NUREG/IA-0440

International Agreement Report

The Alternate Mitigation Strategies Study of Chinshan BWR/4 by Using the LOCA and SBO Analysis of TRACE

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Washington, DC 20555-0001

Manuscript Completed: September 2013
Date Published: March 2014

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

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ABSTRACT

Chinshan nuclear power plant is the first NPP in Taiwan which is the BWR/4 plant. This research focuses on the development of the Chinshan NPP TRACE model and the LOCA combined SBO accident analysis. From the accident at the Japanese Fukusima NPP, an extreme event beyond the design basis is realized to be possible. The current mitigation strategies for the emergency core cooling systems (ECCSs) can be easily voided in the event of an extended station blackout (SBO), where all the onsite and offsite electrical power is failed. Although the electrical power of the critical control systems can be recovered by portable electrical generators, the electrical pumps are difficult to recover by any portable device. The only possible driving force of the pumps in SBO is the steam generated by residual heat. The current strategies in an extended SBO are mostly focused on low pressure injection, but the reactor water level will decrease sharply while the reactor pressure is reduced and that results in a higher PCT. In this report, the alternate mitigation strategies adopting the turbine driven pumps, the high pressure injection systems, are analyzed to maintain an “enough” water level before the reactor pressure is reduced. Three break sizes, 100%, 10% and 1%, on the recirculation suction line of Chinshan NPP which is the most serious LOCA in BWR/4 reactor are analyzed with three sensitivity studies: (1) the scram time, (2) the increase of RCIC injection flow rate, and (3) the earlier HPCI injection.

FOREWORD

The US NRC (United States Nuclear Regulatory Commission) is developing an advanced thermal hydraulic code named TRACE for nuclear power plant safety analysis. The development of TRACE is based on TRAC, integrating RELAP5 and other programs. NRC has determined that in the future, TRACE will be the main code used in thermal hydraulic safety analysis, and no further development of other thermal hydraulic codes such as RELAP5 and TRAC will be continued. A graphic user interface program, SNAP (Symbolic Nuclear Analysis Program) which processes inputs and outputs for TRACE is also under development. One of the features of TRACE is its capacity to model the reactor vessel with 3-D geometry. It can support a more accurate and detailed safety analysis of nuclear power plants. TRACE has a greater simulation capability than the other old codes, especially for events like LOCA.

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of TRACE. INER (Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C.) is the organization in Taiwan responsible for the application of TRACE in thermal hydraulic safety analysis, for recording user's experiences of it, and providing suggestions for its development. To meet this responsibility, the TRACE model of Chinshan NPP has been built. In this report, we focus on the LOCA combined SBO accident of TRACE analysis.

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EXECUTIVE SUMMARY

An agreement in 2004 which includes the development and maintenance of TRACE has been signed between Taiwan and USA on CAMP. INER is the organization in Taiwan responsible for applying TRACE to thermal hydraulic safety analysis in order to provide users' experiences and development suggestions. To fulfill this responsibility, the TRACE model of Chinshan NPP is developed by INER.

According to the TRACE user's manual, it is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. Therefore, in the future, NRC has ensured that TRACE will be the main code used in thermal hydraulic safety analysis, without further development of other thermal hydraulic codes such as RELAP5 and TRAC. Besides, the 3-D geometry model of reactor vessel is one of the features of TRACE. It can support a more accurate and detailed safety analysis of NPPs.

In the NPP safety, the safety analysis of the NPP is very important work. Especially in the Fukushima NPP event occurred, the importance of NPP safety analysis has been raised and there is more concern for the safety of the NPPs in the world. Chinshan NPP was building in 1970. It is the first NPP in Taiwan which is the BWR/4 plant and the original rated power for each unit is 1775 MWt. After the project of MUR (Measurement Uncertainty Recovery) for Chinshan NPP, Unit 2 started MURPU (Measurement Uncertainty Recovery Power Uprate) from April 6, 2008 for Cycle 23 and Unit 1 started MURPU from November 8, 2008 for Cycle 24. The thermal power of Chinshan NPP is 1828MWt now.

This research focuses on the development of the Chinshan NPP TRACE model and the LOCA combined SBO accident analysis. From the accident at the Japanese Fukusima NPP, an extreme event beyond the design basis is realized to be possible. The current mitigation strategies for the emergency core cooling systems (ECCSs) can be easily voided in the event of an extended station blackout (SBO), where all the onsite and offsite electrical power is failed. Although the electrical power of the critical control systems can be recovered by portable electrical generators, the electrical pumps are difficult to recover by any portable device. The only possible driving force of the pumps in SBO is the steam generated by residual heat. The current strategies in an extended SBO are mostly focused on low pressure injection, but the reactor water level will decrease sharply while the reactor pressure is reduced and that results in a higher PCT. In this report, the alternate mitigation strategies adopting the turbine driven pumps, the high pressure injection systems, are analyzed to maintain an"enough"water level before the reactor pressure is reduced. Three break sizes, 100%, 10% and 1%, on the recirculation suction line of Chinshan NPP which is the most serious LOCA in BWR/4 reactor are analyzed with three sensitivity studies: (1) the scram time, (2) the increase of RCIC injection flow rate, and (3) the earlier HPCI injection. Through this report, the alternate mitigation strategy using the turbine driven pumps and residual steam is evaluated for the emergency operational procedures (EOPs) and the severe accident mitigation guidelines (SAMGs).

ABBREVIATIONS

ADS	Automatic Depressurize System
CAMP	Code Applications and Maintenance Program
CHAN	A Component of TRACE to Simulate Fuel Bundles
CS	Containment Spraying
ECCS	Emergency Core Cooling System
EOPs	Emergency Operational Procedures
FSAR	Final Safety Analysis Report
HPCI	High Pressure Core Injecting
HTC	Heat-Transfer Coefficient
JETPUMP	A Component of TRACE to Simulate Jet Pumps
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss-Of-Coolant Accident
LPCI	Low Pressure Core Injecting
LPCS	Low Pressure Core Spraying
MSIV	Main Steam Isolation Valve
NPP	Nuclear Power Plant
PCT	Peak Cladding Temperature
SAMGs	Severe Accident Mitigation Guidelines
SBO	Station Blackout
SEPD	A Component of TRACE to Simulate Separators and Dryers
SRV	Safety Relieve Valve
TAF	Top of Active Fuels
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
USNRC	U.S. Nuclear Regulatory Commission
VESSEL	A Component of TRACE to Simulate the Reactor Vessel

1. INTRODUCTION

In the NPP safety, the safety analysis of the NPP is very important work. Especially in the Fukushima NPP event occurred, the importance of NPP safety analysis has been raised and there is more concern for the safety of the NPPs in the world. The TRACE code, the TRAC/ RELAP Advanced Computational Engine, is the latest component-based and best-estimate reactor system code being developed by the U.S. nuclear regulatory commission (USNRC) for analyzing the neutronic and thermal-hydraulic behaviors, operational transients and other accident scenarios in light water reactors. Through the international cooperative program, the Code Applications and Maintenance Program (CAMP), many organizations participate and adopt the TRACE code for various applications.

From the accident at the Japanese Fukusima NPP, an extreme event beyond the design basis is realized to be possible. The current mitigation strategies for the emergency core cooling systems (ECCS) can be easily voided in the event of an extended station blackout (SBO) because the low pressure injecting system is out of service without AC power and causes a serious low water level. In the Fukusima accident, following the current EOPs and SAMGs, the reactor pressure was released for the low pressure injecting system. In fact, the reactor pressure was still too high for the low pressure injecting system and caused the fuel bundles to become uncovered and damaged. It is necessary to analyze more different cases in the extreme accident and provide more safety precautions and more operation strategies in an accident.

The most extreme event is the LOCA combined with the station blackout (SBO) event, where all the onsite and offsite AC electric power is failed except the DC battery power for the control system. Since the low pressure ECCS system relies on electrical pumps, it will fail in SBO and the automatic depressurize system (ADS) would not be activated because of an interlock with the low pressure core injecting system (LPCI). In an SBO event, the electrical power of the critical control systems can be recovered by portable electrical generators, but the electrical pumps are difficult to recover by any portable device. The only possible driving force for the pumps in SBO is the steam that is generated by residual heat. If the reactor pressure is depressurized, the turbine pumps of the RCIC and the HPCF will fail. Making the best use of residual steam as a driving force for the high pressure core injecting systems is a possible mitigation strategy in the SBO. The two most important targets after the reactor scram are to maintain the reactor coolant above the TAF and to remove the residual heat. The turbine driven pump can fulfill these two requirements. The current strategies in the extended SBO are mostly focused on low pressure injection, but the reactor water level will decrease sharply while the reactor pressure is released and that results in a higher PCT. In this report, the alternate mitigation strategies using the turbine driven pumps and the high pressure injection system are analyzed to maintain an “enough” water level before the reactor pressure is released. With sufficient coolant, the fuel temperature will be maintained to prevent fuel bundles being damaged.

Chinshan nuclear power plant with the same BWR/4 reactor as the Fukusima NPP is the first nuclear power plant in Taiwan. After the MUR (Measurement Uncertainty Recovery) project in 2008, the rated thermal power is 1828MWt with the rated steam flow at 3.5 Mkg/h, the core flow at 24 Mkg/h and the reactor pressure at 6.98 MPa. The power and the core flow analyzed are 1864 MWt and 18 Mkg/h, respectively (102% power and 75% core flow). There are 408 fuel bundles, two recirculation loops, 20 Jetpumps and 130 separators. In Chinshan NPP, the RCIC system and the HPCI system are equipped with turbine driven pumps that provide the injecting mass flow at 25.2 liters/sec and 267.8 liters/sec, respectively.

The TRACE model of Chinshan NPP, consisting of the specific components like 3D VESSEL, JETPUMP, SPED and CHAN, was developed using the plant design data and benchmarked with the steady state of FSAR, the start-up data and the transient results of the RETRAN data. In this research, a more serious case than the Fukusima accident, the LOCA combined SBO, is analyzed with the two turbine driven pumps, the RCIC pump and the HPCI pump. Three break sizes, 100%, 10% and 1%, on the recirculation suction line which is the most serious LOCA in BWR/4 reactor are analyzed with three sensitivity studies: (1) the scram time, (2) the increase of RCIC injection flow rate, and (3) the earlier HPCI injection. Different break sizes will result in different discharging flows, a different declining curve of the reactor pressure, a different scram time and a different recovery time of the reactor water level. The reactor scram can be triggered by the three signals, the drywell high pressure, the reactor low water level and the high reactor pressure. The first two scram signals are analyzed with the three different break areas. The scram signal of the drywell high pressure is found to be activated earlier than the other two scram signals and the impact is clearer in a small break LOCA. If the drywell high pressure signal fails, the reactor will be scrammed by the reactor low water level signal (Level-3) which is delayed by several seconds. The SRV is activated to maintain the reactor pressure in the 1% break area.

