Transactions of the Korean Nuclear Society Virtual Spring Meeting
July 9-10, 2020

Preliminary Evaluation of Small LOCA under Severe Accident Conditions Using ISAAC and M-CAISER in Wolsong Plants

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1. Introduction

To benchmark the severe accident analysis codes for pressurized heavy water reactors (PHWR or CANDU), IAEA organized a coordinated research project (CRP)[1], titled "Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications." Seven institutes joined the CRP using their own codes, among which dedicated codes for CANDU severe accident analysis were only two, MAAP-CANDU and MAAP(1)-ISAAC (Integrated Severe Accident Analysis code for CANDU plants). The CRP results show pretty large difference in accident progression and behavior but validation work was identified to be hard mainly due to lack of PHWR experiment data. Another difficulty for this was lack of a detailed and mechanistic code, such as MELCOR for LWRs, as dedicated tools for CANDU severe accident analysis. According to this demand for an accurate and detailed code in a CANDU society, a new severe accident code called CAISER (**C**ANDU **A**dvanced **I**ntegrated **SE**ve**R**e code) [2] is being developed at KAERI (Korea Atomic Energy Research Institute). At present, the code development reached to the stage of simulating the severe accident progression in a calandria vessel. As a main feature, CAISER code has recently been coupled with a detailed system themal-hydraulic code for design accident analysis resulting in concrete analysis in a primary system (PHTS).

The main purpose of this paper is to evaluate a small loss of coolant accident (SBLOCA) under severe accident conditions using MAAP-ISAAC code and to compare a themal-hydraulic part with CAISER code until fuel channel failure starts. The target plants are Wolsong (WS) NPPs which are a typical CANDU-6 type. Current study basically uses MAAP-ISAAC version 4.03 [3][4]. It is constructed in modules covering individual regions in the plant: PHTS, pressurizer, steam generators, calandria vessel, reactor vault, end-shields, degasser condenser tank, and the reactor building (RB). The code provides an efficient and integrated tool for evaluating in-plant effects of a wide range of postulated accidents, for which a wide spectrum of phenomena including steam formation, core heat-up, cladding oxidation, hydrogen evolution and vessel failure can be evaluated.

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(1) MAAP[5] is an Electric Power Institute (EPRI) software program that performs severe accident analysis for nuclear power plants including assessments of core damage and radiological transport. A valid license to MAAP4 and/or MAAP5 from EPRI is required.

2. ISAAC and M-CAISER Models

The ISAAC models a broad spectrum of physical processes in the core that might occur during accident, such as the:

- Fuel/cladding temperature excursions, degradation and interaction with moderator system
- Zirconium-steam exothermic reaction
- Thermal mechanical failures of fuel channels
- Disassembly of fuel channels
- Formation of suspended debris beds
- Motion of solid and molten debris
- Interaction of the core debris with steam

In particular, the ISAAC models the CANDU feeders, end-fittings, fuel channels and fuel. The models in the ISAAC concentrate on the behavior of these core components within the calandria vessel as the fuel channels disassemble, form suspended debris supported by intact channels, and relocate to the debris bed within the calandria vessel. Each characteristic channel represents a larger number of channels (known as associated channels) with similar powers, elevations and feeder geometries. The ISAAC thermal hydraulic (T/H) models in PHTS are simplified by using assumptions such as coarse nodalization, equilibrium within a fluid phase, a uniform loop pressure and a single global void fraction at which phase separation occurs. The ISAAC PHTS T/H results cannot be expected to be as accurate as those from more detailed PHTS models associated with a T/H code such as CATHENA. Most importantly, however, ISAAC is an integrated code that models the interactions amongst many systems that are modelled in an integrated fashion. Thus, ISAAC calculates the effects of the interplay between the RB, calandria vessel, PHTS, reactor vault, core, etc.

