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International Agreement Report

BEPU Analysis and Benchmark with IIST 2% SBLOCA Experiment Using TRACE/DAKOTA

Prepared by:
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ABSTRACT

There are two licensing approaches for evaluation of Loss of Coolant Accidents consequences: conservative methodology and best estimate methodology. Recently, the trend of nuclear reactor safety analysis reveals an increasing interest to substitute best estimate for conservative methodologies to achieve the safety margins and regulate the licensing and operations of nuclear reactors.

The Institute of Nuclear Energy Research Integral System Test (IIST) facility is a test facility to simulate the thermal hydraulics of a Westinghouse 3-loop Pressurized Water Reactor at Maanshan Nuclear Power Plant. The research purposes of the IIST facility are: (a) to enhance the understanding of thermal hydraulics during transients as well as Small Break Loss of Coolant Accidents, (b) to contribute to the evaluations and developments of safety computer codes, (c) to validate the Emergency Operation Procedures during the transients. The scaling factors of the IIST facility for height and volume in the Reactor Coolant System are approximately 1/4 and 1/400, respectively, and the maximum operating pressure is 2.1 MPa. The scaling of hot leg is based on the Froude number criterion to simulate the transition of flow regimes in the horizontal pipes during transients and accidents.

This study is developed the BEPU methodology and the uncertainty results were compared with the IIST experiment data. The IIST TRACE model consists of 89 hydraulic components, 243 control blocks, 39 heat structures and a power component. The data interactions and communications between TRACE and DAKOTA were controlled by SNAP. Finally, correlations between input parameters and output data are calculated for sensitivity study and ranking to investigate what input parameters dominate the contribution of uncertain distribution of PCT.

There are two tasks in this study. One is benchmark of the simulation capability of IIST TRACE model by comparing with the IIST SBLOCA experiment results. The other is BEPU in SBLOCA analysis, several important parameters were considered in the uncertainty quantified. The BEPU analysis is focused on the discussion of measurement uncertainty and model uncertainty. Furthermore, the IIST experimental data were used to benchmark the results of uncertainty analysis. There are 5 parameters taken into account in this uncertainty analysis. An uncertainty band is formed by the 59 calculations, and the benchmark results show the IIST experimental data were located in the uncertainty band. In addition, this study used two methods to calculate correlation coefficients between an input and an output variable: Pearson's correlation coefficient and Spearman's rank correlation. The formula quantifies the correlations between the input and output parameter. Even though the Spearman's rank correlation employs the rank data is difference than the Pearson's correlation, their results are similar trend. The results indicate that choked flow coefficient is the most sensitive parameter. The correlation coefficients of Pearson's and Spearman's are -0.72405 and -0.70199 respectively. Although the measurement error may influence the PCT calculated, its effect is smaller than the model uncertainty.

FOREWORD

The US NRC (United States Nuclear Regulatory Commission) is developing an advanced thermal hydraulic code named TRACE for nuclear power plant safety analysis. The development of TRACE is based on TRAC, integrating RELAP5 and other programs. NRC has determined that in the future, TRACE will be the main code used in thermal hydraulic safety analysis, and no further development of other thermal hydraulic codes such as RELAP5 and TRAC will be continued. A graphic user interface program, SNAP (Symbolic Nuclear Analysis Program) which processes inputs and outputs for TRACE is also under development. One of the features of TRACE is its capacity to model the reactor vessel with 3-D geometry. It can support a more accurate and detailed safety analysis of nuclear power plants. TRACE usually used in the nuclear power plants analysis. This report showed TRACE can also do the calculation of small system such as dry-storage cask.

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of TRACE. INER (Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C.) is the organization in Taiwan responsible for the application of TRACE in thermal hydraulic safety analysis, for recording user's experiences of it, and providing suggestions for its development. To meet this responsibility, the TRACE IIST model has been built. This study is developed the BEPU methodology and the uncertainty results were compared with the IIST experiment data.

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EXECUTIVE SUMMARY

An agreement which includes the development and maintenance of TRACE has been signed between Taiwan and USA on CAMP. INER (Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.) is the organization in Taiwan responsible for applying TRACE to thermal hydraulic safety analysis in order to provide users'experiences and development suggestions. To fulfill this responsibility, the TRACE model of IIST facility is developed. The IIST facility is a Reduced-High and Reduced-Pressure test facility to simulate the thermal hydraulics of a Westinghouse 3-loop PWR at Maanshan Nuclear Plant since 1992. The scaling factors of the IIST facility for height and volume in the Reactor Coolant System are approximately 1/4 and 1/400, respectively, and the maximum operating pressure is 2.1 MPa. The scaling of hot leg is based on the Froude number criterion to simulate the transition of flow regimes in the horizontal pipes during transients and accidents. The research purposes of the IIST facility are: (a) to enhance the understanding of thermal hydraulics during transients as well as SBLOCAs, (b) to contribute to the evaluations and developments of safety computer codes, (c) to validate the Emergency Operation Procedures during the transients.

The codes used in this research are TRACE v5.0p4 and SNAP v2.2.6. There are two tasks in this study. One is benchmark of the simulation capability of IIST TRACE model by comparing with the IIST SBLOCA experiment results. The other is BEPU in SBLOCA analysis, 5 important parameters were considered in the uncertainty quantified. The BEPU analysis is focused on the discussion of measurement uncertainty and model uncertainty. Furthermore, the IIST experimental data were used to benchmark the results of uncertainty analysis. In addition, this study used two methods to calculate correlation coefficients between an input and an output variable: Pearson's correlation coefficient and Spearman's rank correlation. The formula quantifies the correlations between the input and output parameter. Even though the Spearman's rank correlation employs the rank data is difference than the Pearson's correlation, their results are similar trend. The results indicate that choked flow coefficient is the most sensitive parameter. Although the measurement error may influence the PCT calculated, its effect is smaller than the model uncertainty.

ABBREVIATIONS

BEPU	Best Estimate Plus Uncertainty
CAMP	Code Applications and Maintenance Program
CCFL	Counter-Current Flow Limitation
CSAU	Code Scaling, Applicability and Uncertainty
EOP	Emergency Operation Procedures
HPI	High Pressure Injection
IIST	Institute of Nuclear Energy Research Integral System Test
INER	Institute of Nuclear Energy Research Atomic Energy Council, R.O.C.
LOCA	Loss of Coolant Accidents
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PDFs	Probability Distributions Functions
PWR	Pressurized Water Reactor
RHRP	Reduced-High and Reduced-Pressure
PCT	Peak Cladding Temperature
RCS	Reactor Coolant System
SBLOCA	Small Break Loss of Coolant Accident
SG	Steam Generator
SNAP	Symbolic Nuclear Analysis Program
TRACE	TRAC/RELAP Advanced Computational Engine
US	United States

1. INTRODUCTION

There are two licensing approaches for evaluation of Loss of Coolant Accidents (LOCA) consequences: conservative methodology and best estimate methodology. The conservative methodology refers to the conservative analysis using conservative code and conservative assumptions. While the best estimate methodology refers to best estimate (code) calculation that includes uncertainty analysis. Recently, the trend of nuclear reactor safety analysis reveals an increasing interest to substitute best estimate for conservative methodologies which may apply conservative codes or the combination of best-estimate codes and conservative initial and boundary conditions to achieve the safety margins and regulate the licensing and operations of nuclear reactors [1]. Compared with conservative methodologies, the methodologies of Best Estimate Plus Uncertainty (BEPU) adopt best estimate codes and realistic input data with uncertainties to quantify the limiting values i.e., Peak Cladding Temperature (PCT) for LOCAs.

The use of BEPU methods started from 1974. Research during 1974-1988 provided a foundation sufficient for use of realistic and physically based analysis methods [2]. Large number of experimental programs was completed internationally. The United States Nuclear Regulatory Commission (USNRC) developed the Code Scaling, Applicability and Uncertainty (CSAU) method[1] and demonstrated a licensing acceptable best estimate method which could bring benefit to nuclear plant operators (ex. less conservative, consideration of uncertainties, economic gains). After pioneering CSAU in the next five years several new original methods were developed, such as UMAE method (Italy), NE method (UK) [3], GRS method (Germany) [4], IPSN method (France) etc. All of methods consist of two basic elements which are identification and quantification of important parameters as well as analysis to quantify the combined influence of these uncertainties on output parameters. Uncertainty quantification methods calculate probabilistic information about response functions based on simulations performed according to specified input parameter Probability Distributions Functions (PDFs). Sampling methods are often used in uncertainty quantification to calculate a distribution on system performance measures, and to understand which uncertain inputs contribute most to the variance of the outputs.

