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International Agreement Report

Maanshan PWR FLEX Program Enhance with RELAP5/MOD 3.3

Prepared by:
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ABSTRACT

Taiwan Power Company (TPC) and Atomic Energy Commission (AEC) have signed the CAMP (Code Applications and Maintenance Program) agreements with US NRC (U.S. Nuclear Regulatory Commission) from 2004. Under these agreements, TPC and AEC agree to investigate RELAP5 and TRACE codes for their applications and maintenance and publish NUREG/IA reports following US NRC's specifications. CAMP Program was initiated by the U.S. Nuclear Regulatory Commission in order to ensure the safe operation of nuclear power plants and to promote information exchange, cooperation and experience feedback in the international nuclear energy community. In addition to the continuous establishment and development of safety-related experiments and simulation programs, It is also actively seeking for relevant international experience sharing and feedback implementation. Therefore, it is necessary for our country to continue to participate in the CAMP cooperation agreement. Through international cooperation programs and domestic research analysis and evaluation needs, the results will be written and published NUREG reports and information and experience with international organizations communicate. For domestic energy policy, the proposal is for the supporting nuclear power plant operation safety and decommissioning plan. In this study, RELAP code will apply to assess the nuclear power plants safety, include domestic nuclear power plants accident assessment, verification for TRACE analysis, and assessment of FLEX capability for Maanshan nuclear power plant. Further, this proposal will also prepare one NUREG report every year on SNAP/RELAP5 [2] code application and maintenance based on CAMP agreements. In the next four years, the RELAP5 program will be used for analysis, and this report is expected to complete the establishment of the FLEX model of SNAP/RELAP5 in the Maanshan nuclear power plant, supplemented by TRACE analysis for verification, and at the same time include the comparison of relevant analysis reports commissioned by the Nuclear Energy Research Institute to assist Taipower in the rescue of FLEX equipment in the Maanshan nuclear power plant evaluation etc.

FOREWORD

RELAP5 is a thermal hydraulic analysis code and has been designed to perform best-estimate analysis of LOCA, operational transients, and other accident scenarios for nuclear power plants. Traditionally, RELAP5 models were developed by ASCII files, which was not intelligible for the beginners of computer analysis. A graphic input interface code-SNAP is developed by Applied Programming Technology Inc. and can process the establishment of the RELAP5 models more conveniently.

Taiwan and the United States have signed an agreement on CAMP to obtain the authorization of these codes. NTHU is the organization in Taiwan responsible for the application RELAP5 and SNAP in safety analysis of nuclear power plants. Hence, the RELAP5/SNAP model of Maanshan PWR nuclear power plant has been developed. To expand the applicability of the RELAP5/SNAP model, a thermal hydraulic analysis methodology of the postulated MSLB is established in our previous study. By comparing the RELAP5 results and FSAR [1] data, it indicates that the RELAP5/SNAP model has a respectable accuracy. Hence, this model was used to analysis of Maanshan PWR FLEX program enhance.

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EXECUTIVE SUMMARY

RELAP5 which is MOD3.3 Patch05 code was developed by Idaho National Engineering Laboratory for light water reactor transient analysis. RELAP5 can simulate the operation of NPPs under normal operations and transients, provide an accurate and rapid analysis patterns for NPP systems, and provide transients analysis results to NPPs and regulatory commission. RELAP5/MOD3.3 code is featured with nonhomogeneous and non-equilibrium model for the two-phase system and is a one-dimensional thermal hydraulic analysis code which uses Semi-Implicit method numerical scheme. RELAP5/MOD3.3 code also includes some models to deal with some particular phenomenon, such as critical flow model, reflooding model, metal-water reaction model etc.

SNAP which is developed by Applied Programming Technology, Inc. is a graphic interface code and different from the traditional input deck in ASCII files. SNAP can help users to easily build the RELAP5 models in a graphic interface. Furthermore, SNAP has the animation function to present RELAP5 analysis results. Hence, RELAP5/MOD3.3 and SNAP codes were used in this study.

After the Fukushima disaster in Japan, international attention has been increasingly focused on developing and formulating rescue strategies to counter extreme natural disasters and incidents related to power plants. In response to accidents exceeding design basis, which could significantly impact the safety of nuclear power plants by causing a prolonged loss of offsite power and ultimate heat sink failure, the Nuclear Energy Institute (NEI) in the United States devised a set of severe accident emergency response strategies known as Flexible and Diverse Coping Strategies (FLEX) to address such extreme events.

This study simulated conditions based on the input parameters referenced in the Westinghouse report WCAP-17601-P, including power, decay heat versions, and other necessary parameters. Key parameter information not available in the WCAP-17601-P report was supplemented using data from the National Atomic Research Institute's report "FLEX Case Timeline Analysis and Evaluation Report for Maanshan Nuclear Power Plant" and Tsinghua University's TRACE-related analysis model "Case Analysis Report of FLEX for Maanshan Nuclear Power Plant". After establishing the model, a comparison was made with the National Atomic Research Institute report to validate the credibility of the FLEX model in the SNAP/RELAP5 code.

Subsequently, advanced simulations of FLEX scenarios were conducted to evaluate different injection strategies for their cooling capabilities, ensuring sufficient cooling to maintain core coverage with water and prevent fuel exposure. Finally, by examining the time at which reflux cooling occurs, the study aimed to determine how much margin the power plant needs to recover its power equipment or utilize alternative power sources to restore plant conditions and ensure the plant remains in a safe state.

ABBREVIATIONS AND ACRONYMS

FLEX	Diverse and Flexible Coping Strategies
ACC	Accumulator
CAMP	Code Applications and Maintenance Program
ECCS	Emergency Core Cooling System
FSAR	Final Safety Analysis Report
GPM	Gallon Per Minute
MSIV	Main Steam Isolation Valves
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NTHU	National Tsing Hua University
PWR	Pressurized Light Water Reactor
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SG	Steam Generator
SI	Safety Injection
SNAP	Symbolic Nuclear Analysis Program
TAF	Top Active Fuel

1 INTRODUCTION

The flood caused by the tsunami caused by the 311 Fukushima accident submerged the emergency power supply and transmission system of the Fukushima Daiichi Nuclear Power Plant, making the nuclear power plant in a long-term AC power outage (Extended Loss of Alternating Current Power, ELAP), which eventually led to the loss of the final heat sink LUHS (Loss of Ultimate Heat Sink) severely compromised the integrity of the core cooling system and containment vessel, resulting in damage to the cores of all three reactors. Therefore, various nuclear energy-using countries have launched research on rescue measures beyond the design basis, including the specific major accident strategy guideline SMI (Specific Major Incident, SMI) proposed by Taiwan Power Company.

In addition, the Nuclear Energy Institute (NEI) has also developed a set of diversified and flexible Coping Strategies (FLEX) to establish and analyze the basis for different power plants and strengthen their resilience. The strategic goal is to integrate the existing equipment in the plant, contingency rules and personnel training measures to support the power plant in responding to Beyond Design Basis External Events (BDBEE) and reduce the risk of damage to the power plant.

The U.S. Nuclear Regulatory Commission (U.S. NRC) has continued to carry out safety-related research and simulation program establishment in order to ensure the safety considerations of nuclear power plants during operation, and to promote international information exchange and cooperation in the use of nuclear energy and relevant experience feedback. In addition to development, at the same time seek international experience sharing and feedback so that it can be truly implemented. TAIWAN National Atomic Research Institute and Taipower Corporation signed a new cooperation agreement (Code Application and Maintenance Program with the U.S. Nuclear Regulatory Commission (U.S. NRC) in early 2011 through Taiwan's Taipei Economic and Cultural Representative Office in the United States and the American Institute in Taiwan CAMP) for the application and maintenance of nuclear power plant heat flow programs (RELAP5 and TRACE), to support the safe operation of Taipower's nuclear power plant as the main direction of analysis, and use RELAP5 and new programs to use graphical window interfaces during the planning period SNAP for analysis.

This program uses the thermal-hydraulic assessment program RELAP5 (Reactor Excursion and Leak Analysis Program) approved by the U.S. Nuclear Regulatory Commission (U.S. NRC) for analysis, combined with the advanced graphical user interface program SNAP (Symbolic Nuclear Analysis Package), to establish the Maanshan Power Plant SNAP/ RELAP5 analysis model.

The research direction of this project is to support the safety and operation of Taipower Nuclear Power Plants. According to the WCAP-17601-P [3] reporting conditions proposed by Westinghouse, RELAP5 and the new program use the graphical window interface SNAP, referring to "Maanshan Nuclear Power Plant FLEX Case Timing Analysis and Evaluation Report [4]" which was approved by Nuclear Energy Research in April 2019 The delegate sets the initial conditions and transient actions. By using the data of the reports [5], the SNAP/RELAP5 analysis model of Maanshan power plant is established and verified.

This SNAP/RELAP5 analysis model can be used to support the safety analysis and research of Maanshan power plant."

