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International Agreement Report

Analysis of Maanshan Station Blackout Accident and Rescue Procedures under Different Tube Plugging Situations with TRACE

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ABSTRACT

This research focused on the analysis of URG (Ultimate Response Guideline) procedure and the FLEX (Flexible and Diverse Coping Strategies) strategy after one and four hours when Maanshan nuclear power plant is under station blackout (SBO) accident by using TRACE code. This research then explores the influence on heat transfer between primary side and secondary side when there is a plugged tube in the steam generators of the power plant. The NEI (Nuclear Energy Institute) had proposed a FLEX strategy [1-3] and Taiwan Power Company also has developed URG procedure [4, 5], in order to maintain the safety of plant during a severe disaster. The equipment of plant will become old and deteriorative as time passes by. If there are any problems with the tubes of steam generators, operators will plug the defective tube during outage inspection to prevent them from leaking or bursting. This research uses TRACE to analyze under 2%, 5% and 10% of tube plugging, the effectiveness of the URG procedure and the FLEX strategy used during the SBO accident of Maanshan nuclear power plant. The result shows that even under 10% of tube plugging, the URG procedure and the FLEX strategy won't be affected by tube plugging and still can bring the plant back to safety.

FOREWORD

In March 11, 2011, with earthquake and tsunami, the Fukushima accident brought a huge shock to the world. It shows that when a beyond design basis problem happens, nuclear power plants will face problems, such as whether the operator reactions are fast enough or whether the rescue procedures are effective. Even though the control rods had successfully inserted into the core when the earthquake happened, the following tsunami destroyed much equipment, causing the SBO accident. Because of the destruction of equipment and the lack of experience for operators to deal with these kinds of complicated accidents, operators had missed the final timing to inject sea water into the core, and caused damage of fuel cladding with radiation released.

In order to handle very complicated accidents like Fukushima, NEI had developed a strategy called “Diverse and Flexible Coping Strategies”. For any large non design basis accident like ELAP (Extended Loss of Alternating-Current Power) or LUHS (Loss of Ultimate Heat Sink), the FLEX strategy proposes rescue procedures with diversity and flexibility to deal with these problems. Similarly, the Taiwan Power Company also developed the “Ultimate Response Guideline, URG” for the safety of nuclear power plant in Taiwan, where there is a similar environment with Japan. The concept of URG is that if there’s an accident which will challenge the safety of plant, the rescue procedures must be done in a limited time. Actions are taken to prepare every available water sources, lower the pressure of core and execute water injection to remove the heat from core and keep the plant safe.

During the rescue procedures, the boundary conditions of plant will affect its effectiveness. As the main component for heat transfer in PWR, the integrity of steam generators must be maintained. If there’s any deterioration in the tubes of SG (Steam Generator), then they must be plugged during the outage inspections. This research uses the best-estimate thermal hydraulic program, TRACE, to analysis the effectiveness of URG and FLEX for Maanshan nuclear power plant with different percentage of tube plugging, and study the influence of tube plugging to these rescue procedures.

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EXECUTIVE SUMMARY

TRACE is the best-estimate thermal hydraulic analysis code developed by U.S. NRC. Combining with four analysis codes: TRAC-P, TRAC-B, RELAP5 and RAMONA, TRACE is designed for simulation of operating transient and hypothetical accidents in light water reactors.

SNAP is an interface program of NPP analysis codes which is 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 of SNAP allow every user to easily build the model of nuclear power plant. Furthermore, SNAP has the animation function to present the analysis results.

This research focused on the analysis of URG procedure and the FLEX strategy after one and four hours when Maanshan nuclear power plant is under SBO accident conditions by using TRACE code. Then the research explores the influence on heat transfer between primary side and secondary side when there are plugged tubes in the steam generators of power plant. Maanshan nuclear power plant is the third nuclear power plant, also the only PWR (Pressurized Water Reactor) in Taiwan. The two units, each with three loops, were built by Westinghouse, can generate 980MWe after small increase of power. A hypothetical earthquake was assumed in this research, and a tsunami in the site area caused a SBO accident after the earthquake. With these accident conditions, an input model of Maanshan nuclear power plant of TRACE was used to analyze the effectiveness of URG procedure and FLEX strategy. After that, a tube plugging analysis was made with a revised input model of 2%, 5% and 10% of tube plugging to study the effect of plugged tubes on SBO and rescue procedures of the plant.

The biggest lesson we learned from Fukushima accident, is when a large complex accident happens, any detectors for instruments in the plant could become invalid. Thus it’s very difficult for plant operators to rescue the plant when they can’t find out its real state. Therefore, except for enhancing the equipment of the plant, there could be a large impact on plant safety if there’s an applicable and effective rescue procedure. NEI had proposed the FLEX strategy and Taiwan Power Company also has the URG procedure, in order to maintain the safety of plant from a severe disaster. With the assumption and analysis results in this research, the URG procedure and the FLEX strategy can appropriately deal with SBO accident and keep the peak cladding temperature in safe margin.

During the operation of the plant, the equipment of plant will become old and deteriorate as time passes by. Being the most important boundary of heat transfer between primary and secondary side, if there are any problems with tubes in steam generators, operators will plug the defective tube during outage inspection to prevent them from leaking or bursting while the plant is operating. To predict the effect of heat transfer affected by tube plugging, this research uses TRACE to analyze, under 2%, 5% and 10% of tube plugging, the effectiveness of the URG procedures and the FLEX strategy to respond to the SBO accident at the Maanshan nuclear power plant. The results show that even under 10% of tube plugging, the URG procedure and the FLEX strategy won’t be affected by tube plugging and still can bring the plant back to safety.

ABBREVIATIONS AND ACRONYMS

ACC	Accumulator
AFW	Auxiliary Feedwater
BDBEE	Beyond-Design-Basis External Event
BWR	Boiling Water Reactor
DBA	Design Basis Accident
ECCS	Emergency Core Cooling System
EFS	Emergency Feedwater System
ELAP	Extended Loss of Alternating-Current Power
EOP	Emergency Operating Procedure
FLEX	Flexible and Diverse Coping Strategies
FSAR	Final safety analysis report
HPSI	High Pressure Safety Injection
LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection
LUHS	Loss of Ultimate Heat Sink
MDAFP	Motor Driven Auxiliary Feedwater
MFWP	Main Feedwater Pump
MSIV	Main Steam Isolation Valves
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
PORV	Power-Operater Relief Valve
PWR	Pressurized Water Reactor
PZR	Pressurizer
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
SAMG	Severe Accident Management Guideline
SBO	Station Blackout
SFP	Spent Fuel Pool
SG	Steam Generator
TAF	Top of Active Fuel
TDAFP	Turbine Driven Auxiliary Feedwater Pump

TPC	Taiwan Power Company
URG	Ultimate Response Guideline

1 INTRODUCTION

Maanshan nuclear power plant, known as the third nuclear power plant in Taiwan [6, 7], is located in Pingtung County. The two units of it had started operating in 1984 and 1985. After a small elevation, the thermal power of it were increased to 2822 MWt. The plant was divided to the primary side and the secondary side. The primary side is also called “Reactor Cooling System, RCS”, including pressure vessel, U tube of steam generator and reactor cooling pump, etc. There’re three loops in primary side, and the pressurizer is in only loop 2. On the other hand, there’s steam generator, main steam pipe and feedwater pump in the secondary loop.

TRACE, stands for “The TRAC/RELAP Advanced Computational Engine” [8], is the best-estimate thermal hydraulic analysis program developed by U.S. NRC. Combining with four analysis programs: TRAC-P, TRAC-B, RELAP5 and RAMONA, TRACE was designed for simulation of operating transient and hypothetical accidents in light water reactors. The hydraulic component like PIPE, VALVE, PUMP, etc. in TRACE can be set as the users wish, and can be also divided in to smaller cells for more details of simulation.

SNAP is an interface of NPP analysis codes which developed by U.S. NRC and Applied Programming Technology, Inc. [9]. Different from the traditional input deck in ASCII files, the graphical control blocks and thermal hydraulic connections allow every user to easily build the model of nuclear power plant. Furthermore, it’s also more convenient for everyone to study the structure of plants.

URG is a rescue procedure proposed by the Taiwan Power Company. The highest principle of URG is to remove the residual heat and stabilize the plant when an accident happens. Meanwhile preventing the radioactive from diffusion. There’re three main steps in URG: depressurization, water injection and containment venting. Also there’re three triggers for the activation of URG:

[1] Reactor or steam generators had lost all electricity driven water injection.

[2] The plant had lost every onsite and offsite AC power, including diesel generator and gas turbine generator.

[3] The plant scram because of strong earthquake and there’s warning of tsunami from the Central Weather Bureau.

Whenever activating URG, the operators need to execute:

[1] Depressurization of steam generator

[2] Prepare the path and equipment of alternative water injection of reactor core and steam generator

[3] Prepare containment venting.

When URG is activating, the regular water injection system could probably unavailable. Thus the alternative water injection must be used, like fire pump or engine driven pump. However, the flow rate and working pressure are low in alternative water injection system. Therefore, the depressurization step must be done before the water injection. Furthermore, if there’s hydrogen generates because of the reaction between high temperature steam and fuel cladding during the rescue, containment venting must be executed to prevent the gas from explosion.

The Nuclear Energy Institute had proposed the NEI 12-06 “Diverse and Flexing Coping Strategies Implementation Guide” in 2012. FLEX, as its name, is a rescue strategy with diversity and flexibility. The main purpose of it is to increase the safety margin of the plant. The most special feature of FLEX is to evaluate the characteristic of different plants and analyze their applicability and response capability when facing accidents, and then try to increase their ability dealing with accidents. There’re three phases in FLEX:

[1] Phase 1: Briefly assess the response capability of plant, and apply the existing equipment for rescue and extend responding time.

[2] Phase 2: Verify the damaged safety function of plant, and use portable equipment to draft the strategy of ensuring the safety function.

[3] Phase 3: Make sure that the offside rescue equipment could be obtained, establish a command center and draft a responding plan.

This research focused on the analysis of URG procedure and the FLEX strategy after one and four hours when Maanshan nuclear power plant is under SBO by using TRACE code with SNAP interface. Then, the SBO with 2%, 5% and 10% of SG tube plugging cases are analyzed and simulated in this study to confirm the capacity of URG procedure and the FLEX strategy.

2 MODEL ESTABLISHMENT

Fig. 2-1 is the input model of Maanshan nuclear power plant in TRACE and Fig. 2-2 is the control interface of TRACE/SNAP. There're 133 hydraulic components, 34 heat structures, 2 power component and almost 700 control blocks in the Maanshan nuclear power plant input model of TRACE. The primary side includes reactor pressure vessel and three loops. Accumulator, steam generator U tube, hot leg and cold leg are there in each loops, and pressurizer with its pipeline is in only loop 2. Steam generator downcomer, evaporator, main steam pipe and its relive valve, isolated valve are there in the secondary side. Only heat and no mass exchanging between the primary side and secondary side. The biggest difference between TRACE and other thermal hydraulic analysis program is that TRACE can use a three dimensional model for the reactor vessel. The model of vessel in this input model is a three dimensional cylinder, and can simulate the thermal hydraulic phenomenon of core more precisely. The vessel was divided into 2 rings, 6 azimuths and 12 axial layer like Fig. 2-3. The outer ring is use for reactor downcomer while inner ring for core barrel. The 3 to 6 layer in inner ring is for fuel rod, there're connected with heat structure for the heat power of fuel. The length of fuel is 3.65 meters. The hot leg and cold leg is located at the 8 layer of outer ring and inner ring. Most of the component in loop was built from PIPE, including pipe of hot, cold leg, steam generator U tube, etc. The reactor cooling pump was built from PUMP, with a VALVE and BREAK for the simulation of seal leakage during accident. A FILL was connected at hot leg for water injection during rescue procedure. Two VALVE for check valve are there connected at the outlet of accumulator. The accumulator is built from PIPE, and will automatically start if the pressure of primary side is low enough. Pressurizer and its pipeline are also built from PIPE, connecting on the hot leg of loop 2 and bonding with 4 PIPE on the top of it for PORV and safety valve. Steam generators in secondary side are also built from PIPE, with 2 main steam safety valve and 2 PORV. Another FILL was connected with steam generator downcomer for water injection of steam generator, including fire injection, alternative injection MDAFW, etc.

For the simulation of different operation of plant in transient, there's also many control logics in this input model. Fig. 2-4~2-6 is the control logic of control depressurization of steam generators. With the change of timing and target pressure, one can execute depressurization at any time. Fig. 2-7 is the control block and logic for water injection of secondary side, including fire injection, MDAFW, TDAFW and alternative injection. The flow rate, timing and working pressure can all be adjusting for different sources of injection. Fig. 2-8 is the control block if seal leakage, use for the starting time and flow rate of RCP seal leakage. Fig. 2-9 is the control logic for the timing of reactor tripped when accident happened.

The verification of the TRACE model of Maanshan nuclear power plant were performed in our lab [10-12]. For instance, the correction of flow rate of relief valve and safety valve, and four important start up test: PAT-21, PAT-49, PAT-50 and PAT-51 with comparison to the data from the plant. The result shows that TRACE can successfully analyze the thermal hydraulic parameters and their tendency in plant.