Although the discharging flow of the 1% break area is less than the other two, the time below the TAF level is much longer because the reactor pressure remains high and inhibits the HPCI flow injecting. The reactor water level will found not recover until the success injection of the HPCI flow with the original RCIC flow. The HPCI flow will not be injected into the reactor until the HPCI injecting valve opens at the reactor pressure of 3.44 MPa. The impact on the earlier HPCI injection by increasing the setpoint from 3.44 MPa to 6.21 MPa is found to be significant for the recovery of the reactor water level. The current RCIC flow is not sufficient even in a 1% break area. In this paper, to increase the RCIC flow is also tested with double and triple RCIC flow in a 1% break area. It is found that the reactor water level could recover earlier with the RCIC flow increasing and is more significant in a small break LOCA.

With the sensitivity studies on the three break sizes, the scram signals, the opening setpoint of HPCI injecting valve and the mass flow of the RCIC system in the paper, a different strategy in the extreme accident with SBO is developed. To make the best use of the residual steam as a driving force with the high pressure core injecting systems is a possible mitigation strategy in the SBO. Through the tests, the emergency operating procedures (EOPs) and severe accident mitigation guidelines (SAMGs) can be evaluated again.

2. METHODOLOGY AND MODELING

2.1 The TRACE Code

TRACE, the TRAC/RELAP Advanced Computational Engine developed by USNRC, has been designed to perform best-estimate analyses of loss-of-coolant accidents (LOCAs), operational transients, and other accident scenarios in reactor systems. Through the International Cooperative Program, Code Applications and Maintenance Program (CAMP), many organizations participate and adopt TRACE for various analysis applications. TRACE is a modernized code with the capability to simulate the reactor system and model the thermal-hydraulic phenomena in three-dimensional space. TRACE is a component-based code for fast and integrated inputs of reactor systems. The reactor vessel, fuel bundles, separators and dryers and jetpumps are modeled by the specific components, VESSEL, CHAN, SEPD and JETPUMP. In addition, TRACE is integrated into SNAP (Symbolic Nuclear Analysis Package), a graphic user interface, to assist users developing TRACE input decks. Instead of those out-of-date codes like TRAC and RELAP, the TRACE code provides a new man-machine interface and will become the NRC's flagship thermal-hydraulic analysis tool. The SNAP v2.0.4 and TRACE v5.0p2 were employed in this research.

2.2 Chinshan TRACE Model

The TRACE model of Chinshan Nuclear Power Plant (1) is developed and based on the plant design data; (2) consists of different modules to simulate the reactor systems; and (3) analyzes the 3D thermal-hydraulic phenomena through the 3D VESSEL component (shown in Figure 1). The reactor vessel is divided into 88 cells with eleven axial elevations, four radial rings and two azimuthal sectors. The volume and height of each cell is determined by the plant design data. The form loss, the hydraulic diameter and the flow area can be set individually on each of the six interfaces of a cell to specify fluid-dynamics and heat transfer calculations.
The loss coefficient K for an abrupt expansion is calculated as

$$\mathrm{K}_{\mathrm{expansion}} = \left(1 - \frac{A_j}{A_{j+1}}\right)^2 \tag{1}$$

The loss coefficient K for an abrupt contraction is interpreted as below

Aj+1/Aj	0.0	0.04	0.16	0.36	0.64	1.0
Kcontraction	0.5	045	0.38	0.28	0.14	0.0

Once the loss coefficient K has been determined using the above equation and data, the pressure drop across the abrupt expansion or abrupt contraction may be calculated as follows. The flow at section j has a velocity V_j, while the flow upon reaching section $j+1$ has a velocity V_{j+1} that is lower or higher than V_j because of the abrupt cross-section change. The change in pressure from points j to $j+1$ which is caused by the abrupt area change yields

$$\Delta \mathrm{P}_{\mathrm{j} \to \mathrm{j+1}} = -\left(P_{j+1} - P_j\right) = \rho\left(\frac{V_{j+1}^2}{2} - \frac{V_j^2}{2}\right) + \rho K \frac{V_{j+1}^2}{2} \tag{2}$$

The most external ring is the downcomer while the inner three rings are the core zone. There are 408 fuel bundles and 130 separators. The 408 fuel bundles are modeled by six CHANs, which are located in the axial level 4 and 5. The separators and dryers are simulated by the six SEPDs located from the axial level 7 to 10. The pressure distributions in the reactor vessel, including the core area, separators, dryers and the upper plenum, are also validated. The Chinshan TRACE model includes two recirculation loops with one recirculation pump and ten jet pumps in each, together with four steam lines - each with one safety relieve valve (SRV), one main steam isolation valve (MSIV), two turbine control valves (TCVs), and one turbine bypass valve (TBV). The SRVs and the TBVs function as reactor pressure control systems to prevent the reactor from being over-pressured. The pressure drops along main steam lines and the recirculation loops are calculated per the formulas above and validated through the plant data and the FSAR of Chinshan NPP [1].

Fuel bundles are modeled by six CHANs with point kinetics feedbacks of the delay neutron fraction, Doppler reactivity coefficient and void reactivity coefficient [2]. The water rods, partial length rods, full length rods, tie rods and the leakage paths between the rods are modeled by the CHAN component. The CHAN is divided into twelve axial sections with the last section as the fuel handling. The power shape in axial can be set in the POWER component. The core bypass flow between the fuel bundles is simulated by the VESSEL component. Level 3 is the fuel support that the bypass flow between the fuel bundles can be simulated and divided into six sections. The ratio of the bypass flow to the core flow is validated. The CHAN is divided into twelve axial sections with the last section as the fuel handling. The power shape in axial can be set in the POWER component.

The specific TRACE components, like the JETPUMP, SPEDs and CHANs, are used in the TRACE model of Chinshan NPP. The SEPD component is used to simulate its GE mechanical separators and dryers. The 130 separators are modeled by six SPEDs and located in the level 7 and 8. The SEPD is divided into four cells with three cells to be the main tube and one cell to be the side tube which discharges the separated liquid. The side tube simulates the discharging flow of the separated liquid that the radial mass flow from outsides of separators to the downcomer can be analyzed. The capability to simulate the radial mass flow is important. With the SEPD components used, more thermal-hydraulic behaviors in separators and dryers can be analyzed, including the void fraction, pressure propagation, fluid velocity in separators, and the carryover and carryunder quality. Since the arrangement can be modeled more practically in the stand pipe area, the separators area, and the intermediate area, the radial mass flow of the discharge liquid to downcomer can be simulated. The SEPD component is validated with steady-state baselines and the startup tests, 83% power 75% flow Turbine Trip test and 100% power 100% flow Load Rejection test [3].

The feedwater control system is a three-element control model by the three parameters of the reactor water level, the steam flow rate, and the feedwater flow rate. The normal operating water level is at level 7. For a more practical instrument measurement of the reactor water level, some specific functions are used for the narrow range water level. The RCIC and the HPCI with turbine driven pumps are built with their injection paths via the feedwater lines. The HPCI pump provides injection flow at 267.8 kg/sec when the reactor pressure is between 1.03 MPa and 7.74 MPa. The RCIC pump provides injection flow at 25.2 kg/sec. The driving steam for the RCIC and HPCI pump is modeled to simulate the governor valve of a turbine pump. The Chinshan TRACE model has been benchmarked through several transient cases [2]-[4] with the Chinshan FSAR, the start-up data [5] and the transient results of the RETRAN data [6]-[8]. Besides, the initial condition and animation model of Chinshan NPP is shown in Figure 2.

2.3 Test Cases and Sensitivity Studies

The most limiting LOCA of a BWR/4 reactor is a break on the recirculation suction line. The combined accident of the LOCA and extended SBO will result in an extreme event beyond the DBAs. In this paper, the extreme event of LOCA combined SBO is analyzed with different sizes of break, the 100%, 10% and 1% of the recirculation suction line. The power and the core flow analyzed are 1864 MWt and 18 Mkg/h, respectively (102% power and 75% core flow). The sensitivity studies on the first two scram signals are analyzed with the three different break areas. In addition, sensitivity studies on the two important parameters, the RCIC flow and the early opening of the HPCI injecting valve, are analyzed with three break sizes, 100%, 10% and 1%, on the recirculation suction line.

In this report, the alternate mitigation strategies adopting the turbine driven pumps, the high pressure injection systems, are analyzed to maintain an “enough” water level before the reactor pressure is released. An evaluation from the plant wide to the system level is given. The ADS and LPCI will be out of service as the high pressure injecting systems, the RCIC and the HPCI systems, with turbine driven pumps of are the only available systems. Because the RCIC flow is found not sufficient even in a small break LOCA, the two mitigation strategies, to increase the RCIC flow and to inject the HPCI flow earlier, are analyzed for a possible resolution in the extreme event of the LOCA and the extended SBO.

