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LBLOCA Uncertainty Analysis of Maanshan Nuclear Power Plant with RELAP5/SNAP and DAKOTA

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ABSTRACT

The objective of this study is to assess the applicability of the RELAP5/MOD3.3 model of Maanshan NPP on LBLOCA transient. Maanshan NPP was the first three-loop PWR in Taiwan constructed by Westinghouse. The total power of the two units of the power plant is 2785 MWt, which consist of 2775 MWt for reactor power and 10 MWt for cooling pumps. For the last few years, the TRACE model of Maanshan NPP was developed and several kinds of transient event including the LOCA, AOO and the URG transient were performed and verified. Recently, the RELAP5/MOD3.3 code is another important development priority for our group. In 2015, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface.

An uncertainty analysis methodology of the postulated LBLOCA for Maanshan NPP using RELAP5/MOD3.3, DAKOTA, and SNAP codes are established in this study. The uncertainty parameters are related to ECCS property. The ranges of uncertainty analysis are based on FSAR and Training Manual of Maanshan. The maximum PCT of the uncertainty analysis is 1293 K. The PCT of FSAR is 1383.7 K. Hence, the result of the uncertainty analysis is below the FSAR data. The difference of uncertainty analysis results and FSAR data may be caused by the different calculation procedures, phenomenological modeling, and nodalization, etc. Additionally, the PCT criteria of 10 CFR 50.46 is 1477.6 K (2200°F). The prediction of the uncertainty analysis is below the criteria of 10 CFR 50.46. This implies that Maanshan NPP in LBLOCA transient is at a safety situation when ECCS is performed.

FOREWORD

The U.S. NRC has developed a thermal hydraulic analysis code, RELAP5, has been designed to perform best-estimate analysis of loss-of-coolant accidents, operational transients, and other accident scenarios in reactor systems. Models used include multidimensional two-phase flow, non-equilibrium thermo-dynamics, generalized heat transfer, reflood, level tracking, and reactor kinetics. Traditionally, the RELAP5 code analysis model was developed by ASCII file, which was not intelligible for the beginners of computer analysis. Fortunately, and graphic input interface, SNAP is developed by Applied Programming Technology Inc. and conducted by the U.S. NRC, the model development process becomes more convenient.

To obtain the authorization of these codes, Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of RELAP5 code. NTHU (National Tsing Hua University) is the organization in Taiwan responsible for the application RELAP5 and SNAP in thermal hydraulic safety analysis. Therefore, the RELAP5/MOD3.3 model of Maanshan nuclear power plant has been developed. To expand the applicability of the RELAP5/MOD3.3 model, an uncertainty analysis methodology of the postulated LBLOCA is established in this study.

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EXECUTIVE SUMMARY

RELAP5/MOD3.3 Patch04 code, which was developed for light water reactor transient analysis at Idaho National Engineering Laboratory for U.S. NRC, is applied in this research. This code is often performed to support rulemaking, licensing audit calculations, evaluation of accident, mitigation strategies, evaluation of operator guidelines, and experiment planning analysis. Same as other thermal hydraulic analysis codes, RELAP5/MOD3.3 is based on nonhomogeneous and non-equilibrium model for the two-phase system. However, calculations in this code will be solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. It can produce accurate transient analysis results in relatively short time.

SNAP is an interface of NPP analysis codes which developed by U.S. NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily. Due to these advantages, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface.

DAKOTA code can be used in uncertainty analysis. DAKOTA is applied for the sampling of input parameters, the calculation of correlations, and ranking of input parameters. DAKOTA provides Monte Carlo sampling and Latin Hypercube sampling methods combined with various PDFs including normal, lognormal, uniform, logunifrom, hypergeometric, and user-supplied histograms. In addition, the uncertainty analysis of DAKOTA also can run with the SNAP interface.

Maanshan NPP is located on the southern coast of Taiwan. Its nuclear steam supply system is a type of PWR designed and built by Westinghouse for Taiwan Power Company. In this research, a RELAP5/MOD3.3 model of Maanshan NPP for LBLOCA transient was developed. This research also presents an uncertainty analysis methodology of LBLOCA using RELAP5/MOD3.3, DAKOTA, and SNAP codes.

The results of the RELAP5 analysis for LBLOCA base case indicate that the RELAP5/SNAP model is able to manage a LBLOCA simulation. The phenomenon of the base case is consistent with the LOFT experiment and the analysis results of Westinghouse. According to analysis results, the evaluation of quench front velocity during reflooding phase strongly affects the PCT. The performance of ECCS may affect reflooding phase. Therefore, an uncertainty analysis was performed in this research.

Most of the uncertainty parameters are related to ECCS property in the uncertainty analysis. The ranges of uncertainty analysis are based on FSAR and Training Manual of Maanshan NPP. The maximum PCT of the analysis is 1293 K that includes the uncertainty effect. However, the PCT of Maanshan FSAR is 1383.7 K. The difference of analysis results and FSAR data may be caused by the different calculation procedures, phenomenological modeling, and nodalization, etc. In addition, the PCT criteria of 10 CFR 50.46 is 1477.6 K (2200℉). Hence, the result of the uncertainty analysis is below the criteria of 10 CFR 50.46. This indicates that Maanshan NPP in LBLOCA transient is at a safety situation when ECCS is performed.

ABBREVIATIONS AND ACRONYMS

ACC	Accumulator
BAF	Bottom of Active Fuel
CAMP	Code Applications and Maintenance Program
CLOF	Completed Loss of Flow
ECCS	Emergency Core Cooling System
FWPT	Feedwater pump(s) Trip
FSAR	Final Safety Analysis Report
HHSI	High Head Safety Injection
kg	kilogram(s)
LBLOCA	Large Break Loss Of Coolant Accident
LHSI	Low Head Safety Injection
MPa	Megapascal(s)
MSIVC	Main Steam Isolation Valves Closure
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRWL	Narrow Range Water Level
NSSS	Nuclear Steam Supply System
NTHU	National Tsing Hua University
PDF	Probability Distribution Function
PORVs	Power-Operated Relief Valves
PWR	Pressurized Light Water Reactor
PLOF	Partial Loss of Flow
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
S/G	Steam Generator
SI	Safety Injection
SNAP	Symbolic Nuclear Analysis Program
SRV	Safety/Relief Valves
TAF	Top of Active Fuel
TRACE	TRAC/RELAP Advanced Computational Engine
WRWL	Wide Range Water Level

1 INTRODUCTION

Maanshan Nuclear Power Plant (NPP) is the third NPP in Taiwan. Also, it is the first Pressurized Water Reactor (PWR) located at the south of Taiwan. There are two units in the Maanshan NPP. The total power of the nuclear steam supply system is 2785 MWt, which consist of 2775 MWt for reactor power and 10 MWt for cooling pumps [1]. For the last few years, our group has developed the models of Taiwan NPPs with TRACE code in SNAP interface [2, 3]. Further, it is necessary to perform the NPP transients with several analysis codes so that the data results could be compared with each other to ensure the consistency. Therefore, the RELAP5/MOD 3.3 Patch04 code was chosen to develop a new Maanshan NPP model. Different from the traditional ASCII input deck, the RELAP5/MOD 3.3 model was developed with SNAP interface.

DAKOTA is an uncertainty analysis code [5] and is applied for the sampling of input parameters, the calculation of correlations, and ranking of input parameters. The uncertainty quantification package of DAKOTA provides Monte Carlo sampling and Latin Hypercube sampling methods combined with various PDFs including normal, lognormal, uniform, logunifrom, hypergeometric, and user-supplied histograms.

SNAP is an interface of NPP analysis codes which developed by U.S. NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections make researches comprehend the whole power plant and control system more easily [6]. Due to these advantages, an uncertainty analysis methodology of LBLOCA using RELAP5/MOD3.3 and DAKOTA with SNAP interface is established in this study. Moreover, due to the SNAP interface, the analysis results could be transferred into animations which were more attractive and more understandable. With the animation, interactions of different components and parameters could be easily observed.

To ensure the applicability of this model, three startup tests cases including feedwater pumps trip (FWPT), turbine trip (PAT50) and main isolation valves closure (MSIVC) had been analyzed in 2015 [7]. With the comparison of RELAP5 results and startup tests data, it shows that the RELAP5/MOD 3.3 model of Maanshan NPP is consistent with the startup tests data. To expand the applicability of this RELAP5 model, the postulated LBLOCA is performed. Two main steps are in this study. First, the establishment of RELAP5/MOD3.3 model for the postulated LBLOCA of Maanshan NPP is performed. An analysis of LBLOCA base case is performed by using this model to confirm the phenomenon of LBLOCA. Second, to confirm the uncertainty of ECCS that may affect the PCT, the uncertainty analysis of RELAP5/MOD 3.3 model combining with DAKOTA is performed in this study.

2 MODEL ESTABLISHMENT

As the Maanshan NPP operated in normal conditions, coolant water in primary system will carry the heat generated by the fuel rods to the steam generator. Feedwater in the secondary system then obtain the heat, evaporate and drive turbines to generate electricity. According to the energy conservation principle, internal energy of steam, which had driven turbines, will decrease. This lower internal energy steam will then go through the condenser and be transferred into feedwater and re-injected into the steam generator. However, it is difficult to develop the entire recirculation system with the analysis code. The computational time will be impractically long and the mass balance will be hard to reach. Hence, as developing the RELAP5 model, it is practical to define the feedwater pumps and the turbines as the boundary conditions because the main purpose of this model is to obtain the NSSS reactions during the transient. For the NSSS system of Maanshan NPP, the feedwater pumps, auxiliary feedwater pumps, turbines, safety/relief valves, steam dump valves and PORVs were defined as the boundary conditions and developed by the Time Dependent Volume component in the RELAP5 program [7].

2.1 Hydraulic Components

As mentioned in section 2, there are three recirculation loops in the Maanshan NPP. In each loop, there is a RCP and S/G. On the hot leg of second loop, a pressurizer which can adjust the pressure of RCS with the spray valves and electronic heater was developed. In this analysis model, there are several Branch components developed to simulate the reactor vessel. According to the core arrangement, Branch components from number 140 to 156 were connected together as the average fuel channel. Branch components from number 120 to 136 were connected together as the hottest fuel channel. Branch components from number 100 to 116 were connected together as the bypass flow channel, as shown in Figure 1. Also, these channels will be connected to the heat structure components to obtain the heat and do the reactor kinetic analysis.

For those three recirculation loops in primary side, they were developed by Pipe, Valve, Branch, Jump and Single Volume, as shown in Figure 2. For these three-digit components, the first digit stands for the loop number (2 for first loop, 3 for second loop and 4 for third loop). Further, the other digits of these components represents to the component types. For instance, component 280 is the recirculation pumps in first loop and component 380 is the recirculation pumps in second loop. Though the Pump component in RELAP5 code has been developed with the pump parameters from Westinghouse, pump characters of the RCPs in this model was input according to Taiwan Power Company NPP training materials and past research models which were calculated by RELAP5-3D and TRACE codes, as shown in Figure 3.

In addition to RCPs, another important thermal hydraulic component in the primary side is heat exchanger. Pipe 250, which was developed for heat exchanger in first loop, was divided into eight nodes. According to the geometry of heat exchanger, junction between fourth cell and fifth cell was 180 degree as shown in Figure 4. Further, the heat structure component can be view as structural component once the both side of the heat structure were connected to thermal hydraulic components. Hence, the Pipe 250 was connected to the left boundary of heat structure 2500, as shown in Figure 5. Likewise, the Pipe 350 of secondary loop was connected to the left boundary of heat structure 3500 and Pipe 450 of third loop was connected to the left boundary of heat structure 4500.

