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International Agreement Report

Analysis of Main Steam Line Break Accident for 3-Loop PWR with RELAP5/MOD3.3 Code

Prepared by:
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ABSTRACT

In Taiwan, many prediction analysis for different purposes are investigated with different system codes, such as RELAP5 developed by US NRC. The main purpose of this study was to simulate Maanshan NPP PWR Main Steam Line break (MSLB) inside-containment by using RELAP5/MOD3.3, and then to discuss the results of this model by referring to Maanshan FSAR. Furthermore, sensitivity study of discharge coefficient is also included. In addition, the peak cladding temperature is maintaining under the criteria value in 10CFR50.46 and the results of this RELAP5/MOD3.3 model indicates reasonable thermal-hydraulic phenomena and presents good agreements with FSAR. The sensitivity analysis results show that the variations of break flow rate were governed by discharge coefficient. This study demonstrated that a methodology of RELAP5/MOD3.3 model for Maanshan PWR has been successfully developed, and the present simulation results and analysis can be used for safety analyses and further transient applications.

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 conveniently.

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 analysis methodology of the postulated MSLB is established in this study.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

RELAP5/MOD3.3 Patch05 code [1], which was developed for light water reactor transient analysis at Idaho National Engineering Laboratory for U.S. NRC, is applied in this research. This thermal hydraulic analysis code is provided transient accident analysis for nuclear power plants and regulatory commission, to provide an accurate and rapid analysis patterns for nuclear power plant systems, to simulate the operation of power plants under normal operation and transient accidents, and to formulate appropriate operational specifications. As well as performed to support rulemaking, licensing audit calculations, evaluation of accident, mitigation strategies, evaluation of operator guidelines, and experiment planning analysis. It can be developed to maintain nuclear power plant safety. RELAP5/MOD3.3 is featured with nonhomogeneous and non-equilibrium model for the two-phase system. RELAP5/MOD3.3 is a one-dimensional thermal hydraulic analysis code, which are used Semi-Implicit Method numerical scheme. This method used to enable fast calculation of system transients. It can produce accurate transient analysis results in relatively short time. It also includes several models to deal with some particular phenomenon, such as critical flow model, reflooding model, metal-water reaction model.

SNAP is an interface program of codes and is developed by U.S. NRC and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, SNAP allow every user to easily build the models of nuclear power plants in a graphic interface. Furthermore, SNAP has the animation function to present the analysis results. Therefore, the RELAP5/MOD3.3 with SNAP interface was used in this study.

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. The Maanshan NPP of reactor coolant system has three loops, each of which includes a reactor coolant pump and a steam generator. The pressurizer connects to the hot-leg piping in loop 2.

The MSLB is a kind of DBA analysis, and there are lots similar work has been done before by relevant researchers, but there is no relative research that using the simulated code such as RELAP5/MOD3.3 to analysis the MSLB for Maanshan nuclear power plant. In this research, a RELAP5/MOD3.3 model of Maanshan NPP for MSLB transient was developed. The object of this paper is to develop a complete flow chart for analyzing the nuclear system transient. In order to develop a complete methodology for analyzing the MSLB transient event, sensitivity analysis is added in this report. The results of RELAP5 calculation were compared with those of both FSAR, indicating that the RELAP5 model has the ability to predict the MSLB transient. And the final results show that the result integrity criteria are met.

ABBREVIATIONS AND ACRONYMS

ACC	Accumulator
CAMP	Code Applications and Maintenance Program
ECCS	Emergency Core Cooling System
FSAR	Final Safety Analysis Report
HHSI	High Head Safety Injection
kg	kilogram(s)
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
MPa	Megapascal(s)
MSIV	Main Steam Isolation Valves
MSLB	Main Steam Line Break
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRWL	Narrow Range Water Level
NSSS	Nuclear Steam Supply System
NTHU	National Tsing Hua University
PWR	Pressurized Light Water Reactor
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
RPV	Reactor Pressure Vessel
SG	Steam Generator
SI	Safety Injection
SNAP	Symbolic Nuclear Analysis Program
WRWL	Wide Range Water Level

1 INTRODUCTION

Maanshan NPP is a three-loop PWR located on the southern coast of Taiwan. Its Nuclear Steam Supply System (NSSS) is built by Westinghouse for Taiwan Power Company, which can be divided into primary loop and secondary loop. The designed total power of the NSSS is 2785MWt, which consist of 2775 MWt for reactor power and 10 MWt for cooling pumps. Now, the total power is upgraded to 2822 MWt. In the primary side, there is a Reactor Coolant Pump (RCP) and a Steam Generator (SG) in each loop. Pressurizer locates on the hot leg of the second loop, which can adjust the pressure of the Reactor Coolant System (RCS). Each loop equipped with an Accumulator (ACC) injection system, a High Head Safety Injection system (HHSI) and a Low Head Safety Injection system (LHSI). The designed core flow rate is $49.59*10^6$Kg/hr.

In the past, computer code for safety analysis were used conservative conduction. In some cases, conservative code could not simulate the overall transient accident, such as the Final Safety Analysis Report (FSAR) of Maanshan NPP [2], e.g. the transient event of LOCA. Different code is used at each stage. In addition, demonstration of safety margins is usually given by means of conservative codes that hide great margins without the possibility to quantify results and the result often deviated from the actual physical phenomenon. With the advancement of computer technology and the evolution of safety analysis code, most of the thermal hydraulic system can simulate the entire accident transient, and can simulate more physical phenomenon, the simulation results are also relatively accurate, and the heat transfer empirical formula in the code. It is no longer the most conservative case, but a more accurate estimation code that is closer to the facts.

In order to perform transient analysis for Maanshan NPP. Our group had developed significant that the data results could be compared with each other to ensure the consistency. The main purpose of this research is to develop a RELAP5/SNAP model of Maanshan NPP for MSLB transient. The results of RELAP5/MOD 3.3 calculation were compared with those of FSAR data, indicating that the RELAP5/MOD 3.3 model has the ability to predict the MSLB transient, and the criteria are met.

2 SIMULATION CODE AND MODEL ESTABLISHMENT

2.1 SNAP Interface

Symbolic Nuclear Analysis Package (SNAP) [3] is an interface of NPP analysis codes which developed by U.S. Nuclear Regulatory Commission (NRC) and Applied Programming Technology, Inc. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic components make the user find out the whole NPP and control system more efficiently. The SNAP offers users an environment which is a graphical user interface with pre-processor and post-processor capabilities. In addition, SNAP assists users in developing RELAP5 input decks and running the codes. RELAP5 is a thermal hydraulic code developed for safety analysis of NPPs by the US NRC. In summary, SNAP provides not only a graphical user interface with pre-processor and post-processor capabilities, but also a platform for users to develop RELAP5 models.

2.2 RELAP5/MOD3.3 Model Description

The Maanshan NPP shown in Figure 2-1 is established by Hydraulic Components, such as Pipe, Branch, Valve, Pump, Single-Volume, and so on. And it can be divided into Primary Loop, Secondary Loop, Reactor Pressure Vessel (RPV) and Emergency Core cooling System (ECCS). In this research, there are three loops in the Maanshan NPP. In each loop, there is a RCP and SG. On the hot leg of second loop, a pressurizer which can adjust the pressure of RCS with the spray valves. In this analysis model, there are several Branch components developed to simulate the reactor vessel. According to the core arrangement. Branch components simulate average fuel channel, hottest fuel channel and bypass flow channel. These channels will be connected to the heat structure components to obtain the heat and simulate the heat transfer.

Similar to primary side, the secondary side were developed with Pipe, Valve, Branch, Pump and Single Volume and Time Dependent Junction as shown in Figure 2-2. To simulate the feedwater, auxiliary feedwater and steam dump systems, which flow rate was determined by system feedback, the Time Dependent Junction was used in this model. To save the computational time and simplify the feedwater control system, the 10 steam dump valves simplify to 4 groups and each group developed by a Time Dependent Junction component, which the total steam flow rate was persistent the real NPP operating conditions. And the feedwater control system, which feewater pumps and valves were developed by Time Dependent Volume and Time Dependent Junction respectively.