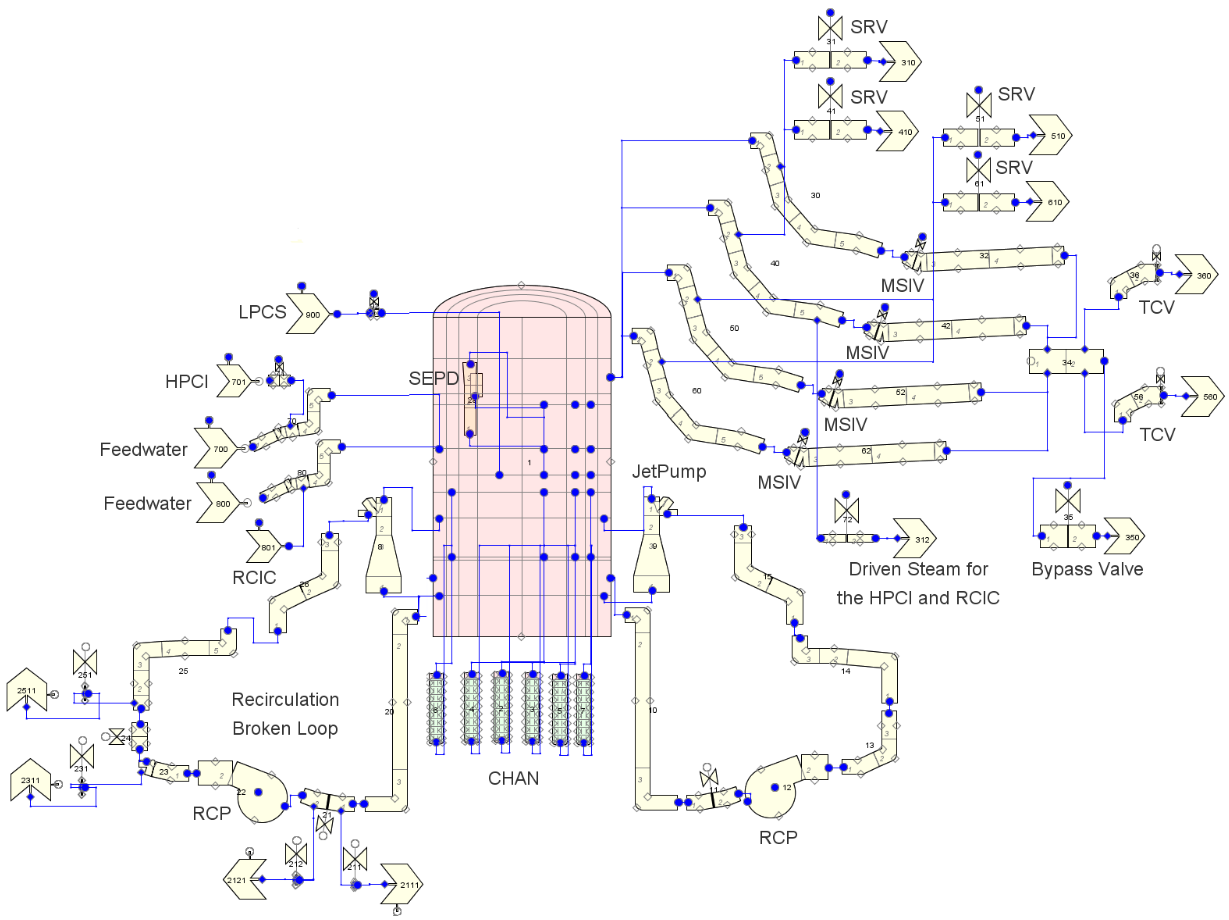

Figure 1 The TRACE model of Chinshan nuclear power plant

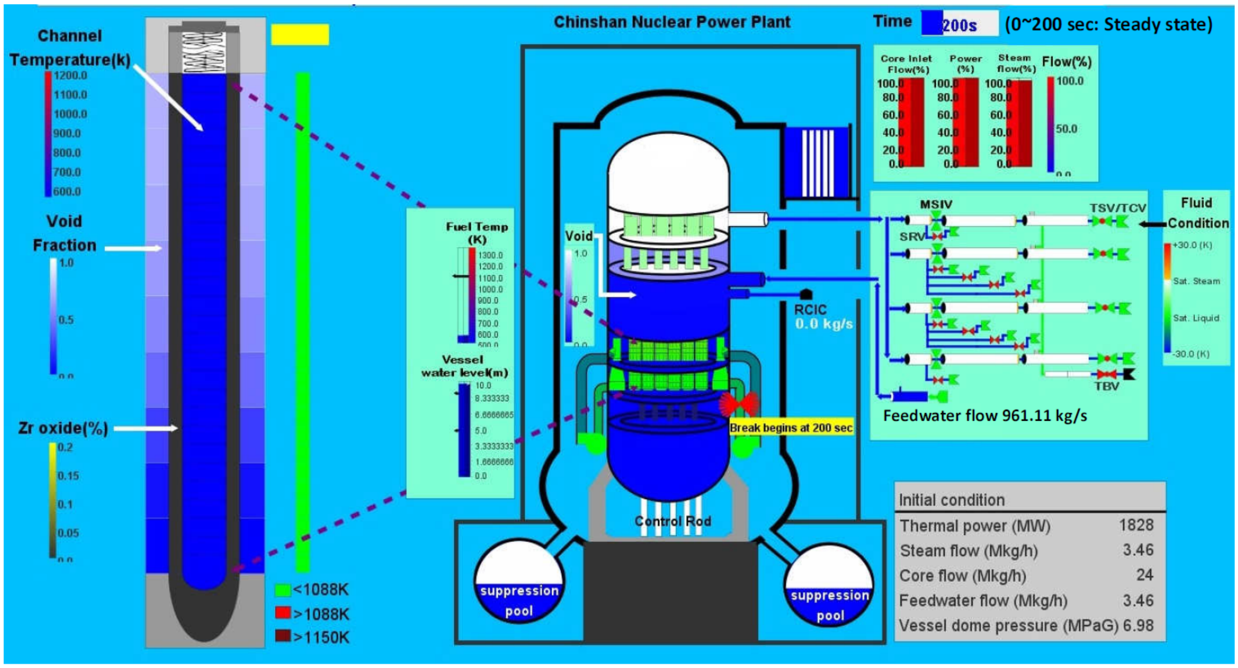

Figure 2　The initial condition and animation model of Chinshan nuclear power plant

3. RESULTS AND DISCUSSION

3.1 The Mitigation Strategies of ECCS System

The ECCS provides a variety of systems for the mitigation strategies of defense in depth which is divided into two segments (see Figure 3). The high pressure system includes the HPCI system. Although the RCIC system is not a typical ECCS, the turbine driven RCIC pump is evaluated as a mitigation injection coolant in this paper. When LOCA, the RCIC system and the HPCI system are activated by the reactor low water level (Level 2) and provide high pressure injecting mass flow rates at 25.2 kg/sec and 267.8 kg/sec, respectively. In a small break, the RCIC system can be started and make up the break flow immediately; in a medium break, the HPCI system provides more injecting coolant into the vessel. If the two high pressure injecting systems still fail to make up the coolant and the reactor water keeps reducing to reactor water level one, the automatic depressurized system (ADS) will be activated to release the reactor pressure for the successful injection of the low pressure systems, the LPCI and the containment spraying (CS), which require AC power for their electrical pumps. The containment spraying (CS), including drywell spraying and wetwell spraying, provides an important pressure reduction method to keep the integrity of the containment.

The RCIC pump, a turbine driven pump, reaches the rated flow in ten seconds and injects coolant directly into the reactor. The HPCI system includes the injecting valve which interlocks with the reactor pressure. If the reactor pressure is more than 3.44 MPa, the HPCI injecting valve will not be opened and the HPCI flow will not inject into the vessel although the HPCI pump has been operating. In the current strategy, the ADS system will be activated to release the reactor pressure for the injection of the LPCI system which utilizes electrical driven pumps. The LPCI system provides a much larger mass flow to make up the reactor coolant in a large break LOCA.

But, in the SBO accident, the mitigation strategy will be voided with the failure of the LPCI system and the CS system. When the reactor pressure is released, the RCIC system and the HPCI system will lose their driving force and at the same time the reactor water level reduces sharply. Without AC power, the low pressure injecting system will be out of service and the ECCS systems, including the high pressure and the low pressure injecting systems, will fail to makeup reactor coolant after the reactor pressure has been released. Although trying to utilize the external injecting system, the reactor pressure is still too high to inject coolant and results in the fuel being uncovered and damaged. Thus, adopting an alternate mitigation strategy to utilize the best of the residual steam through the RCIC system and the HPCI system instead of releasing reactor pressure is necessary in the extreme event of the LOCA and extended SBO.

3.2 The Extreme Event of LOCA Combined SBO

The most limiting LOCA of a BWR/4 reactor is a break on the recirculation suction line. The combined accident of a LOCA and an extended SBO will result in an extreme event beyond the DBAs. All AC power is assumed to be failed, resulting in the out of service of the low pressure injecting system and the automatic depressurized system. The current mitigation strategies of the ECCS as shown in Figure 3 can be easily voided in an extended station blackout (SBO) event, where all the onsite and offsite AC electrical power is failed and results in an LPCI failure. The electrical power of the critical control systems can be recovered by portable electrical generators, but the electrical pumps are hardly recovered by any portable device. The only available driving force for the pumps in SBO is the steam generated by the residual heat. Only the turbine driven pumps of the RCIC and the HPCI are available. If the reactor pressure is depressurized, the high pressure injecting systems with the turbine pumps will fail and the low pressure injecting systems with the electrical pumps also fail. None of injection flow is available. In addition, the reactor pressure is still too high for the external injecting system. Without sufficient coolant, the fuel will be uncovered and the temperature will peak high and cause serious fuel damage. A different strategy in the extreme accident with SBO is necessary. To make the best use of the residual steam as a driving force with the high pressure core injecting systems is a possible mitigation strategy in the SBO.

At the beginning of LOCA, the reactor water level reduces quickly because of the break flow; the reactor is scrammed by the signal of the drywell high pressure which is triggered earlier than the reactor low water level and the reactor high pressure. The important events and setpoints of the Chinshan NPP are listed in the Table 1. The reactor scram can be initialed by the three signals, the drywell pressure high, the reactor low water level and the high reactor pressure. The drywell pressure increases and the reactor water level reduces because of the break discharge; the reactor pressure peaks up immediately after the closure of the MSIV valves. The closure of the MSIV valves is activated by the signal of reactor water level 2. The peaking up of the reactor pressure is more significant in a small break LOCA and the SRV is activated to maintain the pressure in the 1% break area. The SRVs will be activated several times to prevent the reactor pressure from exceeding the limit. The MSIVs are closed by the reactor low water level (Level 2) and the steam flow and the reactor water level reduce very quickly. In the main steam lines, only the SRVs flow and the driving steam flow of the RCIC pump and the HPCI pump remains (shown in Figure 6).

The scram signal of the reactor high reactor pressure will not be triggered in a large break LOCA because the discharging flow is massive that the reactor pressure will not peak up when the MSIV closes. The RCIC system and the HPCI system are initiated at the signal of the reactor low water level (Level 2) and inject coolant into the vessel via the feedwater lines. But the HPCI flow will not be injected into the reactor because the HPCI injecting valve will not be opened unless the reactor pressure lowers than 3.44 MPa. Thus, the reactor pressure is found to be a dominant factor for the HPCI injection in this paper. In a small break, take 1% area break for example, the HPCI flow will not be injected into the vessel although the HPCI pump has started at the reactor water level two. The delay will cause the reactor water lower below the TAF. In any break area, the reactor water level is not recovered unless the HPCI flow injects into the vessel.

3.3 <u>The Various Break Area on Recirculation Suction Line</u>

In this report, the power and the core flow analyzed are 1864 MWt and 18 Mkg/h, respectively (102% power and 75% core flow). The extreme event of the LOCA and the SBO is analyzed with different sizes of break, the 100%, 10% and 1% area of the recirculation suction line. Most of the break mass flow discharges in the beginning and reduces sharply after the reactor water level goes down (shown in Figure 4 and 5). Because the onsite and offsite AC power is failed, the low pressure ECCS systems and the automatic depressurize system (ADS) are failed. The high pressure injecting systems, RCIC and HPCI, with the turbine pumps are the only available systems. The peak cladding temperature increases at the beginning of LOCA and decreases slightly after the reactor scram. Since the reactor water level keeps decreasing, the fuel cladding temperature will keep increasing until the recovery of the reactor water level.

The three different break sizes on the recirculation line will result in different discharging flows, the declining curves of the reactor pressure, the SCRAM times and the reactor water levels. In the Figure 4, the discharging flows of the three are massive in the first fifty seconds then reduce after much of the coolant has been discharged and the water level is below the TAF. Take the time sequences in Table 2 for example; the reactor water levels reduce very fast below the TAF level in the 100% and 10% break area while the 1% break area keeps above the TAF level in the first 65 seconds.

