RELAP5 Analyses of OECD/NEA ROSA-2 Project Experiments on Intermediate-Break LOCAs at Hot Leg or Cold Leg*

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Abstract

Experiments simulating PWR intermediate-break loss-of-coolant accidents (IBLOCAs) with 17% break at hot leg or cold leg were conducted in OECD/NEA ROSA-2 Project using the Large Scale Test Facility (LSTF). In the hot leg IBLOCA test, core uncover started simultaneously with liquid level drop in crossover leg downflow-side before loop seal clearing (LSC) induced by steam condensation on accumulator coolant injected into cold leg. Water remained on upper core plate in upper plenum due to counter-current flow limiting (CCFL) because of significant upward steam flow from the core. In the cold leg IBLOCA test, core dryout took place due to rapid liquid level drop in the core before LSC. Liquid was accumulated in upper plenum, steam generator (SG) U-tube upflow-side and SG inlet plenum before the LSC due to CCFL by high velocity vapor flow, causing enhanced decrease in the core liquid level. The RELAP5/MOD3.2.1.2 post-test analyses of the two LSTF experiments were performed employing critical flow model in the code with a discharge coefficient of 1.0. In the hot leg IBLOCA case, cladding surface temperature of simulated fuel rods was underpredicted due to overprediction of core liquid level after the core uncover. In the cold leg IBLOCA case, the cladding surface temperature was underpredicted too due to later core uncover than in the experiment. These may suggest that the code has remaining problems in proper prediction of primary coolant distribution.

Key words: PWR, ROSA/LSTF, Hot Leg, Cold Leg, Intermediate-Break LOCA, RELAP5 Code

1. Introduction

The investigation on pipe integrity during long-term operation and for life extension of light water reactors (LWRs) has revealed that the occurrence frequency of the pipe break may depend on the pipe size such that complete rupture of a smaller pipe would be more likely to occur than double-ended guillotine break (DEGB) of a large main pipe. The frequency of DEGB of hot and cold legs of pressurized water reactors (PWRs), leading to large-break loss-of-coolant accident (LOCA), would then be quite low. The consideration of rupture of intermediate size piping should become relatively more important than ever in risk informed regulation-relevant safety analyses⁽¹⁾.

The United States Nuclear Regulatory Commission⁽²⁾ proposed risk informed changes to LOCA technical requirements, Appendix K to 10 CFR Part 50, that intermediate-break

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LOCA (IBLOCA) is chosen as a design basis event (DBE) for the assessment of effectiveness of emergency core cooling system (ECCS). For PWRs, DEGB of pressurizer (PZR) surge line connected to one of hot legs and that of one of ECCS lines connected to cold legs are considered as representatives for break location and size of IBLOCA. Many test facilities have contributed to the thermal-hydraulic databases available today especially for LOCAs and operational transients which have been extensively used for the validation of computer codes. The experimental data for IBLOCA (with a break size of over 10%⁽³⁾), however, are quite limited. For the realistic evaluation of safety margin in the IBLOCAs, databases to understand the phenomena and validate analysis methods including best-estimate computer code (e.g., RELAP5 code) are needed by means of systematic experiments simulating IBLOCAs.

Experiments on PWR IBLOCAs with 17% break at hot leg or cold leg to simulate DEGB of PZR surge line or DEGB of one of ECCS lines were conducted using the Large Scale Test Facility (LSTF)⁽⁴⁾ within OECD/NEA ROSA-2 Project. The LSTF simulates a Westinghouse-type four-loop 3423 MW (thermal) PWR by a full-height and 1/48 volumetrically-scaled two-loop system. The main test objectives are to clarify thermal-hydraulic phenomena peculiar to IBLOCAs and to obtain the detailed data which are suitable for the assessment of thermal-hydraulic safety analysis computer codes. The two LSTF IBLOCA experiments were analyzed by RELAP5/MOD3.2.1.2 code⁽⁵⁾ to validate the code predictability. This paper describes major observations in the LSTF experiments to understand thermal-hydraulic responses during the IBLOCAs at hot leg or cold leg with the RELAP5 post-test analysis results. It is noted that the value axis of all the data plot of LSTF test and RELAP5 post-test analysis results are normalized properly due to the agreement of the OECD/NEA ROSA-2 Project.

