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Sensitivity Study of the DEG LBLOCA Transient on the Counter-Current Flow Limitation by Using TRACE

Prepared by:
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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 a sensitivity study on the counter-current flow limitation (CCFL) model. The CCFL model plays a key role in any large break loss of coolant accident (LBLOCA) analysis since it affects the calculated discharge flow, reflooding time and peak cladding temperature (PCT). In this report, a sensitivity study on the CCFL model is performed, by modeling LBLOCA occurring at the Chinshan NPP. The scenario assumes 102% power and 75% core flow, with a double-ended guillotine (DEG) break on the recirculation loop, which is the most limiting LBLOCA for a BWR/4 reactor. Two break locations, i.e. on the suction and the discharge side of a recirculation pump, are evaluated, with high pressure core injecting (HPCI) and low pressure core spraying (LPCS) available whereas low pressure core injecting (LPCI) failed. The TRACE code is used for the analysis. The Chinshan TRACE model was benchmarked against steady-state and transient data contained in the plant FSAR report, as well as start-up data and the transient results using the RETRAN code. The thermal hydraulic phenomena in the lower plenum area and the jet pumps are also analyzed.

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 sensitivity study of the CCFL model of Chinshan NPP.

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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 a sensitivity study on the counter-current flow limitation (CCFL) model. The CCFL model plays a key role in any large break loss of coolant accident (LBLOCA) analysis since it affects the calculated discharge flow, reflooding time and peak cladding temperature (PCT). In this report, a sensitivity study on the CCFL model is performed, by modeling LBLOCA occurring at the Chinshan NPP. The scenario assumes 102% power and 75% core flow, with a double-ended guillotine (DEG) break on the recirculation loop, which is the most limiting LBLOCA for a BWR/4 reactor. Two break locations, i.e. on the suction and the discharge side of a recirculation pump, are evaluated, with high pressure core injecting (HPCI) and low pressure core spraying (LPCS) available whereas low pressure core injecting (LPCI) failed. The TRACE code is used for the analysis. The Chinshan TRACE model was benchmarked against steady-state and transient data contained in the plant FSAR report, as well as start-up data and the transient results using the RETRAN code. The thermal hydraulic phenomena in the lower plenum area and the jet pumps are also analyzed.

ABBREVIATIONS

ADS	Automatic Depressurize System
CCFL	Counter-Current Flow Limitation
CAMP	Code Applications and Maintenance Program
CHAN	A Component of TRACE to Simulate Fuel Bundles
CS	Containment Spraying
DEG	Double-Ended Guillotine
ECCS	Emergency Core Cooling System
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
LPCI	Low Pressure Core Injecting
LPCS	Low Pressure Core Spraying
MSIV	Main Steam Isolation Valve
NPP	Nuclear Power Plant
PCT	Peak Cladding Temperature
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.

The most limiting LBLOCA for a BWR/4 reactor is the DEG break on the recirculation loop, which is 100% break flow on each side. In this paper, two break locations, i.e. on the suction and discharge side of a recirculation pump are evaluated, with HPCI and LPCS available whereas LPCI failed. The TRACE code is used for the analysis. The CCFL model plays a key role in any LBLOCA analysis since it affects the calculated discharge flow, reflooding time and peak cladding temperature (PCT). The current CCFL researches focus on a PWR reactor [1][2], a phenomenon review [3] or a code investigation [4]. In this report, a sensitivity study on the CCFL model is performed, by modelling a LBLOCA occurring at the Chinshan BWR/4 plant. The scenario assumes 102% power and 75% core flow, with a DEG break on the recirculation loop. 10 CFR 50 Appendix K requires that any effect of fuel rod flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. In the events of DEG LOCAs, emergency core cooling system (ECCS) water is injected into the core through the upper plenum, lower plenum, or both. Of particular interest during the reflood period of a LBLOCA, is CCFL and pool formation above the upper core plate following core safety injection. Water that stagnates in the upper plenum and that is held there by CCFL cannot drain into the core and contribute to core cooling. When the CCFL occurs, the mass and heat transfer reduce between the gas and liquid phase, and a water pool forms above the top of fuels. This phenomenon reduces the injecting ECCS water in core area and results in a rapid increase of the fuel temperature. In Damerell's report [5], analysis of the CCFL occurrence is essential to the safety of nuclear reactors. In Issa's paper [3], the CCFL model is applied in the core area.

Chinshan nuclear power plant, the first nuclear power plant in Taiwan, operates at a thermal power of 1828MWt after the measurement uncertainty recovery project in 2008. The rated steam flow is 3.5 Mkg/h with the rated core flow at 24.0 Mkg/h and the reactor pressure at 6.98 MPa. In order to validate the numerical model and to initialize the LOCA simulation, 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 the following parameters have been benchmarked with the steady state of FSAR [6], the start-up data [7], the transient results of the RETRAN data [8], and the vendor's data [9][10]: 1. SEPD components adopted for separators and dryers and benchmarked with the 83% power 75% flow Turbine Trip test and 100% power 100% flow Load Rejection test [11][12], 2. The CHAN components are adopted for fuel bundles with a reactivity feedback and benchmarked with the inadvertent startup of HPCI test. [13], 3. The behaviors of the break flow, core flow, reactor water level and reflood time [14][15].

2. METHODOLOGY AND MODELING

2.1 Description of LBLOCA

The LOCA is a hypothetical accident described in 10CFR50.46 that consists of a loss of reactor coolant resulting from breaks in reactor coolant pressure boundary piping including a DEG break on the largest pipe. The broken pipe is postulated to occur inside the primary containment and upstream of the first isolation valve, which is a general test criteria described in 10CFR50 Appendix A.

The LOCA is generally separated into three phases: the blowdown phase, the refill phase, and the reflood phase. The duration of each phase is strongly dependent upon the break size and location. In the blowdown phase, the reactor pressure and the discharging flow are high and then reduce very soon with loss of reactor water level. During the blowdown phase of a LOCA, the decrease of coolant inventory will result in an increase in fuel cladding temperature due to core flow degradation, and for the larger breaks, the core becomes fully or partially uncovered. The reactor pressure and water level decrease rapidly during the blowdown phase. The safety relieve valve (SRV) and automatic depressurize system (ADS) are not activated in LOCA because the primary coolant system is depressurized rapidly during the blowdown phase. During the early phase of the depressurization, the exiting coolant provides core cooling. The HPCI and LPCS injecting water also provide some heat removal. The end of blowdown is defined to occur when LPCS reaches rated flow.

In the refill phase of a LOCA, the ECCS is functioning and there is a net increase of coolant inventory. During this phase, the core sprays provide core cooling and, along with LPCI, supply liquid to refill the lower portion of the reactor vessel. In this paper, the LPCI is assumed to be failed and only the HPCI and LPCS are available. During the refill phase, high steam pressure in the core region will prevent the penetration of ECCS liquid down into the reactor vessel. This phenomenon is called counter-current flow limitation (CCFL), and results in longer reflood times and consequently higher PCTs. In this stage, the heat removal is mainly by steam; thus the heat transfer from the core to coolant is less than the fuel decay heat rate, which results in continuous increase of fuel cladding temperature during the refill phase.