Emergency Core Coolant Systems include ACC safety injection, HHSI and LHSI. ACC safety injection system is a kind of passive injection system. When the RCS pressure reaches the set point (4.24 MPa), the valve will open automatically and the gravity will drive water into reactor core. The HHSI and LHSI are triggered by Safety Injection (SI) signal.

2.3 Break Model Setting

The break model is developed by three valves shown in Figure 2-3. There are two break valves and one isolation valve, which component number is 886, 887 and 989 respectively. The break valves are closed and the isolation valve is opened in the beginning. When the break transient occurs, the break valves will open and let the coolant flow out of pipe. Simultaneously, the isolation valve will cut the pipe. Boundary condition is set to be 1 atm. In addition, the knowledge about the phenomenon of critical flow is important in evaluation of MSLB. In this research, Henry-Fauske model is used.

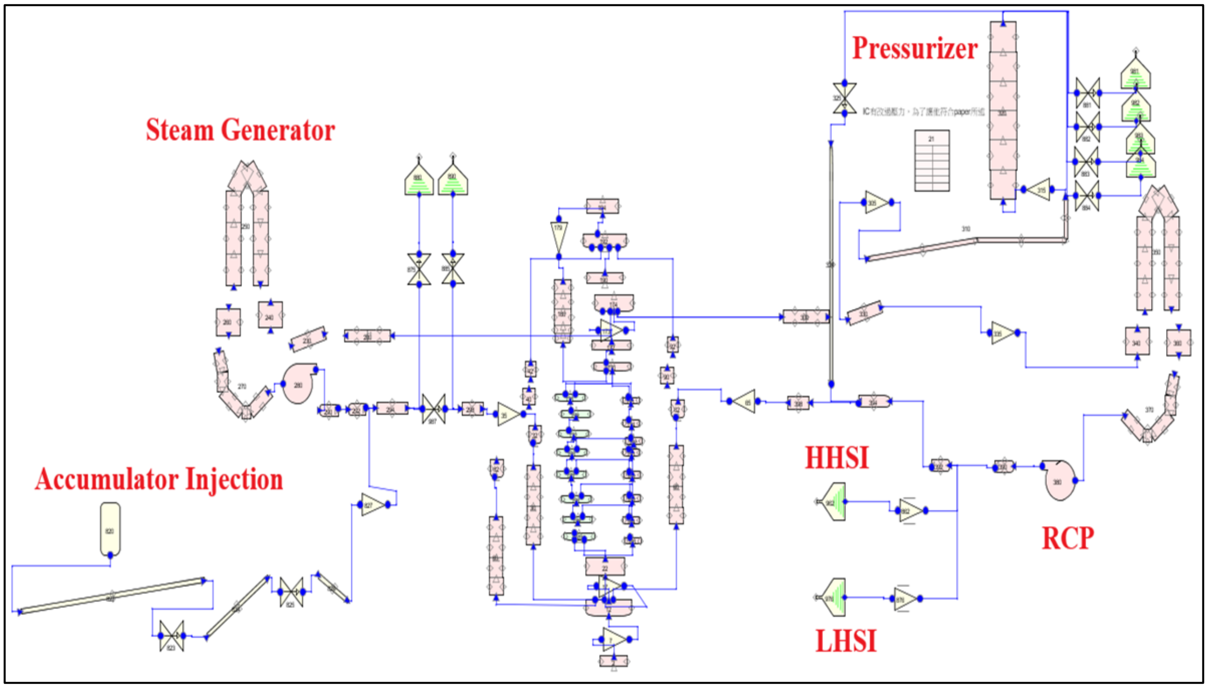


Figure 2-1 Maanshan RELAP5 Model (Primary Side)

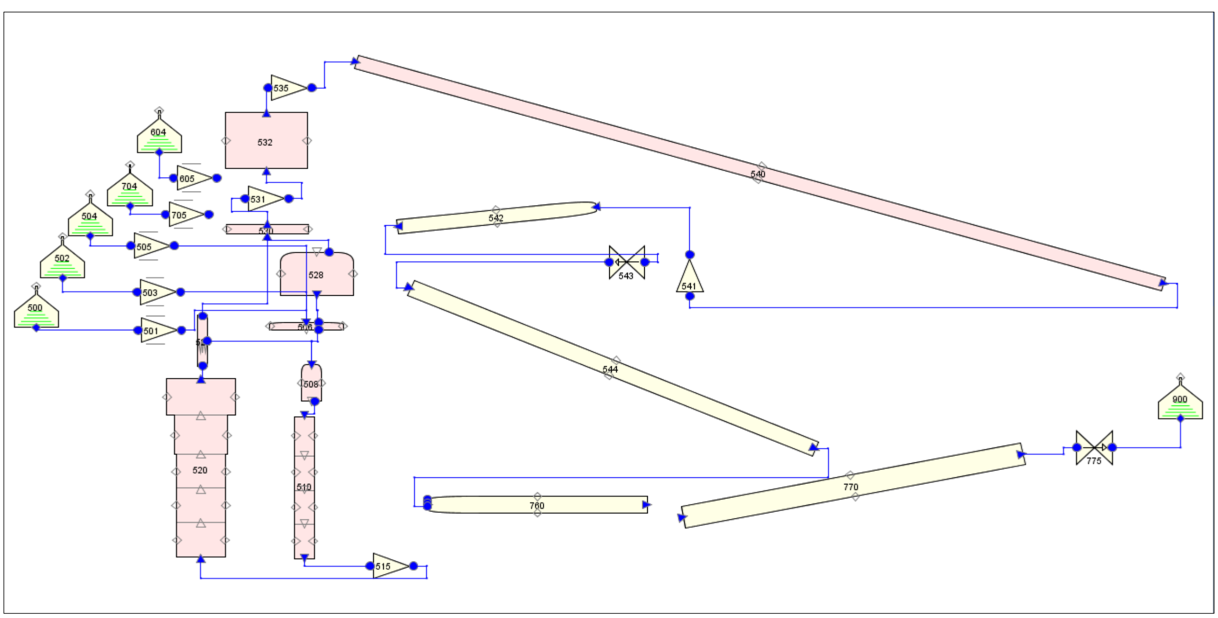


Figure 2-2 Components of Secondary Side of Maanshan NPP in SNAP Interface

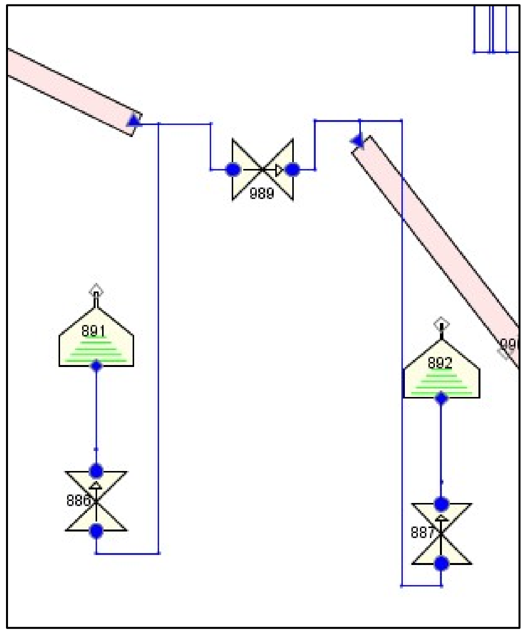


Figure 2-3 RELAP5 MSLB Break Model

3 CASE DESCRIPTION

3.1 MSLB Transient Methodology

The object of this paper is to develop the RELAP5/MOD3.3 model and methodology for analyzing the behaviors of fuel rod, vessel, primary and secondary side during MSLB transient. Figure 3-1 shows the MSLB transient analysis process. MSLB base case was assumed that the break occurred at the secondary side of the loop 1.