2. OECD/NEA ROSA-2 Project

The JAEA started OECD/NEA ROSA-2 Project in 2009 following the ROSA Project⁽⁶⁾ to resolve issues in thermal-hydraulic analyses relevant to LWR safety by using the LSTF of JAEA. Eighteen organizations from 14 NEA member countries have joined the Project to date. In particular, the ROSA-2 Project intends to focus on the validation of simulation models and methods for complex phenomena such as coolant mixing and thermal stratification of highly safety relevance for thermal-hydraulic transients in DBEs and beyond-DBE. The experimental program is defined to provide valuable and broadly usable database to achieve the objectives agreed among participants.

The OECD/NEA ROSA-2 Project consists of six LSTF experiments that mainly include two groups of experiments which are IBLOCAs and steam generator tube rupture (SGTR) accidents. IBLOCA with a break at either hot leg or cold leg of PWR is dealt with as one of important safety issues. Both an accident due to a SGTR and the SGTR accident induced by main steam line break are simulated to evaluate amount of fission product (FP) release and to consider better accident management actions to minimize the FP release.

Fig. 1 Schematic view of ROSA/LSTF

3. ROSA/LSTF

The LSTF simulates a Westinghouse-type four-loop 3423 MW (thermal) PWR by a two-loop system model with full-height and 1/48 in volume. The reference PWR is Tsuruga Unit-2 of Japan Atomic Power Company. Figure 1 shows the schematic view of the LSTF,

being composed of an active SG with 141 full-size U-tubes (inner diameter of 19.6 mm each), primary coolant pump, hot and cold legs in each loop. The inner diameter of hot and cold legs is 207 mm to conserve the volumetric scale (2/48) and the ratio of the length to the square root of pipe diameter to better simulate flow regime transitions in the primary loops (Froude number basis)⁽⁷⁾. Details of the design philosophy are presented in Reference (8).

The LSTF core, 3.66 m in active height, consists of 1008 electrically heated rods in 24 rod bundles to simulate the fuel rod assembly in the reference PWR. Axial core power profile is a 9-step chopped cosine with a peaking factor of 1.495. The LSTF initial core power of 10 MW corresponds to 14% of the volumetric-scaled (1/48) PWR nominal core power because of a limitation in the capacity of power supply. About 1760 instruments are equipped in the LSTF to understand and evaluate thermal-hydraulic responses during simulated accidents and transients.

4. Experiment and Code Analysis Results

4.1 Common Conditions of LSTF Experiments

The 17% break was simulated by using a nozzle, upwardly mounted flush with the leg inner surface, in the loop without PZR (broken loop) as shown in Fig. 2. The break simulation nozzle is 41 mm in inner-diameter (d) and 492 mm in straight portion length (L). The ratio of L to d is designed to be 12 to avoid significant influences of the length onto mass flux through the nozzle⁽⁹⁾. The nozzle flow area corresponds to 17% of the volumetric-scaled cross-sectional area of the reference PWR cold leg.

The experiment was initiated by opening a break valve located downstream of the break nozzle at time zero. Initial steady-state conditions such as primary pressure, fluid temperatures in hot and cold legs were 15.5 MPa, 598 K and 562 K, respectively, according to the reference PWR conditions.

The LSTF core power decay curve after a scram signal was pre-determined based on calculations with the RELAP5 code considering delayed neutron fission power and stored heat in PWR fuel rod⁽¹⁰⁾. The LSTF core power was maintained at the initial value of 10 MW for 18 s after the scram signal until the scaled PWR core decay power dropped to 10 MW. The LSTF core power started to decay afterwards according to the specified core power. To obtain prototypical initial fluid temperatures with this core power, core flow rate was set to 14% of the scaled nominal flow rate. Initial SG secondary pressure was raised to 7.3 MPa to limit the primary-to-secondary heat transfer rate to 10 MW, while 6.1 MPa is nominal value in the reference PWR. Set point pressures for opening and closure of SG relief valves (RVs) are 8.03 and 7.82 MPa respectively, referring to the corresponding values in the reference PWR. The experiments assumed loss of off-site power concurrent with scram and total failure of auxiliary feedwater.