The reactor water level is found not recovered until the HPCI injection. The HPCI injecting valve does not open unless the reactor pressure to be lower than 3.44 MPa. In Figure 6, the reactor pressure of the 1% break area keeps high and the HPCI injecting is inhibited longer than the other two cases although the break discharging is less. The duration below the TAF level of the 1% break area is much longer than the other two break areas because of the HPCI flow being inhibited. It results in a higher peak cladding temperature. The peak cladding temperature keeps increasing when the reactor water level coasts down and reaches the highest temperature before the recovery of the reactor water level (shown in Figure 7 and 8).

The peak cladding temperature of the 1% break area is as high as the PCT of the 100% break area because the reactor pressure of the 1% break area remains high and inhibits the HPCI injection. Thus, a higher setpoint of the HPCI injecting valve opening could be an effective mitigation strategy to maintain the temperature and will be discussed later (see Figure 5 and 12). To maintain the reactor water level above the TAF is more important in the beginning. If the reactor water level reduces below the TAF too early, it will result in a higher peak cladding temperature and a lower reactor water level. Although the reactor water level recovers about at the same time in the 100% and 10% break area, a short recovery of the reactor water level in the beginning of the 10% break area could result in a lower peak cladding temperature (shown in Figure 5 and 12) The peak cladding temperature of the 100% break area is almost double than the 10% break area.

3.4 The Sensitivity of Scram Signals

A reactor scram can be initialed by the three signals, the drywell high pressure, the reactor low water level and the high reactor pressure. The drywell pressure increases and the reactor water level decreases because of the discharging flow, and the reactor pressure peaks up immediately after the MSIV valves close which is activated by the signal of reactor water level 2. The peaking up of the reactor pressure is more significant in the 1% break area and the SRV valve has to be activated for the pressure to be maintained.

In Figure 9, the high drywell pressure is found earlier than the other two signals, the low reactor water level and the reactor high pressure. It is more significant in a small break LOCA. If the high drywell pressure signal fails, the reactor will be scrammed by the low reactor water level (Level 3), which is several seconds after the high drywell pressure signal. The scram signal of the reactor high pressure is the last one and will not be triggered in a large break LOCA.

From the 100% and 10% break areas in Figure 10, the deviation of the reactor water level is not much different between the two scram signals, the high drywell pressure and the low reactor water level. In the 1% break area, the reactor water level with the scram signal of the high drywell pressure drops slower in the first 200 seconds and recovers earlier (about 49 seconds). The trends of the reactor water level of the three break areas can be used to explain the trends of the PCT. Take the 1% break area for example. The peak cladding temperature with the low reactor water level is lower than the one with the high drywell pressure because of an earlier water recovery (see Figure 10 and 11).

The setpoint of the high drywell pressure is 13.8 kPa higher than the normal drywell pressure. The drywell pressure is found to be peaking up quickly even in a small break LOCA. The time sequences of the six cases, the three different break areas with the scram signals of the high drywell pressure and the low reactor water level, are listed in the Table 2 and Table 3.

3.5 The Sensitivity to Increase RCIC Flow

In Figure 5, the designed RCIC mass flow, 25.2 kg/sec, is found to be insufficient for the discharging flow even in the 1% break area and will result in a lower reactor water level although its break flow is less. From Tables 2 and 3, the reactor water levels of the 1% break area are found to be below the TAF longer than the other two break areas.

In this paper, the different RCIC mass flows are analyzed with the double and the triple. To increase the RCIC mass flow is found to be effective for the reactor water recovery and the reactor pressure control, especially in a small break LOCA. Take the 1% break area in the Figure 14 for example, if the original injecting pressure of the HPCI flow is considered, increasing the RCIC flow provides an earlier recovery of the reactor water level. The recovery time of the reactor water level with the double RCIC flow is about 140 seconds earlier and about 540 seconds with the triple RCIC flow. The decline of the reactor water level is slower with the increasing of the RCIC flow. With the triple RICI flow, the time below the TAF is delayed to 300 seconds while the cases with the other two RCIC flows are less than 100 seconds. To increase the RCIC flow will reduce the duration time below the TAF level and is effective to decrease the peak cladding temperature (see Figure 15 and 17).

The reactor pressure declines faster with the increased RCIC flow because the more injecting flow the better cooling effect (shown in Figure 16). The faster pressure decline is good for an earlier HPCI flow injection and in an earlier reactor water recovery (shown in Figure 14). An earlier HPCI injecting flow is found more effective for the reactor water level recovery and the lower peak cladding temperature. In addition, the greater reactor pressure decline will also reduce the discharging flow (shown in Figure 13). To increase the RCIC mass flow is one of the effective mitigation methods in the extreme event of the LOCA and the extended SBO.

3.6 The Sensitivity on Earlier Injecting of HPCI Flow

The HPCI flow provides a larger injecting flow at 267.8 kg/sec whick is ten times that of the RCIC flow. The reactor water level is found to not recover until the successful injection of the HPCI flow in the three different break areas. Although the HPCI pump has started at the same time as the RCIC pump, the HPCI flow will not be injected into the reactor vessel unless the HPCI injecting valve is opened. Since the opening of the HPCI injecting valve is closely related to the reactor pressure, the HPCI injecting flow will be inhibited in a reactor pressure above 3.44 MPa. That is the reason why the reactor water level of 1% break area suffers a longer duration than those of 10% and 100% break areas. If the setpoint of the HPCI injecting valve is increased from 3.44 MPa to 6.21 MPa, the HPCI flow will inject earlier and the impact is more significant than the increasing RCIC flow. The reactor water level is recovered earlier and the peak cladding temperature is reduced more.

In the Figure 14, comparing the two setpoints, 3.44 MPa and 6.21 MPa, it is found that the earlier HPCI injection is more effective for the reactor water level recovery. The recovery time of the reactor water level is 650 seconds earlier while the double RCIC flow is only 150 seconds earlier.

In the Figure 17, the peak cladding temperature reduces more with the earlier HPCI injection when compared with the double and triple RCIC flow. The combined effects of the increasing RCIC flow and the earlier HPCI injection are more significant. With the triple RCIC flow and the earlier HPCI injection in the 1% break area, the reactor water level could be maintained above the TAF level and the peak cladding temperature will not increase.

From the sensitivity studies on the above proposed modifications, it is found that the impact is more significant with an earlier HPCI injection than with the increasing RCIC flow. The earlier injection of the HPCI flow is more effective to maintain the reactor water level and reduce the peak cladding temperature. From the plant owner’s point of view, to change the opening setpoint of the HPCI injecting valve for an earlier HPCI injection is easier and will cost less than to increase the capacity of the RCIC pump.

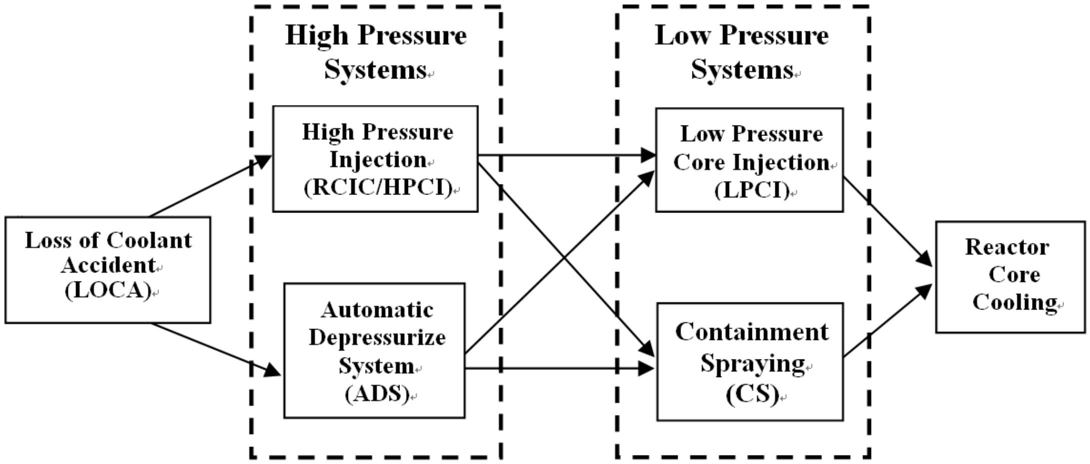

Figure 3 The ECCS system mitigation strategy of Chinshan NPP [1]