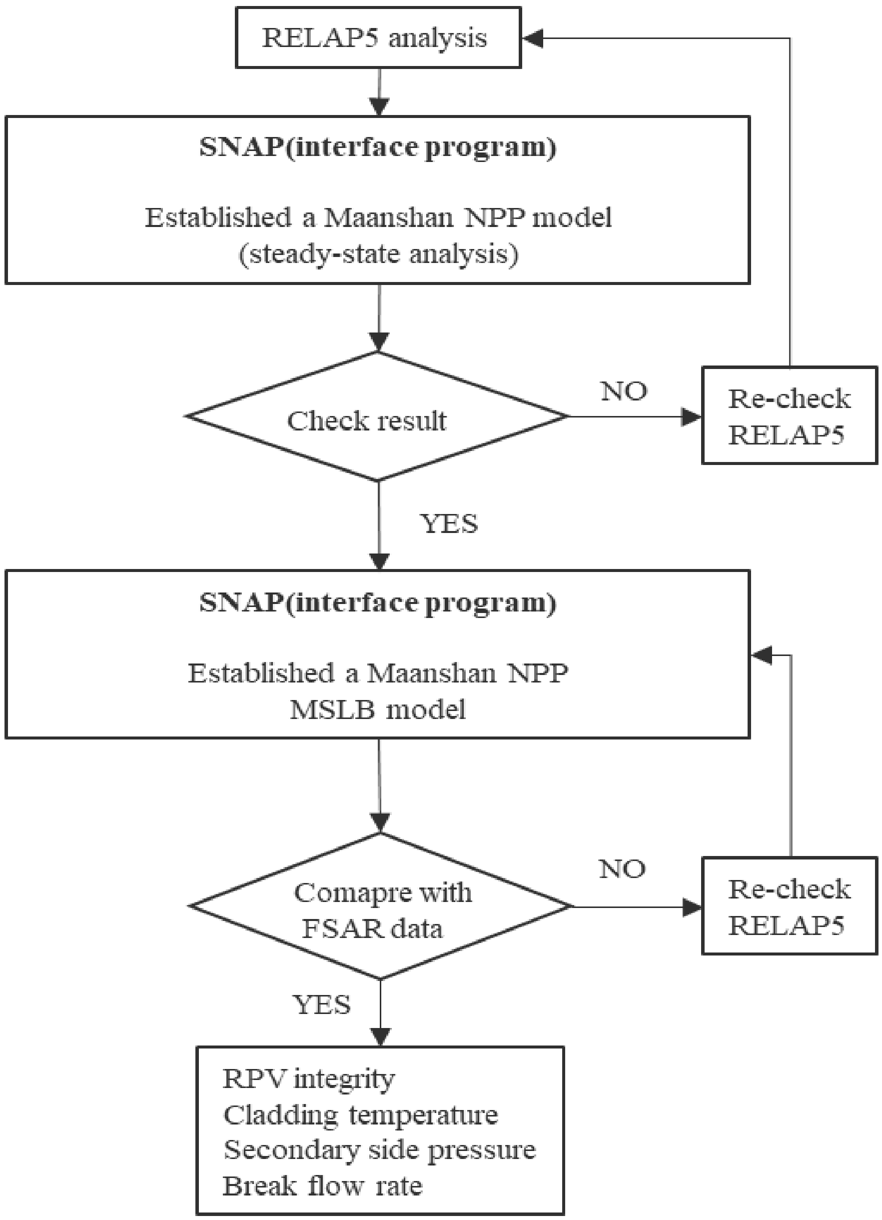


Figure 3-1 1st Loop of Input Model of TRACE

3.2 Assumptions and Initial and Boundary Conditions

The assumptions and initial conditions of the analysis are as follows:

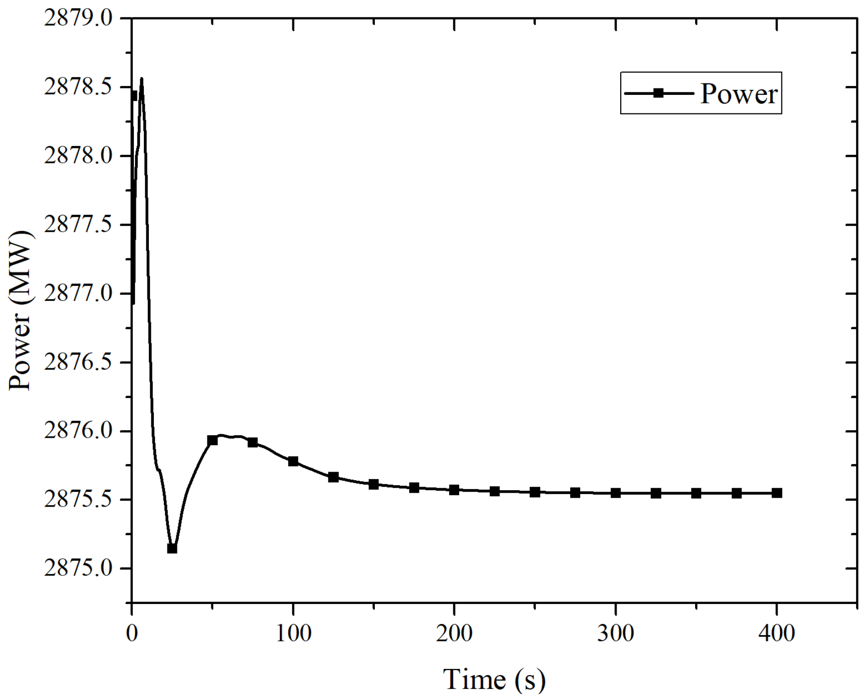
- The initial conditions of important parameters are shown in Table 3-1. The initial power was set as 2875 MWt (102% 2822 MWt).
- Double-ended MSLB break occurred at 400 sec. The broken area of the main steam header side was 0.4367 m^2 (main steam line area).
- MSIVs started to close at 0.5 sec and fully closed at 5.0 sec after MSLB.
- Initial pressure and temperature of primary side were 15.62 MPa and 582.97 K, respectively.
- Initial pressure and temperature of secondary side were 6.96 MPa and 558.64 K, respectively.

To ensure the reactor initial power is 2875 MWt and make sure all the system operated normally. In this research set 400 sec steady-state operation in the beginning shown in Figure 3-2. The Pipe break, which is isolation valve close occurred at 400 sec shown in Table 3-2.

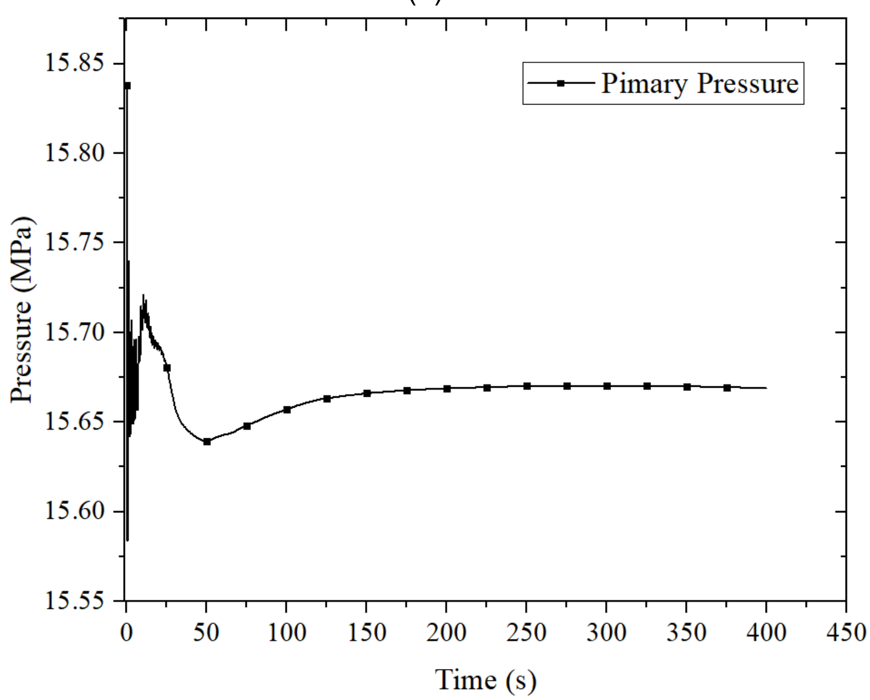
The boundary conditions are set by Time Dependent Volume, which locate at Turbine, Feedwater systems, RCPs, ECCS, Release Valves, Break Valves. Pressure at Turbine is set to be 865 psi. Flow rate of Feedwater systems and ECCS are defined as tables, which are associated with system pressure. Pressure at the breaks is set to be 1 atm and the static quality equal to 1.

