, water mass flow rate into the pressurizer in the early period of the transient event was overpredicted. But water mass flow rate into the pressurizer was able to be decreased by artificially giving the circulation flow between core and the core bypass region.

Introduction

Probable frequency of the loss-of-RHR event during mid-loop operation is relatively high, and the event has actually occurred in PWR plants in the USA [1],[2]. Therefore, various experimental studies [3]-[6] and analysis of the event progression [7] have been undertaken to investigate thermal-hydraulic phenomena and effects of mitigation measures.

A more accurate analysis method is desirable to enhance reliability of mitigation measures and a statistical safety evaluation method with a best estimate analysis code, RELAP5/MOD3.2 for application to the loss-of-RHR event during mid-loop operation is being developed. By applying this statistical evaluation method, the uncertainties of evaluation results can be estimated quantitatively, and as a consequence, excessive conservatism can be reasonably removed to obtain evaluation results with enhanced reliability.

To confirm the code prediction performance for gravity injection which is one of the mitigation measures for this event, it is especially important to verify the prediction performance of the reactor cooling system (RCS) pressure and the reactor vessel water level which are assumed to be the evaluation parameters in the statistical safety evaluation method. Then, this paper reports on the analysis for the Bethsy 6.9a test which was done with RELAP5/MOD3.2 to verify code prediction performance.

1. Gravity injection

Mitigation measures for the loss-of-RHR event during mid-loop operation are reflux cooling by one or two steam generators (SGs) in the case of no openings in the RCS and gravity injection from the refueling water storage tank (RWST) by manual operation in the case of openings in the RCS.

A schematic depiction of gravity injection is shown in Figure 1. After the loss-of-RHR event, core coolant temperature rises and eventually boiling occurs. The steam generated in the core swells the mixture level in the reactor vessel and flows from the opening in the pressurizer top. Some water in the reactor vessel and hot leg may also flow into the pressurizer with the steam flow. Two-phase flow in the pressurizer separate into water and steam because the flow velocity of steam in the pressurizer is relatively slow and water accumulates in the pressurizer. The core uncovery occurs if mitigation measures are not taken because RCS mass inventory decreases due to the outflow of steam from the opening. To prevent core uncovery, gravity injection from the RWST is done by manual operation. The gravity injection flow rate depends greatly on RCS pressure because the driving force for gravity injection is small. The RCS pressure increases due to water accumulation in the pressurizer and pressure loss at the opening in the pressurizer and surge line, and it affects the gravity injection flow rate.

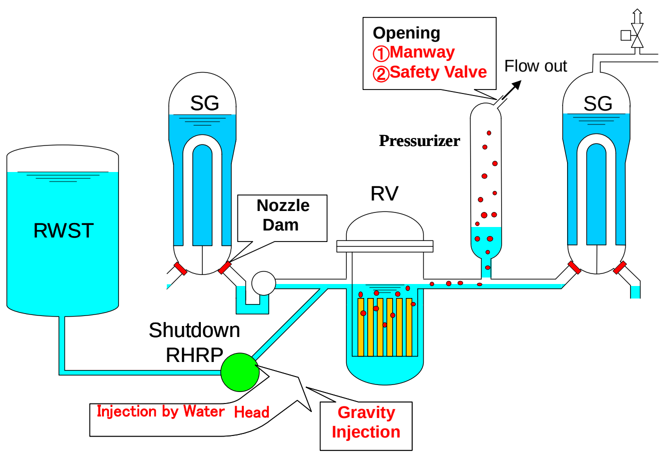

Figure 1 Schematic depiction of gravity injection

2. Analysis of the Bethsy 6.9a test

2.1 Bethsy 6.9a test[4]

The Bethsy facility is a 1/100 volumetrically-scaled, full-height model of a 3-loop (900 MWe) PWR. All loops are identical loops and each contains a SG and a primary coolant pump, but no RHR system.

