Reactor Core Coolability Analysis during Hypothesized Severe Accidents of OPR1000

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1. Introduction

Since the severe accident of the Fukushima Daiichi nuclear power plant occurred in 2011, most of the countries operating the nuclear power plants start to revisit the safety of the plants. Various activities related to the reassessment of the nuclear safety have been performed throughout the worldwide nuclear industries, regulatory bodies, and research organizations. Accordingly, there have been many additional demands to improve the safety systems of the existing and newly constructed power plants. Assessment of the safety features over the hypothesized severe accidents may be performed experimentally or numerically. Due to the considerable time and expenditures, experimental assessment is implemented only to the limited cases. Therefore numerical assessment has played a major role in revisiting severe accident analysis of the existing or newly designed power plants.

Computer codes for the numerical analysis of severe accidents are categorized as the fast running integral code and detailed code. Fast running integral codes are characterized by a well-balanced combination of detailed and simplified models for the simulation of the relevant phenomena within an NPP in the case of a severe accident. MAAP, MELCOR and ASTEC belong to the examples of fast running integral codes. Detailed code is to model as far as possible all relevant phenomena in detail by mechanistic models. The examples of detailed code is SCDAP/RELAP5 [1].

Using the MELCOR, Carbajo. investigated sensitivity studies of Station Black Out (SBO) using the MELCOR for Peach Bottom BWR [2]. Park et al. conduct regulatory research of the PWR severe accident [3]. Ahn et al. research sensitivity analysis of the severe accident for APR1400 with MELCOR 1.8.4 [4]. Lee et al. investigated RCS depressurization strategy and developed a core coolability map for independent scenarios of Small Break Loss-of-Coolant Accident (SBLOCA), SBO, and Total Loss of Feed Water (TLOFW) [5].

In this study, three initiating cases were selected, which are SBLOCA without SI, SBO, and TLOFW. The initiating cases exhibit the highest probability of transitioning into core damage according to PSA 1 of OPR 1000 [6]. Reactor coolability analysis has been performed and as a representative indicator, Jakob number (Ja) was introduced because it indicates the ratio of latent heat to sensible heat. It determines how much the thermal state of a reactor core is deviated from the subcooled condition. Therefore it is proposed that analysis of Ja may enlighten the understanding on the reactor coolability during the accident management.

2. Numerical Description

2.1 Plant Specification of OPR1000

As a reference plant, Korean Optimized Power Reactor (OPR) 1000 was selected for the severe accident analysis using the MELCOR. Table I shows the operating conditions of OPR1000. The OPR1000 consists of 2 loops of nuclear steam supply systems (NSSS). Electrical output is 1000 MWe, core power is 2815 MWt, and cladding material is ZIRLO^{TM}. Nominal operation conditions of OPR1000 are available in the Final Safety Analysis Report (FSAR) [7].

Table I: The operating conditions of OPR1000

Parameter	FSAR
Core Thermal Power [MWt]	2,815
RCS pressure [MPa]	15.5
Core inlet temperature [K]	569
Core outlet temperature [K]	600
Primary flow rate [kg/s]	15,306
Secondary side pressure [MPa]	7.37
Steam flow per S.G [kg/s]	800

2.2 MELCOR Modelling of OPR1000

Fig 1 shows a MELCOR nodalization used for the MELCOR simulation. The MELCOR nodalization of the primary side of the OPR1000 consists of one core, two hot legs, four cold legs, one pressurizer, and two steam generators. There is a dedicated Volume 190 to measure Core Exit Temperature (CET).

