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A Study on Uncertainty and Sensitivity of Operational and Modelling Parameters for Feedwater Line Break Analysis

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급수관 파열사고 해석에 대한 운전변수와 모형변수의 불확실성 및 민감도 연구

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Abstract

Uncertainty analysis of the FLB accident is performed for KNU-1 using the response surface methodology and Monte Carlo simulation. The FLB analyses using the RELAP4/Mod6 were performed a number of times to generate the data base for the uncertainty analysis, along with the EM calculation for comparison purpose. Two kinds of input sets are utilized for response surface method to investigate and compare the effects of the uncertainty of input variables on the RCS peak pressure following a FLB. The first set is composed of six major plant operational parameters and the second set is composed of five major modelling parameters. It is found through the analysis of results that the uncertainties of modelling parameters have more influence on the RCS peak pressure than the uncertainties of plant operational parameters and that the extra margin of 9% of peak presure is gained. And one of the assumptions of EM calculation, which is usually accepted as conservative is found to be erroneous, that is, the initial core inlet temperature is found to act negatively on the RCS pressure following a FLB.

요 약

극한적인 열제거 기능 상실사고인 급수관 파열사고에 대한 불확실성 해석을 반응표면방법과 Monte Carlo모사를 이용해서 원자력 1호기에 대하여 수행하였다. 여러번의 RELAP4/MOD6를 이용한 급수관 파열사고 해석을 통해 불확실성 해석의 Data Base를 마련하였으며, 비교 목적으로 평가모형 계산도 수행하였다. 급수관 파열사고 이후의 원자로 냉각재계통 최대 압력에 미치는 영향을 조사비

교하기 위해 2종류의 입력 Set에 대한 반응표면방법이 활용되었다. 첫 Set는 6개의 주요 발전소 운전변수로 구성되며, 둘째 Set는 5개 주요 모형변수로 구성된다. 결과의 비교 분석을 통해 모형변수의 불확실성이 최대 압력에 미치는 영향이 운전변수 불확실성의 영향보다 매우 큰 것이 밝혀졌고, 최대 압력 증가의 약 9%에 해당되는 여유도 개선도 확인되었다. 또한, 평가모델에서 인정되고 있는 초기 냉각재 노심입구 온도에 대한 가정은 잘못된 것으로 밝혀졌다.

I. Introduction

In the first years of commercial nuclear power, work in the area which is that of system response to complex operational and thermal hydraulic transients, was largely oriented toward large break loss of coolant accident issues and reflected an underlying perception that the whole range of transients could be adequately characterized by the analysis of an extreme limiting case. The TMI accident showed that this was a questionable assumption. In recent years, the emphasis has shifted much more to small break LOCAs and other transients and toward best estimate calculations. One of the main streams of best estimate methodology is to treat statistically the effects of uncertainties of input variables on the output safety parameters. The resulting safety parameters are taken as 95% probability and 95% confidence value and could be used to increase the safety margins.

There are 8 categories of accidents[1] which are to be analyzed to prove the plant safety before the plant is licensed to operate. One of the event categories contains all of the transients which result in a decreas ein the energy removal capability by the secondary system. These events are referred to as undercooling transients because there is a mismatch between the energy being produced in the reactor core and the energy being removed through the secondary system.

The feedwater line break(FLB) accident is generally accepted as the limiting case of the reactor undercooling accident, and is initiated by a break in the main feedwater piping enough to prevent the addition of sufficient feedwater to the steam generator. This transient could result in either a rapid heatup or a rapid cooldown of the plant depending upon the break size and location. However, the cooldown potential of this accident is not considered because the main steam line break accident is generally regarded as the limiting cooldown accident[2]. Therefore the FLB is selected to evaluate the overpressure potential of reactor coolant system (RCS).

In this paper, some statistical methods are utilized to investigate the uncertainty propagation of FLB calculation. The utilized statistical method is the combination of the response surface method and the Monte Carlo simulation[3], and the effect of two sets of input variables on the RCS peak pressure following a FLB are investigated. The first set of input variables is composed of six major plant operational thermal hydraulic parameters, and the second set is composed of five major code modelling parameters. These modelling parameters are varied by controlling the dials of input values. And as a base case, EM calculation of FLB is also performed for a comparative purpose.

