Annals of Nuclear Energy 113 (2018) 332–343

Contents lists available at ScienceDirect

Annals of Nuclear Energy

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

Multi-scale analysis of an ATLAS-MSLB test using the coupled CUPID/MARS code

Ik Kyu Park [a,*], Seok Cho [a], Doo Hyuk Kang [b]

[a] Korea Atomic Energy Research Institute, 1045 Daedeok-daro, Yuseong-gu, Daejeon, Republic of Korea
[b] Sentech, 105 Sinildong-ro, Daedeok-gu, Daejeon, Republic of Korea

ARTICLE INFO

Article history:
Received 8 June 2017
Received in revised form 17 November 2017
Accepted 21 November 2017

Keywords:
Multi-scale
Coupled
Safety analysis
CUPID
MARS

ABSTRACT

This study describes a multi-scale nuclear safety analysis method for a main steam line break accident, which focuses on the 3-dimensional approach for the coolant temperature in the reactor pressure vessel related to the re-criticality and/or pressurizer thermal shock. The direct coupled code, CUPID/MARS, where CUPID is a 3-dimensional two-phase flow analysis code and MARS is a 1-dimensional nuclear system analysis code, was validated against a main steam line break test in the ATLAS facilities, which are a 1/2 scale integral test loop for a Korean APR1400 PWR. The calculation indicates that the suggested 4-step method, which expands the previous 2-step method of steady-state and transient calculations of a 1-dimensional system code, is valid for analyzing a direct 1-D/3-D coupled safety analysis, and the 3-dimensional temperature distribution in the downcomer agrees with the measured result.

1. Introduction

A main steam line break (MSLB) in a pressurized water reactor (PWR) may occur as a consequence of the rupture of a steam line upstream of the cross-connect. The event is characterized by significant space-time effects in the core caused by asymmetric cooling and an assumed stuck-out control rod after the reactor trip. Major concerns for the MSLB accident include return-to-power and criticality. Because of this, the MSLB scenario was based on assumptions that conservatively maximized the consequences for a return-to-power. The coupled 3-D kinetics/core thermal-hydraulic (T/H) code was used to address in the best-estimate manner because the T/H system analysis code with point kinetics did not provide a reliable solution (Todorova et al., 2003).

Return-to-power is a globally occurring phenomenon, and sufficiently accurate prediction of the core power level is possible if the transient inlet coolant conditions and the core reactivity are properly determined by the coupled code. Departure nucleate boiling (DNB), another primary concern in the main steam line break accident, is, however, a local phenomenon, and thus the accuracy of the calculated departure nucleate boiling ratio (DNBR) depends on the accuracy of the power distribution as well as the global core power level (Joo et al., 2003). In this regard, the refined core T/H nodalization feature is desirous because incorporation of the detailed thermal feedback is crucial in producing accurate power distribution.

Pressurized thermal shock (PTS) has also become a primary concern of the MSLB accident from the point of view of the ageing mechanism. The PTS issue is concerned with the possibility of failure of PWR reactor pressure vessels (RPVs) under a very specific set of conditions: the occurrence of reactor transients that subject the vessel to severe thermal shock as well as the normal pressure loading, the existence of sharp defects at the inner surface of the vessel walls, and high enough fast neutron fluence and concentrations of copper and nickel in the vessel walls to result in an extensive radiation-included reduction in the fracture toughness of the vessel material (International Atomic Energy Agency, 2010). MSLB analysis for PTS has two main objectives: to support the transient selection process and to provide fluid temperatures in the downcomer for the structural analyses of the RPV. The 1-D nuclear system code and the integral test experimental data have some limitations for the latter objective.

In the frame of the OECD PKL2 project, a test G3.1 was conducted at the PKL test facility (Umminger et al., 2009). This test investigated a fast cool down transient, which was initiated by a MSLB. The objective of the test was to create experimental data for the qualification of T/H codes against PTS and recriticality aspects. To investigate in more detail T/H behavior inside the RPV, complementary tests on the coolant mixing were conducted at the Rossendorf coolant mixing (ROCOM) test facility (Kliem et al., 2008), which was modeled on a German KONVOI-type

* Corresponding author.
E-mail address: gosu@kaeri.re.kr (I.K. Park).

https://doi.org/10.1016/j.anucene.2017.11.036
0306-4549/© 2017 Elsevier Ltd. All rights reserved.