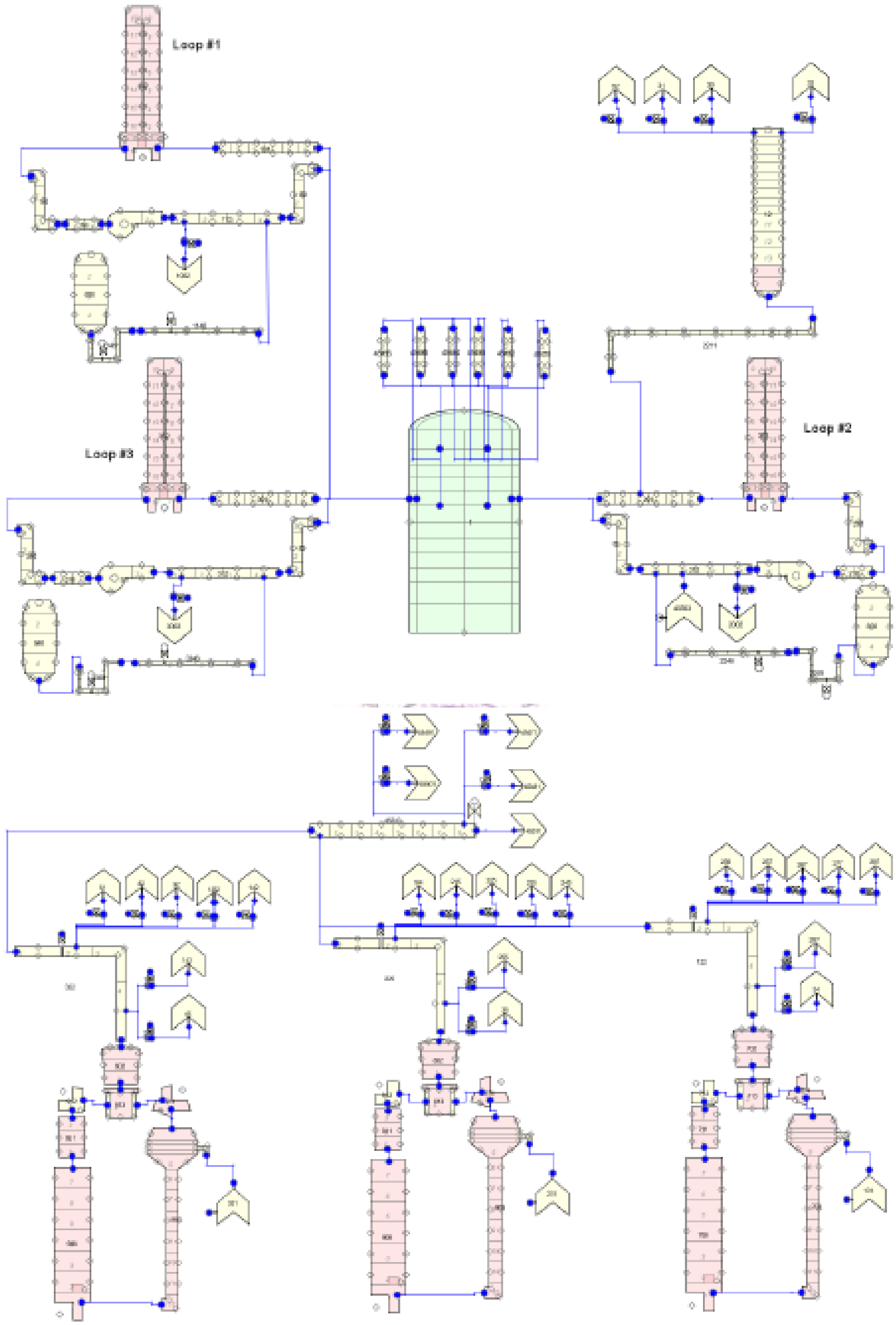


Figure 2-1 Input Model of Maanshan NPP in TRACE

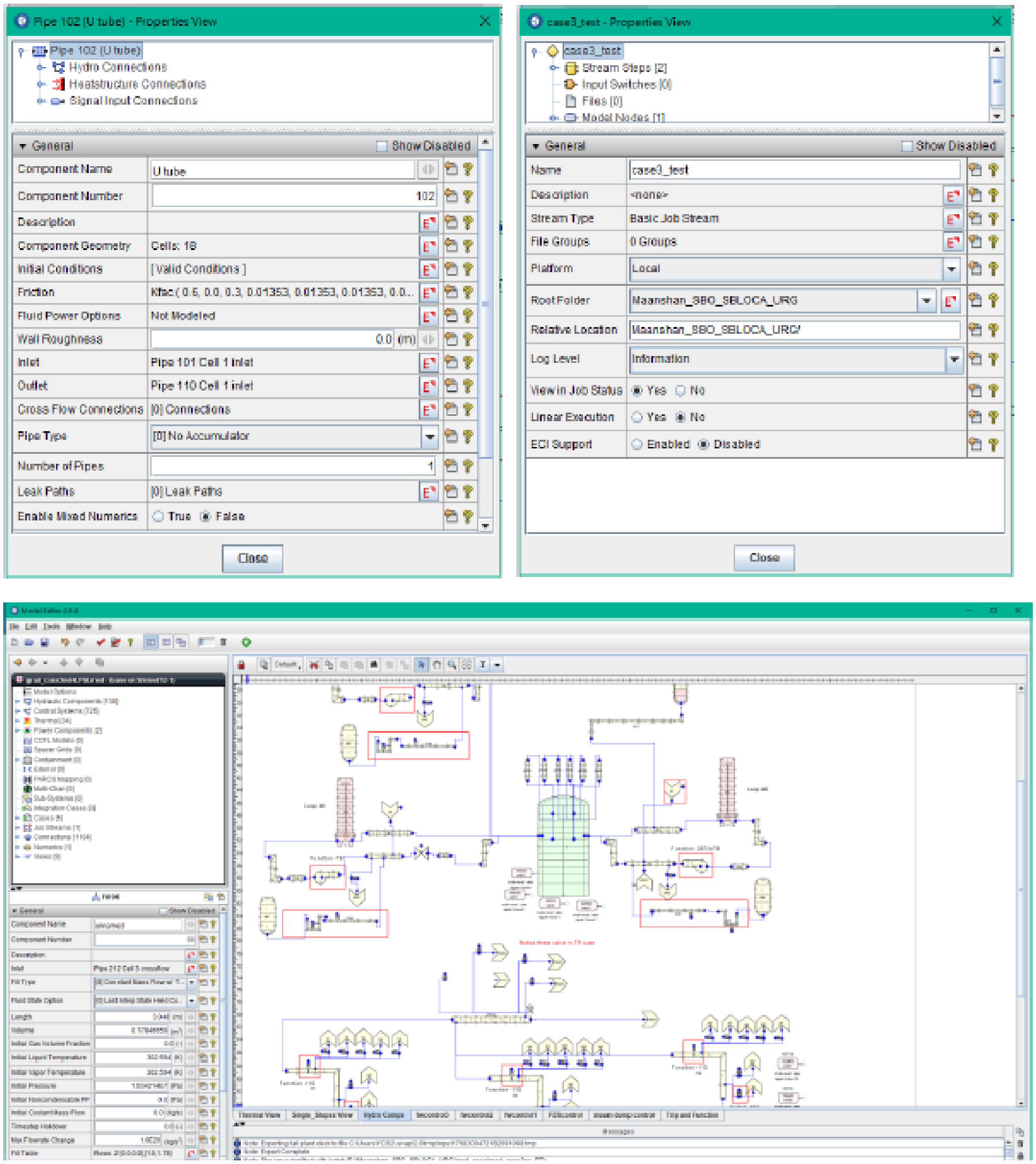
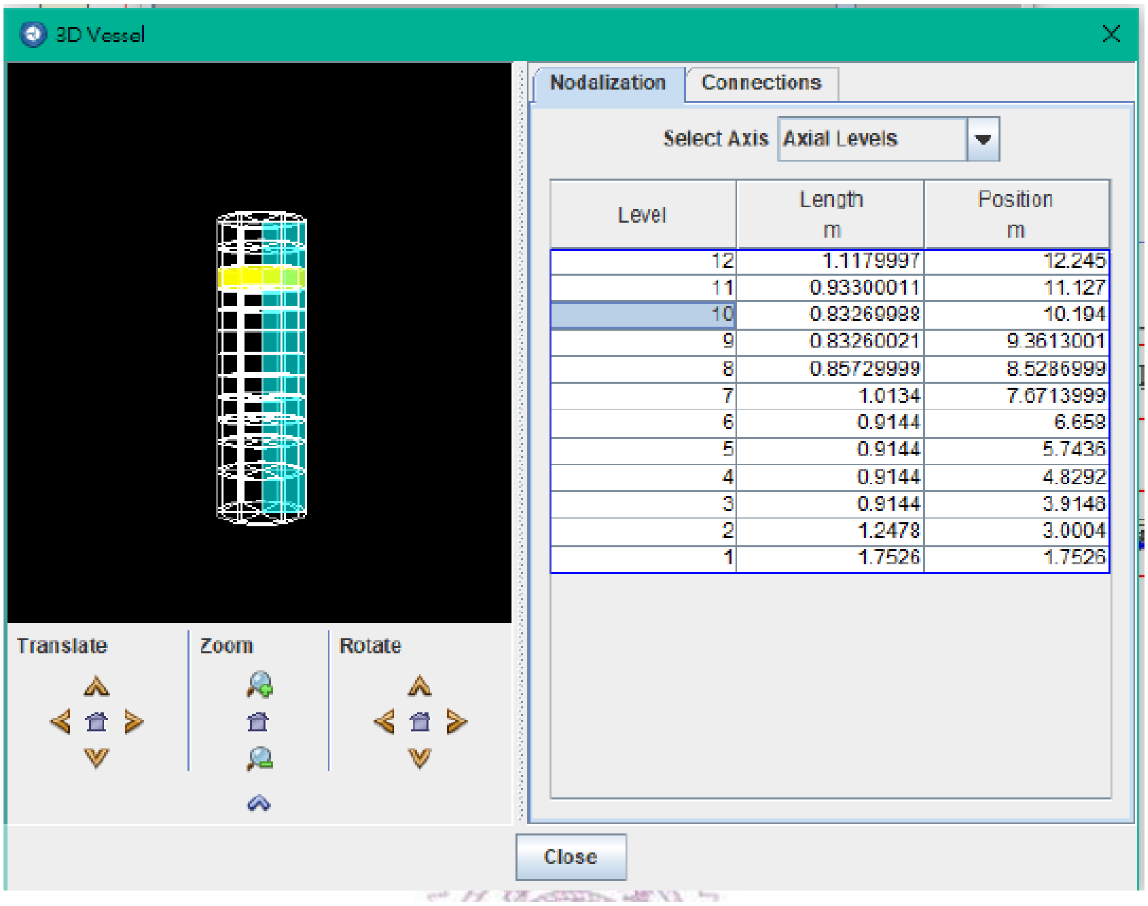


Figure 2-2 Control Interface of TRACE/SNAP



Level	Length m	Position m
12	1.1179997	12.245
11	0.93300011	11.127
10	0.83269988	10.194
9	0.83260021	9.3613001
8	0.85729999	8.5286999
7	1.0134	7.6713999
6	0.9144	6.658
5	0.9144	5.7436
4	0.9144	4.8292
3	0.9144	3.9148
2	1.2478	3.0004
1	1.7526	1.7526