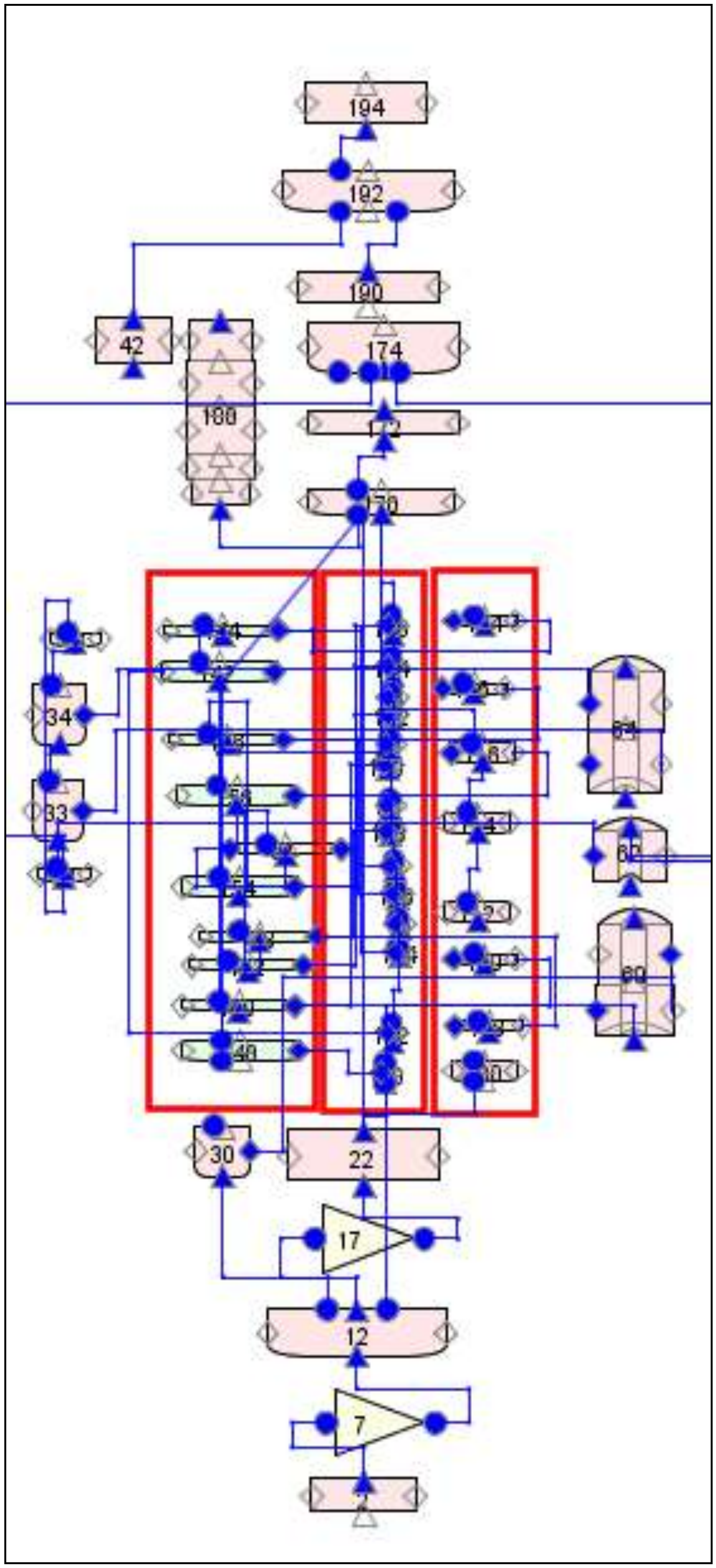

Figure 1 Reactor Core Components of Maanshan NPP in SNAP Interface

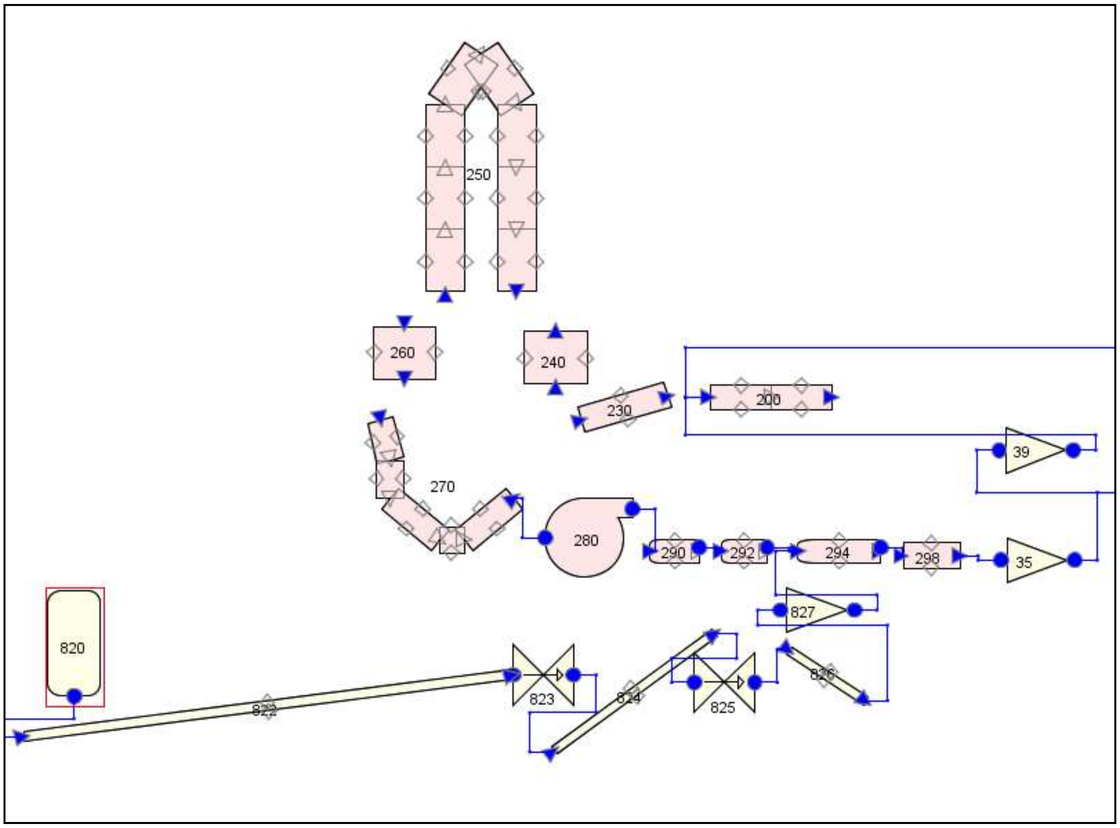

Figure 2 Components of First Loop of Maanshan NPP in SNAP Interface

Similar to primary loops, the secondary loops of Maanshan NPP were developed with Pipe, Valve, Branch, Pump and Single Volume. Specially, to simulate the feedwater, auxiliary feedwater and steam dump systems, which flow rate was determined by system feedback, the Time Dependent Junction was used. With the same rules of primary loops, the components' number in secondary loops was numbered in three digits. The first digit stands for the loop's number and the other two digits stand for the component types. For instance, the component "520" were heat exchanger in first loop because the first digit "5" represents the first loop and the latter two digits "20" represents the heat exchanger. Component 520 was connected to the right boundary of heat structure 2500, which allows the heat transfer from component 250 in primary side to component 520 in secondary side. Due the heat from primary side, the water in component 520 will evaporate and go through the next component 522. Component 522 was a separator which can increase the quality up to 99.7%. This dried steam will then leave the separator and go through the main steam line isolation valve (component 543), turbine control valve (component 774), turbine stop valve (component 775) and drive turbine, as shown in Figure 6.

As mentioned in section 2, the steam dump system was composed by 10 steam dump valves, six turbine bypass valves and several controlling equipment. To save the computational time, this RELAP5 model merged 10 steam dump valves into four groups. Each group was developed by a Time Dependent Junction component which the total steam flow rate was consistent to the operating conditions. Likewise, six turbine bypass valves were developed by two Time Dependent Junction components.

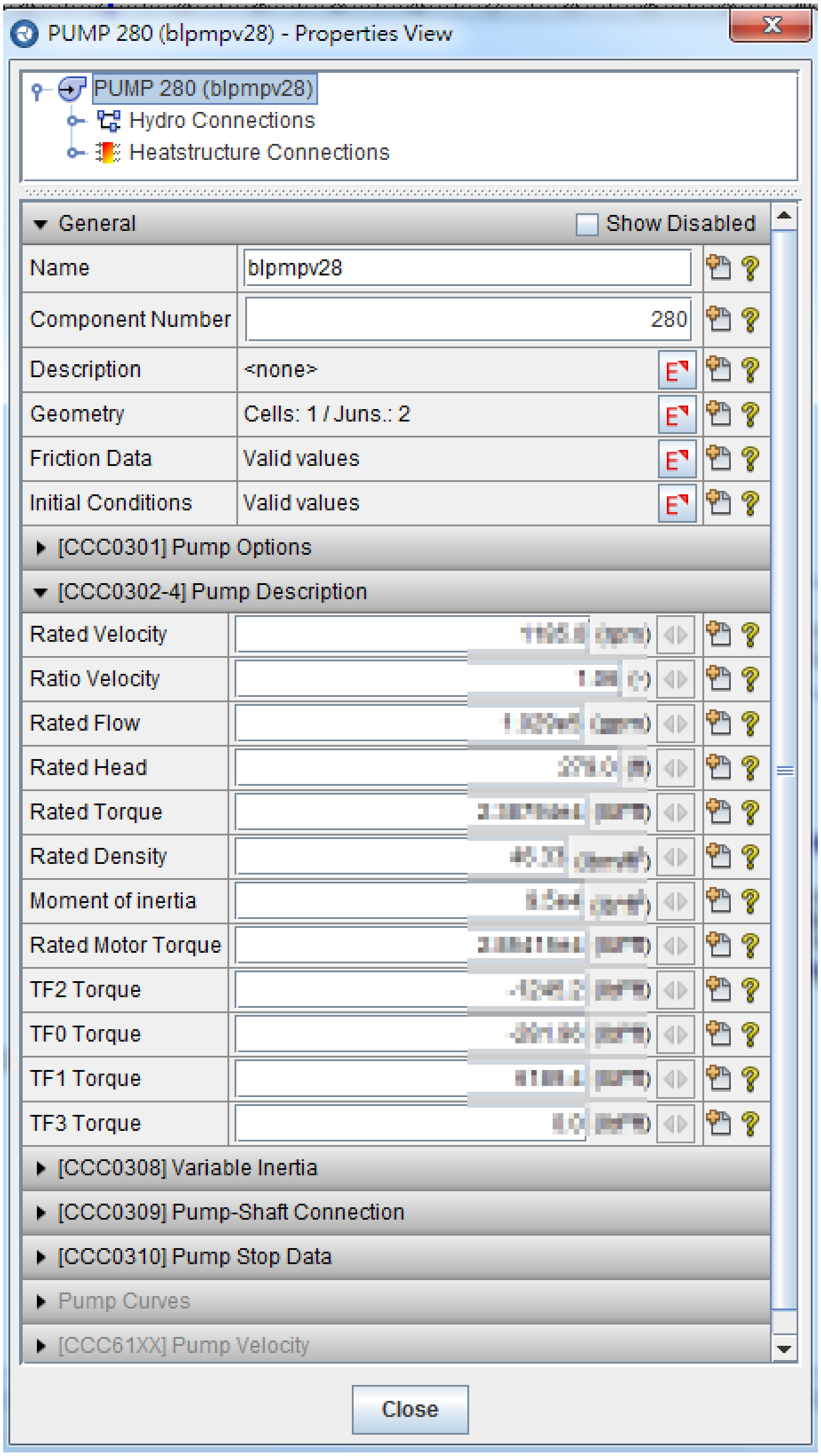

Figure 3 RCP Properties of Maanshan NPP in SNAP Interface

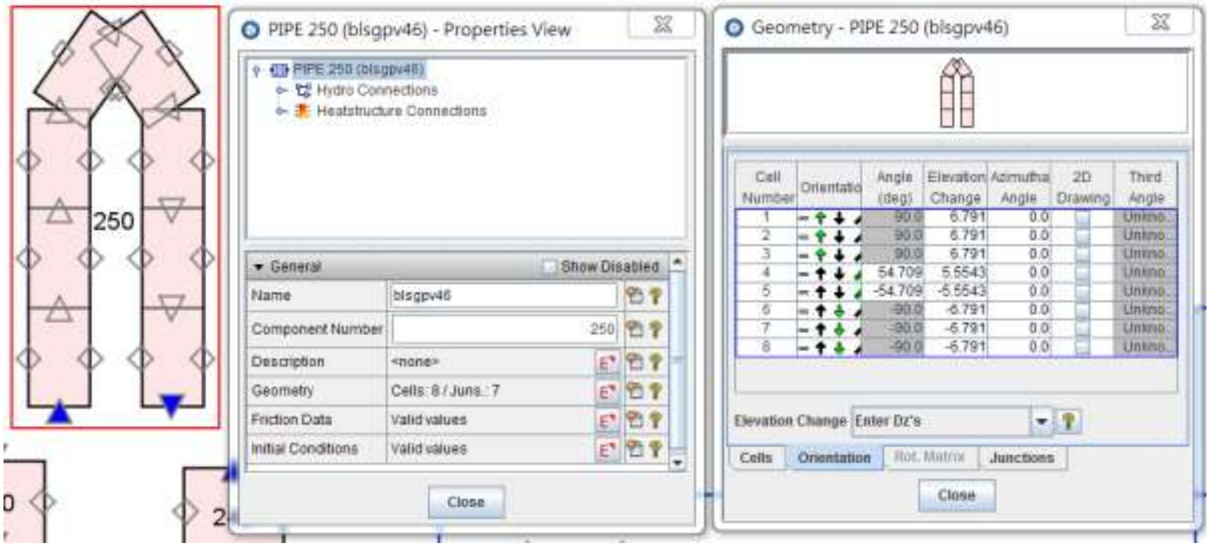

Figure 4 **Heat Exchanger Component 250 of Maanshan NPP in SNAP Interface**

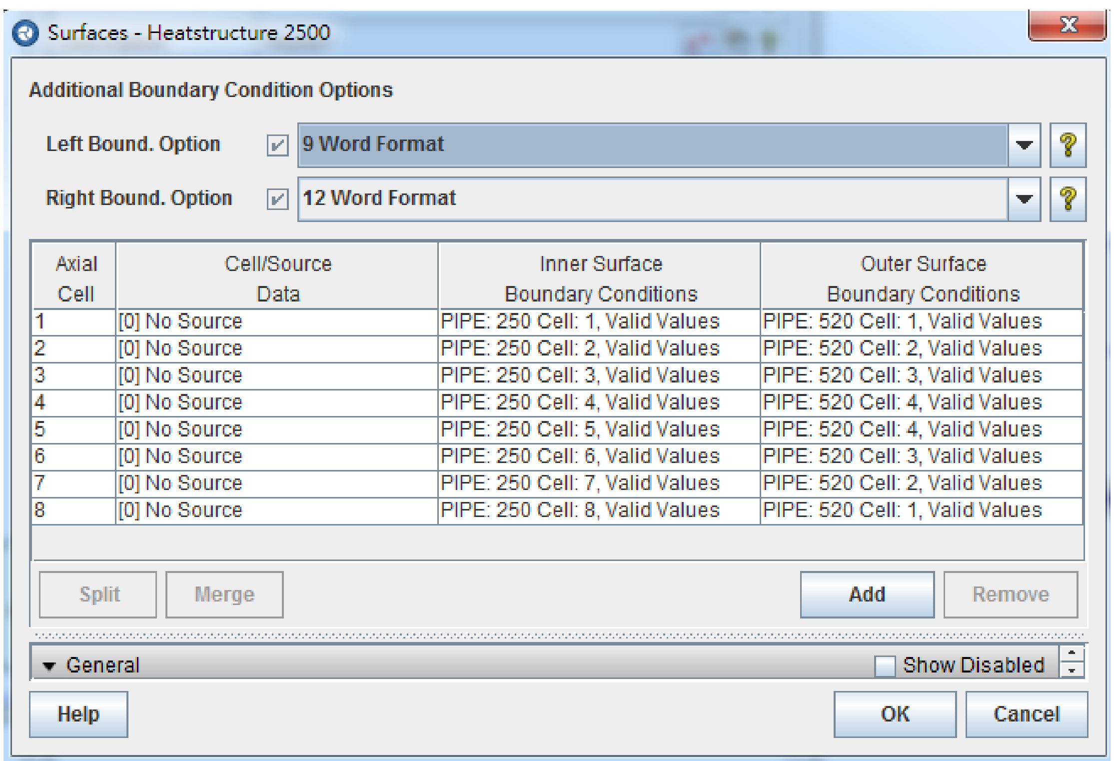

Figure 5 **Heat Structure 2500 Properties in SNAP Interface**

To simplify the feedwater control system, the feedwater pumps and valves were developed by Time Dependent Volume and Time Dependent Junction respectively. For the Time Dependent Volume components, the fluid boundary conditions were referred to the thermal hydraulic properties of feedwater during operation. Therefore, the control system need only concern the effect of NRWL, steam flow rate and feedwater flow rate to determine the feedwater flow rate. Once the flow rate was determined, Time Dependent Junctions which related to the control blocks in the feedwater control system will inject the adequate feedwater into recirculation loops. Details of the feedwater control system will be discussed in the following section.