As aforementioned, the CAISER code simulates core degradation phenomena occurring in a calandria vessel, and it consists of two main modules: a fuel channel module and a calandria vessel module. The fuel channel module simulates the severe accident phenomena happening in a fuel channel, which includes a core uncover, fuel rods heatup, hydrogen generation due to steam-Zr oxidation, fuel rods (pins) slumping, fuel rods melting and relocation, and thermal interaction of relocated molten mass with a pressure tube or calandria tube. The calandria vessel module simulates the overall severe accident phenomena in a calandria vessel, including the sagging of a fuel channel, debris bed formation caused by a fuel channel failure, the molten pool formation, and the calandria vessel wall failure. Recently, the CAISER code has been coupled with the system T/H code to simulate CANDU power plants, which is named M-CAISER [6].

In this paper, SBLOCA is analyzed using ISAAC and M-CAISER codes to compare the T/H results and show the M-CAISER code feasibility for CANDU severe accident analysis in a CANDU6 PHTS configuration (Fig.1).

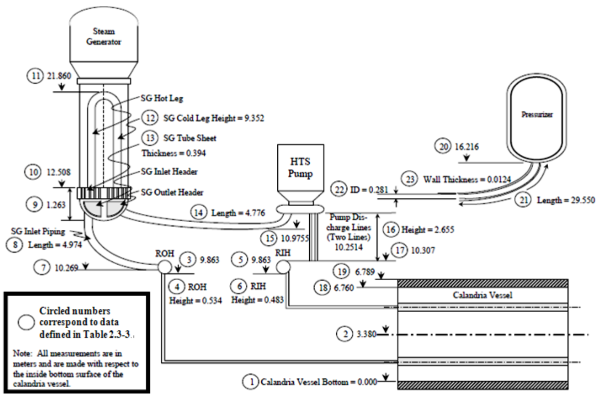

Figure 1 CANDU6 PHTS reference configuration [1]

3. SBLOCA Analysis in ISAAC and M-CAISER

The reference case (SLO-A) is a representative de-pressurization accident defined as 2.5% (0.005327 m^2) reactor inlet header (RIH) break of PHTS Loop-1 and break elevation of 10.696 m referenced to the bottom of calandria vessel with loss of ECC, no moderator cooling, no shield cooling, no steam generator crash cool-down, no steam generator feed water and no local air cooler. It is also assumed that the shutdown systems and PHTS loop isolation are available.

After SBLOCA initiates, the PHTS pressure in broken Loop-1(L-1) decreases, and coolant from the pressurizer and intact Loop-2(L-2) flows into L-1 via common piping. The PHTS and pressurizer are isolated from each other 20 s after the loop isolation signal is generated by the PHTS low pressure below 5.516 MPa(a). The automatic isolation of the two PHTS loops reduces the rate of reactor coolant loss in the event of a loss of coolant accident. While the L-1 coolant continues to vent from the break, L-2 continues to be well cooled by its steam generators, because sufficient primary coolant remains after loop isolation occurs. The L-2 cooling ceases after the steam generator secondary side water inventories boil off. If auxiliary feed water (AFW) is available, it prolongs the heat removal from L-2 fuel channels; otherwise the L-2 coolant heats up and pressurizes until liquid relief valves (LRVs) open at 10.34 MPa(a). The LRVs vent primary coolant into the degasser condenser tank. LRV discharge area from each PHTS loop is modeled with 0.00303 m^2. The availability of auxiliary feed water has a limited effect on L-1 cooling, because the L-1 primary coolant is lost via the RIH break.

Steam generators provide heat sinks for both loops, provided enough primary coolant is available to transport decay heat from the fuel to the steam generators. The steam generator main steam safety valves (MSSVs) are available and they open and close at their set point (5.24/5.11 MPa(a)) to relieve the steam generator secondary side pressure. The L-1 steam generator heat sink capability rapidly decreases as the coolant level drops. The remaining L-1 coolant is boiled off by fuel decay heat, and the steam vents out the RIH break. In ISAAC all four steam generators, MSSVs and the common steam header are modeled. The PHTS pumps are tripped for the pump protection at low void fraction in both loops. The pump trip is initiated when the void fraction in pump inlet water falls below 50%. The determination of cavitation forces the immediate pump stop without pump coastdown for two pumps in each corresponding loop.