II. Feedwater Line Break Analysis

II. 1. Accident description

The FLB transient is initiated by a break in the main feedwater system piping which reduces the ability to remove heat from the RCS. The diversion of the subcooled feedwater flow from both S/Gs to the break causes the S/G temperatures to rise and the S/G mass inventories and water levels to drop. This drastically reduces the primary-to-secondary heat transfer causing

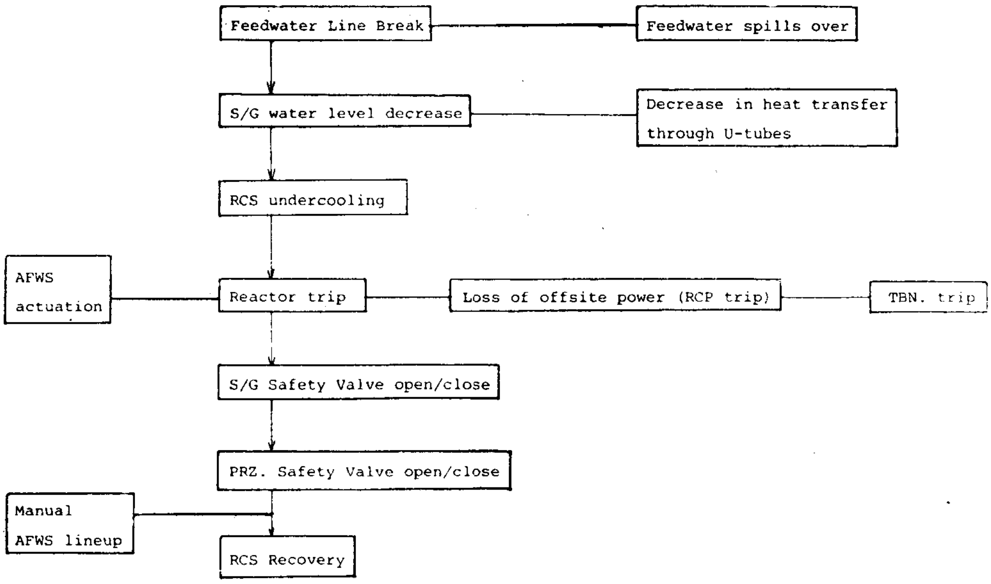

Fig. 1. Event Sequence of FLB

a heatup and pressurization of the primary side of the plant. Eventually the water level in the ruptured S/G decreases to the point where there is a severe reduction in the secondary side heat removal capability. This causes the primary side to heatup even more and culminates in a reactor trip on high primary side pressure. Reactor trip causes the core power and heat flux decline. However, the primary side heatup will still continue after the reactor trip because of the mismatch between the energy removal capability on the secondary side and the energy being produced on the primary side. Eventually this mismatch is eliminated as the core power is reduced to the energy removal capability of the secondary side and a controlled cooldown of the plant can proceed. The severity of the FLB transient depends on a number of system parameters including the break size, initial reactor power, and protection function design, etc., and the schematic event sequence of a FLB is shown in Fig. 1.

II. 2. Method of Analysis

Several detailed analyses using the RELAP4/MOD6 code[4] were performed in order to determine the plant transient following a FLB. An EM calculation was performed first as a base case for the uncertainty analysis of FLB.

The KNU-1 nodalization for FLB analysis is shown in Fig. 2. The nodalization divides the whole system into 36 volumes, 45 junctions, and 11 heat slabs. Initial plant conditions incorporate the instrumental uncertainties of various plant thermal-hydraulic parameters to meet the EM calculation requirements and are shown in Table 1[5,6]. The break size of $0.2 ft^2$ is selected as this break size is reported to be the limiting case[2]. The discharge flow through pressurizer safety valve is simulated by using the negative fill option, and the discharge rate at the design pressure is obtained from the valve specification, which specifies only the steam discharge rate. However, after the pressurizer becoming solid, the discharge through the valve should be based on the two-phase critical flow. As there are no data supplied for the two-phase discharge through safety valve, the data in the reference 7 is utilized. As the S/G secondary side is simulated as one volume, the flow area variation along the elevation could not be

