Best Estimate Plus Uncertainty Analysis of Small-Break Loss-of-Coolant Accidents for IRIS

Changzheng Yin[a*]
[a]University of Science and Technology of China, Hefei, China

Abstract: In recent years, the small modular pressurized water reactor (PWR) has received much attention. International Reactor Innovative and Secure (IRIS) is a small modular PWR, different from conventional PWR in structure and safety characteristics. Regarding structure, the integrated design of the IRIS eliminates large-break loss-of-coolant accidents (LBLOCA). Regarding safety characteristics, the IRIS reactor adopts passive safety facilities to mitigate accidents. This paper uses the reactor safety analysis code RELAP5 and the data analysis platform RAVEN to establish the best estimate plus uncertainty (BEPU) safety analysis framework. In the BEPU framework, the RELAP5 model of the IRIS is used to calculate the reactor's small-break loss-of-coolant accident (SBLOCA) to study the safety characteristics of the IRIS under the accident scenario. The breaks considered in this paper occur in the direct vessel injection (DVI) pipeline and the chemical and volume control system (CVCS) pipeline, respectively. The results of the accident calculation show that the RELAP5 model can reasonably predict accident scenarios and verify that the IRIS has good passive safety characteristics under different accident processes caused by different break locations. Then, the uncertainty quantification of SBLOCAs of the DVI pipeline is studied based on the best estimate. Some essential input parameters of uncertainty are selected to perform uncertainty propagation analysis, and the uncertainty bands of figures of merit are obtained and analyzed. Finally, the contribution weights of different input parameters for the uncertainty of figures of merit are determined by sensitivity analysis.

Keywords: IRIS, SBLOCA, BEPU, Uncertainty Analysis.

1. INTRODUCTION AND MODELING

1.1. Introduction of IRIS

Many safety studies have been conducted for large pressurized water reactors (PWRs). However, because the unique design of small modular reactors (SMRs) differs from that of large PWRs, the process and phenomena in the accident process are different, and the corresponding research needs to be more mature. In order to verify the safety characteristics of small modular reactors, a design basis accident simulation is necessary. The IRIS (International Reactor Innovative and Secure) studied in this paper is a small modular PWR [1]. Since the end of the last century and the beginning of this century, the project has been led by Westinghouse and attracted more than 20 institutions from about 10 countries to participate. The IRIS is designed to be holistic, with almost all primary components, such as the core, reactor coolant pumps (RCPs), steam generators (SGs), and pressurizers, integrated into the pressure vessel. Due to the integrated structure, many components have been redesigned. In addition to the pressure vessel, there are other passive safety systems in the containment vessel, such as emergency boration tanks (EBTs), emergency heat removal systems (EHRSs), and pressure suppression systems (PSSs).

The safety characteristics of the IRIS can be summarized in two aspects: 1) The passive safety characteristics inherited from AP1000 and 2) Inherent safety features generated by design. Inherent safety features can eliminate some accidents by design. For example, since the primary components are integrated into the pressure vessel, the coolant flows only in the pressure vessel, eliminating large-break loss-of-coolant accidents (LBLOCAs). For accidents that have already occurred, design safety features and passive safety facilities together ease the accident process and reduce the consequences of the accident. In this paper, a study on SBLOCAs is carried out to verify that IRIS meets the safety requirements.

1.2. Introduction of BEPU Method

The research on nuclear power plant accidents mainly adopts conservative accident analyses presently. That is, conservative research models are adopted, conservative assumptions are made, and conservative limits are obtained. The predicted results differ significantly from the actual values. Finally, the operation of nuclear power plants is too conservative, and the economy and flexibility of nuclear power plants are sacrificed in exchange for safety. With a more in-depth understanding of reactor operation conditions and various types of operation accidents and the development of various reactor transient analysis programs, the best estimate plus uncertainty (BEPU) analysis method has been developed. According to the IAEA definition, the best estimate analysis should meet the following requirements: No intentional conservatism is introduced into the selected acceptance criteria; Use the best estimate procedure; Includes uncertainty analysis [2]. BEPU method uses unique thermohydraulic safety analysis code to simulate the reactor accident process in detail, and accurately, that is, the ‘best estimate’, and strives to get close to the actual accident situation and get the simulated value close to the actual situation. At the same time, the uncertainty analysis can define the gap between the simulation results and the actual value. The uncertainty mainly comes from the uncertainty of the program itself and the uncertainty of the input parameters of the accident operation. This paper mainly studies the uncertainty of the input parameters.

Compared with the conservative method, the results obtained by the BEPU method are closer to the actual accident process. It can be seen from Figure 1 that the BEPU method can obtain a more considerable safety margin than the conservative method. If the same safety margin is maintained, the BEPU method can reduce the economic and flexibility sacrifice caused by overly conservative assumptions.