4.2 RELAP5 Analysis Conditions

Figure 3 shows a nodding schematic of LSTF for RELAP5 analysis in the hot leg IBLOCA case. The LSTF system is modeled in one-dimensional manner including pressure vessel, primary loops, PZR, SGs and SG secondary-side system. The break unit connected to the

Fig. 2 Schematic view of LSTF break unit

top of the broken hot leg (Fig. 2) was modeled by a vertical pipe, horizontal pipe and oblique pipe between them to simulate corresponding facility configuration. The break nozzle in the vertical pipe was represented by five equal-volume nodes. The analysis employed critical flow model in the code, which may correctly predict the discharge rate through the long break nozzle, with a discharge coefficient (C_d) of 1.0.

The core was represented by vertically-stacked nine equal-height volumes according to 9-step chopped cosine in axial core power profile along the core length. The radial power distribution was then given considering the peaking factor and the number of high-, mean- and low-power rod bundles. Following Wallis correlation⁽¹¹⁾ was applied to simulate counter-current flow limiting (CCFL) at core exit;

$$j_G^{*1/2} + m j_L^{*1/2} = C, \quad (1)$$

where j^* is the non-dimensional volumetric flux. Subscripts G and L denote gas and liquid phases, respectively. Constants of m and C for CCFL at the SG U-tube inlet were 1 and 0.75, respectively, obtained from LSTF test⁽¹²⁾.

These values were selected as the constants in this analysis considering a tendency that water remains on upper core plate in upper plenum to be described in Section 4.3.2, though flow channel structure of the core exit is different from that of the SG U-tube inlet.

Break conditions such as break location and size greatly affect thermal-hydraulic responses. Precise predictions especially for break flow rate and core liquid level may be needed to correctly predict the PCT, which is limited for fuel rod integrity in safety analyses. Prior to the analysis of 17% hot leg IBLOCA test, an analysis with RELAP5 code was thus conducted for LSTF 10% hot leg break LOCA test⁽¹³⁾ as the nearest-size break test to validate the code predictability. The RELAP5 analysis revealed that the break flow rate was predicted reasonably well while the core liquid level after core uncover was overpredicted. Kumamaru et al.⁽¹⁴⁾ have pointed out that the RELAP5 code should overpredict gas-liquid inter-phase drag in the core. Larger inter-phase drag in the core results in overprediction of the core liquid level. In the analysis of 17% hot leg IBLOCA test, inter-phase drag in each node of core was thus reduced to 1/10, referring to the Kumamaru's approach⁽¹⁴⁾.

Conditions for the cold leg IBLOCA analysis are described below. Employed critical flow model, C_d and inter-phase drag in the core were the same as those of the hot leg IBLOCA analysis. The nodalization was the same too except for the break unit connected to the top of the broken cold leg. The Wallis correlation in Eq. (1) was applied to simulate CCFL at the core exit, SG U-tube and SG inlet plenum, based on the test results to be shown in Section 4.4.2. Constants of m and C in the CCFL correlation were given as 1 and 0.75, respectively, similarly to the hot leg IBLOCA analysis, as trial values irrespective of flow channel structure.

Other initial and boundary conditions were determined according to the LSTF test data.

4.3 17% Hot Leg IBLOCA

4.3.1 LSTF test conditions

PZR connected to the intact loop was isolated before the test initiation to avoid influences

Fig. 3 Noding schematic of LSTF for RELAP5 analysis (hot leg IBLOCA case)

of coolant in the PZR onto the primary coolant inventory involved in the LOCA transient. Since the reactor control logic is usually based on the PZR pressure, core scram timing for this particular test was assumed to occur at 4 s after the break. As for ECCS conditions, accumulator (ACC) and low pressure injection (LPI) flow rate was designed to become 3:1 to cold legs of intact and broken loops, respectively, as well as total failure of high pressure injection (HPI) system.