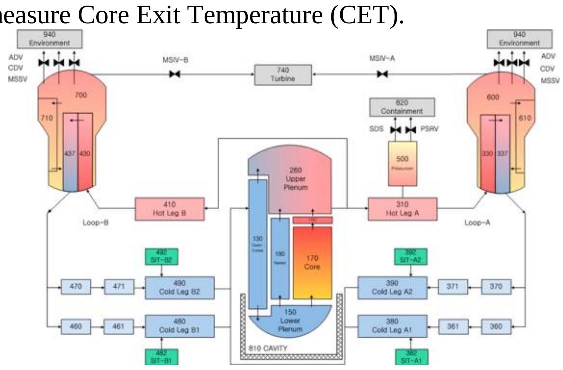

Fig 1. MELCOR nodalization

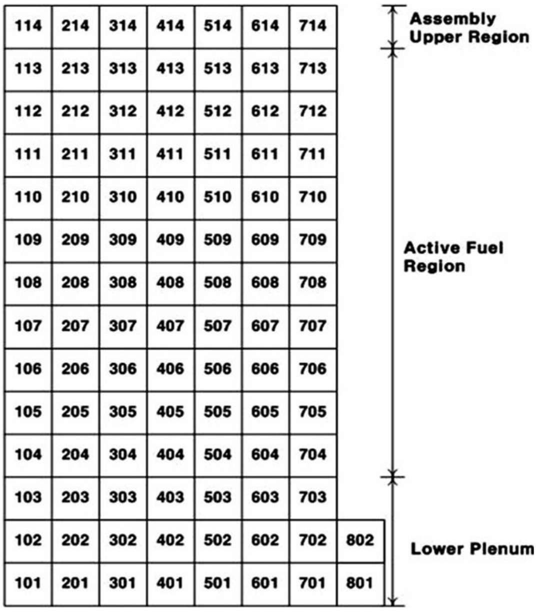

Fig 2. MELCOR nodalization for core

Fig 2 shows a detailed MELCOR nodalization of OPR1000 reactor core. It was divided into three parts: upper region assembly, active fuel region, and lower plenum. The upper region assembly, active fuel region, and the lower plenum consist of 7 cells, 70 cells (7 rings and 10 parts), and 23 cells (8 rings and 3parts), respectively.

2.3 Simulation Matrix

Three initiating events of SBLOCA without SI, SBO, and TLOFW were simulated without employing any mitigation strategies. The SBLOCA assumed that there is a 1.35 inches-break on a cold leg without High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI). The SBO assumed stopping of all off-site power with survival of on-site diesel generator and emergency battery. The TLOFW assumed stopping all feed water. Table II shows the probability initiating events transitioning into core damage.

Table II: The probability of core damage converted by initiating events

Initiating events	Core damage probability (%)
SBLOCA without SI	22.4
SBO	14.4
TLOFW	13.8
Steam Generator Tube Rupture (SGTR)	13.8
Large Break Loss of Coolant Accident (LBLOCA) without SI	12.7
Medium Break Loss of Coolant Accident (MBLOCA) without SI	7.7

3. Results and Discussion

3.1 Major Accident Sequences

Table III shows major accident sequences for the simulated scenarios. The accidents were initiated at time=0 by receiving a reactor trip signal from the pressurizer for SBLOCA, the loss of power signal for SBO, and steam generator low water level signal for TLOFW.

Table III: Major Accident Sequences

	SBLOCA (sec)	SBO (sec)	TLOFW (sec)
Oxidation	8,450	8,234	3,595
Cladding melting	9,480	9,569	4,596
Fuel melting	9,620	9,647	4,673
Fuel relocation to lower head	9,620	10,146	5,338
SITs injection	13,100	-	-
RPV failure	19,100	13,733	8,662

Fig 3 shows the RCS pressure for three initiating events, in which RCS pressure variations differ with the scenarios. It is noted that the SITs are passive devices of injecting borated water, which is actuated by the RCS pressure set point of 4.3 MPa. It is observed that SIT works only when the system pressure decrease and reach to the set point and thus the SIT injection is not observed for the SBO and TLOFW. In case of the SBLOCA, SITs are actuated after molten pool formation.

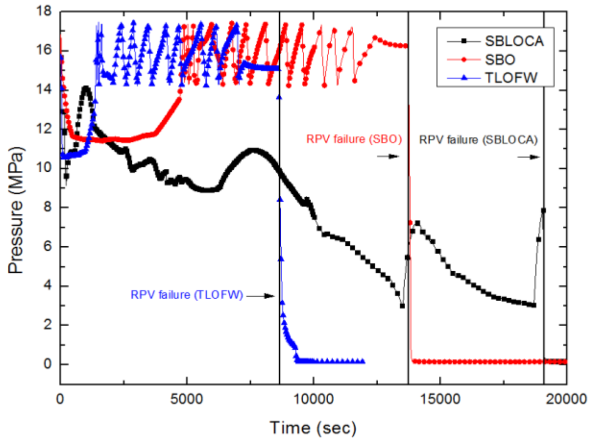

Fig 3. RCS pressure for SBLOCA, SBO and TLOFW

Table IV shows RCS pressure of major accident sequences. As the SBLOCA had a 1.35 inches-break on a cold leg, the pressure of RCS is lower in SBLOCA than in SBO and TLOFW.