The Institute of Nuclear Energy Research Integral System Test (IIST) facility is a test facility to simulate the thermal hydraulics of a Westinghouse 3-loop Pressurized Water Reactor (PWR) at Maanshan Nuclear Power Plant (NPP). This study is developed the BEPU methodology and the uncertainty results were compared with the IIST experiment data.

2. METHODOLOGY

Figure 1 shows the BEPU methodology in this study. First, collecting the IIST facility parameters is needed, such as the vessel geometry, pipe length, and flow area. Then, the TRACE IIST model is established and verified with the experiment data of the IIST facility. If the results of benchmark are not consistent with the IIST experiment data, that modify the TRACE IIST model. If yes, then do the next step for uncertainty analysis. In order to do uncertainty analysis, it must to identify and select the important parameters in the specific transient. Important parameters which will affect transient analysis can be divided into three groups, namely initial conditions, accident boundary conditions, and system settings. To define the uncertainty of the parameters, not only the uncertainty range needs to be quantified, but also the distribution function needs to be specified. Three different kinds of elements contribute the total uncertainty of a particular parameter, which involve normal operational range, measurement uncertainty, and model uncertainty. Once the major parameters have been identified and ranged, random sampling of each parameter needs to be performed to generate a run matrix. DAKOTA code is used to generate random variable and to evaluate response data generated by TRACE/SNAP. The sampling method in DAKOTA could choose the Monte-Carlo or Latin Hypercube method, which can be used with probabilistic variables that have the following distributions: normal, lognormal, uniform, loguniform, triangular, exponential, beta, gamma, gumbel, frechet, weibull, poisson, binomial, negative binomial, geometric, hypergeometric, and user-supplied histograms. With Monte-Carlo method, each random variable is produced from its distribution independent of other variables. When set to Latin Hypercube method, the distribution is sliced into equally likely bins, and a random value is produced from each bin.

Because the required minimum number of TRACE runs is dependent of the values of confidence level and probability, Wilks' formula[5] was employed to determinate the minimum number of runs. The correlations between number of code runs, confidence level, and probability of Wilks' formula are defined:

$$1-\alpha^{n}\geq\beta \qquad \text{Eg. 1}$$

where α is probability, β is the confidence level, and n denotes the number of code runs.

Since the value of PCT is the safety criterion to ensure the integrity of fuel assemblies for LOCAs, the minimum number of 59 was used to generate the maximum bound of PCT which achieve 95/95 criterion.

If more than one output needs to be cited from each trial, the Guba's formula [6] can be used:

$$\beta=\sum_{j=0}^{N-P}\frac{N!}{(N-j)!j!}\gamma^{j}(1-\gamma)^{N-j} \qquad \text{Eg. 2}$$

where N is the sample size and P is the number of output variables.
If output variable is only one, that Guba's formula will reduce to Wilk' formula.

All TRACE runs were defined and executed through SNAP job streams[7-8], and TRACE calculation results were read by AptPlot script. The data interactions and communications between TRACE and DAKOTA[9-10] were controlled by SNAP. Finally, correlations between input parameters and output data (ex. PCT) are calculated for sensitivity study and ranking to investigate what input parameters dominate the contribution of uncertain distribution of PCT.

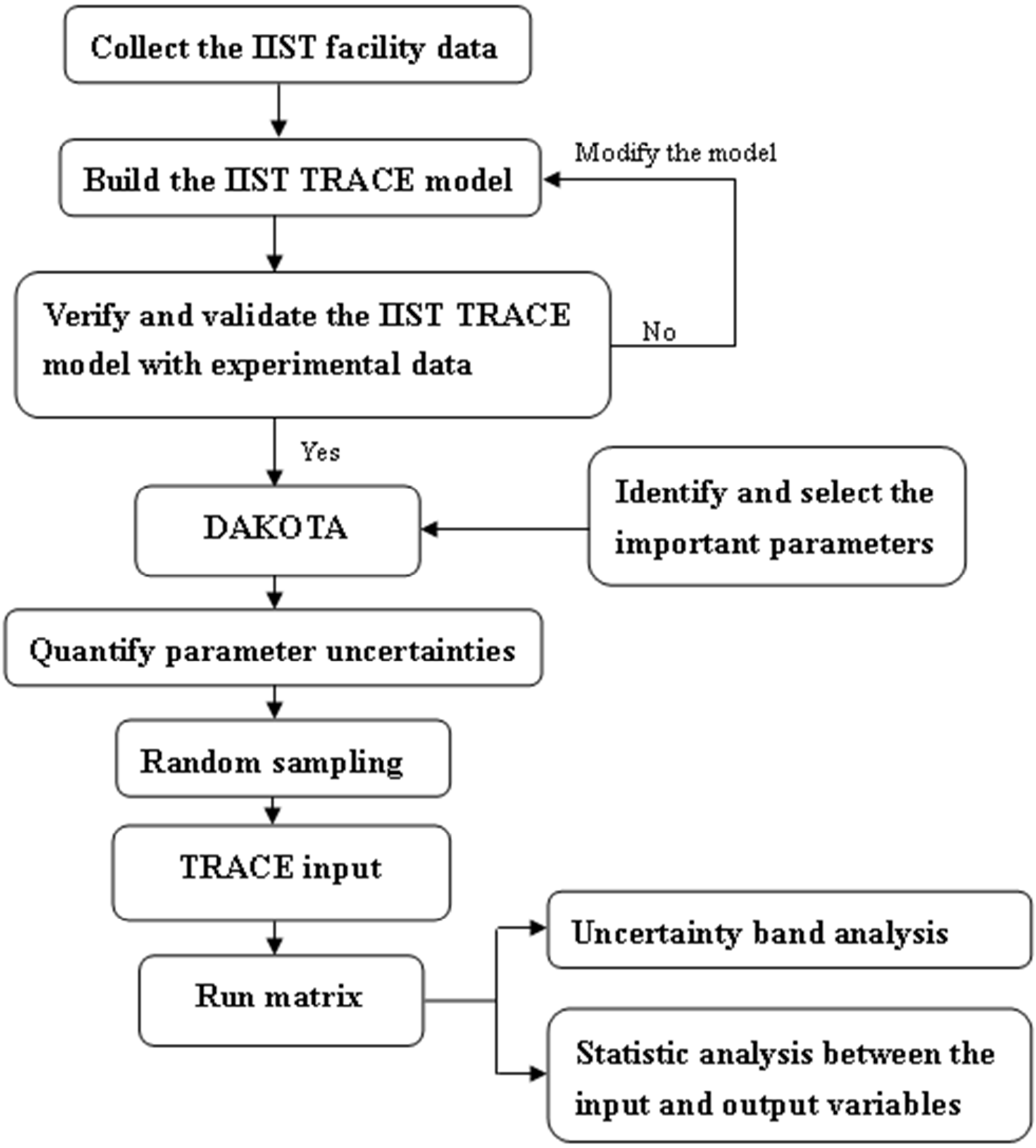

Figure 1 The Flowchart of Uncertainty Analysis

3. FACILITY AND MODEL

3.1 Description of IIST Facility

The IIST facility is a Reduced-High and Reduced-Pressure (RHRP) test facility to simulate the thermal hydraulics of a Westinghouse 3-loop PWR at Maanshan Nuclear Plant since 1992 [11]. The comparisons of key parameters between Maanshan NPP and IIST facility are listed in Table 1. The research purposes of the IIST facility are: (a) to enhance the understanding of thermal hydraulics during transients [12,13] as well as Small Break Loss of Coolant Accidents (SBLOCAs) [14], (b) to contribute to the evaluations and developments of safety computer codes [15,16], (c) to validate the Emergency Operation Procedures (EOP) during the transients [17].

