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International Agreement Report

Application of TRACE V5.0 P2 to China Domestic PWR LBLOCA Analysis

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ABSTRACT

The purpose of this work is to study the behavior of China domestic PWR under LBLOCA scenario using the TRACE (TRAC/RELAP Advanced Computational Engine) code. The work is divided into five parts:

The first part is TRACE model establishment. SNAP（Symbolic Nuclear Analysis Program） program was used to facilitate system modeling work. Important components such as active core, pressurizer, accumulator and steam generator were modeled respectively. These components were tested separately and results were compared with design data to check the accuracy. Key parameters were indentified and properly adjusted to refine the model further. All of the components were incorporated together to build up the integrated TRACE model of China domestic PWR.

The second part is steady state calculation. Steady state of full power operation was simulated by TRACE code and calculation results were compared with design data. Hydraulic frictions were adjusted to keep calculated and designed flow distribution as close as possible. The adjustment work was iterated until all of key parameters were acceptable.

The third part is transient calculation. The LBLOCA scenario was simulated in this part. Restart case of accident scenario was prepared based on the steady state TRACE model established previously. The transient calculation results showed that safety goal was achieved under the assumed accident scenario.

The forth part is sensitivity analysis. Sensitivity analysis of break spectrum and initial accumulator pressure was performed respectively. The most limiting break size and proper initial accumulator pressure were found though the sensitivity analysis.

The last part is accident scenario animation. SNAP was used to create the animation of the LBLOCA Accident scenario. Better understanding of the calculated physical phenomena and transient process was obtained via animation demonstration.

FOREWORD

TRACE is an advanced thermal hydraulic code developed by USNRC. It is a kind of best-estimate safety analysis code widely used in PWR, BWR and test grid thermal hydraulic analysis, such as LOCA, operation transient and other accident scenario. Traditional safety analysis codes, TRAC, RELAP and REMONA for example, are incorporated into TRACE code to build up a modern integrated safety analysis toolset. In the future, TRACE is going to play a very important role in design basis accident analysis and take the place of NRC traditional safety analysis codes mentioned above.

China and U.S. have signed an agreement on CAMP (Code Applications and Maintenance Program). Both sides are responsible for the development and maintenance of CAMP codes such as TRACE. The Nuclear and radiation Safety Center of China (NSC) is a governmental organization in China to provide technical support to the National Nuclear Safety Administration (NNSA). It is responsible for application of TRACE code in thermal hydraulic safety analysis. Users' experience and bugs found in code running should be carefully recorded and reported to USNRC according to CAMP framework. To meet this requirement, we built a TRACE model of China domestic PWR, performed steady state calculation, transient calculation and sensitivity analysis. Transient scenario was animated with SNAP code. Finally this report was prepared to share the code application experience with other CAMP members.

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EXECUTIVE SUMMARY

Couple of years ago, NNSA(China National Nuclear Safety Administration) and USNRC signed an agreement on CAMP（Code Applications and Maintenance Program）. NSC (Nuclear and radiation Safety Center of China) is a government organization to supply technical support to the NNSA(National Nuclear Safety Administration) in China. It has the responsibility to apply the TRACE code in thermal hydraulic safety analysis. Users' experience and bugs found in code running should be carefully recorded and reported to USNRC according to CAMP framework. To meet this requirement, we built a TRACE model of China domestic PWR, performed steady state calculation, transient calculation and sensitivity analysis. We animated the accident scenario and finally prepared this report to share the code application experience.

The simulated China domestic PWR is a 2-loop pressurized water reactor designed by a domestic institution of China. The plant site is located near Shanghai, east coast of China shown in Fig. 1. The reactor rated thermal power is 1930 MW. AFA 3G fuel assembly is used in the core. And fuel cycle length is 18 months. Analysis results show that peak clad temperature is below the safety limit and other key parameter trends are reasonable during LBLOCA simulation.

