Thermal Hydraulic Analysis for Accidents in OPR1000 and Evaluation of Uncertainty for PSA

Tae-jin Kim[a], Yun-je Cho[a], Ho-gon Lim[b], and Goon-cherl Park[a]

[a]*Department of Nuclear Engineering, Seoul National University,*
Gwanak 599, Gwanak-ro, Gwanak-gu, Seoul, Korea, e-mail: ktj1112@snu.ac.kr
[b]*Korea Atomic Energy Research Institute,*
150 Deokjin-Dong, Yuseong-Gu, Daejeon 305-353, Korea

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1 ABSTRACT

Probabilistic Safety Assessment (PSA) is a mathematical tool to evaluate numerical estimates of risk for nuclear power plants (NPPs). For the last couple of years, PSA has been put to practical use for verification of safety in research and industrial fields. However, PSA still has the problems about the quality and reliability. The significant one of the problems is that the quantification of uncertainties from thermal hydraulic (TH) analysis has not been included in the quantification of overall uncertainties in PSA. Since TH phenomena are directly concerned with the accidents occurred in the NPPs, the uncertainties from TH analysis should be treated in PSA. The aim of the present study is to develop a best-estimate thermal hydraulic analysis methodology applicable to PSA and capable to quantify uncertainty from TH analysis. Through the research, the best-estimate TH analyses for large break loss of coolant accident (LBLOCA) of optimized power reactor 1000 MW_{th} (OPR1000) were conducted and probabilistic density function (PDF) of peak cladding temperature (PCT) was obtained using statistical methods. The result of this research can be used to assess the core damage frequency (CDF) from the PSA which was already performed.

2 INTRODUCTION

Recently, the utilization of the risk-informed applications (RIAs), which is based on the result of PSA, have been encouraged in nuclear industry and research area. In step with this movement, U.S. Nuclear Regulatory Commission (NRC) stated "The use of probabilistic risk assessment technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in Probabilistic Risk Assessment methods and data, and in a manner that complements NRC deterministic approach and supports NRC traditional defense-in-depth philosophy." in part of the probabilistic risk assessment policy statement (USNRC, 2000). Besides, PSA is being applied to the nuclear safety regulation by the regulatory bodies in many countries.

Under the background, the quality of PSA must be a key issue. The international atomic energy agency (IAEA) developed quality assurance (QA) programs for PSA and considered an effective and performance-oriented QA to be essential for a reliable PSA. The U.S. NRC provided the guideline necessary to enhance the PSA quality and suggested the phased approach to PSA quality. The Korea Institute of Nuclear Safety (KINS) has also developed a PSA model for the regulation and continues trying to perform the peer review with both domestic and foreign experts.

TH analysis in PSA is performed to obtain the information about construction of event tree and fault tree and about the standard of success/failure for each tree. Thus the result of analysis should be treated necessarily and in detail. Up to the recent, the traditional approach, conservative analysis with the concept of defence in depth, has been applied without consideration of uncertainties from TH analysis. However, as more and more the regulators support the policy for the use of risk-informed regulation (RIR), the best-estimate analysis has been noticed in academic, industrial and researchable world. Examining the result from the best-estimate TH analysis thoroughly, even the accident sequence which turned out to be success from an existing PSA result can have probability to fail, what is worse, the total CDF can also be obtained incorrectly. Therefore the uncertainties originated from TH analysis have to be quantified.

Generally, the uncertainties in the TH analysis were induced from the code models, the input data from the plant operation condition, the nodalization and the inherent uncertainty of the code. In the present study, the TH analysis was performed with considering the uncertainties from the input data and the models with the MARS code and the LBLOCA in OPR1000 was chosen as the pilot accident sequence as a preliminary work to construct the general procedure for the quantification of uncertainties from the TH analysis.

3 SELECTION OF CANDIDATE PARAMETERS

3.1 Target plant and pilot accident sequence

The OPR1000, known as Korean Standard Nuclear Plant (KSNP), was selected for the analysis. The OPR1000 is pressurized water reactor (PWR) and has 2815 MW_{th} of power. The plant consists of two loops, each of which has one steam generator, two cold legs, one hot leg and two reactor coolant pumps. The safety injection from safety injection tank (SIT), the high pressure safety injection (HPSI), and the low pressure safety injection (LPSI) function in the connection with the cold legs. Table 1 shows the main design parameters of the OPR1000 and the input parameters for MARS code calculation.