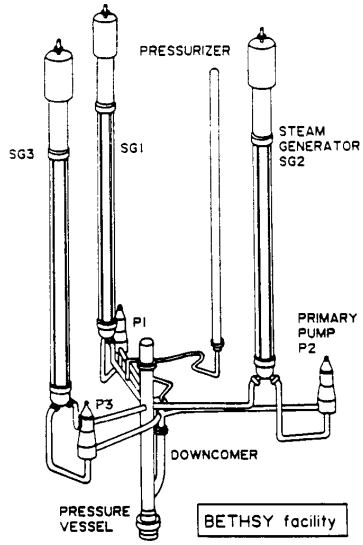

Figure 2 Schematic of Bethsy facility [8]

Table 1 Main conditions of Bethsy 6.9a test

Item	Primary side	Secondary side
1.Initial conditions		
Pressure	Atmospheric	Atmospheric
Fluid temperature	373K	-
Fluid level	Hot leg center	Filled with air
Noncondensable gas	Without	With
Pressurizer manway	Closed	-
Core power	0kW	-
2.Boundary conditions		
Core power	0.5%(140kW)	-
Pressurizer manway	Opened	-
Pressurizer	Connected to hot leg 1	-
Gravity injection point	Cold leg 1	-
Gravity injection temperature	313K	-

Figure 2 shows a schematic of the Bethsy facility. Table 1 shows main test conditions. In the initial conditions, the RCS was in the saturated condition at atmospheric pressure, and noncondensable gas was not present. The SG secondary side was filled with air and isolated. At t=0s the manway which is in the pressurizer top was opened while core power was rapidly raised to 140kW (0.5% NP). When the heater rod cladding temperature reached 523K, gravity injection was triggered. It is effective if the primary pressure in the cold leg is less than 0.15 MPa.

In the test, the primary two-phase level rose after the core power was raised, water flowed into the pressurizer, it accumulated, and RCS pressure rose. Steam flowed out from the pressurizer manway, and temporarily water flowed out too. RCS pressure decreased so that the amount of the steam generation in the core might decrease when the RCS mass inventory decreased due to the outflow from the pressurizer manway, and core uncovery would begin. The gravity injection started when the heater rod temperature rose due to the core uncovery and it reached 523K.

2.2 Analysis

In the analysis, prediction performance of the entire system was confirmed with the focus on the RCS pressure and the reactor vessel water level which are assumed to be the evaluation parameters by the statistical safety evaluation method.

2.2.1 Base case

The base case was executed as the BE analysis without consideration of the uncertainty of the analytical models. In the analysis, not only the hydraulic nodes but also the heat structures were modelled. The heat loss from the test facility was not modelled because it is the event where the temperature doesn't rise very much.

Differential pressure of the pressurizer is shown in Figure 3. The differential pressure rise of the analysis until about 1000s is larger than that of the test data. This is because more water flows into the pressurizer and more water accumulates there. After the differential pressure reaches the peak value (about 0.053MPa) at about 1300s, it decreases until about 1700s and then is steady with a value of about 0.042MPa. In the test data, the differential pressure rise in the initial 300s is faster and becomes gradual later. Differential pressure reaches the peak value (about 0.045MPa) at about 1900s; after that it decreases until about 2400s, and then is steady with a value of about 0.035MPa.

Mass flow rate through the pressurizer top is shown in Figure 4 and the integral of mass flow is shown in Figure 5. The mass flow rate analytical results are less than the test data for the vapor phase single-phase outflow period until about 1000s. The mass flow rate analytical results increase greatly from about 1000 to about 1700s, and are about 0.5kg/s at the peak. A lot of water flows out during this period in the analysis (Fig. 4). It is because more water reaches the pressurizer top because more water flows into the pressurizer. For the test results, the integral of mass flow rate is growing eminently from about 1600 to about 2400s, which suggests that the water is flowing out (Fig. 5), but the water mass flow rate is very small (Fig. 4). The inconsistency is seen for this period in the experimental data. The analytical result changes after about 1700s and the mass flow rate is about 0.06kg/s then. The amount of the steam generation from the core power (140kW) is 0.0629kg/s when the assumed pressure is 0.15MPa. The mass flow rate in the test after about 2400s is almost the same as the amount of the steam generation in the core. The analysis corresponds well with the test at this time (Fig. 4). Therefore, it was thought that the influence of the heat loss from the test facility was small though the heat loss was not modelled in the analysis. Then, mass flow rate decreases from about 3400s in the analysis result because the core uncovery occurs, and the amount of the steam generation decreases.