The scaling factors of the IIST facility for height and volume in the Reactor Coolant System (RCS) are approximately 1/4 and 1/400, respectively, and the maximum operating pressure is 2.1 MPa. The scaling of hot leg is based on the Froude number criterion to simulate the transition of flow regimes in the horizontal pipes during transients and accidents. As shown in Figure 2, the IIST facility consists of a pressure vessel and 3 loops. The pressure vessel has 3 inlet and 3 outlet nozzles. Coolant enters the vessel through the inlet nozzles and flows down through the downcomer, and flows up through the heater rods to the outlet nozzles. The bypass flow from the upper plenum to the downcomer is simulated by three external tubes connected with the valves. Each loop has a steam generator and a coolant pump, and the 3 loops are identical, except that there is a pressurizer in the loop 1. The pressurizer connected with loop 1 equips an electrical heater, spray nozzle and pressure relief valves. The capacity of electrical heater is 10 kW, and the penetrations of spray nozzle and pressure relief valves are located on the top of pressurizer. There are 30 U-tubes in each steam generator. However, the steam dome of a steam generator doesn’t contain separators and dryers, because the steam velocity in the steam dome is not strong enough to entrain liquid into seam line at the low core power during simulation of the decay heat level. The secondary feedwater flow rate is controlled by flow control valve actuated by the water level controller of each steam generator. The IIST facility incorporates a data acquisition system which measures temperature, pressure, flow rate, liquid level, and differential pressure.

3.2 Description of SBLOCA Experiment

The experiment of IIST facility was performed in order to simulate a 2% cold-leg break (the break area is 2% of the scaled cold-leg cross-section area) with total High Pressure Injection (HPI) failure[18,19]. A horizontal break nozzle was installed in the cold-leg of loop 2 which is not connected with pressurizer. The initial conditions of this experiment are listed in Table 2. The SBLOCA experiment started from the break occurred at 0s and the primary pressure of IIST facility dropped until it became only a little higher than the secondary-side pressure of IIST facility. The air flowed through the hot-leg into the SG-1 U-tubes after emptying the pressurizer at 128s. After 164s of the break, the loop 1 flow rate suddenly dropped to near zero, which indicates the decrease of the heat removal capability of SG-1. The noncondensable air obviously slowed the temperature rise in both the primary and secondary sides of SG-1 and caused the sudden decrease of the natural-circulation flow rate in loop 1. The collapsed liquid level of core decreased sharply after the break occurred due to the subcooled liquid discharge in the time period between 0s to 146s. Then, the collapsed liquid level of core decreased slowly when the break flow became a two-phase mixture from 146s to 400s. (Finally, because of no coolant makeup, the core was uncovered with heatup at 1734s.) This experiment was terminated at 1734 s because the uncovering of the core was caused by continuous boil-off of vessel coolant inventory without the actuation of coolant makeup system.

3.3 Description of IIST TRACE Model

The IIST TRACE model, which consists of 89 hydraulic components, 243 control blocks, 39 heat structures and a power component, is showed in Figure 3. This model has three loops and each loop includes the simulation of the hot-leg, Steam Generator (SG) inlet plenum, SG U-tubes, SG outlet plenum, crossover leg, coolant pump, and cold-leg. The pressurizer is located in loop 1 and the break valve is located in loop 2. The models of the three SG are identical, and each of the SG consists of downcomer, boiling section and steam dome. The feedwater flow rates are simulated by time-dependent junctions, and the downstream condition of each steam line is simulated by a break component with constant boundary condition. The steam line is simulated by a break component, which simulates the pressure of steam line during the IIST facility experiments. The break flow area is simulated using a specific valve with the critical flow option. Table 3 shows the comparison of the initial condition between IIST facility and IIST TRACE model. The initial conditions of IIST TRACE model are in good agreement with the IIST facility. The SNAP v 2.2.6 and TRACE v 5.0p4 were employed in this research.

Table 1 The Comparisons of Key Parameters between Maanshan NPP and IIST Facility

Parameter	IIST	Maanshan PWR	IIST/PWR
Design pressure (MPa)	2.1	15.6	1.35×10^{-1}
Maximum core power (MW)	0.45	2775	1.62×10^{-4}
Primary system volume (m^3)	5.37×10^{-1}	2.15×10^{2}	2.50×10^{-3}
Number of loops	3	3	1
Core			
Height (m)	1.0	3.6	2.77×10^{-1}
Hydraulic diameter (m)	1.08×10^{-1}	1.22×10^{-2}	8.85
Bypass area (m^2)	7.2×10^{-5}	1.54×10^{-2}	4.67×10^{-3}
Hot leg			
Inner diameter, D (m)	5.25×10^{-2}	7.35×10^{-1}	7.13×10^{-2}
Length, L (m)	2.0	7.28	2.75×10^{-1}
$L / \sqrt{D}$ ($m^{0.5}$)	8.72	8.48	1.03
U-tube in one SG			
Number	30	5626	5.33×10^{-3}
Average length (m)	4.08	16.85	2.24×10^{-1}
Inner diameter (mm)	15.4	15.4	1.0
Volume (m^3)	2.28×10^{-2}	18.44	1.23×10^{-3}
Cold leg			
Inner diameter D (m)	5.25×10^{-2}	7.87×10^{-1}	6.67×10^{-2}
Length L (m)	5.0	15.7	3.18×10^{-1}
$L / \sqrt{D}$ ($m^{0.5}$)	21.8	17.69	1.22
Downcomer			
Flow area (m^2)	0.0185	2.63	7.03×10^{-2}
Hydraulic diameter (m)	4.12×10^{-2}	4.8×10^{-1}	8.58×10^{-2}
Pressurizer			
Volume (m^3)	9.32×10^{-2}	39.64	2.35×10^{-3}
Surge-line flow area (m^2)	3.44×10^{-4}	6.38×10^{-2}	5.39×10^{-3}

Table 2 The Initial Condition of the IIST SBLOCA Experiment

Parameter	IIST test data
Primary coolant system	
Core power (kW)	126
Pressurizer pressure (MPa)	0.958
Pressurizer water level (m)	1.459
Loop flow rate (kg/s)	0.217
Hot-leg temperature (K)	450
Cold-leg temperature (K)	409
Secondary coolant system	
Secondary-side pressure (MPa)	0.295
Secondary-side fluid temperature (K)	407

Table 3 The Comparison of SBLOCA Experiment Initial Condition between IIST Facility and IIST TRACE Model

Parameter	IIST facility	TRACE/error (%)
Primary coolant system		
Core power (kW)	126	126
Pressurizer pressure (MPa)	0.958	0.964 /0.6
Pressurizer water level (mm)	1459	1463 /0.3
Loop flow rate (kg/s)		
Loop1	0.210	0.219 /4.3
Loop2	0.217	0.219 /0.9
Loop3	0.217	0.219 /0.9
Hot-leg temperature (K)		
Loop1	450	448.7 /0.3
Loop2	449	448.7 /0.1
Loop3	451	448.7 /0.5
Cold-leg temperature (K)		
Loop1	409	409.5 /0.1
Loop2	408	409.5 /0.4
Loop3	409	409.5/ 0.1
Secondary coolant system		
Secondary-side pressure (MPa)		
SG-1	0.301	0.303 /0.7
SG-2	0.295	0.299 /1.4
SG-3	0.295	0.299 /1.4
Secondary-side fluid temperature (K)		
SG-1	407	406.1 /0.2
SG-2	407	405.6 /0.3
SG-3	407	405.6 /0.3

