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ROSA/LSTF test and RELAP5 code analyses on PWR hot leg small-break LOCA with accident management measure based on core exit temperature and PKL counterpart test

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ABSTRACT

An experiment was performed for the OECD/NEA ROSA-2 Project using the large scale test facility (LSTF), which simulated a hot leg small-break loss-of-coolant accident with steam generator (SG) secondary-side depressurization as an accident management measure based on core exit temperature in a pressurized water reactor (PWR). This experiment was conducted under two conditions of high-pressure to meet the PWR pressure condition and of low-pressure to meet the Primärkreisläufe Versuchsanlage (PKL) condition. Core uncovery took place by core boil-off with no reflux coolant from the SGs in the LSTF test. The increase rate of the cladding surface temperatures from top to center of the core relative to the core exit temperature increased according to the linear heat rate in the LSTF test. Some discrepancies appeared between the LSTF low-pressure phase and PKL test results for the core exit temperature increase due to differences in low-temperature structures around the core exit. The RELAP5/MOD3.3 code indicated a remaining problem in the prediction of the core exit temperature due to pseudo coolant mixing. Results of uncertainty analysis for the LSTF low-pressure phase test clarified influences of the combination of the multiple uncertain parameters on peak cladding temperature within the defined uncertain ranges.

1. Introduction

Core exit thermocouples are utilized worldwide as an important indicator to start an accident management (AM) operator action by detecting core temperature excursion during accidents in light water reactors. An experiment denoted as SB-PV-09 was carried out for the OECD/NEA ROSA Project with the rig of safety assessment/large scale test facility (ROSA/LSTF) (The ROSA-V group, 2003), simulating a 1.9% vessel upper head small-break loss-of-coolant accident (SBLOCA) in a pressurized water reactor (PWR) in 2005 (Nakamura et al., 2009). The SB-PV-09 test conditions are described below. Steam generator (SG) secondary-side depressurization was initiated by fully opening the relief valves in both SGs as an AM measure when the maximum core exit temperature reached 623 K: a criterion for Japanese PWR. The break size corresponds to the size of the ejection of one whole penetration nozzle for control rod drive mechanism. In such SBLOCA test with the LSTF, the break size is defined on the basis of the volumetric-scaled cross-sectional area of the reference PWR cold leg (Kukita et al., 1990). High-pressure injection (HPI) system of emergency core cooling system (ECCS) was totally failed adverse to the core cooling. The SB-PV-09 test result raised a safety concern on the reliability of the core exit thermocouples to detect core uncovery and to start an effective AM action.

Scaling problems remain to extrapolate phenomena observed in the scaled-down facilities to the reactor accident conditions (Mascari et al., 2015). Counterpart testing is thus considered preferable for investigating thermal-hydraulic phenomena using integral test facilities that are designed similarly but with different size and pressure, e.g. ROSA/LSTF and Primärkreisläufe Versuchsanlage (PKL) in Germany (Umminger et al., 2012). Table 1 compares the major features of the LSTF and the PKL. Volumetric scaling is 1/48 in the LSTF, whereas it is 1/145 in the PKL. The LSTF runs at full pressure, whereas the PKL pressure is limited to 5 MPa. The number of loops is two in the LSTF, and four in the PKL. The axial core power profile is a 9-step chopped cosine in the LSTF, whereas it is flat in the PKL. The vessel downcomer is cylindrical in the LSTF, but is an annulus in the upper part and a double-pipe in the lower part in the PKL.

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Table 1
Major features of LSTF and PKL.

Item	LSTF	PKL
Height	Full	
Volumetric scaling	1/48	1/145
No. of loops	2	4
Vessel downcomer	Cylindrical	Annulus (upper part); double-pipe (lower part)
U-tubes / SG	141	30
Pressure	Full	Up to 5 MPa
Core power	14% (10 MW)	10% (2.5 MW)
Axial profile	Chopped cosine	Flat
Radial profile	3-region	
ECCS	Full	

An LSTF experiment denoted as SB-HL-18 was performed for the OECD/NEA ROSA-2 Project (Nakamura et al., 2013), which simulated a PWR 1.5% hot leg SBLOCA with an AM measure based on core exit temperature under an assumption of totally-failed HPI system in 2011, as shown in Fig. 1. This experiment was conducted under two conditions of high-pressure to meet the PWR pressure condition and of low-pressure to meet the PKL condition as a counterpart against a subsequent PKL experiment denoted as G7.1 under the collaboration of the OECD/NEA PKL-2 Project. The break size was defined to simulate core uncovery with no reflux coolant from SGs in the LSTF test. As a common AM measure of the LSTF low-pressure phase and PKL tests, SG secondary-side depressurization was initiated by fully opening the secondary-side valves in all the SGs when the maximum core exit temperature reached 623 K. Boundary conditions of the LSTF test were defined based on the PKL test conditions, and volumetric scaling ratio of LSTF to PKL. Meanwhile, some researchers (Freixa et al., 2015; Carlos et al., 2016) have analyzed for the SB-HL-18 test by using TRACE and RELAP5 codes but with no uncertainty evaluation. It is thus necessary to make clear how uncertain parameters affect peak cladding temperature (PCT) through uncertainty analysis for the LSTF test.