ABBREVIATIONS

CAMP	Code Applications and Maintenance Program
NSC	Nuclear and radiation Safety Center of China
NNSA	China National Nuclear Safety Administration
PWR	Pressurized Water Reactor
LPI	Low Pressure Injection
HPI	High Pressure Injection
LBLOCA	Large Break Loss Of Coolant Accidents
NRC	Nuclear Regulatory Commission
SG	Steam Generator
SNAP	Symbolic Nuclear Analysis Program
TRACE	TRAC/RELAP Advanced Computational Engine
US	United States

1. INTRODUCTION

TRACE is an advanced thermal hydraulic code developed by USNRC. It is a kind of best-estimate safety analysis code widely used in PWR, BWR and test grid thermal hydraulic analysis, such as LOCA, operation transient and other accident scenario. Traditional safety analysis codes, TRAC, RELAP, and REMONA for example, are incorporated into TRACE code to build up a modern integrated safety analysis toolset. In the future, TRACE will play a very important role in design basis accident analysis and take the place of NRC traditional safety analysis codes mentioned above.

China and U.S. have signed an agreement on CAMP (Code Applications and Maintenance Program). Both sides are responsible for the development and maintenance of CAMP codes such as TRACE. The Nuclear and radiation Safety Center of China (NSC) is a governmental organization in China to provide technical support to the National Nuclear Safety Administration (NNSA). It is responsible for application of the TRACE code in thermal hydraulic safety analysis. Users' experience and bugs found in code running should be carefully recorded and reported to USNRC according to CAMP framework. To meet this requirement, we built a TRACE model of China domestic PWR, performed steady state calculation, transient calculation and sensitivity analysis. We animated the accident scenario and finally prepared this report to share the code application experience.

Fig. 1 The China Domestic PWR Site

2. METHODOLOGY

SNAP v 2.1.0 and TRACE v 5.0p2 were used in this work. The methodology of the research is as following:

The first step is TRACE model establishment. SNAP（Symbolic Nuclear Analysis Program） program was used to facilitate system modeling work. Important components such as vessel, cold leg, pressurizer, accumulator and steam generator were modeled respectively. These components were tested separately and compared with design data to check the accuracy. Key parameters were indentified and properly adjusted to refine the model further. All of the components were incorporated together and finally formed the integrated TRACE model.

The second step is steady state calculation. Steady state of full power operation was simulated by TRACE code and calculation results were compared with design data. Hydraulic frictions were adjusted to keep calculated and designed flow distribution as close as possible. Iterated adjustment was performed until all of key parameters were acceptable.

The third step is transient calculation. The accident scenario “Large Break LOCA” was simulated in this part. Restart model of accident scenario was prepared based on the steady state TRACE model established in the previous parts. The transient calculation results showed that safety goal was achieved under the assumed accident scenario.

The forth step is sensitivity analysis. Sensitivity analysis of initial accumulator pressure and cold leg break size was performed respectively.

The last step is accident scenario animation. SNAP was used to create the animation of the “Large Break LOCA Accident scenario”. Better understanding of the calculated physical phenomena and transient process was obtained via animation demonstration.

The complete process is presented in Fig. 2.

