Analyses of Severe Accident Progression According to Break Areas in SBLOCAs of SMART

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1. Introduction

KAERI (Korea Atomic Energy Research Institute) performed severe accident sequence analyses using CINEMA (Code for INtegrated severe accident Evaluation and Management) with a helical steam generator model to develop a PAMP (Preliminary Accident Management Program) of SMART (System-Integrated Modular Advanced Reactor) 365 MW_{th} [1-4]. In this study, we analyzed severe accident progression according to break areas in SBLOCAs (Small Break Loss of Coolant Accidents) under the assumption of the failure of all safety systems and not working of the ERVC (External Reactor Vessel Cooling) at SMART. The purpose of this analysis is to compare the severe accident analysis results from a core heat-up to a vessel failure according to the break areas by CINEAM and to see the role of the ADS (Automatic Depressurization System) operation in the SBLOCA simulations.

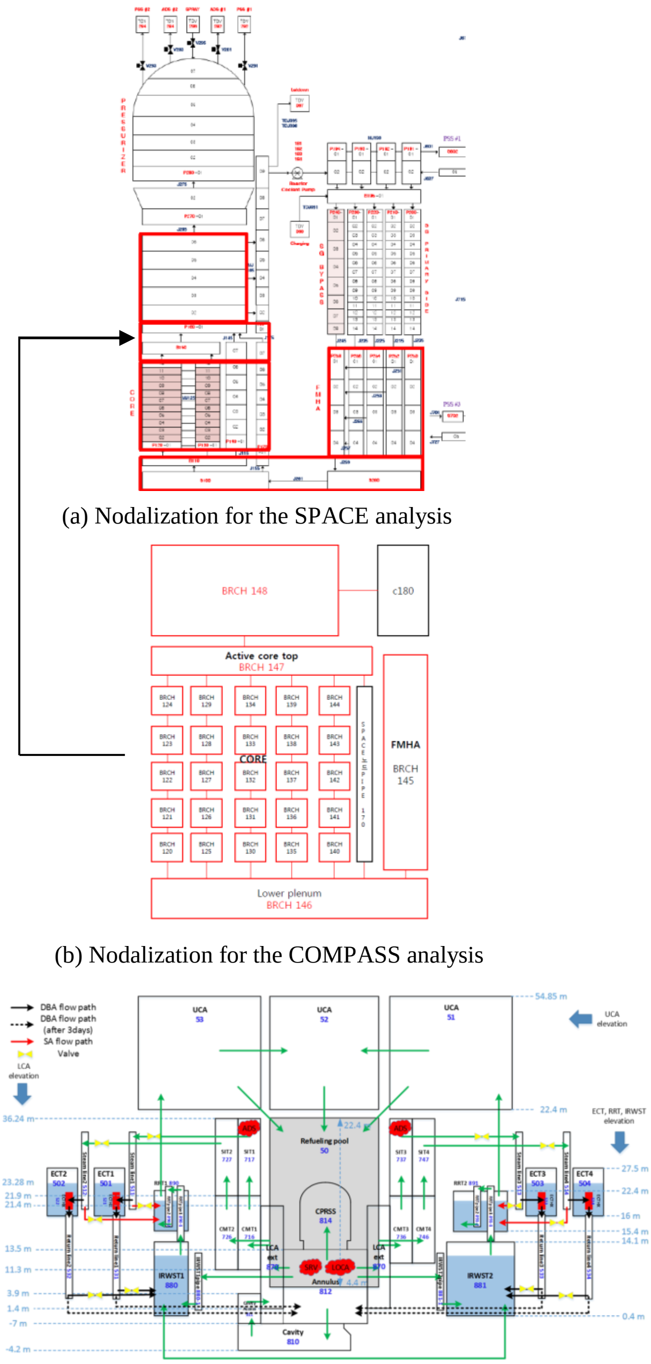
2. CINEMA-SMART Code

The CINEMA-SMART code comprises the following analysis modules: in-vessel phenomenon (CSPACE), ex-vessel phenomenon (SACAP), and fission product (SIRIUS). CSPACE (COMPASS and SPACE) to analyze a severe accident in an RCS (Reactor Coolant System) was developed through connection with COMPASS (COre Meltdown Progression Accident Simulation Software) to simulate the progression of a severe accident in a core and SPACE (Safety and Performance Analysis CodE for nuclear power plants) to simulate thermal-hydraulic behavior in primary and secondary system [3,4].