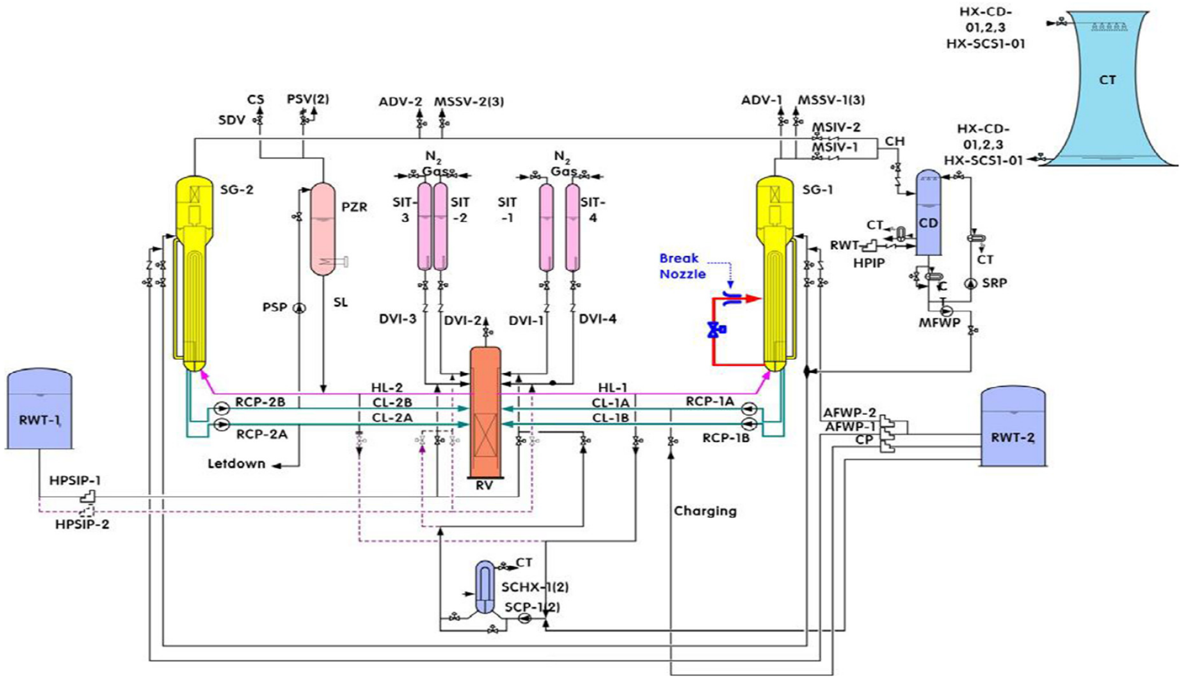

Fig. 1. Schematic diagram of ATLAS facilities.

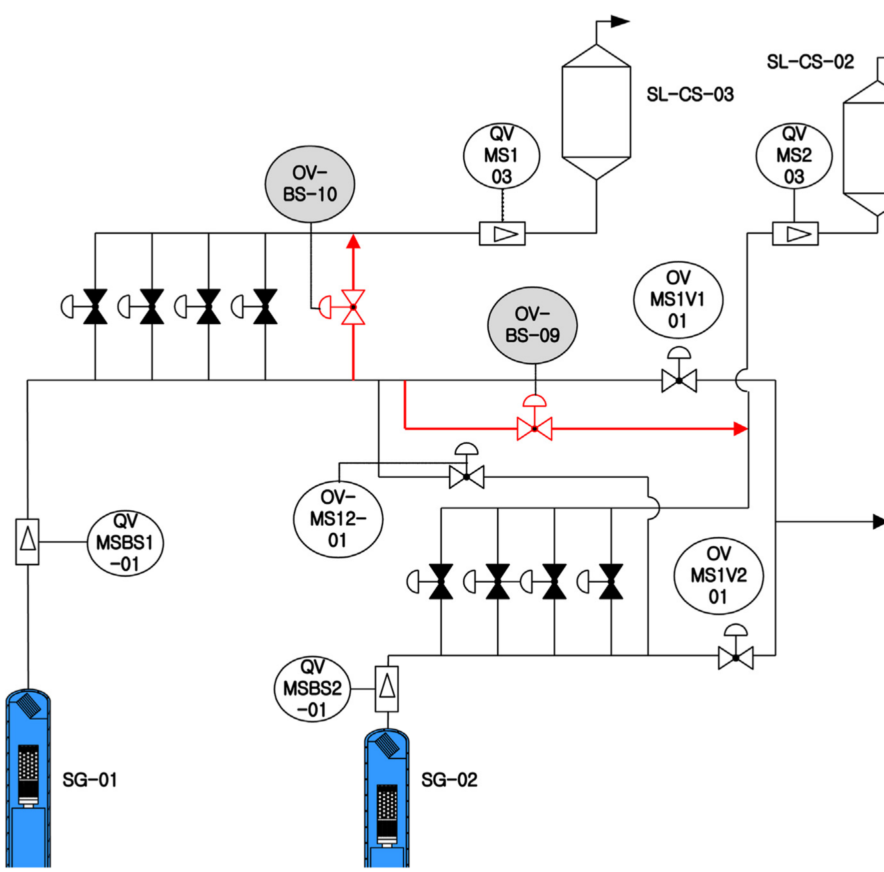

Fig. 2. Schematic diagram of ATLAS main steam line.

reactor. Experimental results at the RPV inlet from the test G3.1 were used as boundary conditions for the ROCOM tests. In the ROCOM test 1.1, the recriticality issue was investigated while test 1.2 examined the coolant mixing in relation to PTS. The PKL experimental data contributed to evaluating the nuclear system analysis code, and the ROCOM experimental data to computational fluid