Transactions of the Korean Nuclear Society Virtual Autumn Meeting
December 17-18

Detailed Core Response Evaluation of SBO Induced Severe Core Damage Using ISAAC for M-CAISER Comparison Basis in Wolsong Plants

Y.M. Song*, J.Y. Kang, D.G. Son, J.Y. Jung, J.H. Bae
Korea Atomic Energy Research Institute, Accident Mitigation Research Team
989-111, Daedeok-daero, Daejeon, Korea
**Corresponding author: ymsong@kaeri.re.kr*

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. 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 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 M-CAISER (**M**ARS plus **C**ANDU **A**dvanced **I**ntegrated **SE**ve**R**e code) [2][3] is being developed at KAERI (Korea Atomic Energy Research Institute). At present, the code development has finished the first stage of simulating the severe accident progression in a core. As a main feature, CAISER code has provided with a concrete core degradation modeling with a detailed nodalization scheme, which makes it possible to simulate channel failure process more accurately and realistically under severe accident conditions.

The main purpose of this paper is to evaluate a station blackout (SBO) accident resulting in hypothetical severe core damage using MAAP-ISAAC and to analyze a core response for providing later M-CAISER comparison basis. The target plants are Wolsong (WS) NPPs which are a typical CANDU-6 type. Current study basically uses MAAP-ISAAC version 4.03 [4][5]. It is constructed in modules covering individual regions in the plant: primary heat transport system (PHTS), pressurizer (PZR), degasser condenser tank (DCT), steam generator (SG), calandria vessel (CV), and the reactor building. The code provides an 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.

----

(1) MAAP[6] 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 Models and Nodalization

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 CV as the fuel channels disassemble, form suspended debris supported by intact channels, and relocate to the debris bed within the CV. 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 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 core, PHTS, CV, reactor vault, reactor building etc. for CANDU severe accident analysis.

In this paper, SBO induced severe core damage is analyzed using ISAAC in a core nodalization scheme (Fig.1) of Wolsong unit 2/3/4 plants which have a typical CANDU6 PHTS configuration.

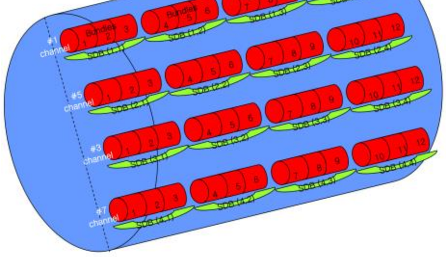

Figure 1 CANDU6 core nodalization (4x4) scheme in ISAAC

3. SBO Analysis in ISAAC

The reference case (SBO-A) is a representative pressurization (= high pressure) accident defined as a transient initiated by a loss of off-site AC (Class IV) power, with the subsequent loss of all on-site standby and emergency electric power supplies (see Table 1). If the high/medium/low-pressure emergency core cooling system (ECCS), SG main (MFW) and auxiliary feed water (AFW) systems, moderator cooling system (MCS) and end-shield cooling system (ESC) are not available, the accident sequence progresses to a severe core damage accident. The PHTS loops are not isolated from each other in this scenario, and the operator intervening actions are assumed unavailable.

Table 1 Status of Major Safety System or Function in SBO-A

Cases	Rx Trip	PHTS loop Isolation	MFW or AFW	ECCS	MCS	ESC	Comments
SBO-A	O	X	X	X	X	X	no AC power

Since the PHTS pumps are not available due to loss of power after the accident initiation, the fuel heats up and a temperature gradient develops in the coolant between the core and the SG region, which promotes natural circulation between the two regions. In SBO-A scenario, the pressure behavior in PHTS loops and in the SGs show similar behavior, respectively in both loops and 4 SGs (Fig.2). As the decay heat is transferred enough to the SG secondary side, the PHTS pressure initially decreases and is maintained as long as the SG secondary side has sufficient heat sink capacity to absorb the decay heat. Meanwhile, the SG secondary pressure stays at the main steam safety valve set point as the valves open and close.

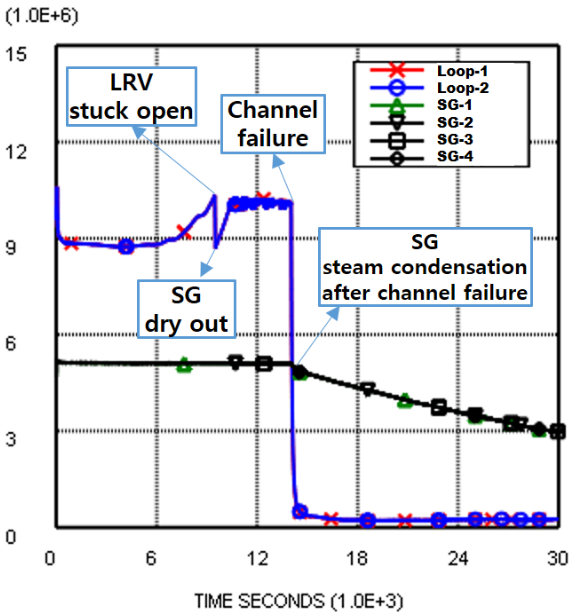

Figure 2 Pressure Behavior in the PHTS and SGs

As the water level in the SGs decreases from boil-off proceeds, the water in the SGs is depleted at about 10,000 s, after which the pressure in the PHTS starts to increase until it reaches the PHTS liquid relief valve (LRV) set point, 10.16 MPa (a), and assumed to fail open resulting in inventory discharge into DCT. The PHTS inventory is lost through the LRVs (Fig.3) and the fuel bundles begin to uncover within the channels (Fig.4).