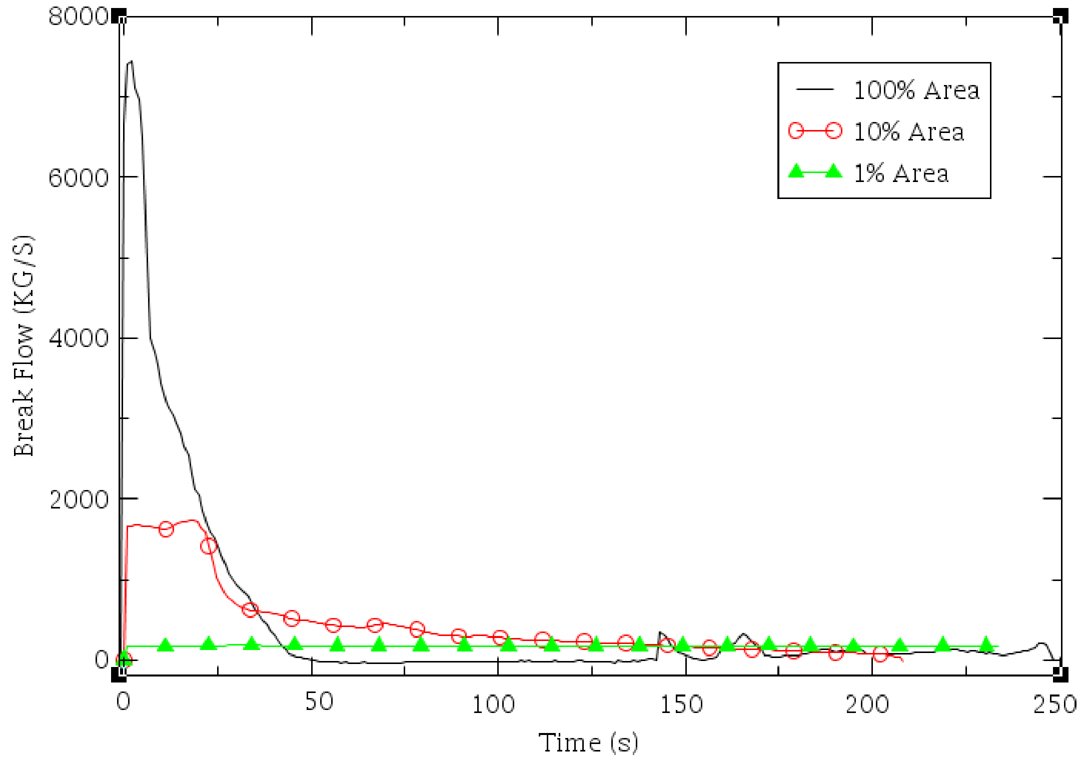

Figure 4 The break flow rate with various break area

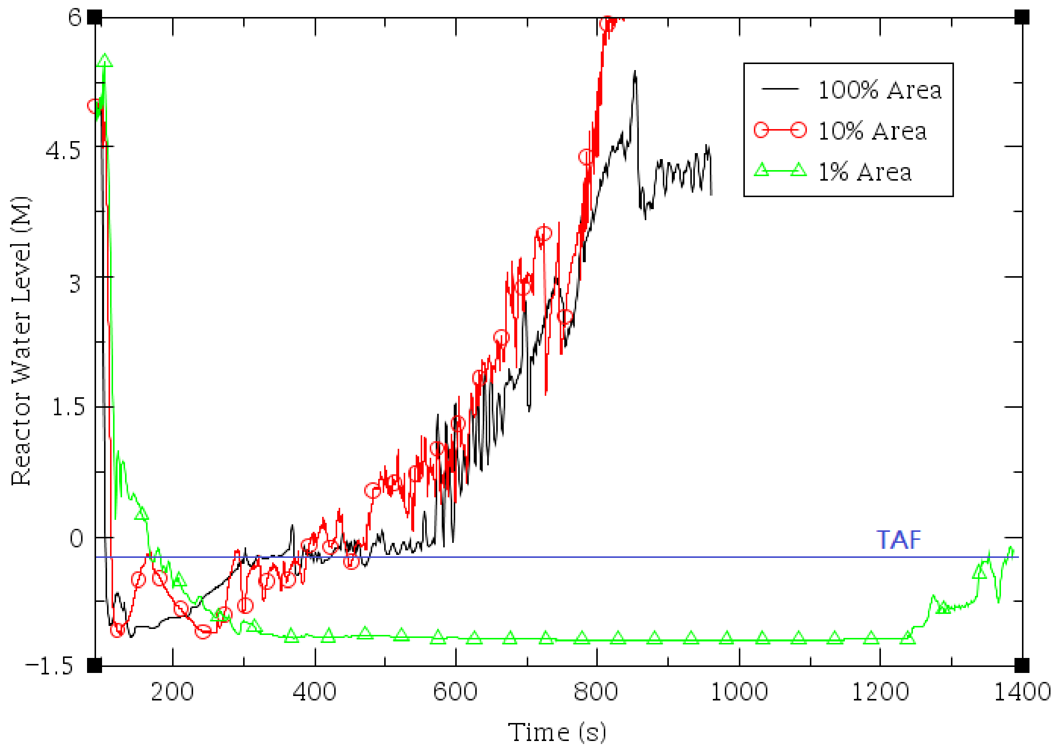

Figure 5 The reactor water level with various break area

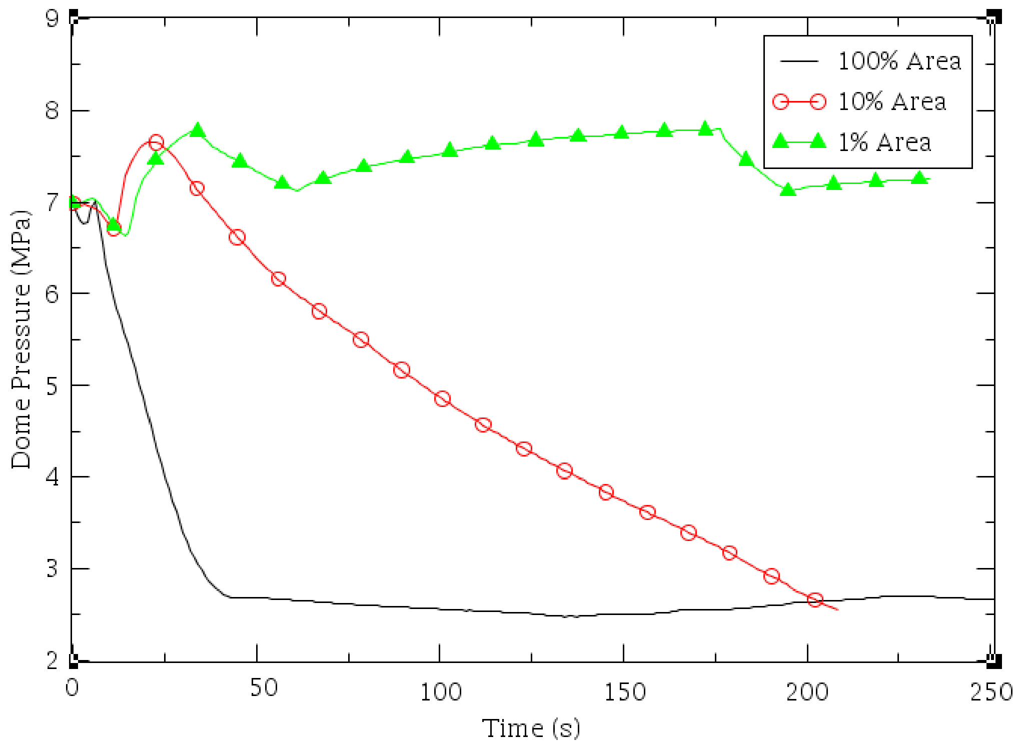

Figure 6 The dome pressure with various break area

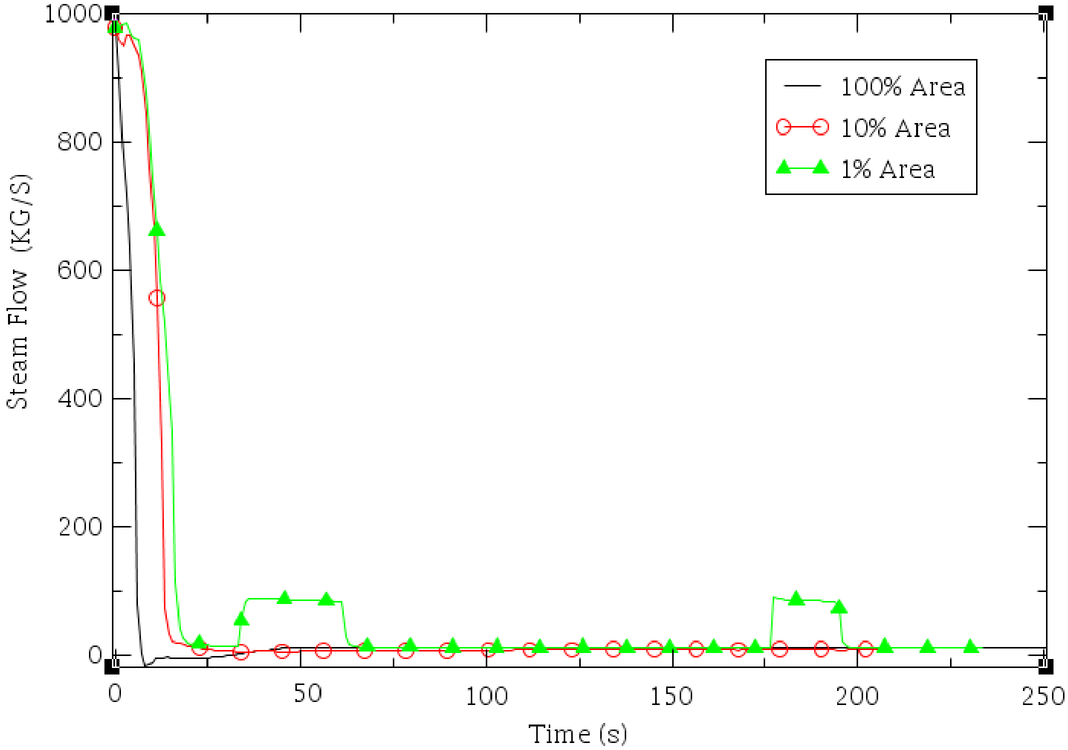

Figure 7 The steam flow rate with various break area

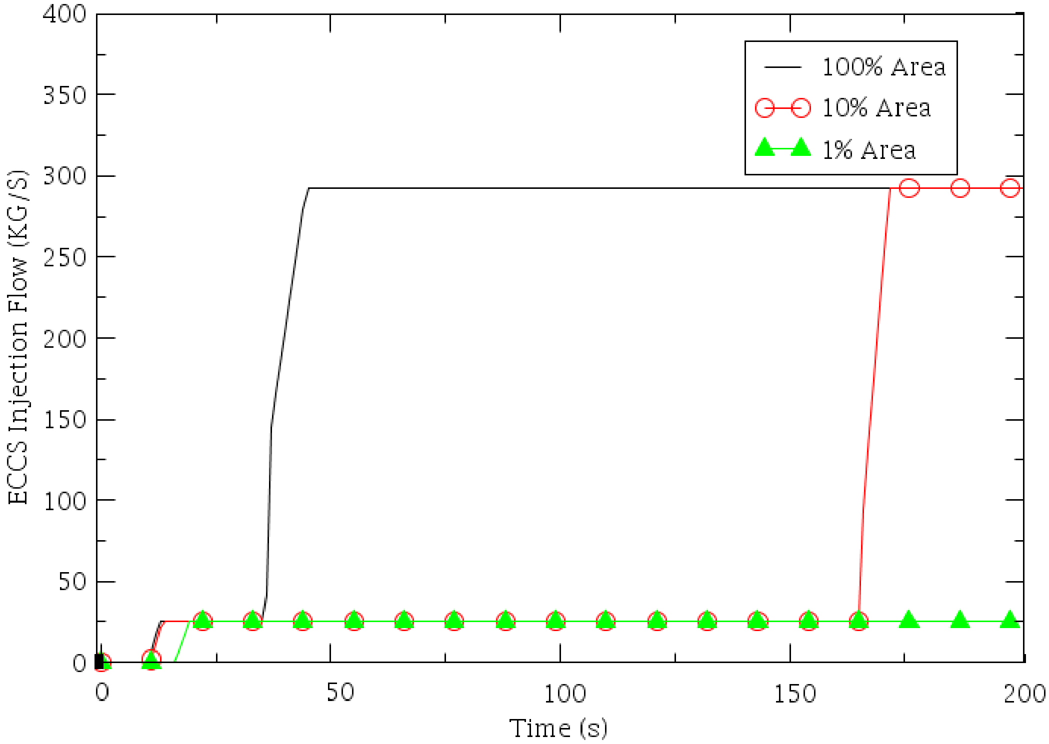