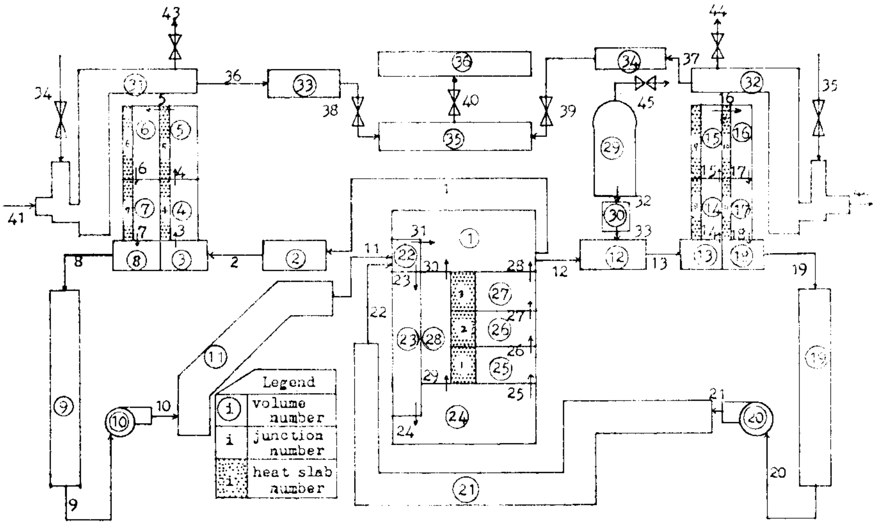

Fig. 2. RELAP4 Nodalization for KNU-1 FLB Analysis

Table 1. Initial Conditions and Assumptions

Parameter	Nominal Value	Used Value	Remark
Core Power, MWt	1,723.5	1,758	2% Calorimetric Error
Core Inlet Temp., °F	541.2	545.2	4°F error
Pressurizer Pressure, psia	2,250	2,280	30 psi error
Pressurizer Water Level, ft.	17.84	18.4346	+2% of level guage
S/G Water Level, ft.	42.16	40.96	−10% of narrow range
Loop Coolant Flow, lb/sec	9,403.4	9,403.4	minimum measured flow
S/G Steam Flow, lb/sec	1,042.99	1,043.06	
S/G Steam Pressure, psia	805	805	
S/G Mass Inventory, lb	9.8714×10^4	9.62053×10^4	
U-tube Heat Transfer Area, ft^2	5.15×10^4	5.15×10^4	
Feedwater Flow, lb/sec	1,042.99	1,043.06	
S/G Height, ft	60.3133	126.4547	Consistent mixture level and mass inventory

simulated. So the S/G height is increased without changing the S/G secondary side total mass and the flow area below the S/G U-tubes to yield the level calculation reasonable.

Before the initiation of transient calculation, it is necessary to simulate the plant steady state condition well to eliminate the error induced by the improper plant thermal-hydraulic input data allocation. To achieve steady state heat balance well, the critical heat flux dial 1.1 is imposed on S/G heat transfer logic between primary to secondary. The dial allows the code user to vary the output from selected RELAP4 correlation and/or models.

To cope with the EM calculation requirements, several assumptions as shown in Table 1 were made.

II. 3. Results and Discussions

Calculated plant parameters for EM calculation following a major feedwater line rupture are shown in figures 3 through 5. Summary of calculated event sequences is shown in Table 2.

As described before, the FLB is analyzed only for the overpressure potential. The overpressure

Table 2. Sequence of Events for EM FLB

Time	Events	Remark
−0.	102% steady state	
0.	Feedline Break	
7.2	'B' S/G lo-lo signal	36.6′
9.8	'B' S/G U-tube top uncovered	33.18′
16.8	'I' S/G lo-lo signal	AFWS actuation signal
22.7	'I' S/G U-tube top uncovered	
39.8	'B' S/G empty	
42.1	Pressurizer hi-p trip signal, Loss of offsite power	2,375 psia, RCP trip, Turbine trip
43.5	PRZ. S/V open	2,500 psia
44.1	Rod begin to drop	delay time : 2 seconds
46.5	PRZ. pressure peak	2,922.41 psia
54.5	PRZ. S/V closed	
76.8	AFWS start to fill intact steam generator	delay time : 60 seconds

limit for the overpressure transient is generally 110% of design pressure[8]. However, for system emergency conditions of which the probability is on the order of 10^{-4}, the limit is 120% of design pressure, resulting in 3000 psia for KNU 1. As the occurring probability of Condition IV accident is regarded on the order of 10^{-4}, the overpressure limit of a FLB is 120% of design pressure[9].