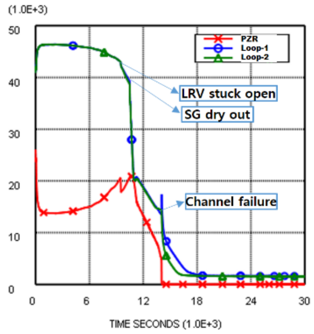

Figure 3 PHTS Water Inventory

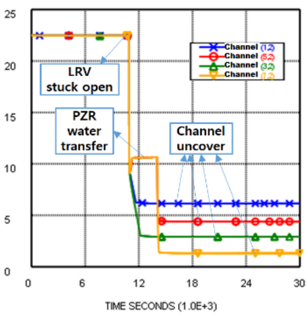

Figure 4 PHTS Water Level

The PHTS inventory is gradually lost through the LRVs resulting in fuel channel dryout. In parallel, the moderator heats up and the water level in the CV decreases gradually since moderator cooling is not assumed available. With the loss of moderator as a heat sink, a lead channel with the highest decay power and the smallest inventory in each loop, which is situated at a high (not always the highest) elevation in the CV, reaches high pressures and temperatures such that the lead channel will not be able to sustain the pressures. As a result, the lead channel, (5,1)channel (refer to Fig.1) ruptures. With the rupture of the lead channel, the PHTS pressure drops rapidly at about 14,000 s (Fig.2/3) and the loop inventory is blown down into the CV (Fig.5). With the rapid blow down of the PHTS inventory into the CV, the pressure inside the CV reaches the set point of the rupture disc and the CV rupture discs burst.

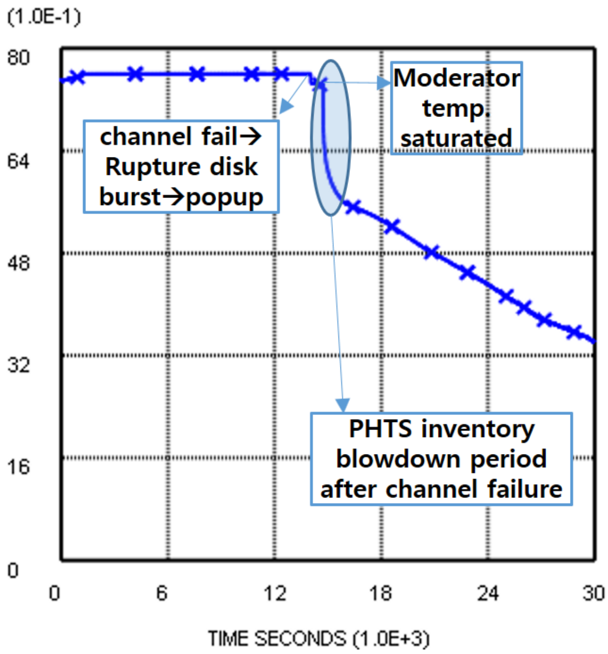

Figure 5 Moderator Water Level in CV

4. ISAAC Core Nodalization and Response Analysis

The CANDU-6 PHTS has two independent loops. Each loop, in turn, make two core passes (figure of eight loop) which is composed of 380 inter-connected U-tubes called channels. The ISAAC code allows the user to group 380 fuel channels into up to 74 core channels based on their elevations, power levels, core passes, and loops. In this simulation, 16 (4x4) core channels together with 4 individual SGs are used to represent 380 fuel channels and these core passes (4x4 with checker board pattern of blue and pink colors) in both loops are entirely represented as shown in Fig.6.

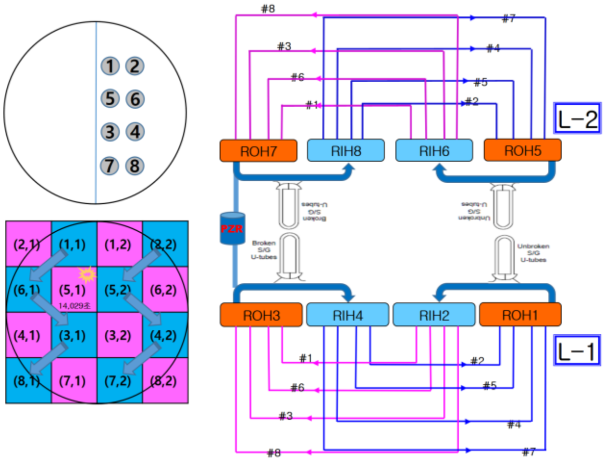

Figure 6 PHTS Core Passes (4x4) with Checker Board Pattern in ISAAC Modeling

When the PHTS circulating system (= pump) stops and phases separate with the water level falling below the reactor header, individual core channel forms a U-tube and 16 U-tubes characterize the 16 independent water pools. Therefore, ISAAC tracks the mass and energy in multiple water pools but in one gas node in the system.