Table 3-1 Initial Conditions of Maanshan NPP in RELAP5/MOD3.3

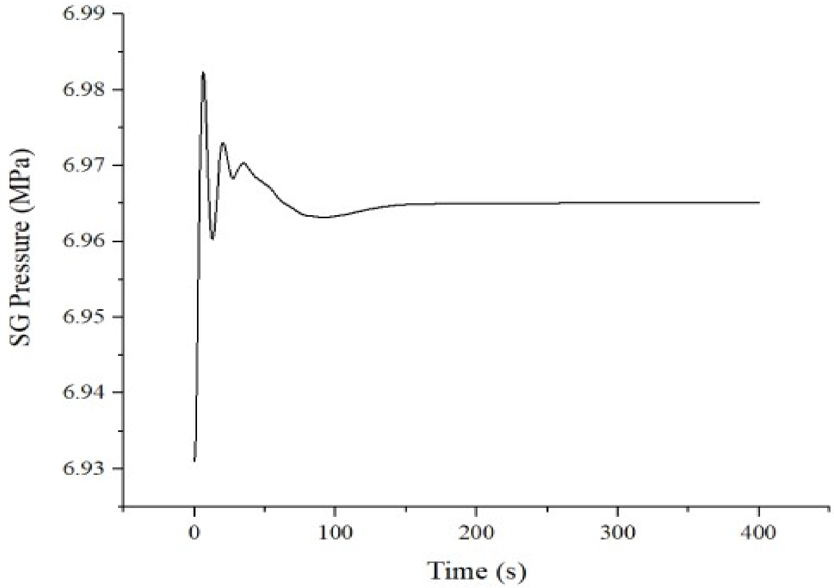
Parameters	RELAP5
Power (MW_t)	2875 (102% of 2822)
Core flow rate (kg/s)	14176
RCS T_{avg} (K)	582.97
RCS Pressure (MPa)	15.862
Scram setpoint (MPa)	12.8
SI signal setpoint (MPa)	11.8
ECCS Electric Delay Time (s)	27
ACC Pressure (MPa)	4.24
SG pressure (MPa)	7.026
SG outlet temperature (K)	558.64
Total outlet steam flow rate (kg/s)	1627.266
Auxiliary feedwater flow rate (kg/s)	7.965
Break Size (m^2)	0.436



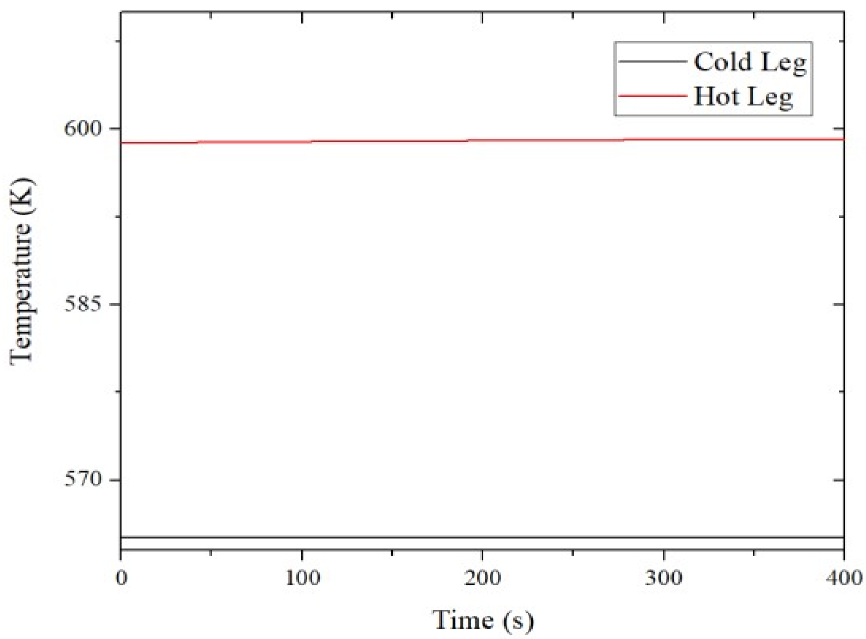
(a) Power



(b) Primary Loop Pressure



(c) SG Pressure



(d) Cold Leg and Hot Leg Temperature

Figure 3-2 Steady-State Results in RELAP5/MOD3.3 (a) Power (b) Primary Loop Pressure (c) SG Pressure (d) Cold Leg and Hot Leg Temperature

3.3 Sequence of MSLB

In the mentioned conditions, a double-ended MSLB is simulated, with all other systems working properly. The most significant events are shown in Table 3-2.

In the beginning, 400 sec steady-state operation would be done to make sure all the systems were operated normally. Immediately after the break, high differential pressure between steam lines causes the actuation of the high pressure safety injection, followed with 27s delay by safety injection trips. Figure 3-3 shows the temperature asymmetry, which will propagate to the core. Because of the boron injected to the vessel by the HPI. Reactor power, shown in Figure 3-4, decreases accordingly.

Table 3-2 Sequence of Events in MSLB

Events	Time
Steady-state	0 (s)
Pipe break	400 (s)
Reactor scram	418 (s)
SI signal	427(s)
Auxiliary feedwater	446 (s)
HHSI	447 (s)
Accumulator injection	541 (s)
End of Accumulator injection	1219 (s)
Transient end	2000 (s)