Figure 2-3　Simulation of Reactor Pressure Vessel in TRACE

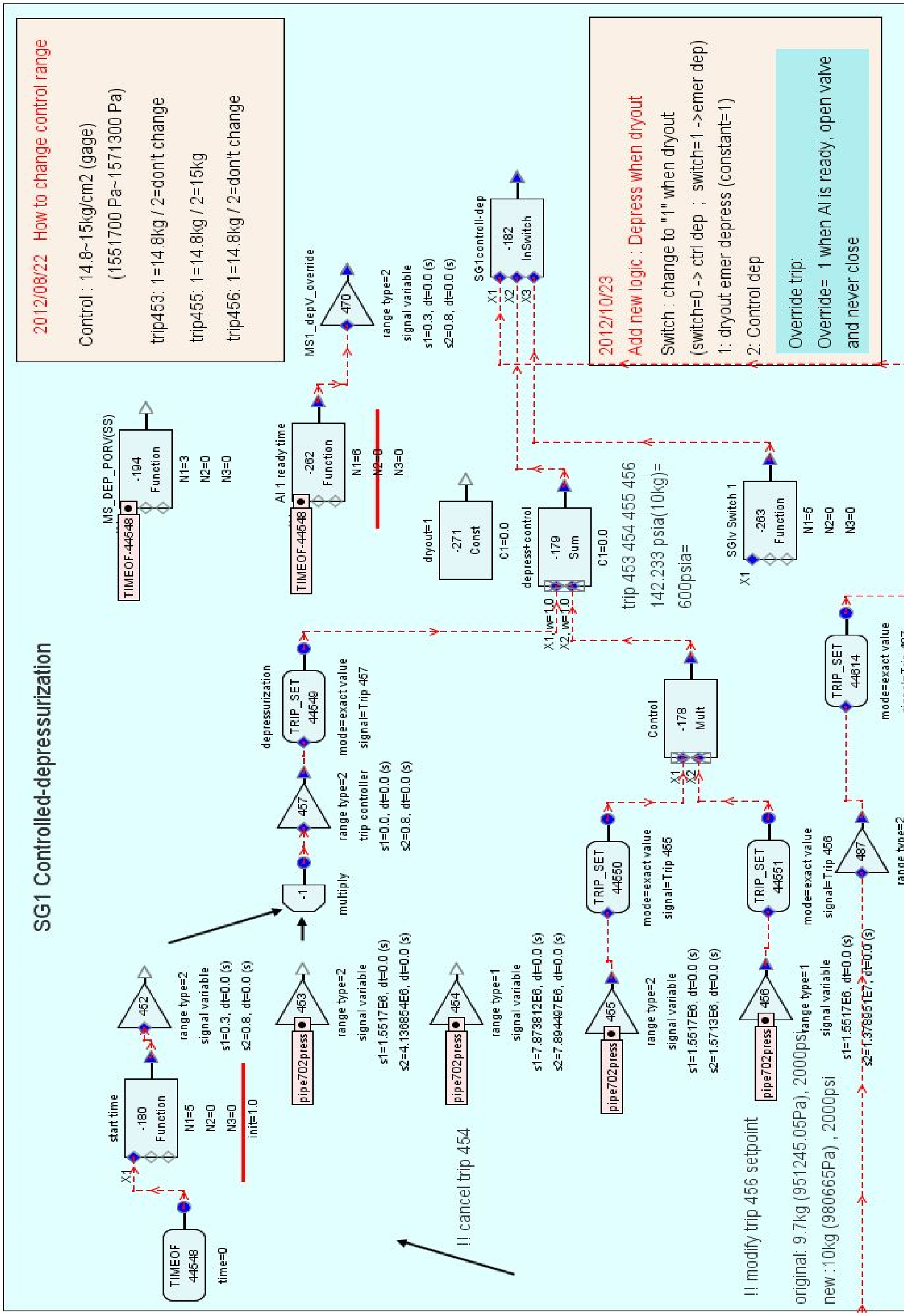


Figure 2-4 Control Block and Logic of Control Depressurization of SG 1

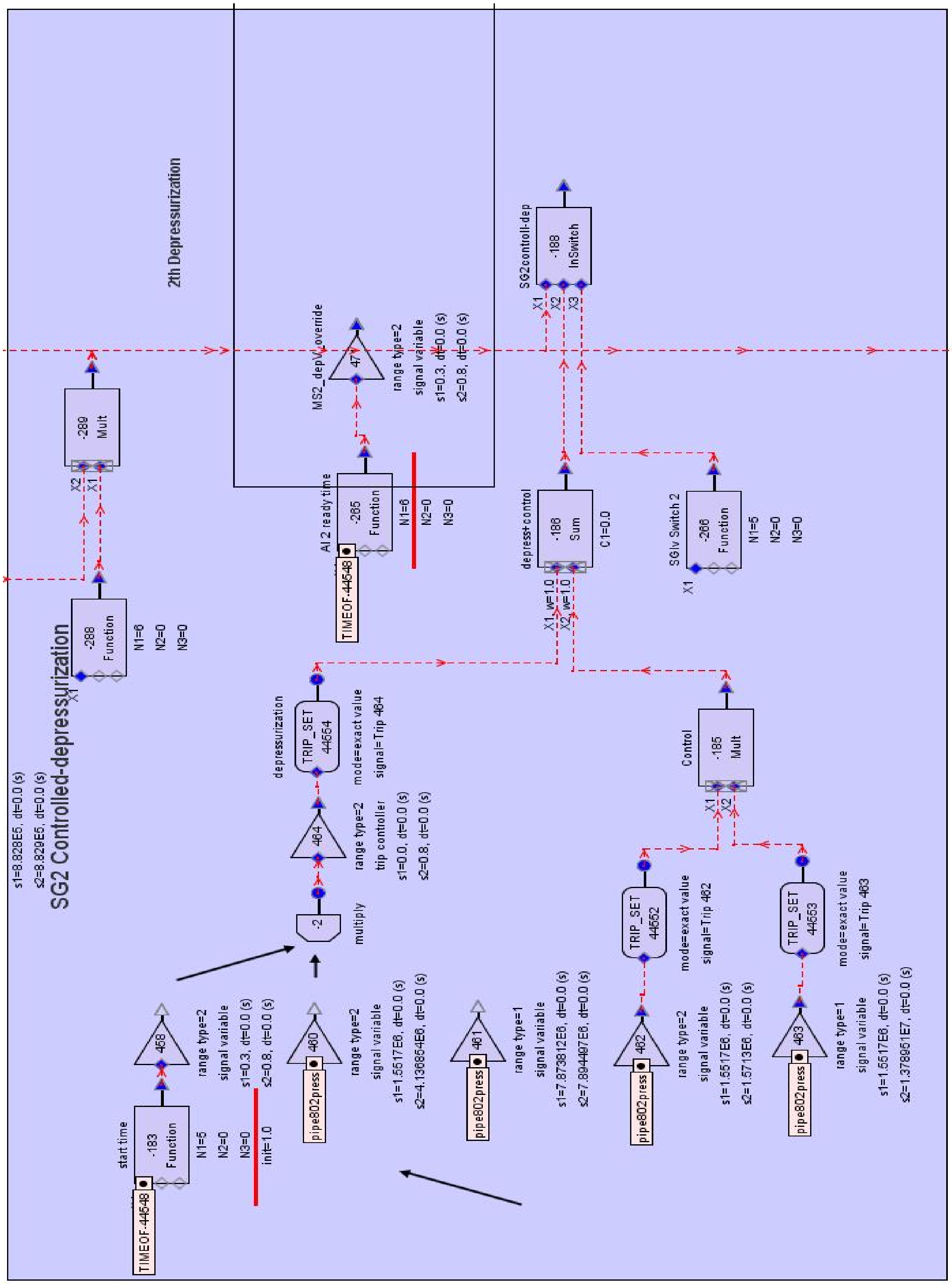


Figure 2-5 Control Block and Logic of Control Depressurization of SG 2

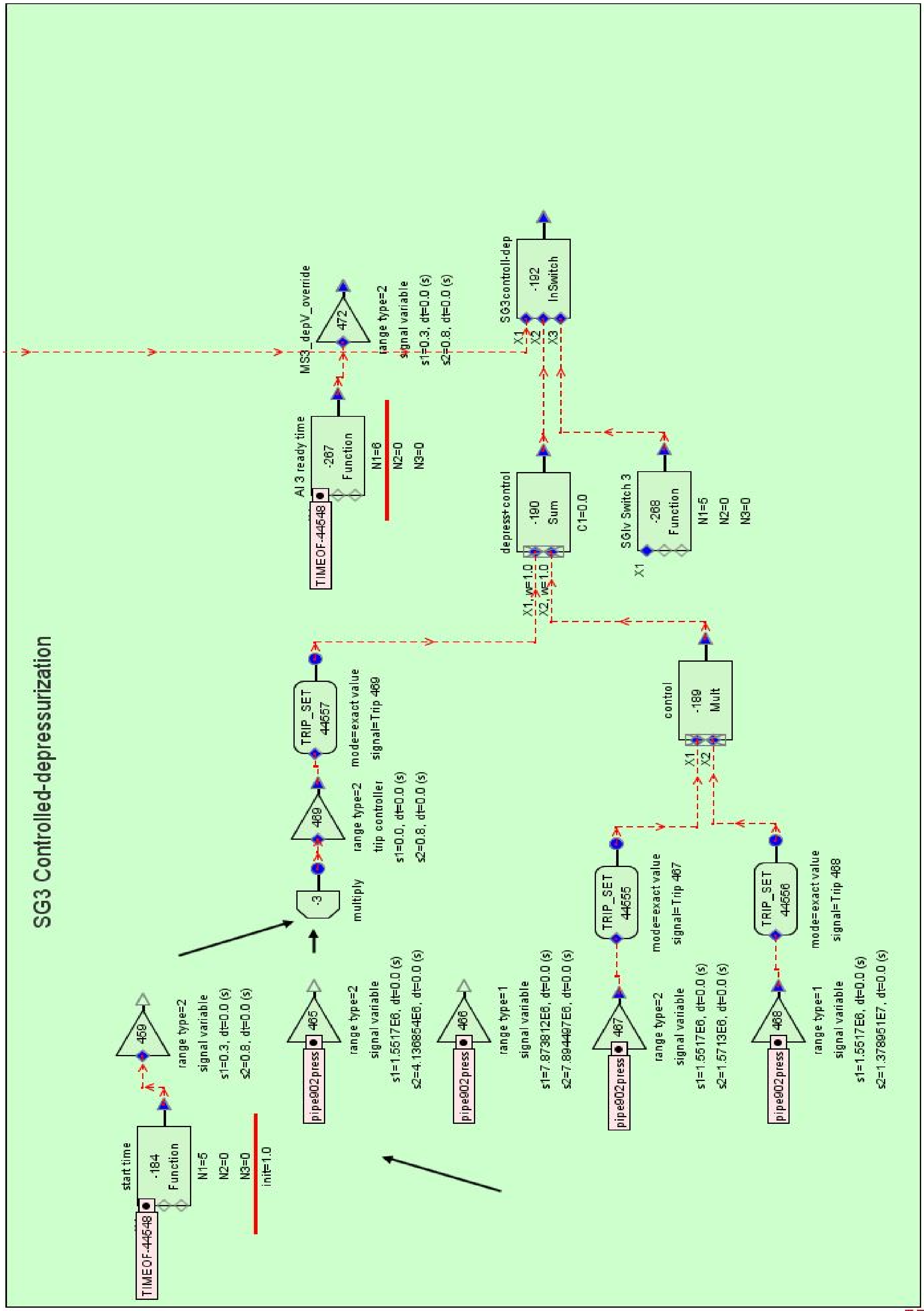


Figure 2-6 Control Block and Logic of Control Depressurization of SG 3

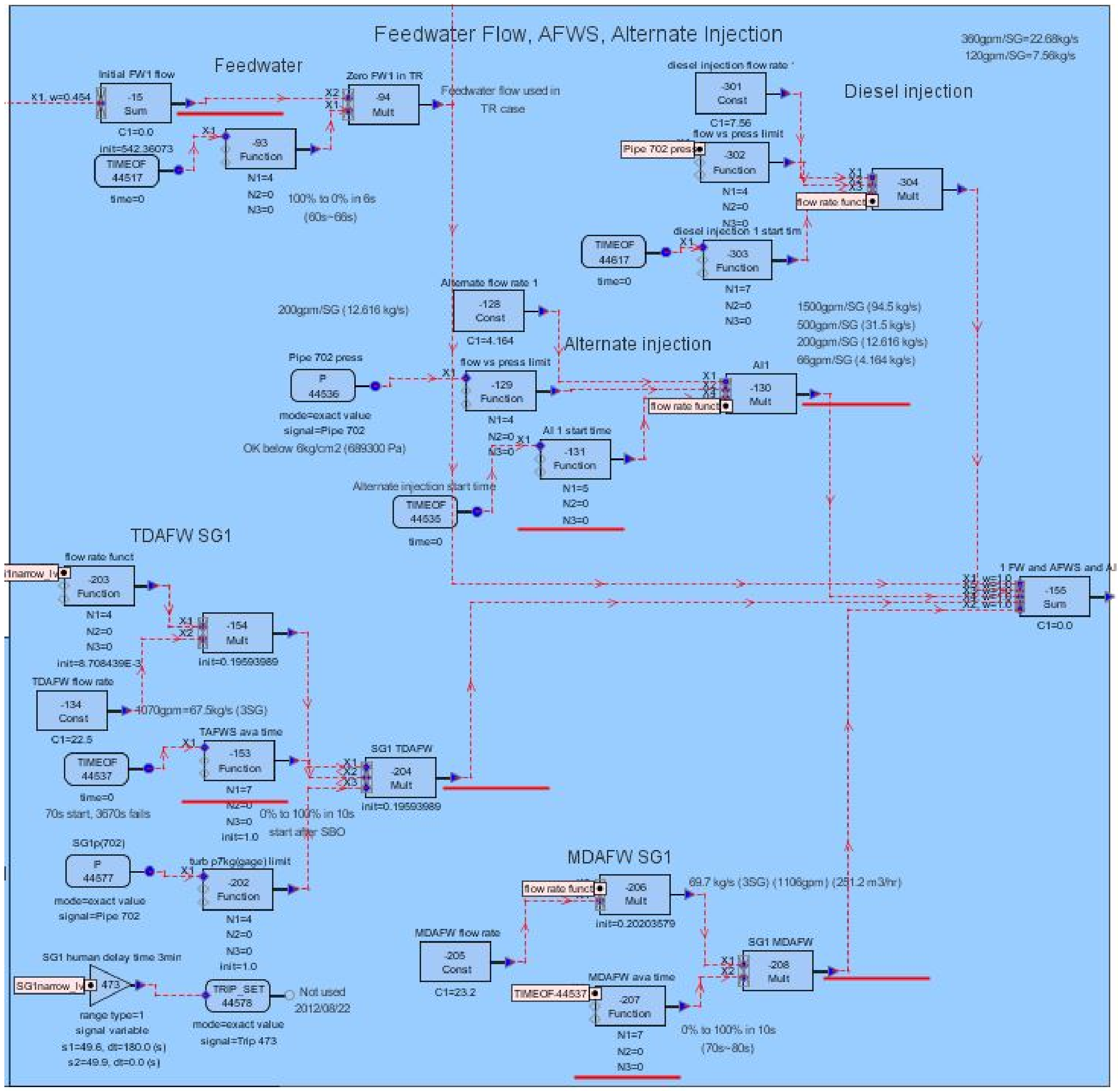


Figure 2-7 Control Block and Logic of Water Injection of Secondary Side

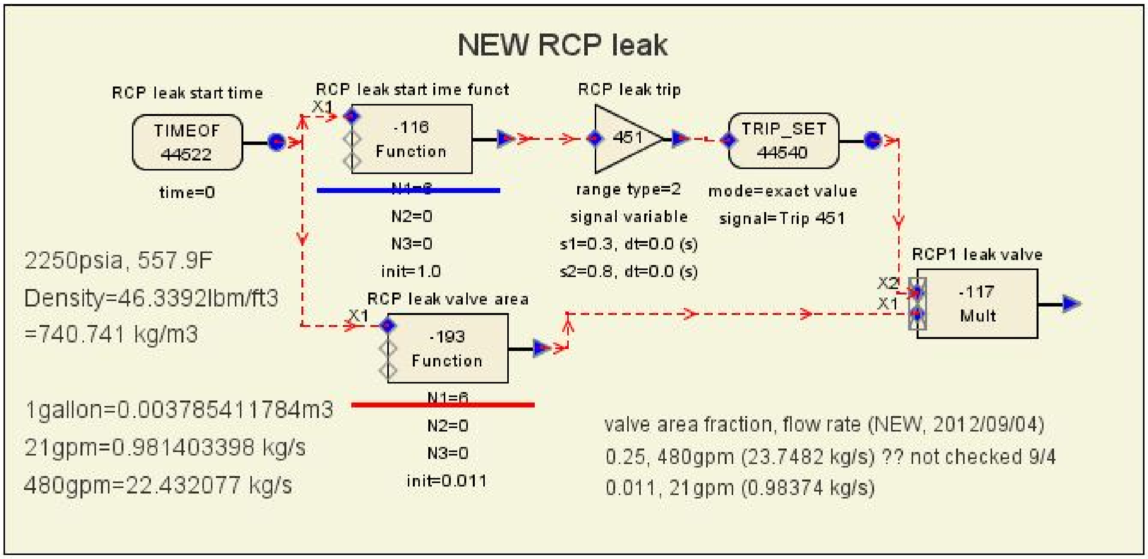


Figure 2-8 Control Block and Logic of RCP Seal Leakage

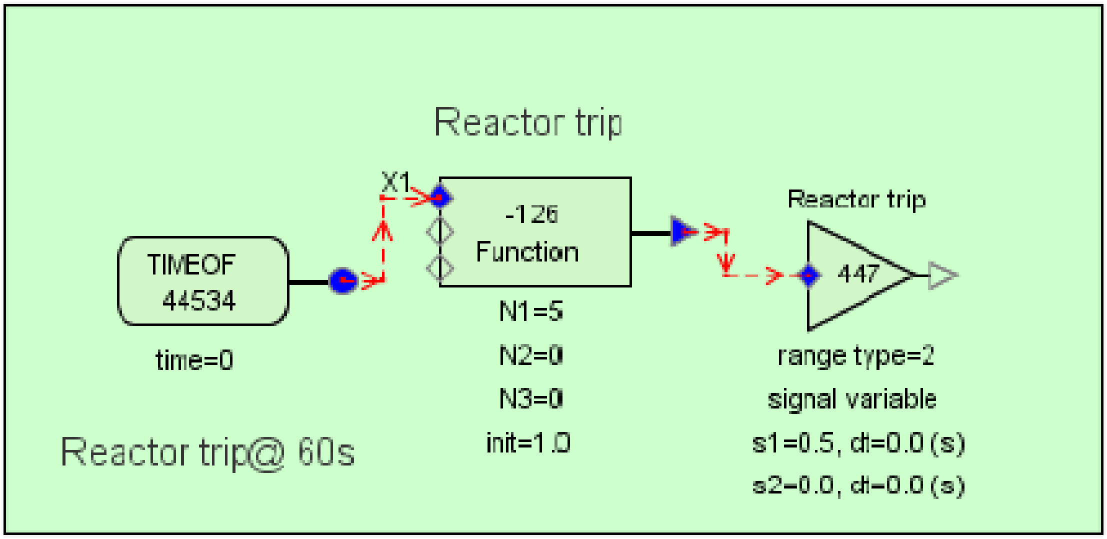


Figure 2-9 Control Block and Logic of Reactor Trip

3 METHODOLOGY

For this research, the sequence of accident must be made at first, then is the setting of input model. At last, we should analysis the result from TRACE. In order to simulate a Fukushima like accident, the SBO accident was used in this research. The definition of SBO is the loss of offsite electric power in 10CFR 50.2, U.S.NRC. Whenever a plant encounter SBO accident, operators shall follow the EOP to stabilize the plant. If the situation of plant become worse, then the SAMP is usable as well, and the evacuation of nearby resident is required if it’s necessary. In this and the following chapter, the simulation of SBO and the rescue procedures from URG and FLEX will be made. According to 10 CFR 50.46, to satisfy the safety margin of core water level and peak cladding temperature, the PCT during the whole accident should remain below 1088K (conservative value) to ensure that the fuel is integrated without any oxidation or damage.

In the following 4 cases, assume that the plant is operating stably from 0 to 60 second with full power. A hypothetical earthquake happened at 60 sec, caused the tripped of reactor, RCP and main feedwater pump. Meanwhile each loops started the seal leakage of RCP 21 gpm. MDAFW started at this moment, help the water injection of steam generators. 30 minutes after the earthquake, a tsunami triggered by the earthquake attacked the site of the plant, caused the damage of diesel engine and electricity supply equipment. The SBO of Maanshan nuclear power plant had begun. Because of the loss of AC power, MDAFW was tripped and unavailable. The TDAFW was boosted by the steam of main steam pipe, so it was supposed to work during the loss of AC power. However, in this research, it was supposed unavailable as well for conservative assumption. When the SBO happened, operators follow the EOP, execute the control depressurization of steam generator, drop the pressure of it and keep it at 15kg/cm^2. Then in the case of URG rescue, Case 1 and Case 2 respectively executes the emergency depressurization of steam generator at 3660 and 14460 sec by the operators, lower the pressure to 6kg/cm^2. After the pressure is below the working pressure, they will execute 800 gpm of water injection of steam generator. For the primary side, execute 25 gpm of water injection by the water-test pump at 3660 and 14460 sec as well.

In the cases (Case 3 and 4) of FLEX rescue, in order to compare with the URG cases, the time of mid-pressure and high-pressure water injection is also assumed at 3660 and 14460 sec. Therefore, the URG and FLEX cases are at the same water injection time point. This can help us to understand the URG and FLEX cases results. The detailed sequence is in Table 3-1 below. In order to simulate the water injection of primary side, add a FILL component at the cold leg of each loop, and give them the control logic of timing and pressure. The modified input model is showed at Fig. 3-1~3-3.

Table 3-1 Sequences of Transients

Time (sec)	Case 1	Case 2
0-60	Plant operating stably	
60	Earthquake happened, reactor tripped, main feedwater pump scrammed, 12gpm of seal leakage started, MDAFW activated, execute control depressurization	
1860	Tsunami attacked, MDAFW tripped, TDAFW supposed unavailable, SBO began	
3660	**Execute emergency depressurization on SG**, 800 gpm of water injection by fire pump on SG, 25 gpm of water injection by water-test pump on primary side	
14460		**Execute emergency depressurization on SG**, 800 gpm of water injection by fire pump on SG, 25 gpm of water injection by water-test pump on primary side
Time (sec)	**Case 3**	**Case 4**
0-60	Plant operating stably	
60	Earthquake happened, reactor tripped, main feedwater pump scrammed, 12gpm of seal leakage started, MDAFW activated, execute control depressurization	
1860	Tsunami attacked, MDAFW tripped, TDAFW supposed unavailable, SBO began	
3660	215 gpm of water injection by fire pump on SG, 40 gpm of water injection by water-test pump on primary side	
14460		215 gpm of water injection by fire pump on SG, 40 gpm of water injection by water-test pump on primary side

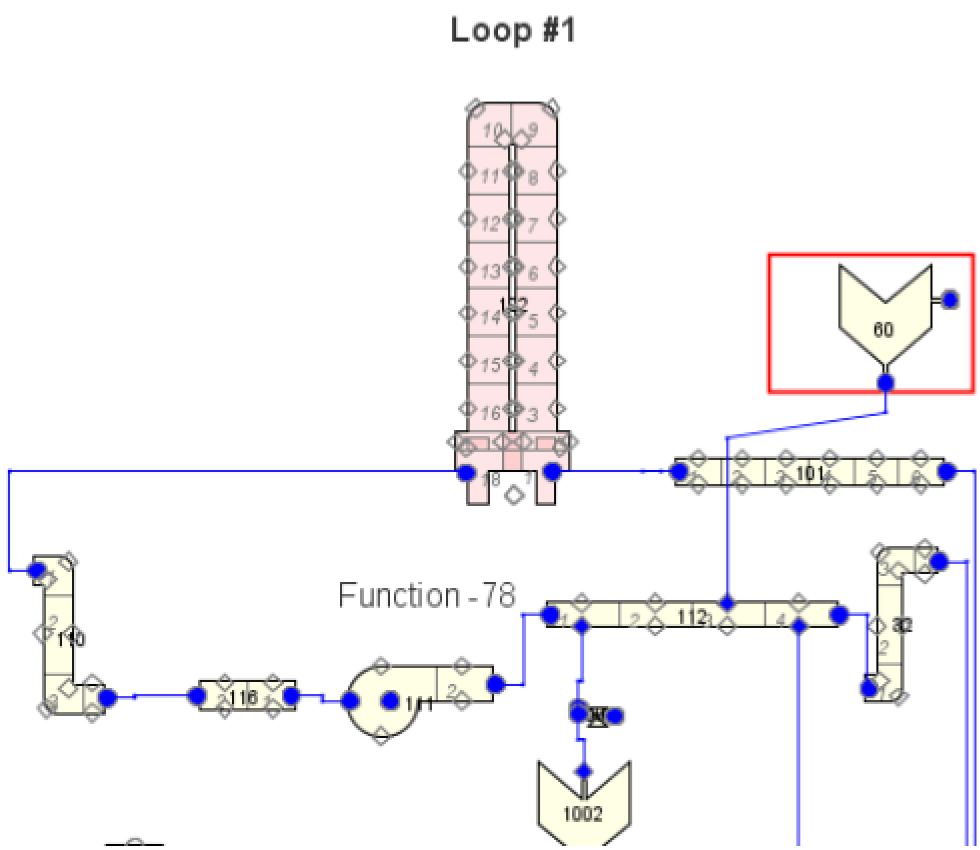


Figure 3-1 1st Loop of Input Model of TRACE

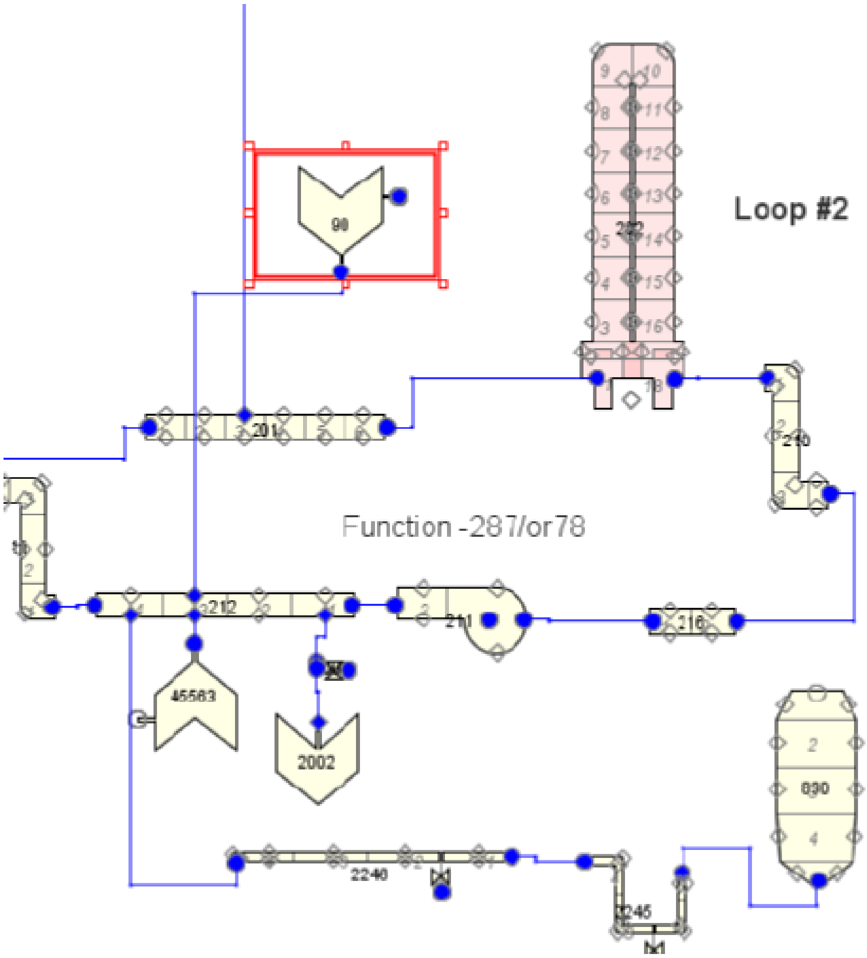


Figure 3-2 2nd Loop of Input Model of TRACE

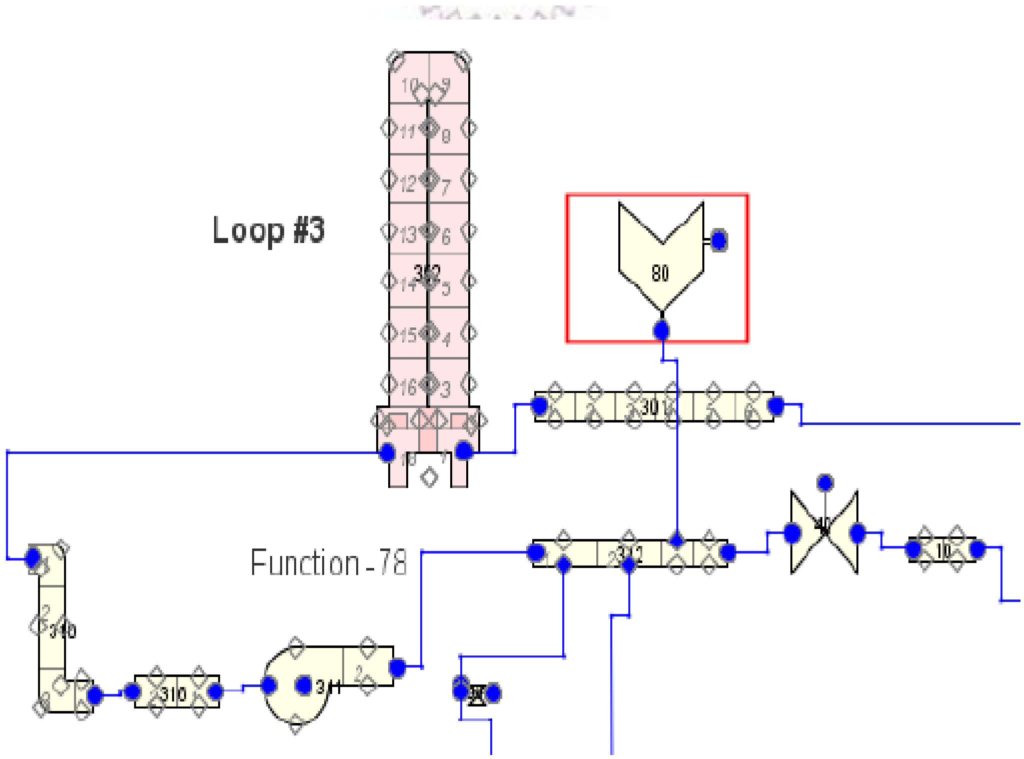


Figure 3-3 3rd Loop of Input Model of TRACE

4 SBO BASE CASE

4.1 Case 1

The earthquake happened at 60 sec, caused the tripped of reactor and reactor cooling pump. At this moment, MDAFW activated automatically, maintain the water level of pressure vessel. Meanwhile, operators opened the PORV to execute the control depressurization of steam generators, lower the pressure to 15kg/cm^2 for the following water injections. Seal leakage also started at this time, as the pressure of primary side continued decrease, the amount of leakage was decreasing obviously. At 1860 sec, the tsunami attacked site area caused the loss of offsite AC power. MDAFW tripped now and TDAFW was supposed to work however failed due to conservative assumption. Operators not only executed the emergency depressurization of steam generator, lower the pressure of them continuously, but also prepared the alternative injection of steam generator. The water injection would start whenever the pressure of steam generator is below 6kg/cm^2. From Fig. 4-1 and 4-2, it could be seen that the water level increased and pressure reached 6kg/cm2 at about 6000 sec then the water injection started. Meanwhile, the hydro-test pump of primary side activated as well, give 25gpm of water injection into primary side covering fuel rods. It is noteworthy that there is a rapid rising of pressure of primary side at about 23000 sec. The reason is the setting in the input model of TRACE. According to procedure 1451, water injection of primary side shall start after the pressure of reactor cooling system is below 7.55MPa. Thus, there are control logic of time and pressure simultaneously in this model. The water injection would start when time reaches the timing of water injection, and the pressure is lower than 7.55MPa. After the water is full in the core, the FILL component of TRACE will keep injecting cause the rise of pressure in primary side. After the pressure rise over the working pressure of hydro-test pump, the control logic would stop the injection, so the pressure of primary side would finally stabilize at 7.55MPa. The result could be seen at Fig. 4-3. After the rising if pressure in primary side, the pressure difference between inside and outside will also rise, lead to a small elevation of amount of seal leakage. The trend of seal leakage could be seen at Fig. 4-4. At last, from Fig. 4-5~4-6, it’s obvious that the coolant was always covering the top of fuel rod, and the PCT was maintain at about 400K till the end. The decay heat can be remove successfully, the core and plant remain safe.