Korea Hydro & Nuclear Power Co. (KHNP) performed PSA for OPR1000 and arranged the CDF for various kinds of accidents. Among the results of LOCAs, the sequence which has the highest CDF was picked up as the pilot accident sequence. The sequence has several assumptions as follows: 1) a hypothesized double-ended cold leg break between the emergency core coolant injection nozzle and the reactor vessel, 2) failure of all LPSI pumps and motor-driven auxiliary feed water pumps, 3) failure of 50% (one of two) HPSI pump and 50% (two of four) SITs, 4) 30.0 seconds time delay for HPSI pumps after the pressurizer pressure decrease to 12.1 MPa.

Table 1. Main parameters of OPR1000

Parameter	FSAR*	MARS
Primary Side		
Reactor power [MW_{th}]	2871	2815
RCS pressure [MPa]	15.51	15.81
Hot leg temp. [K]	600.8	600.6
Cold leg temp. [K]	568.8	569.3
Number of S/G	2	2
SIT		
Total volume of SIT [m^3]	68.0	68.0
Operating temp. of SIT [K]	321.2	299.8
Set pressure of SIT [MPa]	4.033	4.206
RWST		
Water temp. in RWST [K]	299.5	299.8
Set pressure of HPSI [MPa]	12.141	time delay
Flow rate of HPSI [kg s^{-1}]	min. 63.060	table
Set pressure of LPSI [MPa]	1.138	time delay
Flow rate of LPSI [kg s^{-1}]	min.352.681	table
Secondary Side		
S/G pressure [MPa]	7.37	7.43
Set pressure of MSSV [MPa]	8.804	8.618
Set pressure of MSIV [MPa]	6.101	6.105

**FSAR: Final Safety Analysis Report, min.: minimum*

3.2 Analysis code and modeling

The MARS code was developed by the Korea Atomic Energy Research Institute (KAERI) by consolidating and restructuring the RELAP5/MOD3.2 and COBRA-TF codes. The MARS code, best-estimating code, is capable of analyzing a one-dimensional or three-dimensional thermal hydraulic system and the fuel responses of the light water reactor transients. The one-dimensional module of MARS code adopted that of RELAP5. It is known that RELAP and RELAP-based codes can provide the best-estimate results as predicting the TH behavior of nuclear power plants. Since the best-estimate results are preferable for TH analysis in PSA, the MARS code was selected as an analysis code.

The nodalization diagram of the MARS code for the OPR1000 is shown in Fig. 1. All components including the core and the downcomer were simulated by the one-dimensional component models such as a single volume, a time dependent volume, a junction, a pipe, and an annulus. A pressurizer, pumps, accumulators and valves are simulated by the special component models.

Generally, nominal values were so used for most of modeling parameters and other TH parameters that the excessive conservatism was eliminated on the analysis. Table 1 is showing an example that the set pressure of SIT used in the FSAR 4.033 MPa but we used the nominal value of 4.206 MPa as the reference value.

3.3 Phenomena Identification and Ranking Table (PIRT) and candidate parameters

As a first step of the uncertainty quantification in TH analysis, a set of candidate parameters were surveyed for the parameter assessment. Since PCT is considered as the criteria of the reactor core damage, the phenomena and parameters affecting PCT need to be assessed and thus three PIRTs were reviewed: two are for the LBLOCA in an OPR1000 and the other is for a LBLOCA in a Westinghouse four-loop PWR.

From the review, the 31 candidate parameters were set up and the probability distributions, uncertainty ranges and reference values for all parameters were surveyed additionally. Table 2 shows the result of survey. The information on the uncertainties of parameters was found in the technical specification document of FSAR for OPR1000 and previous uncertainty analysis results in the pre-performed PSA. The uncertainty ranges are expressed with a standard deviation when the parameter has the normal distribution and with a maximum and minimum value when the parameter has the uniform distribution, respectively. In case that the probability distribution is unknown, the uniform distribution was assumed.