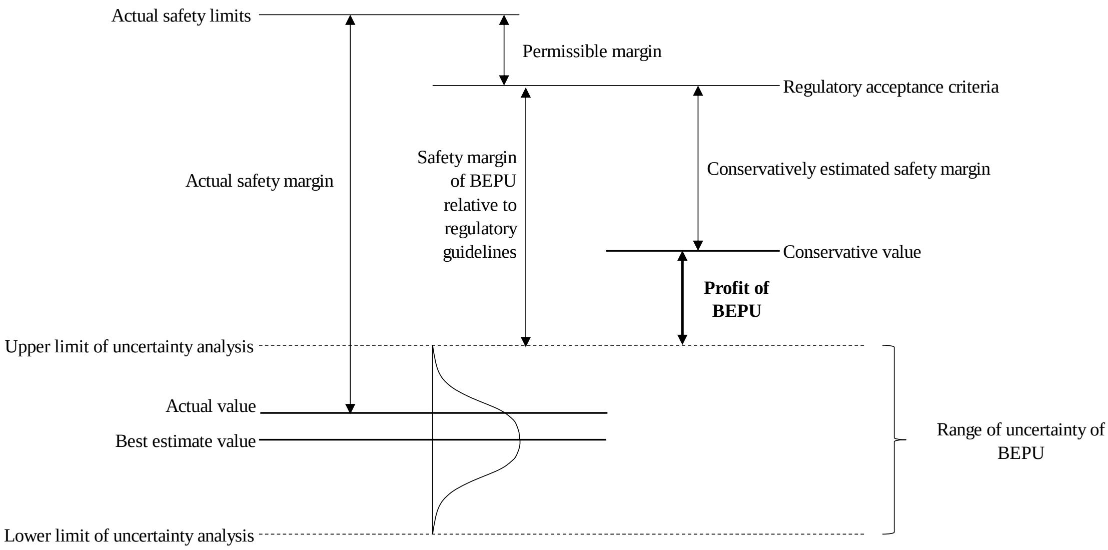

Figure 1. Comparison of Safety Margin between BEPU Method and Conservative Method [3]

The best estimate code used in this paper is the reactor thermal-hydraulic and safety analysis program RELAP5/MOD3.3[4], and the uncertainty analysis software is RAVEN, a data analysis platform developed by Idaho National Laboratory [5]. The specific process of the BEPU framework is as follows: Firstly, RAVEN is used to sample the input uncertainty parameters, then the sampled parameters are input to RELAP5 for accident simulation calculation, and finally, the output data is analyzed. The specific process is shown in Figure 2.

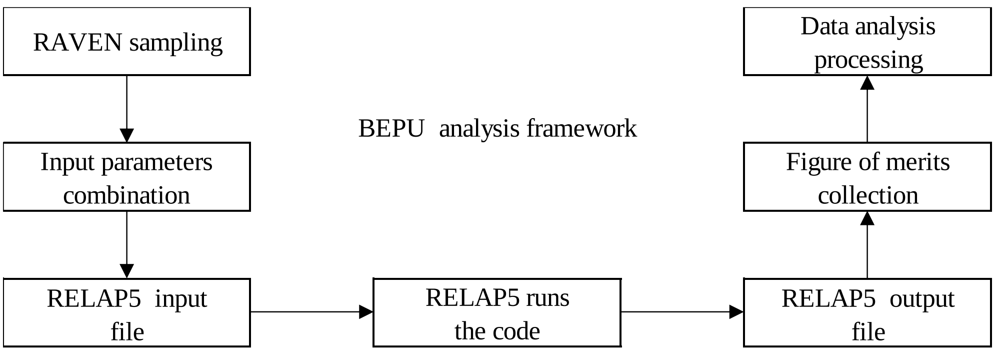

Figure 2. BEPU Analysis Framework Based on RELAP5 and RAVEN

1.3. RELAP5 nodalization of the IRIS

This research establishes a RELAP5 nodalization model of the IRIS, in which the reactor core, pressurizer, SGs, RCPs, EBTs, main feed pipes, main steam pipes, and EHRSs are all modeled. The modeling is shown in Figure 3. Some of the modeling is simplified due to the limitations of the RELAP5 model. For example, the hemispherical part of the pressurizer is simulated by stacked cylinders, and the helical coil steam generators are simulated by inclined straight tubes. Figure 3 is a simplified diagram where only 1 RCP, 1 SG, and 1 EHRS are drawn. In fact, 8 RCPs, 8 SGs, and 4 EHRSs are all modeled.