Figure 8   The ECCS injecting flow rate with various break area

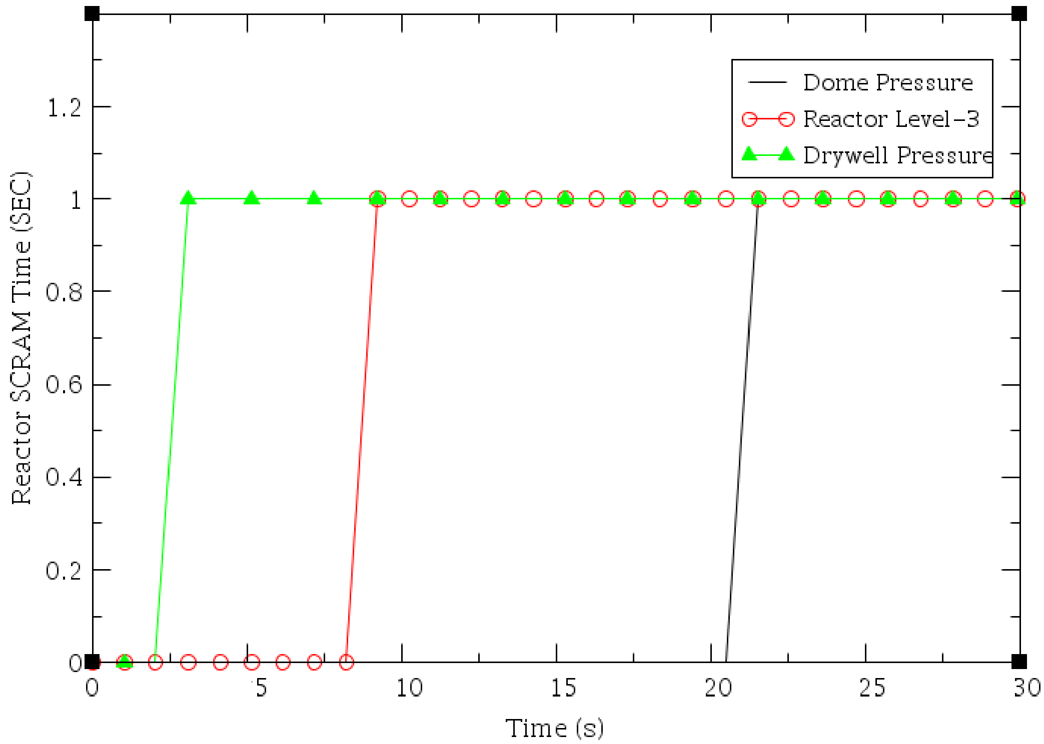

Figure 9 The SCRAM signals in 1% break area

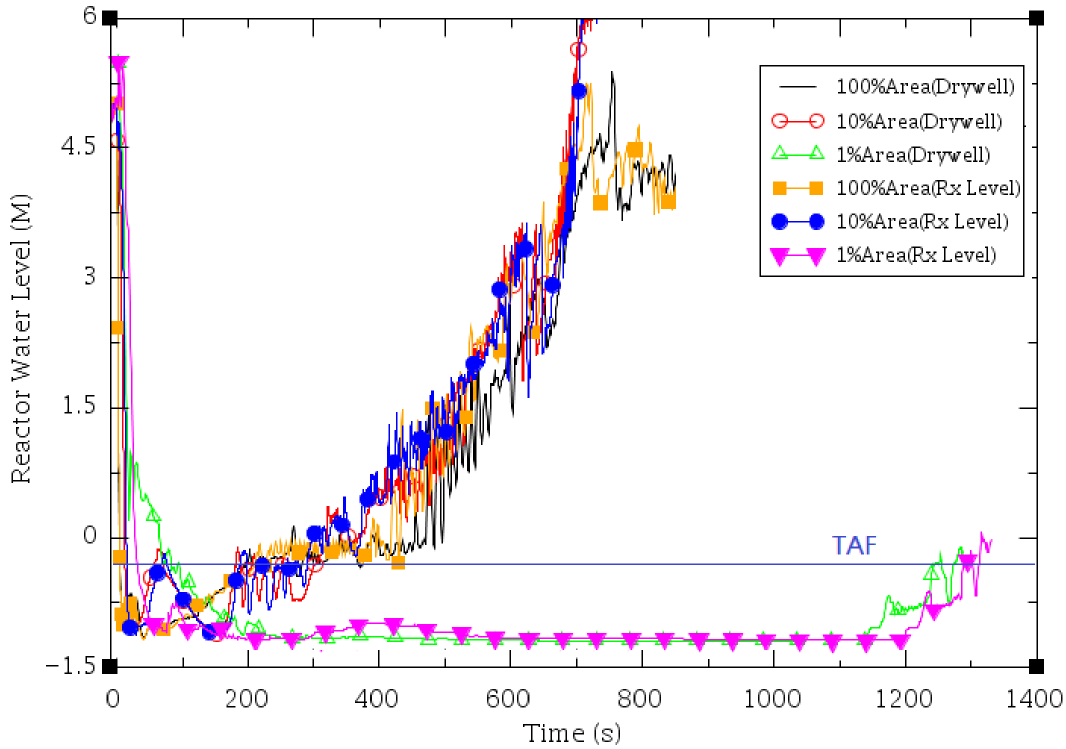

Figure 10　The reactor water level in two SCRAM signals and various break area

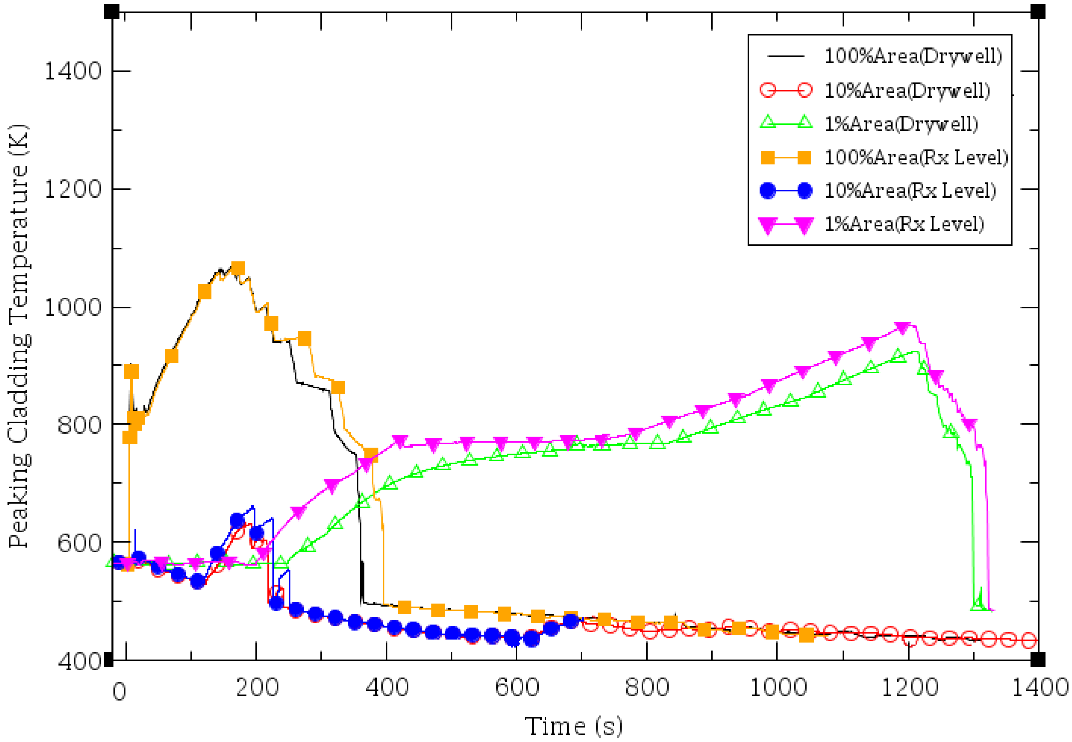

Figure 11 The peak cladding temperature in two SCRAM signals and various break area