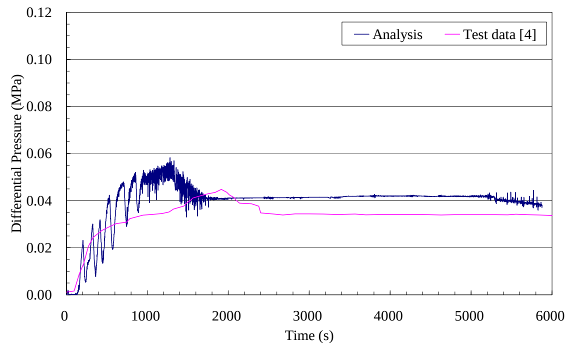

Figure 3 Differential pressure of pressurizer (base case)

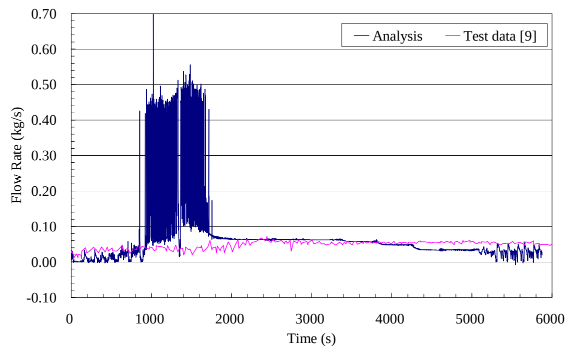

Figure 4 Mass flow rate through the pressurizer top (base case)

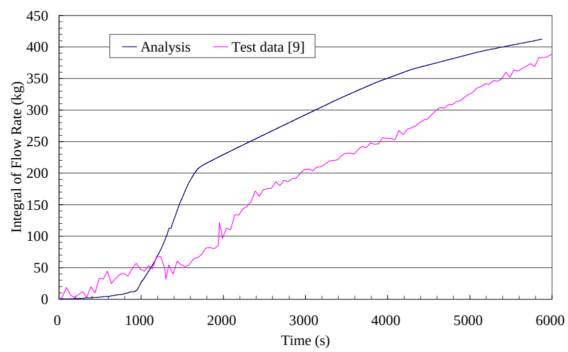

Figure 5 Integral of mass flow rate through the pressurizer top (base case)

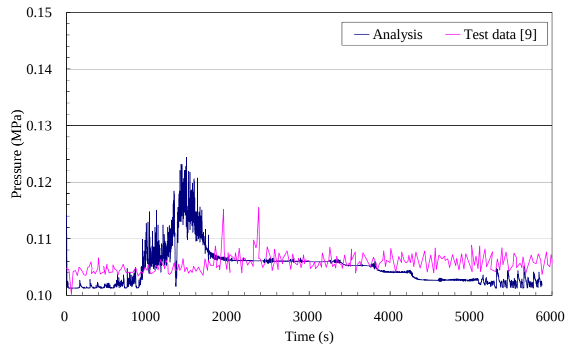

Figure 6 Pressurizer pressure (base case)

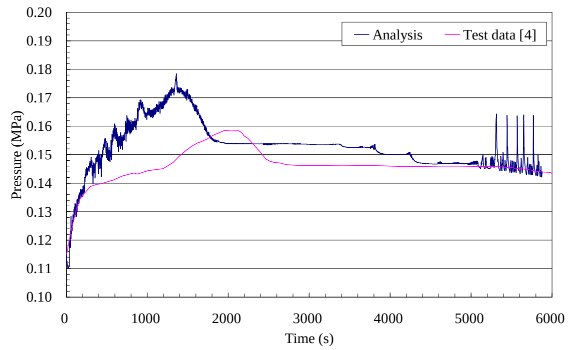

Figure 7 Upper plenum pressure (base case)