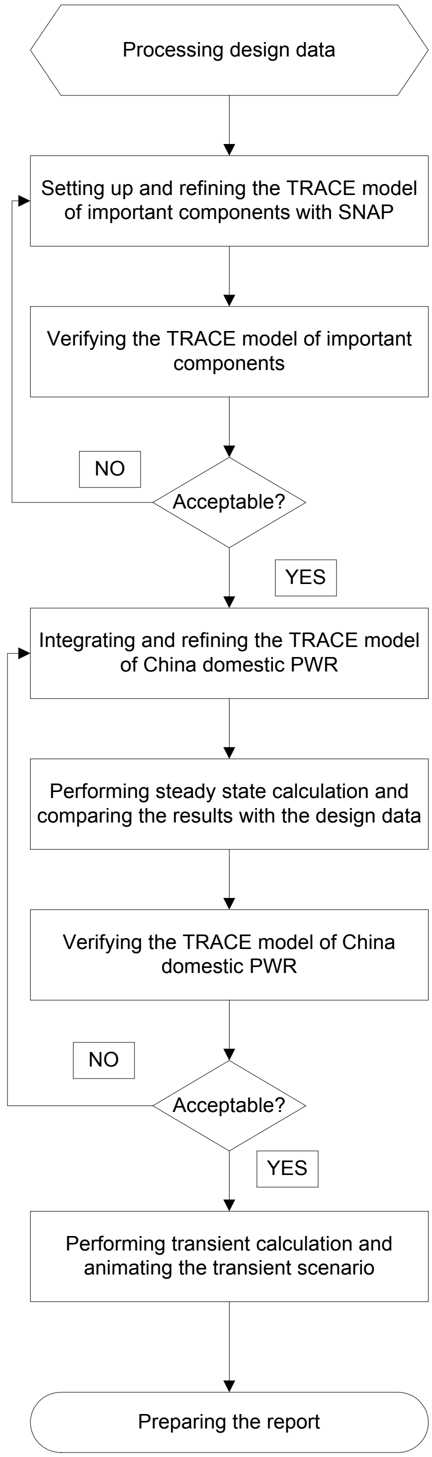

Fig. 2 The Flow Chart of TRACE Model Establishment and Verification

3. ESTABLISHMENT AND VERIFICATION OF CHINA DOMESTIC PWR TRACE MODEL

3.1 General Description

The simulated China domestic PWR is a 2-loop pressurized water reactor designed by a domestic institution of China. The plant site is located near Shanghai, east coast of China shown in Fig. 1. The reactor rated thermal power is 1930 MW. AFA 3G fuel assembly is used in the core. And fuel cycle length is 18 months. Important components such as core, accumulator, hot leg and SG were modeled respectively. These components were tested separately and compared with design data to check the accuracy. Key parameters were indentified and properly adjusted to refine the model further. After refining, all of the components were incorporated together to build up the integrated TRACE model of primary and secondary system.

3.2 Reactor Vessel Component

The height of active core is 3.658m. It consists of 121 fuel assemblies and each assembly is made up of 264 fuel rods. The arrangement of fuel rods in an assembly is 17 X 17, in which 24 rod positions are occupied by control rod guide tubes and 1 rod position is measurement tube. The size of fuel assembly is 21.4cm X 21.4cm. The pitch of fuel rods is 12.6mm. Outer diameter of fuel rod is 9.5mm. Clad thickness is 0.57mm. The diameter of UO_2 pellet is 8.192mm. The designed inlet coolant temperature is 292.8℃ and outlet coolant temperature is 327.2℃.

Pipe component was used to build up reactor vessel, include active core, bypass, downcomer, upper plenum and lower plenum. The heat structure associated with core was used to simulate the average fuel rod and heat transferring from fuel to coolant. The core component and associated heat structure is shown in Fig. 3. The fill and break components were used to model core inlet and outlet boundary conditions. The reactor vessel component was tested separately and the difference between calculated results and designed data were checked to confirm the accuracy of modeling.