2 MODEL AND METHODOLOGY DESCRIPTION

2.1 Maansan RELAP5/SNAP Model Description

Figure 2-1 and Figure 2-2 shows the RELAP5/SNAP model of Maanshan NPP. This model was established in our previous study [4]. This model can be divided into primary side loop, secondary side loop, and ECCS and established by using Pipe, Valve, Branch, Pump and Single Volume, and Time Dependent Junction components. There are three loops in this model. Every loop has a RCP and SG. A pressurizer which can adjust the pressure of RCS with the spray valves connects to the hot leg in the second loop. Some Branch components and heat structure components were used to simulate the reactor vessel and fuels channels. Table2-1 presents the initial conditions of the model for FLEX transient. In addition, the sequence of FLEX transient is shown in Table 2-2.

The rated thermal power of the original reactor of Maanshan Nuclear Power Plant was 2775 MWt. From 1997 to 1998, the Unit 1 and Unit 2 of Maanshan nuclear power plant carried out a Measurement Uncertainty Recapture (MUR) power uprates to increase the maximum design thermal power of the reactor of the Maanshan nuclear power plant. Increased to 2822MWt, the steam supply system is a three-loop pressurized water reactor, which belongs to the indirect loop.

In order to simplify the control of the feed water system, this research mode still uses Time Dependent Junction to simulate the switch of the feed water valve, which can effectively simplify the control system and only need to read the narrow range water level, steam flow and feed water flow, and then pass the corresponding Time Dependent Junction The Dependent Junction can be controlled, and the turbine drives the auxiliary feed water pump TDAFW (Turbine Driven Auxiliary Feedwater, TDAFW) flow setting refers to the version of the National Atomic Research Institute. Refer to Figure 2-3. This research is aimed at the parts required for FLEX rescue conditions, respectively in the secondary side three new medium-pressure water injection path has been added to each circuit, and the high-pressure water injection path in the original model has been used, and the flow and operating pressure limits have been given by referring to the conditions of the TRACE version model. Refer to Figure 2-4, Figure 2-5, and our previous model, the breach component is changed to a substitute valve for simulating shaft seal leakage.

In the part of controlling the decompression, refer to the parameters of the PORV in the original mode and the data of the research report of the National Atomic Research Institute, and use the TMDPJUN (Time Dependent Junction) component to establish the logic of the secondary side control decompression, and correct the decompression process Complies with WCAP-17601-P RCS 70°F/hr cooling rate specification, refer to Figure 2-6.

After the model is established, perform a 1000-second transient simulation steady state to confirm the stability and reliability of the SNAP/RELAP5 model. The verification results are shown in Table 2-3.

2.2 Analysis Methodology Description

The simulation conditions of this study are set with reference to the Westinghouse report WCAP-17601-P input related initial conditions, such as power, decay heat version and other necessary parameters, etc. Because the data of key parameters of the WCAP-17601-P report cannot be obtained, some data from the reports are used in this model. After the model is established, it will be compared with the report of the National Atomic Research Institute to verify the SNAP/RELAP5 model. After the SNAP/RELAP5 model verification, this model is used to evaluate the different water injection strategies to confirm the fuel covered by the water level of the reactor core.

