Preliminary Analysis of SB-LOCA-induced Severe Accident at CANDU-6 Reactor using M-CAISER Code

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1. Introduction

CANDU (Canadian Deuterium Uranium) reactor has its unique design features (reactor vault and moderator tank) as an ultimate heat sink under severe accidents. In case of the loss of coolant accident (LOCA), the moderator cooling system (MCS) keeps cooling down for the calandria tube and can prevent of the fuel channel failure in spite of the loss of emergency core cooling system (ECCS). However, a limited core damage (LCD) may occur despite of the credit on MCS under severe accidents, where in-Fuel channel phenomena such as the melting of 37 fuel-pins and the contact of melted fuel/clad to the pressure/calandria tube [1,2] are critical issues. The present study is focused on an early-phase severe accident (fuel channel failure) at CANDU type reactor under small break LOCA (SB-LOCA), depending on the credit for THE MCS.

2. Methods and Results

2.1 Node system of M-CAISER

Recently, KAERI has been developing the M-CAISER code, which is the coupled version between MARS and CAISER codes [3-6]. MARS roles to simulate the thermal-hydraulics in the primary circuit, while CAISER roles to simulate the core degradation phenomena in a fuel channel and in a calandria tank. CAISER code uses the generalized *Cartesian coordinate system* for 380 fuel channels in a calandria tank and for 37 fuel fins in a fuel channel. In this simulation, a calandria tank is nodalized by (4 x 4) nodes and the fuel channel has the nodes of (2 x 2). The calandria tank and fuel channel has 12 nodes in the flow direction, which reflecting the 12 bundles in a channel. Figure 1 shows the MARS node system for the primary cooling system.

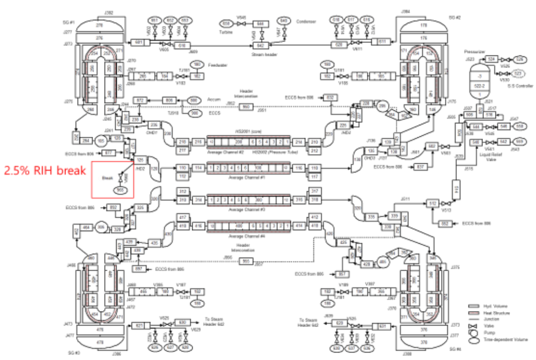


Fig. 1. Node system of general CANDU-6 for M-CAISER

2.2 Sequence of event for SB-LOCA

The present study is focused to analyze the effect of the availability of the MCS based on the scenario of SB-LOCA by simulating two cases: case#1 (available of MCS) and case#2 (loss of MCS) [7]. The MCS affects to the moderator level of the calandria tank. CAISER considers the rapid expulsion of the moderator due to the sudden pressure difference in the fuel-channel, which results in an instantaneous decrease of moderator level. In the scenario of SBLOCA, the 2.5% RIH break is applied according to the reference [8,9]. Emergency core cooling system (ECCS) and crash cool-down for steam generator are unavailable and loop isolation and Reactor coolant pump (RCP) trip are available. Details of SOE are summarized in Table I.

Table I: Sequence of event for SB-LOCA

No.	Event	Criteria
0	SB LOCA start	RIH 2.5% break
1	Reactor trip Main feed water trip Turbine governor valve trip	P_{RH}, P_{ROH} < 8.7 MPa
2	RCP trip (Break loop)	RCP inlet void fraction < 0.5
3	LOCA signal	P_{RH}, P_{ROH} < 5.52 MPa
4	Loop isolation	LOCA signal + 20 s
5	RCP trip (Intact loop)	RCP inlet void fraction < 0.5
6	Liquid relief valve (LRV) open (Intact)	P_{ROH} > 10.34 MPa
7	Degassing condenser relief valve (DCRV) open	P_{DCT} > 10.16 MPa
8	Calandria tank rupture disk open	P_{MCT} > 137.8 kPa

2.3 Results

SBLOCA-induced severe accident was evaluated with two stages: (i) steady state (0-1,000 s) and (ii) transient (after 1,000 s) (Fig. 2). After reactor trip (1,000 s), an rapid depressurization of the RIH break results in the LOCA signal (1,206 s) with low pressure alarm and each loop is isolated by the isolation valve below the pressurizer (1,226 s). Compared to the break loop, the intact loop maintains the forced convection until the RCP trip condition (5,632 s). After the RCP trip, the thermal-hydraulic behavior of the intact loop seems to be similar to that of the station black out accident: high pressure accident accompanying the dryout of the steam

generator at the intact loop (8,528 s), opening of the LRV (8,101 s) and DCRV (8,182 s) [10,11].
As opposed to the fuel and cladding temperature, the calandria tube temperature is governed by the moderator level in the calandria tank. In Fig. 3, the loss of the MCS (case#2) makes the calandria tube temperature to increase about 15,000 sec when the moderator level decreases below the [I][J] = [2][3] node, which means the uncover of the fuel channel. However, the calandria tube temperature (T_ct) for Case#1 keeps the constant value as the saturation temperature under the pressure for calanrdia tank rupture disk open (137.8 kPa). And, it roles to prevent the fuel channel failure in a intact loop even though the inner pressure for calandria tube of intact loop is high, which is same to the LRV open set value (10.34 MPa).