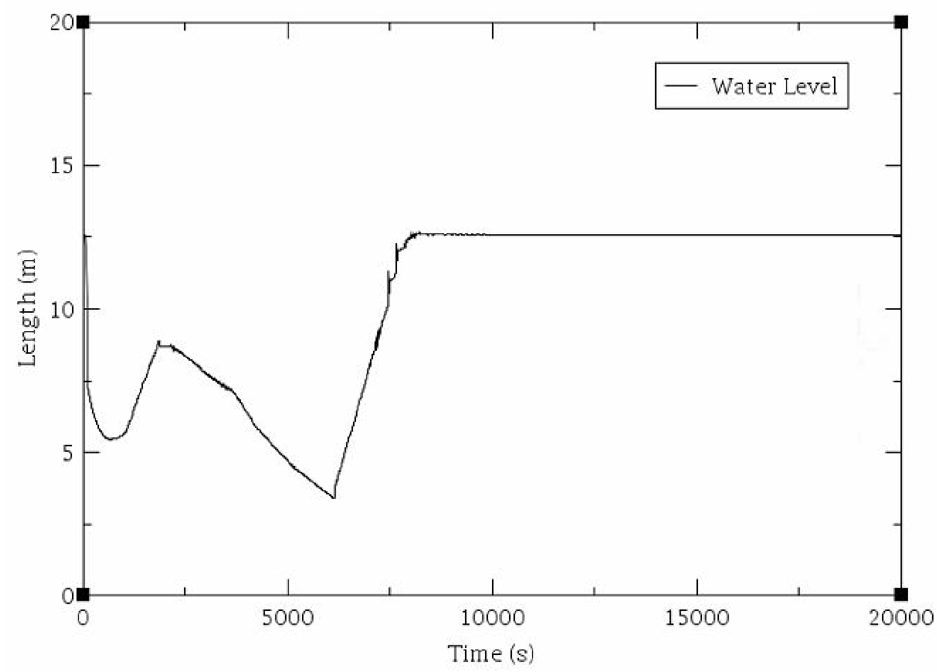


Figure 4-1 Water Level of SG in Case 1 without Tube Plugging

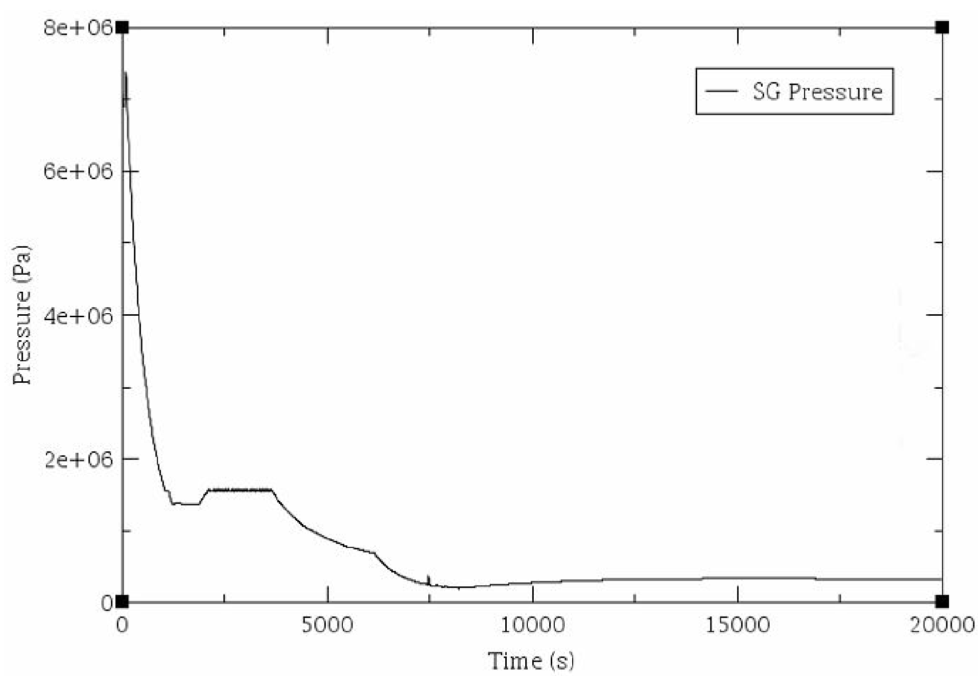


Figure 4-2 Pressure of SG in Case 1 without Tube Plugging

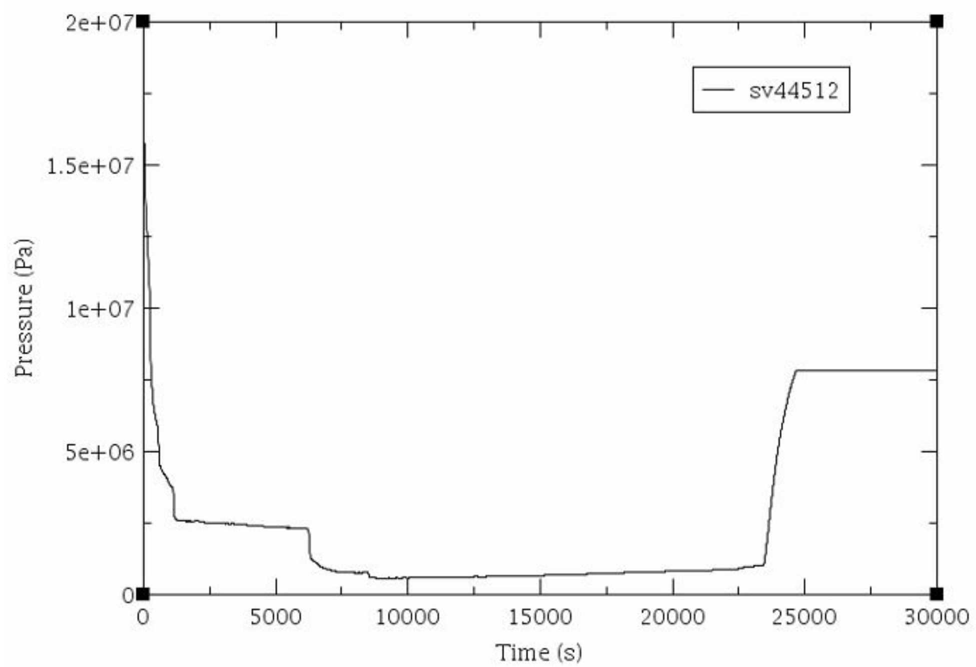


Figure 4-3 Pressure of Primary Side in Case 1 without Tube Plugging

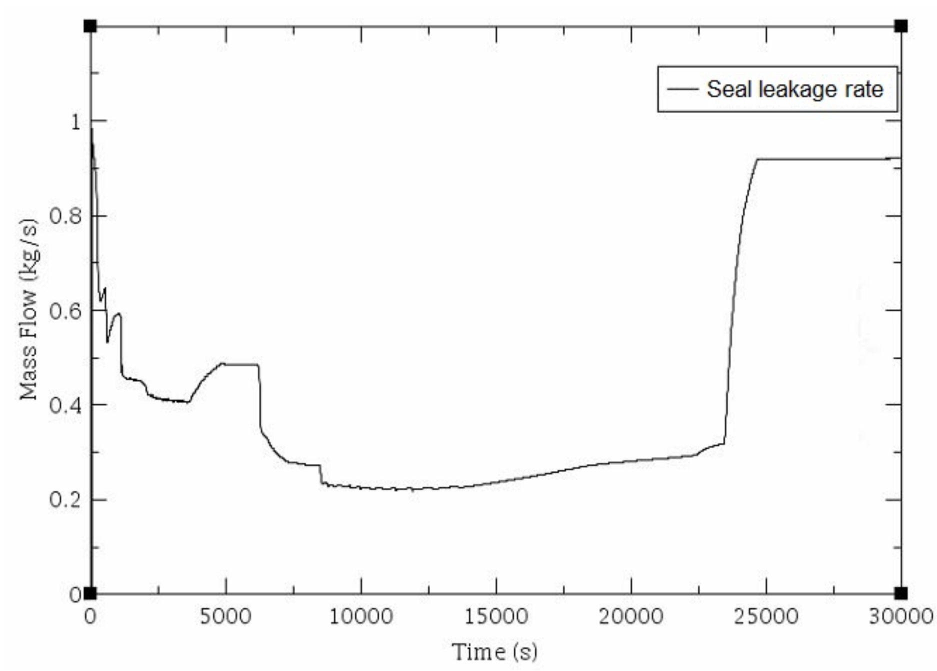


Figure 4-4 Amount of Seal Leakage in Case 1 without Tube Plugging

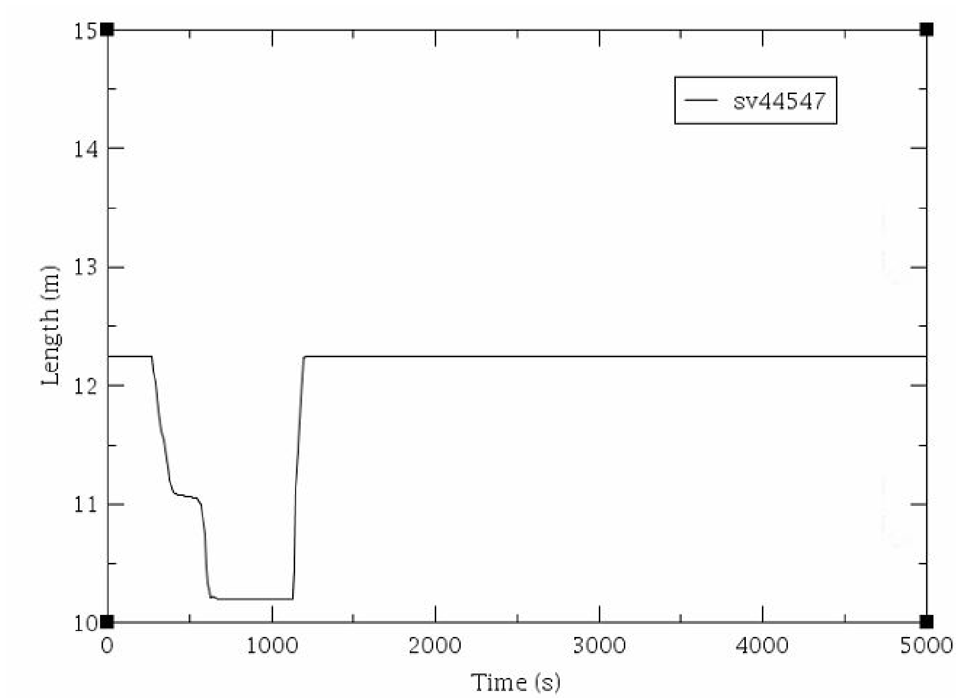


Figure 4-5 Water Level of Primary Side in Case 1 without Tube Plugging

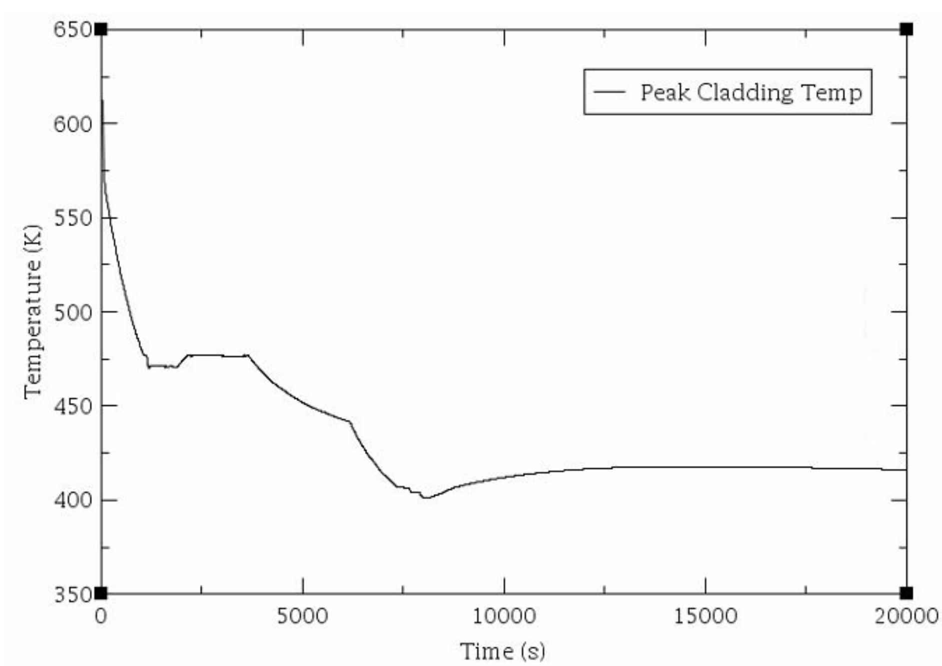


Figure 4-6 Peak Cladding Temperature in Case 1 without Tube Plugging

4.2 <u>Case 2</u>

The sequence of event in Case 2 is similar to Case 1. The only difference is that the rescue procedure at 3660 sec, was delayed until 14460 sec. First, it can be seen from Fig. 4-7 that the steam generator dried out at about 12000 sec, the reactor had lost all of the ability of heat removal from that time. Fig. 4-8 and 4-9 is the pressure and water level of primary side. Because of the loss of ability of heat removal, the pressure of primary side had start rising, and the water level of it had decreased. From Fig. 4-10, it's obvious that the procedure of depressurization finished at 15000 sec, operators started the alternative water injection for steam generator. This action successfully resumes the ability of heat removal of the plant. Thus, after the water injection, the pressure of primary side became lower again, and the water level back to full by hydro-test pump. Because of the delay of rescuing time, there're about an hour of dry out period in steam generator. Causing the vaporization of coolant water and rising of pressure. Luckily, this dry out period was short, the pressure of core was still below the criteria of opening of valve in pressurizer. If the valve of pressurizer were forced to open, then the rapid depressurization would bring a lot amount of water, causing a more complicated result. From Fig. 4-11, it could be seen that the peak cladding temperature had a little increase at 12000 sec, and decrease again because of the water injection. The final temperature was about 400K, while the plant remained safe.