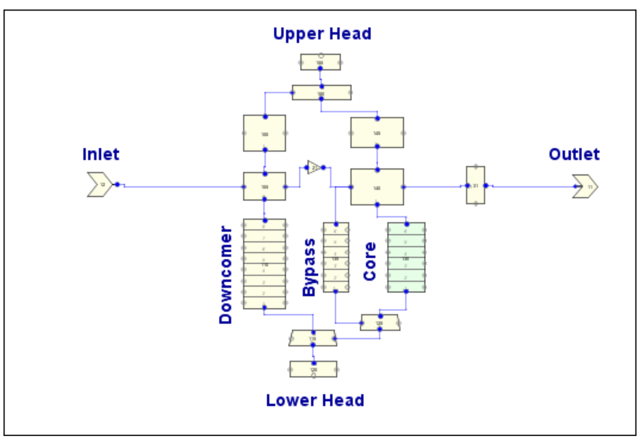

Fig. 3 The TRACE Model of the Reactor Vessel Component for China Domestic PWR

3.3 <u>Power Component</u>

Power component was used to generate power in heat structure standing for fuel rods. The rated thermal power is 1930 MW. Chopped COSINE distribution was assumed for core axial power and flat power distribution was assumed for fuel rod radial power. ANS-94 was used to generate decay heat. In the steady state, initial power of 1930MW was maintained to perform steady state calculation. Afterwards, "Large Break LOCA" accident was initiated and reactor scrammed due to negative void reactivity. After reactor scram, decay heat dominated the heat generating of core. The power component is shown in Fig. 4.

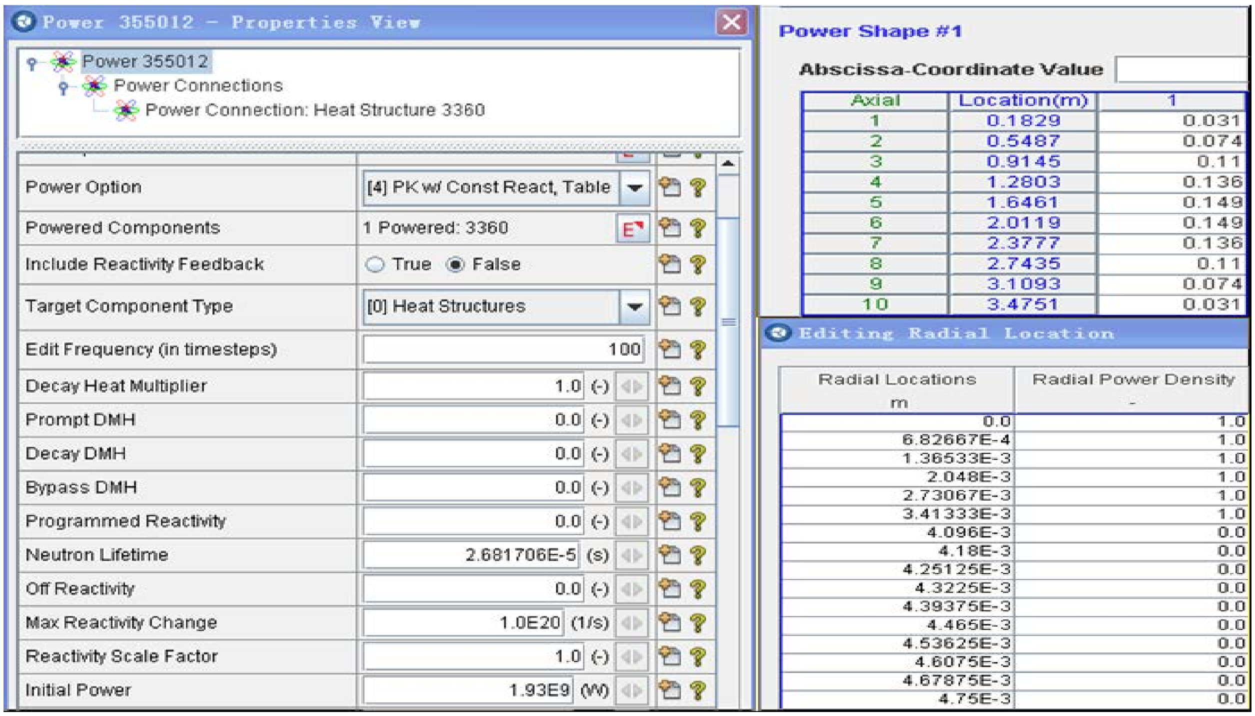

Fig. 4 The TRACE Model of the Power Component for China Domestic PWR

3.4 <u>Primary Loop Components</u>

Primary loop components mainly consist of cold leg pipe, hot leg pipe, pressurizer, accumulators, LPI, HPI, SG tubes and coolant pumps shown in Fig. 5.