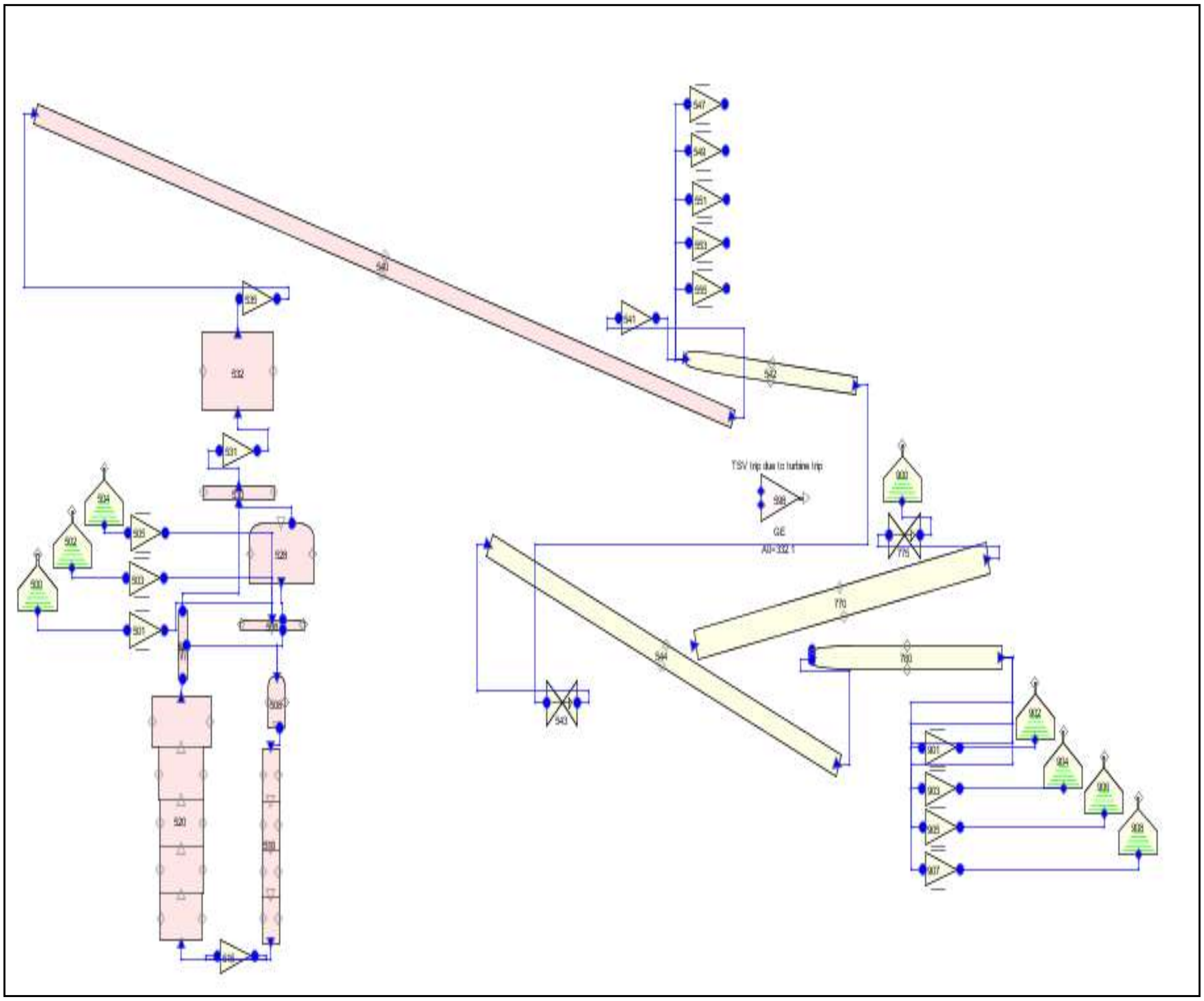

Figure 6 Components in Secondary Side of Maanshan NPP in SNAP Interface

2.2 Control Systems

In operation, the purpose of water level/feedwater control system is to ensure that water in the steam generator can cover the heat exchanger. For Maanshan NPP, the feedwater flow rate was determined by three units including NRWL in steam generator, steam flow rate and feedwater flow rate. As the water level deviate the setting values, the control system will adjust the injection of the feedwater flow rate to maintain the water level of the steam generator. Further, two water level measuring systems including the NRWL and WRWL calculate water level with pressure difference. Different from the TRACE model of Maanshan NPP which our group had developed before, there is no water level sensor signal component in the RELAP5 code. As a result, the measurements of water level were developed and composed with density, pressure and volume signal components which was shown in Figure 7.

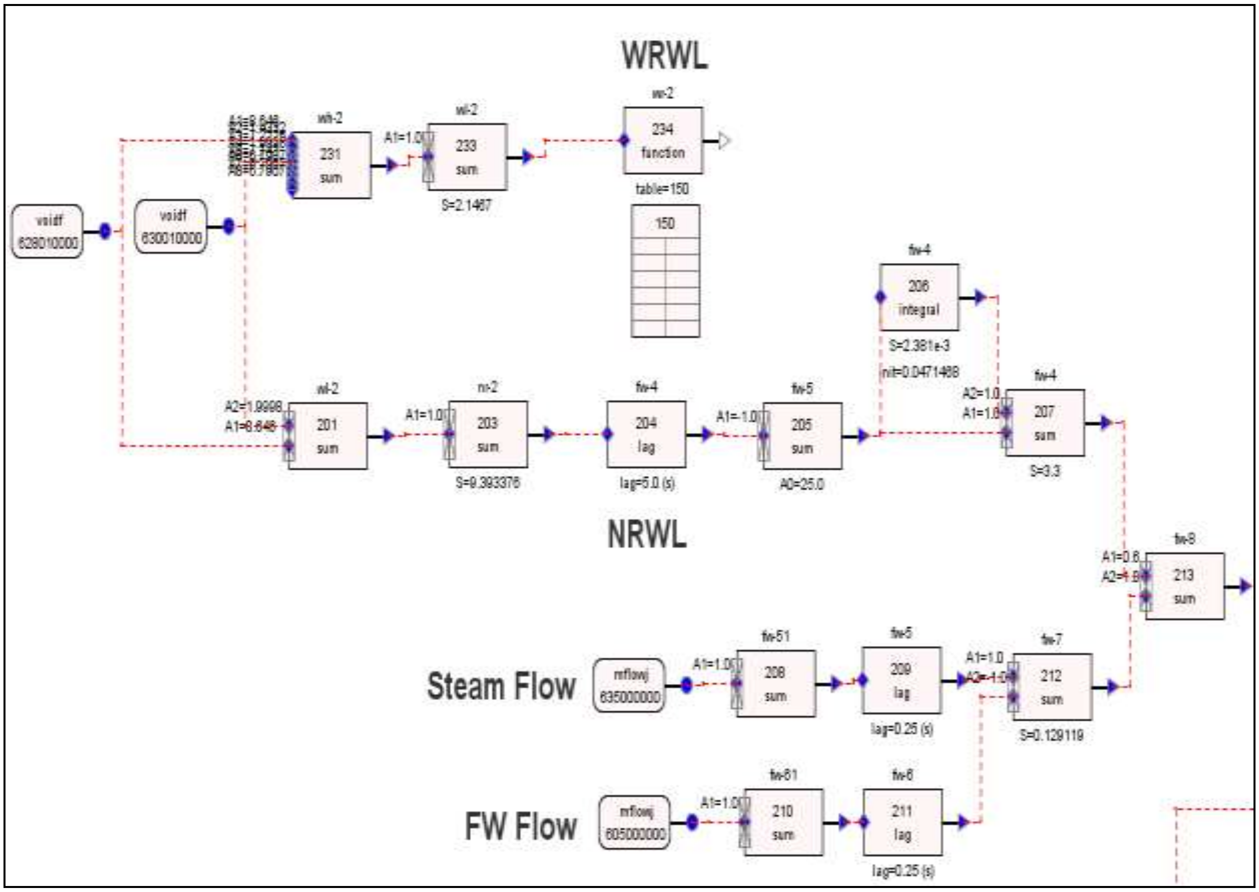

Figure 7 Feedwater Control System in SNAP Interface

In addition to the feedwater control, the steam dump system was also an important response mechanism. As mentioned above, the steam dump system of Maanshan NPP can be divided into two types including the pressure control mode and the Tave mode. The pressure mode was initiated as core power was in range from 0% to 10%, which will not be discussed and applied in this research. Hence, the setting of the steam dump system was only referred to the response of Tave mode. As shown in Figure 8, there are three control blocks with “sum” function calculated the average core temperature values of loop 1 to loop 3 respectively. Then, the control block 308 with “max” function will compare the maximum of average core temperature (Tave) in loop 1 to loop 3 with No Load Temperature (Tno load, 564K in Maanshan NPP). Referring this comparison, the control blocks 318 can convert the difference of Tave and Tno load into steam dump flow rate with Table 15. As the temperature difference exceeded 0% (0°F), the first group of dump valves was opened. As the temperature difference exceeded 16% (15.8°F), the first groups of dump valves were fully opened and the second group of dump valves started to open and so on.

The pressure and water level control system of pressurizer includes the heater and the spray valve. There two types of heater including control heater and backup heater. The control heater and spray valves were applied for adjusting the pressure inside the pressurizer. From Figure 9, the pressure of pressurizer will be compared to rated pressure in control block 120. With the comparison of these two pressure values, the difference can be transferred into the open of the spray valve (control blocks 123) and the power of heater (control block 121 and heat structure 1212 and 1222). However, the control heater is also related to the lower water level of pressurizer (control block 121). As the water level has been lower than 14%, the power of control heater will be zero (control block 124), which means the heater trip. In addition, if the trip setting of control block 121 was assigned to other trip signals, then the control heater can be tripped manually.

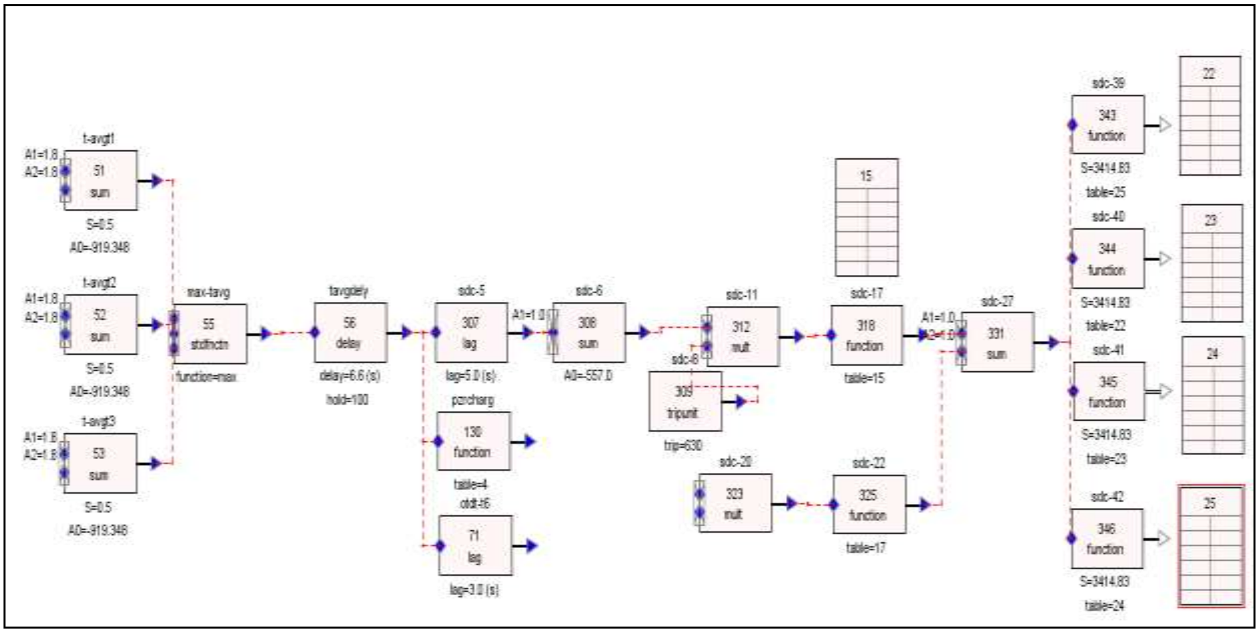

Figure 8 Steam Dump Control System in SNAP Interface

The backup heater is related to the charging control system of pressurizer. As shown in Figure 10, the maximum core temperature will be transferred into program water level through control block 130. Then, the water level will be subtracted from the actual water level. If the difference of these two water levels is larger than 5%, the backup heater will be initiated (control block 132). Further, the water level will be transferred into charging flow rate (control block 136) to adjust the water level inside the pressurizer. However, as the safety injection signal is initiated, the charging flow rate will be forced to zero.