The transient calculation is limited to 75 seconds from the initiation of break because present study is limited to the investigation of peak pressure transient which occurs at the beginning of transient. As a long term, decay heat removal can be achieved by the auxiliary feedwater system, or safety injection system, if RCS condition actuate it.

As shown in Fig. 3, about 39 seconds after the break the broken S/G level decreases to zero resulting in the sharp decrease in the break flow. Before the dryout of broken S/G, the break flow is essentially the saturated water because the break point is simulated to be located at the bottom of S/G shell side. After the dryout, the break flow is the saturated steam in nature. Even after the dryout of broken S/G, the break flow continues and increases

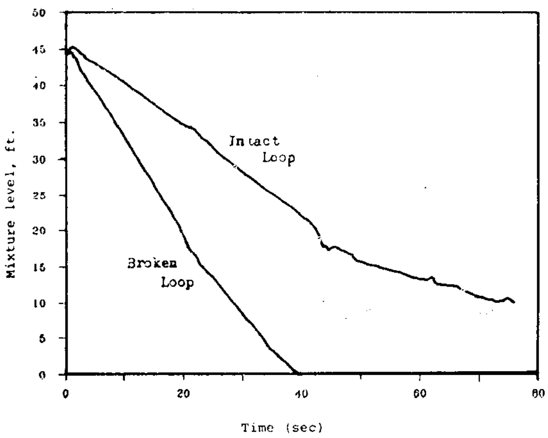

Fig. 3. Steam Generator Mixture Level Transient

slightly. This is due to the fact that the intact S/G level is still maintained as shown in Fig. 3, and the main steam isolation valve is still open to make the steam path through steam line to break, which means the reverse flow through steam line to broken S/G.

Fig. 4 shows the pressurizer pressure transient. The pressure increases slowly until about 42 seconds and after that time it rises sharply for some time. As the broken S/G dries out at about 39 seconds, there are about 3 seconds delay to trigger the pressure jump. This could be explained as follows. As the heat transfer

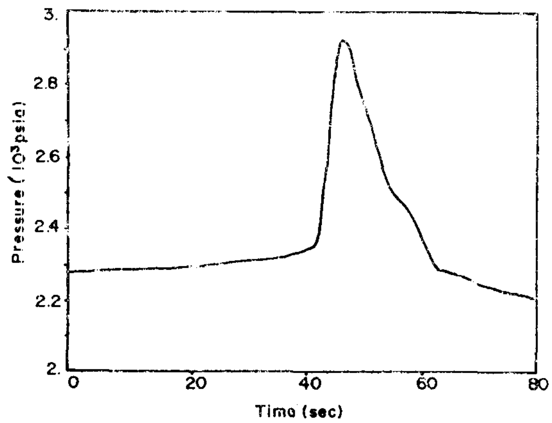

Fig. 4. Pressurizer Pressure Transient

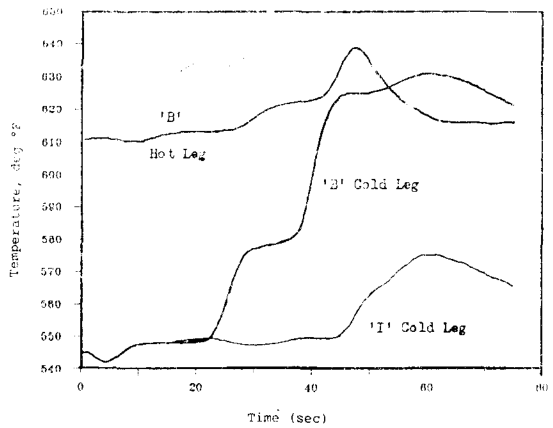

Fig. 5. Loop Temperature Transient

through broken S/G decreases, the primary temperature increases. This in turn increases the temperature differences across the intact S/G U-tube wall to make the heat flux greater. This phenomenon makes the intact S/G level decrease faster to change the heat transfer mode from nucleate boiling to transition or film boiling, this again makes the heat flux through intact S/G U-tubes decrease rapidly at about 42 seconds. And moreover, the pressurizer level goes up to full at about 42 seconds. This also is another reason to explain the pressure jump delay.

