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Thermal-Hydraulic Behavior of LOCA during Shutdown Operation: C4.2 Test Findings from OECD/NEA ATLAS Phase 3 Project

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1. Introduction

In the early stages of nuclear power development, the shutdown state of a nuclear power plant (NPP), including hot shutdown, was widely perceived as safe due to lower residual heat and extended time for operator mitigation. However, the 1987 Diablo Canyon Unit 2 incident [1] revealed that a partially filled reactor coolant system (RCS) could lead to rapid temperature and pressure escalation, exposing critical vulnerabilities even under shutdown conditions. Over the 30-year period from 1972 to 2002, a total of 625 events occurred during low-power and shutdown operations, with 28 involving a loss of coolant accident (LOCA) [2]. Many of these incidents arose from operator errors or procedural oversights during maintenance, allowing coolant to transfer into the in-containment refueling water storage tank (IRWST) or containment sump. Managing such LOCAs is challenging because residual heat remains significant, automatic safety systems are often bypassed, and operator response time is reduced. Acknowledging these concerns, the OECD/NEA ATLAS Phase 3 (hereafter, OECD-ATLAS3) project selected a shutdown LOCA as the C4.2 test scenario. This test investigated natural circulation on the primary loop, passive heat removal on the secondary side, and the grace period before intervention by utilizing the ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) facility. The present experimental database contributes to validating thermal-hydraulic safety codes and refining accident management strategies for beyond-design-basis events.

2. Description of the C4.2 test

The C4.2 test addresses a LOCA scenario during shutdown cooling, approximating conditions 2.05 hours after the reactor trip of APR1400. All test procedures complied with KAERI's quality assurance program [3]. The initial phase involved heat-up, steady-state operation, and feed-and-bleed cooling until the target conditions reached: core power of 265 kW, primary pressure of 2.75 MPa, and a core exit temperature of 177 °C. Data acquisition began once low power operation was maintained, continuing for normalized time, t^*=0.43 with a uniform radial power distribution and a prescribed axial profile in the core heater rods [4]. During this low power operation, the discrepancy between core power input and primary-side heat removal was estimated to be around 1.49%, largely due to steam generator heat losses. The LOCA transient commenced by opening the OV-BS-11 valve on the DVI-3 line, followed by isolation of the main steam and feedwater lines. Two accident management (AM) actions were employed: (1) PAFS (Passive Auxiliary Feedwater System) actuation at t^* = 1.82, reflecting the 30-minute grace period in APR1400, and (2) safety injection pump (SIP) activation upon cladding temperatures exceeding normalized temperature, T^* = 0.8. The test ended at around t^* = 7.57, yielding crucial thermal-hydraulic data for refining LOCA mitigation strategies under shutdown cooling conditions. Due to the confidentiality constraints of the OECD-ATLAS3 project, all data in this manuscript have been arbitrarily normalized to preserve sensitive information.

3. Experimental results and discussion

Figure 1 provides an overview of the major events and key thermal-hydraulic parameters during the C4.2 test, which simulated a DVI-3 (the third direct vessel injection) line break LOCA under shutdown cooling conditions. Upon opening the DVI-3 line (OV-BS-11), the primary system pressure dropped sharply, then rebounded near t^*=0.57 (as depicted in Fig.1. (a)). The secondary system, isolated at the onset of LOCA, experienced a steady pressure rise due to retained steam.

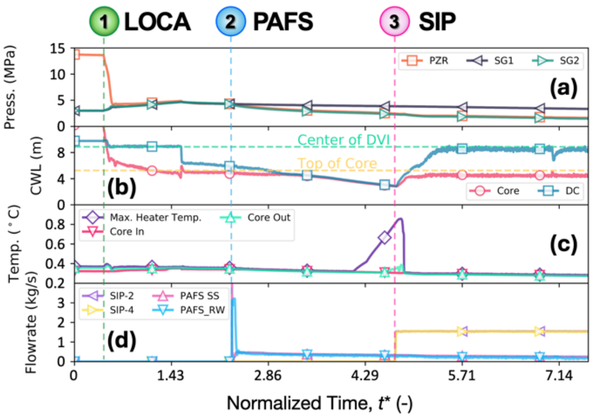

Fig. 1. Major sequence of events and RCS key variable behavior in the C4.2 test.

Simultaneously, the reactor pressure vessel (RPV) upper head emptied rapidly, and the downcomer water level fell to the DVI nozzle elevation (as shown in Fig.1. (b)), transitioning the coolant discharge from single-phase to two-phase flow and reducing the release rate.

Loop seal clearing (LSC) occurred around t^* = 1.57 when steam generated in the core broke through and displaced the coolant trapped in the intermediate legs, allowing steam to flow into the cold legs. At t^*=2.26, PAFS was actuated, generating steam condensation in the secondary side and returning subcooled water to the steam generator (as shown in Fig.1. (d)). Although this action briefly raised the water level in the intermediate leg (loop seal reformation, LSR), it did not restore core inventory. Consequently, the core heater surface temperature rose above T^* = 0.8 at t^*=4.72, prompting safety injection pump (SIP) activation through the DVI-2 and DVI-4 lines (as shown in Fig.1. (d)). The added injection water replenished the core and quenched the heater, ending the transient at approximately t^*=7.57.

Figure 2 depicts water levels dropping to the top of the horizontal pipe. Because this accident scenario operates under reduced power, pressure, and temperature compared to steady-state conditions, LSC took longer, and no heat excursion occurred within the initial grace period. The discharged steam gradually depleted the core coolant, although PAFS in loop 2 condensed steam, enabling loop seal reformation in intermediate leg (IL) 2A. Nonetheless, loop 1 remained cleared, preventing pressure buildup from the hot leg to the downcomer. These results highlight the importance of loop seal behavior in coolant inventory management and demonstrate the role of PAFS in mitigating steam release during shutdown cooling LOCA scenarios.

4. Conclusions

This paper investigated a LOCA scenario under shutdown conditions simulated in the C4.2 test, highlighting key thermal-hydraulic phenomena such as LSC and LSR.

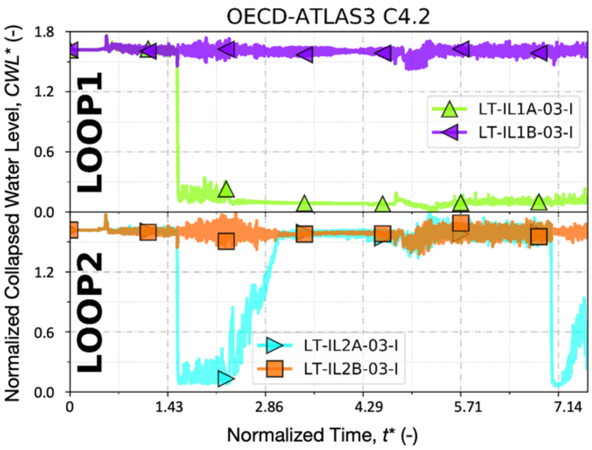

Fig. 2. Water level on intermediate legs (vertical downward pipe, steam generator side).

The test revealed that, while PAFS can reduce steam release, it alone cannot maintain adequate core cooling without safety injection. The present test data support improved accident management strategies by informing operator actions, guiding reactor design enhancements, and enabling refined thermal-hydraulic code validation. These findings underscore the importance of safety analyses for beyond-design-basis events in shutdown operations and warrant research.

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