The SG tubes were modeled with pipe component with 8 cells and heat structure was associated to simulate heat transferring from primary loop to secondary loop. The pressurizer was modeled with prizer component to stabilize system pressure and the pressure setpoint is 15.5 MPa. The accumulators were modeled with pipe component to supply emergency water injection during the refill stage of LBLOCA and actuating setpoint is 4.235 Mpa. The HPI and LPI were modeled with fill component to supply safety water injection during reflood stage of LBLOCA and the mass flow depends on injection point pressure.

Break was modeled by break and valve component connected to broken loop cold leg. Discharge coefficient of 0.7 was selected according to final safety analysis report.

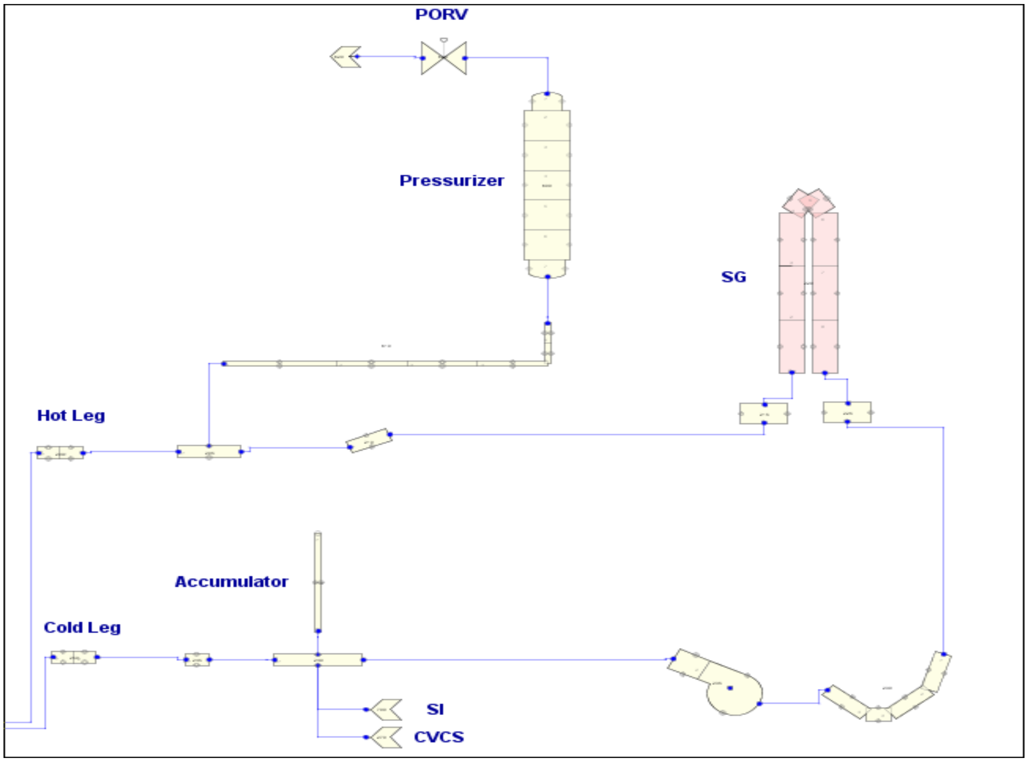

Fig. 5 The TRACE Model of the Primary Loop Components for China Domestic PWR

3.5 Secondary Loop Components

Secondary Loop Components mainly consist of SG secondary side, main feed water, auxiliary feed water, steam line and related valves.

