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Preliminary Validation Analysis of Severe Accident Progression on LOCA without Safety Injection of Real Power Plant for CINEMA Computer Code

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1. Introduction

As an integrated severe accident computer code development in Korea, CINEMA (Code for INtegrated severe accidEnt Management Analysis) has been developed for a stand-alone severe accident analysis [1]. The basic goal of this code development is to design a severe accident analysis code package by exploiting the existing domestic DBA (Design Basis Analysis) code system for the severe accident analysis. The CINEMA computer code are composed of CSPACE [2], SACAP (Severe Accident Containment Analysis Package) [3], and SIRIUS (SImulation of Radioactive nuclide Interaction Under Severe accident) [4], which are capable of core melt progression with thermal hydraulic analysis of the RCS (Reactor Coolant System), severe accident analysis of the containment, and fission product analysis, respectively.

The CSPACE is the result of merging the COMPASS (COre Meltdown Progression Accident Simulation Software) [5] and SPACE (Safety and Performance Analysis CodE for nuclear power plants) models [6, 7], which is designed to calculate the severe accident situations of an overall RCS thermal-hydraulic response in SPACE modules and a core damage progression in COMPASS modules. For the purpose of CSPACE verification for a real power plant, LOCA (Loss Of Coolant Accident) without SI (Safety Injection) of severe accidents for the real power plant of the APR1400 has been analyzed in this study. This validation analysis has been performed to estimate the efficiency of the CINEMA computer code and the predictive qualities of its models from an initiating event to a containment performance. Best estimate calculations from the initiating events of the SBLOCA (Small Break LOCA) of 2-inch equivalent diameter and the LBLOCA (Large Break LOCA) of 9.6-inch equivalent diameter without SI (Safety Injection) have been performed by using the CINEMA computer code. This paper is focus on the severe accident progression after reference [8].

2. CINEMA Input Model

The input model for the CINEMA calculation of the APR1400 [9] was a combination of the SPACE and COMPASS input models. Heat structures for the fuel rods and the lower part of the reactor vessel in the SPACE input model were replaced by the COMPASS input models. In the SPACE model, the reactor core was simulated as three channels to evaluate the thermal-hydraulic behavior in detail, and each channel was composed of five axial volumes, as shown in Fig. 1. A surge line and a pressurizer were attached to one of the hot legs in the primary coolant loop. In the COMPASS input model of this analysis, the component numbers for the fuel and control rods were three and five. A steady state calculation was performed in order to verify the input nodalization of CSPACE for the APR1400. The steady state results for a selected set of parameters were in very good agreement with the operating conditions of the APR1400. The maximum error of the steady state results using the CSPACE computer code with operating values of the APR1400 is approximately 5% in the coolant mass flow rate at the core inlet. The steady state conditions obtained from the simulation were used as initial conditions for the transient calculation.

Fig. 2 shows the CINEMA input model for APR1400 containment. The reactor cavity (1), reactor cavity access area and corium chamber room (2), reactor vessel annulus (3), refueling pool (4), broken steam generator compartment A (5), unbroken steam generator compartment B (6), pressurizer compartment (7), containment upper compartment (8), containment dome (9), containment annular compartment (10), holdup volume tank (11), IRWST (In-Containment Refueling Water Storage Tank) (12),and environment (13) are simulated for the containment.

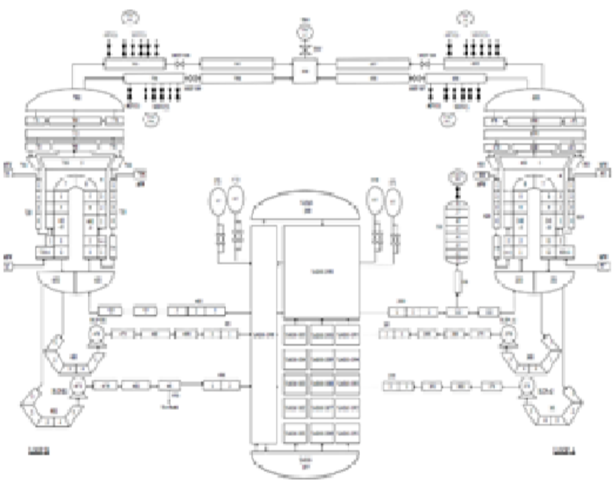


Fig. 1. CINEMA input model for NSSS (Nuclear Steam Supply System).