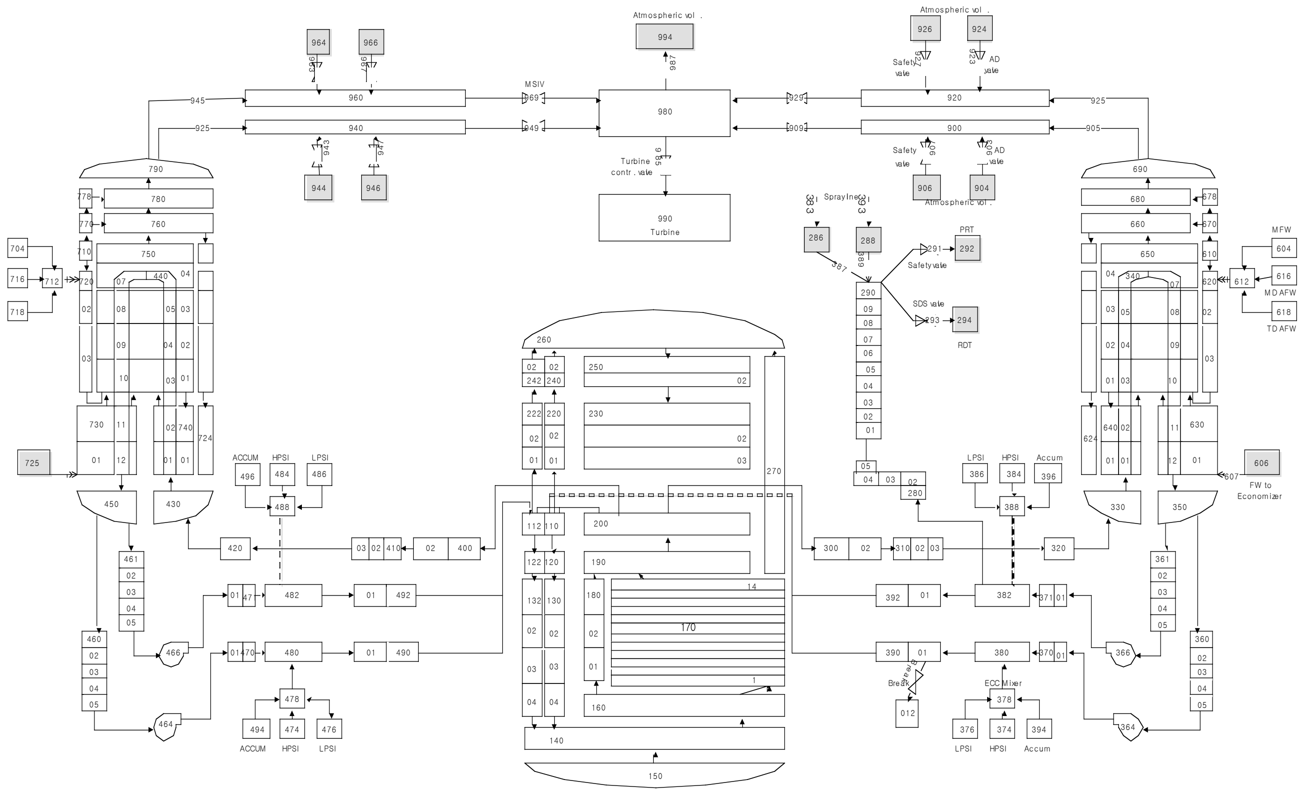

Figure 1. Nodalization diagram for the OPR-1000

4 PARAMETER ASSESSMENT

To perform the sensitivity calculations of the candidate parameters, the several values within uncertainty range of each parameter were applied to the input data of MARS code, respectively. The effect on the PCT by the uncertainty of each parameter was estimated and all parameters were ranked based on the result of the parameter assessment. Five high-ranked parameters were identified as the effective parameters to find PDF.