The reflood phase begins when the core is being reflooded with ECCS water and the mixture level reenters the core region. During reflood, cooling above the mixture level is provided by entrained reflood liquid and below the mixture level by pool boiling. Large axial variations of wall temperature and of heat fluxes occur in a very narrow region close to the quench front, which results in steep axial temperature gradients. Steam generated mainly at the quench front entrains water slugs, fragments, and droplets upward. The cladding quenches, very rapidly cooling to the saturation temperature as the cladding surface becomes wetted. The cladding temperatures were observed to turn around very shortly after the onset of reflood [9][10][16].

2.2 Chinshan TRACE Model

The TRACE model of Chinshan Nuclear Power Plant (1) is developed, 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 (Figure 1). The reactor vessel is divided into 88 cells with eleven axial elevations, four radial rings and two azimuthally divided sectors. There are 408 fuel bundles and 130 separators. The 408 fuel bundles are modeled by the CHAN component, a specific TRACE component which is located in axial level 4 and 5 with point kinetics and dynamic reactivity feedback. The separators and dryers are simulated by the SEPD component which is located from axial level 7 to 10. The Chinshan TRACE model includes two recirculation loops, with one recirculation pump and ten jet pumps in each, and four steam lines with one SRV, one main steam isolation valve (MSIV), one turbine control valve (TCV) and one turbine bypass valve (TBV) in each line. The HPCI injects coolant into the upper downcomer and flow through the lower plenum to the core area; the LPCS water is injected in the upper plenum area. The HPCI provides the injection mass flow at 267.8 kg/sec. The LPCS provides the coolant directly on the top of fuels with the mass flow at 472.2 kg/sec per loop.

The specific VESSEL component of TRACE code is able to model a more practical reactor vessel and its associated internals. The component is 1-, 2-, or 3D in a cartesian or cylindrical geometry and uses a six equation, two-fluid model to evaluate the flow through and around all internals so that more thermal-hydraulic phenomena can be analyzed. The internals of a reactor vessel can be modeled practically by the 3D VESSEL component including the downcomer, fuel-assembly, core area, core bypass flow, and upper and lower plenum. Modeling options and features incorporated into the VESSEL component are designed mainly for LOCA analysis, but the VESSEL component can be applied to other transient analyses as well. The energy transportation between the fluid and the reactor structure is important for LOCA analysis and is modeled by the heat-structure component of TRACE code. Structures that can exchange heat with the fluid in a reactor vessel include downcomer walls and the support plate and they are modeled as slabs and vessel rods. The convective heat transfer from any structure into the different fluid phases is modeled using Newton's law of cooling to represent the energy exchange rate between the structure and the fluid phase [17]. The energy equations of the different fluid phases for various heat-transfer regimes are included in the TRACE code to provide better correlation of the heat transfer coefficient (HTC) between the wall and fluid. Heat transfer from the fuel rods and other structures is calculated using flow-regime-dependent HTC obtained from a generalized boiling curve based on a combination of local conditions and history effects. Inner- and/or outer surface convection heat-transfer and a tabular or point-reactor kinetics with reactivity feedback volumetric power source can be modeled. The coupling algorithm is implicit in terms of the wall temperature, liquid, vapor, and saturation temperatures and explicit in terms of the heat transfer coefficient (1). For each new-time step, the wall HTCs (h) of a given structure are evaluated using the surface wall temperatures (T_w) and the fluid conditions obtained for the last time step. The new-time fluid-dynamics and conduction equations are solved using these HTCs, the new-time surface temperatures, and the new-time fluid and saturation temperatures where the total energy flux between the heat structure surface and the fluid cell can be written as

$$Q_{total}^{n+1} = h_l^n\left(T_w^{n+1} - T_l^{n+1}\right) + h_g^n\left(T_w^{n+1} - T_g^{n+1}\right) + h_{sat}^n\left(T_w^{n+1} - T_{sat}^{n+1}\right) \quad (1)$$

The term $h_{sat}^n\left(T_w^{n+1} - T_{sat}^{n+1}\right)$ represents heat flux to the liquid that goes directly to boiling and $h_l^n\left(T_w^{n+1} - T_l^{n+1}\right)$ is the liquid heat flux that contributes to heating of the liquid. Since both the heat structure and fluid use the same heat flux term at each time step, energy is conserved for the coupling between the conduction and flow equations.

2.3 Pressure Drops of Reactor Vessel and Recirculation Loop

The flow resistances in the reactor vessel and the recirculation loop are crucial in the LOCA analysis and closely related to the break flow rate, the pressure transient and the reflood time [16]. The pressure distributions in the vessel are validated with plant data, including the lower plenum, fuel support plate, fuel zone, separators, dryer and dome area [12]. The pressure distribution in the recirculation loop including the recirculation pump is also validated. (Figure 2) The recirculation system provides forced flow through the reactor core for better cooling efficiency and to be one of the power modulating methods of a BWR reactor. There is one recirculation pump and ten jet pumps of each recirculation loop. The suction of the recirculation loop is connected to the lower downcomer and is pressurized by the recirculation pump as a driven flow of the jet pumps which is about 40% of the core flow. The recirculation pump adopts a TRACE built in homologous pump which is based on the standard homologous-curves approach. These curves represent the performance of the pump in a normalized format, giving the normalized pump head as a function of the normalized volumetric flow and normalized pump speed. Homologous curves (one curve segment represents a family of curves) are used for this description because of their simplicity. These curves describe, in a compact manner, all operating states of the pump obtained by combining positive or negative pump-impeller angular velocities with positive or negative fluid volumetric flows. Each recirculation pump includes a motor-generator set which provides excess moment force and core flow to mitigate the impacts while trips. The recirculation pump is modeled by a centrifugal pump which calculates the pressure differential across the pump impeller and the pump impeller's angular velocity as a function of the fluid flow rate and fluid properties including two-phase effect. In this paper, the homologous pump is used with a two phase flow with the pressure and flow rate validated. When reactor water level is below level-2, the recirculation pump will be tripped and cause the decreasing of core flow. After the recirculation pump trips, the pump speed will decrease with a ramp rate at 10~13 seconds.

Since the breaks in this paper are located in the recirculation loop, the pressure distribution along the loop is important for analyzing the discharge flow. The flow resistance in the recirculation loop is critical in LOCA analysis and is closely related to the break flow rate and the pressure transient. The pressure distribution in the recirculation loop, including the recirculation pump, is validated with plant data as shown in Figure 2. The elevation and the flow resistance along the recirculation loop are built based on design data. The recirculation loop suction connects to the lower downcomer area. The upstream of recirculation pumps is the lowest point of the recirculation loop. The break flow at the recirculation pump suction is larger than the one at the discharge side because of the higher pressure and lower flow resistance. When coolant flows through the recirculation pump, the pressure is increased. The pressure peaks when the coolant flows through the nozzle of the jet pumps which have an abrupt cross area.