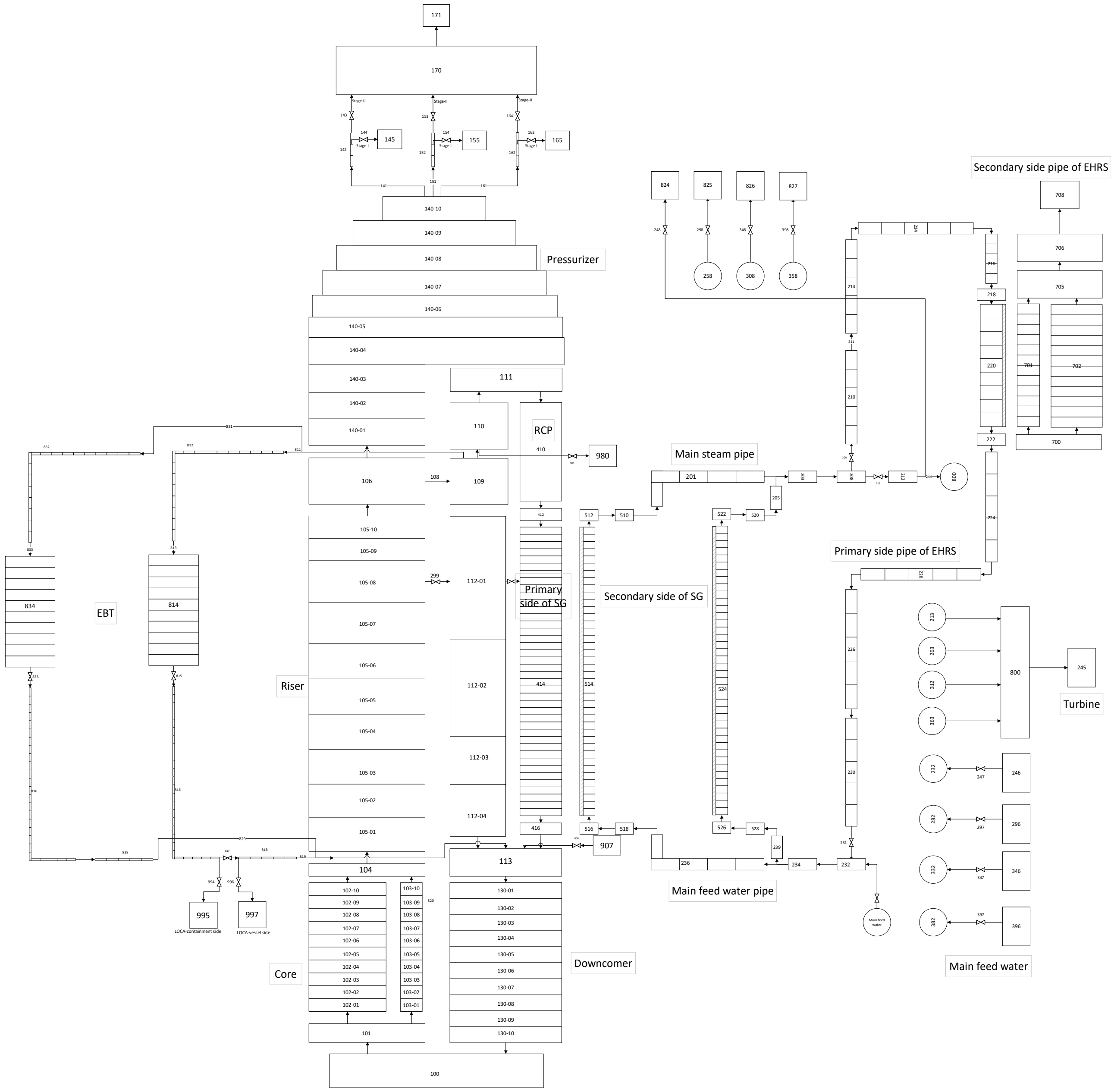

Figure 3. RELAP5 Nodalization of the IRIS

2. BEST ESTIMATE ACCIDENT ANALYSIS

2.1. Steady-State Operating Parameter

Before the accident analysis, it is necessary to ensure that the reactor is in a steady state before the accident. Therefore, before the subsequent transient analysis, the RELAP5 model is verified by steady-state regulation to ensure that the initial state of the built model meets the requirements of the starting point of the accident analysis. The steady-state operating parameters calculated in this paper are compared with those in the literature [6], as shown in Table 1, indicating that the steady-state values of the model built meet the requirements.

Table 1. Steady-State Operating Parameters

Parameter	Unit	Reference Value	Calculated Value
Pressurizer pressure	MPa	15.5	15.5
Core flow	kg/s	4504	4506.0
Vessel flow	kg/s	4707	4707.7
Core inlet temperature	K	565.2	565.13
Core outlet temperature	K	601.5	601.27
SG pressure	MPa	5.8	5.79
Steam exit temperature	K	590.2	591.38
Total steam flow	kg/s	502.8	502.82
Core pressure drop	kPa	52.0	62.1
SG pressure drop primary/secondary	kPa	72.0/296	64.8/272
RCP head	m	19.1(18.3-21.3)	19.7

2.2. Accident Process

Due to its integrated design, the IRIS has no large pipes in the primary circuit, so it eliminates LBLOCAs. In the case of the SBLOCAs, the IRIS mitigates the accident process by limiting the loss of coolant [7], which is inherently safe by design, in contrast to conventional PWR measures of continuously injecting water into the pressure vessel. After the occurrence of SBLOCA, the reactor goes through roughly three stages [7], which are as follows:

1) Blowdown phase. After the accident, the coolant is released from the break and enters the containment vessel. The pressure in the pressure vessel decreases. Then, the reactor shuts down, and the main feedwater valve and the main steam valve of the secondary circuit are closed. Then, the RCP shuts down, and the primary circuit enters the natural circulation state. After the pressure drops further, EBTs are used, and borated water is injected into the core. EHRSs are used to help carry away core heat.