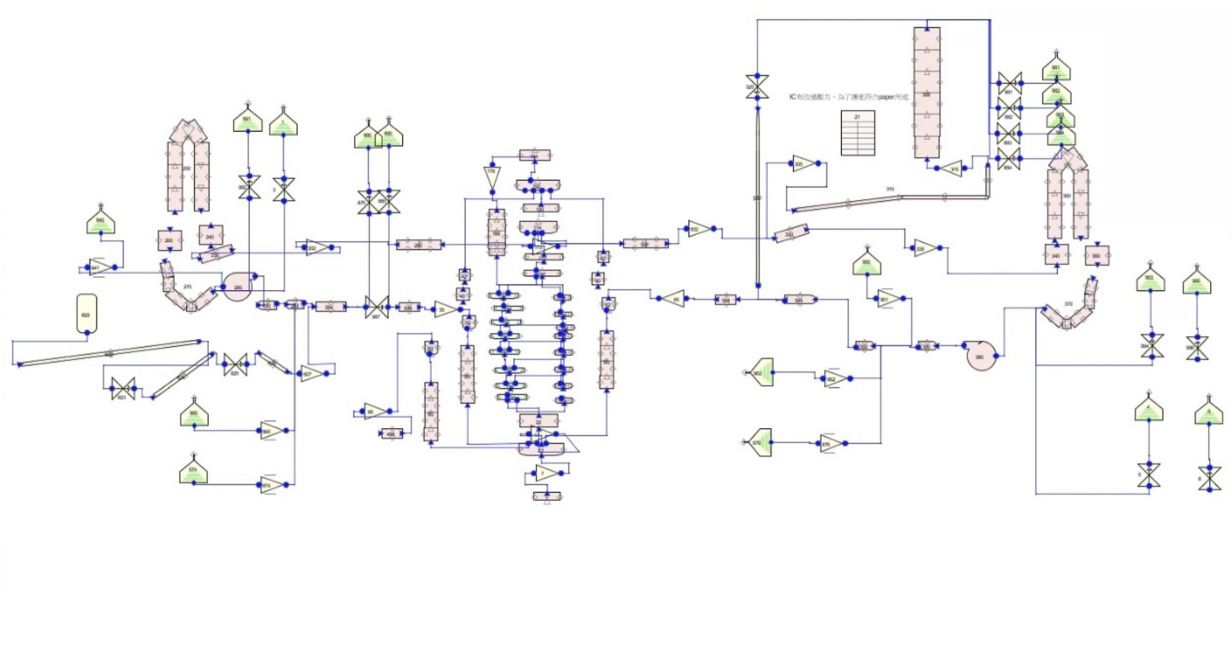


Figure 2-1 The RELAP5/SNAP Model of Maanshan NPP Primary Side

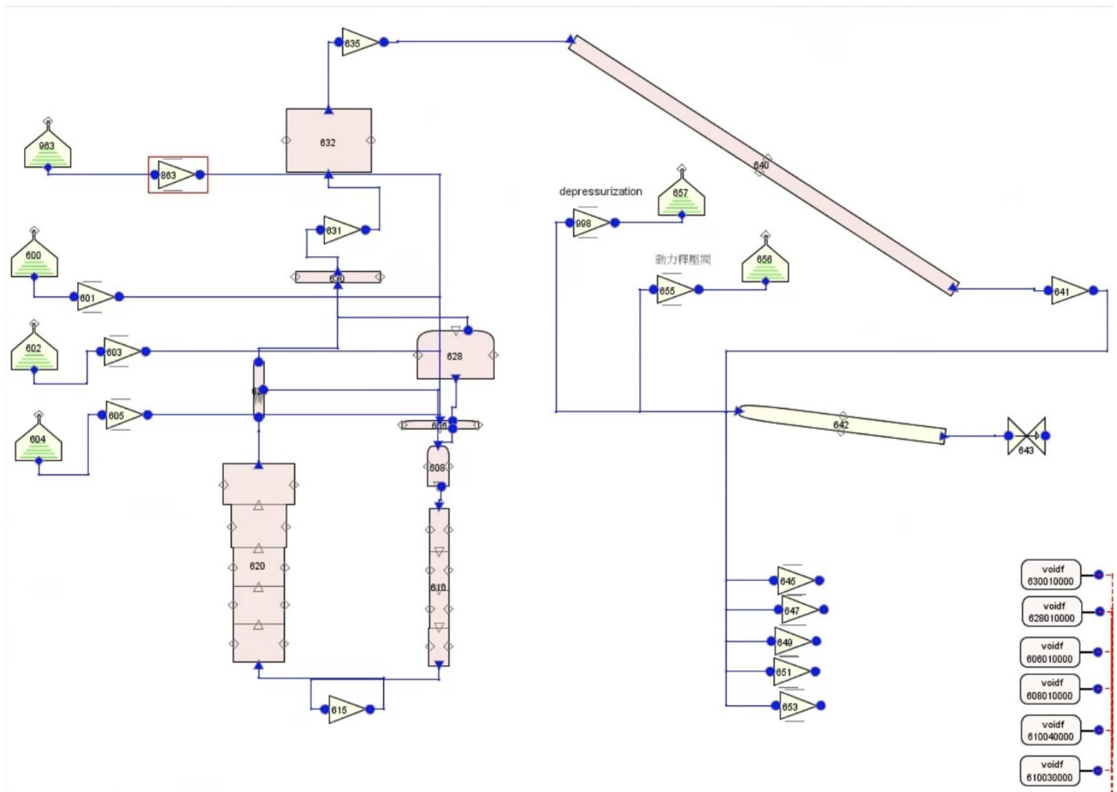


Figure 2-2 The RELAP5/SNAP Model of Maanshan NPP Secondary Side

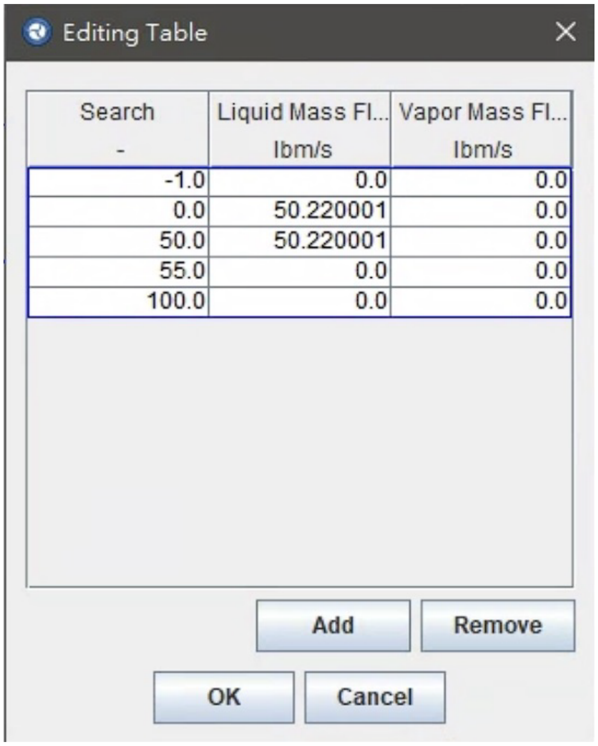


Figure 2-3 TDAFW Flow

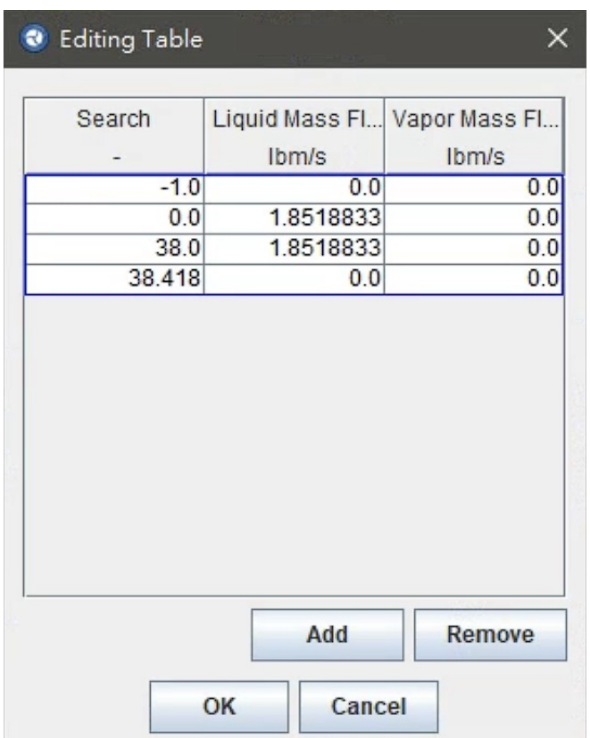
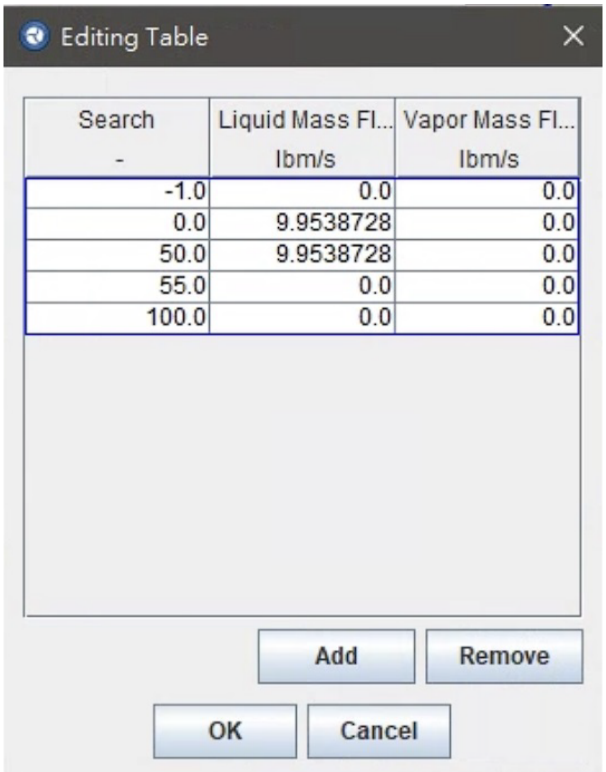


Figure 2-4 FLEX High Pressure Injection Flow



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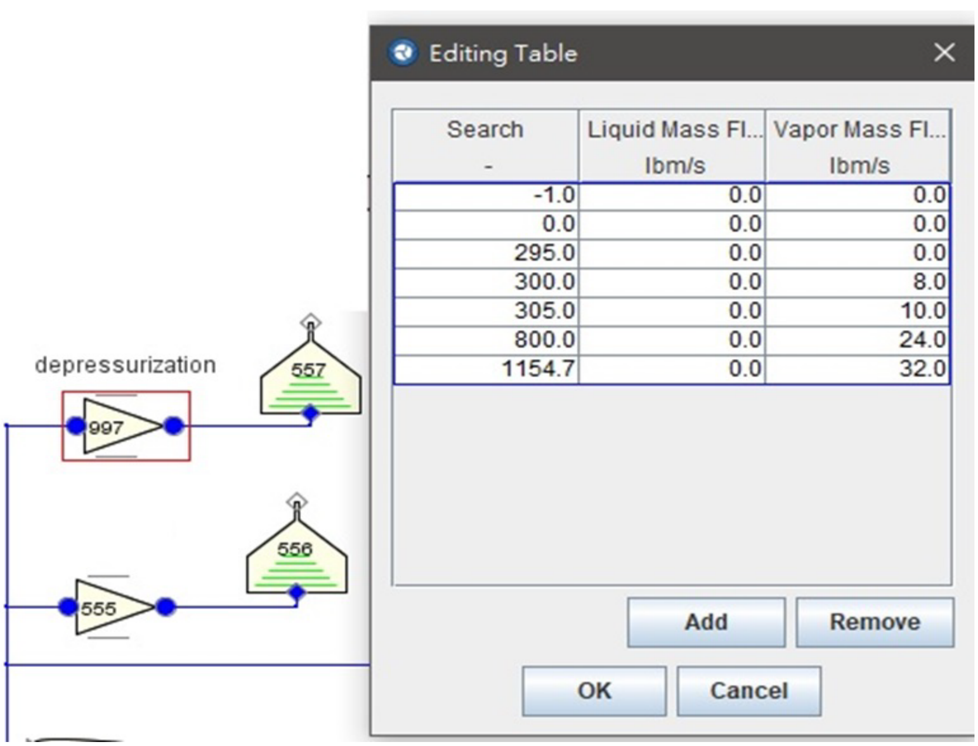
0.0	9.9538728	0.0
50.0	9.9538728	0.0
55.0	0.0	0.0
100.0	0.0	0.0

Figure 2-5 FLEX Medium Pressure Injection Flow



Search -	Liquid Mass Fl... lbm/s	Vapor Mass Fl... lbm/s
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0.0	0.0	0.0
295.0	0.0	0.0
300.0	0.0	8.0
305.0	0.0	10.0
800.0	0.0	24.0
1154.7	0.0	32.0

Figure 2-6 WCAP-17601-P Cooling Rate Specification

Table 2-1 Initial Conditions of Maanshan NPP

Parameter	SNAP/RELAP5
Core Power (MW)	2823
Core Average Temperature (°F)	587.2
RCP Flow (gpm)	223437
PZR Pressure (psia)	2262
Steam Generator Pressure(psia)	972
Steam Temperature (°F)	541
SG NR Water Level (%)	50.0
Main Steam Flow, total (Mlb/hr)	12.54
Feedwater Pressure (psia)	1100
Feedwater Temperature (°F)	440

Table 2-2 Sequence of Events in FLEX

Events	Time (sec)
Steady-state	0
SBO, Reactor scram & RCP seal (5gpm)	1000
Auxiliary feedwater injection	1060
RCP seal increase to 64gpm	1780
SG depressurization	8300
Accumulator injection	13100
SG pressure drop to 300 psia	18500
End of Accumulator injection	27300
Reflux cooling	62280
RCS water level lower than TAF	139600
Transient end	159939

Table 2-3 The Steady State Verification Results

Parameter	SNAP/RELAP5 transient 100 sec	SNAP/RELAP5transient 1000 sec(Case 0 sec)	Difference %
Core Power(MW)	2823	2823	0
Core Average Temperature (°F)	587.9	589.1	0.2
RCP Flow (gpm)	238891	238891	0
PZR Pressure (psia)	2276	2274	-0.08
Steam Generator Pressure(psia)	991	991	0
Steam Temperature (°F)	541	541	0
SG NR Water Level (%)	50.0	50	0
SG WR Water Level (%)	88.41	88.58	0.19
Main Steam Flow, total (Mlb/hr)	13.09	13.09	0
Feedwater Pressure (psia)	1100	1100	0
Feedwater Temperature (°F)	440	440	0

3 THE ANALYSIS RESULTS OF THE CASES

3.1 The Results of Maanshan RELAP5 Flex Model (Base Case)

This chapter supplements the missing key parameters in WCAP-17601-P through reports from NARI[4] and NTHU[5]. It also simulates Case A, Case B, Case 3, and Case 4 in the TRACE version that implements the FLEX operation. Since NARI uses the same assumptions, including initial conditions such as power, pressure, and the same decay heat settings, mutual comparisons are made to verify the accuracy of the SNAP/RELAP5 program of the FLEX model of the Maanshan nuclear power plant.