4.3.2 LSTF test results and comparison with code analysis results

The LSTF experiment results are shown in Figs. 4 through 12, being compared with RELAP5 post-test analysis results for such parameters as the primary and secondary pressures, break flow rate, liquid levels in broken hot and cold legs, flow rates of intact ECCS, collapsed liquid levels in intact crossover leg, upper plenum and core, and cladding surface temperature. In these figures, the normalized value of 1 corresponds to the initial or maximum value obtained through the post-test analysis.

Observations in the Experiment

The break flow rate is derived from the differential of the time-integrated break flow evaluated from liquid level increase in the break flow storage tank. Break flow turned from single-phase liquid to two-phase flow in a very short time after the break. Rather large size of break caused a fast primary depressurization. The scram signal caused the closure of SG main steam isolation valves (MSIVs) and the coastdown of primary coolant pumps. The SG secondary-side pressure increased rapidly up to 8 MPa after the closure of the SG MSIVs. The SGs no longer served as the heat sink after the primary pressure became lower than the SG secondary-side pressure.

Liquid levels appeared in upper plenum and hot leg due to flashing of fluid because of the fast primary depressurization. Liquid level appeared in the cold leg when the primary pressure became lower than the SG secondary-side pressure. Water remained on the upper core plate in the upper plenum due to CCFL because of significant upward steam flow from the core to the break, affecting cladding surface temperature of simulated fuel rods. Crossover leg upflow-side once became empty of liquid at 140 s due to steam by flashing of cold leg fluid. Core uncover started at 165 s simultaneously with liquid level drop in crossover leg downflow-side before loop seal clearing (LSC). The LSC was induced due to manometric level change in the crossover leg because of the pressure difference between the hot and cold legs by steam condensation on ACC coolant injected into the cold leg at 155 s after the primary pressure decreased to 4.51 MPa. Core liquid level rapidly recovered after the LSC took place at 180 s. The crossover leg upflow-side became voided by 195 s.

The peak cladding surface temperature (PCT) of 607 K was observed at Position 5 (=core center), which was lower than the initial temperature of 623 K. Due to the ACC coolant injection, cold legs became refilled with subcooled water. The hot leg liquid level recovered temporarily and greatly after the whole core recovery. After the termination of ACC coolant injection, the LPI system was actuated at 505 s when the primary pressure decreased to 1.24 MPa. The hot leg IBLOCA test was terminated when the primary and SG secondary-side pressures reached nearly-equilibrium condition with well-cooled core after the LPI actuation.

Fig. 4 LSTF and RELAP5 results for primary and secondary pressures in hot leg IBLOCA case

Fig. 5 LSTF and RELAP5 results for break flow rate in hot leg IBLOCA case

Fig. 6 LSTF and RELAP5 results for broken hot leg liquid level in hot leg IBLOCA case

Fig. 7 LSTF and RELAP5 results for broken cold leg liquid level in hot leg IBLOCA case

Fig. 8 LSTF and RELAP5 results for flow rates of intact ECCS in hot leg IBLOCA case

Fig. 9 LSTF and RELAP5 results for intact crossover leg collapsed liquid level in hot leg IBLOCA case

Fig. 10 LSTF and RELAP5 results for upper plenum collapsed liquid level in hot leg IBLOCA case

Fig. 11 LSTF and RELAP5 results for core collapsed liquid level in hot leg IBLOCA case

Fig. 12 LSTF and RELAP5 results for cladding surface temperature in hot leg IBLOCA case

Comparison of Analysis Results

The RELAP5 calculated results agreed reasonably well with the primary parameters measured in the LSTF experiment due to good prediction of break flow as has been compared in Figs. 4 through 12. LSC induced by steam condensation on ACC coolant injection appeared slightly earlier than in the experiment due to underprediction of the crossover leg liquid level because of a bit earlier initiation of ACC system. Some discrepancies from the test data appeared in the timings of empty of liquid in the crossover leg upflow-side due to influences of the difference between the test data and prediction for the hot leg liquid level. The liquid levels started to drop in the core, upper plenum and hot leg earlier than in the experiment due to earlier termination of ACC coolant injection.