Each core channel is nodalized into 12 axial horizontal nodes (= bundles) as shown in Fig.1. Each node contains four components such as fuel, cladding, pressure tube (PT) and calandria tube (CT). Heat transfer between each components with the coolant and moderator is modeled. After the fuel channel dries up, it starts to heat up. At a sufficiently high temperature, the PT and/or the CT start to sag. The code represents the impact of CT sagging by disabling heat transfer to surrounding coolant and moderator. This accelerates the temperature escalation and eventually leads to the relocation of the core node material to lower channels or the CV bottom. As the accident progresses, the CV inventory decreases and finally dries out.

The temperature trend of four core components in the 9th bundle of the lead channel(5,1) which fails first are shown in Fig.7. Until channel failure occurs, the fuel/cladding/PT temperatures are the same but higher than the CT temperature which is the moderator temperature. After channel failure, PT follows the CT temperature and all four components finally reach the melting temperature together after the channel uncovery in CV.

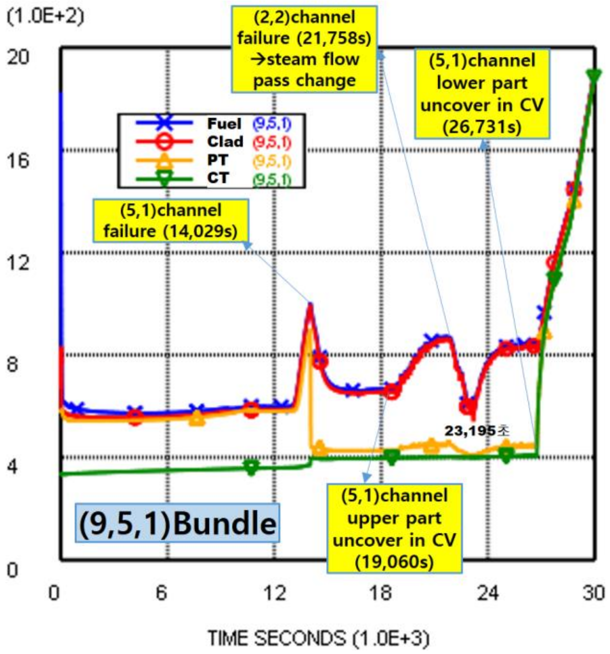

Figure 7 Lead Channel (= bundle(9,5,1)) Temp. Behavior

5. Summary

Detailed core response evaluation of SBO induced severe core damage using ISAAC is made for WS 2/3/4 which are typical CANDU-6 types. The main objective is to evaluate a SBO accident resulting in hypothetical severe core damage using MAAP-ISAAC, especially to evaluate a core response for providing detailed M-CAISER comparison basis. The following table shows the comparison result for main event timing of the concrete analysis with M-CAISER code [7] using the same core passes (4x4).

Event Timing [sec]	MAPP-ISAAC	M-CAISER
LRV open	9473	9258
SG dry out	10041	9188
Channel uncover	12751	11483
PZR empty	14034	16720
PT failure	14029	13362
Fuel Relocate in CV	22649	17795
CV dry out	40054	25687

The main results of this study are as follows:

- The SBO accident progress is simulated using fast-running ISAAC having features like non-mechanical models and empirical correlations and the response of core with checker board pattern is analyzed in detail.
- According to the draft comparison of accident progression timing with M-CAISER, two codes show similar trend until PT failure but shows a difference after PT failure which requires additional analysis for corium behavior in CV afterwards.

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).

REFERENCES

[1] IAEA TECDOC 1727 - Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications, November (2013).
[2] Jun Ho Bae, Dong Gun Son, Ki Hyun Kim, Jun Young Kang, Yong Mann Song, Jong Yeob Jeong, Sang Ho Kim, Bo Wook Rhee, Modelling and Simulation of CANDU Severe Accident Analysis Code, CAISER, To Be Published (2020).
[3] KAERI, CANDU severe accident LOCA scenario analysis using MARS-KS/CAISER coupled code system, KAERI/CM-2867/2020 (2020).
[4] KAERI (Korea Atomic Energy Research Institute), ISAAC Computer Code User's Manual, KAERI/TR-3645/2008 (2008).
[5] Y.M. Song et. al., Severe Accident Progression and Consequence Assessment Methodology Upgrades in ISAAC for Wolsong CANDU6, Transactions of the Korean Nuclear Society Spring Meeting, Jeju, Korea, May 7-8 (2015).
[6] Modular Accident Analysis Program (MAAP) for CANDU Reactors, ANS (1992).
[7] J.Y. Kang et. al., Preliminary Analysis about SBO-induced Severe Accident at CANDU-6 Using M-CAISER Code, Transactions of the Korean Nuclear Society spring Meeting, JeJu, Korea, July 9-10 (2020).