Case A is the benchmark case of this study. It is assumed that the power plant will be completely blacked out (SBO) when the external power is lost at the beginning. At this time, the reactor will immediately shut down safely, the main steam pipe isolation valve (MSIV) will be closed, and the main feed water pump and motor will drive the auxiliary power supply. The feedwater pumps all tripped, and steam was used to drive the auxiliary feedwater pump to replenish water to the steam generator until the end of the simulation time. During the simulation of the case, it was assumed that the RCP leaked, each loop was 5 gpm, and the system leaked 1 gpm due to unknown reasons, total 16 gpm, and the seal will fail due to high temperature after the accident 780 seconds, and the seal leakage will increase to a total of 64 gpm.

Two hours after the accident, the secondary-side controlled depressurization was implemented, and the accumulators began to replenish water to the RCS until the secondary-side pressure dropped to 300 psia, and then the auxiliary feedwater continued to provide residual heat removal capacity until the natural circulation stopped, to observe Subsequent water level changes, transient action integration reference Table 3-1.

Table 3-1 Case A Sequence of Event for Maanshan FLEX Model

Event	WCAP-17601-P	NARI Case A sec(hr)	NTHU Case A sec(hr)
Notice	1. seal leakage will increase to a total of 64 gpm 2. ACC available 3. SG Depressurization *Real injection time delay by Narrow range water level		
ELAP occurs causing the following: ■ Reactor Trip ■ Turbine Trip ■ Loss of Charging, Letdown ■ Loss of Pressurizer Heaters ■ Loss of Main Feedwater (coast down in 5 sec) ■ RCP Trip ■ Lost Seal cooling	T = 0.00 (0.0)	T = 0.00 (0.0)	T = 1000 (0.27)
AFW Flow Begins to all SGs	T=60(0.017)	T=60(0.017)	T=1060(0.27)
RCP seal leakage increases to 21 gpm/seal package	T=780(0.217)	T=780(0.217)	T=1780(0.49)
Operators commence remote cooldown to approximate SG pressure of 300 psia	T=7200 (2.0)	T=7200 (2.0)	T=8300 (2.3)
Accumulators start to inject water to the RCS	T=12750(3.54)	T=12050(3.347)	T=13100(3.63)
Operator ceases cooldown, maintains SG pressure at current ~ 300 psia	T=14900 (4.13)	T=17900(4.97)	T=18500(5.13)
Accumulators isolate	T=177000 (47.2) Accumulators empty	T=28400(7.89)	T=27300(7.58)
Loop Natural Circulation ceases, Reflux cooling begins	T=120000 (33.3)	T=65520(18.2)	T=62280(17.3)
Core uncover Occurs	T=196800 (54.67)	T=129900(36.08)	T=139600(38.77)
End of time	T=203619 (56.56)	T=131850(36.63)	T=159939(44.42)

Case A refers to the WCAP-17601-P report. Two hours after the accident, the PORV on the secondary side is turned on to control the pressure drop to 300 psia and maintain stability, as shown in Figure 3-1 . As shown, while maintaining the cooling rate of the primary side (RCS) at 70°F per hour (70°F/hr), the results show that the pressure on the primary side dropped rapidly following the trend of the secondary side after the reactor tripped, and then it was close to and maintained At about 300 psia, as shown in Figure 3-2, the change in leakage of the shaft seal is also maintained at a stable leakage after the primary side pressure is stabilized, as shown in Figure 3-3.

The temperature result of hot-leg is shown in Figure 3-4. It indicates that the overpressure water in the RCS is changed to the saturated water until 119,250 seconds later. The water in the SG primary side was almost empty and the water level of the reactor core began to drop. There was a brief rise at about 13,000 seconds due to the water injection of the accumulators. Then, the water level continued to drop because of the leakage of the RCPs. 38.47 hours after the accident, the water level was lower than TAF (21.785ft), as shown in Figure 3-5. In the end, when the final heat sink was lost, the temperature of the hot channel part of the fuel assembly would rise sharply by more than 1500°F, causing the RELAP5 program to fail to continue the calculation and stop , as shown in Figure 3-6.

