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RELAP5/MOD3 Assessment Against a ROSA-IV/LSTF Loss-of-RHRS Experiment

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Abstract

An analysis of a loss of residual heat removal system (RHRS) event during midloop operation after reactor shutdown was performed using the RELAP5/MOD3 thermal-hydraulic computer code. The experimental data of a 5% cold leg break test conducted at the ROSA-IV Large Scale Test Facility (LSTF) to simulate a main coolant pump shaft seal removal event during midloop operation of a Westinghouse-type PWR were used in the analysis. The predicted core boiling time and the peak primary system pressure showed good agreements with the measured data. Some differences between the calculational results and the experimental results were, however, found in areas of the timing of loop seal clearing and the temperature distribution in a pressurizer. Other calculational problems identified were discussed as well.

1. Introduction

Thermal-hydraulic phenomena following a loss of residual heat removal (RHR) system during reactor cold shutdown have drawn much attention since recent risk studies have shown that abnormal events during reactor shutdown may have significant contribution to the reactor overall risks and several incidents involving loss of RHR system have been occurred during shutdown operation. To cope with these safety issues during reduced level operation, the U. S. Nuclear Regulatory Commission (NRC) has suggested expeditious and programmed actions including analyses for understanding of thermal-hydraulic behavior and nuclear steam supply system performance following a loss-of-RHR system in a generic letter by the NRC [1]. Various safety issues on loss-of-RHR events during reduced inventory operation were studied in detail to provide guidance for plant specific analyses by Naff *et. al.* [2], Ward *et. al.* [3], and Ward [4]. Sample representative calculations for three different plants needed to address identified safety issues were performed in those reports.

For inspection and maintenance of the steam generator U-tubes and/or reactor coolant pumps (RCP) seal, the liquid level of the reactor coolant system (RCS) is reduced to levels below the top of the loop piping, and the RHR system is used to remove the core decay power during this specific mode commonly called midloop operation. A loss-of-RHR system can be caused by a failure of a RHR pump due to noncondensable gas ingestion into the RHR suction line, or a loss of vital alternating current power, or inadvertent closure of the RHR isolation valves. Following the loss-of-RHR system, liquid boiling and steam generation due to the decay power begin in the core. Then, the primary system pressure can be increased by the accumulation of steam and the integrity of RCS temporary boundaries may be challenged from the increased system pressure.

Several experiments simulating loss-of-RHRS events during the midloop operation have been conducted at the ROSA-IV/LSTF facility as reported by Nakamura *et. al.* [5-8]. Main concerns of the experiments were the steam migration behavior in the RCS, the reflux cooling in the steam generator U-tubes and the influences of the opening location on the system integral responses. The experiments were conducted at various RCS configurations, *i.e.*, the open system with a break nozzle at the cold leg, or the hot leg, or the pressurizer, and at the closed system. In the present paper, the configuration with opening at the cold leg, which simulates the RCP seal removal, is of concern because of the prolonged vessel level depression until venting through the loop seal occurs.

2. Experiment and RELAP5 Model Descriptions

Initial conditions of the cold leg opening experiment attempted to simulate the system transients during maintenance of the RCP were summarized in Table 1. The area of the opening was equivalent to a 5 % of the cold leg pipe cross sectional area. The initial liquid level in the primary loop was set approximately at the centerline elevation of the horizontal legs. Both SG secondary sides were filled to the normal level (~10 m above the tube sheet) with water of room temperature. The core power was 0.6 % (430 kW) of the scaled nominal PWR power and was kept at this value throughout the experiment to simulate the decay power at approximately one day after the reactor shutdown. The primary coolant temperature was controlled using the RHR system at 334 and 318 K for the hot and cold legs, respectively. The coolant was taken out the nozzles at the bottom of both hot legs, pumped through a simulated RHR heat exchanger and injected into both cold legs through the ECCS nozzles. The initial pressure was atmospheric in both primary and secondary systems, with the relief valves on the pressurizer and SGs latched open. The upper portion of the primary and secondary systems above the water level was filled with air. To initiate the transient, a horizontally oriented opening was made in the cold leg in the loop without the pressurizer (loop B), and the RHR system from the primary loop and the pressurizer relief valves were isolated at the same time. No operator action was taken unless the core temperature exceeded ~700 K.