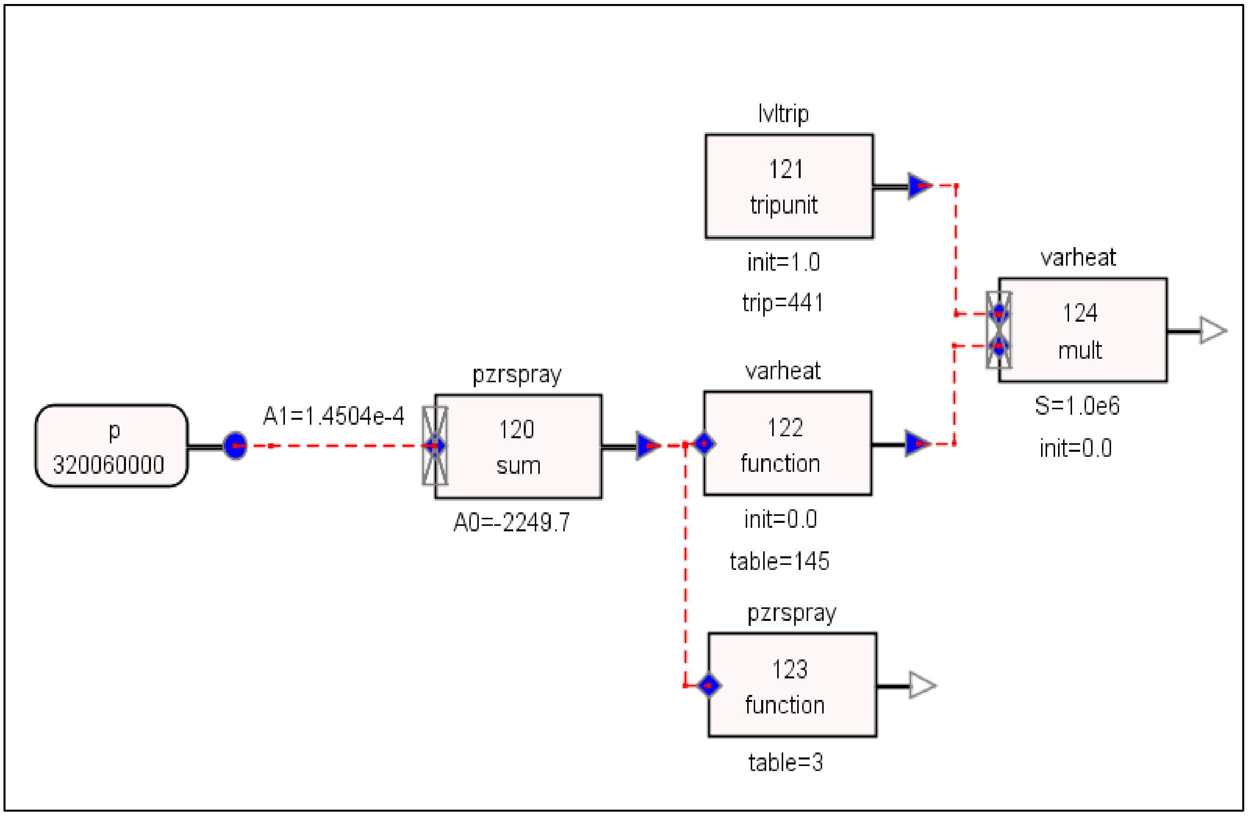

Figure 9 Heater of the Pressurizer Control System in SNAP Interface

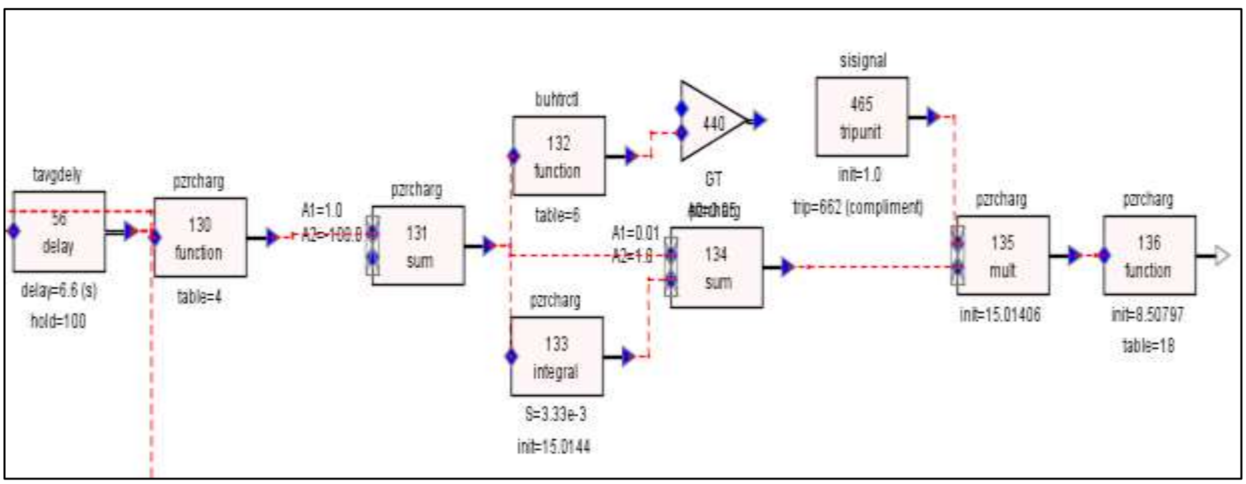

Figure 10 Pressurizer Injection Control System in SNAP Interface

2.3 <u>Reactor Kinetics</u>

In this RELAP5 model of Maanshan NPP, there are two sets of heat structures, which component numbers are 1201 and 1601, developed to simulate the hot fuel channel and average fuel channel. These heat structures were divided into 16 nodes (shown in Figure 11) in axial and 7 nodes in radial (shown in Figure 12). For the axial nodes, they were connected to the Brach components of the reactor core respectively. For radial nodes, the first 4 nodes stand for fuel pellets; the fifth node is filled helium inside the fuel rod and the sixth and seventh nodes are fuel cladding. The materials of each node can be defined manually. In this model, thermal properties (thermal conductivity and thermal capacity) of material 1 for the first 4 nodes were referred to that of Uranium dioxide. The material 2 for node 5 was referred to the helium thermal properties and the material 3 for node 6 and 7 was referred to that of zircalloy.

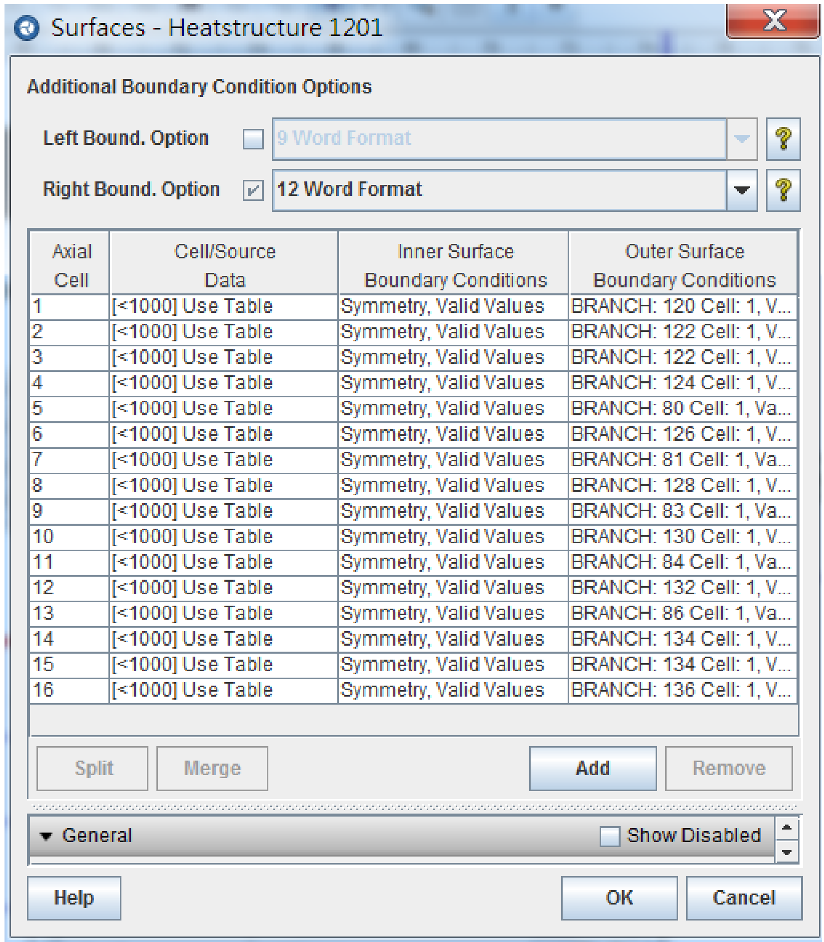

Figure 11 Properties in Axial Direction of Heat Structure 1201 in SNAP

Heat source of the heat structure can be set with the total reactor power or power table. In this model, at the beginning of the model assessment the heat source was set with power table which was referred to the startup test data results of Maanshan NPP to ensure the applicability of the thermal hydraulic components. After that, heat source of the heat structure would be set with total reactor power to ensure the point kinetic feedback calculations. For those heat structure components which were developed as the fuel bundles, the left boundaries were set as “symmetry” and the right boundaries were connected to the Branch components. For these connections of Brach components and heat structures, the power ratio should be defined (as shown in Figure 13) respectively to calculate the correct heat transfer. The power ratio setting of this RELAP5 model was referred to the TRACE models which were fully developed and assessed before.

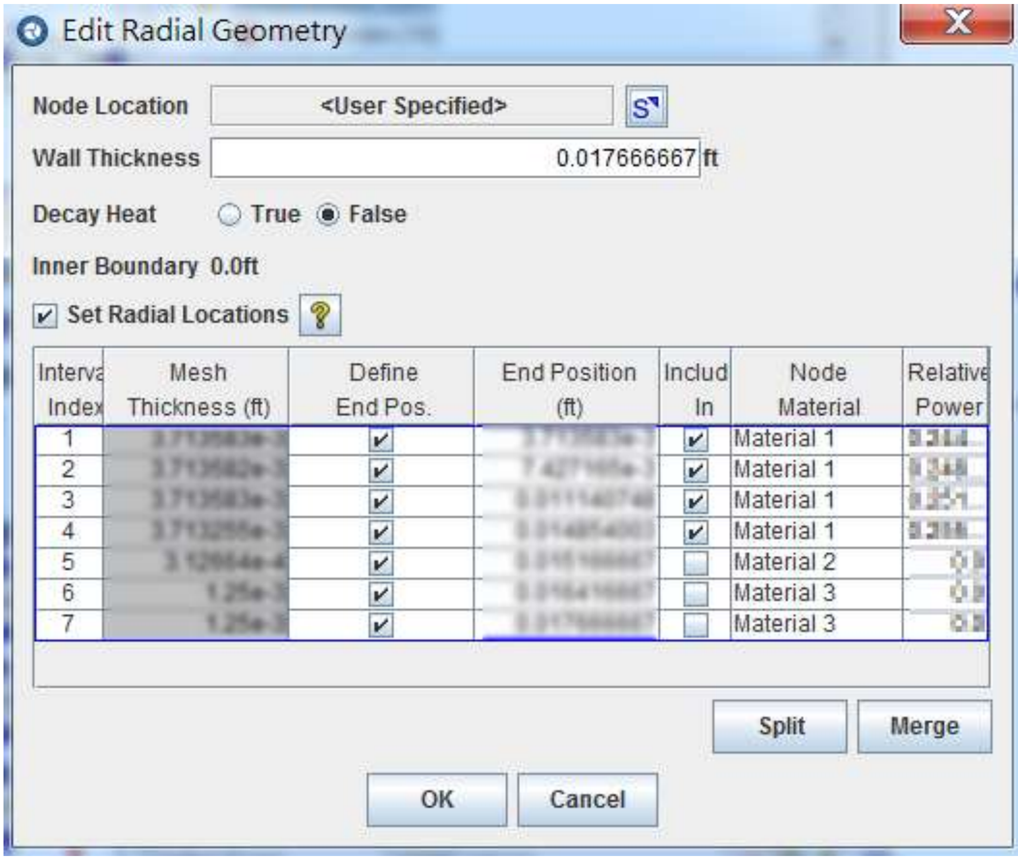

Figure 12 Properties in Radial Direction of Heat Structure 1201 in SNAP

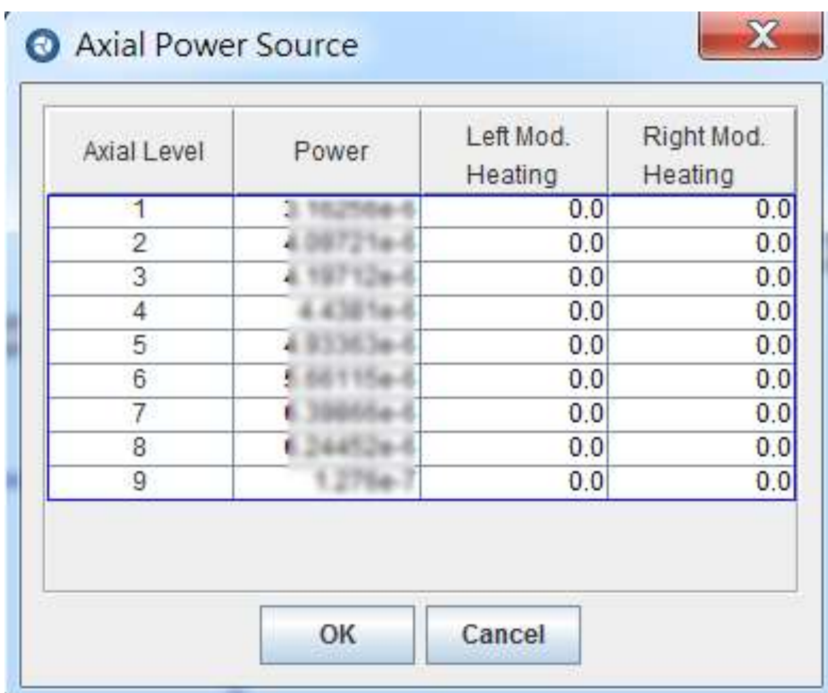

Figure 13 Power Ratio for Branch 100 to 116 of Heat Structure 1201

For the point kinetic model, in addition to defining the power ratio of each node of heat structures, the ratio and position of reactivity feedback should also be defined. The reactivity feedback is dominated by Doppler Effect and Moderator density effect. The previous one is related to the temperature of fuel rods; hence, the check list of fuel temperature and reactivity should be added into the Power components. With this table (as shown in Figure 14), the RELAP5 code can calculate the corresponding reactivity feedback due to fuel temperature. Further, the fuel temperature feedback ratio should also be defined manually in the “Heat Weighting” settlement of Power component (shown in Figure 15). Similarity, to calculate the Moderator feedback, the checklist of coolant density and reactivity should be defined (shown in Figure 16). Then, with the volume weighting list (shown in Figure 17), the Branch components which were developed as the reactor core would be connected to the point kinetic calculation. With these settings, the RELAP5 code could calculate the power variation due to temperature and density changes inside the reactor core during transient events.

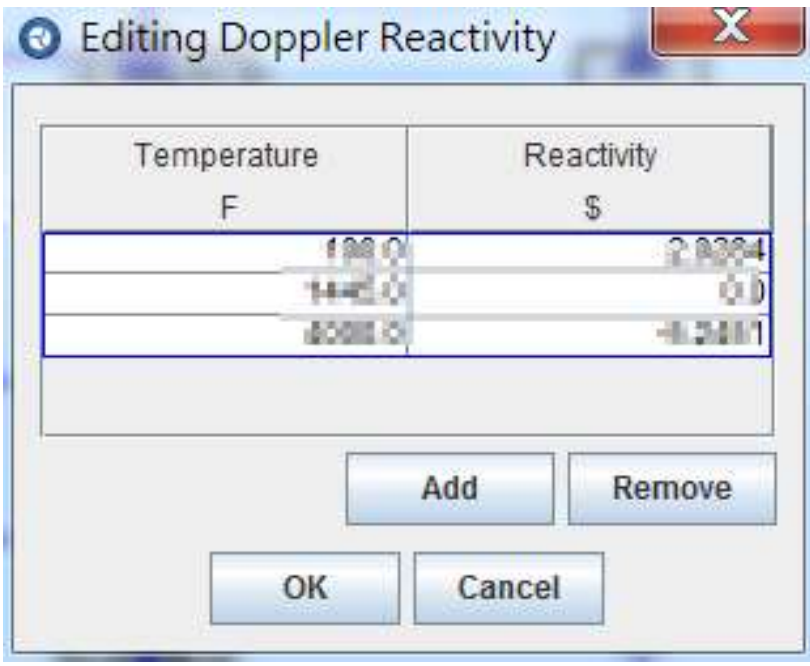

Figure 14 Doppler Effect Reactivity Feedback Table

As mentioned above, the startup assessment transient events were calculated with power table first to ensure the applicability of thermal hydraulic components. Then, the point kinetic model would be applied to do the whole assessment of Maanshan RELAP5 model. As performed with power table, the RELAP5 model needs no control system to simulate reactor scram. However, when performed with point kinetic model, the reactor scram control is required. For instance, Table 100 is the scram reactivity feedback table which will start to dominate the power variation once the trip logic/variable gate is initiated as shown in Figure 18. From this figure, it is obvious that the table could cause a large negative reactivity feedback in few seconds to simulate the control rods insertion.