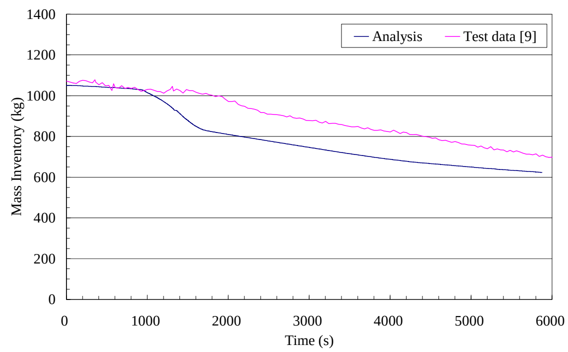

Figure 8 RCS mass inventory (base case)

Pressurizer pressure is shown in Figure 6. The analytical results rise rapidly from about 1000s, and exceed the test data greatly. The analytical results decrease to the same level as the test data at about 1700s. The period where the pressure of the analysis rises greatly corresponds to the period when the mass flow rate through the pressurizer top increases greatly. This is because pressure loss at the opening in the pressurizer top increases. In the analysis, pressurizer water level rises more than the test data indicate and more water also flows out from the opening than in the test data due to more water flows into the pressurizer than in the test data. In the test, water mass flow rate is small and pressure does not have great fluctuations either.

Upper plenum pressure is shown in Figure 7. The analytical result pressures are higher than the test data from about 200s until about 1800s. The analytical results are also higher than the test data peak pressures, and the analysis peaks at an earlier time. The analytical peak pressure (0.178MPa) is reached in about 1400s while test peak pressure (0.158MPa) is reached in about 2000s. The difference between the analytical and test values becomes smaller from about 2400s. However, the analytical results are higher (about 0.01MPa difference) than test data until about 4000s. The difference in the pressurizer differential pressure behavior influences these.

RCS mass inventory is shown in Figure 8. The analytical results and the test data correspond well for about the first 1000s. However, the distribution of coolant is presumed to be different between them; water accumulation in the pressurizer is large in the analysis, while water accumulation in the RCS, except for the pressurizer, is large in the test. In the analysis, RCS mass inventory decreases rapidly from about 1000 to 1700s because the water flows out from the pressurizer. However, the outflow of the water is delayed until about 1600s in the test. Moreover, because the outflow discharge is small, the decrease in the RCS mass inventory is also gradual.

It was thought that the upper plenum pressure of the analysis was higher than that of the test, and the RCS mass inventory was less than the test data because the water mass flow rate into the pressurizer was more than that indicated by the test data from the above-mentioned results. Core interfacial friction, upper plenum interfacial friction, hot leg horizontal stratified criterion, hot leg entrainment model, and pressurizer surge line CCFL model are regarded as parameters that influence water mass flow rate into the pressurizer.

Core interfacial friction influences the core mass inventory. Upper plenum interfacial friction influences the upper plenum mass inventory and the water mass flow rate from the reactor vessel to the hot leg both. The void fraction rises when the interfacial friction is large, and the water mass flow rate from the reactor vessel to the hot leg increases.

Hot leg horizontal stratified criterion influences the water mass flow rate from the hot let to the pressurizer surge line by changing the criterion under which the flow regime of the hot leg changes from nonstratified flow to horizontal stratified flow. The water mass flow rate from the hot leg to the pressurizer surge line decreases because the interfacial friction becomes smaller than the nonstratified flow when the flow is made stratified.

The hot leg entrainment model applies to the junction that connects the hot leg with the pressurizer surge line, and it influences the water mass flow rate from the hot leg to the pressurizer surge line through the junction quality.

The pressurizer surge line CCFL model influences mass flow rate from the pressurizer to the pressurizer surge line. The amount of the drop in the water mass flow rate from the pressurizer to the pressurizer surge line decreases when the CCFL criterion is severe, and the substantial water mass flow rate flows into the pressurizer increases.