The RELAP5/MOD3 code was developed at the Idaho National Engineering Laboratory (INEL) to provide best-estimate predictions of postulated accidents and transients in light water reactor (LWR) systems. The code features a two-phase, two-fluid nonequilibrium hydrodynamic model with many generic component models and special process models. However, the code has limited capabilities to deal with noncondensable gases mixed with steam. Since thermal and mechanical equilibrium between noncondensable gas and steam were assumed, the code was not capable of simulating separate migration behavior for steam and noncondensable gas.

The RELAP5/MOD3 nodalization was composed of 142 volumes and 155 junctions. Heat transfer between components and environmental heat loss were modeled by 176 heat structures. A formation of multidimensional natural circulation without net flow through the core in the reactor vessel during the initial core coolant heat-up phase was observed in the experiment. The core nodalization composed of a pair of vertical channels connected by cross-flow junctions were used for better representation of the multidimensional circulatory flow, and form loss coefficients at the junction connecting the core to the lower plenum increased to reduce the manometric flow oscillation between the core and the downcomer. Some options such as the water packing and the vertical stratification were turned off at all junctions. The internal choking option was not used at all junctions except the break nozzle.

To obtain the initial steady state conditions, the RHR systems were modeled using a set of time dependent volumes and time dependent junctions at each loops. The steady state results summarized in Table 1 were obtained by running a transient calculation. Although flow oscillation of a small magnitude was revealed during the calculation, the predicted steady state conditions were almost identical to the data except fluid temperature at the upper head. Predicted water level at the hot leg and cold leg were at the mid-height of the piping, while the measured data were slightly below the mid-height level and were asymmetrical between two loops.

	Experiment	RELAP5
Primary side:		
Pressure (MPa)	0.1	0.1016
Temperature (K)		
Hot Leg	334.1	334.1
Cold Leg	317.7	318.0
Upper Head	334.2	317.9
Pressurizer	320.2	320.0
Core Power (kW)	430	430
Upper Plenum Level (m)	5.53	5.51
Average void Fraction		
Hot Leg	0.46	0.51
Cold Leg	0.37	0.49
Secondary side:		
Pressure (MPa)	0.1	0.1
Fluid Temperature (K)	317.2	317.2
Level (m)	1.11	10.0 (WR)

Table 1. Steady State Conditions

3. Results and Discussions

Much attention should be paid during analyzing the calculational results since mass errors at the condition of low pressure and low flow have been reported by many RELAP5 users. The mass conservation is checked after solving a system pressure matrix in the semi-implicit numerical solution scheme of RELAP5. A mixture density calculated from the state relations is compared with the mixture density calculated from the unexpanded form of the mixture continuity equation which is numerically mass preserving, and the difference between these two densities is used to control the time step. The mass error is confined to a small value at every time advancement. However, the cumulative error in the total mass increases monotonously, which is believed to be caused by a code deficiency. One of practical methods to overcome this difficulty is to use a small time step since most of the numerical truncation errors is generated during inversion process of a system matrix. A small time step increases the diagonal dominance of the matrix and reduces the numerical truncation error. In the present calculations, a time step of 0.001 second was taken although very long computer CPU time of about 90 hours for a transient at a HP-735 workstation was needed.

Most of the mass error occurred during boiling of the core liquid as shown in Fig.1. The system total mass error of the present calculation was within the 7 % with respect to the initial fluid mass in the primary system. The largest mass error at every time step occurred at the components which the steam was interfaced with the noncondensable gas. The volumes which had the largest mass error were the upper plenum shortly after boiling in the core, the steam inlet plena and the pressurizer bottom.