3. Severe Accident Analysis Using CINEMA-SMART

3.1 Input Model

Nodalizations for the CINEMA-SMART analysis of SMART are shown in Fig. 1. In Fig. 1(a), the regions of the upper plenum, core, lower plenum, and FMHA (Flow Mixing Head Assembly) in the nodalization for the SPACE calculation are replaced by COMPASS nodes (Fig. 1(b)). The SACAP nodes used in the calculation are shown in Fig. 1(c). A connection between the CSPACE node and the SACAP node is made at the reactor annulus node, the Safety Injection Tank (SIT) compartment nodes, and the reactor cavity node shown in Fig. 1(c) because we assume that the primary coolant through the break park and the ADS discharges to the reactor annulus and the SIT compartment #1 and #3 in the containment.



(a) Nodalization for the SPACE analysis

(b) Nodalization for the COMPASS analysis

(c) Nodalization for the SACAP analysis

Fig. 1. Nodalization for the CINEMA-SMART analysis

3.2 Steady State Analysis

A steady state calculation was performed to prepare conditions for a transient analysis to simulate a severe accident initiated by a SBLOCA. The calculated steady state results by CINEMA-SMART show a good agreement when compared to the major design parameters of SMART (Table I).

Table I: Calculation Results of Steady State [5]

Parameter	**SMART**	**CINEMA**	**Error(%)**
Core power [MW]	365.0	365.0	0.00
PZR pressure [MPa]	15.0	14.86	-0.95
PZR level [%]	70.0	69.96	-0.06
Core inlet temp. [℃]	295.5	298.38	0.97
Core outlet temp. [℃]	320.9	323.25	0.73
Total RCS flow [kg/s]	2507.0	2507.31	0.01
Steam pressure [MPa]	5.76	5.79	0.57
FW pressure [MPa]	6.66	6.68	0.25
Steam temp. [℃]	302.3	299.67	-0.87
FW temp. [℃]	230	230.04	0.02
Total FW flow [kg/s]	190.61	190.61	0.00

3.3 Transient Analysis

A transient calculation was started by breaking a two-inch pipe at the PSIS (Passive Safety Injection System) line connected to the RCP (Reactor Coolant Pump) discharge pipe after completing the steady state calculation. The transient calculation was performed repeatedly by varying the broken pipe diameter from 2 inch to 0.22 inch and 1 inch to check how the times of major events change according to the break areas. If the pipe break occurs at 2000.0 s, the RCS pressure rapidly decreases to 0.1 MPa owing to the discharge coolant to the containment through the break part in the results of the broken pipe diameter of 1 inch and 2 inch (Fig. 2).