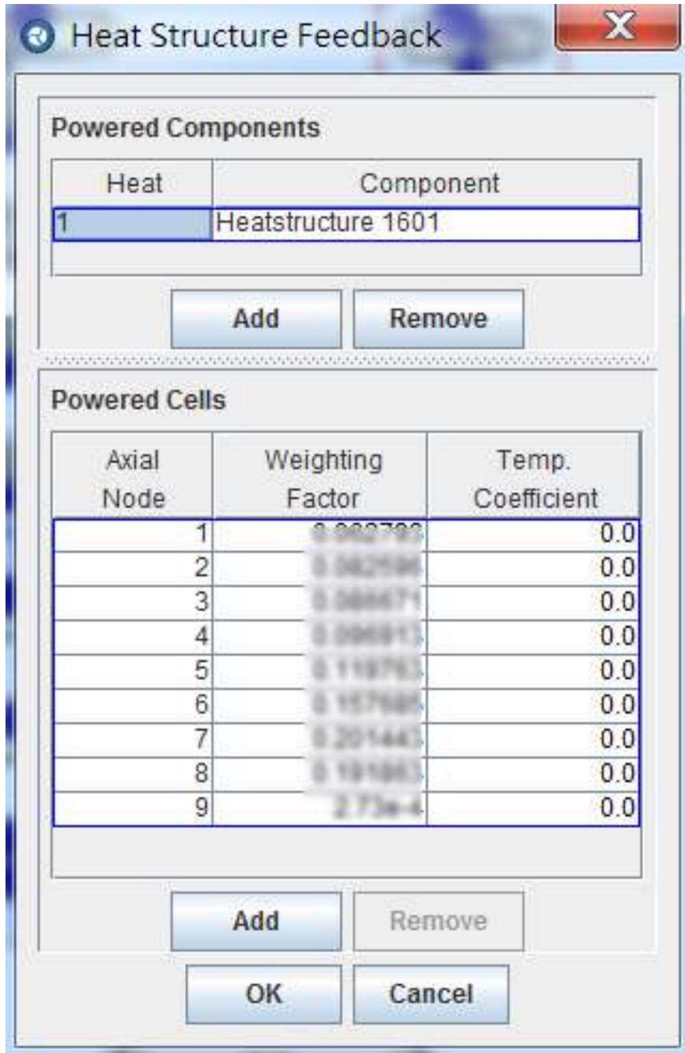

Figure 15 Doppler Effect Heat Structure Weighting Factor

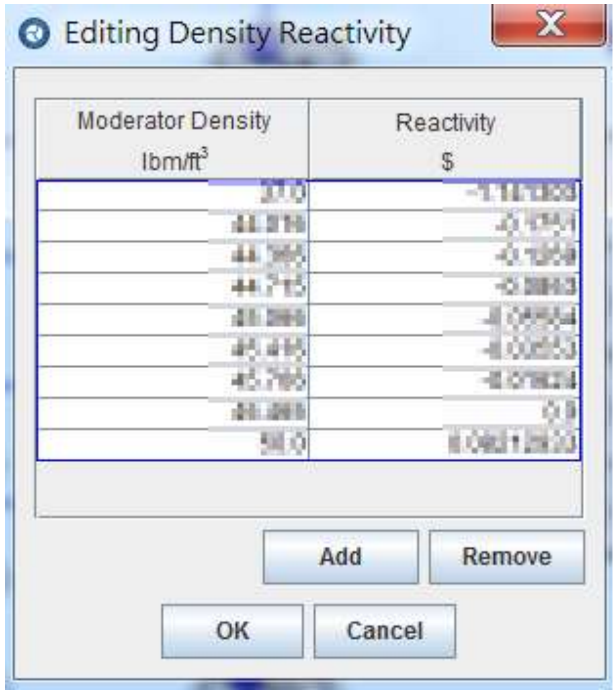

Figure 16 Density Effect Reactivity Feedback Table

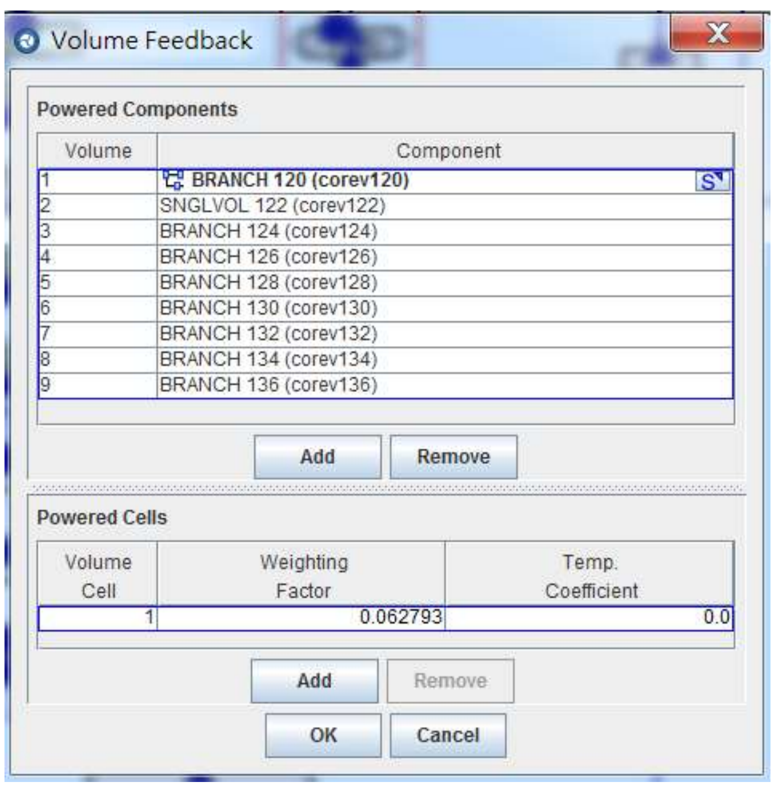

Figure 17 Moderator Density Effect Volume Weighting Factor

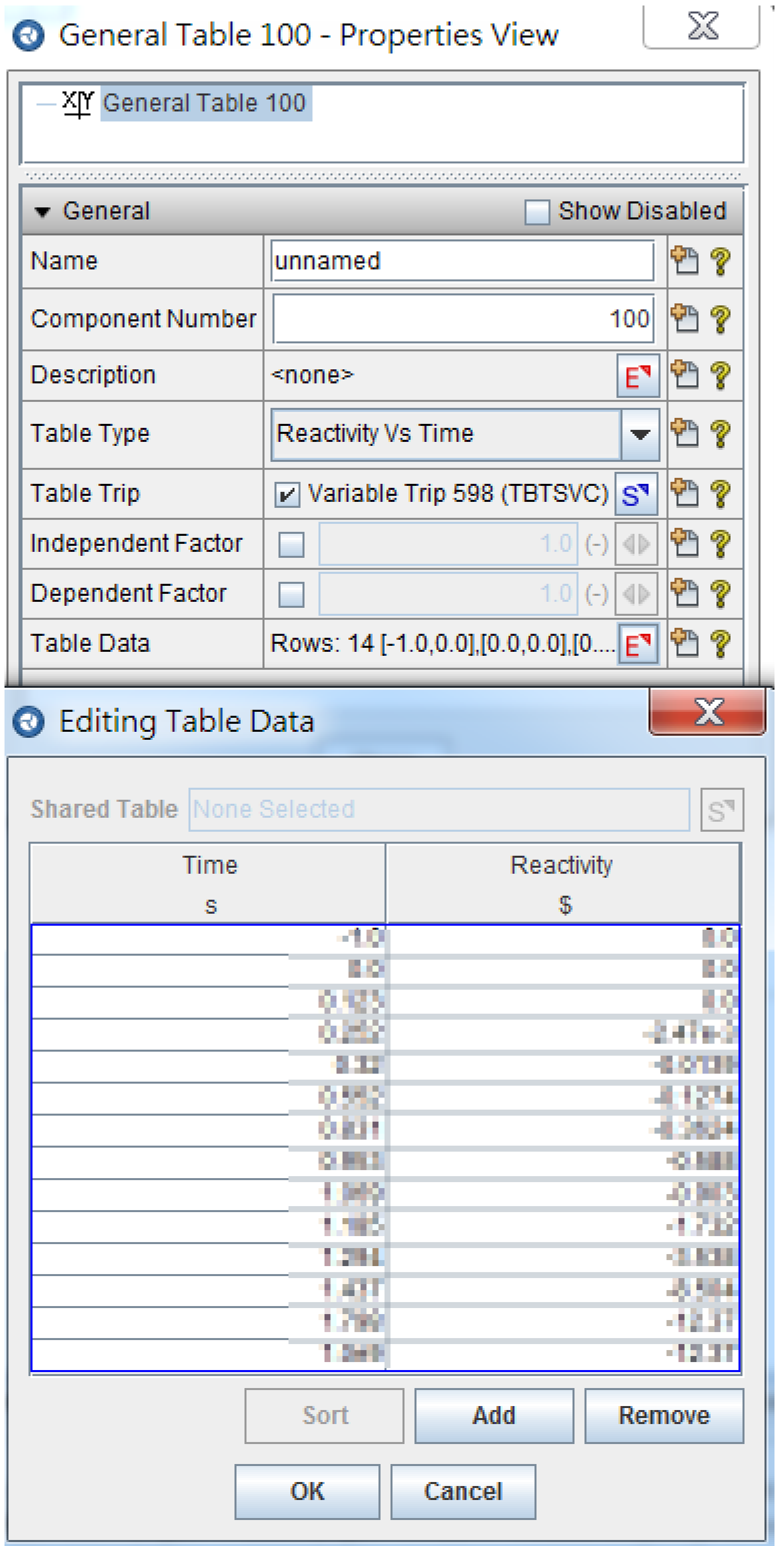

Figure 18 Reactor Scram Reactivity Feedback Properties

3 METHODOLOGY

This methodology has four steps which are (1) develop the basic analysis model using RELAP5. (2) ranking of plant status parameters. (3) ranging of plant status uncertainty parameters (4) generate the input files by random sampling.

(1) Develop the basic analysis model using RELAP5:

RELAP5/MOD3.3 code with SNAP interface is used to do the calculations in this research. The RELAP5 Maanshan NPP analysis model had been developed few years ago. Several verifications and validations had been done, including main steam line isolation valve closure (MSIVC), feedwater pumps trip (FWPT), turbine trip (PAT50), partial loss of flow (PLOF), complete loss of flow (CLOF) [7].

(2) Ranking of plant status parameters:

Before the uncertainty analysis, it is necessary to do the preliminary analysis or study to determine which uncertainty parameters are important. In this research, a LBLOCA base simulation was done to identify the uncertainty parameters. Initial conditions were according to the real operation conditions. Initial power was set to be 2822MWt and assumed all the systems were operated in the normal status. The uncertainty parameters were evaluated based on the results of the base case, including pressurizer pressure, accumulator (ACC) pressure, ACC volume, ACC water temperature, safety injection (SI) water temperature, feedwater temperature. Detail of the LBLOCA base analysis is indicated in next chapter.

(3) Ranging of plant status uncertainty parameters:

To define the uncertainty parameters of plant status, the uncertainty range and the distribution function are both needed. The uncertainty range in this research are according to Maanshan FSAR and Maanshan Training Manual. The distribution function is set to be uniform distribution in this research. Detail of the uncertainty parameters are listed in chapter.

(4) Generate the input files by random sampling:

DAKOTA code is used to do the sampling in this research. RELAP5 and DAKOTA code are coupled on the SNAP interface. The input uncertainty parameters were sampled and generated the RELAP5 input files. Figure 19 is an example of parameters input, uncertainty range and distribution function setting in DAKOTA.

Wilk's formula was adopted to determine the minimum number of runs. In nuclear safety analysis, only upper bond of PCT is the most concern. For the one-sided statistical tolerance limits, the minimum number of runs is given by Wilk's formula:

$$1 - \alpha^n \geq \beta$$

α is the percentage of tolerance limit, β is the confidence level, n is the minimum number of runs. For example, the analysis results in this research achieve α/β = 95/95 criterion, it means there is 95% confidence level that the results are below the 95% of tolerance limit. To achieve the criterion, the minimum number of runs n = 59.