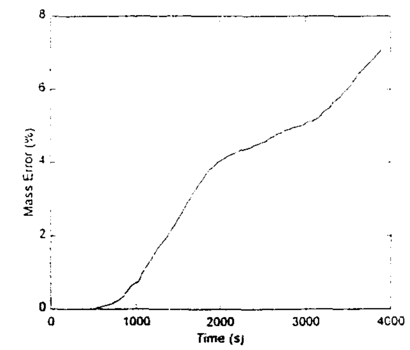

Fig.1 Proportion of the Cumulative Mass Error to the Initial Mass in the Primary System

The calculated primary pressures in both the hot leg and the cold leg agreed reasonably with the experimental results as shown in Fig. 2. Following the transient initiation, the liquid temperature in the core began to rise and started to boil at about 500 s. The hot leg pressure increased monotonously until the steam generated in the core began to discharge to the upper head through the guide tubes. The pressurization rate for this period predicted well except the pressure oscillation associated with large void fraction changes in the core. A pressure drop due to the steam discharge into the upper head was observed in the experiment, while the pressurization rate just reduced in the calculation. The pressurization rate increased again after the upper head was filled by steam, however, it reduced soon because of heat transfer to the secondary side. This reduced pressurization rate resulted in a delayed LSC. The pressure decreased rapidly owing to a circulatory flow due to the LSC. The pressure increase in the cold leg during the boiling phase was underpredicted, while the peak pressure obtained at just before the LSC was predicted well.

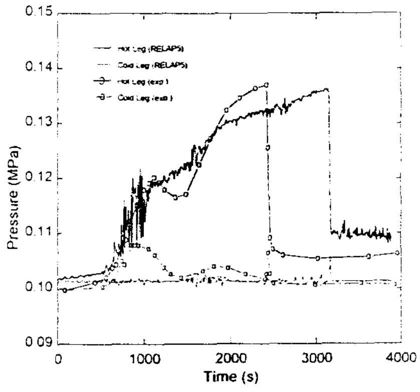

Fig.2 Pressures at the Broken Loop

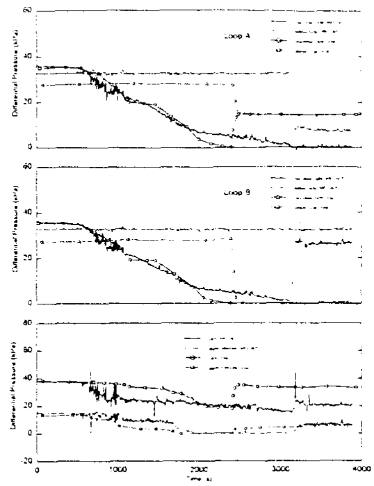

Fig.3 Differential Pressures in the Crossover Legs and the Reactor Vessel

The differential pressures at the crossover legs and the reactor vessel were plotted in Fig. 3. The calculated liquid level decrease in the loop seal downflow side agreed fairly well with the data. Differences were associated with the system pressurization rate affected by steam venting into the upper head and condensation heat transfer in the U-tubes. The LSC occurred partially in loop A and completely in loop B in the experiment, while it occurred only in loop A in the calculation. The upper plenum liquid level was overpredicted, but the core liquid level was underpredicted because of the overpredicted void fraction in the core and the mass error. The temperature excursion observed at the high power bundle was not predicted because the fuel rods were modeled at the average power level. The core liquid level in the core was recovered by the liquid moved from the crossover leg for the course of the LSC.

4. Conclusions

An assessmental study of the RELAP5/MOD3 code was performed by comparing with the ROSA-IV/LSTF loss-of-RHR experiment with a 5 % cold leg break during the midloop operation. The calculational difficulties caused at the system conditions characterized by low pressure, low core power and the presence of a noncondensable gas, were successfully overcome. All major thermal-hydraulic phenomena such as the primary system pressurization were predicted qualitatively well. In the calculation, however, flow oscillations during the boiling phase and a delayed loop seal clearing occurred. It was found that those problems were associated with limitations of the one dimensional code to simulate multidimensional phenomena and with the numerical truncation errors.

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