2.4 The CCFL Model of the TRACE Code

10 CFR 50 Appendix K requires that any effect of fuel rod flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The restrictions of Appendix K generally cause ECCS evaluation models to predict the PCT to occur during the reflood phase of the LBLOCA. Best estimate codes, however, have generally shown that the PCT occurs during the blowdown phase because of the improved understanding of the various reflood phenomena. In the event of a LOCA [16], ECCS water is injected into the core through the upper plenum, lower plenum, or both. Of particular interest during the reflood period of a LBLOCA, is CCFL and pool formation above the upper core plate following core safety injection. The ECCS water that is stagnated in the upper plenum and held there by CCFL cannot drain into the core and contribute to core cooling. Water flows downward from the upper plenum into the core under the influence of gravity. This downward flow is opposed by steam flow upward from the core. When the CCFL occurs, the mass and heat transfer reduce between the gas and liquid phase, and a water pool forms above the top of fuels. As ECC water enters the hot core, some of it will turn into steam. The increase in pressure tends to suppress the reflooding velocity in the core. This retarding effect of steam on the reflood rate is referred to as steam binding. This phenomenon reduces the injecting ECCS water in core area and results in a rapid increase of the fuel temperature [5]. Analysis of the CCFL occurrence is essential to the safety of nuclear reactors [3]. The CCFL behavior in BWRs is important because steam generation in the fuel bundles can restrict the emergency core cooling spray water from entering the top of the bundles. In LBLOCA, the reactor water level reduces rapidly because of its massive discharging flow, and the HPCI is not sufficient. In this paper, only the LPCS is assumed to be available and sprays coolant from the top of fuels. During the refill phase of a LOCA, the counter-current flow would occur at the top tie plate and the core support plate in both the BWR and PWR reactor vessels. When the LPCS sprays coolant into the top of fuels, the coolant forms a pool on the top of core because the pressure in the core area is high enough to prevent liquid flow down. This phenomenon is called the countercurrent flow limitation (CCFL), where only penetration liquids get into the core area. The cladding temperature increases rapidly when CCFL occurs because there is little mass and heat transfer between liquid and steam. When CCFL occurs, the ECCS

coolant would be prevented from penetrating down to the fuel zone which why this is called the ECCS bypass model.

The CCFL is defined when the downward flowing liquid in a countercurrent flow is about to change its direction and flow upward. This condition can be obtained if the steam flow rate is increased while keeping the liquid flow rate constant in a countercurrent flow. Depending upon the geometry of the equipment used, the occurrence of the CCFL could vary. In reactor applications, CCFL can occur at flow area restrictions when liquid flows downward through rising gas or vapor. For example, in the tie-plate region during reflood, the upward flow of steam can prevent or limit the down flow of liquid. Since the TRACE code is well designed for the LOCA analysis, both categories of CCFL phenomena, the single pipe and the perforated plate, are adopted and validated. The TRACE code is capable of calculating the CCFL and predicting the CCFL by resolving the conservation equation of the CCFL model. The TRACE code models four principle regimes, the three bubbly/slug flow regimes and one annular/mist flow regime, on the vertical and horizontal flows. The CCFL correlation of the TRACE code can be applied in its 3D component and 1D component. The CCFL correlations describe a superficial liquid mass flux in down-flow versus a superficial gas mass flux in up-flow which are represented as three types, the Wallis type (6) [18], the Kutateladze type (7) [19] and the Bankoff type (8) [20]. The CCFL model of the TRACE code adopts the three correlations and is verificated [4][21].

$$j_g^{*1/2} + m_W j_f^{*1/2} = C_w \tag{6}$$

$$K_g^{*1/2} + m_K K_f^{*1/2} = C_K \tag{7}$$

$$H_g^{*1/2} + m_B H_f^{*1/2} = C_B \tag{8}$$

Where m and C are constants determined by experiments, j_k^* , K_k^* and H_k^* are dimensionless superficial velocity, k is liquid or gas phase.

$$j_k^* = j_k \left[\frac{\rho_k}{gd(\rho_f - \rho_g)} \right]^{1/2} \tag{9}$$

$$K_k^* = j_k \left[\frac{\rho_k^2}{g\sigma(\rho_f - \rho_g)} \right]^{1/4} \tag{10}$$

$$H_k^* = j_k \left[\frac{\rho_k}{gw(\rho_f - \rho_g)} \right]^{1/2} \tag{11}$$

Where j_k and ρ_k are the superficial velocity and the density of phase k , d is the hydraulic diameter, σ is the surface tension, and w is the interpolative length scale.