Table IV: RCS pressure of Major Accident Sequences

	SBLOCA (MPa)	SBO (MPa)	TLOFW (MPa)
Oxidation	10.24	14.41	14.22

Cladding Melting	8.36	16.25	16.62
Fuel melting	8.18	14.33	17.39
Fuel Relocation to lower head	8.18	16.62	16.47

3.2 Indication of Reactor Core Coolability by Jakob Number

Jakob number is the dimensionless number formulated by Bosnjakovic. Ja indicates the ratio of latent heat to sensible heat. In general, Jakob number is used to derive an approximate formula for the growth of a bubble in a uniformly superheated liquid and in nonuniform temperature fields [8]. Extended use of Ja is possible for the analysis of the reactor core coolability. For example, if sufficient cooling is not implemented, the core water inventory will be heated up and significant superheat is produced. Accordingly Ja will increases as the reactor core heats up. Thus the introduction of Ja is beneficial in figuring out the reactor core state. It is hypothesized that there may exist a critical Jakob number which represent the significant physical change of the reactor core. As such, it is our intention to collect the MELCOR simulation data and display into the interpretable set of reprocess to better judge the reactor core coolability.

$$\text{Ja} = \frac{c_{p,f}\rho_f (T_0 - T_{sat})}{h_{fg}\rho_g} \qquad (1)$$

ρ_f: Density of fluid [kg/m^3]
ρ_g: Density of vapor [kg/m^3]
$c_{p,f}$: Specific heat capacity of fluid [kJ/kg^{o}K]
h_{fg}: Latent heat of vaporization [kJ/kg]
T_o: Temperature of the superheated liquid [K]
T_{sat}: Saturation temperature [K]

Accident management is never a routine practice that can be easily implemented. It involves the various readings of the plant data of CET, pressure, core water level, hot-leg and cold-leg temperatures, SG water level, to mention a few. Major accident monitoring parameters are pressurizer pressure and CET [1]. Thus, monitoring the CET with T_{sat} under the pressurizer pressure renders subcooling margin in OPR 1000 [9]. Note that subcooling margin monitor (SMM) is equipped in the OPR1000 but the collected data are not reprocessed into the interpretable set of processor. In this study, since Ja includes the thermal properties corresponding to the pressure and temperature variables, Ja should be eligible parameter in determining how the reactor core is deviated from subcooled condition.

By replacing the temperature in Eq. (1) with maximum cladding temperature (MCT) and CET, reprocessing parameters of Ja_{MCT} and Ja_{CET} are defined as following.

$$\text{Ja}_{MCT} = \frac{c_{p,f}\rho_f (T_{MCT} - T_{sat})}{h_{fg}\rho_g} \qquad (2)$$

$$\text{Ja}_{CET} = \frac{c_{p,f}\rho_f (T_{CET} - T_{sat})}{h_{fg}\rho_g} \qquad (3)$$

Figs. 4 to 6 show Ja_{MCT} and Ja_{CET} for three initiating events of SBLOCA, SBO, and TLOFW, respectively. There are five distinctive regimes in graph. These are numbered as (1) oxidation, (2) cladding melting, (3) fuel melting, (4) fuel relocation to lower head, and (5) RPV failure. It was observed that Ja_{MCT} and Ja_{CET} increase rapidly from oxidation to cladding melting due to the additional heat generation by exothermic reaction. Since the cladding melting, Ja_{MCT} and Ja_{CET} for all cases behave similarly. Jakob number after RPV failure increases sharply in SBO and TLOFW