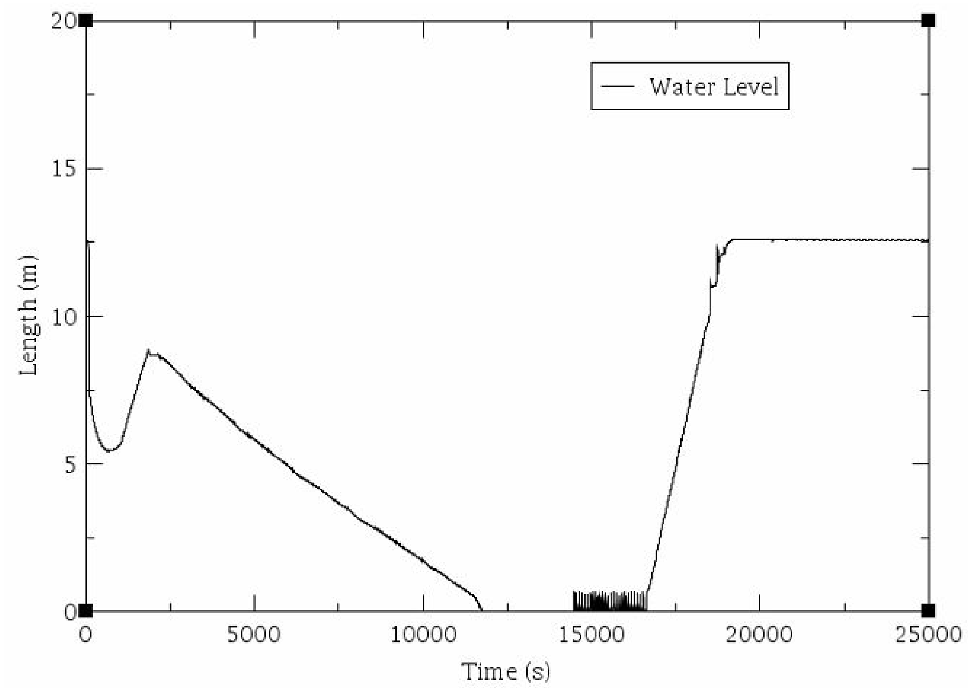


Figure 4-7   Water Level of SG in Case 2 without Tube Plugging

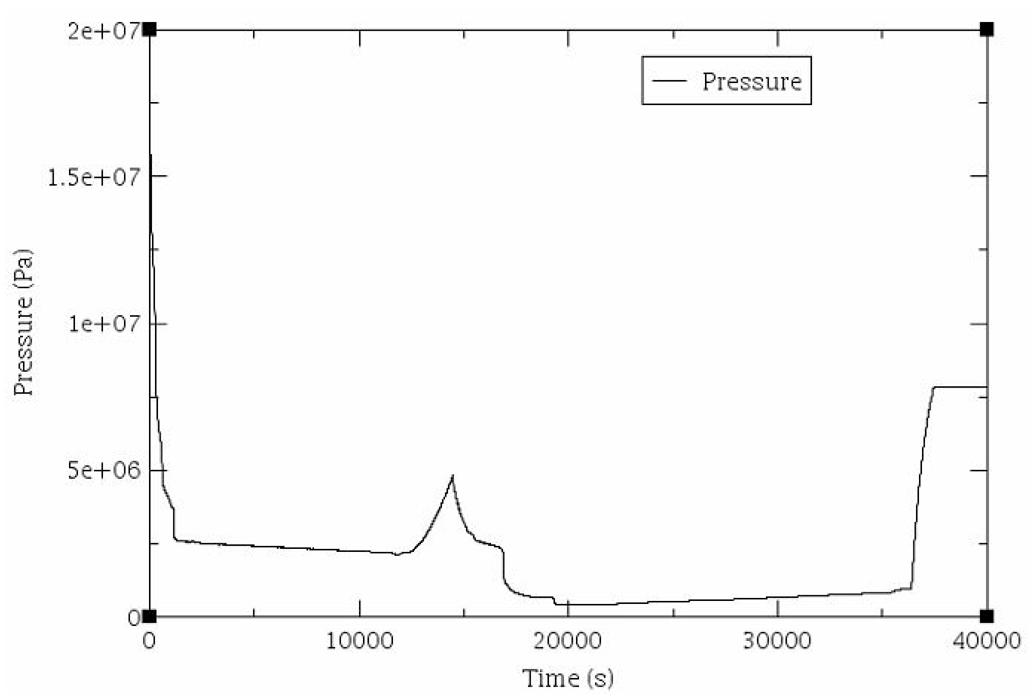


Figure 4-8   Pressure of Primary Side in Case 2 without Tube Plugging

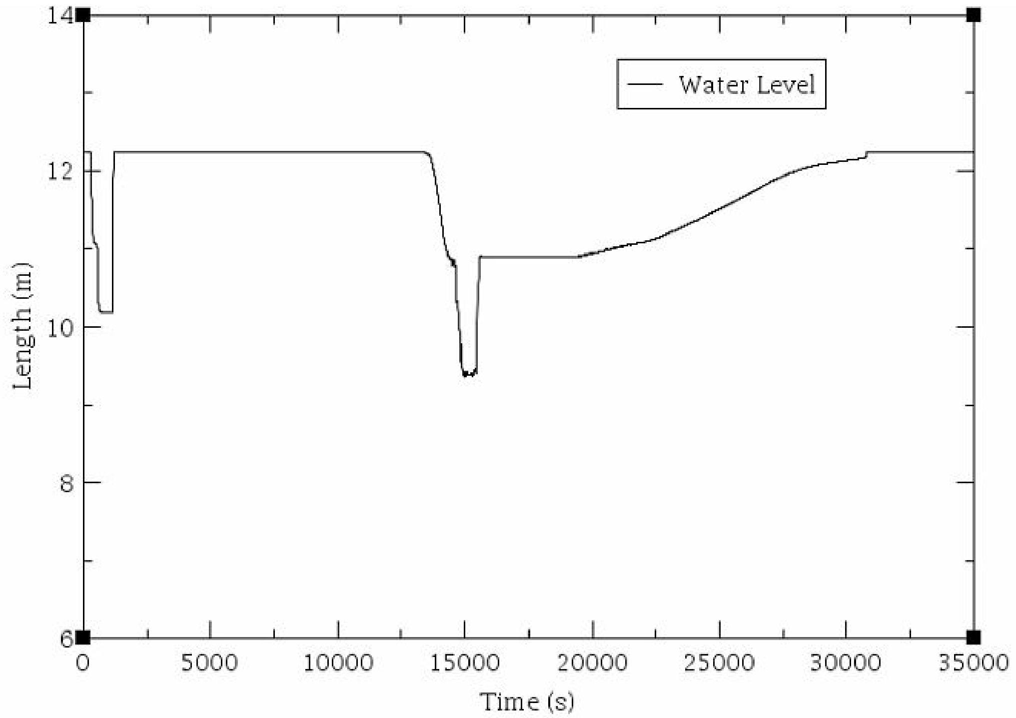


Figure 4-9 Water Level of Primary Side in Case 2 without Tube Plugging

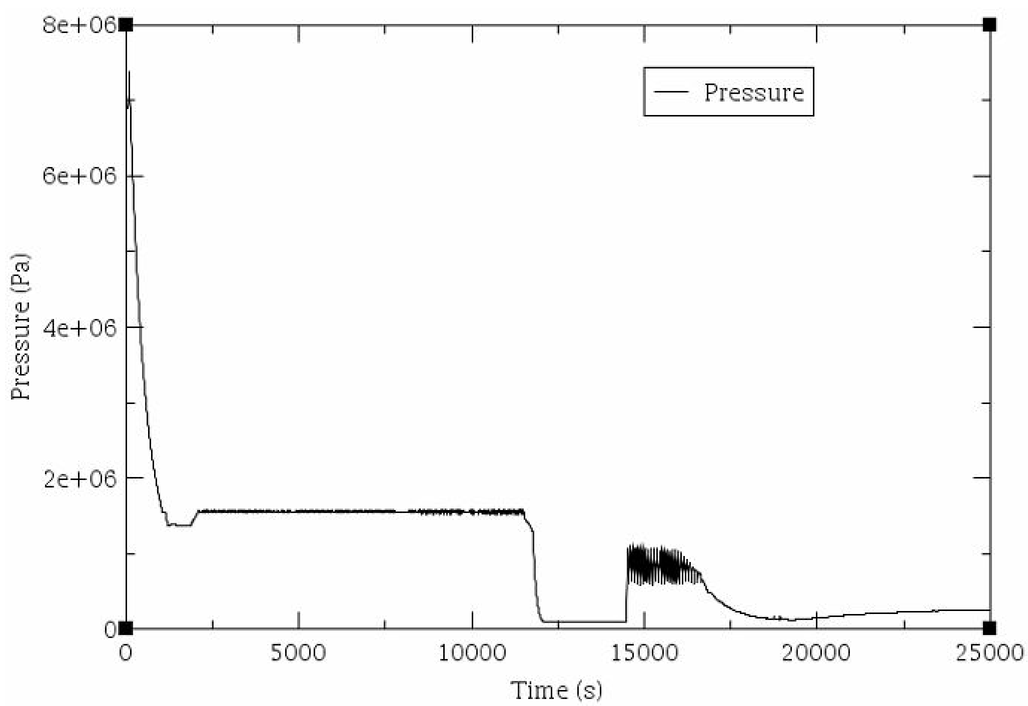


Figure 4-10 Pressure of SG in Case 2 without Tube Plugging

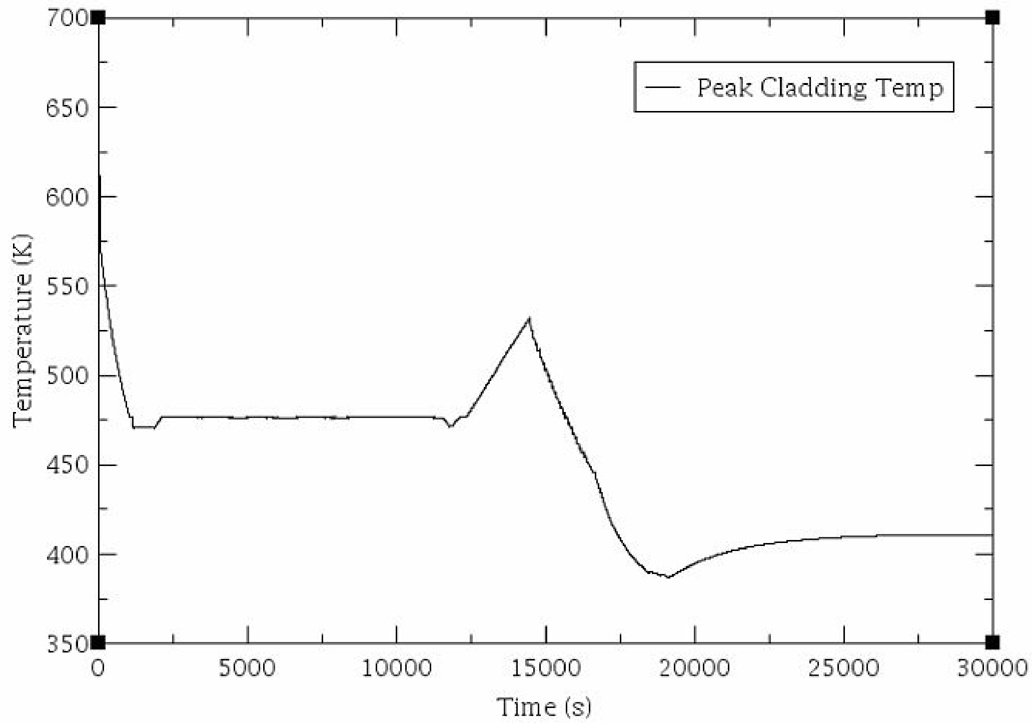


Figure 4-11 Peak Cladding Temperature in Case 2 without Tube Plugging

4.3 Case 3

The plant operated stably until 60 sec then the earthquake happened, caused the tripped of reactor and reactor cooling pump. Fig. 4-12 shows that before 1860 sec, the MDAFW keeps the water level of steam generator. The tsunami at 1860 sec caused the loss of AC power, and SBO began. At this time when MDAFW and TDAFW were both unusable, the water level of steam generator started to decrease until 7.5 m. After 3700 sec, the mid-pressure water injection activated to help the water level of steam generator. In the period, the heat transfer between primary and secondary side had never interrupted. Fig. 4-13 shows that the pressure of steam generator decreased because of the control depressurization at 60 sec, and stabilized at 1.8MPa. Without any other procedure of depressurization, the pressure had remained this value until the end of simulation. On the other side, Fig. 4-14 shows that there's no significant fluctuation in the pressure of primary side. Because of the difference of pressure between primary side and outside had decreased, the amount of seal leakage also decreased in Fig. 4-15, which means it's easier to maintain the water level in primary side. At last, when the time came to 3660 sec, the injection procedure began, 40gpm of high-pressure water injection keep the water full. In the period where the water of core was decreasing, it had never lower than the TAF, 6.67m. Fig. 4-16 shows that the water in core had always covering the fuel rods properly. As mentioned before, because of the setting of input model in TRACE, it will keep injecting water into primary after it's too full and stop when the pressure is beyond 1500psia, which is the working pressure of high-pressure water injection. Finally, Fig. 4-17 shows that under there rescue procedures, peak cladding temperature was maintained at 475 K, far from our safety margin.