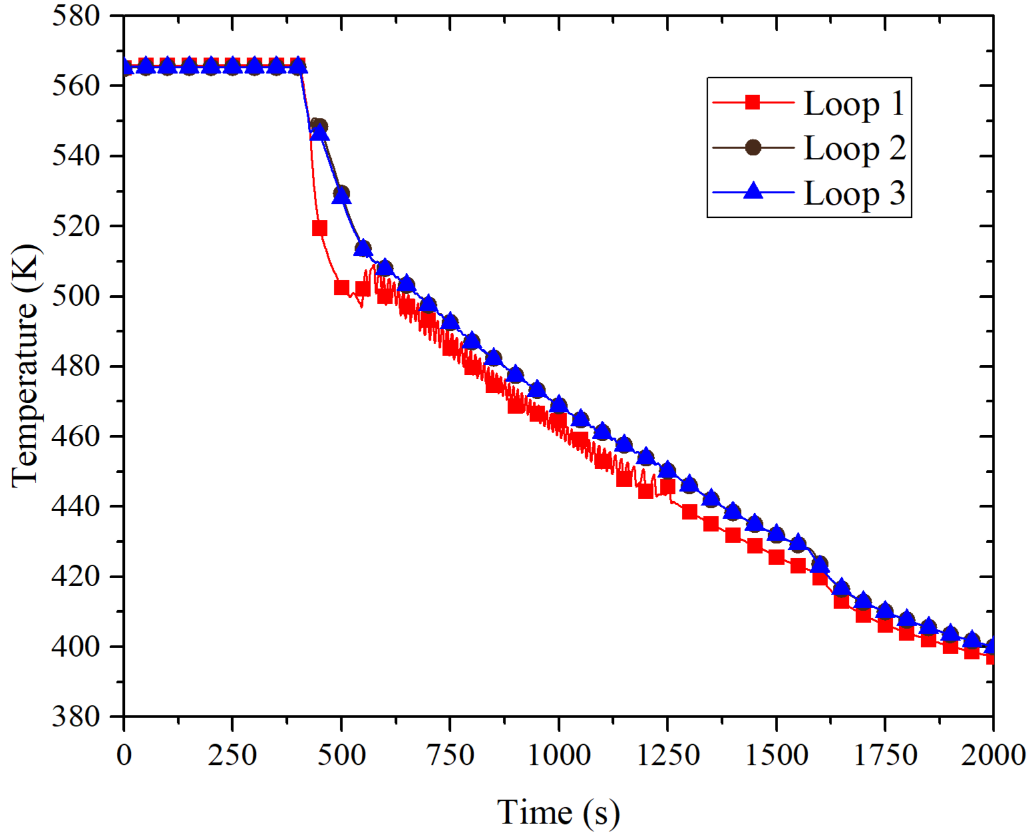


Figure 3-3 Cold Leg Temperature Results

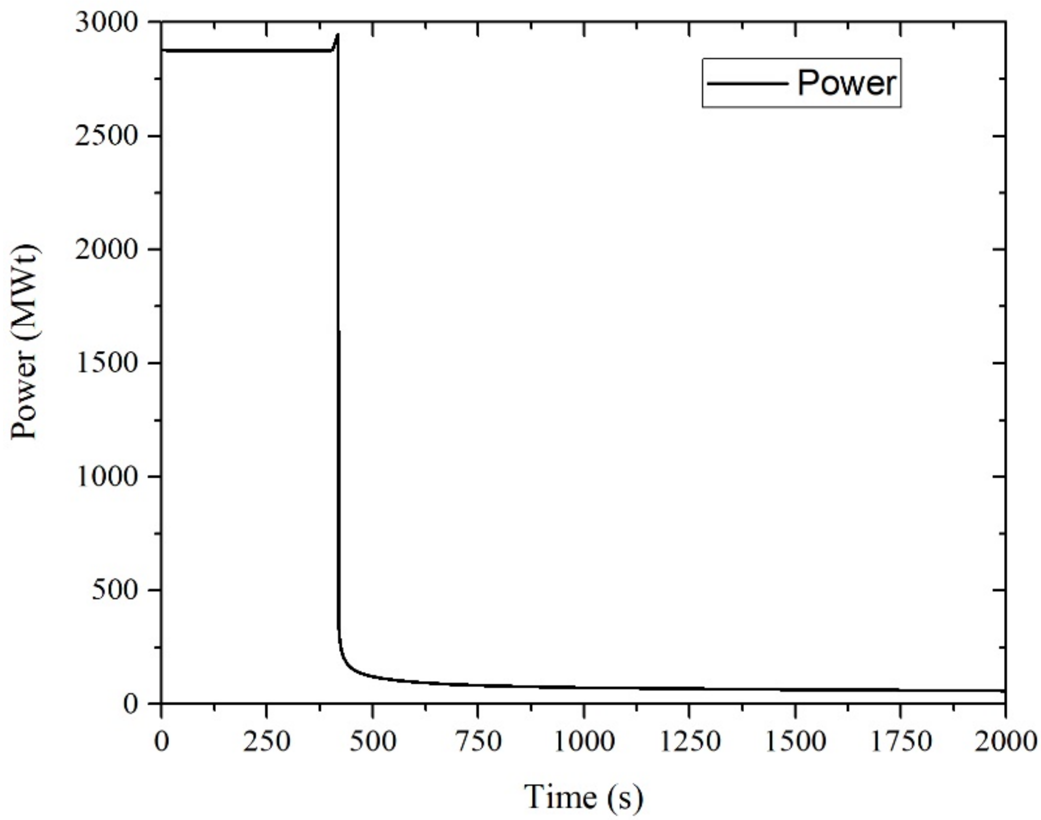


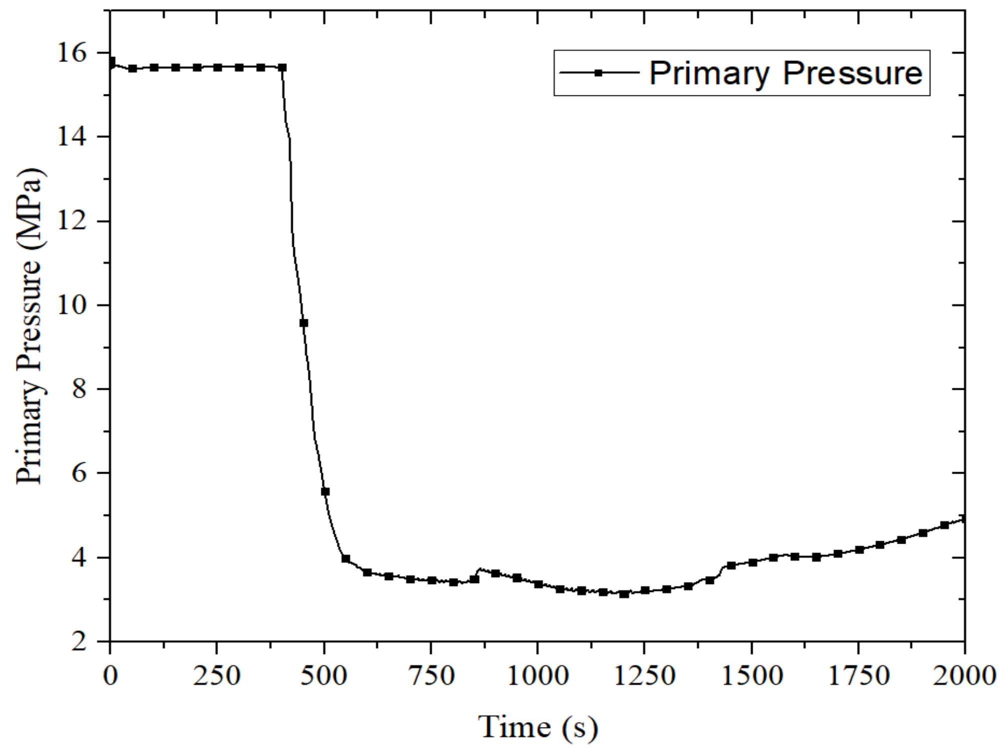
Figure 3-4 Power Result

4 RELAP5/MOD3.3 MSLB RESULTS

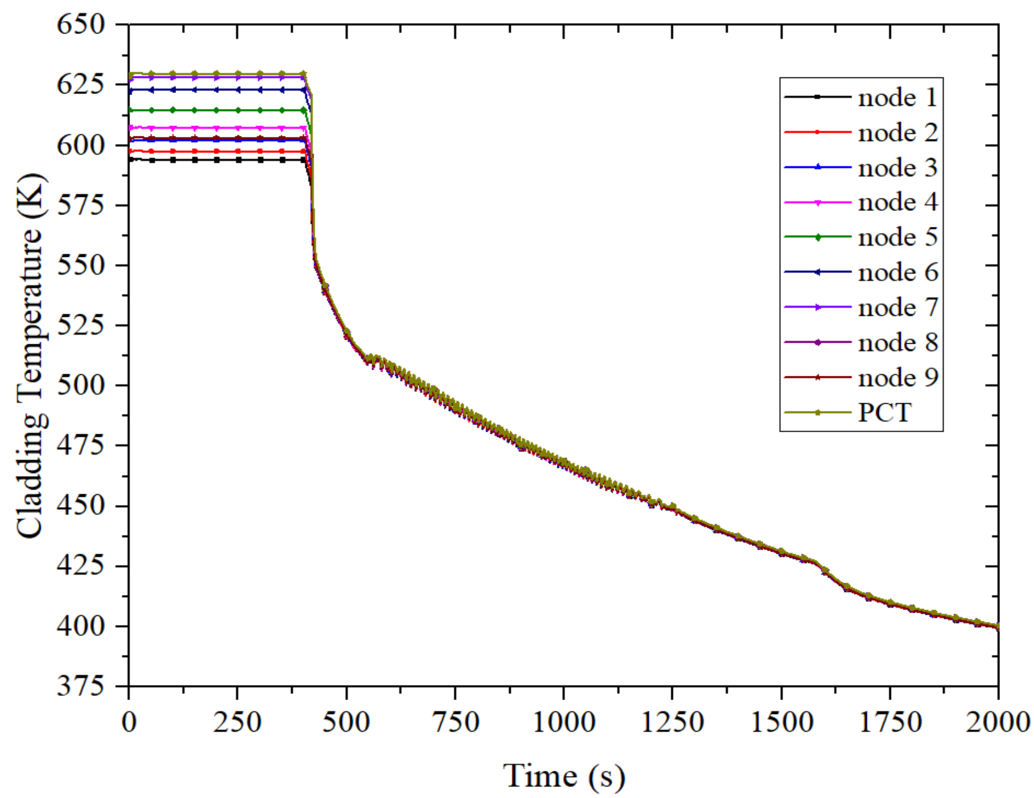
In general, the MSLB accident sequence can be divided into three phases based on the pressure trend in the RPV as shown in Figure 4-1. The first phase is blowdown phase, which secondary side pressure has a rapid depressurization of the RPV. The second phase is the safety injection phase. As the RPV depressurizes, the pressure difference between the RPV and the HHSI becomes small enough to allow HHSI enter the RPV. The third phase is the long-term cooling phase [4]. The MSLB accident in a PWR consists in a break of one steam line upstream of steam header. Immediately after the break, high differential pressure between steam lines causes the actuation of the HPIS, followed with 27s delay by safety injection trips. As a result of the break, broken secondary loop mass flow will highly increase and produce an important heat extraction in the corresponding SG. The primary side of such SG will quickly cool to a temperature rather cold. When cold water reaches the core, it will be an asymmetric cooling in the core.