Discussion

Influences of analysis conditions, especially focusing on CCFL at core exit and inter-phase drag in the core, onto the collapsed liquid levels in the upper plenum and core and cladding surface temperature were investigated further as shown in Figs. 10 through 12. The case with the CCFL at the core exit and reduction of inter-phase drag to 1/10 in the core is defined as 'base case'. The constants of m and C in the CCFL correlation are then 1 and 0.75, respectively.

In the case with no CCFL at core exit different from 'base case', the cladding surface temperature at Position 7 (=about 0.8 m above the core center) started to increase at 155 s. The PCT was 547 K due to overprediction of the core liquid level, while the upper plenum became empty of liquid. In the case with original inter-phase drag in the core different from 'base case', the cladding surface temperature at Position 7 began to increase at 150 s and the PCT reached 582 K. By considering the CCFL at the core exit and reduction of inter-phase drag to 1/10 in the core as 'base case', the cladding surface temperature at Position 8 (=about 1.2 m above the core center) started to increase at 140 s and the PCT reached 590 K. The core liquid level after core uncover was overpredicted due to earlier LSC because of earlier initiation of ACC coolant injection, though the upper plenum liquid level was overpredicted. The elevation where the PCT appeared in the analysis was about 1.2 m above the elevation in the experiment due to overprediction of the core liquid level.

Improvement of constants of the CCFL correlation at the core exit and inter-phase drag model in the core as well as condensation heat transfer model under ACC coolant injection condition at the cold leg may be necessary for better prediction, as the remaining subjects.

4.4 17% Cold Leg IBLOCA

4.4.1 LSTF test conditions

The scram signal was generated when the PZR pressure decreased to 12.97 MPa. As for ECCS conditions, HPI, ACC and LPI systems were actuated in the intact loop only to simulate the ECCS line break. Further assumption was single-failure of diesel generators related to both the HPI and LPI flow rates.

4.4.2 LSTF test results and comparison with code analysis results

The LSTF experiment results are shown in Figs. 13 through 22, being compared with RELAP5 post-test analysis results in a similar way to the hot leg IBLOCA case. Compared parameters are the primary and secondary pressures, break flow rate, liquid levels in broken hot and cold legs, flow rates of intact ECCS, collapsed liquid levels in intact crossover leg, inlet plenum and U-tube (typically medium tube) of intact SG, upper plenum and core, and cladding surface temperature. In these figures, the normalized value of 1 corresponds to the initial or maximum value obtained through the post-test analysis, similar to the hot leg IBLOCA case.

Observations in the Experiment

Break flow turned from single-phase liquid to two-phase flow in a very short time after the break, similar to the hot leg IBLOCA test. Primary depressurization was faster than in the hot leg IBLOCA test due to larger break flow rate. The SG RVs were opened for about 30 s due to the SG secondary-side pressure increase after the MSIVs closure. The primary pressure soon became far lower than the SG secondary-side pressure. The SG U-tube liquid level started to decrease greatly due to the fast primary depressurization and steam flow from the inlet plenum soon after the break. The liquid level decrease in the SG U-tube downflow-side continued down to the crossover leg to cause LSC. Liquid levels started to decrease in the upper plenum and core at 25 s simultaneously with liquid level drop in the crossover leg downflow-side. Core dryout took place at 35 s due to rapid liquid level drop in the core before the LSC, resulting in the temperature excursion in the core. The coolant flowed towards the cold leg through the hot leg, inlet plenum and U-tube of SG from the core. Liquid was accumulated in upper plenum, SG U-tube upflow-side and SG inlet plenum before the LSC due to CCFL by high velocity vapor flow, causing further decrease in the core liquid level. Water remained on the upper core plate in the upper plenum due to CCFL, similar to the hot leg IBLOCA test. The HPI system was started almost simultaneously with the core dryout, but was ineffective on the core cooling because the injection flow rate was far smaller than break flow rate. After the LSC took place at 40 s, water remained in the crossover leg upflow-side moved to the cold leg gradually by steam flow towards the break. The crossover leg upflow-side became voided by 100 s.