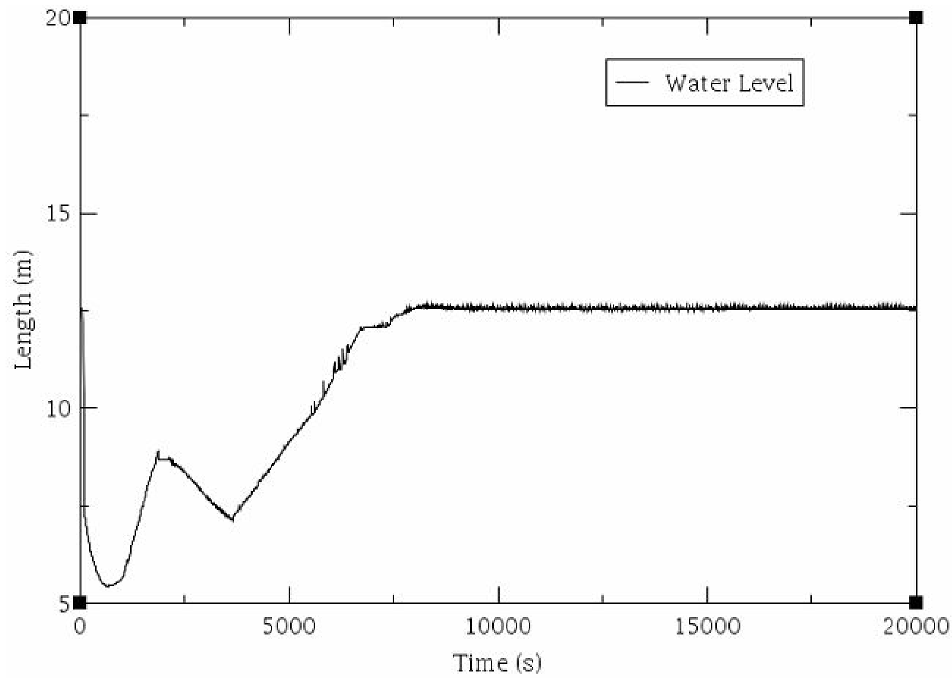


Figure 4-12 Water Level of SG in Case3 without Tube Plugging

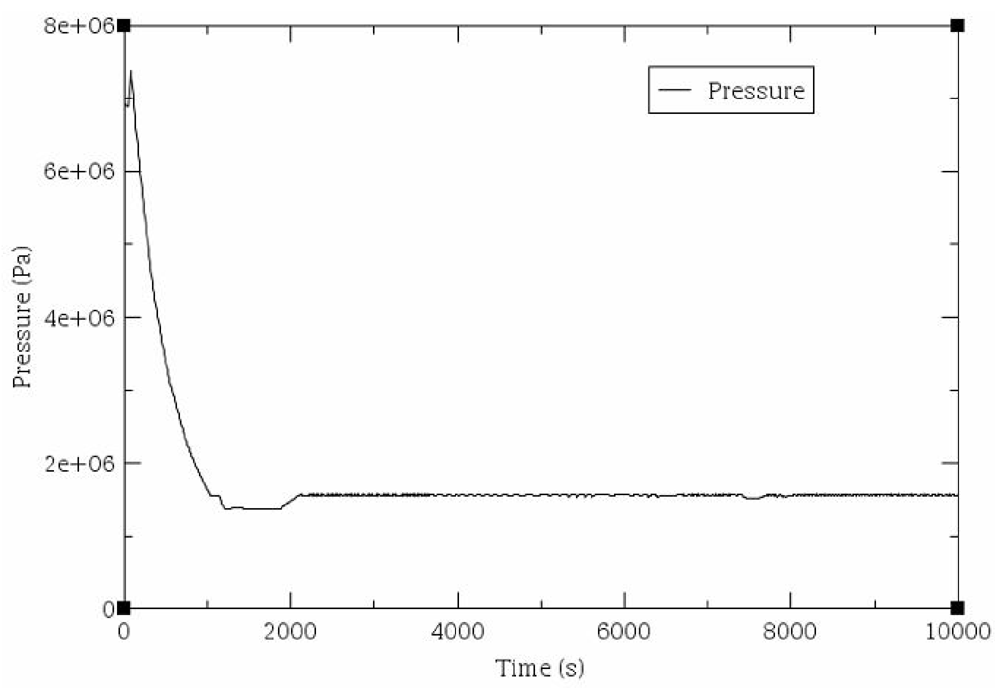


Figure 4-13 Pressure of SG in Case 3 without Tube Plugging

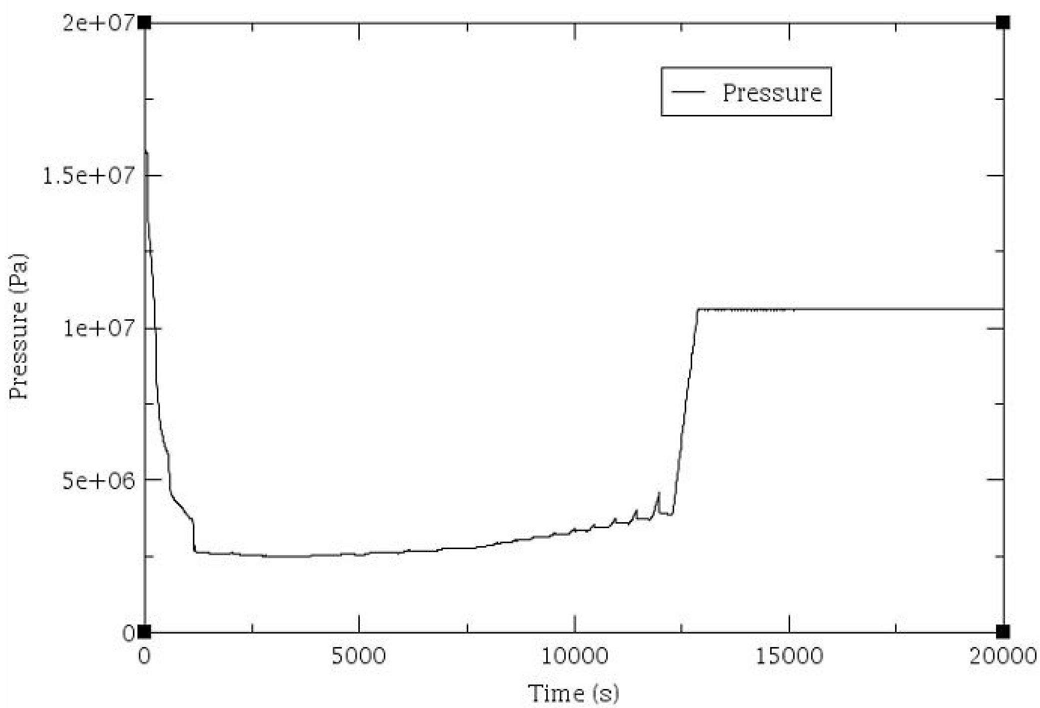


Figure 4-14 Pressure of Primary Side in Case 3 without Tube Plugging

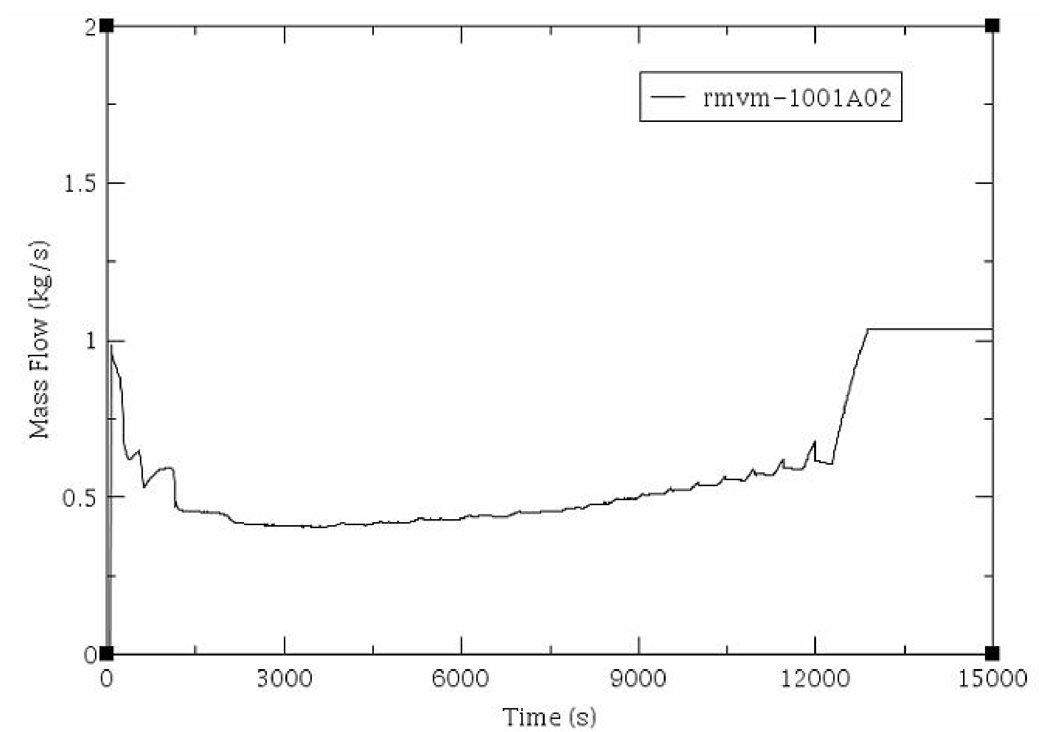


Figure 4-15 Amount of Seal Leakage in Case 3 without Tube Plugging

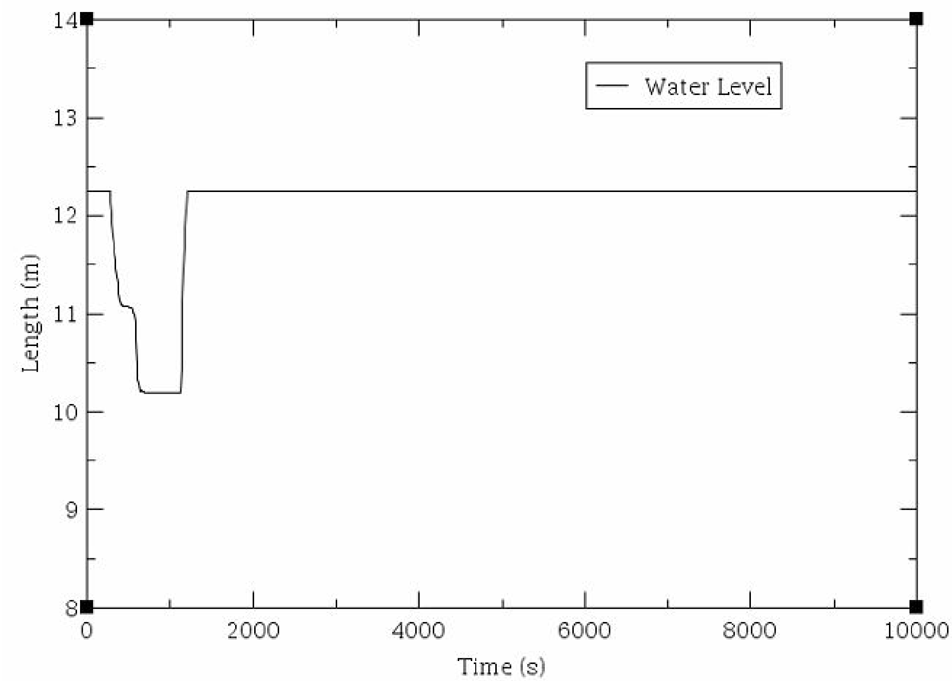


Figure 4-16   Water Level of Primary Side in Case 3 without Tube Plugging

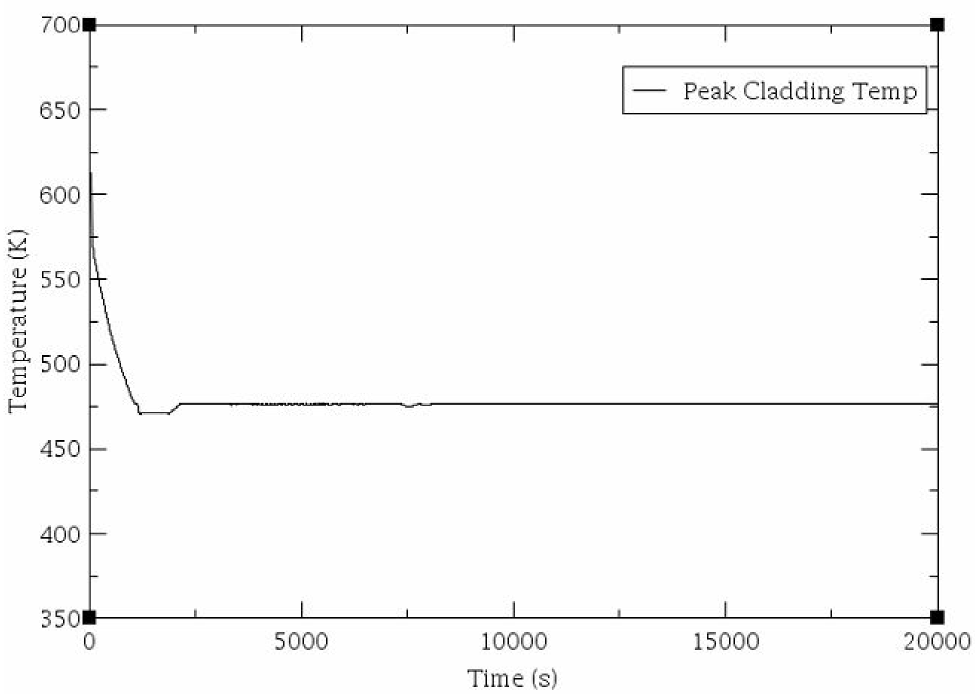


Figure 4-17   Peak Cladding Temperature in Case 3 without Tube Plugging

4.4 Case 4

The rescue procedures in Case 4 is similar to Case 3, but the water injection of Case 3 in 3660 sec was changed to 14460 sec. From Fig. 4-18, it could be seen that when SBO happened at 1860 sec, water level of steam generator was decreasing. Then dried out at 12000 sec and last for 2500 seconds. When the steam generator was dried out, the coolant in primary side started to vaporize due to the loss of heat removal. Then the vaporization of coolant had caused the rising of pressure in core. Fig. 4-20 show that the increasing of pressure in core had continued until 5MPa at 14500 sec. Meanwhile, the water level of core had decreased to 9.5 m at 15000 sec (see Fig. 4-19). Although it's still covering the fuel rod, but if the steam generator kept unavailable, the core would finally dry out. After the water injection of both side at 14460 sec by operators, the core had regained the ability of heat removal and the vaporization of coolant in primary side had stopped. Meanwhile the water level of core had also backed to full by the high-pressure water injection. Fig. 4-21 show that the peak cladding temperature had gone to 540K at 15000 sec. During the whole period of SBO, the dry out of steam generator occurred at 2.78 hours after the accident. If there's no special reason, execute water injection in 2.78 hours when an accident happened would be a safer option. In this case, the delay of water injection after 2400 seconds had caused a small rising of PCT, although this rising had not affect the safe of plant.

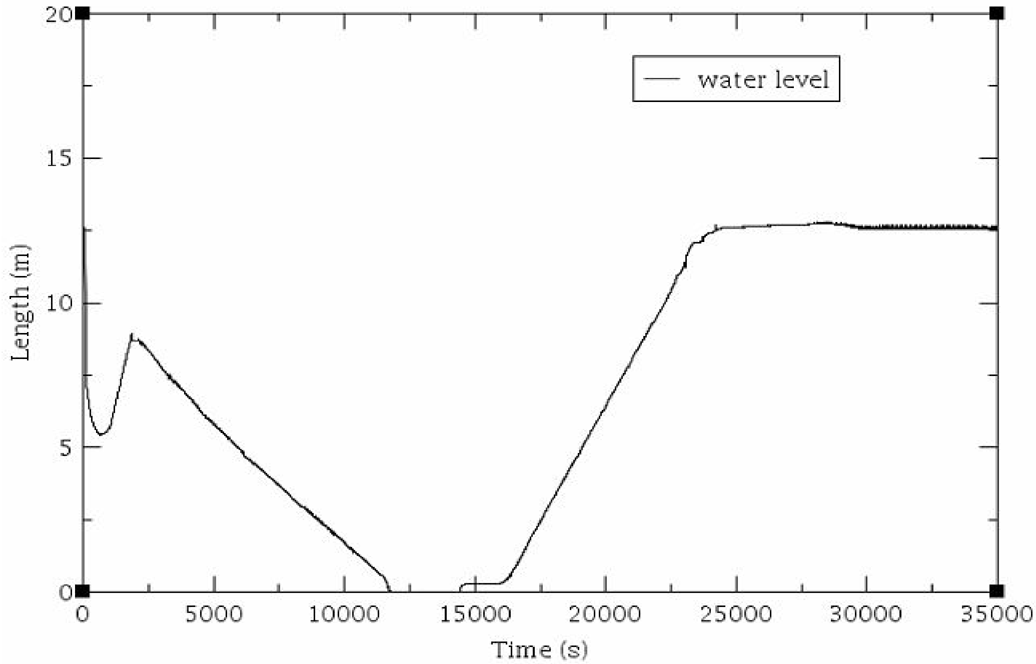


Figure 4-18 Water Level of SG in Case 4 without Tube Plugging

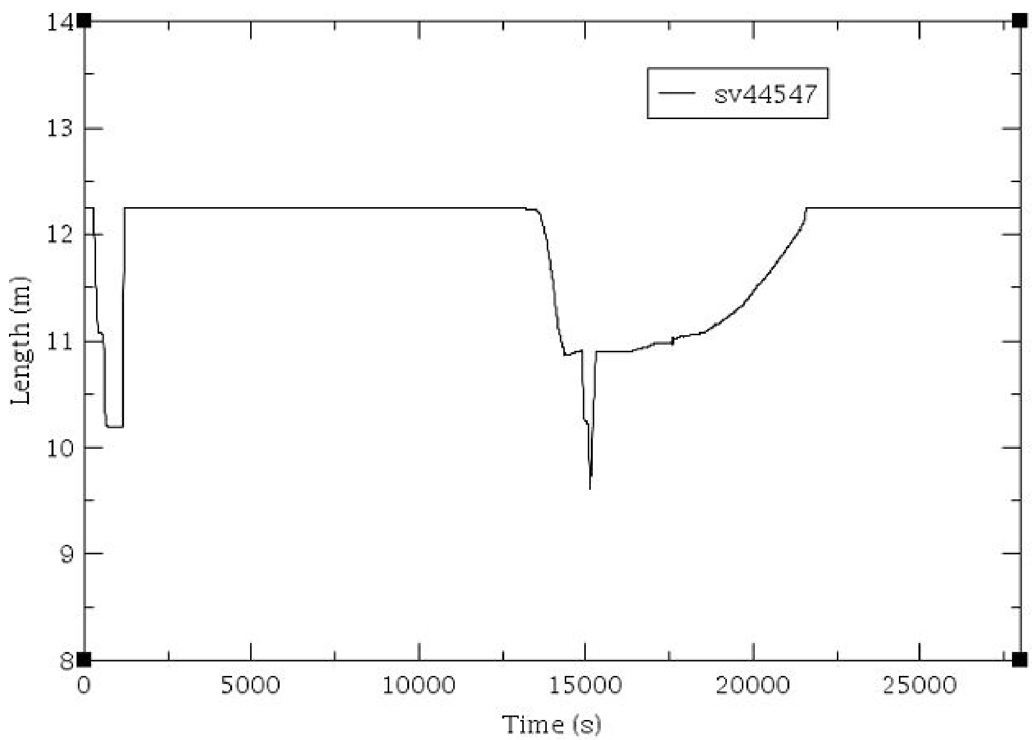


Figure 4-19 Water Level of Primary Side in Case 4 without Tube Plugging

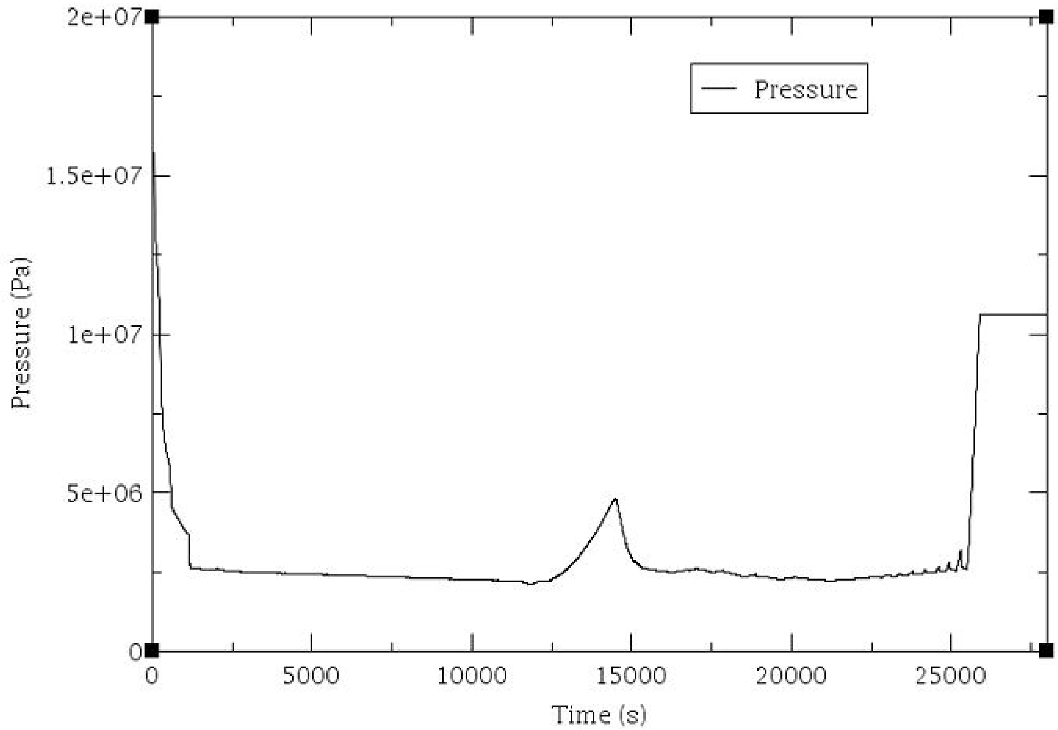


Figure 4-20 Pressure of Primary Side in Case 4 without Tube Plugging

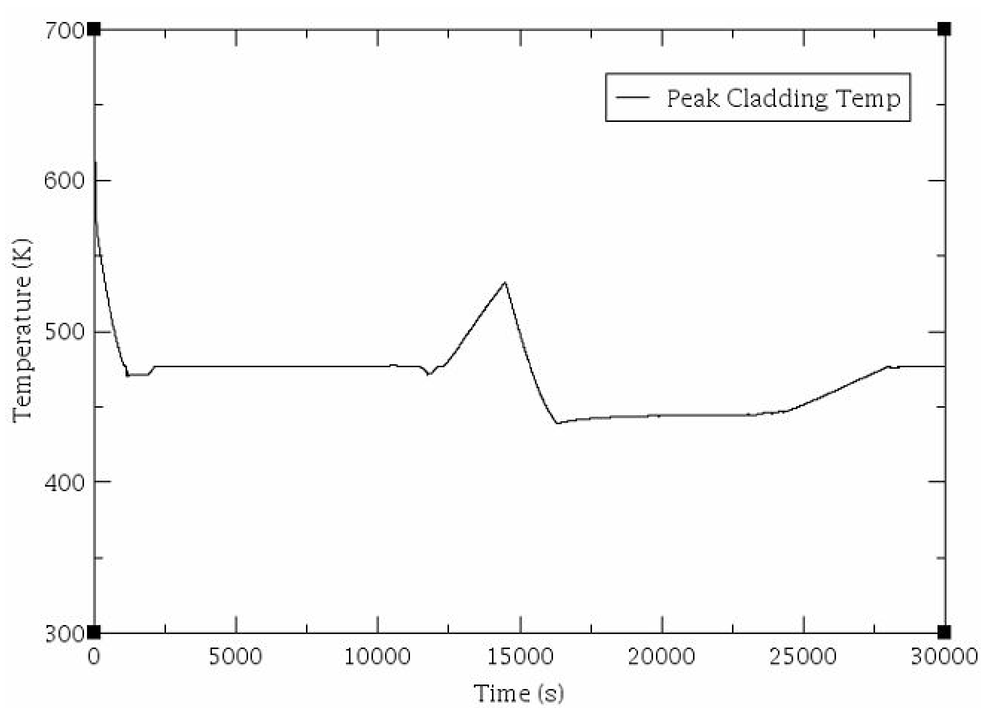


Figure 4-21 Peak Cladding Temperature in Case 4 without Tube Plugging

5 SBO CASE WITH TUBE PLUGGING

5.1 Methodology of Tube Plugging Analysis

In this section, the input model of TRACE will be modified to simulate different percentage of tube plugging. As mentioned before, the tube will gradually deteriorate as time passes by. If the defect of a tube in steam generator is over 40%, then it should be plugged for safety. According to the document of the 22th 23th outage inspection of Maanshan nuclear power plant [13, 14], the number of tube plugging in unit 1 is: S/G-A 143, S/G-B 111, S/G-C 77, and the percentage of tube plugging is: S/G-A 2.45%, S/G-B 1.97%, S/G-C 1.26%. The number of tube plugging in unit 2 is: S/G-A 107, S/G-B 60, S/G-C 39, and the percentage of tube plugging is: S/G-A 1.9%, S/G-B 1.07%, S/G-C 0.69%. Thus for conservative analysis, 2%, 5%, 10% of tube plugging will be simulated in this research. To simulate tube plugging in the input model, first step is to decrease the flow area of steam generator U tube by 98%, 95% and 90%. Then for heat exchanging area, decrease the number of pipe of HTSTR from 5624 to 5512 for 98%, 5343 for 95% and 5062 for 90%. Fig. 5-1 is the U tube of input model. Table 5-1 and 5-2 are the setting of HTSTR and Table 5-3 is the detail value of setting.