The L-1 channels are now dry, and the fuel heats up. The channels are surrounded by cool moderator that limit the channel and fuel heat up. Fuel sheaths may be hot enough to oxidize with the L-1 steam, producing hydrogen (some vents into RB via RIH break). When AFW is not available, the L-2 coolant boils off and vents out the LRVs. The L-2 water level decreases and fuel channels begin to dry out. The L-2 pressure remains high, because the loop is still intact, so dry channels overheat and strain outwards (balloon) until at least one fails. Following the L-2 fuel channel failure, the flow from the loop heats up and pressurizes the calandria vessel. The calandria rupture disks at the top of calandria opens and discharge heavy water to the RB (SG room).

4. Results and Discussion

The accident progression timing is compared between ISAAC and M-CAISER codes in Table 1, which shows T/H progression can be calculated in a similar manner with ISAAC after reactor scram in M-CAISER.

Table. 1 Accident Progression in ISAAC & M-CAISER

Event in SBLOCA	ISAAC	CAISER
RIH Break (0.005327 m^2) in L-1 Shield cooling off ECCS (HPI/MPI/LPI) off No SG MSSV manual open (No SGCC) PHTS loop isolation (after receiving LOCA signal)	0.0	
Reactor scram SG MFWS and AFWS off, and Turbine stop valve closed	48	48
PZR empty (<2.5% nominal)	120	120
LOCA signal received	195	264
PZR L-1 & 2 isolation	215	284
Primary pump off in L-1	254	182

Core has uncovered in L-1	(801)	
Core water is empty in L-1	(2040)	
CV (Calandria Vessel) water pool begins to boil off	(5140)	
SG broken loop dryout in L-2	**5646**	**7050**
Primary pump off in L-2	6663	4600
CV overpressure rupture valve open	(7694)	
Core has uncovered in L-2	(7900)	
Pressure tube rupture (L-1)	(8666)	
Fuel bundle failure (L-1)	(8776)	
Pressure tube creep rupture (L2)	(8811)	
Fuel bundle failure (L-2)	(10214)	

As a key T/H output variable, steam generator (SG) water level (collapsed) in both loops is shown in Fig. 2 and Fig. 3, respectively for ISAAC and M-CAISER.

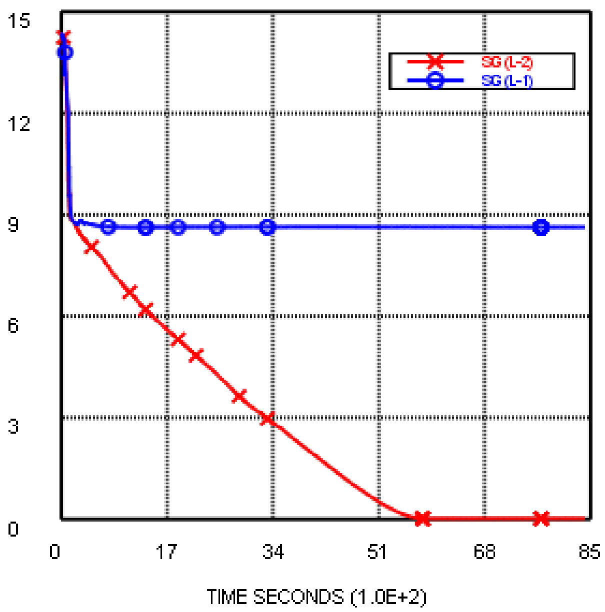

Figure 2 SG collapsed water level in ISAAC [m]