Fig. 5 shows the loop temperature transients. The cold leg temperature of broken loop rises rapidly from about 22 seconds. This is related to the broken S/G heat flux decrease. However, as the intact S/G still has its heat sink function, the intact cold leg temperature does not increase so much. For some time interval, broken loop cold leg temperature is higher than hot leg temperature of that loop due to the fact that the broken S/G has almost lost its heat sinkfunction and the mass velocity of coolant decreases because of the reactor coolant pumps trip. In other words, the broken cold leg temperature does not decrease so much as the intact loop temperature dose while the coolant flows through the S/G. But the hot leg temperatures are almost same for both legs because of the flow mixing between intact and broken loop flow.

III. Uncertainty and Sensitivity Assessment of FLB Analysis

III. 1. Uncertainty Analysis of Plant Operational Parameters

As the general method of RSM is well described elswhere[3,10], no detailed description on it is provided in this paper.

1. Variable Selection and Uncertainty Distribution

Based on the results of sensitivity study of FLB to the plant thermal hydraulic and operational parameters which have been performed for the KNU-1[11], six parameters selected important to RCS peak pressure are; initial values of core power, core inlet temperature, pressurizer pressure, steam flow rate, S/G water level, and pressurizer water level.

The values of mean, standard deviation, and their uncertainty distribution types are shown in Table 3. This values are extracted from reference 3 and 12.

2. Building RSM

The thermal hydraulic system analysis code, RELAP4/Mod 6, is used to calculate the RCS

Table 3. Statistical Data of Plant Operational Parameters

Variable No.	Parameters	Mean	Standard Deviation	Uncertainty Distribution
Z_1	Initial Power (% of Rated Thermal Power)	100	1.15	Uniform
Z_2	Core Inlet Temperature, °F	541.2	2.31	Uniform
Z_3	Pressurizer Pressure, psia	2,250	17.32	Uniform
Z_4	Steam Flow Rate, lb/sec	1,043.06	12	Normal
Z_5	Steam Generator Level, ft.	42.16	0.24	Normal
Z_6	Pressurizer Level, ft.	17.84	0.5946	Normal

Table 4. Fractional Factorial Design Points(2^{6-2}) with Peak Pressure Responses

Run No. \ Variables	x_1	x_2	x_3	x_4	x_5	x_6	Reactor Trip time	Peak Pressure time	peak pressure
1	+	+	+	+	+	+	44.3	49	2,875.22
2	+	+	+	−	−	−	44.3	48.5	2,873.91
3	+	+	−	+	−	−	44.3	48.5	2,862.61
4	+	+	−	−	+	+	44.3	49.5	2,874.37
5	+	−	+	+	−	+	45.8	51.5	2,924.07
6	+	−	+	−	+	−	47.8	54	2,869.17
7	+	−	−	+	+	−	48.3	54	2,868.81
8	+	−	−	−	−	+	46.3	51.5	2,914.24
9	−	+	+	+	−	−	45.3	49.5	2,828.72
10	−	+	+	−	+	+	45.3	50	2,852.93
11	−	+	−	+	+	+	46.8	50	2,860.75
12	−	+	−	−	−	−	45.3	49.5	2,824.32
13	−	−	+	+	+	−	49.3	54.5	2,879.16
14	−	−	+	−	−	+	46.8	52.5	2,917.55
15	−	−	−	+	−	+	47.3	52.5	2,920.31
16	−	−	−	−	+	−	48.3	54	2,868.49

peak pressure according to the experimental design plans for the prescribed six statistical parameters. Two level fractional factorial design (2^{6-2}) is employed to obtain the response surface coefficients. The 2^{6-2} design requires sixteen design points, the levels of which are listed in Table 4 together with RCS peak pressure responses calculated by RELAP4/Mod6.

