Nuclear Engineering and Technology xxx (2018) 1–8

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Technical note

ROSA/LSTF test and RELAP5 code analyses on PWR steam generator tube rupture accident with recovery actions

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ARTICLE INFO

Article history:
Received 6 March 2018
Received in revised form
30 April 2018
Accepted 11 May 2018
Available online xxx

Keywords:
Pressurized water reactor
Large-scale test facility
Steam generator tube rupture
Recovery action
High-pressure injection system
RELAP5 code

ABSTRACT

An experiment was performed for the OECD/NEA ROSA-2 Project with the large-scale test facility (LSTF), which simulated a steam generator tube rupture (SGTR) accident due to a double-ended guillotine break of one of steam generator (SG) U-tubes with operator recovery actions in a pressurized water reactor. The relief valve of broken SG opened three times after the start of intact SG secondary-side depressurization as the recovery action. Multi-dimensional phenomena specific to the SGTR accident appeared such as significant thermal stratification in a cold leg in broken loop especially during the operation of high-pressure injection (HPI) system. The RELAP5/MOD3.3 code overpredicted the broken SG secondary-side pressure after the start of the intact SG secondary-side depressurization, and failed to calculate the cold leg fluid temperature in broken loop. The combination of the number of the ruptured SG tubes and the HPI system operation difference was found to significantly affect the primary and SG secondary-side pressures through sensitivity analyses with the RELAP5 code.
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1. Introduction

When a steam generator tube rupture (SGTR) accident happens in a pressurized water reactor (PWR), operators need to take proper actions to terminate primary coolant discharge to steam generator (SG) secondary-side and to suppress the amount of radionuclide release to environment as low as possible. The radionuclide release to atmosphere during SGTR accident takes place primarily through cycle opening of relief valve of broken SG. Typical operator recovery actions include intact SG secondary-side depressurization to assure the heat sink for the primary system through natural circulation, and auxiliary spray in a pressurizer (PZR) to equalize primary and broken SG secondary-side pressures and to recover the PZR liquid level. Meanwhile, the primary pressure is kept higher than the SG secondary-side pressure when high-pressure injection (HPI) system of emergency core cooling system is under operation. Several analytical studies have been done for recovery actions from SGTR accidents of PWRs to mitigate their consequences by using best-estimate computer codes [1–4].

A SGTR accident occurred at the Mihama Unit-2 of the Kansai Electric Power Co., Ltd. in 1991, as one of worldwide SGTR incidents [5]. An experiment denoted as SB-SG-06 was conducted on the Mihama Unit-2 SGTR incident [6] with the large-scale test facility (LSTF) [7] under full-pressure conditions in the rig of safety assessment (ROSA) program at Japan Atomic Energy Agency in 1991. The LSTF simulates a Westinghouse-type four-loop 3,423 MW (thermal) PWR by a full-height and 1/48 volumetrically-scaled two-loop system, and is 1/21 scale as compared to the two-loop Mihama Unit-2. The SGTR size was equivalent to a double-ended guillotine break of one of the SG U-tubes in the two-loop Mihama Unit-2. The SB-SG-06 test results have reproduced the event transition and the pressure behavior in the Mihama Unit-2 SGTR accident well. Although the frequency of rupture of several SG U-tubes is quite low, some researchers [8,9] have evaluated the effectiveness of recovery actions from multiple SGTR events through the LSTF simulation tests and the RELAP5 code analyses. Many of databases relevant to PWR SGTR accidents with recovery actions have been obtained by using such integral test facilities as Semiscale [10], LOFT [11], LOBI [12], BETHSY [13], IIST [14], and ATLAS [15]. The experimental data, however, would be inadequate to clarify specific thermal-hydraulic phenomena and the recovery actions effectiveness due to such atypical features as small volume and low height.

An LSTF experiment denoted as SB-SG-15 was carried out for the OECD/NEA ROSA-2 Project [16], simulating a PWR SGTR accident with recovery actions in 2010. A long nozzle used to simulate the

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https://doi.org/10.1016/j.net.2018.05.002
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Please cite this article in press as: T. Takeda, ROSA/LSTF test and RELAP5 code analyses on PWR steam generator tube rupture accident with recovery actions, Nuclear Engineering and Technology (2018), https://doi.org/10.1016/j.net.2018.05.002

SGTR was the same as that in the SB-SG-06 test mentioned earlier. The recovery actions were defined, referring to the SB-SG-06 test conditions. As for the main test difference the HPI coolant was injected into the cold legs in both loops for the SB-SG-15 test, whereas it was injected into not only both the cold legs but the vessel upper plenum for the SB-SG-06 test. In this study, the author performed post-test analysis for the SB-SG-15 test by using RELAP5/MOD3.3 code [17] to assess the code predictive capability. Moreover, the author conducted sensitivity analyses based on the post-test analysis with the RELAP5 code to investigate the influences of the number of the ruptured SG tubes, the HPI system operation difference, and the onset timing of the intact SG secondary-side depressurization on major thermal-hydraulic responses. This paper is concerned with major results from the LSTF test and the RELAP5 code analyses.