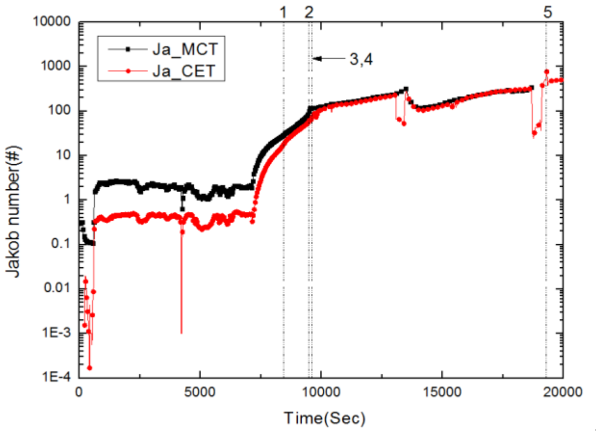

Fig 4. Jakob number for base SBLOCA

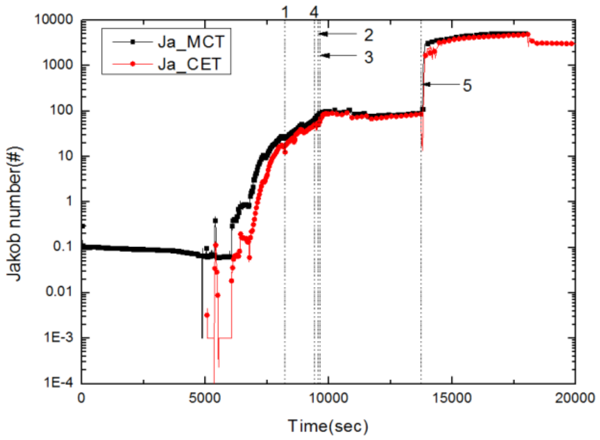

Fig 5. Jakob number for base SBO

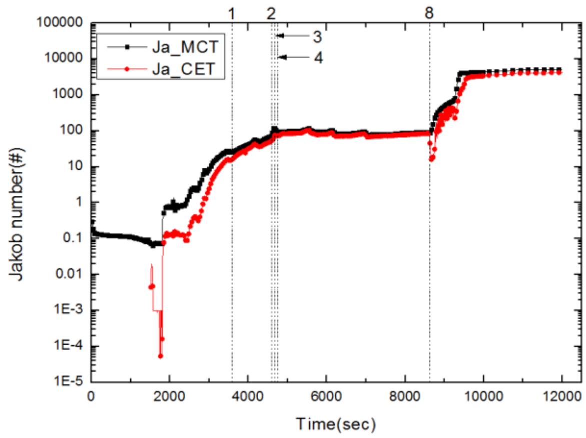

Fig 6. Jakob number for base TLOFW

Tables V and VI show numeric values of Ja_{MCT} and Ja_{CET}, respectively for the major accident sequences of the initiating events. Since the MCT is always higher than the CET, Ja_{MCT} is always higher than Ja_{CET}. The threshold Ja_{CET} for the oxidation ranges 25 to 29, which are on the order of 30. As well, it is observed that the values of Ja_{CET} are more or less similar regardless of the different scenarios.

Table V: Jakob number of MCT

	SBLOCA	SBO	TLOFW
Oxidation	29.57	25.69	26.12
Cladding melting	99.22	82.52	87.28
Fuel melting	119.40	91.53	121.26
Fuel relocation to lower head	119.40	103.57	101.47

Table VI: Jakob number of CET

	SBLOCA	SBO	TLOFW
Oxidation	18.09	12.47	16.14
Cladding melting	57.60	50.45	53.42
Fuel melting	72.29	60.34	73.01
Fuel relocation to lower head	72.29	89.35	87.22

4. Conclusions

The objective of this study is to investigate the reactor core coolability during hypothesized severe accidents of OPR1000. As a representative indicator, we have employed Jakob number and developed Ja_{CET} and Ja_{MCT} using the MELCOR simulation. Although the RCS pressures for the respective accident scenarios were different, the Ja_{MCT} and Ja_{CET} showed similar trends. Therefore, Ja_{CET} is expected to provide judgmental information to the reactor operators for the proper severe accident management.

5. Acknowledgements

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REFERENCES

[1] International Atomic Energy Agency, Approaches and tools for severe accident analysis for nuclear power plants, Vienna (Austria): International Atomic Energy Agency, 2008.
[2] J. J. Carbajo, MELCOR Sensitity Studies for a Low-Pressure, Short-Term Station Blackout at The Peach Bottom Plant, Nuclear Engineeting and Design, Vol 152, pp.287-317, 1994
[3] J. H. Park, H. Kim, C. Lee, Regulatory Research of The PWR Severe Accident, Korea Institute of Nuclear Safety, 2000
[4] K. I. Ahn, S-Y Park, S-W Cho, MELCOR 1.8.4 Sensitivity Analysis of The Severe Accident Evolution During The APR 1400 LOCA, Annals of Nuclear Energy, Vol 33, pp.71-83, 2006
[5] S. Lee, K. S. Ha, H-Y. Kim and S. J. Kim, Validation of RCS Depressurization Strategy and Core Coolability Map for Independent Scenarios of SBLOCA, SBO, and TLOFW, Journal of Nuclear Science and Technology, Vol. 51, No. 2, pp.181-195, 2014
[6] Korea Power Engineering Company, OPR1000 Level 1 PSA Report, 1999
[7] Korea Hydro & Nuclear Power, Shin Kori 1&2 final safety analysis report, 2008.
[8] N. Zuber, The Dynamics of Vapor Bubbles in Nonuniform Temperature Fields, J. Heat Mass Transfer, vol.2, pp.83-98,1961.
[9] Korea Hydro & Nuclear Power, Basic System of Nuclear Power Plant (III), pp67-68, 2008.