The number of unknowns of the response surface coefficients, b for 2^{6-2} design are seven, and are calculated to be the values as shown in Table 5.

Table 5. Response Surface Model for Plant Operational Parameters

Parameter	Variable	Coefficient	Regression Coefficient	Regression Coefficient for Zi	Sensitivity Factor(%/%)
Constant	1	b_0	2,875.9144	6,037.82	—
Initial Power	x_1	b_1	6.8858	5.9617	0.2073
Core Inlet Temperatue	x_2	b_2	−19.311	−8.3596	−1.5731
Pressurizer Pressure	x_3	b_3	1.6769	0.09682	0.07575
Steam Flow Rate	x_4	b_4	1.5419	0.1280	0.04642
Steam Generator Level	x_5	b_5	−7.3019	−10.537	−0.15447
Pressurizer Level	x_6	b_6	16.516	48.15	0.29869

As a result, the RSM can be represented by the following analytical approximation.

$$y=2875.9144+6.8858x_1-19.3106x_2 +1.6769x_3+1.5419x_4-7.3019x_5 +16.5156x_6 \quad (1)$$

Replacing the coded value x_i with the real value z_i gives an alternative expression for RCS peak pressure as

$$y=6037.82+5.9617z_1-8.3596z_2 +0.09682z_3+0.1280z_4-10.537z_5 +48.150z_6 \quad (2)$$

Above equation is the final form of the fast running approximation to be used in the Monte Carlo simulation[13].

Eq. 2 can also be used for sensitivity study of input variables. The RCS peak pressure sensitivity factor, which is defined as the percent variation of RCS peak pressure due to the percent variation of input variable is calculated as shown in Table 5.

As a result, the coefficient of determination, R^2 is calculated to be 0.865.

3. Monte Carlo Analysis

1200 Monte Carlo trials using the MOCUP code are performed to produce a sample RCS peak pressure. From the sample, the medium and standard deviation values of RCS peak pressure following a FLB are estimated. The medium value of peak pressure is 2,876 psia and the standard deviation is found to be 30.275 psia as shown in Table 6.

The one side higher limit value at the 95% probability is 2,922 psia. The limit value of pressure is then selected by considering the tolerances of peak pressure itself additional on the 95% confidence level basis, that is,

$$2922+1.645\ \sigma/\sqrt{n}=2923.44$$

where

σ=population standard deviation

n=number of sampling

1.645=one side 95%point of standard normal distution

Table 6. Result of MOCUP Calculation

Item	Operational Parameters	Modelling Parameters
No. of Runs	1,200	1,200
Mean, psia	2,876	3,150
5% Value, psia	2,826	2,860
95% Value, psia	2,922	3,432
Minimum Value, psia	2,780	2,640
Maximum Value, psia	2,970	3,792
Standard Deviation, psia	30.28	174.1

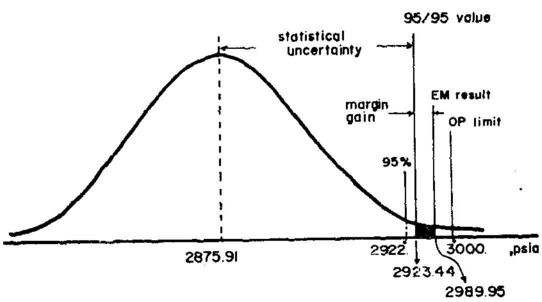

Fig. 6. Peak Pressure Uncertainty Distribution Schematic (Plant Operational Parameters)

Figure 6 shows the schematic RCS peak pressure distribution for this case.

III.2. Uncertainty Analysis of Modelling Parameters

Uncertainty Analysis of code modelling parameters is performed to investigate the effects on the RCS peak pressure following a FLB. This is performed by the use of special feature of "Dial" in RELAP4/Mod6 of which the purpose is to allow the sensitivity or uncertainty analyses to be made. Dial means a multipler or coefficient that will be applied by the program to the value otherwise computed or used. Five code modelling parameters selected important to RCS peak pressure are critical flow model, critcal heat flux correlation, pre-CHF heat transfer coefficients, post-CHF heat transfer coefficients, and pressurizer safety valve discharge flow. The values of mean and standard deviation of modelling paramers, and their uncertainty distribution types are shown in Table 7. and are extracted from reference 12 and 14.