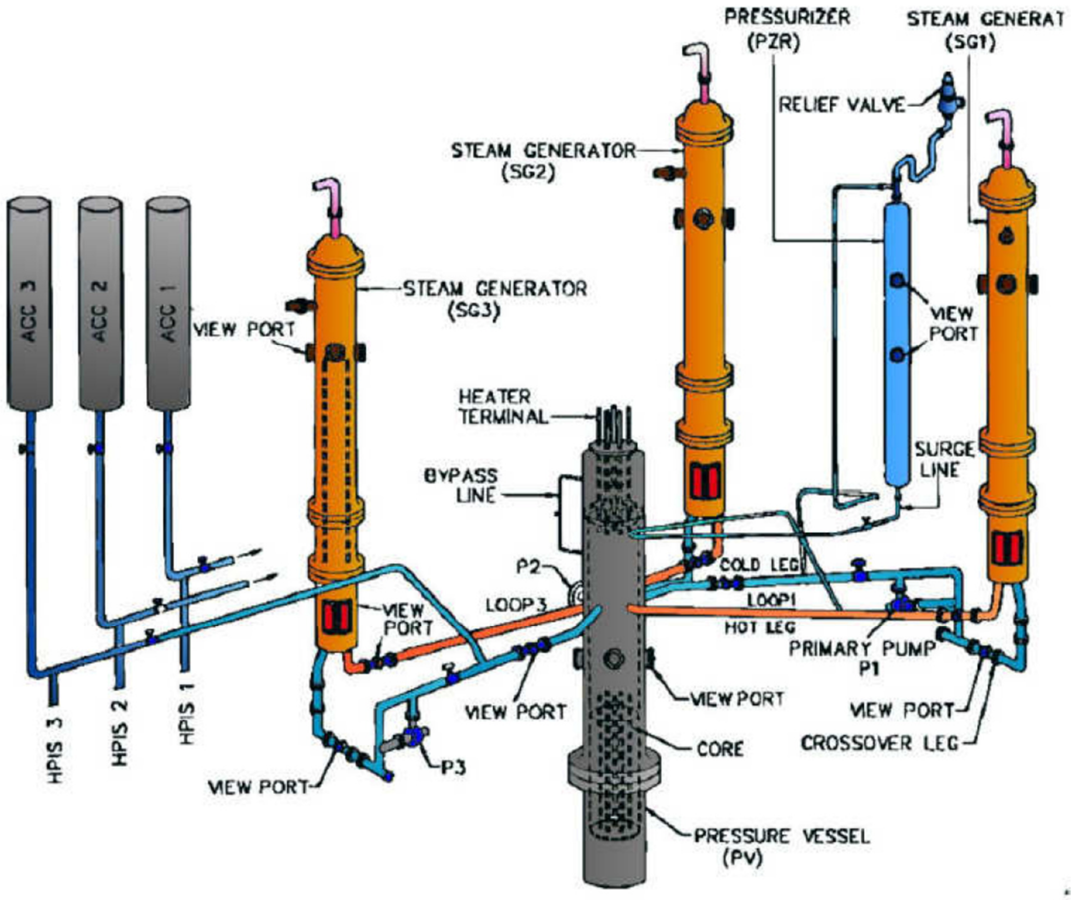

Figure 2 The Schema of IIST Facility

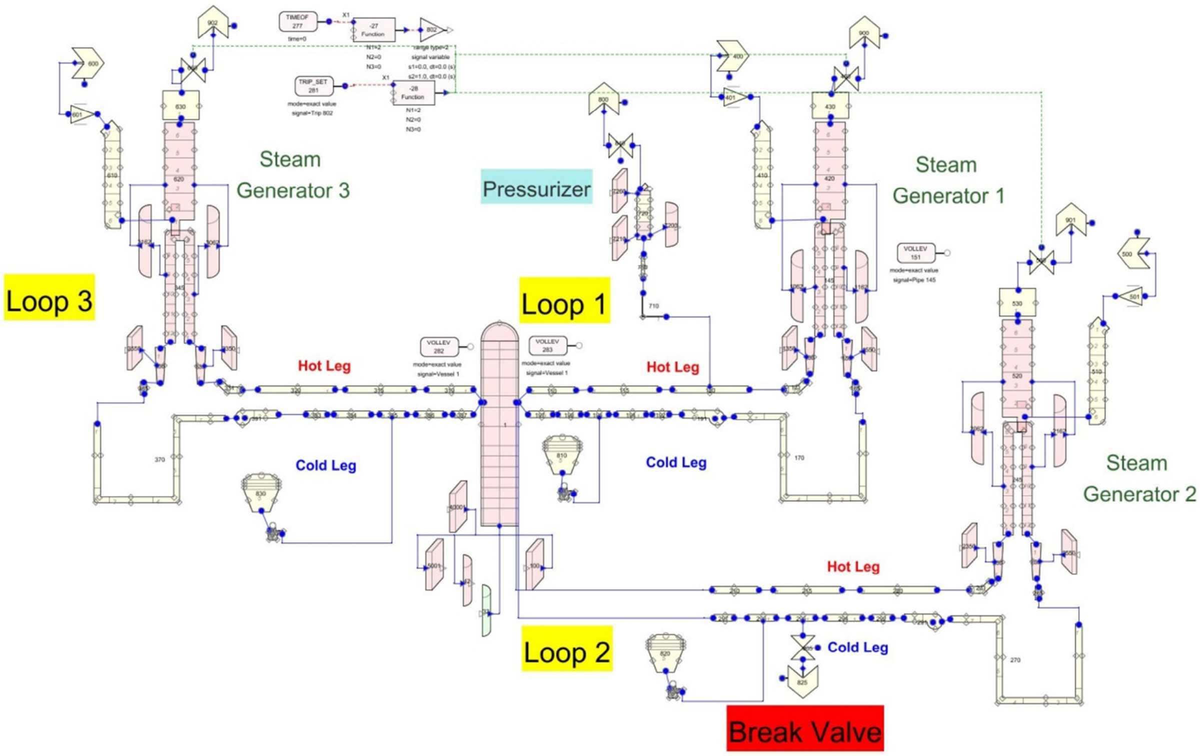

Figure 3 The TRACE IIST Facility Model

4. RESULTS AND DISCUSSIONS

4.1 Best Estimate Results

The first task of this study is to benchmark the simulation capability of IIST TRACE model by comparing with the IIST SBLOCA experiment results. The parameters including break flow rate, system pressure, core water level and peak cladding temperature, etc, which affect the plant safety directly should be taken into consideration in SBLOCA analysis. Figure 4 and Figure 5 show the comparisons of break flow rate and primary system pressure between IIST data and TRACE data. The break flow rate and primary system pressure trends of TRACE simulated results are similar with the IIST data. Figure 6 shows the comparisons of the core water level between IIST data and TRACE data. The trends of their curves are similar. Natural-circulation flow rate can be observed in the integral loop (i.e. loop-1 and loop-3) during the IIST SBLOCA experiment. Figure 7 shows the comparison of loop-3 flow rate between IIST data and TRACE data in the SBLOCA experiment. From the analysis results, there are underpredicted flow rate at 200~400s and overpredicted flow rate at 400~600s in loop-3. Figure 8 shows the fluid temperatures of the cold-leg and hot-leg in loop-3. There is no significant difference in the loop-3 fluid temperature (hot-leg and cold-leg) between the TRACE prediction and the IIST experiment data. Figure 9 shows the comparisons of cladding temperature between IIST data and TRACE data. The PCT rising time of IIST experimental data and TRACE calculated are 1782s and 1869s respectively.

In summary, the thermal-hydraulic results calculated by TRACE are in agreement with those of IIST facility experiments data.