The MSIV is closed 5 seconds after the break occurs. The results show that the water level and pressure of the SG loop 1 at the beginning of the event caused the water level and pressure to drop, but the electric of the NPP was not lost. The auxiliary feedwater pump continued to replenish water and returned to the full water level. The water level of the SG loops 1 and 2 was slightly reduced due to the impact of the break at the beginning of the event, but the Auxiliary feedwater system continue injected water to SG and refilled to the full water level (shown in Figure 4-2).

After MSLB occurs, a large amount of steam-water mixture rushes out through the break and leads to the closure of MSIVs. Furthermore, all feedwater stops injection immediately, and then reactor scrams. In Figure 4-5, there is a clear trend of decreasing break flow after MSIV closed. However, break flow rate at the first few seconds in this transient. At first few seconds, rapid discharge of reactor coolant decreases the secondary side pressure as shown in Figure 4-3 and causes the flashing of the coolant in the SG. Due to this phenomenon, a lot of voids are produced and heat transfer mechanism changes to film boiling which causes U-tube heat flux as shown in Figure 4-4. In addition, PCT calculated by RELAP5/MOD3.3 which is 630 K which is under required criteria of 2200 ºF in 10CFR50.46 as shown in Figure 4-1. In this research, MSLB accident focus on break flow rate. The break flow rate has been influenced by the void fraction. When the vapor is flash out to the containment, the break flow rate decreases as the void fraction increases as shown in Figure 4-6. Due to the differences of critical flow model and discharge coefficient between FSAR and RELAP5/MOD3.3, the break flow rate may have a little different as shown in Figure 4-5. As shown in Figure 4-5, the RELAP5 break flow rate is more than FSAR data.