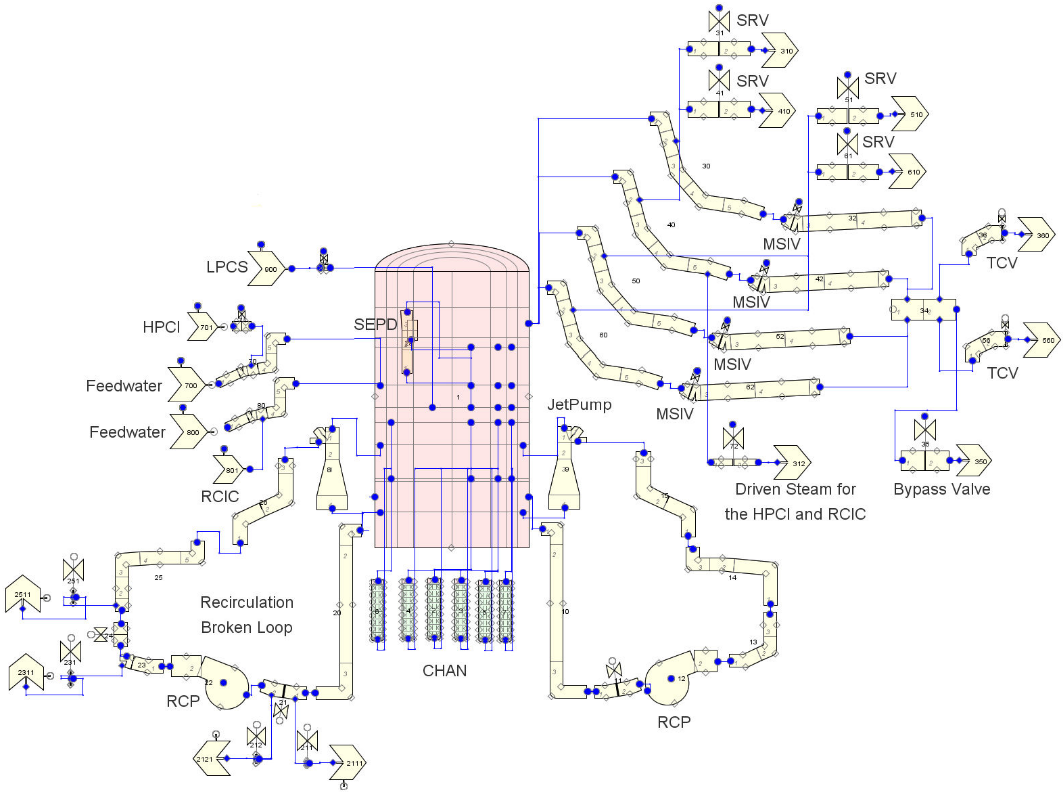

Figure 1 The TRACE model of Chinshan nuclear power plant

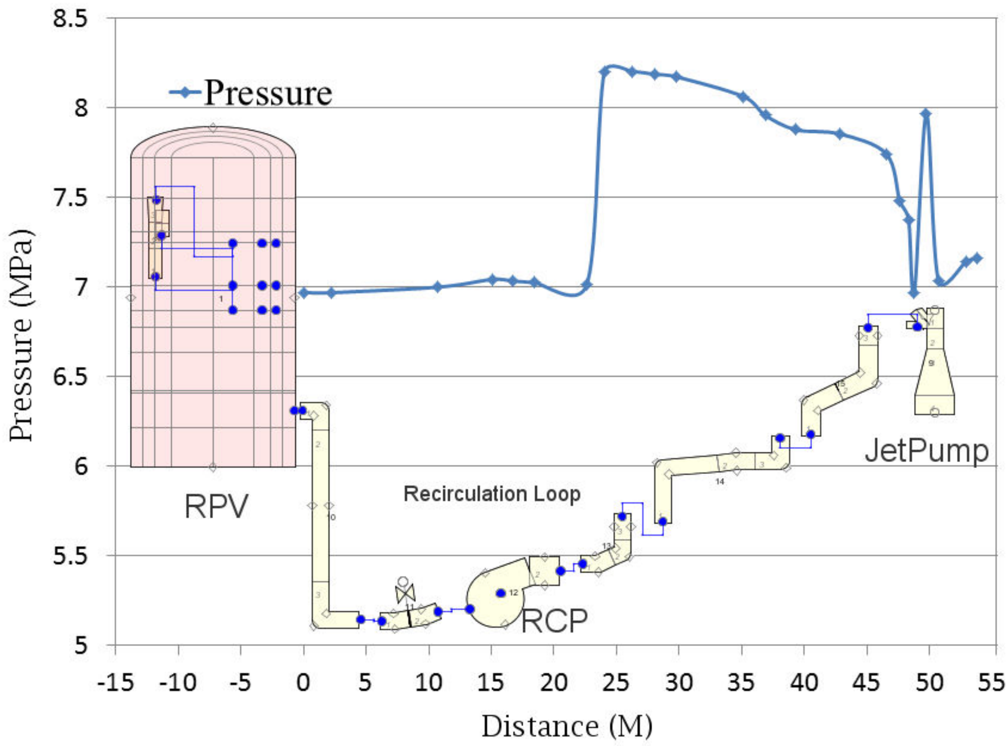

Figure 2 Pressure distributions in recirculation loop

3. RESULTS AND DISCUSSION

When DEG break on recirculation loop begins, reactor water level decreases fast because of massive discharging flow. In this report, Table 1 shows the initial conditions and the reactor is assumed to scram by the signal of reactor low water level (Level-3) that occurs in less than one second soon after the DEG break begins. In Figure 3, the break flow is large in the first twenty seconds and most of the coolant in the reactor inventory is discharged. In the first few seconds, the break flow at the recirculation pump discharge peaks higher than at the suction because the recirculation pump is still running, but it decreases rapidly after the recirculation pumps trip at reactor water level 2. After the closure of MSIVs, the different discharging flow can be noted from a peak of reactor pressure at the recirculation pump discharge (Figure 4). The reactor pressure at the recirculation pump suction does not increase because of its larger break flow.

When the pressure in lower plenum decreases, a flash phenomenon occurs and results in a peak of core flow (Figure 5) and cladding temperature decrease (Figure 6). The flash phenomena occur earlier and more significant at the recirculation pump suction because of its larger discharging flow. The flash at the recirculation pump suction occurs earlier at 8.5 seconds while the one at the discharge occurs at 15.3 seconds (Table 2). From Figure 5 and 6, the core flow is found to be a main effect on the cladding temperature. When the DEG break occurs, the cladding temperature increases immediately because of the massive break flow, fast decreasing core flow and reactor water level. The sudden increase of core flow impacted by the flash phenomenon provides a better cooling effect and results in the decrease of cladding temperature (Figure 6). Although the cladding temperature decreases for a while, the massive discharging coolant will result in continuous loss of coolant inventory and consequently the increasing cladding temperature.

When the DEG break occurs, the reverse flow is found in the path from the lower plenum to downcomer through jet pumps and becomes a part of break flow. The core flow decreases rapidly when the DEG breaks occurs and the reverse flow is observed as a negative value of the jet pumps exit flow in Figure 7. The jet pumps exit flow transfers from positive to negative, and then turns to zero after most of the coolant having been discharged. The jet pumps uncover and recirculation loop uncover occur in less than twelve seconds, and it is earlier at the recirculation pump suction. The coolant flashes in lower plenum soon after the recirculation pipes uncover. When the flash occurs, a sudden pressure increase in the lower plenum will drive out the coolant in the lower plenum through the core area, jet pumps and downcomer. In Figure 8, the jet pumps suction flow also peaks up when the flash occurs.

In this report, LPCI and containment spraying (CS) are assumed to be failed. The MSIV closes and the recirculation pump trips at level-2. After the closure of MSIVs, the main steam flow stops. The HPCI is initiated at the signal of reactor water level-2 and injects coolant into the vessel via the feedwater line B. The LPCS is initiated at the signal of reactor water level-1 and injects coolant directly on the top of active fuels (TAF). The cladding temperature increases rapidly because of the fast decrease of core flow. Since the break flow is larger than the HPCI injecting flow, reactor water level keeps decreasing until the LPCS injected. At the beginning of LPCS injection, the reactor water level is still below the fuel support plate and the increasing cladding temperature keeps increasing. In the mean time, the steam fills the whole fuel zone with high pressure that prevents the LPCS injecting water from flowing into the fuels bundles and forms a pool on the top of fuels. The coolant will penetrate or dump into the fuel zone when the static pressure of the pool is greater than the steam pressure. The phenomenon is called CCFL. In this stage, the conductive heat transfer from liquid to steam is not efficient, resulting in a higher cladding temperature. With the more coolant penetrating or flowing to the fuels zone, more thermal heat is removed and the steam pressure is reduced, and the lower plenum begins to refill. But fuel temperature will keep increasing until reflood.

3.1 The Break Flow of the Recirculation Loop

The DEG breaks at the recirculation pump suction and the discharge are analyzed. The break flow at the recirculation pump discharge is less because the flow resistance of the recirculation pump reduces the break flow (Figure 3), which results in a slower decline of reactor pressure (Figure 4) and a later decline of reactor water level (Figure 9). The effect with a lower resistance is more significant in the first few seconds while the break flow is large. From Figure 9, the duration of the fuel uncover is longer at the recirculation pump suction and results in higher PCT (Figure 6). The break flows with and without the CCFL model are almost the same. Several data are about the same cures with and without the CCFL model in the first 40 seconds from Figure 3 through Figure 13, which implies that the CCFL model only affect the reflood time and PCT.

3.2 The Transient Curves of Core Flow

In Figure 5, the core flow decreases rapidly soon after the DEG break begins. The fast decreasing core flow results in a rapid increase of cladding temperature. The coolant in the lower plenum becomes reverse-flow to the downcommer through jet pumps as the negative values of the jetpump exit flow and discharge flow indicated in Figure 7 and 8. In addition, the reverse-flow is also found in the driven flow of jet pumps at the broken loop, but it is smaller because of its higher elevation and flow resistance of nozzles. Because of the massive discharging flow, the suction end of jet pumps uncovers in 2.5 and 5.3 seconds at the recirculation and the discharge, respectively. The recirculation pump trips at the signal of reactor water level-2. Because the discharging flow on the recirculation loop is massive, the reactor water level decreases very fast.

Because the break flow at the recirculation pump suction is much larger than at the discharge (Figure 3), the reverse-flow is more significant at the former one. In Figure 10, the reverse-flow at the driven nozzle of jet pumps is found smaller at the recirculation pump suction because its driven nozzle connects to the recirculation pump with flow resistance added .