The analyses changing these parameters within the range of uncertainty of the parameters were done, and the influence to the water mass flow rate into the pressurizer was confirmed. About three parameters (core interfacial friction, upper plenum interfacial friction, and hot leg horizontal stratified criterion), parameters were combined so that sensitivity may be large and the influence was confirmed. In the combination thought that the water mass flow rate into the pressurizer increased, three parameters were assumed to be 1.7, 1.7, and 0.5 [10] respectively. In the combination thought that the water mass flow rate into the pressurizer decreased, three parameters were assumed to be 0.3, 0.3, and 4.0 [10] respectively. But no large sensitivity was seen.

It was presumed that the water mass flow rate from the reactor vessel to the pressurizer in the test was less than that of the analysis because core average void fraction was smaller than that of the analysis. The core flow oscillated in the analytical result of the base case, and, as a result, there was a possibility that the boiling region in the core was broadened. Then, the core axial void fraction distribution was calculated by the hand calculation in the steady condition that assumed the pool boiling. The result was very steady with the analytical result. It could not explain the boiling region in the core was broadened only by the influence of the oscillation. Next, it was thought that the circulation flow in the reactor vessel was one of the factors that influenced the core boiling region. It was presumed that the boiling region in the test shifted up than a stagnant pool boiling because subcooled water moved into the upper part of core and boiled due to minute circulation flow. The saturation temperatures of each part of core are different depending on the water head difference. So it became subcooled water when saturated water in the upper part of core flowed into core bottom by circulation flow. This judged as a peculiar effect to the low power test. The circulation flow was natural circulation between the low temperature region and the high temperature region, and the low temperature region in the Bethsy facility was thought to be equivalent to the core bypass region.

The core heat generation is about 0.5% of the rated power, so if there is a mass flow rate of about 2.2kg/s at the flow velocity of about 5.3cm/s, the core boiling region rises to the center of the core. This mass flow rate can be achieved with a water head of about 1m. The amount of water that moves from the core and the upper plenum to the pressurizer decreases, and water accumulation in the pressurizer decreases if the core boiling region moves to the upper part of the core. The base case also models the core bypass flow path with another channel. But, the circulation flow by a minute driving force is erased by the pressure fluctuation according to boiling and vibration of the flow, and the influence of the circulation flow does not appear. Therefore, the sensitivity analyses that artificially gave the circulation flow in the constant mass flow rate were executed and the influence of the circulation flow was confirmed.

2.2.2 Sensitivity analysis case

Then, to confirm the influence of the circulation flow, the core bypass exit junction was changed to a flow boundary junction and mass flow rate corresponding to the circulation mass flow rate between the core and the core bypass was given in the direction from the upper plenum to the core bypass. To confirm the sensitivity in consideration of the uncertainty of the circulation flow rate, mass flow rate were assumed to be 3kg/s that was more than 2.2kg/s and 6kg/s doubled in addition.

The core average void fraction decreases when the circulation flow is modelled, especially in the lower part of core there is a small void fraction, and the boiling region moves to the upper part. Therefore, the mass inventory in the core increases, and the water mass flowing to the pressurizer decreases. The differential pressure of the pressurizer becomes small (Figure 9), and, as a result, the outflow discharge from the pressurizer top decreases (Figures 10 and 11). The pressurizer pressure decreases (Figure 12) because the water outflow from the pressurizer top decreases. The upper plenum pressure decreases and it approaches the test value (Figure 13) because the pressurizer pressure and the differential pressure of the pressurizer decrease. The water outflow from the pressurizer top decreases, the decrease in the RCS mass inventory becomes more gradual, and the inventory values approach the test data (Figure 14).

However, there are still differences between the test data and the analytical results, and further examination is necessary.