2) RV/CV depressurization phase. In this phase, the break flow rate is reduced rapidly. EHRSs continuously remove heat from the reactor pressure vessel and depressurize. Automatic Depressurization System (ADS) is also used to accelerate depressurization. In this paper, ADS is not used. EHRS alone has achieved an excellent depressurization effect.

3) Long-term cooling phase. At this stage, the long-term gravity makeup system (LGMS) operates to inject cooling water into the core. Unlike EBT, LGMS provides a slight cooling water flow and a long injection duration, providing long-term cooling against core decay heat. This phase is not involved in the simulation of the accident process in this study, so the nodalization model of the IRIS omits the LGMS.

2.3. Calculation Results Analysis

In order to study the influence of different break locations on the accident process, the SBLOCA with breaks in the direct vessel injection (DVI) pipeline and the chemical and volume control system (CVCS) pipeline are selected for simulation. The DVI pipeline is located in the area between SG and the core. It is a pipeline where borated water is injected into the core. Its diameter is 0.0508m, and the cross-sectional area is about 0.002 m^2. The CVCS pipeline, located above the RCPs and SGs, is the largest pipe in the primary circuit, twice the diameter of the DVI pipeline. For the two pipelines, the same size of the break, that is, half of the cross-sectional area of the DVI pipeline (0.001 m^2), is selected to simulate the accident.

This paper simulates the accident from the beginning of the break to 3000 seconds after the break, including the blowdown and depressurization phases. This paper refers to the reactor trips and safeguards actuation signals and delays given by the IRIS Preliminary Safety Assessment by Westinghouse [7]. The final SBLOCA accident process is shown in Table 2. Figure 4 to Figure 9 shows the changes in some critical parameters during the accident process.

Table 2. SBLOCA Accident Process

Trip Description		Setpoint	Trip Time Delay	Trip Time in Calculation	
				DVI Case	**CVCS Case**
Reactor trip		Pressurizer pressure < 13.79MPa	2	118	182
Main feedwater valve closure		Pressurizer pressure < 13.79MPa	2	118	182
Main steam valve closure		Pressurizer pressure < 13.79MPa	2	118	182
RCP trip		Pressurizer level < 1.299m	15	129	193
EBT put into use		Pressurizer pressure < 12.41MPa	2	241	331
EHRS put into use	2 Loops	Steam line pressure>9MPa	2	124	188
	4 Loops	Pressurizer pressure < 12.41MPa	2	241	331