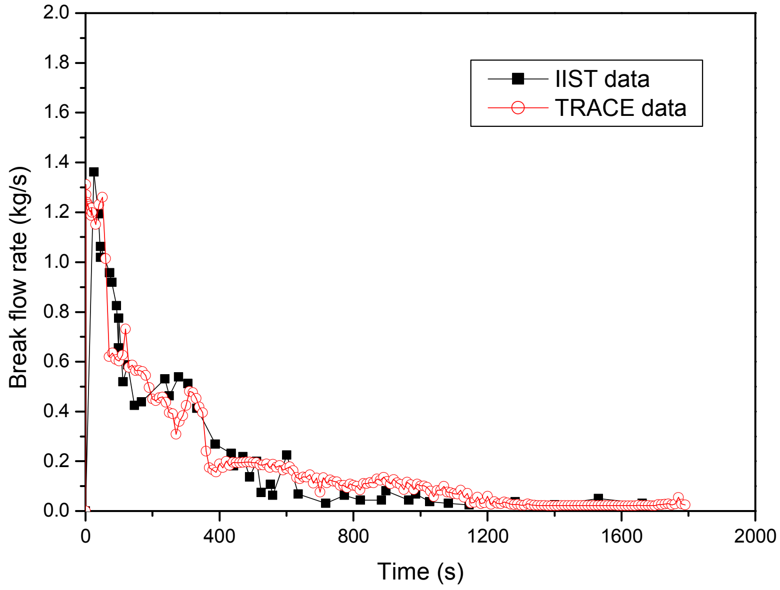

Figure 4 The Comparisons of Break Flow Rate between IIST Data and TRACE Data

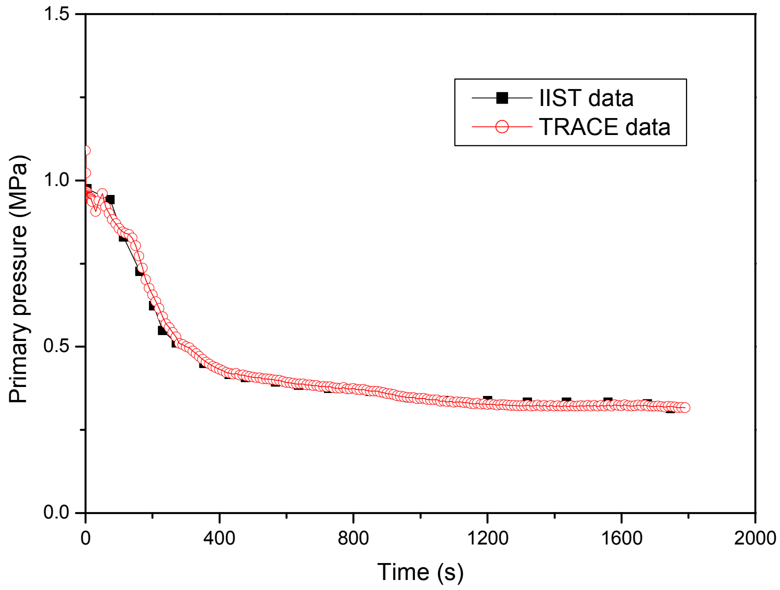

Figure 5 The Comparisons of Primary System Pressure between IIST Data and TRACE Data

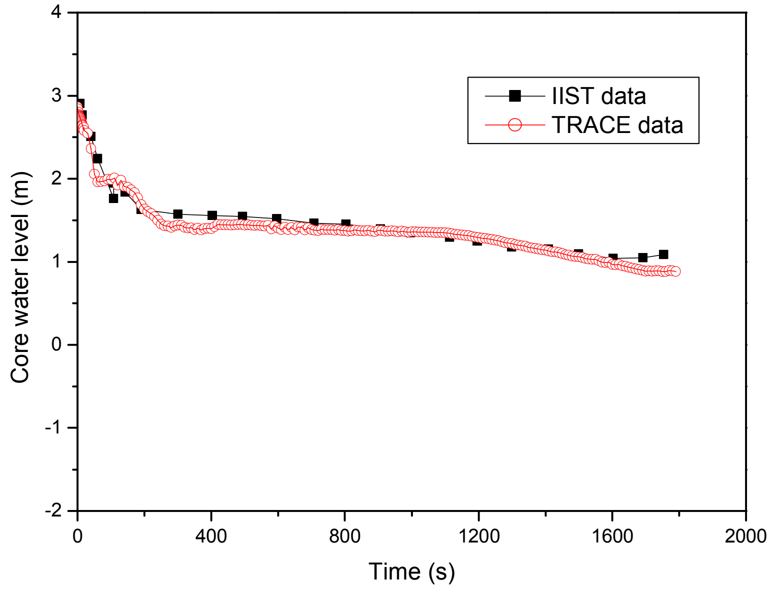

Figure 6 The Comparisons of Core Water Level between IIST Data and TRACE Data

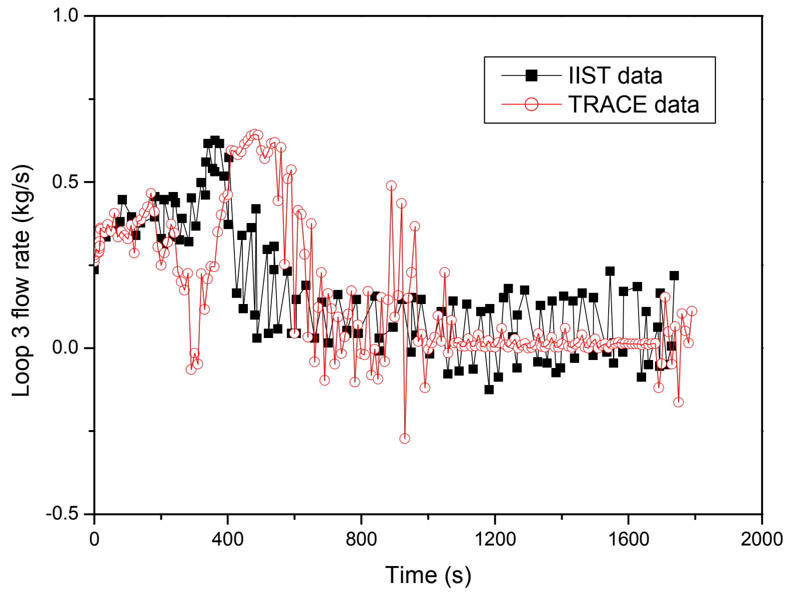

Figure 7 The Comparisons of Loop-3 Flow Rate between IIST Data and TRACE Data

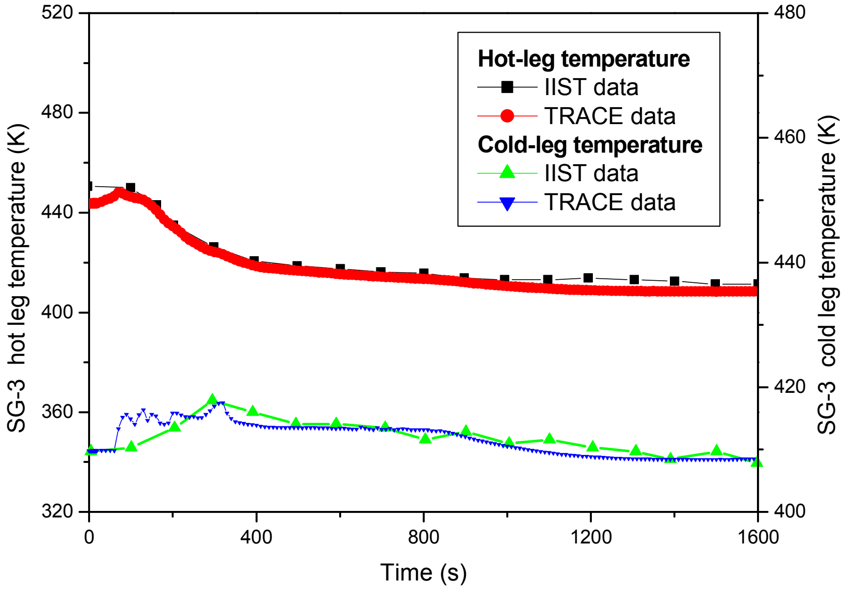

Figure 8 The Comparisons of Loop-3 Hot-Leg and Cold-Leg Temperature between IIST Data and TRACE data