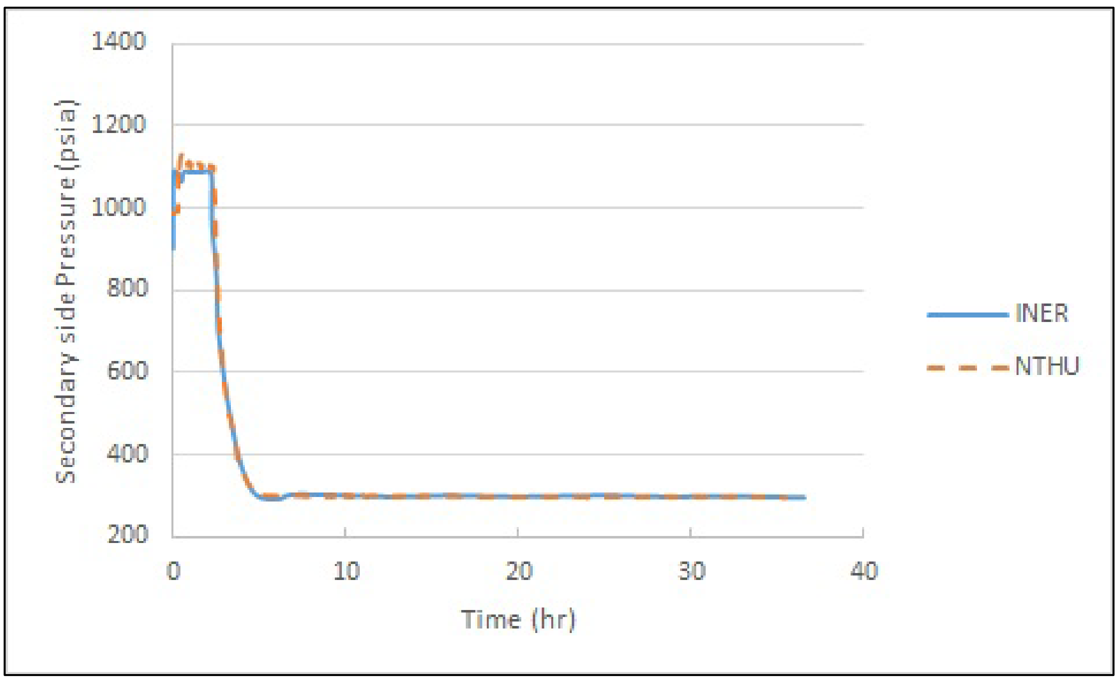

Figure 3-1 Case A Secondary Side Pressure

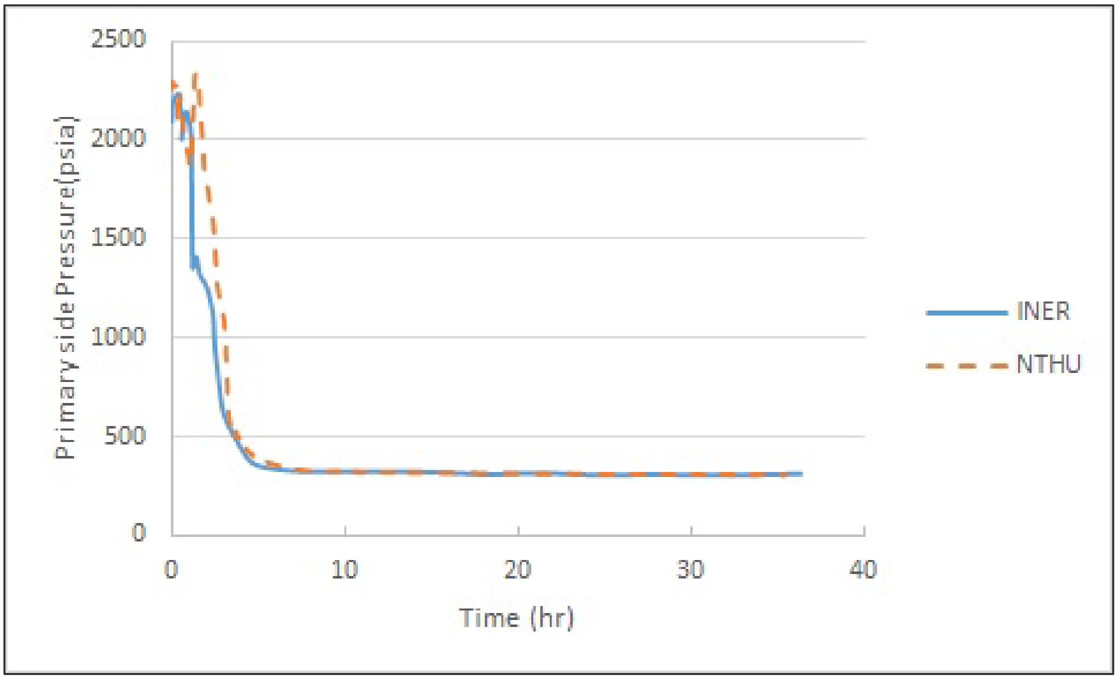

Figure 3-2 Case A Primary Side Pressure

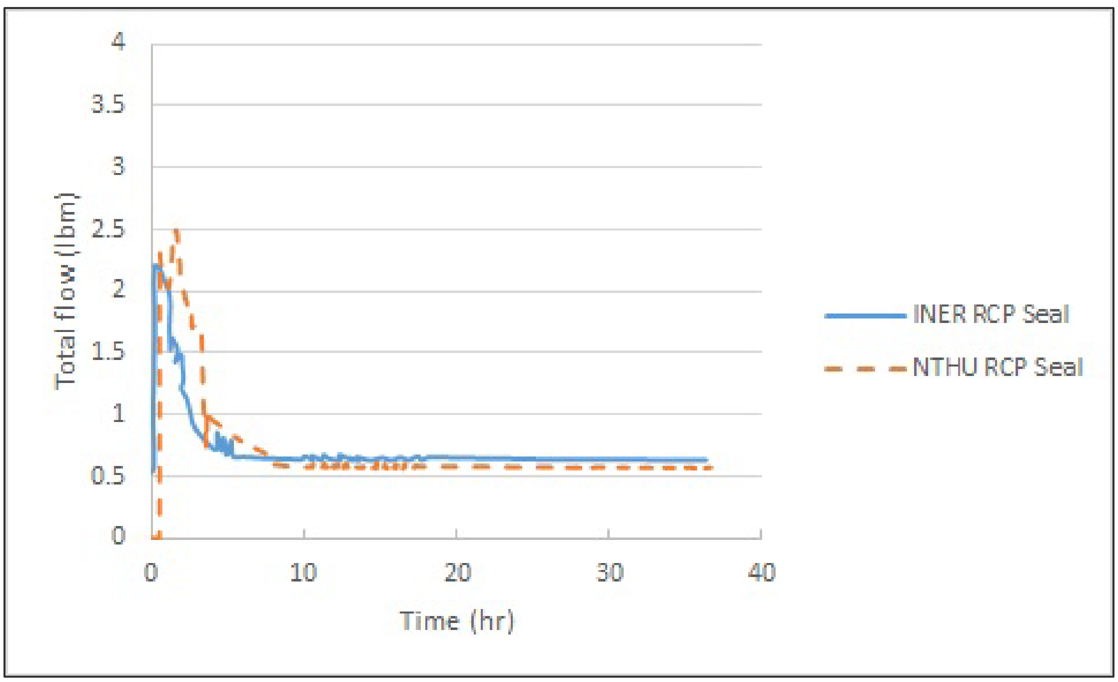

Figure 3 3 Case A RCP Seal Leakage

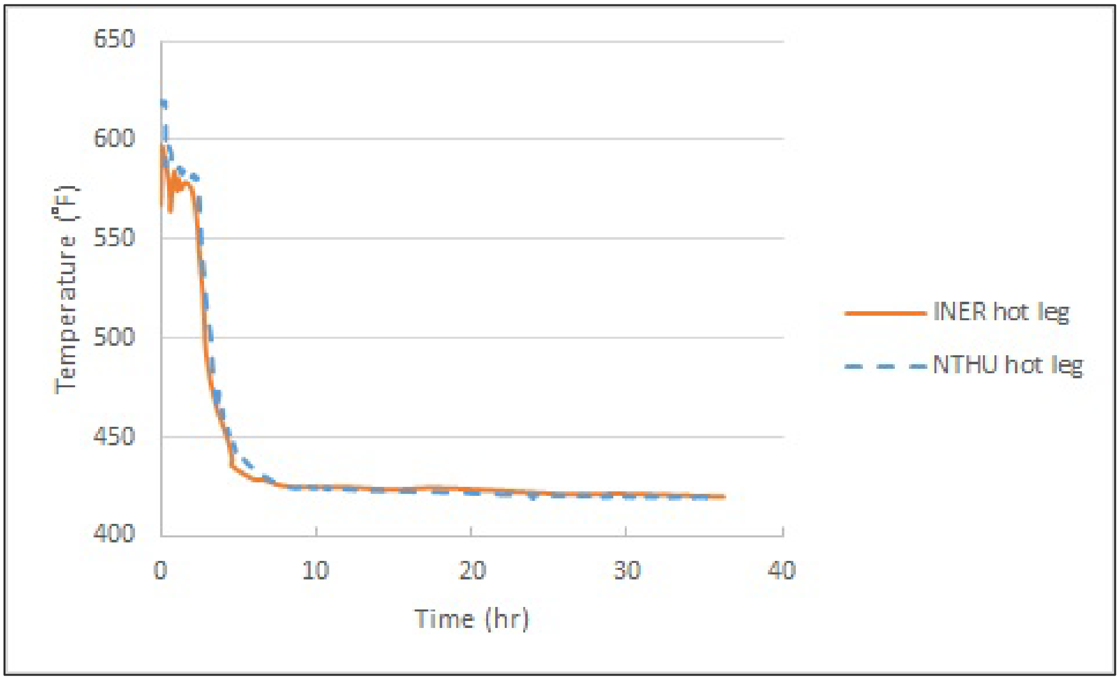

Figure 3-4   Case A Hot Leg Temperature

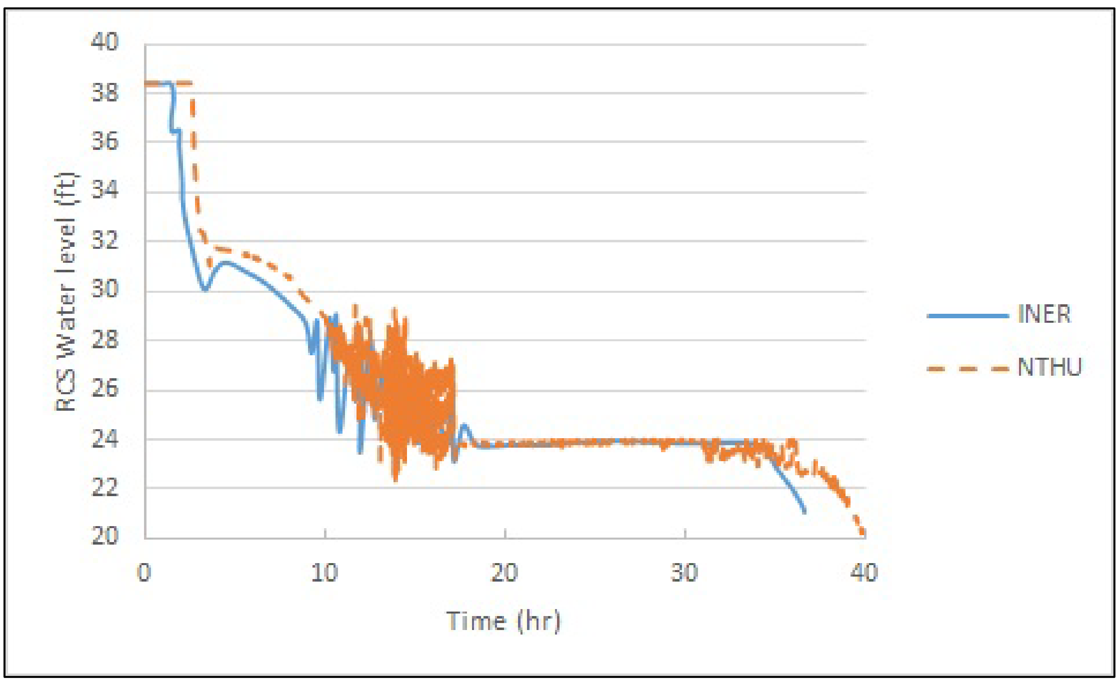

Figure 3-5   Case A RCS Water Level

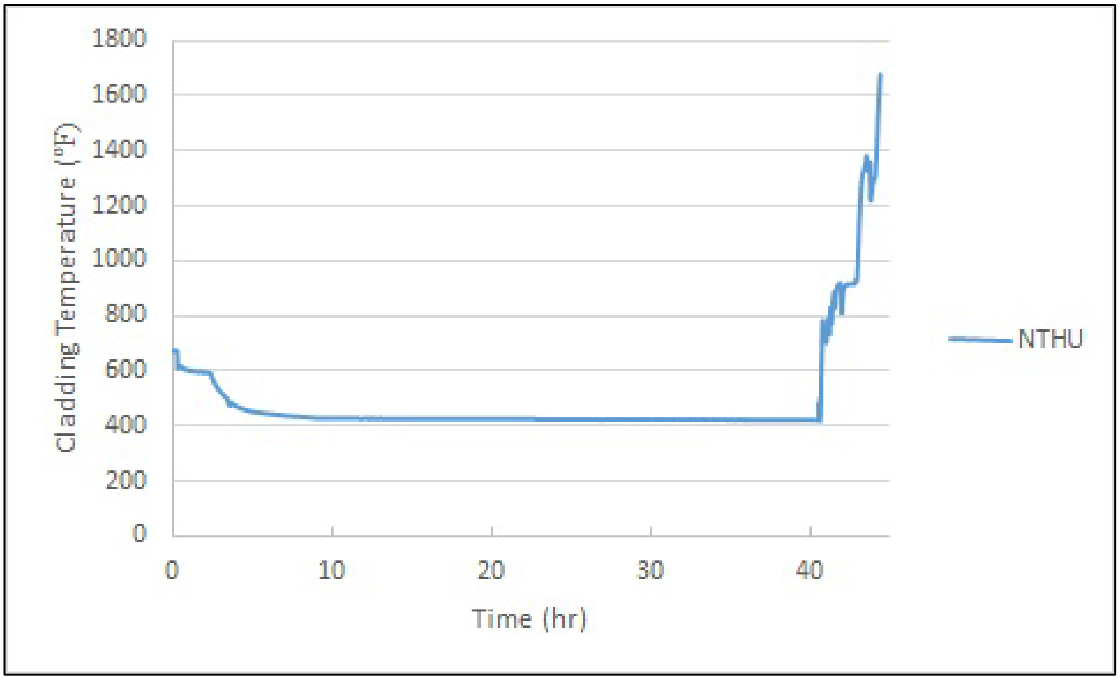

Figure 3-6 Case A Cladding Temperature

3.2 Comparison of RELAP5 and TRACE Analysis Results

This study also refers to the FLEX case simulated by the TRACE program "NNEERF-105-REP-20 Maanshan FLEX analysis", and imports its setting parameters to observe the differences in the results.