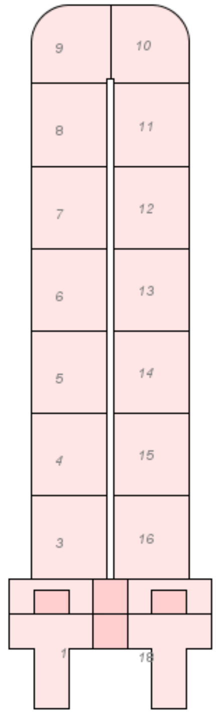


Figure 5-1   Steam Generator U Tube in the TRACE Input Model

Table 5-1 Input Parameters of U Tube in the TRACE Input Model

Cell Number	Volume(m^3)	Length(m)	Flow Area(m^2)	DZ(m)
1	2.8415956	1.063752	2.6712952	0.531876
2	1.4248896	0.5334	2.671334	0.5334
3	1.3256418	1.2583973	1.0534367	1.2583973
4	1.3256418	1.2583973	1.0534367	1.2583973
5	1.3256418	1.2583973	1.0534367	1.2583973
6	1.3256418	1.2583973	1.0534367	1.2583973
7	1.3256418	1.2583973	1.0534367	1.2583973
8	1.3256418	1.2583973	1.0534367	1.2583973
9	1.3256418	1.2583973	1.0534367	0.62919864
10	1.3256418	1.2583973	1.0534367	-0.62919864
11	1.3256418	1.2583973	1.0534367	-1.2583973
12	1.3256418	1.2583973	1.0534367	-1.2583973
13	1.3256418	1.2583973	1.0534367	-1.2583973
14	1.3256418	1.2583973	1.0534367	-1.2583973
15	1.3256418	1.2583973	1.0534367	-1.2583973
16	1.3256418	1.2583973	1.0534367	-1.2583973
17	1.4248896	0.5334	2.671334	-0.5334
18	2.8415956	1.063752	2.6712952	-0.531876

Table 5-2 Input Parameters of HTSTR in the TRACE Input Model

Axial Nodes	14 Axial Cells
Critical Heat Flux	AECL_IPPE
Fuel Rod Option	Not Fuel Rod
Axial Plane	Z-Direction
Geometry	Cylindrical
Radial Geometry	Radial Nodes
Initial Temperature	Temperature 14, 3
Liquid Level Tracking	False
Axial Conduction	True
Pitch to Diameter Ratio	1.33
Metal Water Reaction	off
Fuel-Clad Interaction	dynamic gas-gap model is off
Max. FCI Calculations	0
Fine Mesh Reflood	False
Maximum Axial Nodes	15
Minimum Node Distance	5.0E-3
Gas Gap HTC	0
Stand Alone Supplemental Rods	False
Surface Multiplier	5624

Table 5-3　Detailed Valve in the Setting of Tube Plugging

%	Flow area of U tube（m^2）	Number of tube of U tube
0%	1.05344	5624
2%	1.032368	5512
5%	1.000765	5343
10%	0.948093	5062

5.2 Case 1 with Tube Plugging

In the tube plugging analysis of Case 1, curves with different percentages of tube plugging of one parameter was put in a figure, so it's easier to compare the influence of tube plugging. Fig. 5-2 is the water level of steam generator. A little fluctuation in the beginning of transient, rising and declining as the activated and tripped of MDAFW. It could be noticed that different percentage of tube plugging affect nothing with the water level of steam generator. Fig. 5-3 is the trend of steam generator pressure. Control pressurization began 60 seconds after the transient, and emergency depressurization began at 3660 sec when water injection is ready. In figure could be seen that no significant difference between each curves. Fig. 5-4 is the temperature of steam generator U tube. One can figure out if tube plugging has any effect on the heat exchanging of both side by the change of temperature. From the curves, there's no big change on temperature in different percentage of tube plugging. In Fig. 5-5, water level of primary side, there's a little bit difference at the timing when the water was being injected. However, the difference was so tiny that could be neglected. In Fig. 5-6, it's obvious that that pressure of primary side in 10% tube plugging is about 500 seconds earlier when it raised because of water injection and then back to stable like other percentage of tube plugging. With these figure of parameters, it could say that even under 10% of tube plugging when plant encounter SBO accident, there's no influence on the URG procedure by tube plugging. Fig. 5-7 is the PCT of whole event. The PCT in different percentage of tube plugging are all the same, which means that even under 10% of tube plugging can the URG procedure become effective and bring the plant back to safe. To see the detail difference of parameters, three different moments of these parameters had been chosen in Table 5-4: before control depressurization (50 sec), after control depressurization (3000 sec), after emergency depressurization (7000 sec). It's easy to say that the difference of these parameter is about 0.5% to 1%. The small difference has no influence on these thermal hydraulic parameters.

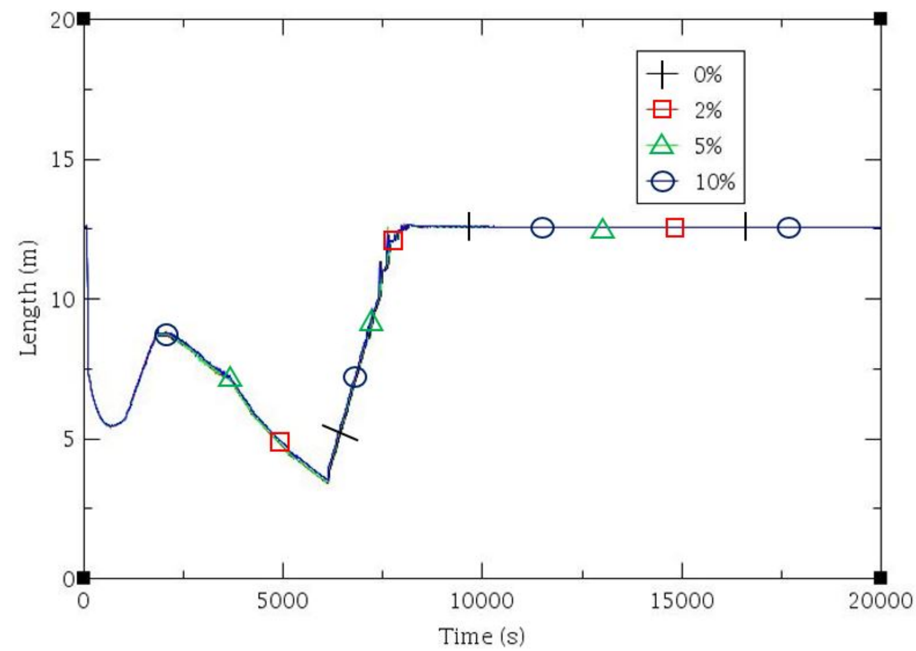


Figure 5-2 Water Level of SG in Case 1 with Tube Plugging

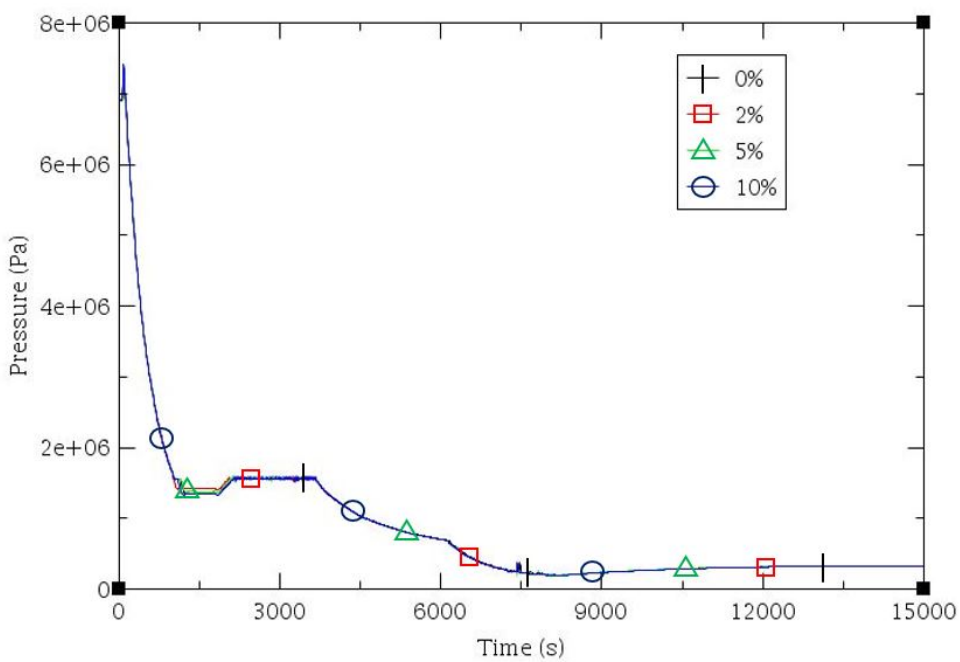


Figure 5-3 Pressure of SG in Case 1 with Tube Plugging

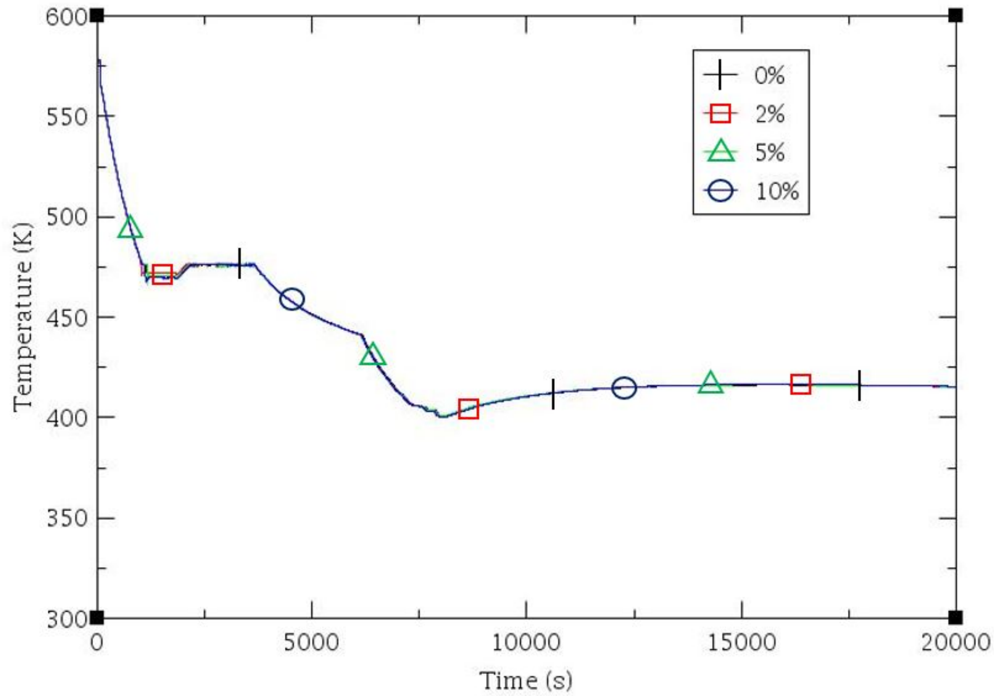


Figure 5-4 Temperature of U Tube in Case 1 with Tube Plugging

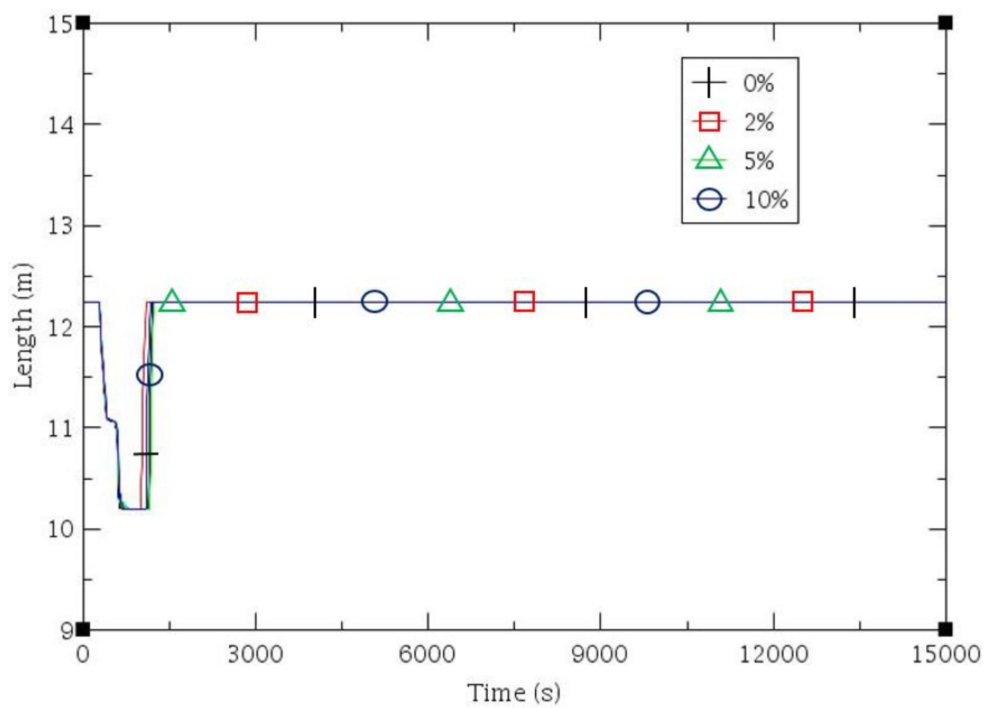


Figure 5-5 Water Level of Primary Side in Case 1 with Tube Plugging

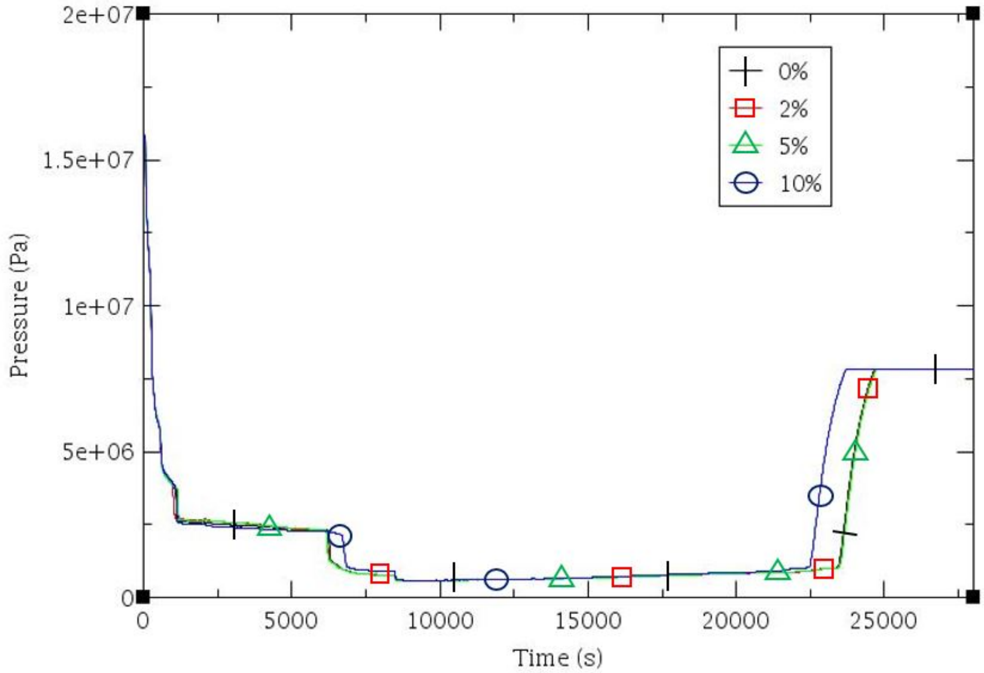


Figure 5-6 Pressure of Primary Side in Case 1 with Tube Plugging

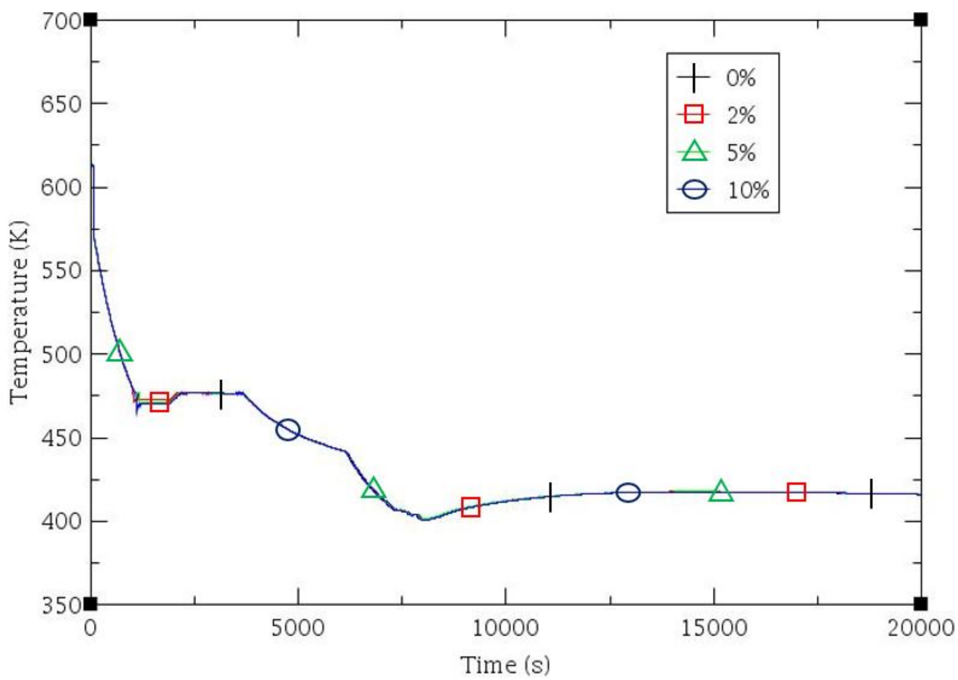


Figure 5-7 Peak Cladding Temperature in Case 1 with Tube Plugging

Table 5-4 Thermal Hydraulic Parameters in Case 1

Percentage	0%	2%	5%	10%
Peak cladding temperature(K)				
Before control depressurization(50 sec)	611.986	612.15	612.408	612.882
After control depressurization(3000 sec)	476.522	476.538	476.556	476.708
After emergency depressurization(7000 sec)	414.005	413.34	413.347	413.294
Pressure of primary side(MPa)				
Before control depressurization(50 sec)	15.727	15.744	15.772	15.83
After control depressurization(3000 sec)	2.492	2.581	2.58	2.407
After emergency depressurization(7000 sec)	0.885	0.874	0.878	1.011
Temperature of U tube(K)				
Before control depressurization(50 sec)	576.64	576.9	577.3	578.023
After control depressurization(3000 sec)	476.03	475.92	475.88	476.01
After emergency depressurization(7000 sec)	413.33	412.94	412.823	412.831
Pressure of steam generator(MPa)				
Before control depressurization(50 sec)	6.903	6.904	6.905	6.905
After control depressurization(3000 sec)	1.56	1.57	1.564	1.574
After emergency depressurization(7000 sec)	0.316	0.316	0.314	0.31
Water level of U tube(m)				
Before control depressurization(50 sec)	12.59	12.59	12.58	12.58
After control depressurization(3000 sec)	7.75	7.75	7.75	7.85
After emergency depressurization(7000 sec)	8.02	8.15	8.18	8.27

5.3 Case 3 with Tube Plugging

First, Fig. 5-8 is the water level of steam generator. MDAFW activated at 60 sec to keep it high, and tripped at 1860 sec lead to the decreasing of it. Until the water injection at 3660 sec, the variety between different percentages of tube plugging is small. Fig. 5-9 is the pressure of steam generator, without the emergency depressurization, the pressure remained steady after control depressurization and the curves were nearly the same. Fig. 5-10 is the water level of primary side, it's easy to see that 0% and 2% were a little faster when it raised. Fig. 5-11 is the temperature of U tube, it was supposed to observe the influence by temperature changing while there's no significant difference between them. The pressure of primary side in Fig. 5-12 is similar to plugged Case 1. The difference only happened at the timing when the pressure raised by control logic, while there's no difference at any moment before. Fig. 5-13 is the PCT in the transient. During the whole accident, there's no difference between these curves of tube plugging. With these analysis, it shall prove that when the percentage of tube plugging in Maanshan nuclear power plant is at most 10%, the plugged tube has no influence on the FLEX rescue procedure, and the PCT is still within our safety margin. Similarly, for a clearer view of difference between each curves, Table 5-5 was made as before. Although there's no emergency depressurization in this case, three timing were still chosen to see the parameters at these moments. It's obvious that difference of every parameter are about 0.5% to 1% and cause nothing to our rescue procedure of plant.