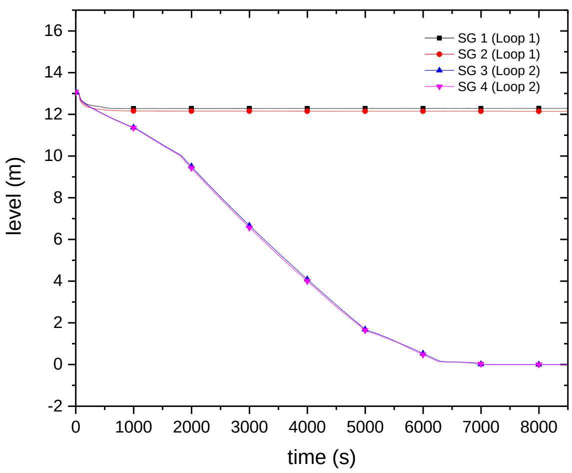

Figure 3 SG collapsed water level in M-CAISER [m]

As a key event, SG dryout time in L-2 is appeared about 30% faster in ISAAC. In order to find out cause variables, three variables (such as initial SG coolant inventory, reactor trip time and PHTS pump trip time) are analyzed through a sensitivity study. Fig. 4 shows the results which are summarized in Table 2. According to this, SG dryout time is shortened if the inventory decreases or the trip time increases (as more primary heat is transferred to SG secondary side in case of delay in reactor trip or PHTS pump trip)

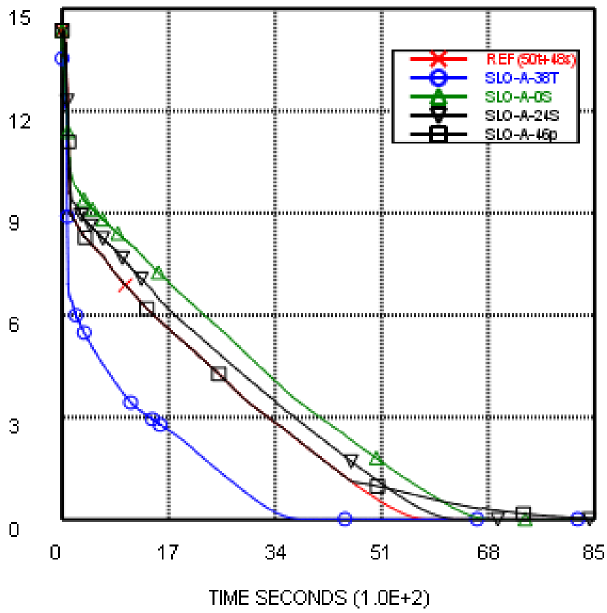

Figure 4 Sensitivity study for SG(L-2) dryout time in ISAAC

Table. 2 ISAAC sensitivity summary for SG(L-2) dryout time

Case	SG water inventory [ton]	SG dryout time [s]
SLO-A-38t	37.7	3647
	Reactor trip time [s]	
SLO-A-0S	0.1	6656
SLO-A-24S	24	6120
	PHTS pump trip time [s]	
SLO-A-46p	4600	8280

The main results of this study are as follows:

- M-CAISER code has been tested for a capability to simulate the T/H conditions under severe accident conditions by comparing with ISAAC results. The comparison shows T/H progression is calculated in a similar manner in two codes with some difference, for example, about 30% difference in SG dryout time.
- ISAAC has a chance to revise the input model such as SG initial inventory and reactor trip time by comparing with more detailed and mechanistic models, which must have been impossible without M-CAISER code. It also suggest PHTS void fraction calculations resulting in pump trip needs a further review in ISAAC model because the 30% difference

in SG dryout time becomes a half when M-CAISER pump trip time is applied to the ISAAC calculation.

ACKNOWLEDGMENTS

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea Government (Ministry of Science, ICT, and Future Planning) (No. NRF-2017M2A8A4017283).

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