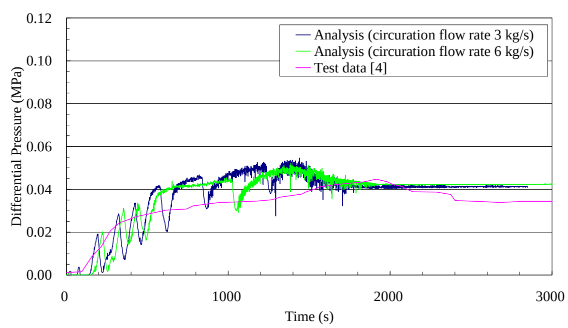

Figure 9 Differential pressure of pressurizer (sensitivity study case)

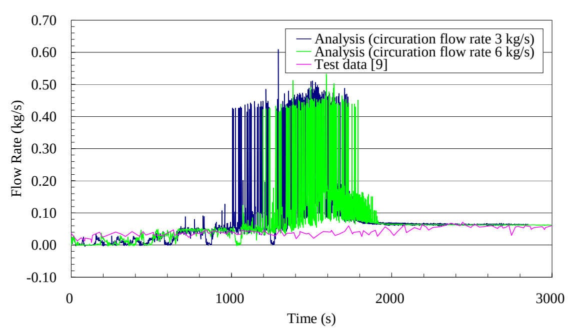

Figure 10 Mass flow rate through the pressurizer top (sensitivity study case)

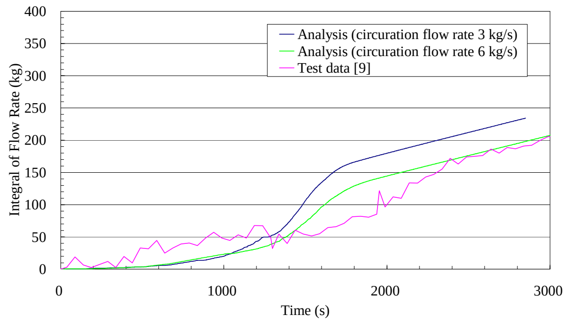

Figure 11 Integral of mass flow rate through the pressurizer top (sensitivity study case)

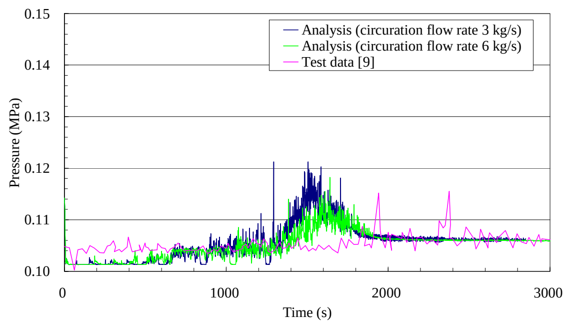

Figure 12 Pressurizer pressure (sensitivity study case)

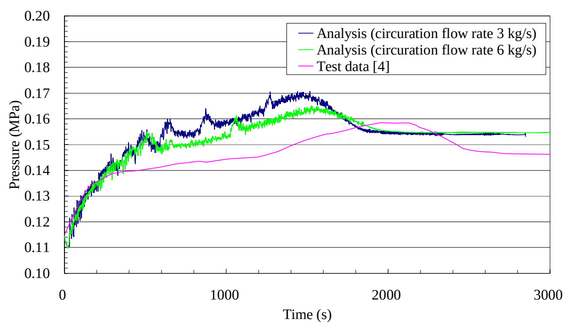

Figure 13 Upper plenum pressure (sensitivity study case)

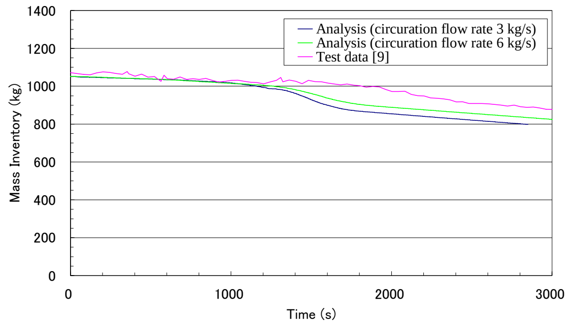

Figure 14 RCS mass inventory (sensitivity study case)

3. Conclusion

To confirm the code prediction performance for gravity injection which is a mitigation measure for the loss-of-RHR event during mid-loop operation, analysis of the Bethsy 6.9a test was carried out with the RELAP5/MOD3.2 code. The prediction performance of the RCS pressure and the reactor vessel water level which are assumed to be the evaluation parameters in the statistical safety evaluation method was especially considered.