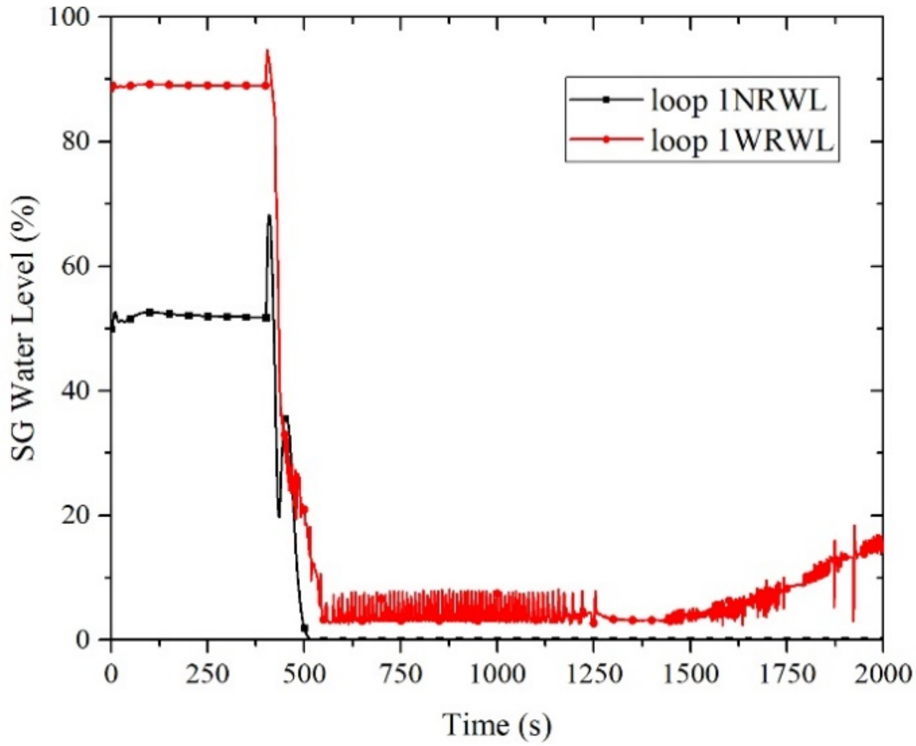


(a) Primary Side Pressure

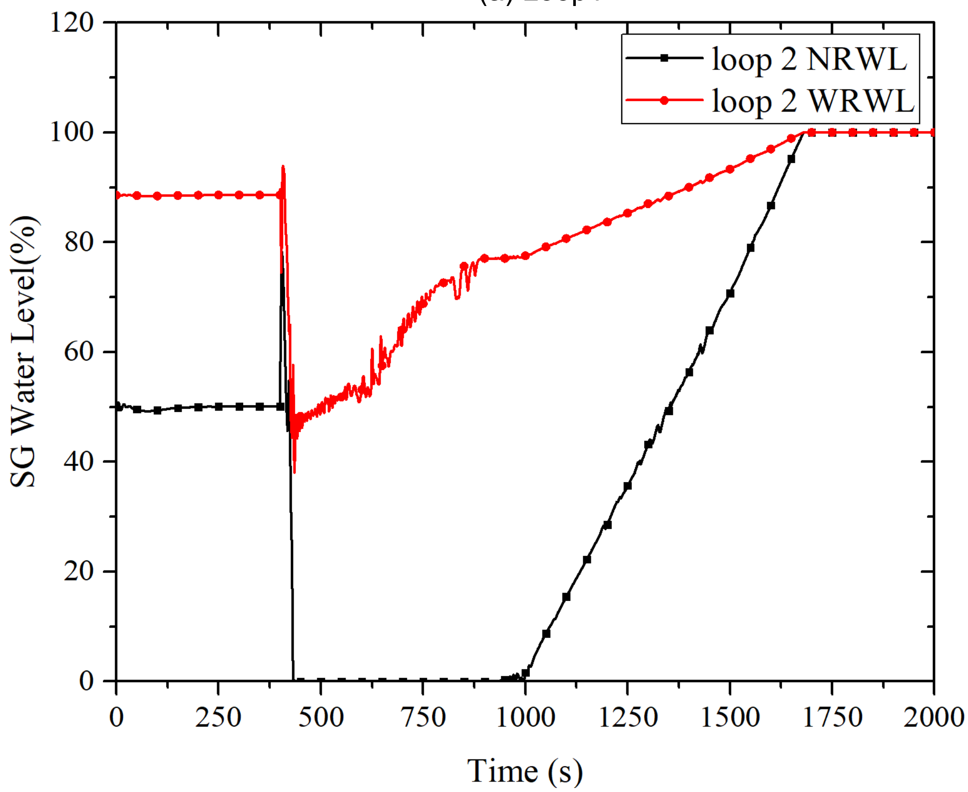


(b) PCT

Figure 4-1 The Analysis Results of RELAP5 (a) Primary Side Pressure (b) PCT



(a) Loop1



(b) Loop 2

Figure 4-2 SG NRWL and WRWL Results (a) Loop1 (b) Loop 2

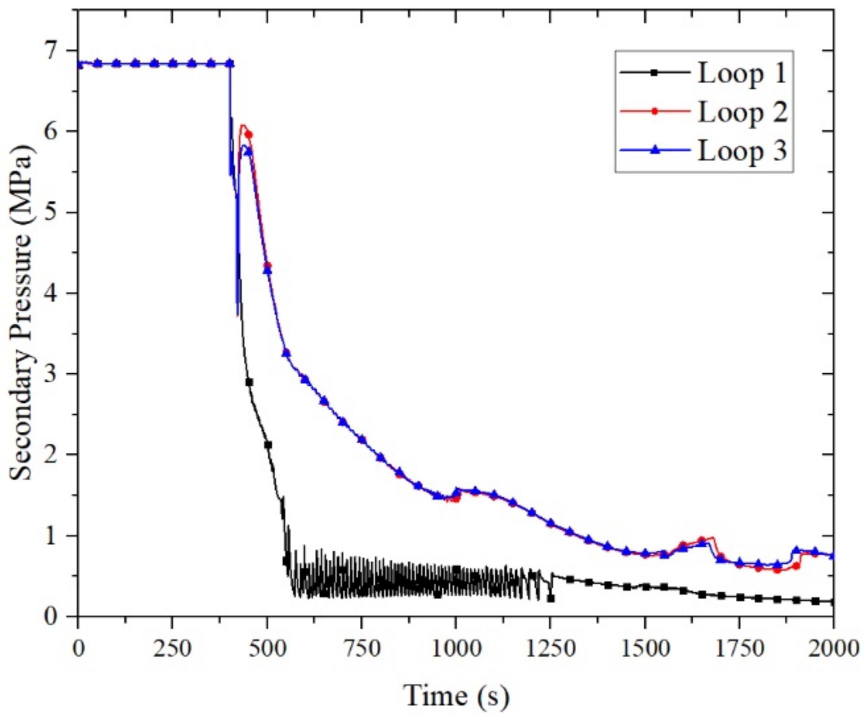


Figure 4-3 Secondary Side Pressure Results

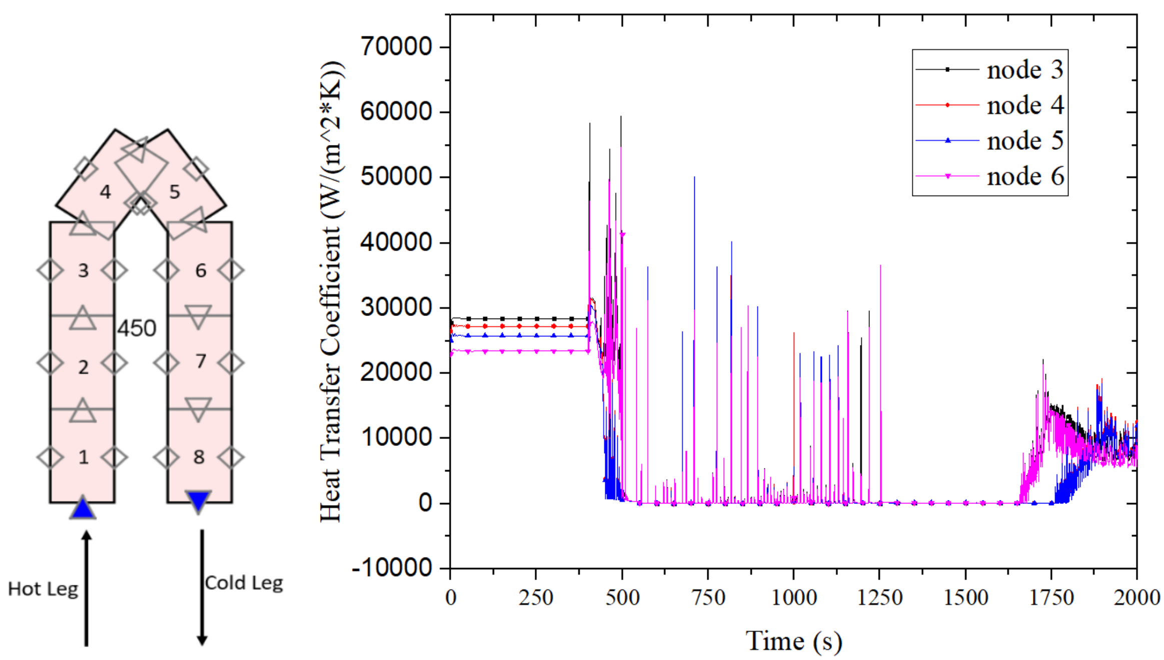
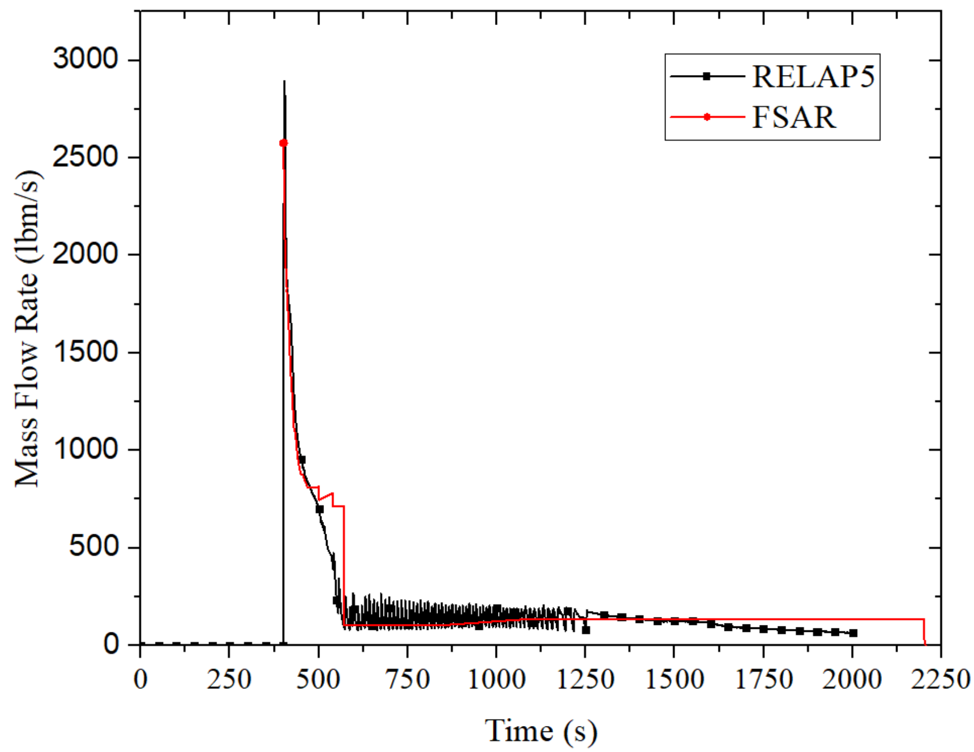
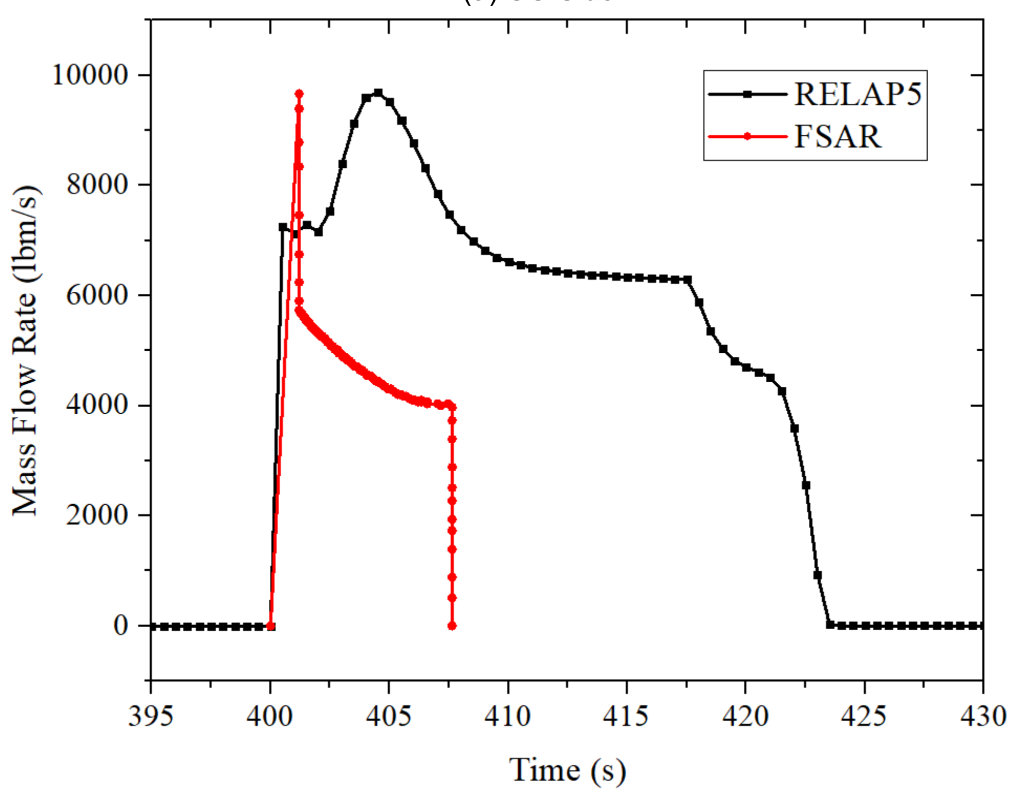