Table 7. Statistical Data of Important Modelling Parameters

Variable No.	Parameters	Mean	Standard deviation	Distribution function
Z_1	Critical flow dial	1	0.375	Uniform
Z_2	Critical heat flux dial	1.1*	0.08	Normal
Z_3	Post-CHF heat transfer coefficient dial	1	0.124	Normal
Z_4	Pre-CHF heat transfer Coefficient Dial	1	0.124	Normal
Z_5	Pressurizer Safety Valve Discharge Rate Dial	1	0.375	Uniform

* Mean value of CHF Dial has been adjusted for proper RELAP4 steady-state initialization.

Table 8. Fractional Factorial Design Points(2^{5-1}) with Peak Pressure Responses

Run No. \ Variable	x_1	x_2	x_3	x_4	x_5	Reactor trip time, sec.	Peak presusre time, sec.	Peak pressure, psia
1	+	+	+	+	+	44.1	48	2,278.26
2	+	+	+	−	−	43.2	48	2,904.93
3	+	+	−	+	+	43.8	48	2,788.86
4	+	+	−	−	−	47.1	53	2,990.78
5	+	−	+	+	−	26.8	33	3,414.39
6	+	−	+	−	+	33.4	39	3,237.98
7	+	−	−	+	−	24.1	31	3,453.63
8	+	−	−	−	+	29.8	35.5	3,311.47
9	−	+	+	+	−	54.8	60	3,145.01
10	−	+	+	−	+	54.7	59.5	3,051.06
11	−	+	−	+	−	54.7	60	3,176.19
12	−	+	−	−	+	54.4	59.5	3,073.75
13	−	−	+	+	+	31.6	37	2,995.89
14	−	−	+	−	−	45.6	53.5	3,453.34
15	−	−	−	+	+	22.9	28	3,185.28
16	−	−	−	−	−	37.6	44	3,432.15

As for the case of plant operational parameter uncertainty analysis, RELAP4/Mod6 is used to calculate the RCS peak pressure according to the experimental design plans for the five prescribed statistical parameters. Two level fractional factorial design (2^{5-1}) is employed to obtain the response surface surface coefficient. The 2^{5-1} design requires sixteen design point, the levels of which are listed in Table 8 together with the peak pressure responses calculated.

The number of unknowns of the response surface coefficients, for this case are six, and

Table 9. Response Surface Model for Modelling Parameters Uncertainty Analysis

Parameter	Variable	Coefficients	Regression Coefficients	Regression Coefficients for Zi	Sensitivity Factor (%/%)
Constant	1	b_0	3,149.56	6,204.5	—
Critical Flow Model	x_1	b_1	−39.5231	−105.39	-3.3462×10^{-2}
CHF Correlation	x_2	b_2	−160.9559	−2,011.9	-7.0334×10^{-1}
Post-CHF Heat Transfer Coefficient	x_3	b_3	−26.9531	−217.36	-6.9013×10^{-2}
Pre-CHF Heat Transfer Coefficient	x_4	b_4	−32.3719	−261.06	-8.2888×10^{-2}
Safety Valve Discharge	x_5	b_5	−96.7419	−257.98	-8.1910×10^{-2}

are calculated to be the values as shown in Table 9.

As a result, the RSM can be represented by following analytical expression.

$$y=3149.56-39.5231x_1-160.9556x_2 -26.9531x_3-32.3719x_4-96.7419x_5 \quad (3)$$

Replacing x_i with z_i gives an alternative expression for RCS peak pressure as

$$y=6204.5-105.39z_1-2011.9z_2-217.36z_3 -261.06z_4-257.98z_5 \quad (4)$$

Sensitivity factors as a result of regression analysis are shown in Table 9 and the coefficient of determinati is calculated to be 0.81.