However, due to the earlier research time of the TRACE results (2016), some settings are different from the current situation of the power plant, such as the decay heat version and standards. Differences in differential pressure, differences in control and decompression logic, and shaft seal leakage, etc., are compared with the published version of the TRACE version, and the data are slightly simplified, so the comparison results are only used as a reference for model verification, sequence of event reference Table 3-2.

The shaft seal leakage is fixed at 21 gpm for each circuit, and another 1gpm system leaks for unknown reasons, a total of 64 gpm. The important transient action is similar to the aforementioned Case A of RELAP5, while in the FLEX rescue mode, the water injection of the medium pressure pump and the high pressure pump are used simultaneously.

In this way, water is injected into the primary side (1.8518833 lbm/s) and the secondary side (9.9538728 lbm/s) respectively, and their rescue capabilities are observed.

Case B is mainly divided into two stages. The first stage is 0~8 hours, using the steam turbine to drive the auxiliary feed water TDAFW to replenish water on the secondary side, and the accumulators ACC to replenish water to the primary side, and to control the pressure reduction on the secondary side; the second stage 8~80 hours to start FLEX rescue, the medium and high pressure equipment will replenish water to the primary and secondary sides respectively.

In contrast, the pressure result of RELAP5 is more stable than that of TRACE, as shown in Figure 3-7, and steam generator water level behaves very similarly, as shown in Figure 3-8, but the rising time point and stable state of the RCS water level between RELAP5 and TRACE are different, as shown in Figure 3-9, which leads to the influence of seal leakage, as shown in Figure 3-10.

Table 3-2 Case B Sequence of Event for RELAP5 and TRACE Comparison

Event	TRACE Case 3 Sec (hr)	SNAP/RELAP5 Sec (hr)
Notice	1. seal leakage will increase to a total of 64 gpm 2. ACC available 3. SG Depressurization *Real injection time delay by Narrow range water level	
ELAP occurs causing the following:	T=0~60(0~0.0167 hr)	T=0~1000 (0~0.27)
■ Reactor Trip ■ Turbine Trip ■ Loss of Main Feedwater (coastdown in 5sec) ■ Seal cooling lost and began leaking 21gpm/loop ■ SG Depressurization to 300 psia	T=60(0.0167hr)	T = 1000(0.27)
Accumulators start to inject water to the RCS	T=785(0.218hr)	T=1785(0.27)
FLEX High pressure pump injection to RCS TDAFW Off FLEX medium voltage start injection to S/G	T=28860(8.00hr)	T=29800(8.27)
Accumulators isolate	T=29448(8.18hr)	T=29400(8.17hr)
Core Uncover	NA	NA
End of time	T=288000(80.00hr)	T=298000(82.77)