2. ROSA/LSTF

The LSTF simulates a Westinghouse-type four-loop 3,423 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. Fig. 1 shows the schematic view of the LSTF that is composed of a pressure vessel, PZR, and primary loops. Each loop includes an active SG, primary coolant pump, and hot and cold legs. Loops with and without PZR are designated as intact loop and broken loop, respectively. Each SG is furnished with 141 full-size U-tubes, inlet and outlet plena, boiler section, steam separator, steam dome, steam dryer, main steam line, four downcomer pipes, and other internals. The tube inner-diameter of 19.6 mm is the same as that in the reference PWR. To better simulate the flow regime transitions in the primary loops, the hot and cold legs (inner-diameter of 207 mm each) are sized to conserve the volumetric scale (2/48) and the ratio of the length to the square root of pipe diameter (Froude number basis) [18]. The time scale of simulated thermal-hydraulic phenomena is one to one to that in the reference PWR. To simulate the fuel rod assembly in the reference PWR, the LSTF core (active height of 3.66 m) consists of 1,008 electrically heated rods in 24 rod bundles. Axial core power profile is a nine-step chopped cosine with a peaking factor of 1.495. The LSTF maximum core power of 10 MW corresponds to 14% of the volumetrically scaled PWR nominal core power.

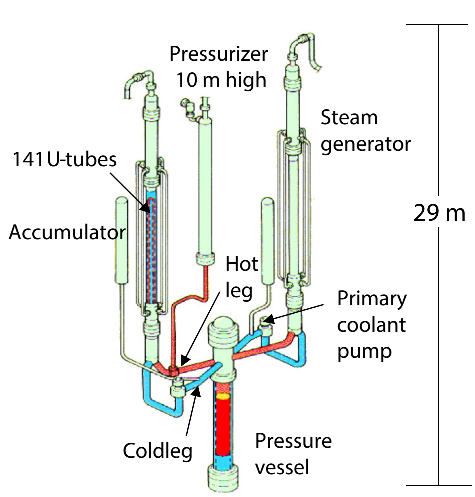

Fig. 1. Schematic view of ROSA/LSTF.

3. LSTF test and RELAP5 code analysis conditions

3.1. LSTF test conditions

The SGTR was simulated by using a 1.8 m-long nozzle with inner-diameter of 6.2 mm in the break unit in a piping connected between nozzles at inlet plenum and at secondary boiler section bottom of SG in broken loop without PZR, as shown in Fig. 2. The nozzle size corresponds to a double-ended guillotine break of the 1/21 volumetrically-scaled cross-sectional area of one of SG U-tubes in the Mihama Unit-2. The nozzle length simulates pressure drop of fluid in the broken SG U-tube of the Mihama Unit-2.

Table 1 shows the major test conditions. The experiment was initiated by opening a break valve installed in the break unit (Fig. 2) at time zero. Initial steady-state conditions such as PZR pressure and fluid temperatures in the hot and cold legs were 15.5 MPa, 598 K, and 562 K, respectively, according to the reference PWR conditions. A scram signal was generated when the PZR pressure decreased to 12.97 MPa. Loss of off-site power was assumed to occur concurrently with the scram signal, causing the closure of a SG main steam stop valve and the coastdown of primary coolant pumps. Main feedwater was terminated in both SGs 31 s after the scram signal. The LSTF core power decay curve after the scram signal was pre-determined on the basis of some calculations with the RELAP5 code considering delayed neutron fission power and stored heat in PWR fuel rod [19]. Initial SG secondary-side 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. Initial SG secondary-side collapsed liquid level was set to 10.3 m which corresponds to the SG medium tube height. Set point pressures for opening and closure of SG relief valves are 8.03 and 7.82 MPa, respectively, referring to the corresponding values in the reference PWR.

As a recovery action, the intact SG secondary-side depressurization was initiated by fully opening the relief valve 720 s after the scram signal. Auxiliary feedwater was injected into the secondary-side of both SGs 70 s after a safety injection signal when the PZR pressure decreased to 12.27 MPa. The auxiliary feedwater was terminated in broken loop when the broken SG secondary-side collapsed liquid level reached 12.85 m, while it continued till the test end in intact loop. The HPI system was initiated coolant

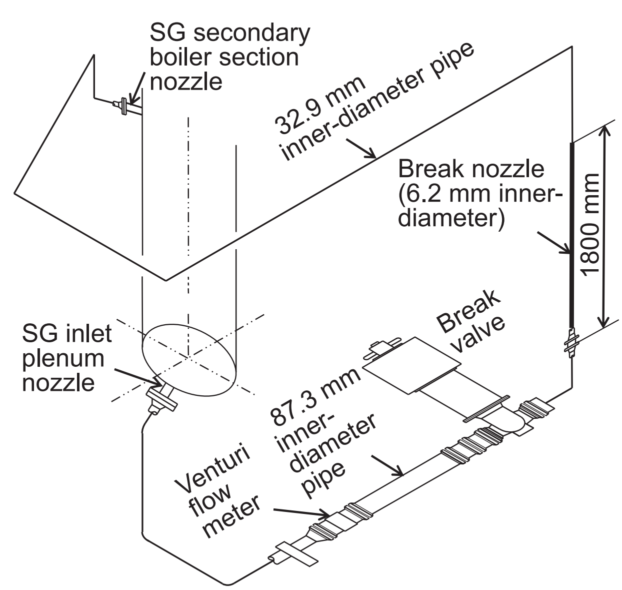

Fig. 2. Schematic view of LSTF break unit. SG, steam generator.

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