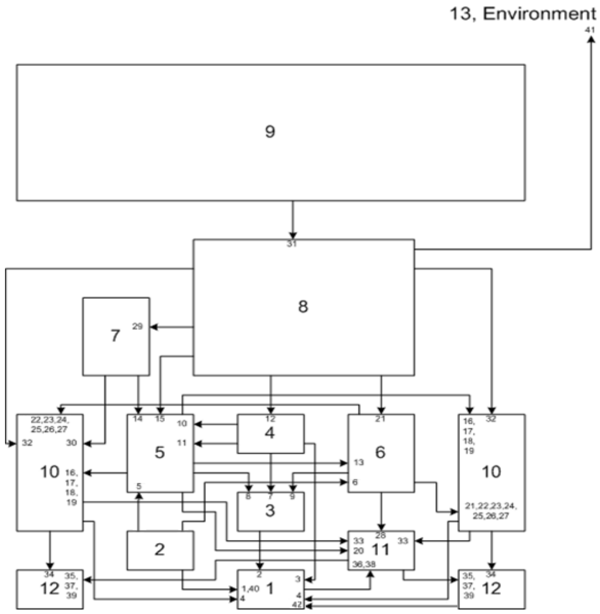


Fig. 2. CINEMA input model for containment.

3. Results and Discussion

Table I shows CINEMA results on the major events. The accident was initiated by producing 2-inch of the SBLOCA and 9.6-inch of the LBLOCA equivalent diameter breaks in the cold leg. The reactor and the RCP (Reactor Coolant Pump) s were assumed to be tripped at an accident initiation time. The RCS water inventory rapidly decreased and a boiling started in the core because the safety injection pumps were not actuated. The fuel began to heat up when the core was uncovered. Oxidation of the fuel cladding began when the cladding surface temperature reached 1,000K and produced an oxidation heat. The fuel cladding was failed by a sausage-type ballooning. When the cladding surface temperature reached 1,700K, oxidation of the zircaloy was accelerated as the steam was supplied from the bottom of the reactor vessel.

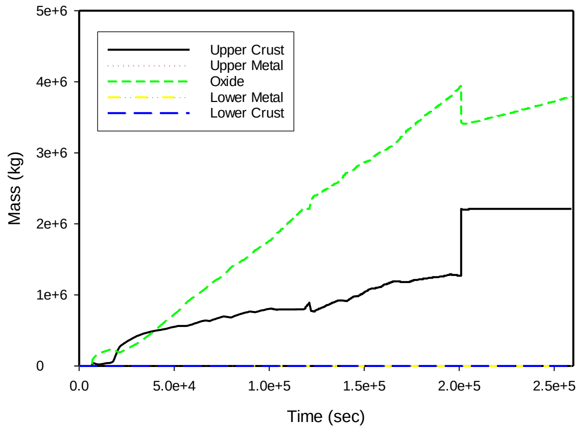
At about 2,129K of the cladding surface temperature, the zircaloy inside the oxide shell began to liquefy and the outer portion of the fuel pellets was dissolved. The relatively thin ZrO_2 shell ruptured at about 2,390K because the shell strength decreased with the temperature increase. The bottom of the core dried out because a hot mixture of liquefied fuel and cladding had relocated downward. The debris formed at the bottom of the fuel rods, where the liquefied mixture had resolidified. The melting temperature of the zirconium dioxide is 2,390K, and that of the uranium dioxide is 2,400K in these calculations. The flow blockage in the lower part of the core region occurred because of a fuel melting and a cohesive debris formation. The melted core material had relocated to the lower plenum of the reactor vessel. Finally, the reactor vessel was failed by a creep through a melt thermal attack.

After the LOCAs occur at 0sec, the pressurizer pressure rapidly decreases to the saturation pressure corresponding to the hot leg temperature at the beginning of the transient. As the coolant began to boil, the expansion of the coolant caused by a boiling was able to compensate for the break flow, and the pressure maintained a saturation pressure. In the LOCAs without the SI, the volumetric flow out through the break is greater than the coolant expansion caused by a boiling, and the pressure began to decrease again. The steady decrease in the pressurizer pressure stopped after the SITs began a coolant injection to the RCS. When the injected liquid entered the core, it boiled, then raising the pressurizer pressure. When the pressure was low enough again, more coolant was injected. This cycling of the SITs actuation slowed down the depressurization. When the molten core material relocated to the lower plenum, the pressurizer pressure increased because of a coolant boiling in the lower plenum. In the SBLOCA transient, frequent cycling showed because the break size is small. The SIT actuation depends on the RCS pressure. In the SBLOCA without the SI, the reactor vessel may be failed at an early time, because much coolant was lost by the break. The total hydrogen generation masses in the SBLOCA and LBLOCA without SI are approximately 280kg and 500kg, respectively. The total hydrogen generation mass of the LBLOCA case is larger than that of the SBLOCA case, because of the core damage time and coolant inventory difference by a coolant loss inventory through the break, which are reasonable value.