In Figure 5, the core flow peaks up in both the DEG breaks at the recirculation pump and the discharge because the coolant in the lower plenum flashes and results in a peak of pressure. The flash phenomenon can be evaluated through Figure 11 when the steam flow in lower plenum increases suddenly and then peaks up and down. The flash phenomenon occurs earlier and more significantly at the recirculation pump suction because its water level in the lower plenum reduces faster (Figure 12).

3.3 The Curve of Peak Cladding Temperature

The cladding temperature is found to be closely related to core flow at the beginning of LOCA. In Figure 6, the cladding temperature peaks up fast while the core flow decreases rapidly. The core flow reduces rapidly because of the massive break flow and the trip of recirculation pumps. In Figure 5, the core flow peaks at the recirculation pump suction while a rapid cladding temperature decrease is observed in Figure 6. The rapid temperature decrease is not observed at the recirculation pump discharge. It is the reason that the first peak of the cladding temperature at the recirculation pump discharge is higher than at the suction in Figure 3 although its break flow is less? After the first peak of cladding temperature, the temperature decreases for a while because of the sudden peak of core flow caused by the flash phenomena in lower plenum (Figure 5). In Figure 6, the peak cladding temperature at the recirculation pump suction is lower and decreases earlier than at the discharge. This can be seen in Figure 5 with an earlier peak of core flow at the recirculation pump suction. The core flow peaks up because of the flash phenomena in lower plenum. In Figure 5, the core flow decreases fast soon after the break occurs and results in rapid increase of PCT in Figure 6. When the reactor pressure and water level decrease causing by massive break flow, the coolant flash occurs in the lower plenum and results in a peak of core flow (Figure 5) and the flash phenomenon is more significant at the recirculation pump suction.

In addition, the HPCI is initiated at the signal of reactor water level-2 and it provides excess coolant to the vessel at 265.5 kg/sec via the feedwater line B which is also helpful to reduce the cladding temperature. Since the HPCI flow is still less than the break flow, the reactor water level will not recover before LPCS injection. In this stage, the transient curve of cladding temperature is found closely related to reactor water level after the sensitivity studies with the core flow and scram time. When comparing Figure 6 with Figure 9, the PCT is found to be closely related to the recovery of reactor water level. The higher break flow at the recirculation pump suction will result in a faster decline and lower reactor water level with faster increase and higher peak cladding temperature. The time to reflood at the recirculation pump suction is longer than at the discharge (Figure13). The later recovery of reactor water level will result in a higher PCT. The PCT at the recirculation pump discharge is the lowest one among the four tests and is the highest one at the recirculation pump suction with CCFL model.

3.4 The Impact of CCFL Effect

In Figure 6, the second peak of cladding temperature with CCFL effect is found to be higher than without CCFL effect because less ECCS water penetration provides less cooling in the fuel zone. Although the cladding temperature decreased slightly after the sudden increase of core flow, it keeps increasing because of the continuing loss of coolant inventory. In this paper, the CCFL model of the core zone is analyzed. The impact on the steam pressure of the fuel zone during the refill phase is found to be critical for the reflood time and PCT. The ECCS water will penetrate into the fuel zone only when the static pressure of the pool is higher than the steam pressure in the fuel zone. The more coolant penetrating into the core area results in a more heat removal and a faster temperature decline. In Figure 9 and 12, although the HPCI and LPCS are initiated almost at the same time in the four test cases, a later recovery of reactor water level is found with the CCFL model because the LPCS flow is blocked and forms a pool above the top of fuels. The curve of reactor water level decline at the beginning of LOCA is found not affected by the CCFL effect. The reason is that the reactor water level decline is mainly related to the break flow.