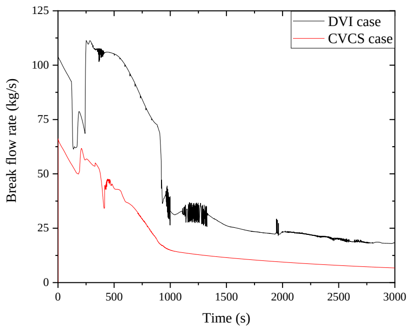

Figure 4. Break Flow Rate

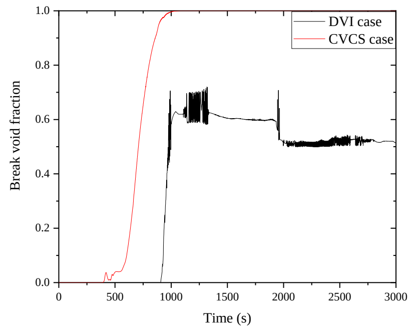

Figure 5. Break Void Fraction

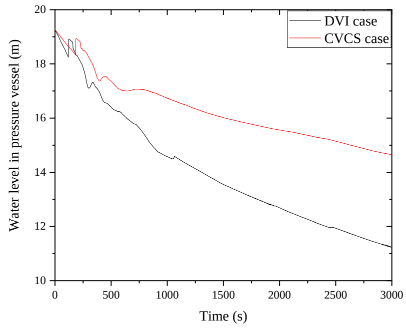

Figure 6. Water Level in Pressure Vessel

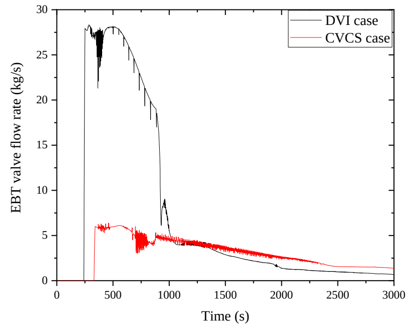

Figure 7. EBT Valve Flow Rate

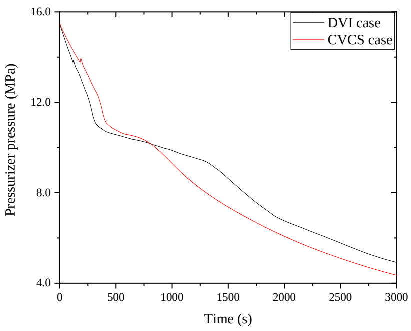

Figure 8. Pressurizer Pressure

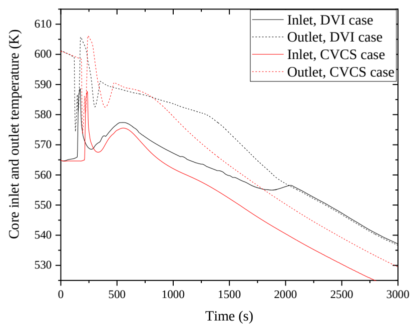

Figure 9. Core Inlet and Outlet Temperature

It can be seen that the accident process of the CVCS case is slower than that of the DVI case, which indicates that the location of the break will have a decisive impact on the accident process [8]. The processes of SBLOCA are as follows.

1) After the break occurs, the coolant is discharged from the break into the containment, which causes the core coolant flow rate, pressure, and water level to decrease rapidly. The back pressure of the containment increases, resulting in a slow decline in the break flow rate. In the initial case, the break position of the

CVCS case is higher than that of the DVI case, and the pressure at the break is lower, so the break flow rate is significantly lower, making the core pressure drop more slowly for the CVCS case.

2) The core pressure drops to the critical value, triggering an emergency shutdown. The main feedwater valves, steam valves, and RCPs are closed in succession. The reactor power rapidly decreases to the decay heat level, and the core flow rate decreases rapidly, leading to a rapid decrease in break flow. As the core power decreases, the outlet temperature decreases rapidly to approach the inlet temperature. Then, the outlet temperature rises rapidly due to the continuous release of decay heat from the core. In the secondary circuit, the steam pipe pressure rises rapidly, putting EHRSs into use and taking away heat from SGs through natural circulation. After some time, the excess heat in the primary circuit is discharged, the core inlet and outlet temperature drop again, and the break flow rate drops.

3) As the core pressure drops further, the EBTs are used, and borated water is injected into the pressure vessel. For the DVI case, the break occurs in the pipeline where water is injected into the core, resulting in a rapid increase in the break flow rate again. For the CVCS case, the increase of break flow caused by EBT operation is not as severe as that in the DVI case. At the same time as the EBTs are used, the remaining two EHRSs are in service to continue to help with core cooling and pressure relief.

4) The break discharge state changes when the system pressure drops to the critical coolant temperature. In the DVI case, the discharge state changes from subcooled to two-phase discharge while the void fraction at break increases from 0 to about 0.6. The break and EBT flow rate rapidly decreases. In the CVCS case, due to the high position of the break, the discharge state changes rapidly from subcooled to superheated discharge. Pure steam, rather than the mixture of steam and water in the DVI case, is discharged through the break, so the pressure and temperature in the pressure vessel drop faster in the CVCS case than in the DVI case in this stage.

5) In the DVI case, the inlet and outlet temperatures become nearly identical about 2000 seconds after the break. In the CVCS case, the difference between inlet and outlet temperature decreases continuously. Finally, the reactor’s temperature and pressure continue to decline slowly, the primary flow, break flow, and EBT flow are maintained at a low level, and the accident will be transferred to the long-term cooling phase.

US Nuclear Regulatory Commission (USNRC) denoted that the IRIS was designed to comply with all applicable USNRC criteria [9]. The peak cladding temperature (PCT), an essential parameter for evaluating safety measures, cannot exceed 1477.6K in the LOCA accident guidelines formulated by USNRC [10]. This criterion has been applied by Oriani et al. [11] to the SBLOCA analysis of the IRIS. PCT in the two cases is shown in Figure 10. After the accident, the variation trend of PCT is similar to that of core inlet and outlet temperature. In the whole process of the accident, the PCT does not exceed the PCT of the steady-state operation stage before the accident, and the maximum PCT is only about 625K, which is far below the safety limit, indicating that the IRIS has a high safety margin under the LOCA of break at different locations.

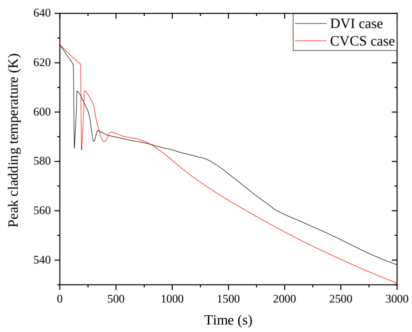

Figure 10. Peak Cladding Temperature

3. UNCERTAINTY ANALYSIS

In this chapter, uncertainty analysis is carried out based on the best estimate of the DVI pipeline SBLOCA case.

3.1. Sampling Method

There may be deviations between the best estimate and the value of the actual operation. This deviation is due to the error of the calculation model and the error between the actual operating parameters and the set value. Uncertainty analysis aims to determine the upper and lower limits of the error and ensure that the actual value falls within the error range to facilitate the setting of the corresponding regulatory limits. The object of uncertainty analysis in this paper is mainly input parameters. The uncertainty distribution of each parameter is determined and sampled according to the distribution. The data generated by sampling is input into the best estimate program to obtain the upper and lower limits of the output results.