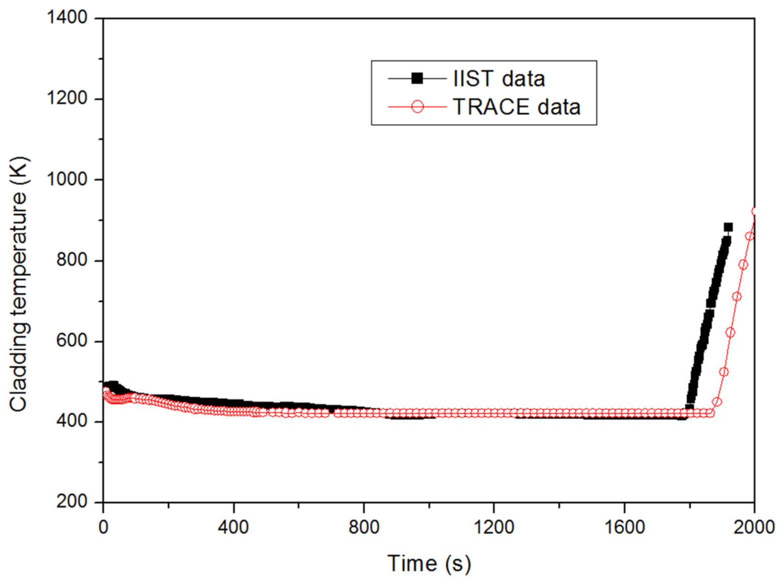

Figure 9 The Comparisons of Cladding Temperature between IIST Data and TRACE Data

4.2 Uncertainty Analysis

In BEPU of LOCA analysis, several important parameters were considered in the uncertainty quantified. The uncertainty attributes have been divided for several kinds, such as measurement uncertainty, code uncertainty, boundary condition uncertainty, model uncertainty, etc. This study is focused on the measurement uncertainty and model uncertainty analysis. Furthermore, the IIST experimental data were used to benchmark the results of uncertainty analysis. The uncertainties of system pressure and coolant average temperature are major contributed by measurement uncertainty (i.e., Venturi tube, thermocouple, etc.). However, the model uncertainty is dependent on the coefficient of model by user's definition. Table 4 lists the 5 parameters (i.e., primary pressure, core outlet temperature, loop flow rate, choked flow coefficient, and Counter-Current Flow Limitation (CCFL) coefficient) taken into account in this uncertainty analysis, which are defined as the SNAP user-defined numeric variables and linked with uncertainty configuration to generate TRACE input files. By coupling with DAKOTA, the important parameters with uncertainties were generated randomly based on specified PDFs. In particular, the statistical theory predicts that 59 calculations are required to simultaneously bound the 95th percentile of one parameters (PCT) with a 95-percent confidence level.

In LOCA simulation, the coefficient of choked flow model will influence the calculated results of critical flow rate at the break pipe. Except considering the measurement error, the model uncertainty was taken into account in this study. Figure 10 displays the 59 break flow rates as a function of time. The benchmark results show the IIST experimental data were located in the uncertainty band.

Figure 11 displays the 59 PCTs as a function of time. In BEPU analysis, an uncertainty band is formed by the 59 calculations PCTs. The faster PCT rises, the more serious condition in transient happens. The minimum and maximum PCT rising time are 1642s and 2237s respectively. The IIST experimental data shows the PCT rising time is 1782s.

Table 4 The Important Parameters for the Uncertainty Analysis

Input parameters	Nominal values	Uncertainty range	PDFs
System pressure	1.0293 (MPa)	[-2%, +2%]	ND
Hot-leg temperature	440.4 (K)	[-0.3%, +0.3%]	ND
Loop flow rate	0.217 (kg/s)	[-1.3, +1.3]	ND
Choked flow coefficient	1.0 (--)	[0.4, 1.2]	UD
CCFL coefficient	0.4 (--)	[0.2, 0.8]	UD

Note: ND means normal distributions and UD means uniform distributions

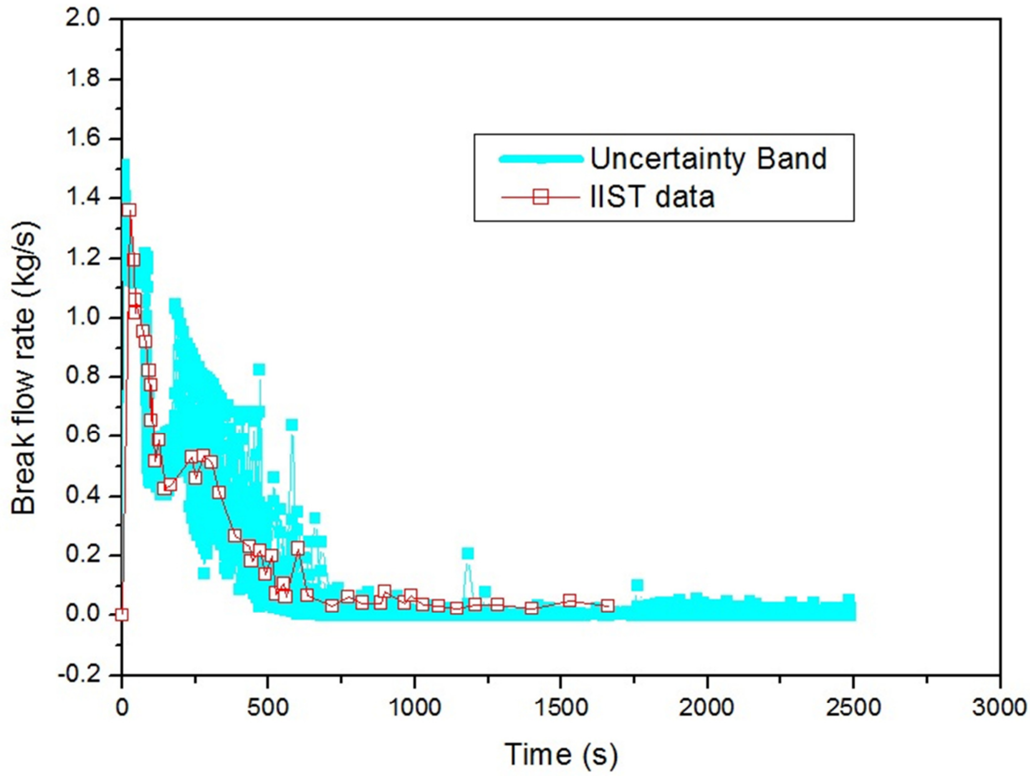

Figure 10 The Break Flow Rates during the SBLOCA

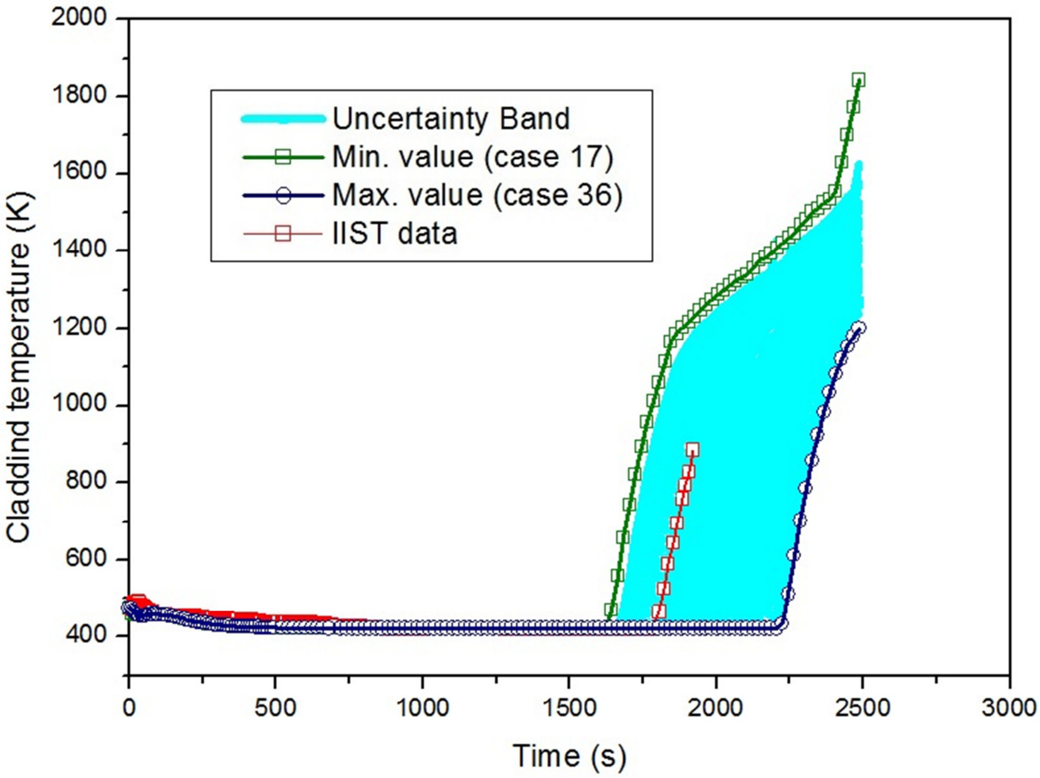

Figure 11 The PCTs during the SBLOCA

4.3 Correlation Analysis

Correlations are always calculated between two sets of sample data. One can calculate correlation coefficients between two input variables, between an input and an output variable, or between two output variables. Therefore, correlations between input parameters and PCTs (output parameters) are calculated for sensitivity study and ranking to investigate what input parameters dominate the contribution of uncertain distribution of PCT. In this study, there are two methods to present the correlations between input variables and output variables, Pearson's correlation coefficient and Spearman's rank correlation. The Pearson's correlation is shown in Eq. 3.