The core bottom-up quench started after the incipience of ACC coolant injection at 110 s. A temperature excursion, however, continued in the core even after the initiation of reflooding due to the ACC coolant injection. Therefore, core power was decreased to a certain low level in several steps to protect the LSTF core after the maximum cladding surface temperature exceeded pre-determined criterion of 958 K. The PCT of 978 K was observed at Position 6 (=about 0.4 m above the core center). The core power decreased down to 5% of the pre-determined power level as the PCT reached 978 K. Whole core was quenched due to reflooding. Due to the ACC coolant injection, the cold leg became refilled with subcooled water in the intact loop, but the liquid level fluctuated greatly in the broken loop. The hot leg became empty of liquid temporarily after the incipience of ACC coolant injection. Liquid level recovered in the crossover leg in the intact loop only due to coolant back-flow from the cold leg after the LPI actuation. The cold leg IBLOCA test was terminated when the primary and SG secondary-side pressures reached nearly-equilibrium condition with well-cooled core after the LPI actuation, similar to the hot leg IBLOCA test.

Fig. 13 LSTF and RELAP5 results for primary and secondary pressures in cold leg IBLOCA case

Fig. 14 LSTF and RELAP5 results for break flow rate in cold leg IBLOCA case

Fig. 15 LSTF and RELAP5 results for broken hot leg liquid level in cold leg IBLOCA case

Fig. 16 LSTF and RELAP5 results for broken cold leg liquid level in cold leg IBLOCA case

Fig. 17 LSTF and RELAP5 results for flow rates of intact ECCS in cold leg IBLOCA case

Fig. 18 LSTF and RELAP5 results for intact crossover leg collapsed liquid level in cold leg IBLOCA case

Fig. 19 LSTF and RELAP5 results for collapsed liquid levels in inlet plenum and U-tube of intact SG in cold leg IBLOCA case

Fig. 20 LSTF and RELAP5 results for upper plenum collapsed liquid level in cold leg IBLOCA case

Fig. 21 LSTF and RELAP5 results for core collapsed liquid level in cold leg IBLOCA case

Fig. 22 LSTF and RELAP5 results for cladding surface temperature in cold leg IBLOCA case

Comparison of Analysis Results

The RELAP5 code roughly predicted the overall trend of the LSTF experiment as has been compared in Figs. 13 through 22. Significant drop in the cold leg liquid level started slightly later than in the experiment due to a bit later LSC because of overprediction of the crossover leg liquid level. The primary pressure was overpredicted probably due to smaller steam discharge through the break because of overprediction of the cold leg liquid level, though break flow rate and HPI flow rate were predicted reasonably well. Core uncover following significant drop in the core liquid level before LSC started slightly later than in the experiment. Liquid accumulation in the SG U-tube upflow-side and SG inlet plenum before LSC was well calculated, though with tendencies that the U-tube upflow-side was slightly overpredicted and the inlet plenum was slightly underpredicted.

Discussion

Similar to the hot leg IBLOCA analysis case, influences of the CCFL at core exit and reduction of inter-phase drag to 1/10 in the core onto the collapsed liquid levels in the upper plenum and core and cladding surface temperature were investigated as shown in Figs. 20 through 22. Definition of 'base case' is the case with application of the Wallis CCFL correlation to the core exit, SG U-tube and SG inlet plenum and reduction of inter-phase drag to 1/10 in the core. The constants of m and C in the CCFL correlation are then 1 and 0.75, respectively. No core power control depending on the maximum cladding surface temperature was made in the analysis.