In the base case which was executed as the BE analysis without consideration of the uncertainty of the analytical models, water accumulation in the pressurizer and mass flow rate through the pressurizer top were larger than in the test data because water mass flow rate into the pressurizer in the early period of the transient event was overpredicted by the code. The upper plenum pressure was overpredicted because water head due to water accumulation in the pressurizer and two-phase pressure loss by water outflow through the pressurizer top increased. Moreover, because the mass flow rate from the pressurizer top was more than the test data, the RCS mass inventory was underpredicted.

The analysis was done that changed parameters regarded as influencing water mass flow rate into the pressurizer (core interfacial friction, upper plenum interfacial friction, hot leg horizontal stratified criterion, hot leg entrainment model, and pressurizer surge line CCFL model) within the range of the uncertainty of the parameters. But large sensitivity to the water mass flow rate into the pressurizer was not seen.

In addition, the cause for overpredicting the water mass flow rate into the pressurizer in the base case was investigated and it was attributed to not being able to model the natural circulation between the low temperature region (core bypass) and the high temperature region (core). In the analysis that assumed the core bypass exit junction to be a flow boundary junction and modelled the circulation flow, the core void fraction decreased, especially there was a small void fraction in the lower part of the core, and the boiling region moved to the upper part. Therefore, the mass inventory in the core increased, and the water mass flowing to the pressurizer decreased.

However, there are still differences between the test data and the analytical results, and further examination is necessary.

4. Acknowledgment

The Bethsy facility design data for the analysis were provided by the OECD/NEA data bank. We wish to express our gratitude for providing the test data.

5. References

[1] USNRC, "Loss of residual heat removal system Diablo Canyon Unit 2 April 10, 1987", NUREG-1269, 1987.

[2] USNRC, "Loss of vital AC power and the residual heat removal system during mid-loop operations at Vogtle Unit 1 on March 20, 1990", NUREG-1410, 1990.

[3] H. Nakamura, J. Katayama and Y. Kukita, "Loss of Residual Heat Removal (RHR) event during PWR mid-loop operation: ROSA-IV/LSTF experiment without opening on primary loop pressure boundary", *FED ASME Power Plant Transients*, Vol. 140, 1992, pp. 9-16.

[4] D. Dumont, G. Lavialle, B. Noel and R. Deruaz, "Loss of residual heat removal during mid-loop operation: BETHSY experiments", *Nuc. Eng. Design*, Vol. 149, 1994, pp. 365-374.

[5] R. Mandl, K. Umminger and J. Logt, "Failure of PWR-RHRs under cold shutdown conditions experimental results from the PKL test facility", 18th Water Reactor Safety Research Information Meeting, 1990, pp. 329-351.

[6] C. Lee, T. Liu, Y. Way and D. Hsia, "Investigation of mid-loop operation with loss-of-RHR at INER Integral System Test (IIST) facility", *Nucl. Eng. Design*, Vol. 163, 1996, pp. 349-358.

[7] S.A. Naff, G.W. Johnsen, D.E. Palmrose, E.D. Hughes, C.M. Kullberg and W.C. Arcieri, "Thermal-hydraulics processes during reduced inventory operation with loss of residual heat removal", NUREG/CR-5855, 1992.

[8] Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency, "Bethsy experiment 9.1b 2" cold leg break without HPSI and with delayed ultimate procedure, comparison report", Vol.1, 1992.

[9] S. Bouabdallah, "RELAP5/MOD3.2 validation using BETHSY test 6.9a", NUREG/IA-0188, 2000.

[10] I. Kinoshita, H. Nagumo, M. Yamada, Y. Sasaki and Y. Yoshida, "Quantification of the RELAP5 model uncertainty for application to the loss-of-RHR event during the mid-loop operation", Proceedings of the 17th International Conference on Nuclear Engineering (ICONE17), ICONE17-75550, Brussels, Belgium, 2009 July 12-16.