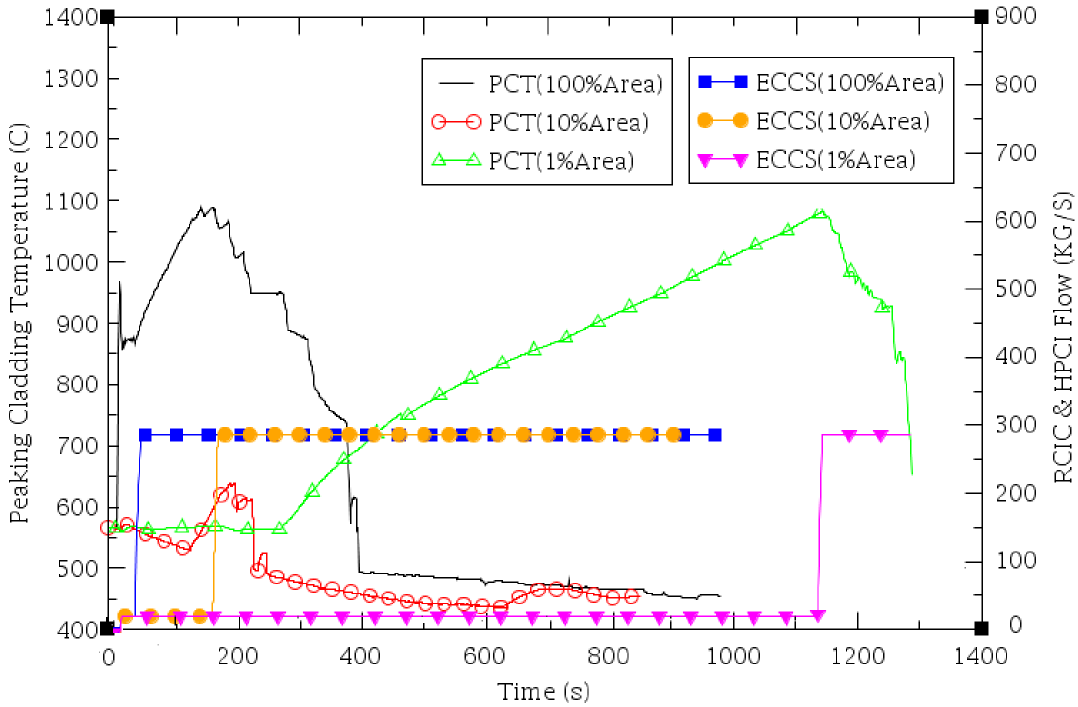

Figure 12 The peak cladding temperature with various break area

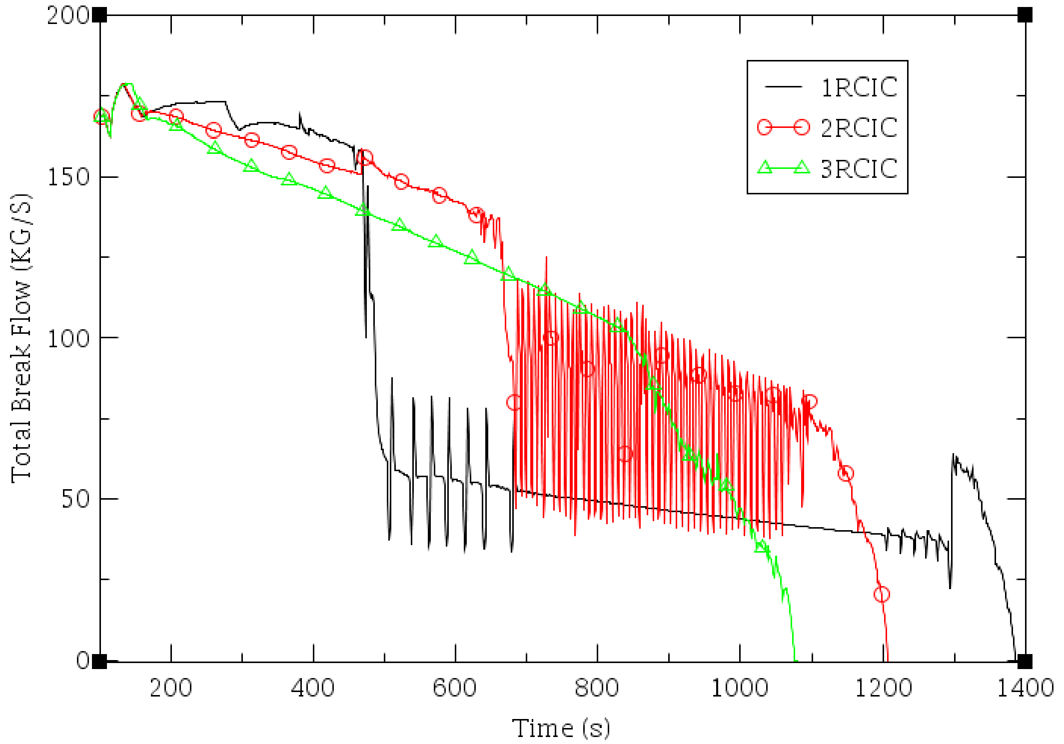

Figure 13 The total break flow with various RCIC flow

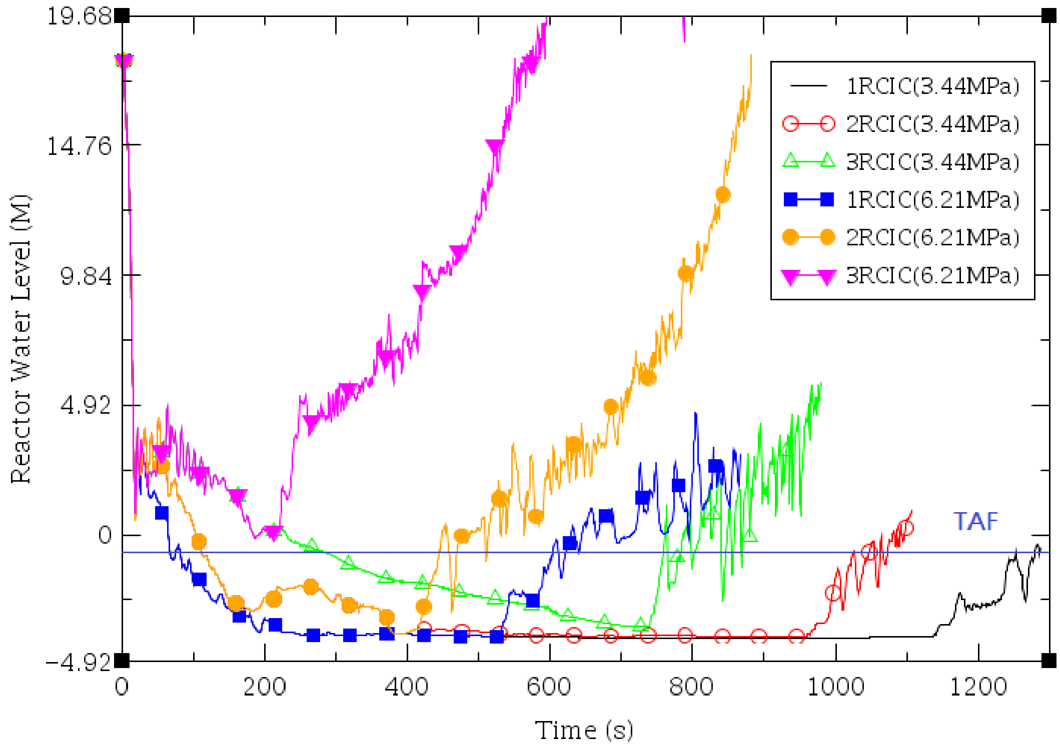

Figure 14 The reactor water level with various RCIC flow and HPCI injecting time

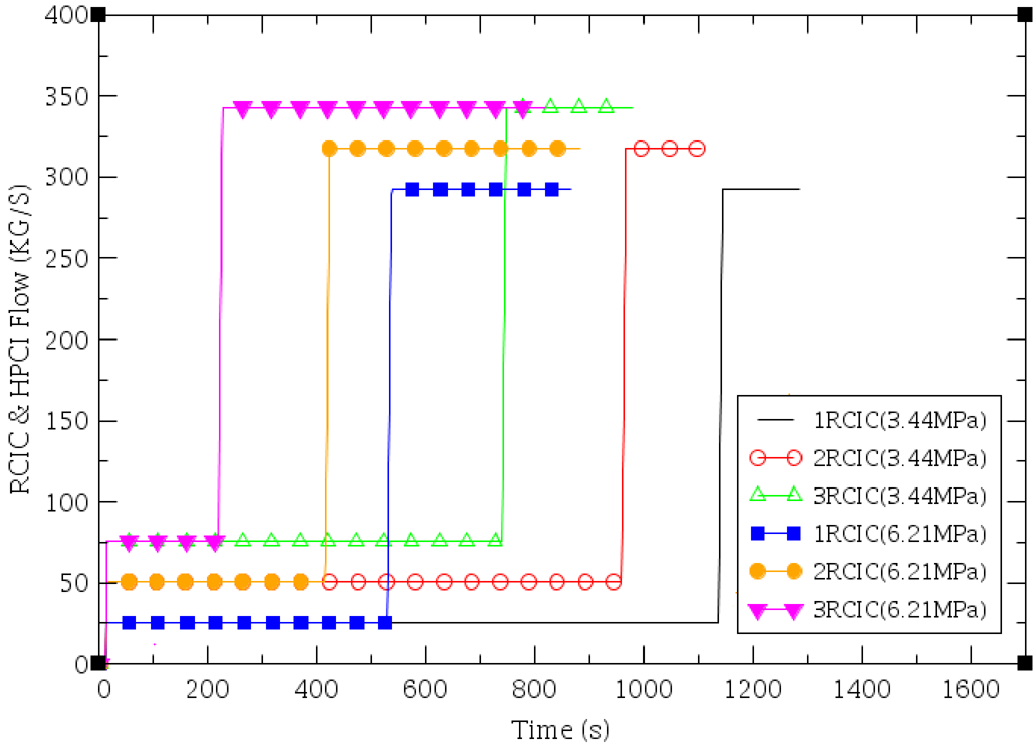

Figure 15 The ECCS injecting flow with various RCIC flow and HPCI injecting time

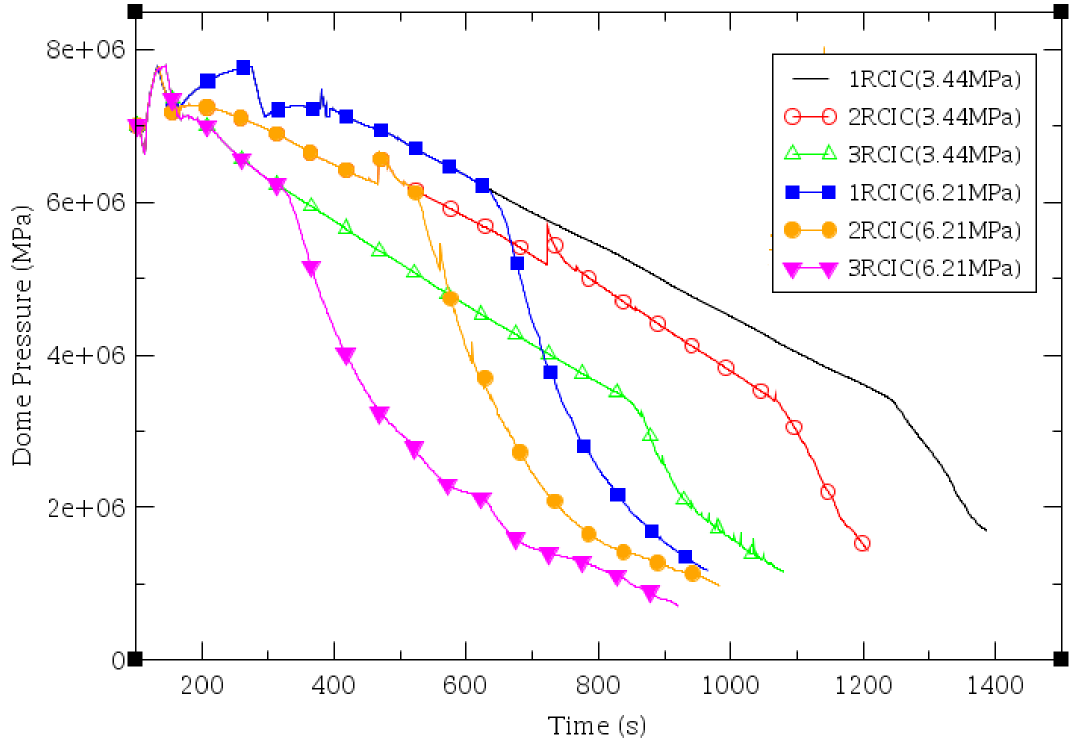

Figure 16　The dome pressure with various RCIC flow and HPCI injecting time

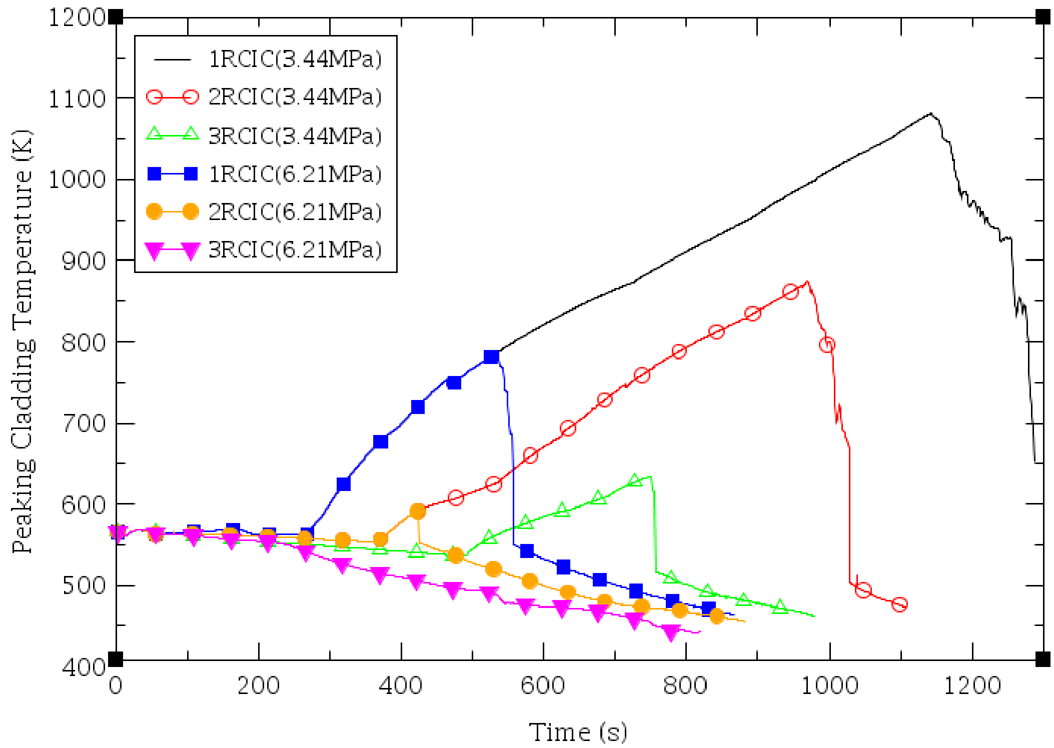

Figure 17 The peak cladding temperature with various RCIC flow and HPCI injecting time

Table 1 Setpoints of Chinshan NPP

Parameter/Event	Value
Thermal power (MW)	1828
Steam flow (Mkg/h)	3.46
Core flow (Mkg/h)	24
Feedwater flow (Mkg/h)	3.46
Vessel dome pressure (MPaG)	6.98
Number of reactor internal pumps	20
Reactor Scrams	
(1)Drywell Pressure High	Original +13.8 kPa
(2)Rx Low Water Level	Level 3(TAF+455cm)
(3)Dome High Pressure	7.54 MPa
RCIC Pump initialed	delay10 seconds
(1) Drywell Pressure High	Original +13.8 kPa
(2)Rx Low Water Level	Level 2(TAF+312cm)
HPCI Pump Initialed	delay 30 seconds
(1) Drywell Pressure High	Original +13.8 kPa
(2)Rx Low Water Level	Level 2(TAF+312cm)
HPCI Injecting Valve Open	Reactor Pressure<3.44 MPa
Recirculation Pump Tripped	Level 2
MSIV Closed	Level 2

Table 2 Event sequences (scrammed by high drywell pressure)

Events	Time Sequences (second)		
	100%Area	10%Area	1%Area
Break Begins	0	0	0
Reactor Scrams(High Drywell Pressure)	0.54	2.2	6.2
Dome Pressure High	None	14.0	21.4
Rx Level-3	0.04	4.7	8.1
Rx Level-2	1.0	7.8	12.5
Rx Level-1	3.0	11.0	19.6
RCIC Injecting	11.0	17.8	22.5
MSIV close	1.0	7.8	12.5
Rx Level below TAF	4.4	12.5	65
HPCI starts Injecting($P < 3.44$ MPa)	35.8	161	1202
Rx Level above TAF	246.8	194.1	1265
PCT reached	242.4	181.6	1100

Table 3 Event sequences (scrammed by low reactor water level)

Events	Time Sequences (second)		
	100%Area	10%Area	1%Area
Break Begins	0	0	0
Reactor Scrams(Rx Level-3)	1.0	6.0	13.0
Dome Pressure High	None	14.0	23.0
Rx Level-3	0.5	5.6	12.5
Rx Level-2	1.3	10.1	18.8
Rx Level-1	3.5	13.9	26.0
RCIC Injecting	11.3	20.1	28.8
MSIV close	1.3	10.1	18.8
HPCI starts Injecting (P < 3.44 MPa)	36.0	163	1198
Rx Level above TAF	214	194	1314
PCT reached	209.8	179.3	1285

4. CONCLUSIONS

(1) In the SBO event, the low pressure injecting systems with electrical pumps fail and the reactor pressure is still too high for the external low pressure injecting system. The electrical power of the critical control systems can be recovered by portable electrical generators after some improvements from the Fukusima experience, but the electrical pumps are difficult to recover by any portable device. The only possible driving force for the pumps in SBO is the steam generated by the residual heat. The current strategies in the extended SBO are mostly focus on the low pressure injection, but the reactor water level will decrease sharply while the reactor pressure is released that results in a higher PCT. To utilize the residual steam through the turbine driven RCIC system and HPCI system is an alternate mitigation strategy to makeup the reactor coolant than just to release the reactor pressure.

(2) The current RCIC injecting flow is only sufficient for the SBO event without LOCA. With the current RCIC flow, the duration of the reactor water level below TAF in the 1% break area is longer than in the 100% and 10% break areas because its reactor pressure is higher and results in a later HPCI injection. To increase the RCIC flow is helpful for the recovery of the reactor water level and the reactor pressure decline, but the impact is significant only in a small break LOCA. The reactor pressure declines faster with the increasing RCIC flow because more RCIC injecting flow results in a better cooling effect.

(3) The reactor water level is not recovered until the successful injection of HPCI in three break areas. Although the HPCI pump is started at the same time as the RCIC pump, the HPCI flow will not be injected into the reactor vessel until the open of the HPCI injecting valve. That is the reason why the reactor water level of 1% break area suffers a longer duration than those of 10% and 100% break areas. If the setpoint of the HPCI injecting valve is increased from 3.44 MPa to 6.21 MPa, the HPCI flow will inject earlier and the impact is more significant than the increasing RCIC flow. The reactor water level is recovered earlier and the peak cladding temperature is reduced more.

(4) Two alternative mitigation strategies, the increase of the RCIC flow and the earlier injection of the HPCI flow, are effective for the reactor water recovery in the combined accident of LOCA and SBO. The earlier injection of the HPCI flow is more effective to maintain the reactor water level and to decrease the peak cladding temperature. The combined effects of the increasing RCIC flow and the earlier HPCI injection are more significant. With the triple RCIC flow and the earlier HPCI injection in the 1% break area, the reactor water level could be maintained above the TAF level and the PCT will not increase. From the plant owner's point of view, to change the opening setpoint of the HPCI injecting valve for an earlier HPCI injection is easier and costs less than to increase the capacity of the RCIC pump.