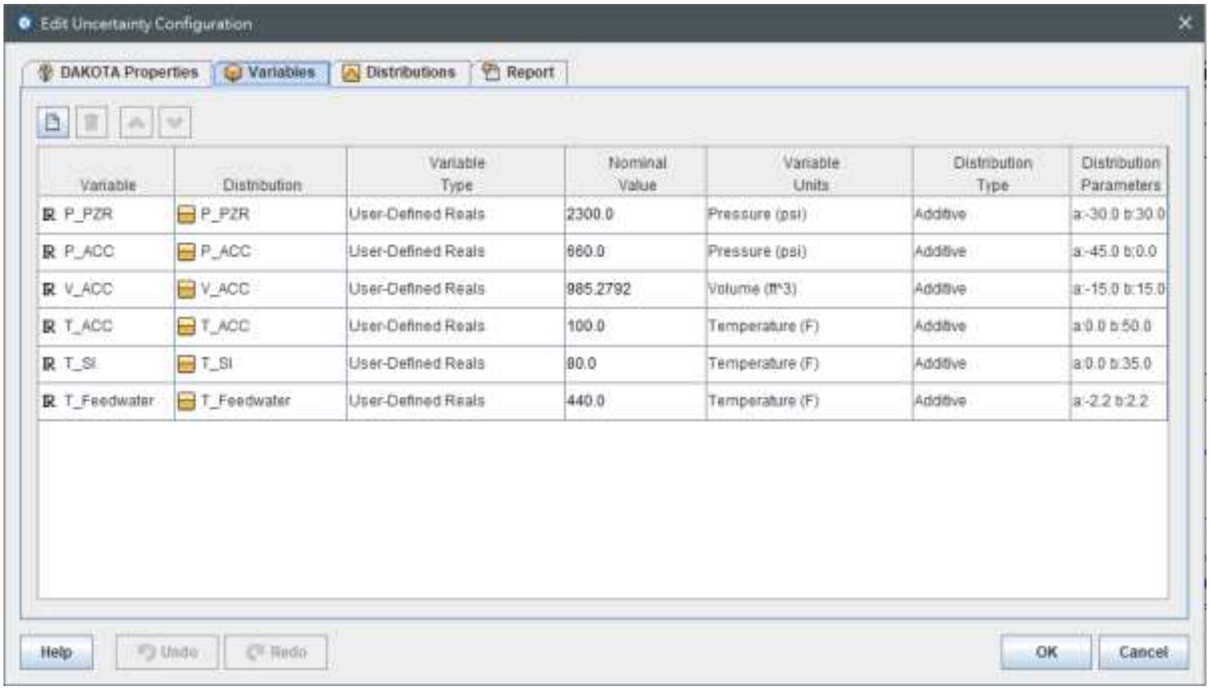

Figure 19 Uncertainty Parameters Setting in DAKOTA

4 LBLOCA BASE CASE

4.1 Initial Condition and Basic Assumption

The initial conditions and the ECCS properties are listed in Table 1. The initial power is 2822 MWt. The LBLOCA base case was performed in this study. It was assumed that the break occurred at the primary side, at first loop cold-leg near the vessel. The break type was assumed to be double-ended and the break model is shown in Figure 20. Before the pipe rupture, 400 sec steady-state calculation had been done to ensure all the system operated correctly. In the LBLOCA transient, High Head Safety Injection (HHSI), ACC injection, and Low Head Safety Injection (LHSI) are included. It was assumed that all the safety injection systems could operate normally. The safety injection signal setpoint is 11.8 MPa and delay time is 27 sec. The ACC injection set point is 4.24 MPa. The minimum operating pressure of Low Head Safety Injection (LHSI) is 1.15 MPa.

Table 1 Initial Condition and ECCS properties

Parameters	Values
Initial power (MWt)	2822
Core flow rate (kg/s)	14176
Tavg* (K)	582.97
RCS* pressure (MPa)	15.862
Scram setpoint (MPa)	12.8
SI* signal setpoint (MPa)	11.8
SI signal delay (s)	27
ACC liquid volume (m^3/tank)	27.9
ACC injection setpoint (MPa)	4.24
ACC liquid temperature (K)	311
LHSI operating pressure(MPa)	1.15
Break size (m^2)	0.38

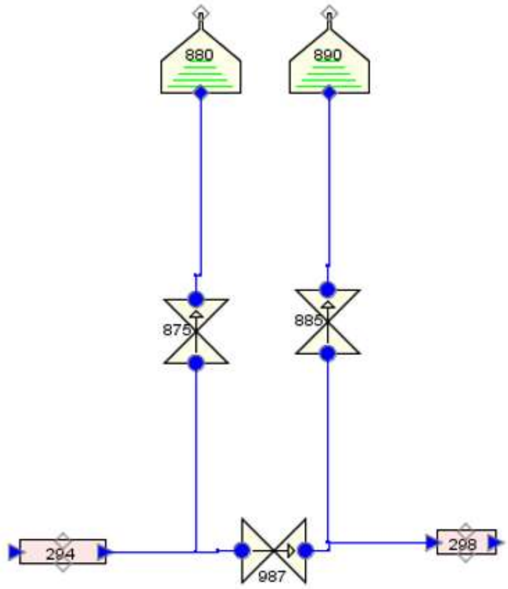

Figure 20 RELAP5/SNAP LBLOCA Break Model

4.2 Analysis Results

The LBLOCA transient can be divided into six stages, which are critical heat flux phase, upward core flow phase, downward core flow phase, refill phase, reflood phase, and long term cooling phase. The exact postulated event sequence of Maanshan NPP is listed in Table 2.

Table 2 Postulated LBLOCA Events Sequence of Maanshan NPP

Action	Time (sec)
Pipe rupture	0
Scram	1.8
RCP trip	1.8
SI signal	2.8
ACC injection	5.5
HHSI	29.8
LHSI	29.8
ACC injection ended	69.5
End of event	700

4.2.1 Critical Heat Flux Phase

Immediately following the cold-leg break, the fluid discharged out of the pipe with very high flow rates, which can be observed in Figure 21. Due to the extremely high break flow rate, the system pressure drop rapidly. This depressurized phenomenon resulted in water flashing in the reactor vessel. The void fraction of vessel is shown in Figure 22, which indicates that the void fraction went up to one in very few seconds. Because of the flashing phenomenon in reactor core, the fuel cladding became fully uncovered. As shown in Figure 23, the cladding temperature increasing dramatically to the first peak in the very early period.

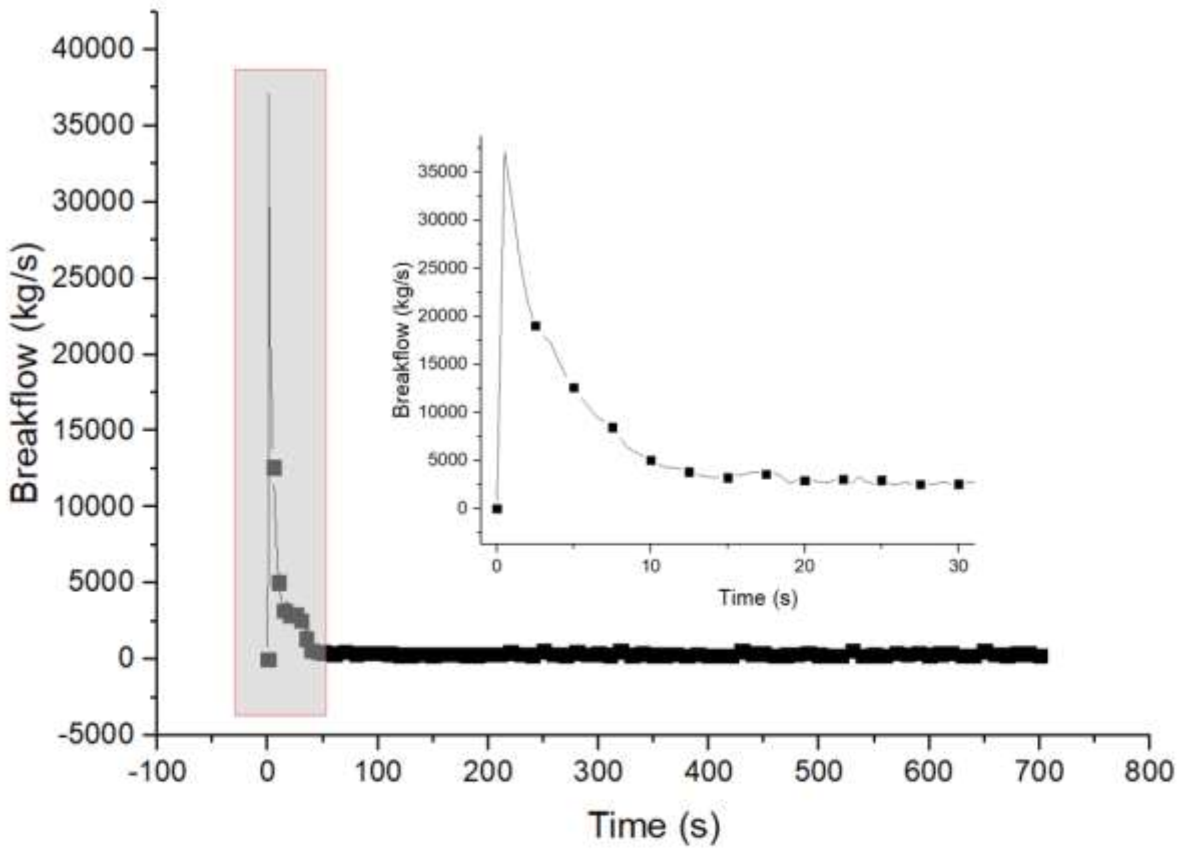

Figure 21   LBLOCA Break Flow Rate

4.2.2   Upward Core Flow Phase

The coolant in loop system still had great inertia at the very beginning (0 ~ 5 sec), and flashing phenomenon in the reactor vessel also generated a great pressure drop. Therefore, there was some two-phase mixture could be pulled into the core, which made the cladding temperature decrease. Because of the bottom-up mixture cooling effect, the temperature drop was more distinct at the lower part. This phenomenon can be seen in Figure 23, there was a temperature drop just after the first peak.

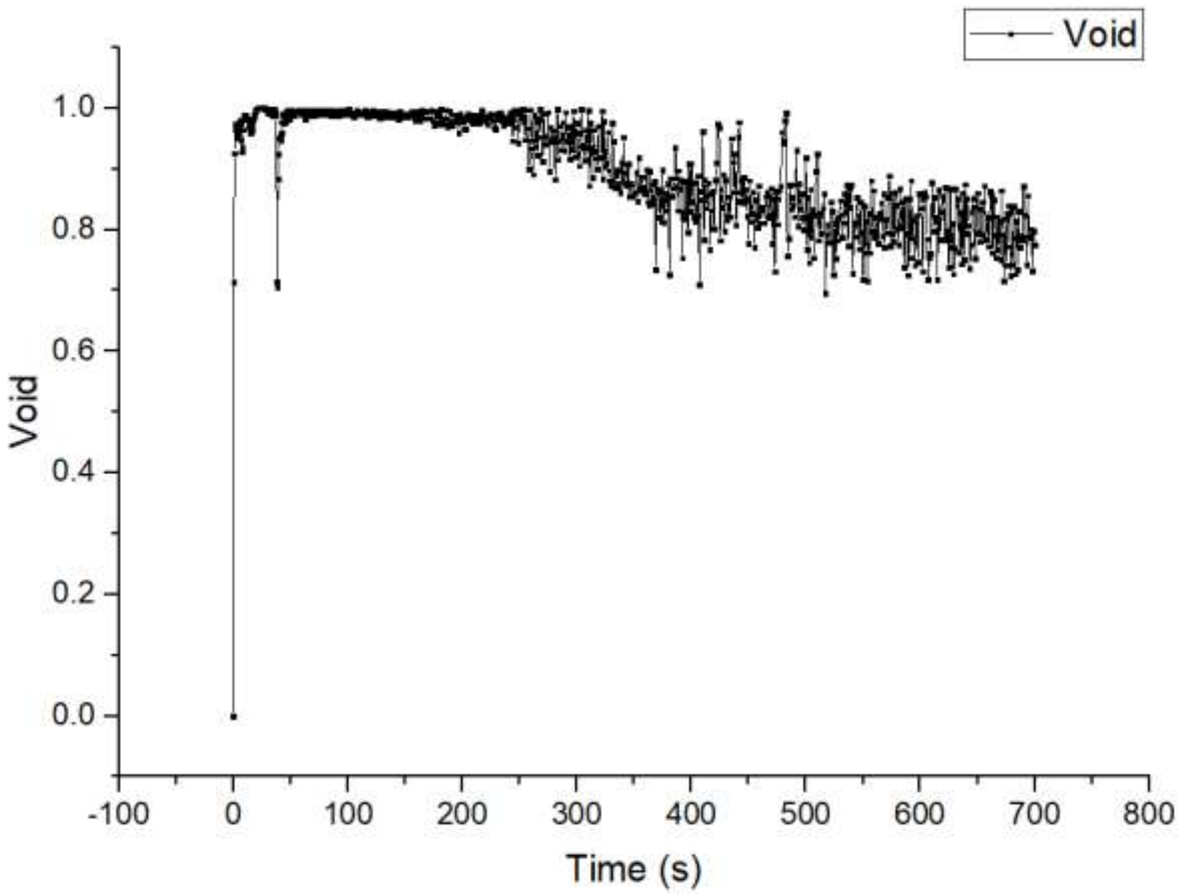

Figure 22 Void Fraction in Vessel

4.2.3 Downward Core Flow Phase

As pump head continues to degrade, and the coolant slow down. The pressure drop at break started to fully dominate and pull flow down through the core, up through the downcomer to the break. Figure 24 shows the vapor velocity in reactor core, which indicate a velocity increasing downward at about 6 ~ 10 sec. High speed vapor flow also provided good core cooling. This phenomenon can be seen in Figure 23, which shows a second temperature drop at about 10 sec.

4.2.4 Refill Phase

Following the blowdown phase, the water level was below the BAF (see Figure 25). Therefore, the cladding temperature increased (see Figure 23). This period is featured of rapid cladding temperature increasing because of lacking water and steam cooling.