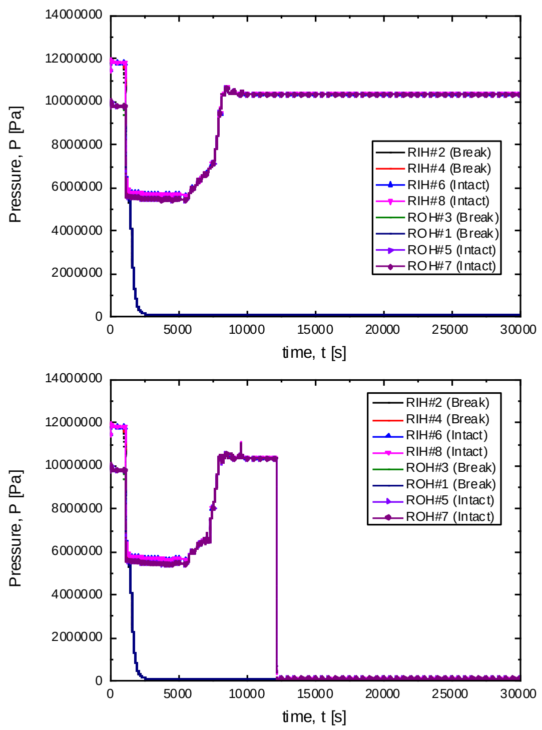
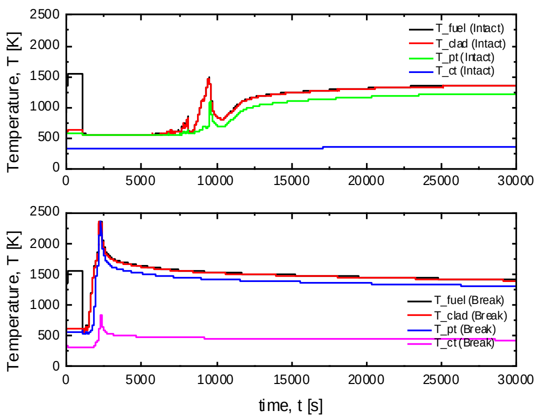


Fig. 2. Pressure of primary heat transport system (PHTS): case #1 (top)_ and case#2 (bottom).



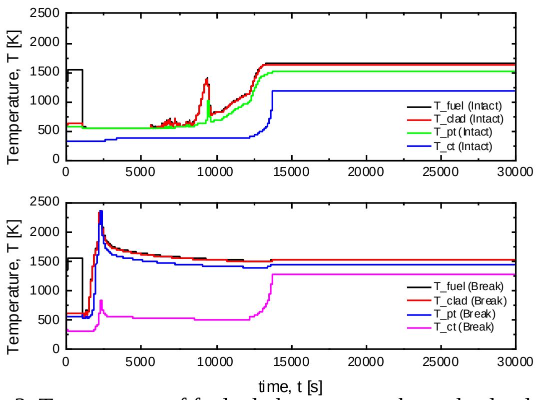


Fig.3. Temperature of fuel, clad, pressure tube and calandria tube ([I][J][K] = [2][3][5]): case #1 (top) and case#2 (bottom).

CAISER have four failure modes for the pressure and calandria tube, respectively (average temperature, local temperature, ultimate strength criteria and creep failure criteria). Calandria tube of the break loop at the case#1 is failed by the temperature-based criteria of which indicates the rapid heat-up by the dryout of the coolant into the fuel channel. On the other hands, calanrdia tube of the break loop for the case#1 was not failed because of its low temperature. Creep failure model in the CAISER is applied by the Larson-miller parameter (LMP) approximation for the pressure (Zry-2) [12] and calandria tube (Zry-2.5% Nb) [13], respectively. According to the LMP, high pressure nearly identical to the normal operation (~ 10MPa) with the low temperature of calandria tube guarantees the long time for the creep rupture failure of the fuel channel.