This paper adopts Wilks's non-parametric sampling method [12,13] to sample all parameters simultaneously. The advantage of this method is that the sampling number is not necessarily related to the number of parameters selected but is related to the required confidence level and tolerance limit. The method establishes the relationship between confidence level, tolerance limit, and sampling number. Suppose that the probability distribution of a figure of merit x is f(x), given the upper and lower limits L and U of a tolerance interval, then the Formula (1) states that the confidence that x has at least a share of γ falling between the interval (L, U) is β. For the two-sided tolerance limit, Wilks's Formula for the number of sampling N is as Formula (2).

$$P\left(\int_L^U f(x)dx \geq y\right) = \beta \tag{1}$$

$$\beta = 1 - \gamma^N - N(1-\gamma)\gamma^{N-1} \tag{2}$$

The statistical accuracy typically considered for safety parameters in BEPU applications is 95/95 [14], meaning γ and β are 95%. From Formula (2), N can be calculated as 93. In this paper, considering the possibility of program failure in the calculation process, the number of sampling calculations is 100.

3.2. Distributions of Input Parameters

Two considerations are made in the selection of sampling parameters: First, the selected parameters would have a significant impact on the accident process; Second, considering that it would take much time to re-calculate and determine the initial parameters to adjust the reactor operation steady state, the parameters that do not affect the core steady state after the change are selected, and more input parameters will be considered in later study.

Firstly, parameters related to the break are considered. For SBLOCAs, the break's size and discharge coefficient are closely related to the flow rate, directly affecting the subsequent accident process. Therefore, this paper selects the area of break and three discharge coefficients (subcooled, two-phase, and superheated, respectively) for uncertainty sampling. Secondly, parameters related to safety measures are considered. During the accident process, the two EBTs are very important, and the borated water they inject plays a role in supplementing the coolant and helping to cool the core. The change in temperature and pressure of borated water in EBT will affect its cooling effect and thus affect the accident process, so these two parameters are selected for uncertainty sampling. The distribution should be determined after selecting the uncertain sampling parameters. Because the uncertainty parameters used in this paper are relatively few, and the total input uncertainty is small, a more conservative uniform distribution is adopted in the distribution, and the variation range of each parameter is ±5%. Parameter distributions are listed in Table 3.

Table 3. Uncertainty Distributions of Input Parameters

Parameter	Reference Value	Distribution	Range
The area of the break	$0.001m^2$	Uniform	$(0.00095m^2, 0.00105m^2)$
Discharge coefficient (subcooled)	1.00	Uniform	(0.95,1.05)
Discharge coefficient (two-phase)	1.00	Uniform	(0.95,1.05)
Discharge coefficient (superheated)	1.00	Uniform	(0.95,1.05)
EBT initial temperature	322.0K	Uniform	(305.9K,338.1K)
EBT initial pressure	15.535MPa	Uniform	(14.758MPa,16.312MPa)

These parameters are combined by RAVEN using the Monte Carlo method for non-parametric sampling. The input parameters are randomly sampled during each sampling to form a set of sampling data, which becomes a set of inputs for RELAP5 calculation. In this paper, 100 working conditions are sampled, and each group of working conditions is independent.

3.3. Results and Analysis

Considering that the drastic changes in figures of merit (FOMs) during the accident occur within the 1500s after the break, and 100 sampling calculations would consume much time, RELAP5 is used to calculate working conditions for each group only until the 1500s after the accident. The uncertainty envelopes of some FOMs are shown in Figure 11-16.