In the case with no CCFL at the core exit different from 'base case', the cladding surface temperature at Position 6, where the PCT appeared, started to increase at 85 s which was much later than in the experiment. The PCT was 826 K due to overprediction of the core liquid level, while the upper plenum became empty of liquid. In the case with original inter-phase drag in the core different from 'base case', the cladding surface temperature at Position 5 (=core center) began to increase at 45 s but the whole core was once quenched due to temporary recovery of the core liquid level and thus temporal overprediction of the core liquid level. The cladding surface temperature started to increase again after 90 s due to the core liquid level decrease. The PCT was almost the same as that in the case with no CCFL at the core exit. When the CCFL at the core exit and reduction of inter-phase drag to 1/10 in the core were considered as 'base case', the PCT was 963 K as the cladding surface temperature at Position 5 started to increase at 45 s. The core liquid level, however, was underpredicted after around 50 s due to overprediction of the upper plenum liquid level probably because of coolant remain in the crossover leg upflow-side under influences of the CCFL at the core exit.

Anyhow, it was confirmed that both the CCFL at the core exit and reduction of inter-phase drag in the core greatly affect the core liquid level behavior and thus the cladding surface temperature. Coefficients of the CCFL correlation at the core exit and inter-phase drag model in the core should then be improved for better prediction.

Analysis results by such best-estimate computer codes as RELAP5 code include uncertainties in physical models especially concerning two-phase flow phenomena. IBLOCA analyses of LSTF and PWR using best estimate plus uncertainty (BEPU) methods may be the next step. Both constants of the Wallis CCFL correlation at the core exit and gas-liquid inter-phase drag in the core would then be important uncertainty parameters which may affect the PCT significantly.

5. Conclusions

Two LSTF experiments were conducted in OECD/NEA ROSA-2 Project, simulating PWR IBLOCAs with 17% break at hot leg or cold leg. The results of the LSTF IBLOCA experiments were compared with the post-test analysis results by RELAP5/MOD3.2.1.2

code to clarify the remaining subjects. Obtained results are summarized as follows;

(1) Hot leg IBLOCA

- (a) Core uncover started simultaneously with liquid level drop in crossover leg downflow-side before loop seal clearing (LSC) induced by steam condensation on accumulator (ACC) coolant injected into the cold leg. Water remained on the upper core plate in the upper plenum due to counter-current flow limiting (CCFL) because of significant upward steam flow from the core. The peak cladding surface temperature was 607 K, which was lower than the initial temperature.
- (b) Overall trend of the LSTF experiment was well calculated by the RELAP5 code due to good prediction of the break flow. The cladding surface temperature, however, was underpredicted due to overprediction of core liquid level after the core uncover.

(2) Cold leg IBLOCA

- (a) Core dryout took place due to rapid liquid level drop in the core before LSC. Liquid was accumulated in upper plenum, steam generator (SG) U-tube upflow-side and SG inlet plenum before the LSC due to CCFL by high velocity vapor flow, causing enhanced decrease in the core liquid level. A temperature excursion continued in the core even after the initiation of reflooding due to ACC coolant injection, which induced actuation of automatic core power decrease to protect the LSTF core. The peak cladding surface temperature was 978 K.
- (b) The RELAP5 code roughly predicted overall trend of the LSTF experiment. Some discrepancies from the measured data, however, appeared in liquid levels in cold leg and crossover leg and primary pressure. The cladding surface temperature was underpredicted due to later core uncover than in the experiment.

(3) Common findings

Prediction of primary coolant distribution is important for the realistic evaluation of safety margin in the IBLOCAs. The RELAP5 code, however, has remaining problems in the proper prediction of primary coolant distribution. Improvement of constants of the Wallis CCFL correlation at the core exit and gas-liquid inter-phase drag model in the core as well as further survey of other parameters, which may affect the core liquid level and thus the cladding surface temperature, may be needed for better prediction. As for the improvement, separate-effect experiments focusing on such complex phenomena as CCFL at the core exit should be necessary to perform under well-defined boundary conditions.

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