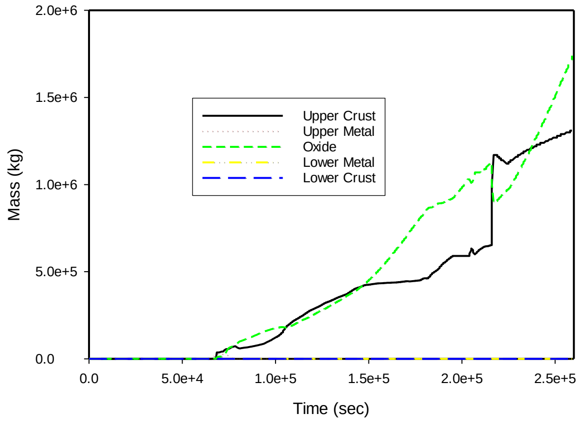
Table I: CINEMA results on the major events.

Major Events	LBLOCA (sec)	SBLOCA (sec)
Transient Initiated	0	0
SIT Injection/End of SIT Injection	191/421	4,870/33,900
Core Uncovery	1,015	2,637
Initiation of Fuel Cladding Oxidation	1,650	3,197
Start of FP Gap Release (T_{clad}=1,173 K)	1,773	3,420
Core Dryout	1,953	3,718
Corium Relocation into Lower Head	3,890	65,500
Reactor Vessel Failure	6,630	74,995
End of Calculation	259,200	259,200
Total Hydrogen Generation Mass	228.4 kg	451.2 kg

Fig. 3 shows the CINEMA results on the corium mass in the reactor cavity for the LOCA without the SI of APR1400. The corium relocated to the reactor cavity initially at a reactor vessel failure, which were 6,630 s and 74,995 s in the LBLOCA and SBLOCA, respectively. After that, corium mass in the reactor cavity increased.



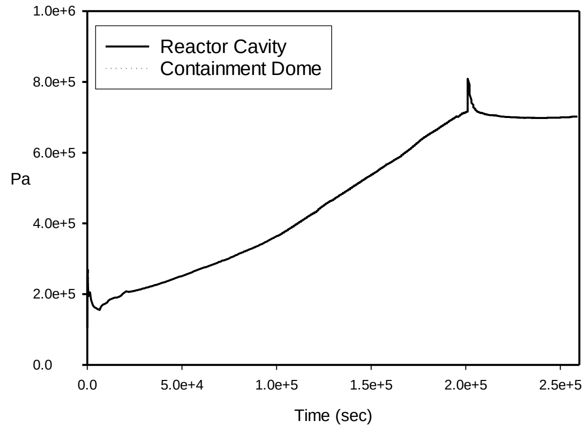
(LBLOCA)



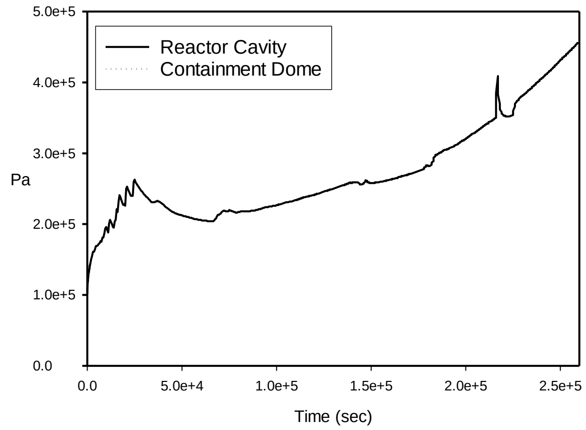
(SBLOCA)

Fig. 3. CINEMA results on corium mass in the reactor cavity for the LOCA without the SI.

Figs. 4 and 5 show the CINEMA results on the containment pressure and temperature in the LOCA without the SI of the APR1400, respectively. After reactor vessel failure, containment pressure and temperature increase gradually. After reactor vessel failure, reactor cavity temperature rapidly increases. Fig. 6 shows the CINEMA results on the hydrogen concentration of containment in the LOCA without the SI of APR1400. In general, containment pressure and temperature increase gradually. After reactor vessel failure, hydrogen concentration of the reactor cavity rapidly increases. After that, it decreases due to the hydrogen move to the containment. After reactor vessel failure, hydrogen concentration of the containment dome increases gradually.