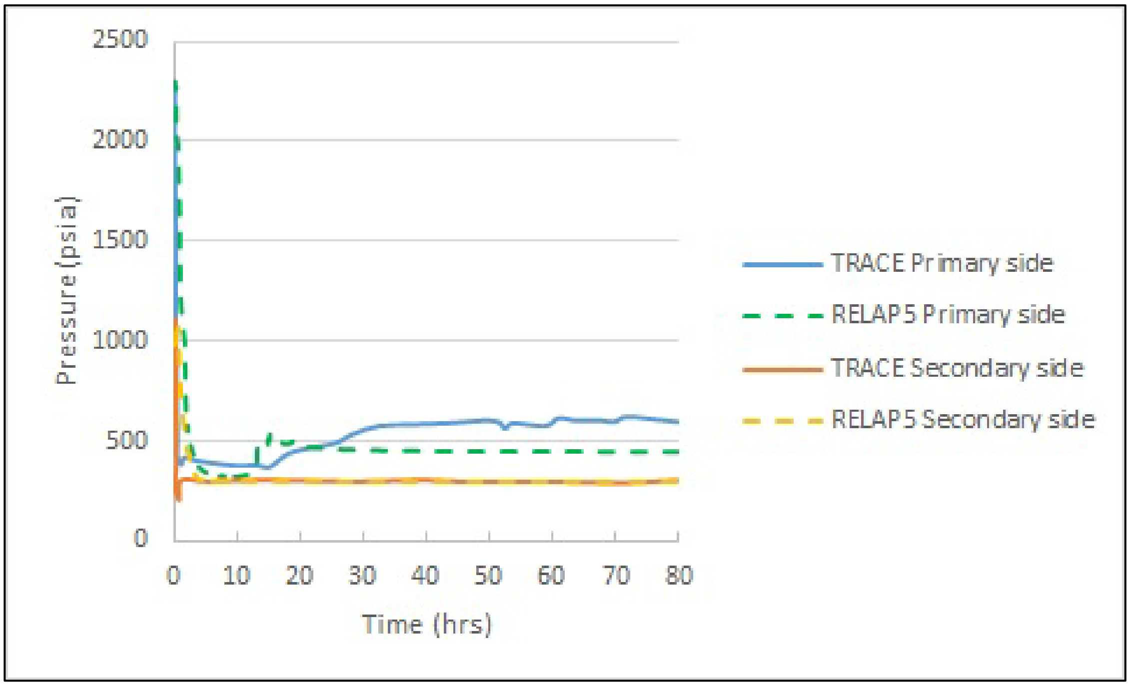

Figure 3-7 Case B Primary & Secondary Side Pressure

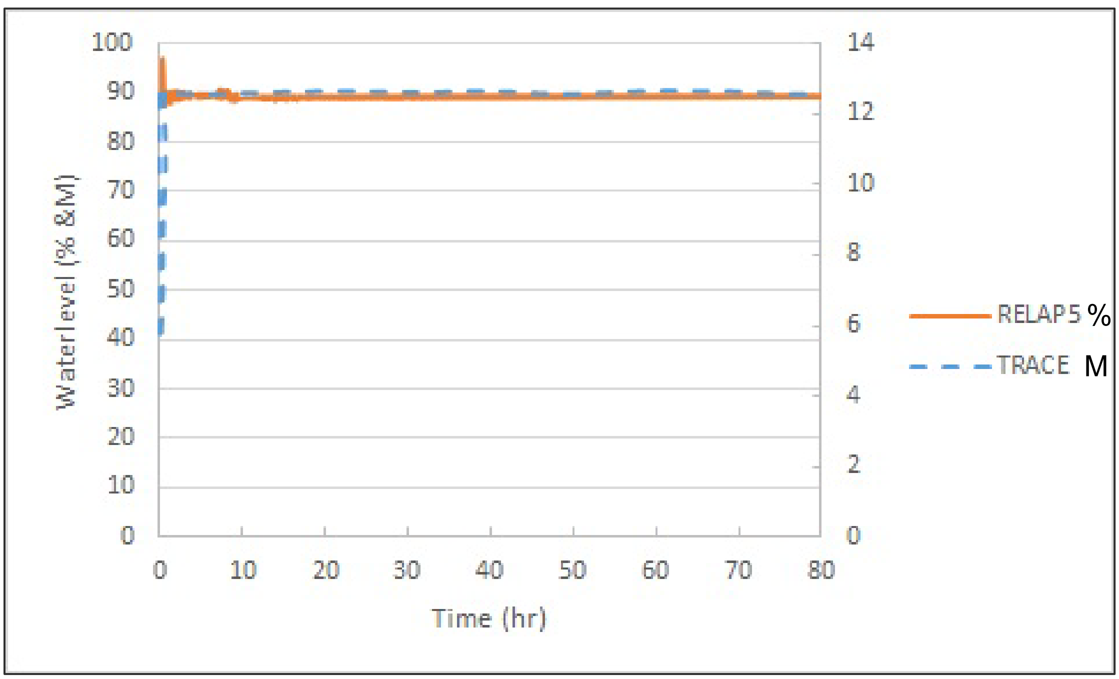

Figure 3-8 Case B SG Water Level

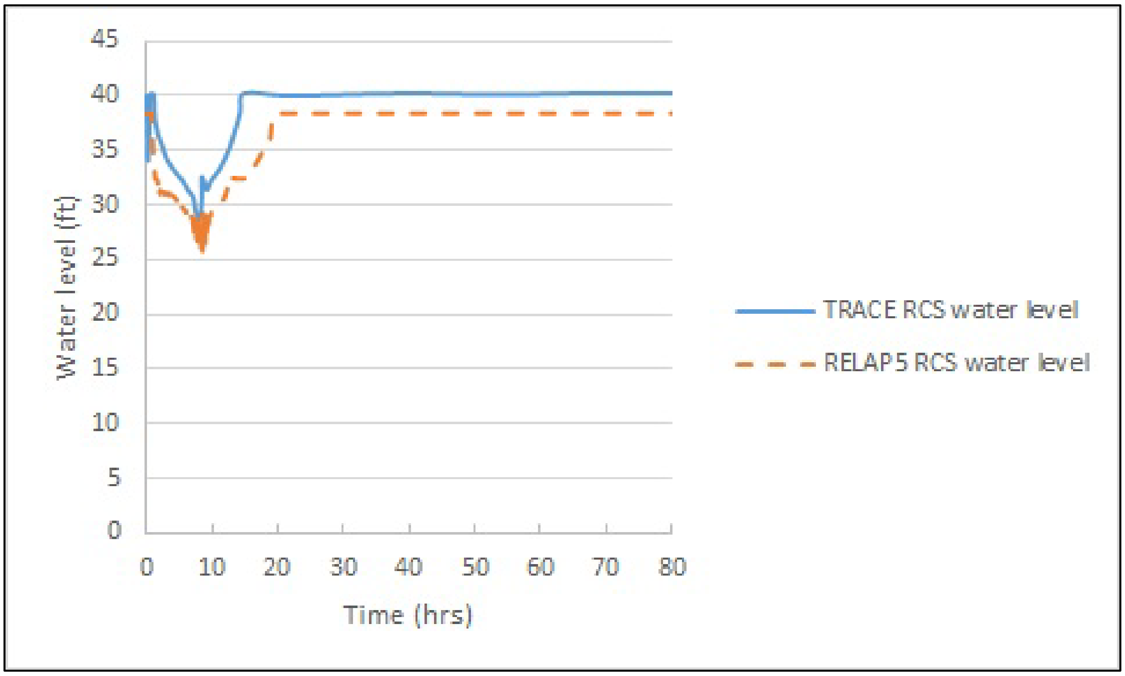

Figure 3-9 Case B RCS Water Level

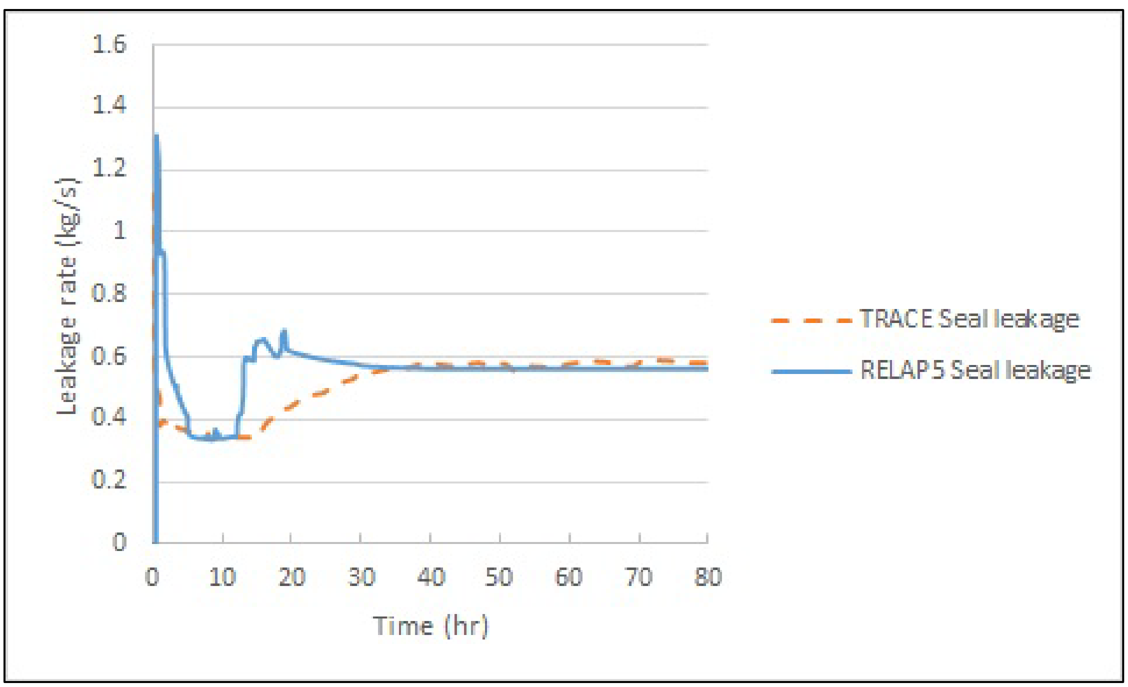

Figure 3-10 Case B RCP Seal Leakage

3.3 Maanshan RELAP5 FLEX Sensitivity Analysis

The water level of SNAP/RELAP5 Maanshan in Case A began to drop below TAF at around 40 hours, this section will discuss the FLEX rescue time point based on the current value of this time, and take the transient condition of Case A as the basic case to observe whether the safety of the power plant can still be guaranteed when rescue measures are delayed, sequence of Case C shown in Table 6 and the sensitivity analysis of FLEX injection time of cases shown in Table 3-3.

Table 3-3 Case C Sequence of Event

Event	FLEX Sensitivity Analysis sec (hr)
Notice	1. seal leakage will increase to a total of 64 gpm 2. ACC available 3. SG Depressurization *Real injection time delay by Narrow range water level
ELAP occurs causing the following: ■ Reactor Trip ■ Turbine Trip ■ Loss of Charging, Letdown ■ Loss of Pressurizer Heaters ■ Loss of Main Feedwater (coastdown in 5 sec) ■ RCP Trip ■ Lost Seal cooling	T = 1000 (0.27)
AFW Flow Begins to all SGs	T=1060(0.27)
RCP seal leakage increases to 21 gpm/seal package	T=1780(0.49)
Operators commence remote cooldown to approximate SG pressure of 300 psia	T=8300 (2.3)
Accumulators start to inject water to the RCS	T=13100(3.63)
SG depressurizes to 300 psia	T=18500(5.13)
Accumulator isolation	T=28800(8)
Loop Natural Circulation ceases, Reflux cooling begins	T=62280(17.3)
Core uncover	T=13700(38.05)
FLEX injection	T=137800(38.27)
RCS water level rise up to initial (38.14 ft)	T=207400(57.61)
End of Time	T=298400(82.88)

From the results of the reactor core temperature, although the core water levels in the four cases were all lower than TAF, as shown in Figure 3-11, the time in Case C-1 was too short to observe a significant increase in cladding temperature (see Figure 3-12). In Case C-2 and Case C-4, it can be found that because the water level of the reactor core has been lower than TAF for a long time, the cladding temperature of the fuel increases significantly, and the highest observed temperature is 903°F (757K) and 915°F (763K), as shown in Figure 3-12. Case C-3 only performs the water injection of the medium-pressure pump. Although the flow rate is smaller than TDAFW, it can continue to replenish water to the steam generator. However, without the help of high-pressure injection, the water level on the primary side cannot be maintained. As a result, the heat sink is lost and the water level in the reactor core drops. The cladding temperature will rise to over 1500°F at 150,200 (41.77 hours) after the accident, so that the RELAP5 program cannot continue to calculate and stops.

From the results of all advanced case, although the ELAP mode can support about 40 hours, in practice, it is still necessary to establish a long-term cooling mode as soon as possible, especially for the cooling capacity of the primary side. If the FLEX configuration cannot be completed within 40 hours, it may have an irreversible impact on the reactor core, the result of Case C reference Table 3-4.

Table 3-4 Case C Rescue Operation Schedule

Case/Event	FLEX High Pressure injection (hr)	FLEX Medium Voltage injection (hr)	Rescure result ○success ×fail
C-1	38	38	○
C-2	41	41	○
C-3	malfunction	38	×
C-4	41.5	41	○