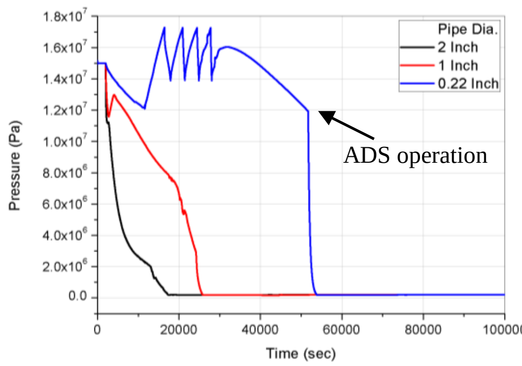
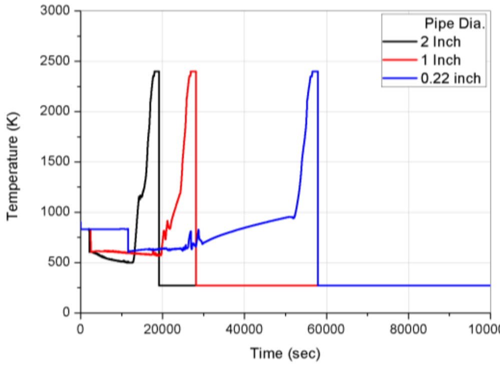
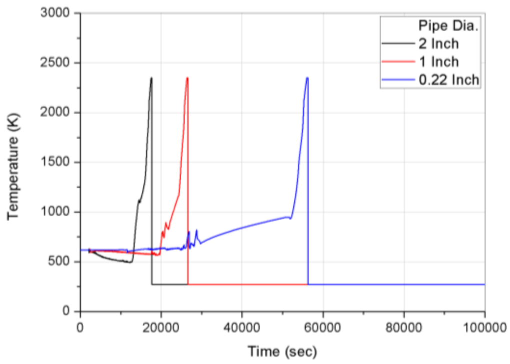


Fig. 2. RCS pressure behaviors according to the break areas



(a) Fuel temperature



(b) Cladding temperature

Fig. 3. Fuel and cladding temperatures at the middle height in the center ring (Ring 1) according to the break areas

Table II: Times of major events according to the break areas

Major Events	**Time (sec)**		
	2 inch	**1 inch**	**0.22 inch**
Occurrence of Initial Event SBLOCA	2,000.0	2,000.0	2,000.0
LPP Signal Actuation	2,096.2	2,436.1	11,563.7
Reactor Trip	2,097.3	2,437.2	11,564.8
RCP Trip	2,097.3	2,437.2	11,564.8
Entry of Severe Accident*	14,233.0	22,378.5	49,916.6
Start of H_2 Generation	14,870.0	23,560.0	53,140.0
Start of Fuel Relocation	15,350.0	24,090.0	53,450.0
ADS Opening by Operator	16,033.0	24,178.5	51,716.7
Complete of Core Uncover**	16,671.0	22,151.0	28,681.0
RV Rupture	41,570.0	44,616.6	73,451.1

*Core outlet temperature reaches 923.15K
**Water empty at the core bottom

A reactor trip signal is generated by an LPP (Low Pressurizer Pressure) signal if the pressurizer pressure decreases to 12.13 MPa. And then, the reactor is shut down as the control rods are inserted into the reactor core by the reactor trip signal after a time delay of 1.1 s from the actuation of the LPP signal. It is assumed that the turbine trip and loss of off-site power occur at the same time as the reactor trip signal is generated. In addition, we assumed the failure of all safety systems during this accident progression and no operation of the ERVC. Therefore, the fuel and cladding temperatures are greatly increased and arrives the melting temperature as the core uncover phenomenon completes (Fig. 3 Table II). Table II shows the times of major events according to the break areas. The comparison of the major event sequences between the break of the pipe diameter 2 inch and 1 inch shows similar severe accident sequences each other except a time delay of approximately 340 s – 3500 s for all major events in the result of 1 inch pipe break.

The calculation result of the break of the pipe diameter 0.22 inch shows the different RCS pressure behavior when compared to other results (Fig. 2). The RCS pressure increases to approximately 17 MPa from 12.13 MPa after the actuation of the reactor trip signal. This may be explained by the fact that the loss of secondary heat removal, the RCP trip, and the decay heat in the core increases the coolant temperature in the RCS. In addition, the energy removed by the discharged coolant through the 0.22 inch pipe is a small amount compared the coolant energy stored in the RCS. However, the ADS operation by the operator can decrease the RCS pressure through the discharging the coolant in the RCS into the SIT compartments.