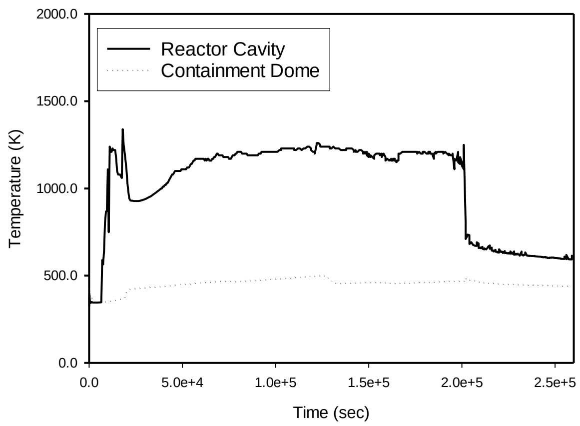


(LBLOCA)

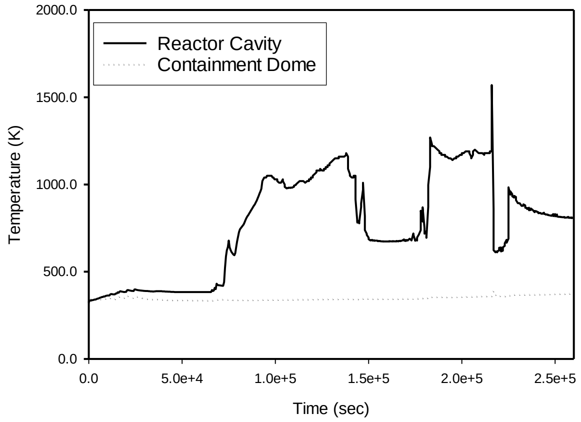


(SBLOCA)

Fig. 4. CINEMA results on containment pressure in the LOCA without the SI.

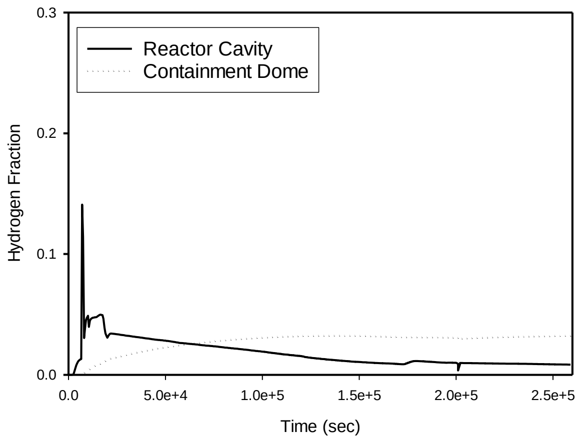


(LBLOCA)

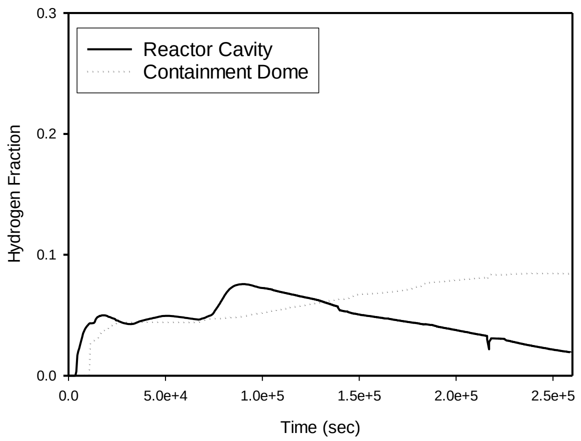


(SBLOCA)

Fig. 5. CINEMA results on containment temperature in the LOCA without the SI.



(LBLOCA)



(SBLOCA)

Fig. 6. CINEMA results on hydrogen concentration of containment in the LOCA without the SI.

4. Conclusion

As part of CINEMA verification for a real power plant, a preliminary analysis of the LOCAs without SI for the real power plant has been performed using the CINEMA computer code. The pressure behavior, fuel mass change by melting and relocation, hydrogen generation mass, relocated corium mass in the reactor cavity, and containment pressure and temperature, and hydrogen concentration of the containment showed the reasonable values. More preliminary analysis of containment performance after the reactor vessel failure wil.be performed for the real power plant. In addition, preliminary analysis of SIRIUS models for the fission product behavior will be performed as part of CINEMA validation and verification.

ACKNOWLEDGMENTS

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