Figure 4-4 The Results of U-tube Heat Transfer Coefficient



(a) SG Side



(b) MSIV Side

Figure 4-5 The Results of Break Flow Rate (a) SG Side (b) MSIV Side

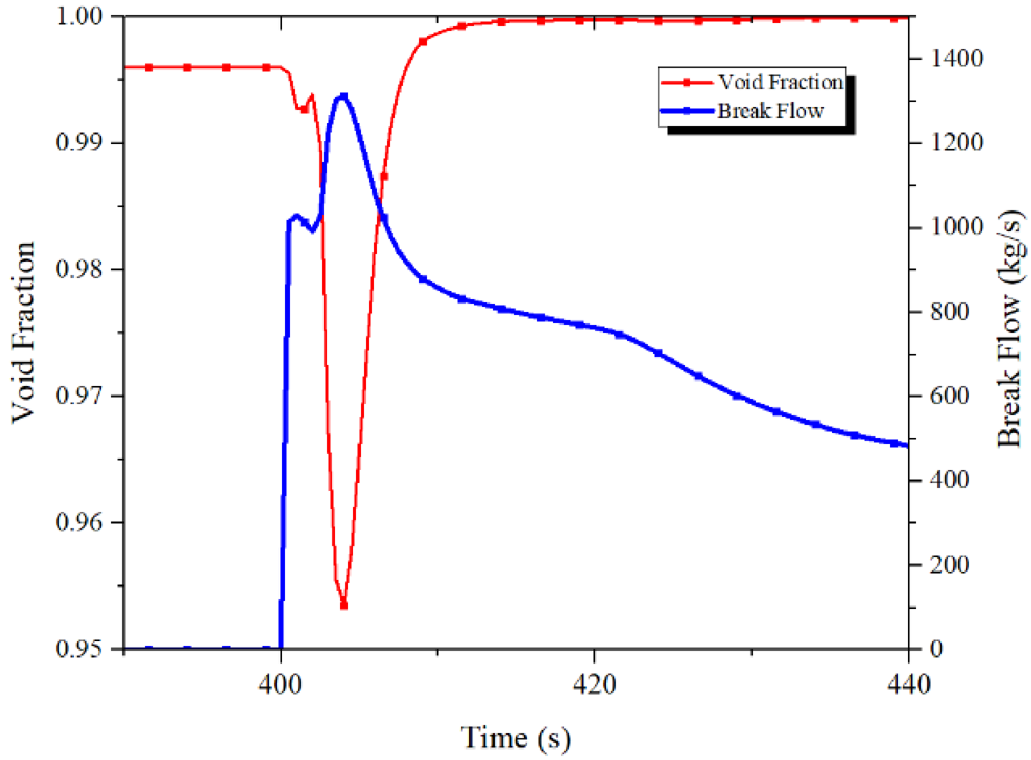


Figure 4-6 Break Flow Rate and Void Fraction Results

5 SENSITIVITY STUDY OF DISCHARGE COEFFICIENT

In RELAP5/MOD3.3, there is a critical flow model and a discharge coefficient that affects the calculation of break flow rate. The critical flow model limits the fluid velocity beyond the speed of sound, which is embedded in RELAP5/Mod3.3. Therefore, the default preset cannot be changed. The default is the Henry-Fauske model [5]. Fortunately, discharge coefficient can be change. In this research, the sensitivity analysis of discharge coefficient value is 0.1, 0.2, 0.3, 0.4, 0.5, 0.6, 0.7, 0.8, 0.9. As can be seen from the Figure 5-1, a strong relationship between break flow rate and the discharge coefficient. In this research, Observe the great discharge coefficient comes great break flow rate. The discharge coefficient is much higher than the other results in the whole pipe break stage (t < 50 s). The discharge coefficient results only have a different in the first 50 seconds, and the break flow rate calculated after 150 seconds is not much different.

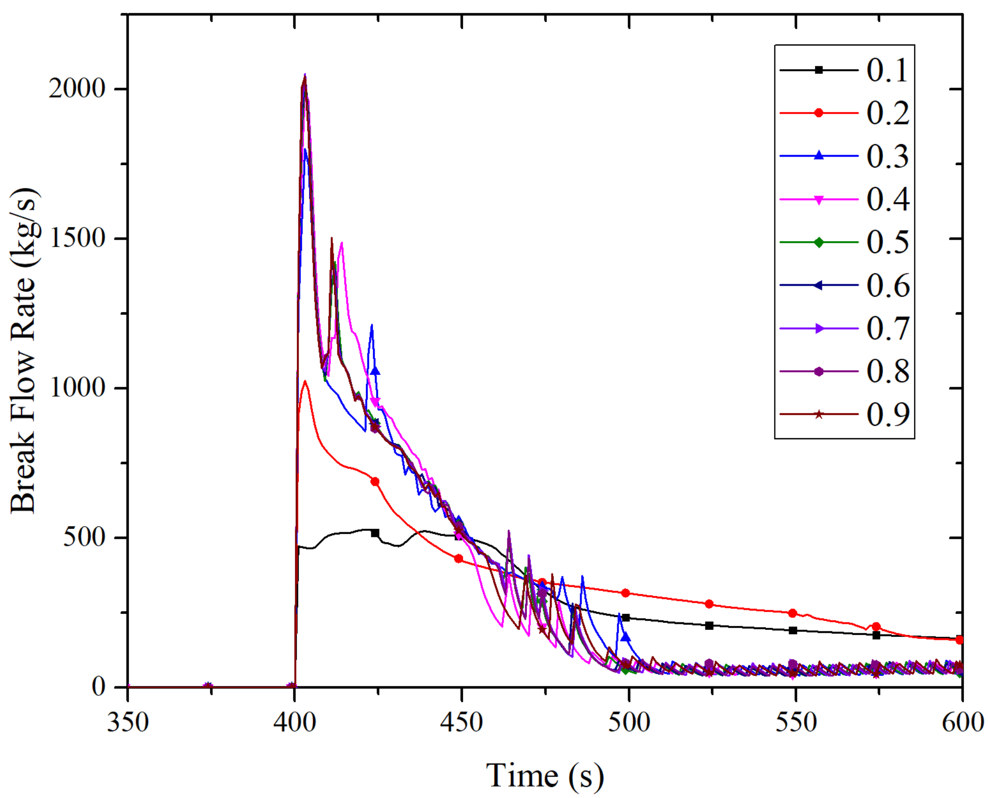


Figure 5-1 Discharge Coefficient Sensitivity Analysis of Break Flow

6 CONCLUSION

This research presents the results of the simulation of a MSLB inside-containment transient event by using RELAP5/MOD3.3 with primary circuit closed and full of water. These results also indicate this RELAP5/MOD3.3 model of the Maanshan NPP for the MSLB inside-containment LOCA is capable of predicting thermal hydraulic phenomena reasonably. From the results obtained in the previous sections, it can be observed that the main phenomenology is reproduced. The injection from HPIS allows maintain the primary in safe situation. In general, RELAP5/MOD3.3 results for the Maanshan NPP model are coherent with FSAR data. Additionally, the relationship between break flow rate and the discharge coefficient has been observed that the great discharge coefficient comes great break flow rate. In conclusion, the model presents good achievements, giving improved ability to manage safety margins. This study successfully demonstrated a RELAP5/MOD3.3 model for Maanshan NPP model, and this methodology can be applied to nuclear reactor safety analysis for transient accident events.

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11. ABSTRACT (200 words or less)

In Taiwan, many prediction analysis for different purposes are investigated with different system codes, such as RELAP5 developed by US NRC. The main purpose of this study was to simulate Maanshan NPP PWR Main Steam Line break (MSLB) inside-containment by using RELAP5/MOD3.3, and then to discuss the results of this model by referring to Maanshan FSAR. Furthermore, sensitivity study of discharge coefficient is also included. In addition, the peak cladding temperature is maintaining under the criteria value in 10CFR50.46 and the results of this RELAP5/MOD3.3 model indicates reasonable thermal-hydraulic phenomena and presents good agreements with FSAR. The sensitivity analysis results show that the variations of break flow rate were governed by discharge coefficient. This study demonstrated that a methodology of RELAP5/MOD3.3 model for Maanshan PWR has been successfully developed, and the present simulation results and analysis can be used for safety analyses and further transient applications.

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