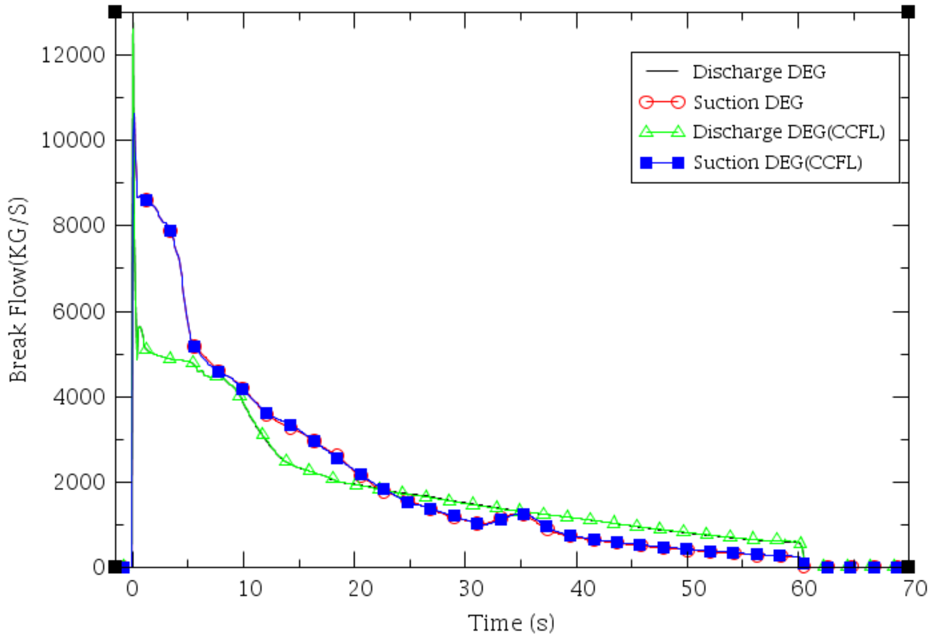

Figure 3 Comparison of the total break flow

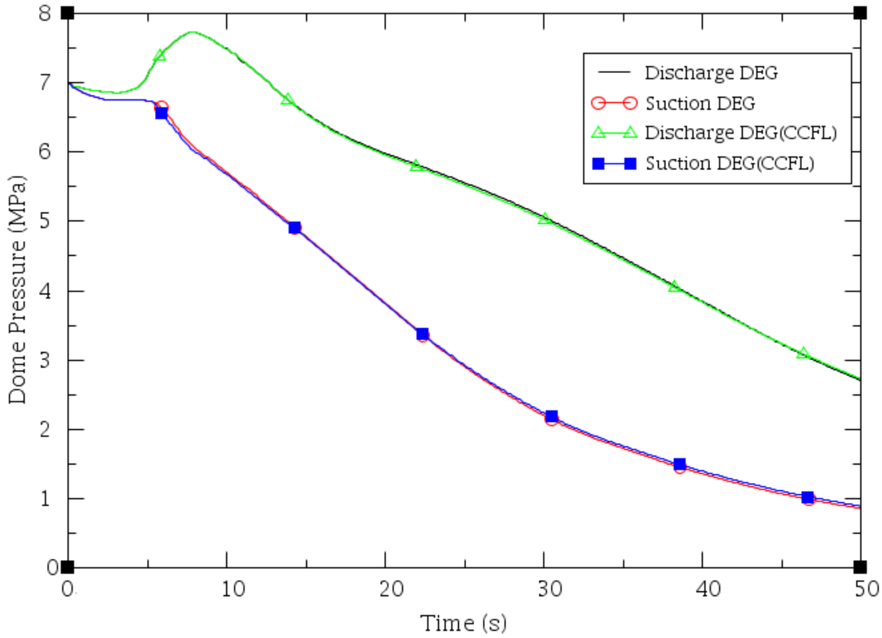

Figure 4 Comparison of reactor pressure

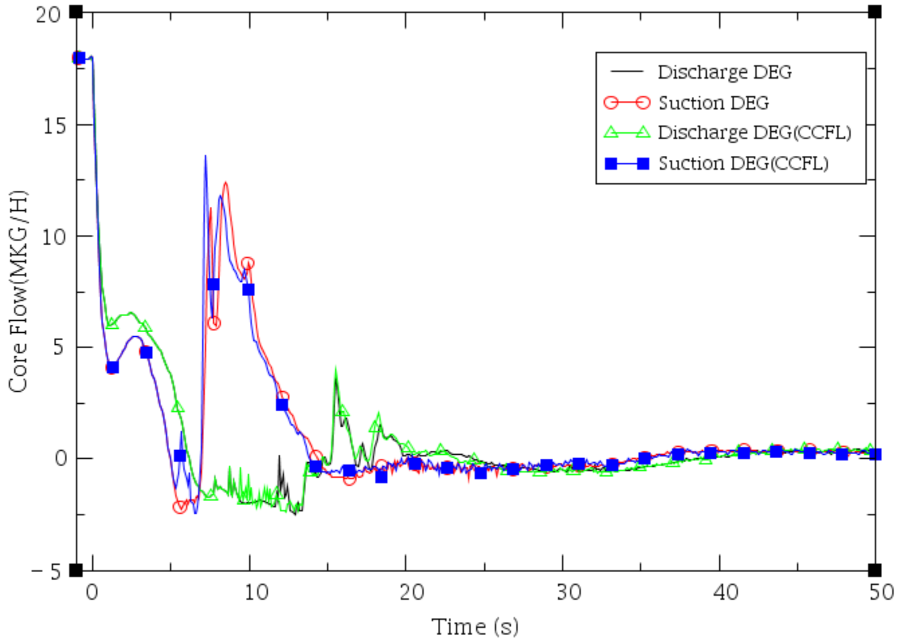

Figure 5 Comparison of the core flow

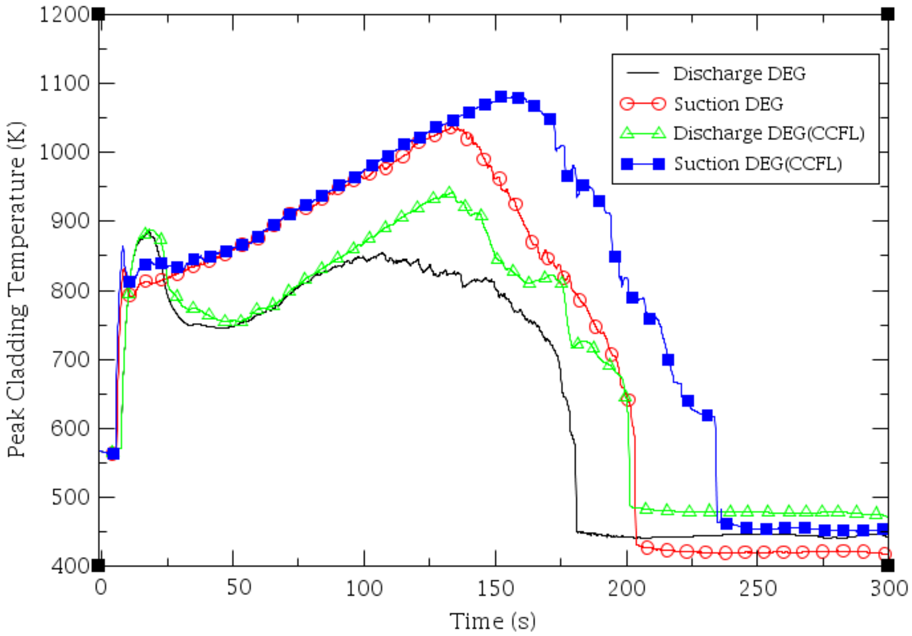

Figure 6 Comparison of peak cladding temperature

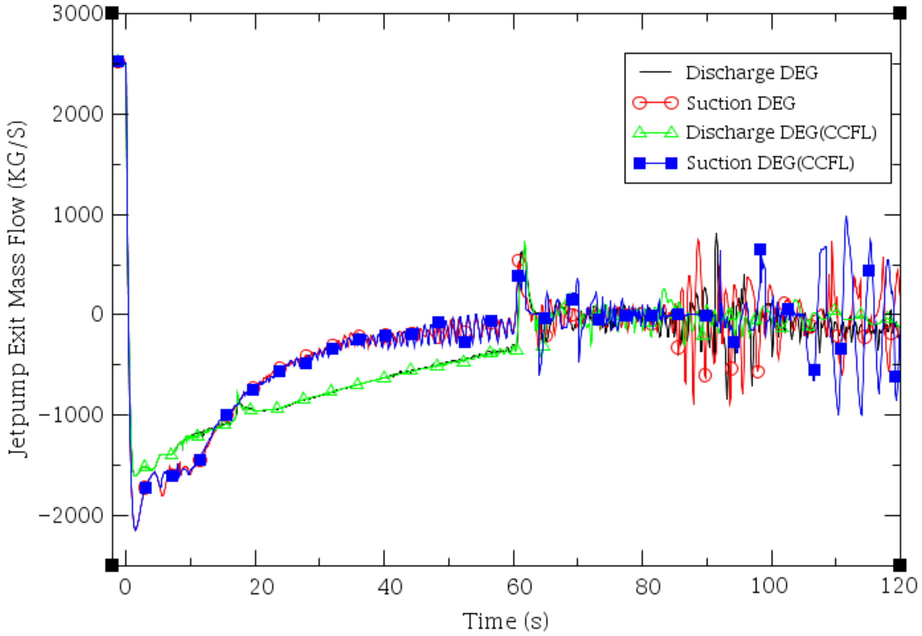

Figure 7 Comparison of the jet pumps exit flow rate at the broken loop

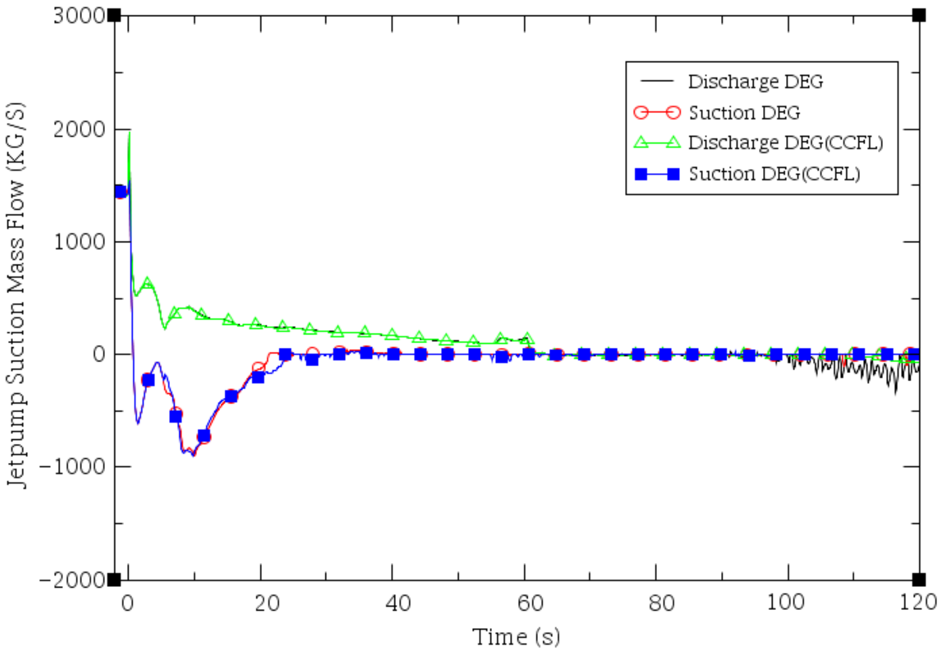

Figure 8 Comparison of the jet pumps suction flow rate at the broken loop

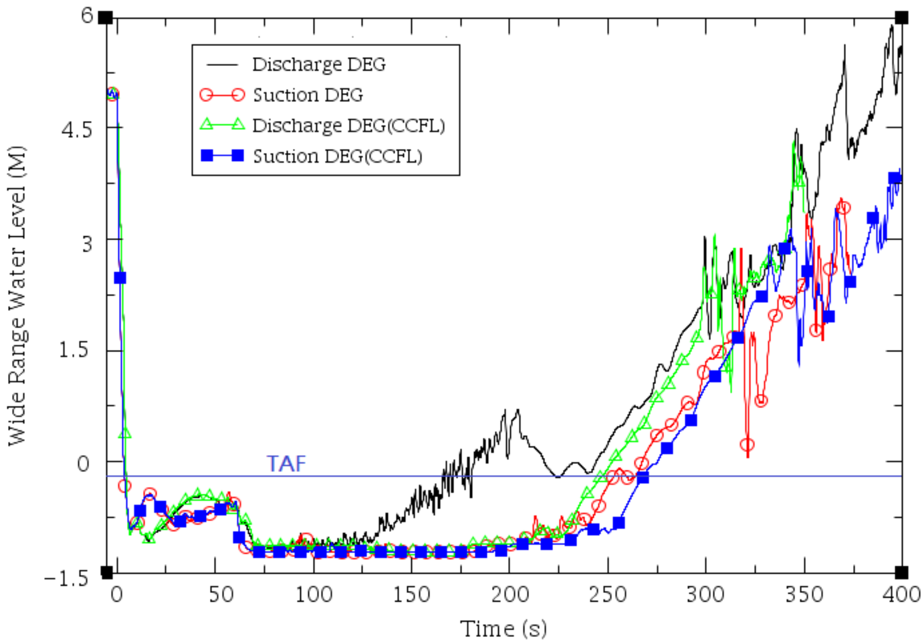

Figure 9　Comparison of the reactor water level

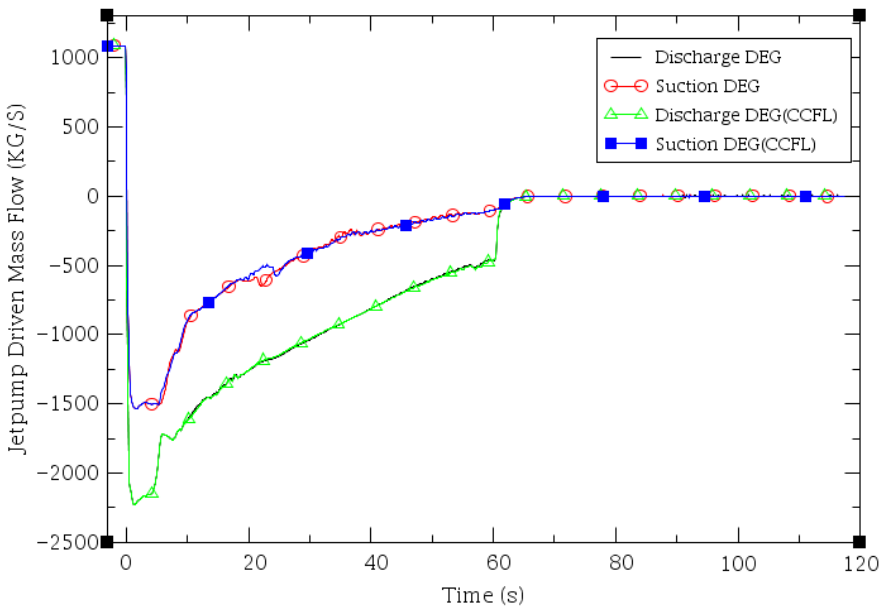

Figure 10　Comparison of driven flow rate of jet pumps at the broken loop

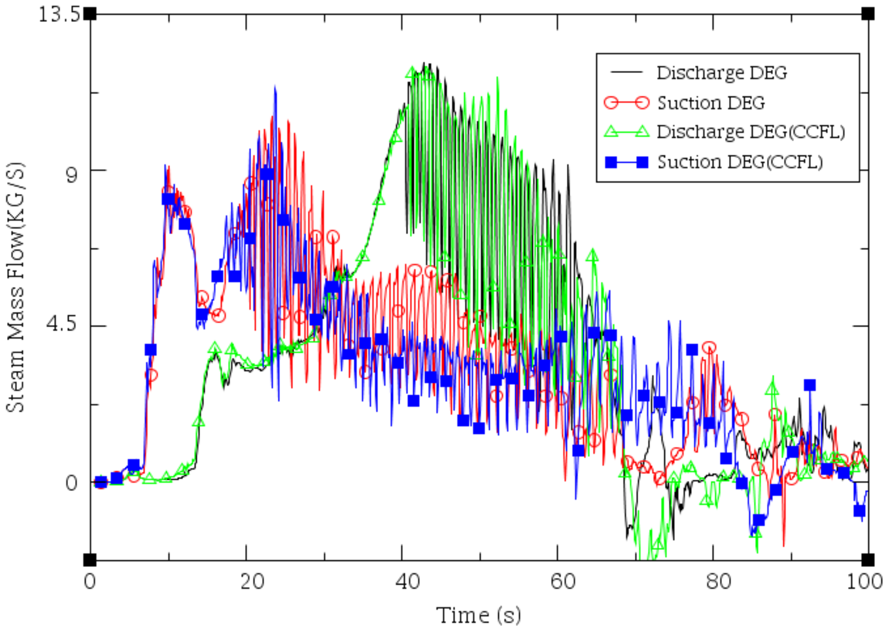

Figure 11 Comparison of the steam flow in lower plenum

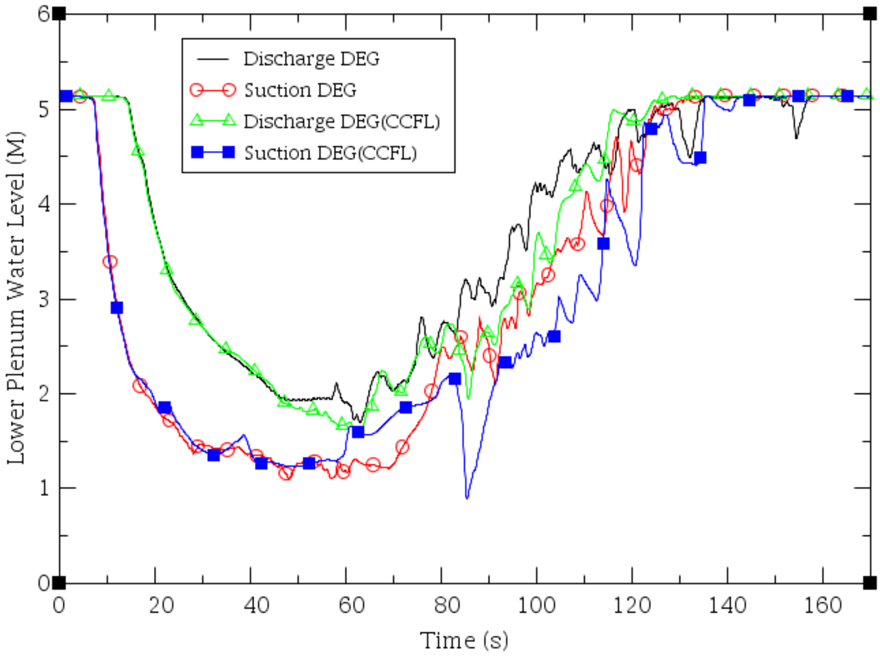

Figure 12 Comparison of the water level in lower plenum

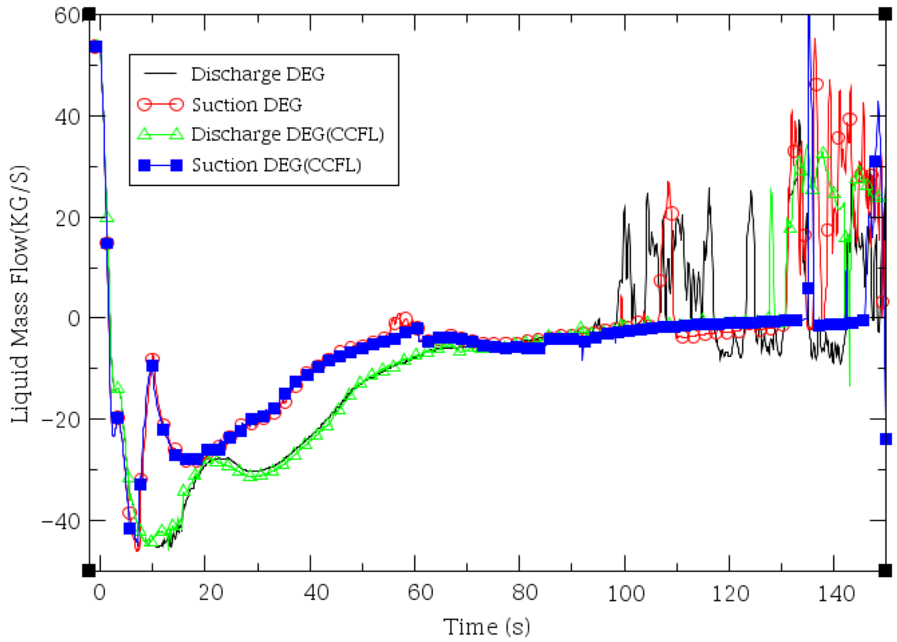

Figure 13 Comparison of the liquid flow of fuel support

Table 1 The Initial conditions for the LBLOCA of Chinshan NPP

Parameters	102%POWER 75% FLOW
Thermal Power (MW)	1864
Steam Flow (Mkg/h)	3.54
Core Flow (Mkg/h)	18
Feedwater Flow (Mkg/h)	3.54
Recirculation Loop Flow (Mkg/h)	3.95
Dome Pressure (MPaG)	6.98
Middle Core Pressure (MPaG)	7.09

Table 2 Event Times for LBLOCA of Recirculation Loop of Chinshan NPP

Event/Setpoint	Value(Seconds) Discharge DEG	Suction DEG	Discharge DEG(CCFL)	Suction DEG(CCFL)
Initial break	0	0	0	0
Initiate Scram(reactor water level 3)	0.5	0.3	0.5	0.3
Reactor water level 2	2.1	1.0	2.1	1.0
Steam flow stop(MSIV closed)	2.4	1.3	2.4	1.3
Reactor water level 1	4.1	3.0	4.1	3.0
Jet pump suction uncover	5.3	2.5	5.3	2.5
Recirculation Pipe uncovers	11.4	6.0	11.4	5.8