(5) The containment pressure is found to increase very quickly even in a small break. The containment spraying (CS) provides a spraying function to reduce the drywell and wetwell pressure. But the current containment spraying pump uses an electrical pump which will fail in an SBO event. The containment pressure will be out of control and will lose containment integrity in a combined accident of LOCA and SBO. A backup containment spraying with a turbine driven pump is necessary. The excess HPCI flow can be utilized as an alternate mitigation strategy for the containment spraying function because the HPCI flow is more than the break flow in a small break LOCA.

In this report, the extreme event of a LOCA and an extended SBO is analyzed using the Chinshan TRACE model. Two alternative mitigation strategies with three different break areas are analyzed. From the experience of the Fukusima accident, although the reactor pressure is reduced, it is still too high for the alternative low pressure injecting system and will cause a serious low water level. Thus, an alternative mitigation strategy to utilize the best of the residual steam through the high pressure injecting systems, RCIC and HPCI, is necessary. The current design for RCIC flow is not sufficient even in a very small break LOCA. The reactor water level is not recovered until the successful injection of the HPCI flow in a LOCA. Two alternative mitigation strategies, increasing the RCIC flow and injecting the HPCI flow earlier, are effective for an earlier reactor water recovery and a lower peak cladding temperature in the extreme event of LOCA and SBO. The current strategies in the extended SBO are mostly focus on the low pressure injection, but the reactor water level will decrease sharply while the reactor pressure is released that results in a higher PCT. In this report, the alternate mitigation strategies adopting the turbine driven pumps, the high pressure injection systems, are analyzed to maintain an “enough” water level before the reactor pressure is released. As a result of this report, the recommended actions could be adopted for evaluations of the emergency operational procedures (EOPs) and severe accident mitigation guidelines (SAMGs).

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NRC FORM 335 U.S. NUCLEAR REGULATORY COMMISSION
(9-2004)
NRCMD 3.7

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
NUREG/IA-0440

2. TITLE AND SUBTITLE
The Alternate Mitigation Strategies Study of Chinshan BWR/4 by Using the LOCA and SBO Analysis of TRACE

3. DATE REPORT PUBLISHED

MONTH	YEAR
March	2014

4. FIN OR GRANT NUMBER

5. AUTHOR(S)
Jong-Rong Wang, Chun-Yu Chen*, Hao-Tzu Lin, Chunkuan Shih*

6. TYPE OF REPORT
Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*
Institute of Nuclear Energy Research
Atomic Energy Council, R.O.C.
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National Tsing Hua University
101 Section 2, Kuang Fu Rd., HsinChu
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9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*
Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
K. Tien, NRC Project Manager

11. ABSTRACT *(200 words or less)*
Chinshan nuclear power plant is the first NPP in Taiwan which is the BWR/4 plant. This research focuses on the development of the Chinshan NPP TRACE model and the LOCA combined SBO accident analysis. From the accident at the Japanese Fukusima NPP, an extreme event beyond the design basis is realized to be possible. The current mitigation strategies for the emergency core cooling systems (ECCSs) can be easily voided in the event of an extended station blackout (SBO), where all the onsite and offsite electrical power is failed. Although the electrical power of the critical control systems can be recovered by portable electrical generators, the electrical pumps are difficult to recover by any portable device. The only possible driving force of the pumps in SBO is the steam generated by residual heat. The current strategies in an extended SBO are mostly focused on low pressure injection, but the reactor water level will decrease sharply while the reactor pressure is reduced and that results in a higher PCT. In this report, the alternate mitigation strategies adopting the turbine driven pumps, the high pressure injection systems, are analyzed to maintain an "enough" water level before the reactor pressure is reduced. Three break sizes, 100%, 10% and 1%, on the recirculation suction line of Chinshan NPP which is the most serious LOCA in BWR/4 reactor are analyzed with three sensitivity studies: (1) the scram time, (2) the increase of RCIC injection flow rate, and (3) the earlier HPCI injection.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*
Taiwan
BWR/4
Chinshan NPP
Alternate Mitigation, LOCA, SBO
Peak cladding temperature (PCT)
Code Application and Maintenance Program (CAMP)
INER (Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C.)
SNAP (Symbolic Nuclear Analysis Program)

13. AVAILABILITY STATEMENT
unlimited

14. SECURITY CLASSIFICATION
(This Page)
unclassified
(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program

NUREG/IA-0440

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March 2014