$$r = \frac{\sum_{i=1}^{n}(x_i-\bar{x})(y_i-\bar{y})}{\sqrt{\sum_{i=1}^{n}(x_i-\bar{x})^2 \sum_{i=1}^{n}(y_i-\bar{y})^2}} \qquad \text{Eq. 3}$$

where r is the Pearson's correlation coefficient, n is the number of samples, and x and y denote two quantities.

The formula of Spearman's rank correlation is the same as Pearson's; however, the difference is that Spearman's rank correlation employs the rank data which substitute the ranked values for raw data.

Figure 12 shows the distribution histograms of the five input parameters and 59 resultant PCTs. The parameters of primary pressure, core outlet temperature, and loop flow rate are uniform distribution, but choked flow model and CCFL coefficient are normal distribution. In addition, it could see that the PCT distribution is similar to normal distribution.

Figure 13 (a) shows the calculated results of Pearson's correlation coefficient, which include the correlation of each two parameters. The coefficients represent negative correlation. It means the small value of parameter, the quickly time of temperature raised. The correlation between input parameters and PCT are shown in Figure 13(b), and the results indicate that choked flow coefficient is the most sensitive parameter. The coefficient which user-defined has more influence on the PCT calculated results. Although the measurement error may influence the PCT calculated, its influence is smaller than the model uncertainty.

Figure 14 shows the results of Spearman's correlation coefficient. The calculated results are similar to the Pearson's correlation coefficient. The results also indicate that choked flow coefficient is the most sensitive parameter. Pearson's correlation coefficient and Spearman's correlation coefficient are -0.72405 and -0.70199 respectively.

By the Pearson's coefficient and Spearman's coefficient, it quantifies the correlations between the input and output parameter. Even though the Spearman's rank correlation employs the rank data is difference than the Pearson's correlation, their results are similar trend.

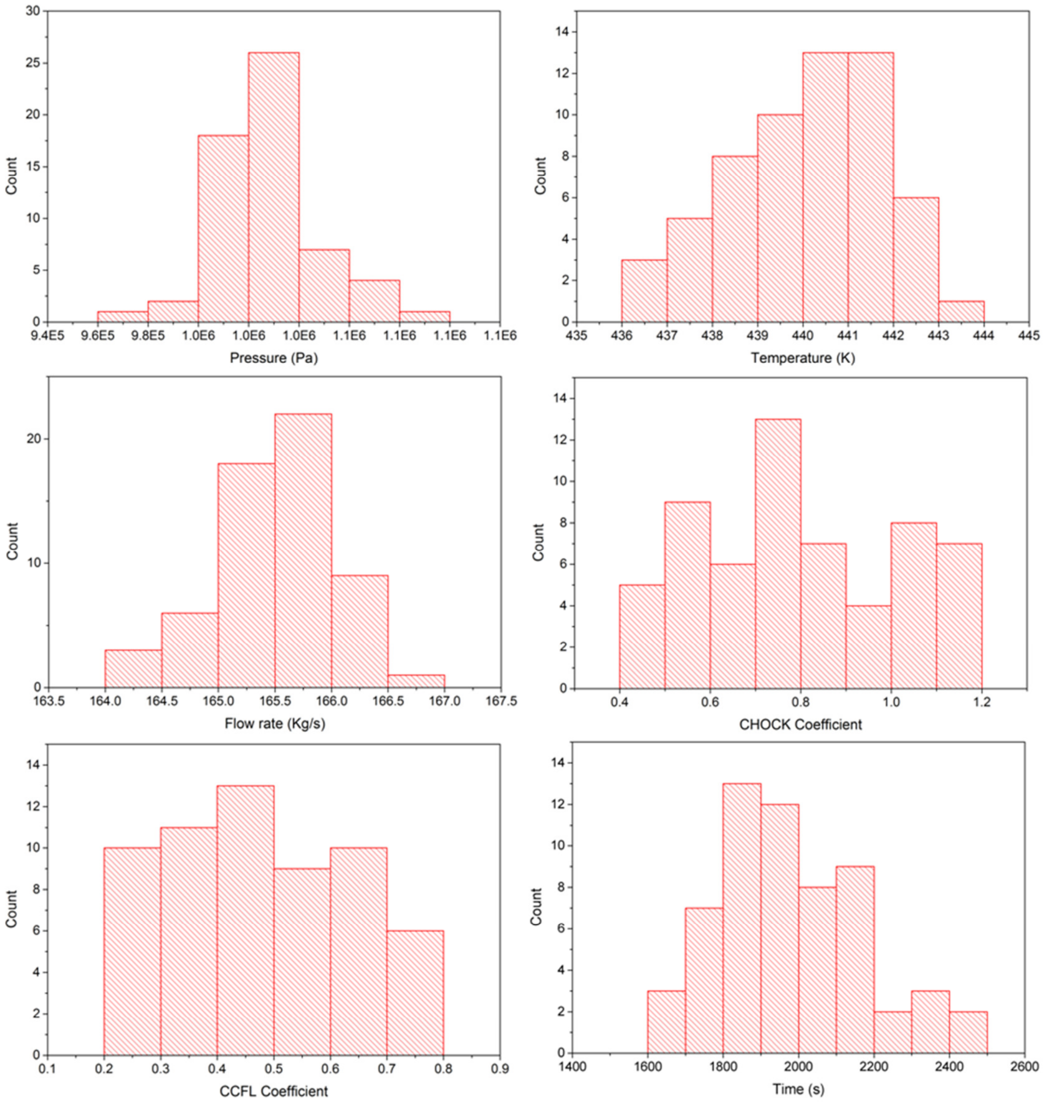

Figure 12 The Histograms of the Input Parameters and Resultant PCTs

Pearson Correlations

		Primary Pressure	Outlet Temperature	Flow rate	Chock Coef.	CCFL Coef.	PCT
"Primary Pressure"	Pearson Corr.	1	-0.03771	0.02841	0.01978	0.0277	-0.12322
	Sig.	--	0.77675	0.83089	0.88177	0.83502	0.35249
"Outlet Temperature"	Pearson Corr.	-0.03771	1	0.01934	0.02321	-0.00527	-0.07711
	Sig.	0.77675	--	0.88442	0.86149	0.96837	0.56157
"Flow rate"	Pearson Corr.	0.02841	0.01934	1	0.00813	0.01269	-0.06131
	Sig.	0.83089	0.88442	--	0.95128	0.92399	0.64461
"Chock Coef."	Pearson Corr.	0.01978	0.02321	0.00813	1	0.0206	-0.72405*
	Sig.	0.88177	0.86149	0.95128	--	0.87691	9.14255E-11
"CCFL Coef."	Pearson Corr.	0.0277	-0.00527	0.01269	0.0206	1	-0.03725
	Sig.	0.83502	0.96837	0.92399	0.87691	--	0.7794
"PCT"	Pearson Corr.	-0.12322	-0.07711	-0.06131	-0.72405*	-0.03725	1
	Sig.	0.35249	0.56157	0.64461	9.14255E-11	0.7794	--