Lower Plenum Flashes	15.3	8.5	15.3	8.0
Recirculation Isolation Valve Begins to Close	17.6	7.3	17.6	7.1
HPCI Flow Starts	22	21	22	22
LPCS Flow Starts	54	53	54	53
Recirculation Isolation Valve Fully Close	64	54	64	54
Start Reflooding	98	99	128	135
PCT Reached (Temperature, K)	17 (883K)	133 (1035K)	133 (950K)	155 (1081K)
Water Above BAF	144	223	204	233
Water Above TAF	246	267	252	227

4. CONCLUSIONS

With the CCFL effect considered, the impacts on the steam pressure in the fuel zone during the refill and reflood phase are critical for the time to reflood and PCT. Although the HPCI and LPCS are initiated almost at the same time, the time to reflood at the recirculation pump suction takes longer than at the recirculation pump discharge because the later with a flow resistance of recirculation pump, which results in a less break flow. The CCFL effect does not affect the break flow but closely related to the reflood time. The time to reflood with the CCFL model also takes longer than without the CCFL model because the ECCS water forms a pool above the fuels with less cooling, which results in a much higher PCT.

The break flow at the recirculation pump suction is larger than at the recirculation pump discharge because of the flow resistance of recirculation pumps, which affect the break flow more at the beginning of DEG LOCA. The break flow rate is massive at beginning that the flow resistance of recirculation pumps plays an important role and results in a higher break flow at the recirculation pump suction. The break flow at the recirculation pump suction is larger, which results in a faster decline of reactor water level and pressure. The cladding temperature peaks immediately at the beginning of LOCA because of the fast decreasing core flow. The cladding temperature