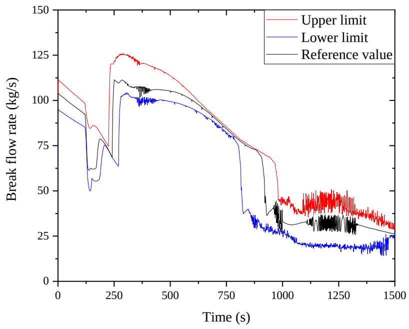

Figure 11. Break Flow Rate

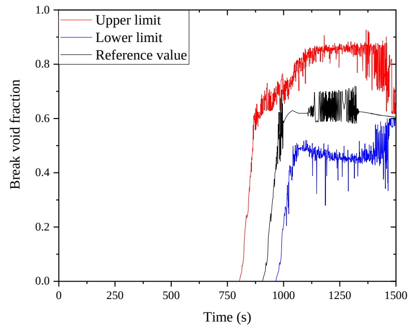

Figure 12. Break Void Fraction

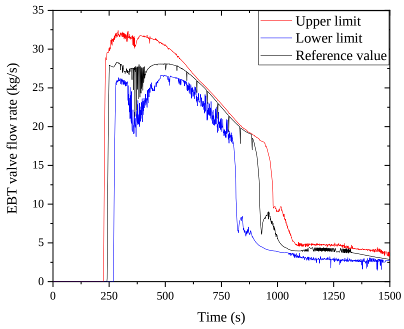

Figure 13. EBT Valve Flow Rate

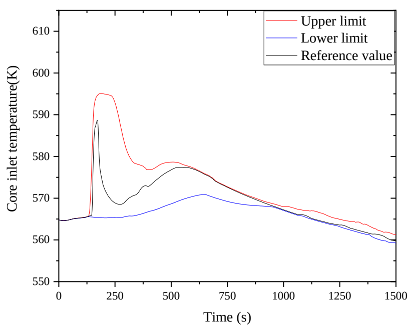

Figure 14. Core Inlet Temperature

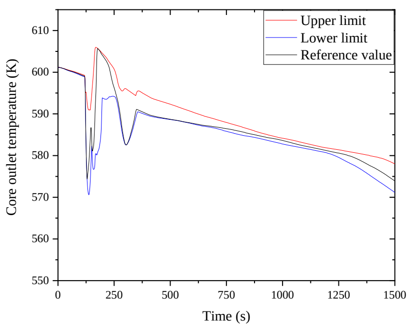

Figure 15. Core Outlet Temperature

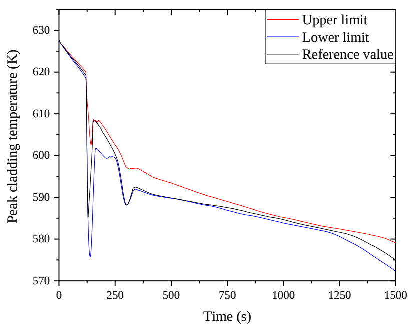

Figure 16. Peak Cladding Temperature

In these figures, the black lines represent the best estimates (reference values), while the red and blue lines represent the upper and lower limits of uncertainty analysis, respectively. After calculation by RELAP5, the uncertainty of the input parameters also forms an uncertainty distribution near the reference value of the FOMs after propagation, which is reflected as the envelope of the upper and lower limits in the figure. After uncertainty analysis, the following conclusions can be drawn:

1) The uncertainty of input parameters does not cause significant changes in the accident process. Compared with the reference condition, the variation trend of PCT under each uncertain condition is similar. During the accident process, maximum PCT occurs at the end of the steady-state operation and just after the break occurs, and the uncertainty tolerance interval is (627.5548K,627.5576K), which is far lower than the prescribed safety limit of 1477.6K, indicating that the change of input working conditions does not affect the safety of the IRIS during SBLOCAs.

2) The change trend of the envelope band of most parameters is from narrow to wide and then narrow. In the middle stage of the accident process, the uncertainty of FOMs is significant due to the difference in the trip time of safety measures. While at the beginning of the accident and the later stage of the accident process, the uncertainty of FOMs is smaller.

3) The disturbance of input parameters has a different influence on different FOMs. For example, the uncertainty envelope of PCT is always very smooth, and the uncertainty range is small, while the uncertainty envelope of the break void fraction changes sharply, and the uncertainty range is extensive.

4) The main factors affecting the uncertainty of the same FOM vary in different periods. For example, the uncertainty of the break flow rate after the break is affected by the uncertainty of the break area and discharge coefficient, and the influencing factors are simplex. Hence, the uncertainty envelope bandwidth is relatively stable. In the middle of the accident process, the uncertainty envelope bandwidth of the break flow rate becomes more significant due to the uncertainty of the trip time of safety measures and EBT parameters. In the later stage of the accident process, the uncertainty envelope band of the break flow rate changes significantly and oscillates violently due to the drastic change in void fraction uncertainty.

3.4. Sensitivity Analysis

Sensitivity analysis can be used to find input parameters that significantly impact FOMs. Spearman's rank correlation coefficient is suitable for sensitivity analysis of multiple input parameter combinations. The FOMs of sensitivity analysis are the maximum break flow rate, the maximum EBT flow rate, and the maximum core inlet and outlet temperature. The input parameters are the relevant parameters in the uncertainty analysis. Among the discharge coefficients, only the subcooled discharge coefficient is analyzed. The figures below show the Spearman's sensitivity coefficients for several FOMs.