The predicted amounts of hydrogen generation in the core during the severe accident according to the break areas are shown in Fig. 4. Hydrogen is generated during the zirconium oxidation in the fuel cladding by a chemical reaction of zirconium with steam as the cladding temperature increases after the core uncover phenomenon begins. Thus, the amount of hydrogen generation is usually dependent on the cladding temperature and the steam flow rate thorough the core [3]. The calculation result shows that the accumulated hydrogen mass is decreased as the break area reduces (Figs 4 and 5).

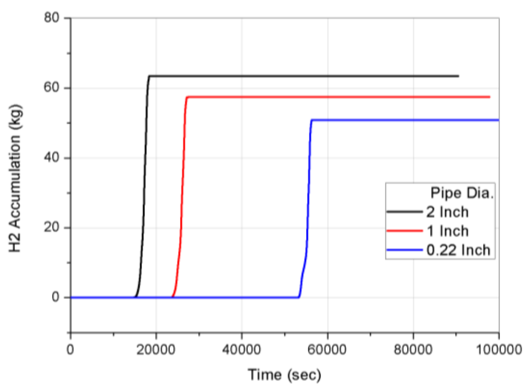


Fig. 4. Amount of hydrogen generation in the core according to the break areas

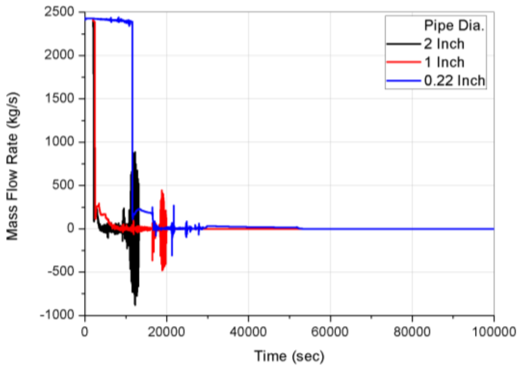


Fig. 5. Steam flow rate at the core inlet according to the break areas

4. Conclusions

We analyzed the severe accident progression initiated by SBLOCAs according to the break of the 0.22 inch pipe, 1 inch pipe, and 2 inch pipe under the assumption of the failure of all safety systems and not working of the ERVC at SMART using the CINEMA-SMART code. We found that the CINEMA-SMART results are reasonable through the comparison of the major event sequences between three cases. In addition, we can know that the ADS operation can be effectively used to reduce the RCS pressure in the SBLOCA with the small break area such as 0.22 inch pipe. It is also believed from the CINEMA-SMART results that the reactor vessel may not ruptured if the ERVC is operated at SMART because the ERVC can prevent the temperature increase of the reactor vessel.

ACKNOWLEDGMENTS

This work was supported by the Korea Atomic Energy Research Institute (KAERI) grant funded by the Korea government.

REFERENCES

[1] Han Ok Kang, SMART Pre-Project Engineering between KAERI and K.A.CARE, Proceedings of KNS Autumn Meeting, October 29-30, 2015, Jeju, Republic of Korea.
[2] IAEA, Implementation of Accident Management Program in Nuclear Power Plants, Safety Report, No. 32, Vienna, Austria, 2004.
[3] KHNP, KAERI, FNC, and KEPCO E&C, Theory Manual of the CINEMA Code, Technical Report, S11NJ16-2-E-TR-7.4, Rev.0, Daejeon, Republic of Korea, 2017.
[4] J. H. Bae, et al., Development Document for CINEMA-SMART, Technical Report, S-916-NB437-001, Rev.1, KAERI, 2018.
[5] H. S. Kang, D. G. Son, R.-J. Park, S. H. Park, and C. H. Park, An Evaluation of Severe Accident Sequence Initiated by a SBLOCA Using the CINEMA Code, Proceedings of NTHAS-11, November 18-21, 2018, Busan, Republic of Korea.