*:Correlation is significant at the 0.05 level

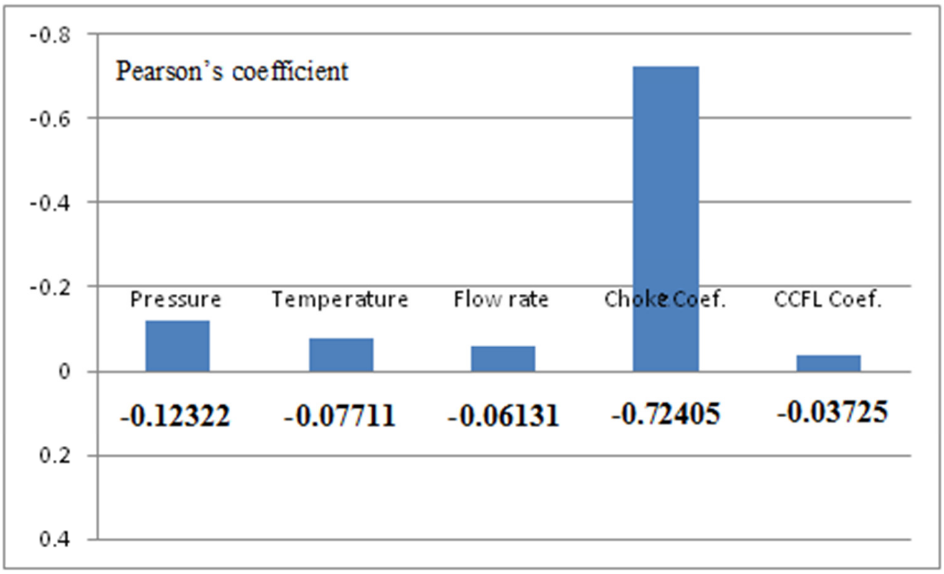

Figure 13 The Pearson's Correlation Coefficients between Input Parameters and PCT

Spearman Correlations

		Primary Pressure	Outlet Temperature	Flow rate	Chock Coef.	CCFL Coef.	PCT
"Primary Pressure"	Spearman Corr.	1	0.05371	8.7668E-4	0.00579	0.00719	-0.09351
	Sig.	--	0.68619	0.99474	0.96531	0.95691	0.48115
"Outlet Temperature"	Spearman Corr.	0.05371	1	0.01841	-0.01426	-0.01788	-0.05558
	Sig.	0.68619	--	0.88993	0.91463	0.89305	0.67586
"Flow rate"	Spearman Corr.	8.7668E-4	0.01841	1	-0.00994	-0.02624	-0.05032
	Sig.	0.99474	0.88993	--	0.94046	0.8436	0.70506
"Chock Coef."	Spearman Corr.	0.00579	-0.01426	-0.00994	1	-0.00427	-0.70199*
	Sig.	0.96531	0.91463	0.94046	--	0.97442	5.84034E-10
"CCFL Coef."	Spearman Corr.	0.00719	-0.01788	-0.02624	-0.00427	1	-0.10094
	Sig.	0.95691	0.89305	0.8436	0.97442	--	0.44686
"PCT"	Spearman Corr.	-0.09351	-0.05558	-0.05032	-0.70199*	-0.10094	1
	Sig.	0.48115	0.67586	0.70506	5.84034E-10	0.44686	--

*:Correlation is significant at the 0.05 level

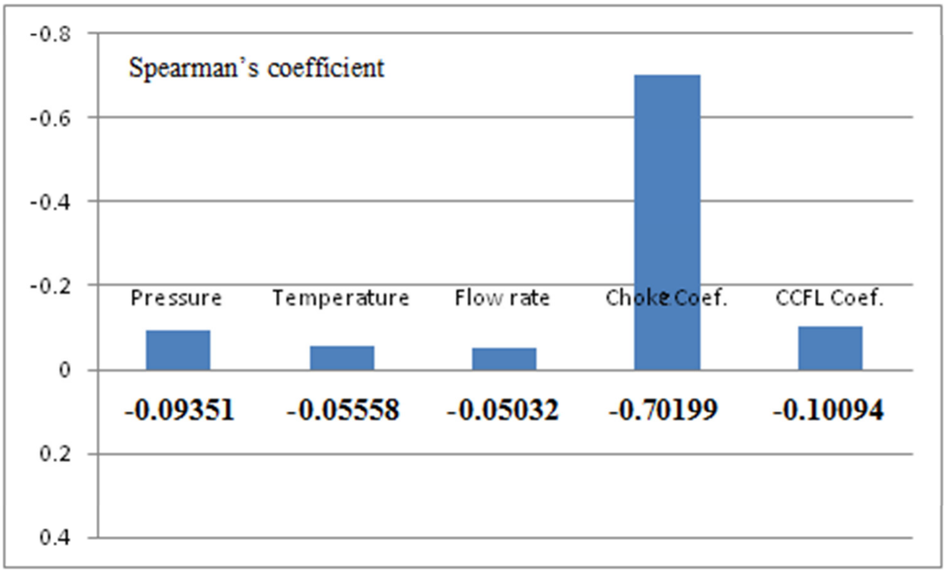

Figure 14 The Spearman's Correlation Coefficients between Input Parameters and PCT

5. CONCLUSIONS

There are two tasks in this study. One is benchmark of the simulation capability of IIST TRACE model by comparing with the IIST SBLOCA experiment results. The parameters including break flow rate, system pressure, core water level and peak cladding temperature, etc, which affect the plant safety directly should be taken into consideration in SBLOCA analysis. In summary, the thermal-hydraulic results calculated by TRACE are in agreement with those of IIST facility experiments data.

The second task is BEPU in SBLOCA analysis, several important parameters were considered in the uncertainty quantified. In this study, the BEPU is focused on the discussion of measurement uncertainty and model uncertainty analysis. Furthermore, the IIST experimental data were used to benchmark the results of uncertainty analysis. There are 5 parameters taken into account in this uncertainty analysis. An uncertainty band is formed by the 59 calculations, and the benchmark results show the IIST experimental data were located in the uncertainty band. The minimum and maximum PCT rising time are 1642s and 2237s respectively. The IIST experimental data shows the PCT rising time is 1782s.

In addition, this study used two methods to calculate correlation coefficients between an input and an output variable: Pearson’s correlation coefficient and Spearman’s rank correlation. The formula quantifies the correlations between the input and output parameter. Even though the Spearman’s rank correlation employs the rank data is difference than the Pearson’s correlation, their results are similar trend. The results indicate that choked flow coefficient is the most sensitive parameter. The correlation coefficients of Pearson’s and Spearman’s are -0.72405 and -0.70199 respectively. Although the measurement error may influence the PCT calculated, its effect is smaller than the model uncertainty.

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11. ABSTRACT (200 words or less)

This study is developed the BEPU methodology and the uncertainty results were compared with the IIST experiment data. The IIST TRACE model consists of 89 hydraulic components, 243 control blocks, 39 heat structures and a power component. The data interactions and communications between TRACE and DAKOTA were controlled by SNAP. Finally, correlations between input parameters and output data are calculated for sensitivity study and ranking to investigate what input parameters dominate the contribution of uncertain distribution of PCT. There are two tasks in this study. One is benchmark of the simulation capability of IIST TRACE model by comparing with the IIST SBLOCA experiment results. The other is BEPU in SBLOCA analysis, several important parameters were considered in the uncertainty quantified. The BEPU analysis is focused on the discussion of measurement uncertainty and model uncertainty. Furthermore, the IIST experimental data were used to benchmark the results of uncertainty analysis. There are 5 parameters taken into account in this uncertainty analysis. An uncertainty band is formed by the 59 calculations, and the benchmark results show the IIST experimental data were located in the uncertainty band. In addition, this study used two methods to calculate correlation coefficients between an input and an output variable: Pearson's correlation coefficient and Spearman's rank correlation. The formula quantifies the correlations between the input and output parameter. Even though the Spearman's rank correlation employs the rank data is difference than the Pearson's correlation, their results are similar trend. The results indicate that choked flow coefficient is the most sensitive parameter. The correlation coefficients of Pearson's and Spearman's are -0.72405 and -0.70199 respectively. Although the measurement error may influence the PCT calculated, its effect is smaller.

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