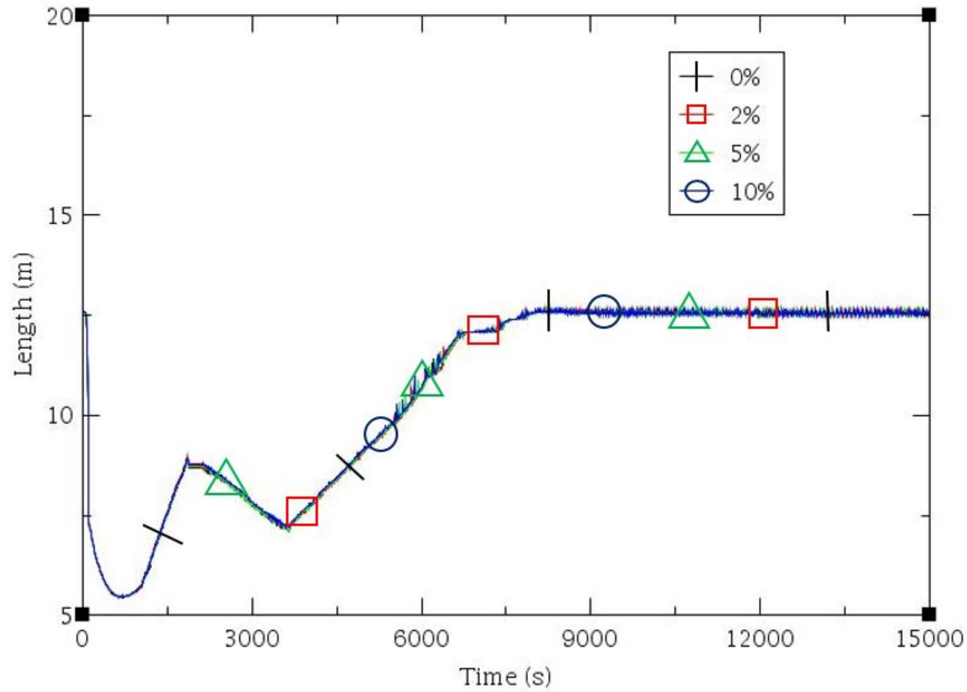


Figure 5-8 Water Level of SG in Case 3 with Tube Plugging

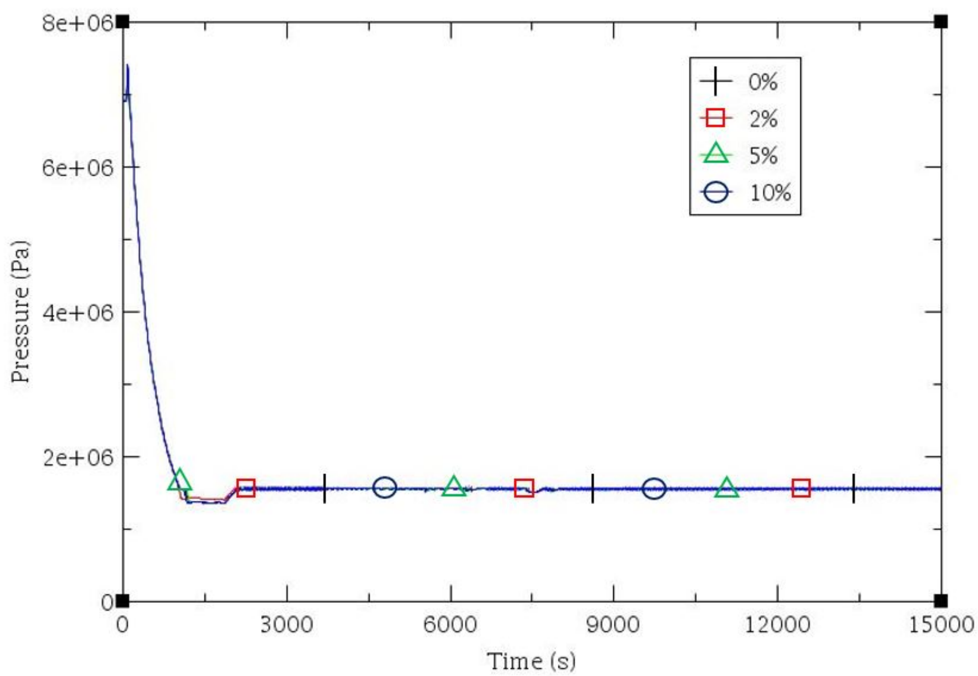


Figure 5-9 Pressure of SG in Case 3 with Tube Plugging

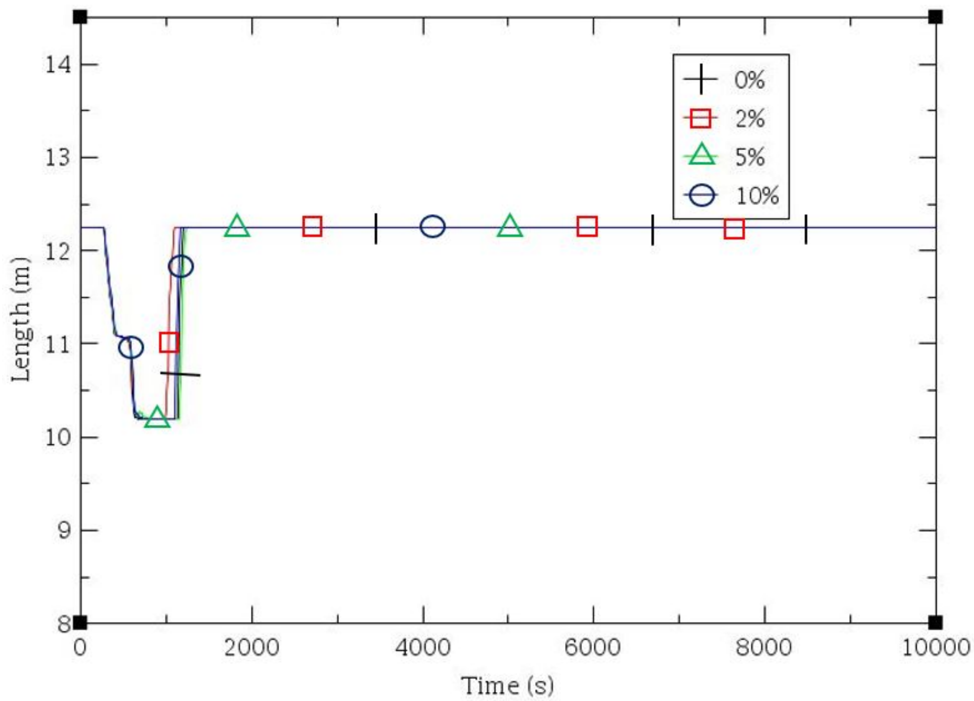


Figure 5-10 Water Level of Primary Side in case 3 with Tube Plugging

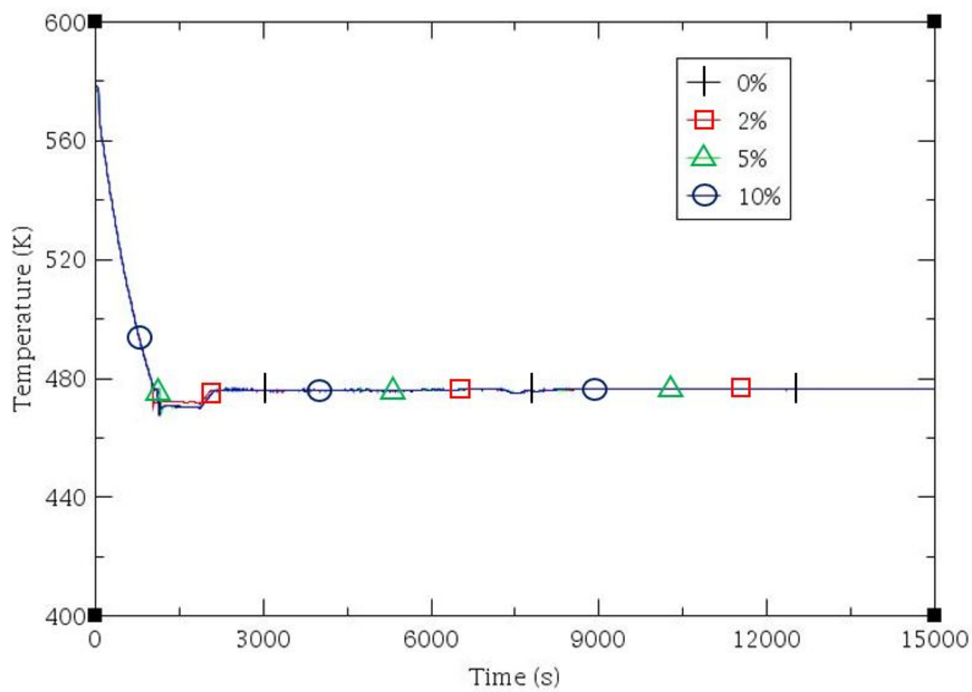


Figure 5-11 Temperature of U Tube in Case 3 with Tube Plugging

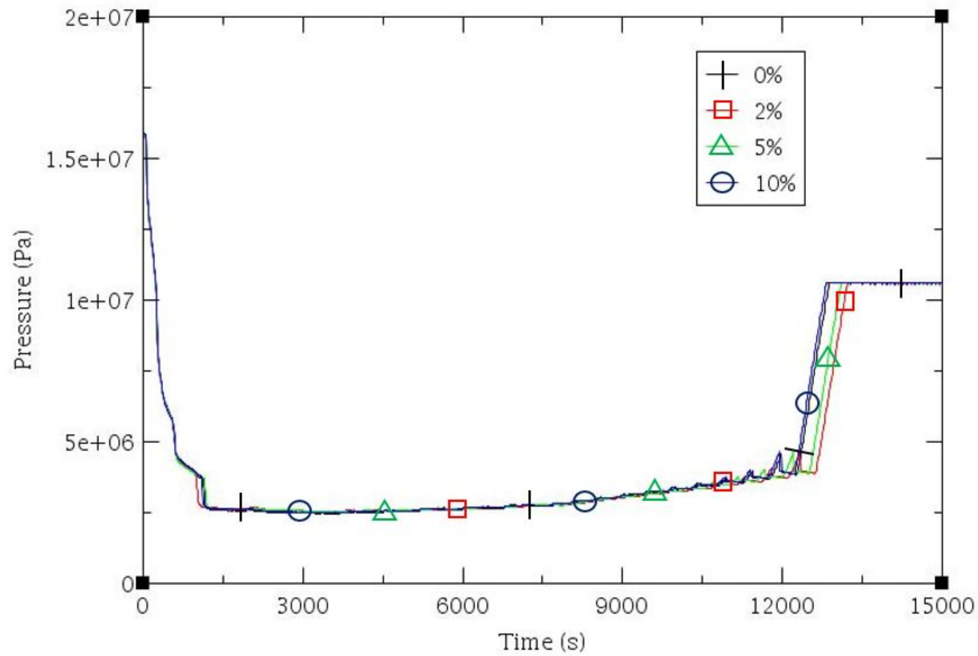


Figure 5-12 Pressure of Primary Side in Case 3 with Tube Plugging

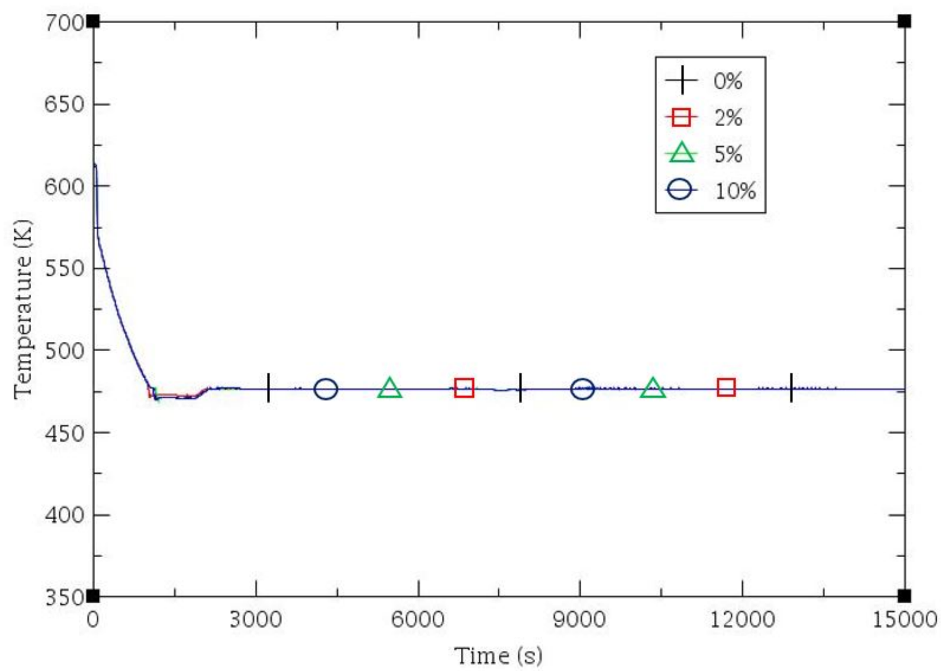


Figure 5-13 Peak Cladding Temperature in Case 3 with Tube Plugging

Table 5-5 Thermal Hydraulic Parameters in Case 3

%	0%	2%	5%	10%
Peak cladding temperature(K)				
Before control depressurization(50 sec)	611.986	612.15	612.408	612.882
After control depressurization(3000 sec)	476.522	476.538	476.556	476.708
After emergency depressurization(7000 sec)	476.742	476.75	476.778	476.791
Pressure of primary side(MPa)				
Before control depressurization(50 sec)	15.727	15.744	15.772	15.83
After control depressurization(3000 sec)	2.492	2.581	2.58	2.407
After emergency depressurization(7000 sec)	2.781	2.695	2.767	2.76
Temperature of U tube(K)				
Before control depressurization(50 sec)	576.64	576.9	577.3	578.023
After control depressurization (3000 sec)	476.028	475.92	475.88	476.01
After emergency depressurization(7000 sec)	476.337	476.28	476.335	476.438
Pressure of SG(MPa)				
Before control depressurization(50 sec)	6.903	6.904	6.905	6.905
After control depressurization(3000 sec)	1.56	1.57	1.564	1.574
After emergency depressurization(7000 sec)	1.571	1.569	1.551	1.565
Water level of SG(m)				
Before control depressurization(50 sec)	12.59	12.59	12.58	12.58
After control depressurization(3000 sec)	7.75	7.75	7.75	7.82
After emergency depressurization(7000 sec)	12.07	12.07	12.11	12.07

6 CONCLUSION

In this research, the best estimate thermal hydraulic program TRACE had been used to analyze when Maanshan nuclear power plant encounter a SBO accident, the effectiveness of URG and FLEX rescue procedure. Furthermore, the analysis of effectiveness of rescue procedure when the plant has plugged tube were also finished. This research had successfully used an existing TRACE model with additional control block and logic for simulation of water injection to simulate the URG and FLEX procedure. While in chapter 5, 2%, 5% and 10% of tube plugging analysis were analyzed by changing the boundary conditions of input model.

During the analysis of URG and FLEX, the rescue procedures are assumed to begin 1 and 4 hours after the accident to compare URG and FLEX cases at the same water injection time point. The results by TRACE show that if rescue an hour after the accident, then the plant remains safe in the whole transient. No exposure of fuel rod and no interrupt of heat transfer between primary and secondary side. If rescue 4 hours after the accident, then a dry out of steam generator will happen and last about 2000 to 3000 seconds. During this dry out period, the pressure of core will rise and water level will decrease because of the loss of heat transfer. However, this short term dry out period is causing nothing to the plant, the following rescue could also bring the plant back to safe. While encounter a complicated Fukushima-like accident, instrument and sensors could be broken during accident, the condition of plant might also not precise enough. If there’s any delay on the rescue procedure, it’s hard to say that nothing would happen during the delay. To ensure the integrity and safety of plant, whenever preparing the URG and FLEX for the plant, one should be as fast as possible to bring the plant back to safe rapidly. In addition, recovering AC power of any other power source should also be doing quickly, so the other following process could be properly carried on.

Furthermore, in the analysis of tube plugging, 2%, 5% and 10% of simulation had been done by changing the flow area of U tube and number of pipe of HTSTR. The results show that no matter the pressure and water level in primary of secondary side, the influence of tube plugging is so tiny and can be neglected. Even in the highest 10% of tube plugging, both URG and FLEX procedure can keep the plant safe. According to the simulation by TRACE, there’s no need to do other treatment about tube plugging for Maanshan nuclear power plant. The present rescue process can properly ensure the safety of plant and prevent any other danger.

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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

This research focused on the analysis of URG (Ultimate Response Guideline) procedure and the FLEX (Flexible and Diverse Coping Strategies) strategy after one and four hours when Maanshan nuclear power plant is under station blackout (SBO) accident by using TRACE code. Then explore the influence on heat transfer between primary side and secondary side when there's plugged tube in the steam generators of power plant. The NEI (Nuclear Energy Institute) had proposed the FLEX strategy and Taiwan Power Company also has the URG procedure, in order to keep the safety of plant from severe disaster. The equipment of plant will become old and deteriorative as time passes by. If there's any problem in tubes of steam generators, operators shall plug the defective tube during outage inspection to prevent them from broken. This research use TRACE to analyze under 2%, 5% and 10% of tube plugging, is it still effective to use the URG procedure and the FLEX strategy rescuing the SBO accident of Maanshan nuclear power plant. The result shows that even under 10% of tube plugging, the URG procedure and the FLEX strategy won't be affected by tube plugging and still can bring the plant back to safety.

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