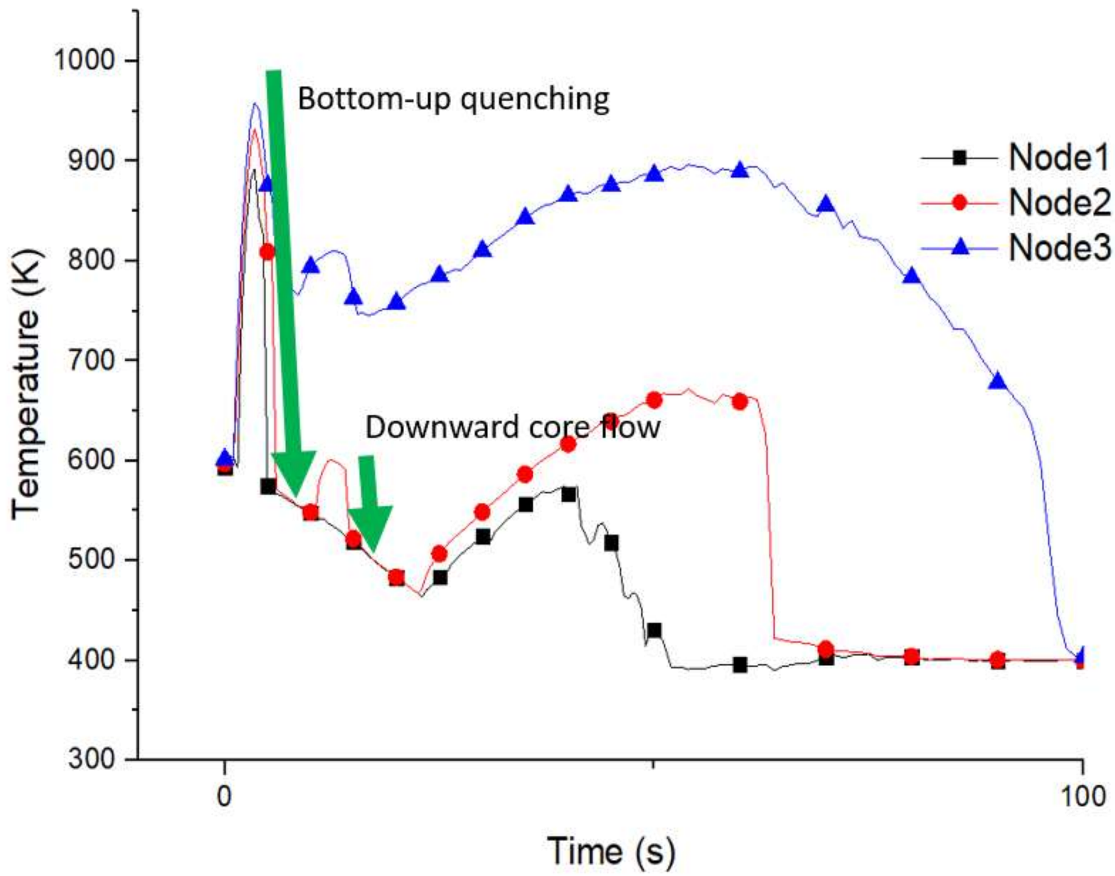

Figure 23 Cladding Temperature at the Lower Part

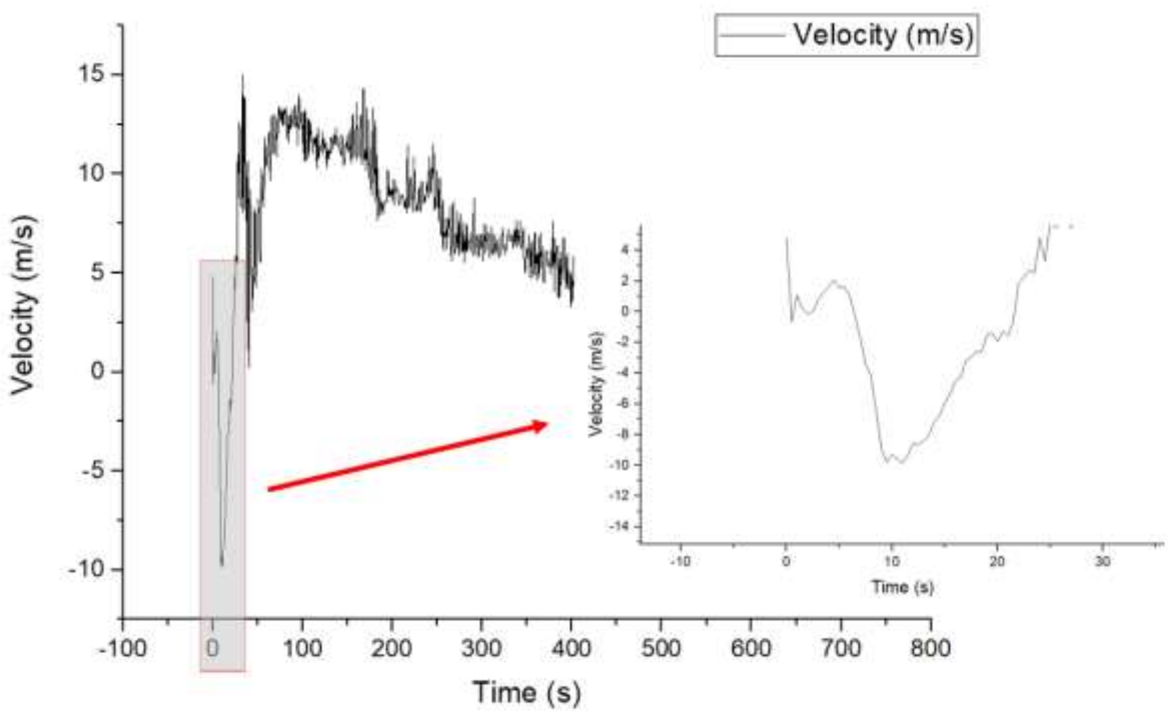

Figure 24 Core Vapor Velocity

4.2.5 Reflood Phase

Due to the ACC injection, the water level continued to increase. Figure 25 shows the core collapsed water level. At about 19 sec, the coolant reached the BAF and lower core region began to quench. During this period, core cooling might increase because of upward vapor flow and liquid entrainment. Therefore, the cladding temperature could decrease before the coolant recovered the fuel cladding.

In general, if the income core flow velocity exceeded certain value, it was assumed that the flow regime near the quench front was Inverted-Slug flow. The references [8~10] indicate that the minimum core coolant inlet velocity should be 0.15m/s to form an Inverted-Slug flow regime. These studies also indicate that the quenching temperature depends strongly on the core inlet coolant velocity.

The flow regime during reflooding phase was shown in Figure 26. Under the quench front, fuel cladding was covered by liquid and cladding temperature dropped to saturation temperature. At inverted-annular and droplet region, the two phase mixture still provided good cooling effect that made the cladding temperature go down. Therefore, this period is the most important in LBLOCA analysis. Figure 27 shows the quench and rewet points for the nodes.

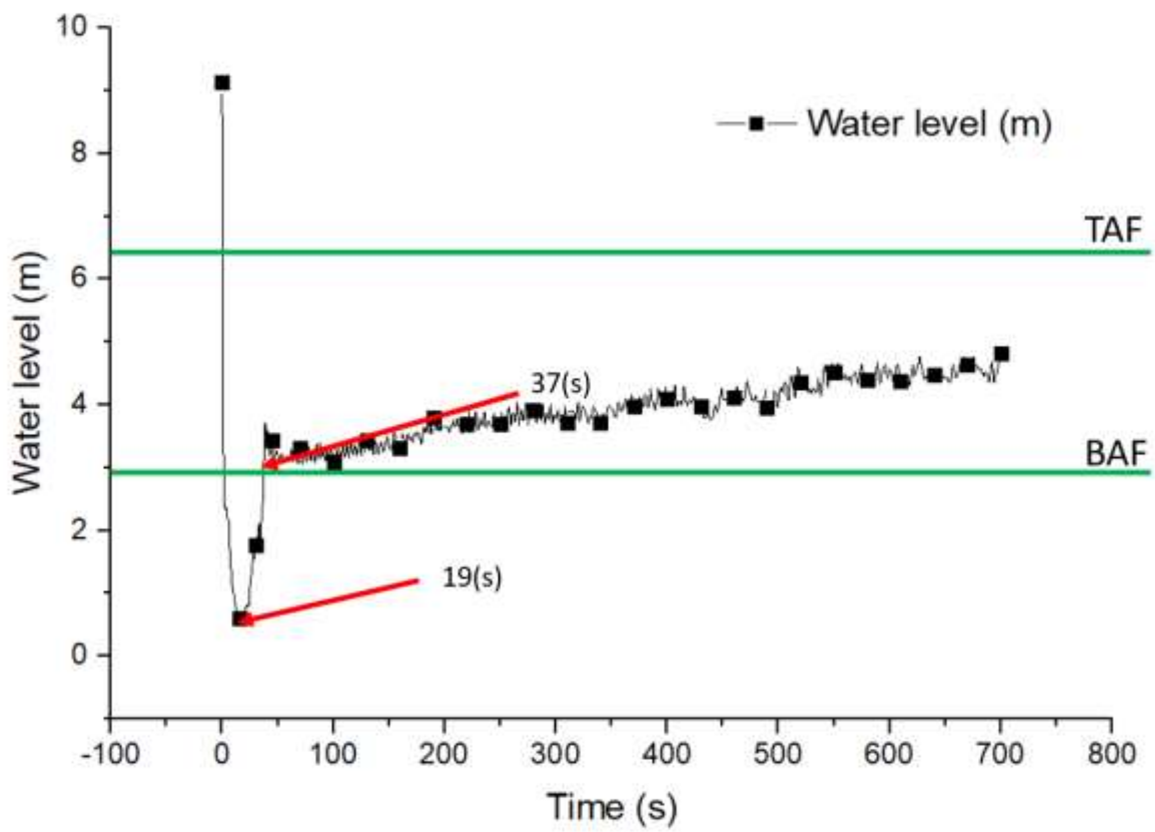

Figure 25 Core Collapsed Water Level

4.2.6 Long-Term Core Cooling

Whole fuel cladding was recovered by coolant at about 500 sec, which can be seen in Figure 27. With the pumped safety injection operating continuously, the water will never below the TAF again.

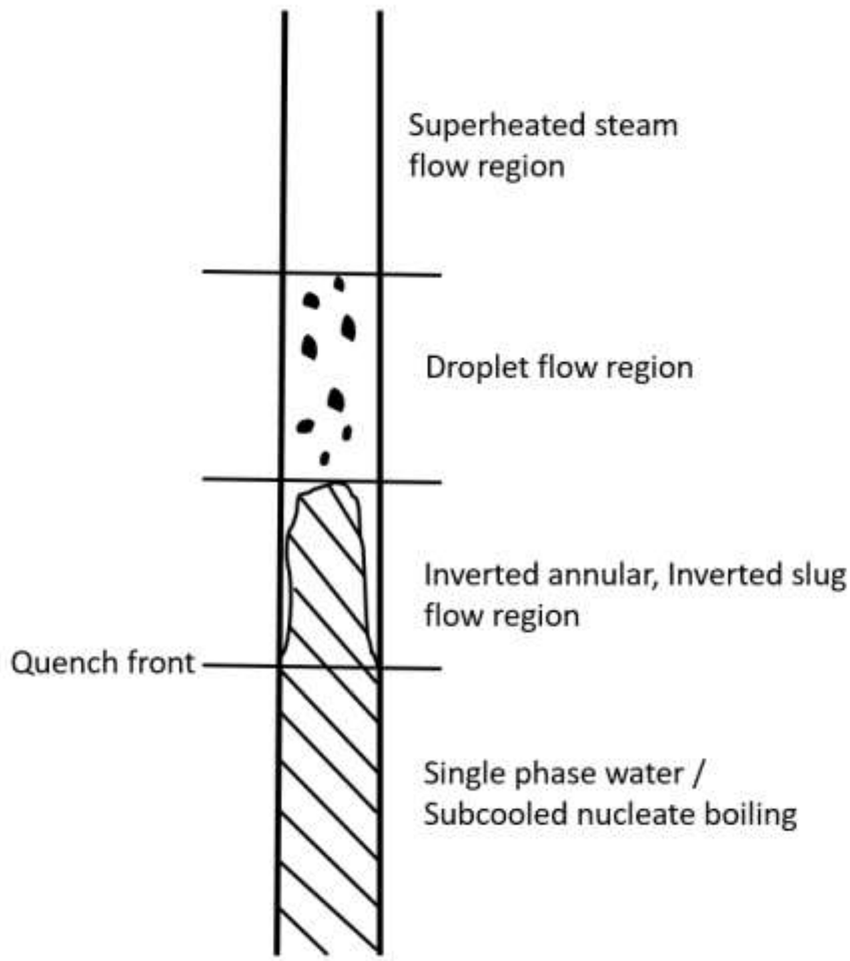

Figure 26 Core Flow Regime during Reflooding phase

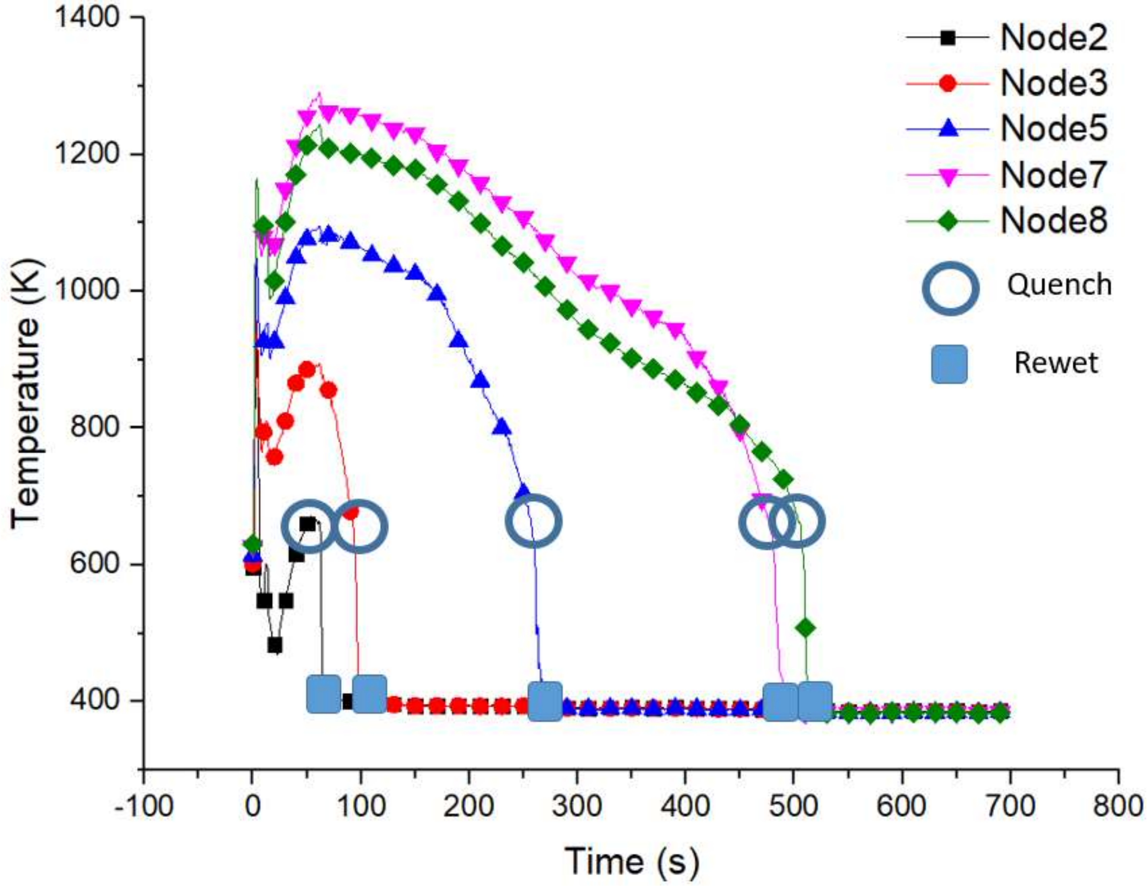

Figure 27 Cladding Temperature Axial Profile

5 LBLOCA UNCERTAINTY ANALYSIS

5.1 Uncertainty Parameters

According to the previous Chapter, the evaluation of quench front progressing during the reflooding phase strongly affects the cladding temperature. The quench front velocity is a relation between TQ and h [11, 12]. TQ is the quenching temperature, and h is the heat transfer coefficient of the coolant under the quench front. TQ and h strongly depend on system pressure and subcooling of the coolant. In LBLOCA case, the performance of ECCS may affect the PCT, system pressure, and subcooling of the coolant. Therefore, the uncertainty analysis by using RELAP5, DAKOTA, and SNAP codes was performed in this study. The uncertainty parameters which are mainly the properties of ECCS are presented in Table 3. These parameters include pressurizer pressure, accumulator (ACC) pressure, ACC volume, ACC water temperature, safety injection (SI) water temperature, and feedwater temperature. The uncertainty ranges are based on FSAR and Training Manual of Maanshan NPP. The distribution functions are all set to be uniform distribution. According to the Wilk's formula, 59 of runs should be conducted to achieve the 95/95 criterion. These uncertainty parameters would be randomly sampled and generated 59 RELAP5 input files.