In this study, the author carried out mutual comparison of the LSTF low-pressure phase and PKL tests to investigate influences of system scaling on major thermal-hydraulic responses. The author performed post-test analysis for the LSTF test by using RELAP5/MOD3.3 code (USNRC Nuclear Safety Analysis Division, 2001) to clarify the remaining subjects. In order to define uncertain parameters, the author made an attempt to set up phenomena identification and ranking table (PIRT) (Wilson and Boyack, 1998) for the low-pressure phase of the hot leg SBLOCA on the basis of the data analysis and the post-test analysis with the RELAP5 code for the LSTF low-pressure phase test from the viewpoint of the importance of phenomena in determining the PCT. The author conducted uncertainty analysis for the LSTF low-pressure phase test with the RELAP5 code using the PIRT to investigate influences of the multiple uncertain parameters on the PCT. With regard to the uncertainty analysis conditions, some of the parameters and ranges, the number of the computer code calculations and the sampling method were different from those employed in the author's previous work on the LSTF test concerning the cold leg intermediate-break LOCA with single-failure ECCS (Takeda and Ohtsu, 2017). This paper describes major results from the LSTF test and the RELAP5 code analyses, and the PKL counterpart test.

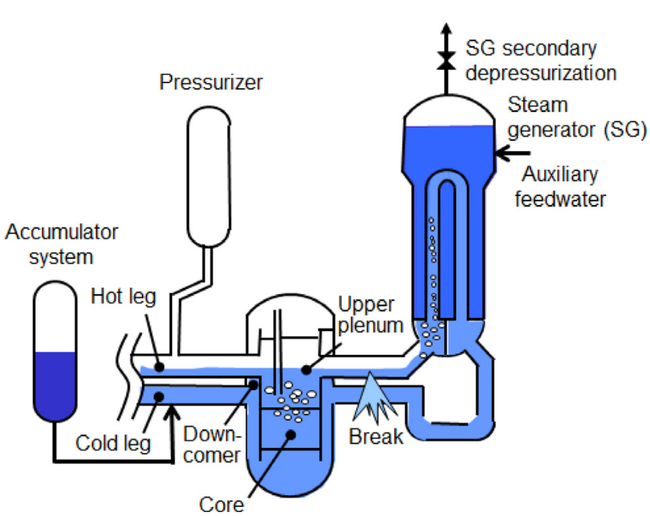

Fig. 1. Coolant behavior during PWR hot leg SBLOCA with AM measure.

2. LSTF and PKL facilities

The LSTF simulates a Westinghouse-type four-loop 3423 MW (thermal) PWR by a two-loop system model with full-height and 1/48 of volume. The reference PWR is Tsuruga Unit-2 of Japan Atomic Power Company. Fig. 2 shows the schematic view of the LSTF that is composed of a pressure vessel, pressurizer (PZR), and primary loops. Each loop includes an active SG with 141 full-size U-tubes (inner-diameter of 19.6 mm each), primary coolant pump, and hot and cold legs. The hot and cold legs, 207 mm in inner-diameter, are sized to conserve the volumetric scale (2/48) and the ratio of the length to the square root of pipe diameter to better simulate the flow regime transitions in the primary loops (Zuber, 1980).

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. The axial core power profile is a 9-step chopped cosine with a peaking factor of 1.495. Fig. 3 shows the horizontal cross-section of the LSTF core. The rod bundles denoted as B13–B20, B21–B24, and B01–B12 are used for high-, mean-, and low-power rod bundles, respectively. The LSTF initial core power of 10 MW corresponds to 14% of the volumetrically-scaled (1/48) PWR nominal core power because of a limitation in the capacity of power supply. The core power after the test initiation is then

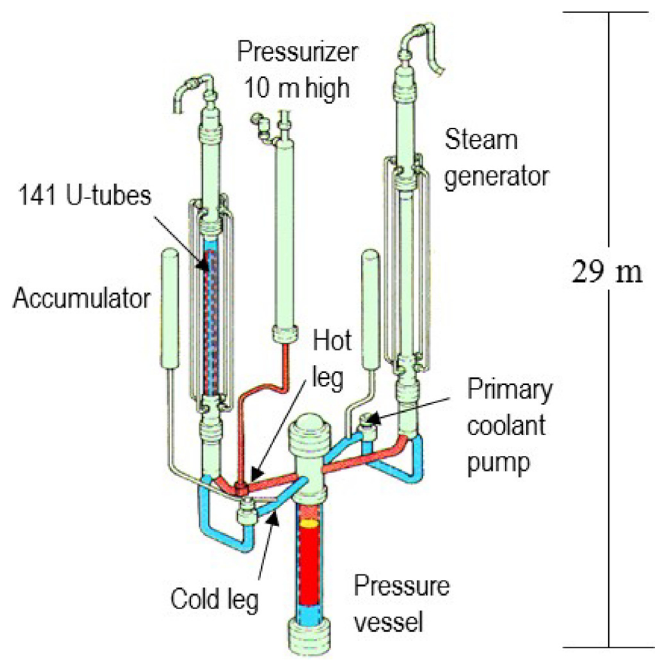

Fig. 2. Schematic view of ROSA/LSTF.