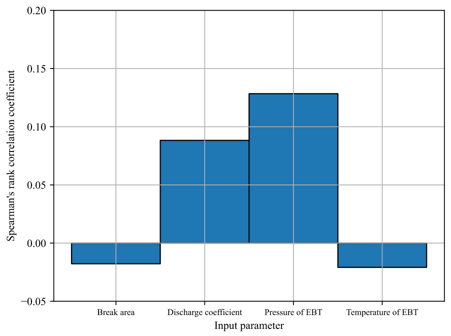

Figure 17. Sensitivity to Maximum Break Flow Rate

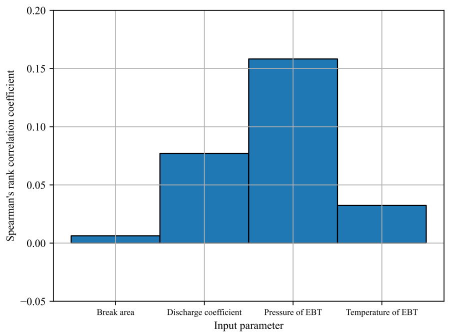

Figure 18. Sensitivity to Maximum EBT Flow Rate

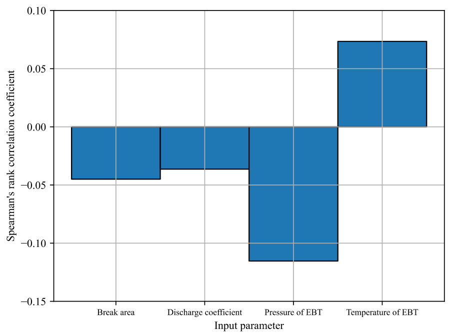

Figure 19. Sensitivity to Core Inlet Temperature

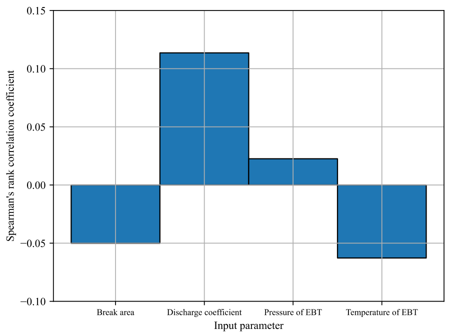

Figure 20. Sensitivity to Core Outlet Temperature

The following conclusions can be drawn from the analysis of the above figure:

1) For break flow rate and EBT flow rate, the sensitivity coefficients of break discharge coefficient and EBT pressure are higher, and both are positive, indicating that the larger the break discharge coefficient and the higher EBT pressure are, the more significant coolant loss and faster EBT injection will be caused.

2) The sensitivities of the break area to several FOMs are low, and the absolute values of Spearman's coefficients are less than 0.05, indicating that the break size is not a decisive factor affecting the accident process.

3) The temperature of EBT has little effect on the break flow rate and the operation of EBT, but it will significantly affect the coolant temperature of the core.

4. CONCLUSION

In this paper, the BEPU framework built by RELAP5 and RAVEN is used to analyze SBLOCAs of the IRIS. The case study first estimates the SBLOCAs induced by breaks in the DVI and CVCS pipelines. The best estimate results show that the reactor's design safety characteristics and passive safety facilities can alleviate the accident process in time and make the core cool down and decompress steadily without causing severe consequences. Meanwhile, the influence of different break locations on the accident process is also compared. Then, by coupling RAVEN and RELAP5, uncertainty analysis is conducted based on the DVI case. The results show that the uncertainty range of PCT is far below the safety threshold, indicating that IRIS has a high safety margin during SBLOCAs. Meanwhile, the influence of input parameter uncertainty on different FOMs is also discussed in the study by sensitivity analysis.

References

[1] Carelli M D. IRIS: A Global Approach to Nuclear Power Renaissance. Nuclear News. 46(10), 32-42, 2003.
[2] IAEA. Accident Analysis for Nuclear Power Plants. Safety Reports Series. 23, 2002.
[3] Ye Q. Best Estimate Plus Uncertainty Analysis for SBLOCA on a Small Modular Reactor (in Chinese). Huazhong University of Science and Technology, Wuhan, 2019.
[4] USNRC. RELAP5/MOD3.3 Code Manual Volume I: Code Structure, System Models and Solution Methods. Rockville, 2001.
[5] Alfonsi A, Rabiti C, et al. RAVEN Theory Manual, Revision 4. Idaho National Laboratory, Idaho, 2021.
[6] Bajs T, Oriani L and Conway L. Development of RELAP5 Nodalization for IRIS Non-LOCA Transient Analyses. American Nuclear Society Topical Meeting in Mathematics & Computations, Gatlinburg, 2003.
[7] Westinghouse. IRIS Preliminary Safety Assessment. Pittsburgh, 2003.
[8] Larson T K, Moody F J, et al. Iris Small Break LOCA Phenomena Identification and Ranking Table (PIRT). Nuclear Engineering and Design. 237(6), 618-626, 2007.
[9] USNRC. IRIS International Reactor Innovative and Secure, Plant Overview. https://www.nrc.gov/docs /ML0336/ML033600086.pdf. 2002.
[10] USNRC. 50.46 Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors. https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0046.html. 2021.
[11] Oriani L, Conway L E and Grgic D. Small Break Loss of Coolant Accident Analysis for the International Reactor Innovative and Secure (IRIS). 5th International Conference on Nuclear Option in Countries with Small and Medium Electricity Grids, Dubrovnik, 2004.
[12] Wilks S S. Statistical Prediction with Special Reference to the Problem of Tolerance Limits. Annals of Mathematical Statistics. 13(4), 400-409, 1942.
[13] Wilks S S. Determination of Sample Sizes for Setting Tolerance Limits. Annals of Mathematical Statistics. 12(1), 91-96, 1941.
[14] IAEA. Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation. Safety Reports Series. 52, 2008.