Table 3 Uncertainty Parameters

	Parameter	Deviation	PDFs	Reference
1	PZR pressure	15.86 MPa ($\pm$0.21MPa)	Uniform distribution	FSAR
2	ACC pressure	4.14 - 4.55 MPa		Training manual
3	ACC volume	27.9 m^3 ($\pm$0.425 m^3)		Training manual
4	ACC temperature	310.93 - 338.71 K		Training manual
5	SI temperature	300 K (+19.4 K)		FSAR
6	Feedwater temperature	500 K ($\pm$1.22 K)		Training manual

5.2 Results

The core collapsed water level can be seen in Figure 28, which shows the time of end-of-blowdown is estimated at about 19 sec. Reflooding begin at about 37 sec, the time which the coolant reaches the BAF. PCT of 59 cases was shown in Figure 29. During blowdown and refilled phases, the temperature results of 59 cases are very similar. The time of the first temperature peak of 59 cases all occur at about 7 sec and the value of all cases are also almost identical (1293 K). In the beginning of the pipe break, a severe depressurization causes liquid in reactor core suddenly transfer to vapor phase. The heat transfer coefficient of vapor is much lower than liquid makes the cladding temperature reach to the first peak at about 7 sec. It is obvious that the first peak is caused by the heat transfer coefficient difference between liquid and vapor. Therefore, these six uncertainty parameters, which are related to ECCS, have no effect on the first peak. After the coolant reaches BAF, both PCT and water level curves start to have distinct deviations. During this period, the coolant started to attach the lower part of fuel cladding. Some liquid droplets would be entrained by vapor from lower part to upper part that makes the cladding temperature do not increase dramatically like the first peak. Therefore, the deviations of ECCS properties made the cooling effect different and makes different PCT. The red curve in Figure 29 shows the highest PCT of all cases, which is about 1293 K. The PCT of LBLOCA in Maanshan FSAR is 1383.7 K. The difference of analysis results and FSAR data may be caused by the different calculation procedures, phenomenological modeling, and nodalization, etc. In addition, the PCT criteria of 10 CFR 50.46 is 1477.6 K (2200°F). Comparing the criteria and the results of uncertainty analysis, the results of uncertainty analysis are below the criteria.

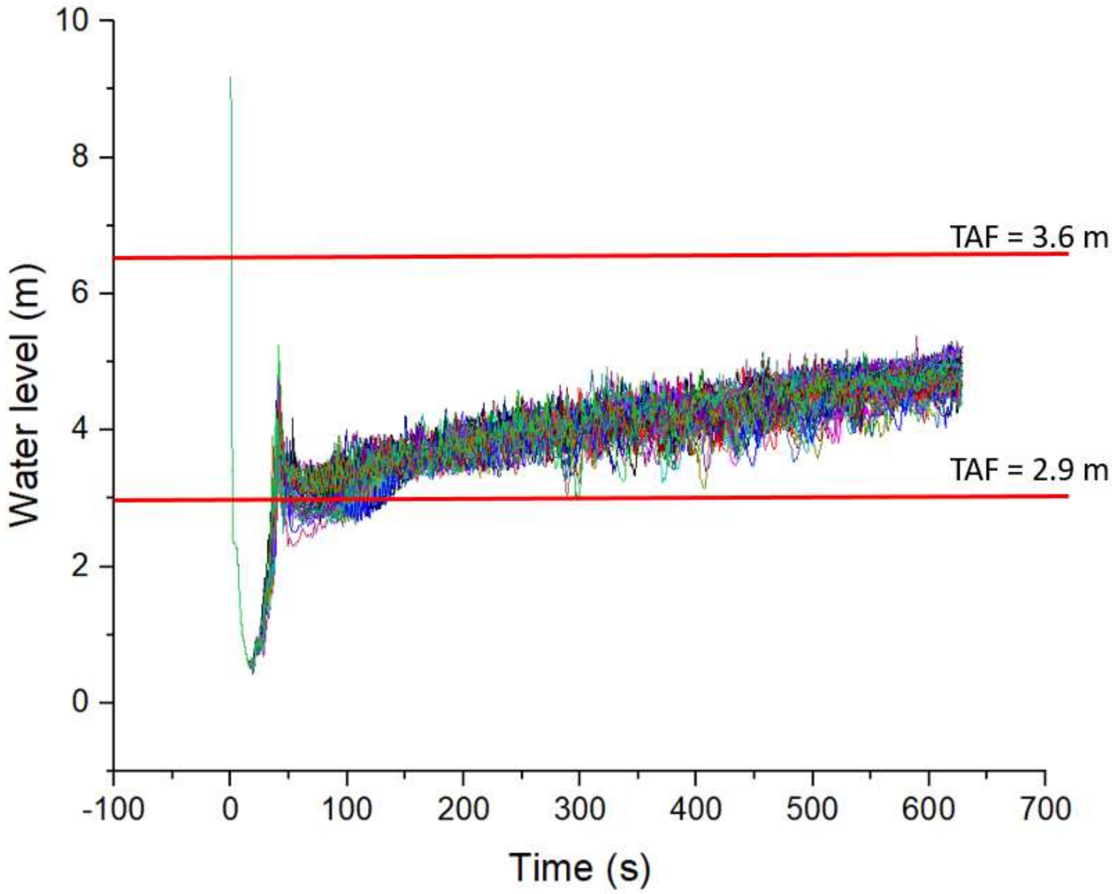

Figure 28 Core Collapsed Water Level of 59 Cases

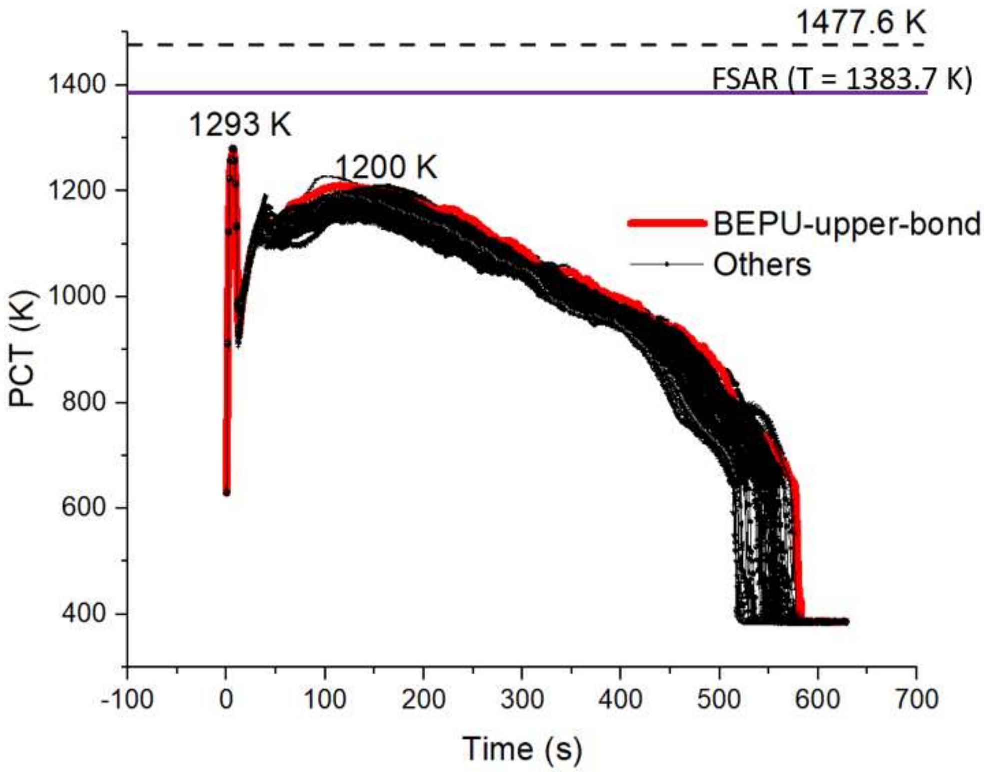

Figure 29 PCT of 59 Cases

6 CONCLUSIONS

This study presents an uncertainty analysis methodology of a postulated LBLOCA of Maanshan NPP using RELAP5/MOD3.3, DAKOTA, and SNAP codes. The results in base case analysis shows that the model of Maanshan NPP is able to manage a LBLOCA simulation. The phenomenon of the base case is consistent with the LOFT experiment and the analysis results of Westinghouse [13, 14]. The evaluation of quench front velocity during reflooding phase strongly affects the PCT. The performance of ECCS may affect reflooding phase. Therefore, an uncertainty analysis was performed in this study.

For the uncertainty analysis, most of the uncertainty parameters are related to ECCS property. The ranges of uncertainty analysis are based on FSAR and Training Manual of Maanshan NPP. The maximum PCT of the analysis is 1293 K. This result includes the uncertainty effect. The PCT of Maanshan FSAR is 1383.7 K. Therefore, the prediction of the uncertainty analysis is below the FSAR data. The difference of analysis results and FSAR data may be caused by the different calculation procedures, phenomenological modeling, and nodalization, etc. In addition, the PCT criteria of 10 CFR 50.46 is 1477.6 K (2200°F). Hence, the result of the uncertainty analysis is also below the criteria of 10 CFR 50.46. This indicates that Maanshan NPP in LBLOCA transient is at a safety situation when ECCS is performed.

7 REFERENCES

1. Taiwan Power Company, Final Safety Analysis Report for Maanshan Nuclear Power Station Units 1&2 (FSAR), Taiwan Power Company, Republic of China (Taiwan), 1983.

2. Jong-Rong Wang, Che-Hao Chen, Hao-Tzu Lin, Chunkuan Shih, Assessment of LONF ATWS for Maanshan PWR Using TRACE Code, NUREG/IA-0436, 2014.

3. Hao-Tzu Lin, Jong-Rong Wang, Kai-Chun Huang, Chunkuan Shih, Show-Chyuan Chiang, Chia-Chuan Liu, Station Blackout Mitigation Strategies Analysis for Maanshan PWR Plant using TRACE, Annals of Nuclear Energy, 89, 2016.

4. Information Systems Laboratories, Inc., RELAP5/MOD3.3 Code Manual Volume I: Code Structure, System Models, and Solution Methods, Information Systems Laboratories, Inc., Rockville Maryland, 2010.

5. Adams, B. M., et al., DAKOTA, a Multilevel Parallel Object-oriented Framework for Design Optimization, Parameter Estimation, Uncertainty Quantification, and Sensitivity Analysis, Sandia National Laboratories, 20-21, 66-67, 99, 2011.

6. Applied Programming Technology, Symbolic Nuclear Analysis Package (SNAP) User's Manual, Applied Programming Technology Inc., Bloomsburg, 2007.

7. Chunkuan Shih, Jong-Rong Wang, Shao-Wen Chen, Hao-Chun Chang, Show-Chyuan Chiang, and Tzu-Yao Yu, RELAP5/MOD3.3 Model Assessment of Maanshan Nuclear Power Plant with SNAP Interface, NUREG/IA-0472, 2017.

8. A.D. Siegel, Rewetting Temperature during Top and Bottom Reflood, M.S. Thesis, University of Illinois, 1979.

9. A.D. Siegel and J.J. Carbajo, Trans. ANS 35, pp. 327-328, 19.

10. Kevin D. Kimball, Quench Front Propagation during Bottom Reflooding of a Heated Annular Channel, Nuclear Engineering and Design 76, pp. 79-88, 1983.

11. Duffey, R.B., Porthouse, D.T.C., The Physics of Rewetting in Water Reactor Emergency Core Cooling. Nuclear Engineering and Design 25, pp. 379~394, 1973.

12. B.D.G. Piggott and D.T.C. Porthouse, A Correlation of Rewetting Data. Nuclear Engineering and Design 32, pp. 171~181, 1975.

13. S. M. Modro, S. M. Modro, V. T. Berta, A. B. Wahba, Review of LOFT Large Break Experiments, NUREG/IA-0028, 1989.

14. Westinghouse Company, Best-Estimate Analysis of the Large-Break Loss-of-Coolant Accident for Maanshan Units 1 and 2 Nuclear Power Plant Using the ASTRUM Methodology, WCAP-17054-P, 2009.

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10. SUPPLEMENTARY NOTES
K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The objective of this study is to assess the applicability of the RELAP5/MOD3.3 model of Maanshan NPP on LBLOCA transient. Maanshan NPP was the first three-loop PWR in Taiwan constructed by Westinghouse. In 2015, the RELAP5/MOD3.3 model of Maanshan NPP was developed with SNAP interface. An uncertainty analysis methodology of the postulated LBLOCA for Maanshan NPP using RELAP5/MOD3.3, DAKOTA, and SNAP codes are established in this study. The uncertainty parameters are related to ECCS property. The ranges of uncertainty analysis are based on FSAR and Training Manual of Maanshan. The maximum PCT of the uncertainty analysis is 1293 K. The PCT of FSAR is 1383.7 K. Hence, the result of the uncertainty analysis is below the FSAR data. The difference of uncertainty analysis results and FSAR data may be caused by the different calculation procedures, phenomenological modeling, and nodalization, etc. Additionally, the PCT criteria of 10 CFR 50.46 is 1477.6 K (2200°F). The prediction of the uncertainty analysis is below the criteria of 10 CFR 50.46. This implies that Maanshan NPP in LBLOCA transient is at a safety situation when ECCS is performed.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	
Probability Distribution Function (PDF)	13. AVAILABILITY STATEMENT unlimited
Feedwater Pump(s) Trip (FWPT)	14. SECURITY CLASSIFICATION
Complete Loss of Flow (CLOF)	*(This Page)* unclassified
High Head Safety Injection (HHSI)	*(This Report)* unclassified
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