Table 2. Candidate parameters and uncertainty ranges

	Parameter	**Probability Distribution**	**Uncertainty Range**	**Reference Value**	**Description**
1	Gap Conductance	Uniform	0.05 mm – 0.360 mm	0.092 mm	3
2	Power Peaking Factor	Uniform	1.50535 - 1.71465	1.61	2(± 5.6%)
3	Operating Plant Power	Normal	98% - 102%	2815	
4	Decay Heat	Normal	93.4% - 106.6 %	ANS79-1	2(± 6.6%)
5	Axial Power Distribution	Top/bottom	Assumption	8th node	±3 nodes
6	Fuel Thermal Conductivity	Normal	90% - 110 %	table	Function of temperature
7	Cladding Thermal Conductivity	Normal	88% - 112 %	table	Function of temperature
8	Pellet Heat Capacity	Normal	90% - 110 %	table	Function of temperature
9	Time Step Size	Uniform	0.0025 – 0.01	0.005	
10	Loop Mass Flow Rate	Uniform (assumption)	3800.5 kg $m^{-2} s^{-1}$ - 3854.0 kg $m^{-2} s^{-1}$	3827.25 kg $m^{-2} s^{-1}$	
11	Rod Internal Pressure	Uniform	2 MPa –15 MPa	8 MPa	FSAR
12	Density Reactivity	Uniform	table	table	Function of moderator density
13	Radial Power Distribution	Discrete	Assumption	1	
14	RCP Coastdown/Running	Uniform (assumption)	90% - 110%	4985.8 ·	Momentum of inertia
15	Core Bypass Flow	Discrete	2.5% - 3.0%	2.75%	Bypass flow fraction to total cold leg mass flow
16	Cladding Swell & Rupture Model	Discrete	0~0.00007 m	0	Variation of gap size
17	RCP 2-1Phase Head Multiplier	Uniform	0 - 1.0	0.5	
18	Doppler Reactivity	Uniform	table	table	
19	Cladding Heat Capacity	Uniform	90% - 110%	table	Function of temperature
20	HPSI Flow Rate	Uniform	table	table	
21	SIT Gas Pressure	Uniform	39.92 MPa - 44.20 MPa	42.058 MPa	
22	SIT Effective Water Volume	Uniform	50.531 - 54.411	52.471 m^3	
23	SIT Water Temperature	Uniform	280.30 K – 319.30 K	299.7964 K	
24	Containment Initial Pressure	Uniform (assumption)	0.9777 bar -1.0494 bar	1.01353 bar	
25	Containment Net Free Volume	Uniform (assumption)	77220.5 m^3- 81468.0 m^3	79287.17 m^3	
26	Oxidization Dial	Normal	on/off	on	
27	Groeneveld CHF Dial	Normal	0.9 - 1.1	1.0	
28	Pump K-Factor	Uniform	0.02 - 0.8	0.41	
29	Pump Torque Multiplier	Uniform	0 - 1.0	0.5	
30	Pressurizer Pressure	Uniform	153.31 bar - 156.89 bar	155.1 bar	
31	CCFL Model	-	Wallis/Kutateladze	Wallis	

4.1 Input preparation

Minimum, maximum and reference values for one parameter were input to the MARS code input, respectively. Consequently, total 63 calculations for 31 parameters were performed: when the reference value is applied, only one calculation is required for all parameters since the reference case of each parameter is all the same. The reference value represents the nominal value which comes from the operating and design conditions and thermo-physical data.

Since the uncertainty ranges of the axial power distribution and the rod internal pressure were unknown, the proper assumptions were applied as below. In the case of the axial power distribution, we set the reference case in which 8th node of fourteen nodes for the fuel rods has the biggest power density. Furthermore, for sensitivity analysis, we considered two more cases that 5th and 11th node have the biggest power density, respectively. In regard of rod internal pressure, by referring the safety analysis report in the FSAR of OPR1000, the reference, maximum and minimum values were set as 8.0 MPa, 2.0 MPa and 15.0 MPa, respectively.

For the best-estimate calculation, the pressure of the secondary side was calculated by MAAP code with same initial and boundary conditions. The calculated pressure was considered as boundary condition in the MARS code.

4.2 Result of reference case

The transient of water level in the core for the reference case is shown in Fig. 2 to understand the procedure of the accident. As soon as the cold leg is broken, the reactor comes to trip out at 0.27 seconds. The core is uncovered in 1.0 second and begins to recover at 37.0 seconds due to the safety injection from the SIT and reaches to fully recovered state at 76.0 seconds. After the SIT is depleted at 77.0 seconds, the core water level is decreased again and fully recovered near 3600.0 seconds due to the HPSI.

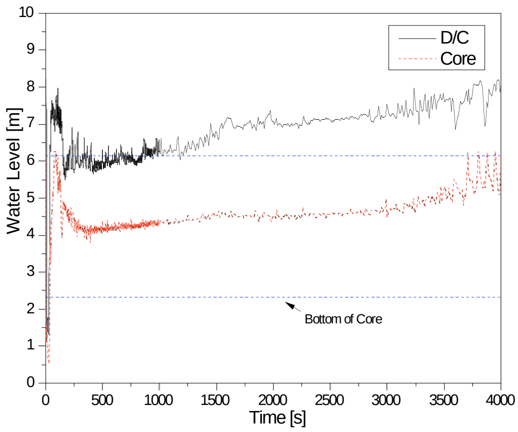

Figure 2. Transient of water level in reference case

The pressure of primary side is sharply decreased to about 0.15 MPa in 34.0 seconds and reaches to equilibrium state with the pressure of the secondary side at 0.3 MPa. The SIT starts the injection at 12.5 seconds and HPSI is started with 30.0 seconds time delay after the pressure of pressurizer is reduced to 12.1 MPa.