decreases for a while because the sudden peak of core flow, which is caused by the flash phenomena, provides a better cooling. But the reactor water level is found not to recover without LPCS injection that the cladding temperature will keep increasing before the core is reflooded. With the Chinshan TRACE model adopted, more LOCA events will be analyzed in the future. The sensitivity study on the pellet-cladding gap is the next planned analysis.

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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 a sensitivity study on the counter-current flow limitation (CCFL) model. The CCFL model plays a key role in any large break loss of coolant accident (LBLOCA) analysis since it affects the calculated discharge flow, reflooding time and peak cladding temperature (PCT). In this report, a sensitivity study on the CCFL model is performed, by modeling LBLOCA occurring at the Chinshan NPP. The scenario assumes 102% power and 75% core flow, with a double-ended guillotine (DEG) break on the recirculation loop, which is the most limiting LBLOCA for a BWR/4 reactor. Two break locations, i.e. on the suction and the discharge side of a recirculation pump, are evaluated, with high pressure core injecting (HPCI) and low pressure core spraying (LPCS) available whereas low pressure core injecting (LPCI) failed. The TRACE code is used for the analysis. The Chinshan TRACE model was benchmarked against steady-state and transient data contained in the plant FSAR report, as well as start-up data and the transient results using the RETRAN code. The thermal hydraulic phenomena in the lower plenum area and the jet pumps are also analyzed.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
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