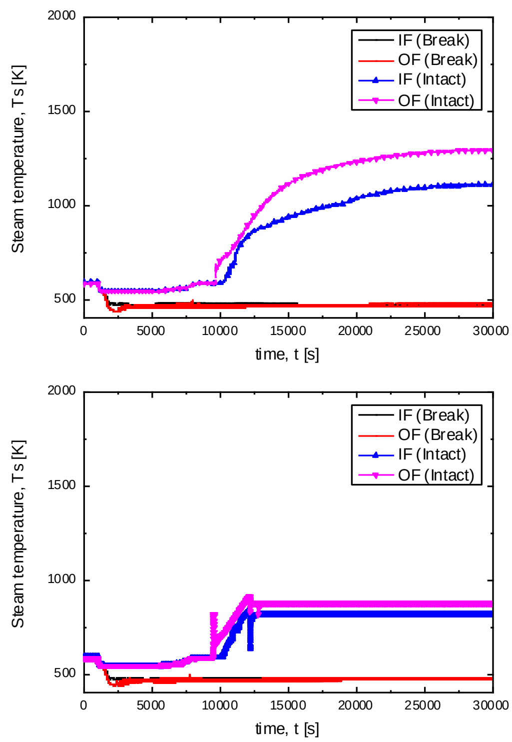


Fig. 4. Steam temperature of inlet feeder (IF) and outlet feeder (OF): case#1 and case#2

Another interesting point of the case#1 is that it has higher steam temperature (nearly 1300 K) at the feeder pipes in an intact loop compared with that of the case#2 (below 1000 K) (Fig. 4). It can be explained by the natural convection of the superheated steam [14]. Compared to the case#2, the fuel channel (or, calandria tube) in an intact loop is not failed in the case#1, which makes the strong natural convection in a intact loop, while the natural convection in a intact loop for the case#2 is negligible because of the low pressure in a loop. As a result, the feeder pipe for the case #1 is heated up by the strong natural convection in an intact loop. This leads to the feeder pipe of carbon steel to have a high temperature in which the steel oxidation can be generated and makes the unexpected hydrogen generation under steam environment [15].

3. Conclusions

Early phase of the SB-LOCA-induced severe accident was evaluated by the M-CAISER code depending on the loss of the MCS. Compared to the case of a no credit for the MCS (case#2), the case of a credit for the MCS (case#1) shows LCD at the broken loop without failure of the fuel channel. Superheated steam of the PHTS can lead to temperature increase in the feeder pipe at the case#1, which then leads to the additional the generation of hydrogen gas from steel oxidation.

REFERENCES

[1] C. Geradi, J. Buongiorno, Investigation of pressure-tube and calandria-tube deformation following a single channel blockage event in ACR-700, MIT, 2005

[2] B.R.Sehgal, Nuclear safety in light water reactors: severe accident phenomenology, Academic press, 2011

[3] J.H. Bae et al., Theory manual of CANDU Advanced Integrated Severe accident code (KAERI/TR-7734/2019), 2019

[4] J. Kang et al., User manual of CANDU Advanced Integrated Severe accident code (KAERI/TR-7733/2019), 2019

[5] J.H. Bae et al., Core degradation modeling of CANDU severe accident code, CAISER, Transactions of the Korean Nuclear Society, Spring Meeting, Jeju, Korea, May 16-18, 2018

[6] J.H. Bae et al., Simulation of CS28-1 experiment by using CANDU severe accident code, CAISER, Transaction of the Korean Nuclear Society, Autumn Meeting, Yeosu, Korea, Oct 25-26, 2018

[7] Y. M. Song et al., Preliminary evaluation of small LOCA under severe accident conditions using ISAAC and M-CAISER in wolsong plants, Transaction of the Korean nuclear society, spring meeting, Online, 2020

[8] Final Safety Analysis Report, Wolsong Unit II (Chp. 15), 2006

[9] T. Kim, J. Park, A containment analysis for SBLOCA in the refurbished wolsong-1 nuclear power plant, Nuclear engineering and design, 2011

[10] J. Kang et al., Preliminary analysis about SBO-induced severe accident at CANDU-6 using M-CAISER code, Transaction of the Korean Nuclear Society, spring meeting, Online, 2020

[11] Y.M. Song et al., Detailed Core Response Evaluation of SBO Induced Severe Core Damage Using ISAAC for M-CAISER Comparison Basis in Wolsong Plants, Transaction of the Korean Nuclear Society, Autumn meeting, 2020.

[12] S. Arai and H. Murabayashi, Failure correlation for zircaloy-2 fuel cladding under high temperature transient conditions, Journal of nuclear science and technology, 1987.

[13] K. Guguloth et al., Mechanism of creep deformation with evolution of microstructure and texture of Zr-2.5Nb alloy, Material science and engineering A (2018)

[14] Development of analysis methodology for beyond design basis accident based on RD-310 for CANDU, 1301007-0113-SB110, NESS (2014)

[15] Y.M. Song et al., Reconsideration of hydrogen source term in CANDUs to include oxidation steel, Transaction of the Korean Nuclear Society, spring meeting, 2018

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