4.3 Result of sensitivity calculation

Figure 3 shows the result of the sensitivity calculations, which contains the effects of the uncertainties of the candidate parameters on the PCT. Table 3 shows numerically the difference of PCTs between the reference case and the case with maximum or minimum value. All parameters were ranked according to the maximum difference of PCT for each peak. In most cases, the PCT transients show the similar trend with the reference case, the result of which has three peaks: the blowdown PCT in 5.0 seconds, the first reflood PCT between 30.0 and 50.0 seconds, and the second reflood PCT between 110.0 and 140.0 seconds. Exceptionally, the case with maximum time step size shows the different PCT trend from that of the reference case. From the result, the time step size among the parameters might be identified as an important parameter. However, in the present study, not the trend of temperature transient but PCT was of interest and the time step size had less effect on PCT than the other high-ranked parameters did. Hence this study excluded the time step size.

Consequently, the PDF of PCT was obtained only for the blowdown PCT because the temperature range of the blowdown PCT is much higher than two reflood PCTs. As a result of sensitivity calculations, the gap conductance, the power peaking factor, the fuel thermal conductivity, the cladding heat capacity and the RCP 2-1 phase head multiplier were identified as the effective parameters on the PCT.

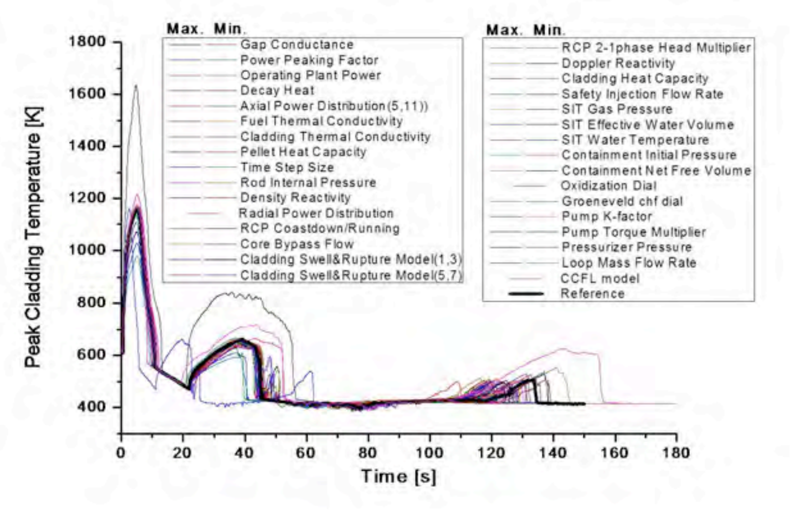

Figure 3. PCT transient for sensitivity calculations

5 QUANTIFICATION OF UNCERTAINTY

To quantify the uncertainties of effective parameters, a statistical method is necessary which can provide enough significant level with the minimum number of calculations and thus statistical methods were reviewed. In this research, the direct Monte-Carlo method (MCM) and the response surface method (RCM) were compared to construct the PDF of the blowdown PCT.

5.1 Statistical methods

The MCM is a statistical method using the randomly sampled values. The algorithm of the Monte-Carlo method is very simple and it is possible to make the proper statistical model by the random sampling when the probability is unknown. However, it requires a lot of calculations to insure enough significant level.

The RSM has been used to identify the important parameter and its error, and to remove the insignificant parameters. The RSM has the advantage that the number of calculations is sharply reduced against the MCM. Therefore, we selected the RSM as a basic method and the result of the direct MCM was compared to estimate the accuracy and efficiency of the RSM.

Table 4. Calculation matrix of Box-Behnken method

	P1	P2	P3	P4	P5	PCT
1	0	1	0	-1	0	1198.1
2	1	0	1	0	0	1537.7
3	0	0	1	1	0	1137.7
⋮						
44	**0**	**0**	**-1**	**-1**	**0**	**1189.6**
45	**0**	**-1**	**1**	**0**	**0**	**1121.1**
46	**0**	**1**	**-1**	**0**	**0**	**1209.5**

5.2 Construction of response surface

At first, the calculation matrix was established by using the Box-Behnken method as shown in Table 4. In the table, P1 to P5 mean the gap conductance, the power peaking factor, the fuel thermal conductivity, the