1200 Monte Carlo trials are performed to produce a sample peak RCS pressure. From this sample, the medium and the standard deviation of peak RCS pressure following a FLB are estimated. The mean value of peak pressure is 3150 psia and the standard deviation is found to be 174.1 as shown in Table 6, together with the results of the operational parameters case. The one side higher limit value at the 95% probability is 3432 psia. The limit value of pressure is then selected to be 3440.27 psia by considering the tolerances of peak pressure itself additional on the 95% confidence basis.

Table 10. Comparison Table

Case		Results, psia
EM without correction		2,922.41
with correction		2,989.95
Operational		
Parameters	mean	2,876
	95%	2,922
	95/95	2,923.44
	Margin gain	66.5
Modelling		
Parameters	mean	3,150
	95	3,432
	95/95	3,440.27

IV. Comparisons and Discussions

RELAP4/Mod6 computer code calculates the peak RCS pressures following FLB according to the experimental design plan. The response surface coefficients are obtained using the least square method. Calculated results for the RCS peak pressures uncertainty distribution indicate the limit value of 2923.44 psia on the 95% probability 95% confidence level basis for the operational parameters input set, and limit value of 3440.27 psia for modelling parameters input set. From the evaluation model calculation, the peak RCS pressure is found to be 2922.41 psia, which is lower than the 95/95 values of the statistical analysis.

It is found, however, that the initial core inlet temperature that is used to be assumed 4°F higher than nominal value in EM calculation has a negative effect on the RCS peak pressure. Present study shows that the volume expansion of reactor coolant due to FLB is greater in the case of the lower initial coolant temperature, if all other thermal-hydraulic conditions are held constant. This could be understood from the thermal-hydraulic standpoint by the following reasons. Lower coolant temperature means the higher coolant density. The pressure increase up to the opening of the pressurizer safety valve is due to the volume expantion of the reactor coolant, since the total mass of the reactor coolant remains unchanged. It is thermal-hydraulically clear that the volume expansion due to RCS heatup is greater in the case of the initially denser reactor coolant, if all other parmeters are held constant. As previously described, the sensitivity factor of core inlet temperature is calculated to be be -1.5731 which means that the percent change of core inlet temperature will decrease the RCS peak pressure 1.5731% of the calculated value.

Taking into account this effect, that is, assuming the core inlet temperature 4°F less than the nominal value would increase the RCS peak pressure by 67.54 psia, resulting in the RCS peak pressure to be 2989.95 psia. This value still has a margin of 10.05 psia to the over-pressure criterion of RCS pressure, 3000 psia which is the 120% of design pressure.

As the statistical analysis of the operational parameters shows the 95/95 value to be 2923.44 psia, the margin is 76.56 psia which increases the EM calculation margin by 66.51 psia.

As the modelling parameter uncertainty and statistical analysis is performed to compare the effects on the RCS peak pressure with the case of operational parameter analysis, its numerical value itself has no great meaning. But it should be noted that the modelling parameters have greater effect on the RCS peak pressure following a FLB than the operational parameters. Therefore, care must be taken deeply in selecting the modelling parameters of the sophiscated system analysis code such as RELAP4/Mod6 for analyzing this kind of accident.

The conficient of determination, R^2, is calculated to be 0.865 for operational parameter case and 0.810 for modelling parameter case. These values would increase a little if three level factorial design plan is employed. But present work is found to be sufficient to predict the general tendency of uncertainty propagation for a FLB.

V. Conclusions

Some conclusions are developed from the present study.

1. RSM and Monte Carlo simulation is a useful method in performing uncertainty and statistical analysis of FLB.
2. The extra margin gained through the operational parameter uncertainty analysis is found to be 66.51 psi which is about 9% of the pressure increase in EM calculation.
3. Initial core inlet temperature has a negative effect on the RCS peak pressure following a FLB, which is opposite to common notion generally acceptded presently.
4. Modelling parameters have much larger effect on the RCS peak pressure following a FLB than the operational parameters.
5. The major parameters are found to be initial core inlet temperature for the operational parameters uncertainty analysis case, and critical heat flux correlation for the modelling parameters case.
6. Future application of RSM can be extended to the statistical assessment of various transient analyses, which is one of the major fields of best estimate methodology.

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