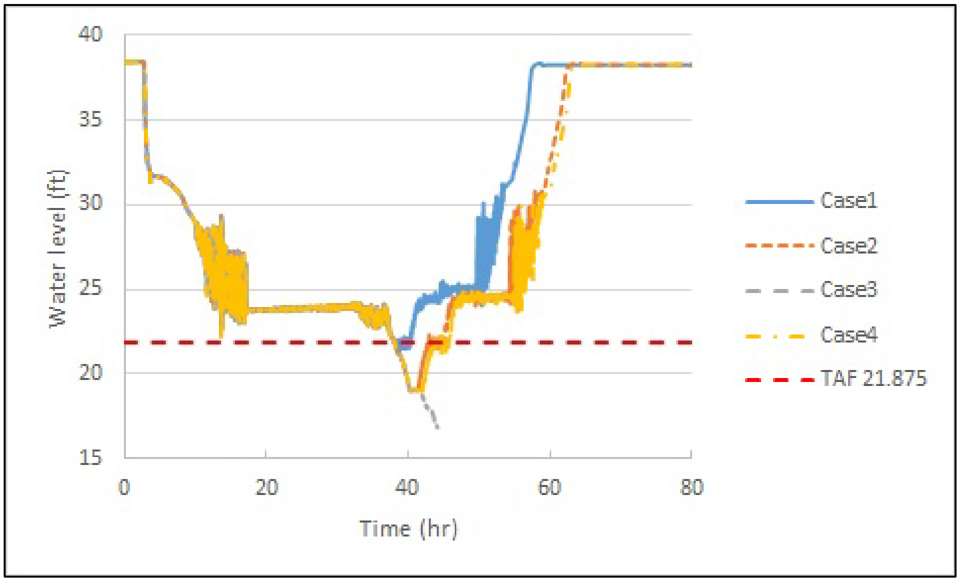

Figure 3-11 Core Water Level

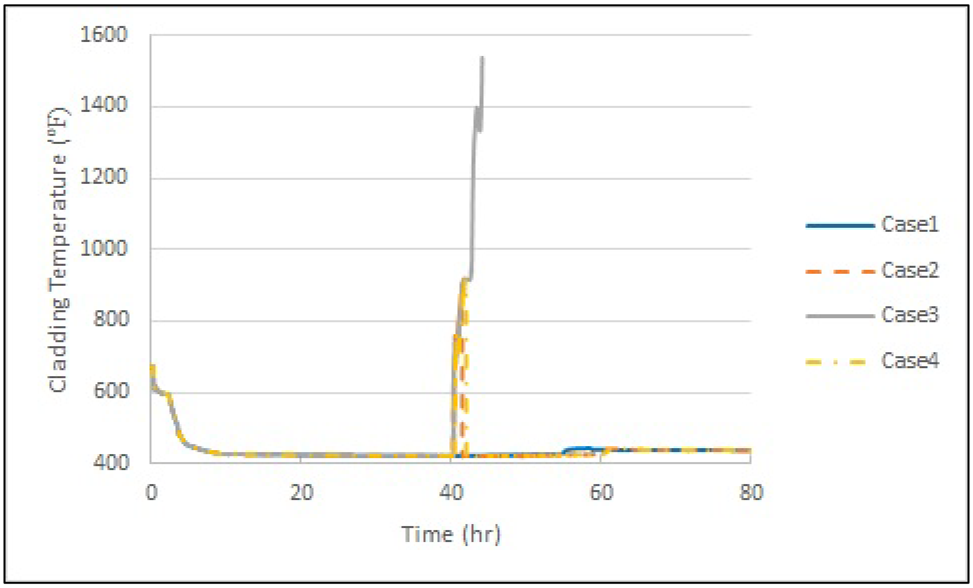

Figure 3-12 Cladding Temperature

3.4 Maanshan RELAP5 FLEX Strategy Enhancement

Since the instrument may not be able to accurately detect the occurrence of reflux cooling, this section will refer to the SOP(Standard Operating Procedures)-1460[6]of Maanshan plant and the WCAP-17792-P report to discuss the preventive methods of reflux cooling and the required measures or equipment to avoid the recritical occurrence of the reactor core. The definition of reflux cooling is shown in Table 3-5.

By refer to the data of the WCAP-17792-P, the Case D is performed by using RELAP5 and the sequence of Case D is shown in Table 3-6. According to Table 3-5 and the result of Case D, the figure that indicates the time of reflux cooling can be drawn. In this figure, the parameter of X-axis is the time that the water level of the core is below the top of the hot-leg (approximately 30.1135ft) and the parameter of Y-axis is the time that the flow quality of the top of SG is below 0.2.

When the core water level drops below the top of the Hot Legs (approximately 30.1135ft), the RVLIS is determined to be reached.
Top of SG Flow quality exceeds a value of 0.2.
Find out the time of reflux cooling from (1) and (2).

Table 3-5 Westinghouse Reflux Cooling Definition

Liquid phase flow through the loops is becoming discontinuous and the RCS is moving into a truly stratified state
Flow quality in the SG U-bend region is showing a rapid, monotonic increase
Flow quality exceeds a value of 0.2
One hour averaged liquid phase flows are in the range of 50 to 100 lbm/sec and decreasing

Table 3-6 Case D Reflux Cooling Sequence of Event

Event	Case D Reflun cooling
Notice	RCP seal leakage are 5gpm and 64gpm
ELAP occurs causing the following: ■ Reactor Trip ■ Turbine Trip ■ Loss of Charging, Letdown ■ Loss of Pressurizer Heaters ■ RCP Trip ■ Loss of Main Feedwater (coastdown in 5 sec) ■ Seal cooling lost ■ RCP seal leakage (5gpm & 64gpm)	T = 1000 (0.27)
AFW Flow Begins to all SGs	T=4660
End of time	T=288000(80)

The data of WCAP-17792-P is also shown in Figure 3-13. The trend of the Case D is similar to the data of WCAP-17792-P, Table 3-7 shows the results of Case D. In addition, the 5 gpm case is commonly used in the analysis and conditions of Maanshan nuclear power plant. According to Figure 3-13, the reflux cooling time did not reach the certification condition until 58 hours after the accident. It indicates that there are more time to deal with the preparation of the water injection pumps and other things in the Maanshan nuclear power plant for 5 gpm case.

Table 3-7 Case D Reflux Cooling Depressurization or Not Comparison

Time (hr)	Without depressurization	
Seal leakage rate	5gpm	64gpm
RCS water level drop to Hot Legs	58.430621	5.8056239
Quality value of 0.2	59.486115	7.0556331
End of Time	80	17.1

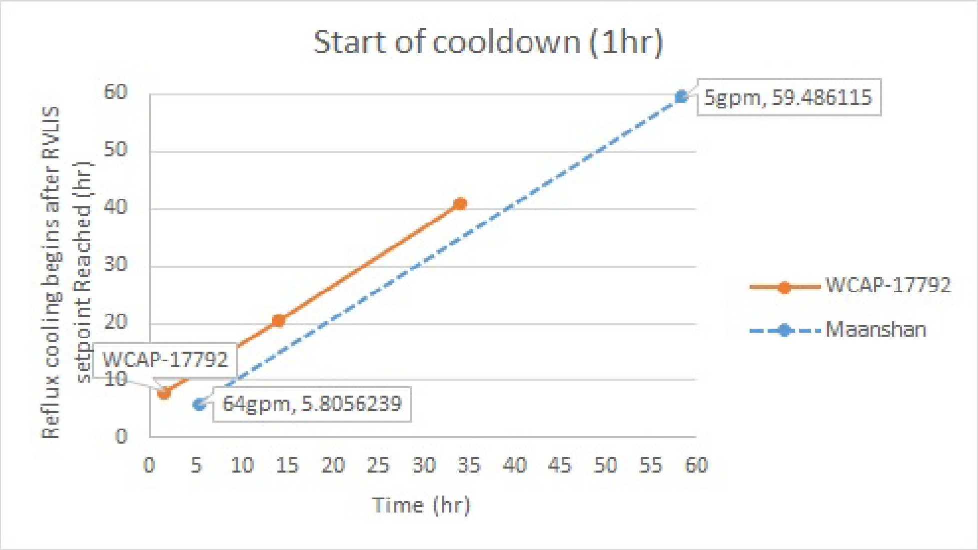

Figure 3-13 Case D Reflux Cooling Time

4 CONCLUSION

This study successfully established the FLEX mode of the RELAP5/MOD3.3 coupling graphical interface program SNAP, and imported the WCAP-17601-P report parameter data. By comparing the cases of the previous TRACE version and the National Atomic Research Institute version, it is verified that this mode can correctly simulate the basic actions of the FLEX case, and then continue to discuss whether the thermal-hydraulic phenomenon maintains a safety margin in the event of an accident.

5 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）, 2017.

[2] Chunkuan Shih, Jong-Rong Wang, Shao-Wen Chen, Hao-Chun Chang, NUREG/IA-0472, RELAP5/MOD3.3 Model Assessment of Maanshan Nuclear Power Plant with SNAP Interface.

[3] WCAP-17792-P Emergency Procedure Development Strategies for Extended Loss of AC Power Event for all
Domestic Pressurized Water Reactor Designs Revision 0-B December 2013.

[4] Maanshan Case Time Series Analysis Evaluation Report, NARI (April 2020).

[5] Maanshan FLEX Analysis Report, NTHU Center for Energy and Environmental Research (November 2016.

[6] Taiwan Power Company Maanshan Nuclear power plant program book 1460.1r0p2.

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K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

Taiwan Power Company (TPC) and Atomic Energy Commission (AEC) have signed the CAMP agreements with US NRC since 2004. Under these agreements, TPC and AEC agree to investigate RELAP5 and TRACE codes for their applications. In addition to the continuous establishment and development of safety-related experiments and simulation programs, It is also actively seeking for relevant international experience sharing and feedback implementation. The results will be written and published NUREG reports and information and experience with international organizations communicate. For domestic energy policy, the proposal is for the supporting nuclear power plant operation safety and decommissioning plan. In this study, RELAP code will apply to assess the nuclear power plants safety, include domestic nuclear power plants accident assessment, verification for TRACE analysis, and assessment of FLEX capability for Maanshan nuclear power plant. The RELAP5 program will be used for analysis of the FLEX model of SNAP/RELAP5 in the Maanshan nuclear power plant, supplemented by TRACE analysis for verification, and at the same time include the comparison of relevant analysis reports commissioned by the Nuclear Energy Research Institute to assist Taipower in the rescue of FLEX equipment in the Maanshan nuclear power plant evaluation etc.

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Maanshan NPP
FLEX Diverse and Flexible Coping Strategies
ECCS Emergency Core Cooling System
FSAR Final Safety Analysis Report
PWR Pressurized Water Reactor
RCS Reactor Coolant System
SBO Station Blackout
TAF Top Active Fuel
NTHU National Tsing Hua University

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