SG secondary side mainly includes downcomer, boiler, separator and steam dome. Main feed water and auxiliary feed water were modeled with fill component connected to a one-cell pipe, which was on top of downcomer. Downcomer was modeled with pipe component with 4 cells. Feed water flows downwards through the downcomer and goes into boiler at the bottom cell. Heat is transferred from SG primary side to the secondary side through SG tube wall and steam is generated in the boiler. The mixture of steam and water rises to the separator to be separated. Separator was modeled with ideal separator component with 1 cell. Dry steam continues to move up to the steam dome and separated water goes down to downcomer again. Proper form loss coefficient must be included to keep specific circulation rate. On top of steam dome, there is a built-in steam flow limiter. Flow area was decreased to around 0.1 m^2 to limit steam flow during main steam line break accident. Steam line was modeled with pipe component and isolation valves, bypass valves and safety relief valves were connected to related steam lines.

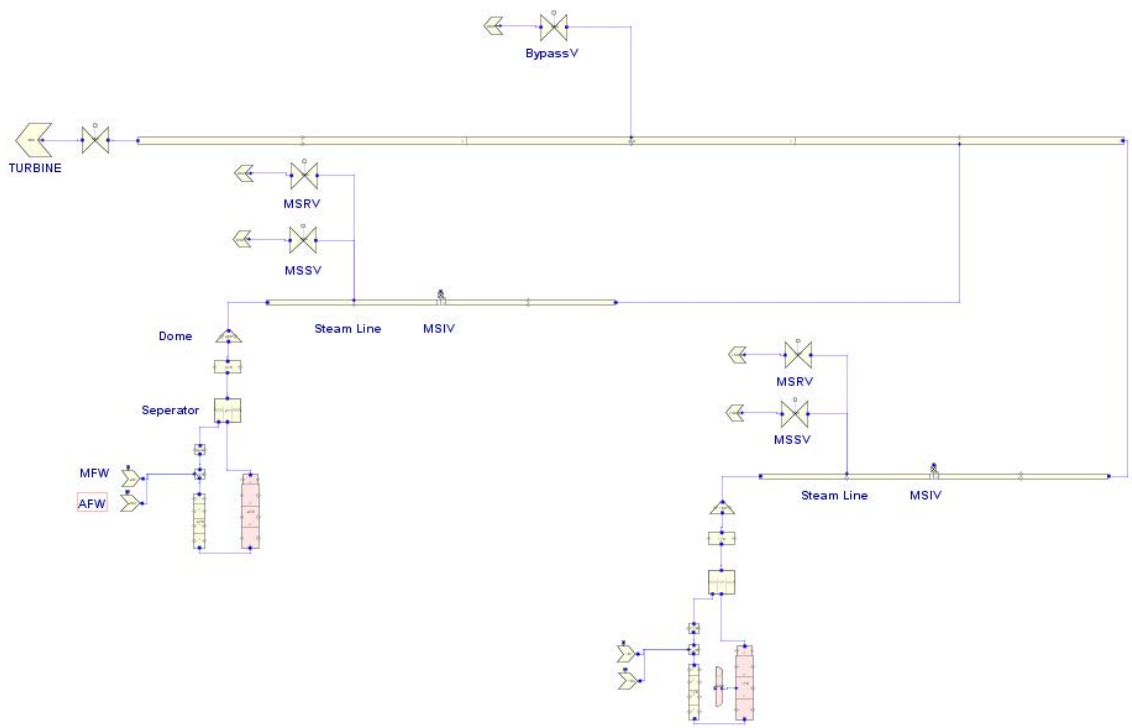

Fig. 6 The TRACE Model of the Secondary Loop Components for China Domestic PWR

4. STEADY STATE CALCULATION

4.1 Model Description

There are 2 loops for the simulated China domestic PWR as shown in Fig. 7. Pressurizer was connected to the intact loop to delay the safety injection signal which was activated by low low pressurizer pressure. While in steady state calculation, there will not be any safety injection signal, which will appear in next transient calculation step. In designed steady state, reactor power is 1930MW, pressurizer pressure is 15.5 MPa, primary side mass flow is 4796 kg/s per loop, bypass flow is 6.5%, main steam pressure is 6.71 MPa, main steam quality is 99.75%, steam mass flow is 541.9 kg/s per SG, circulation rate is 3.4%.

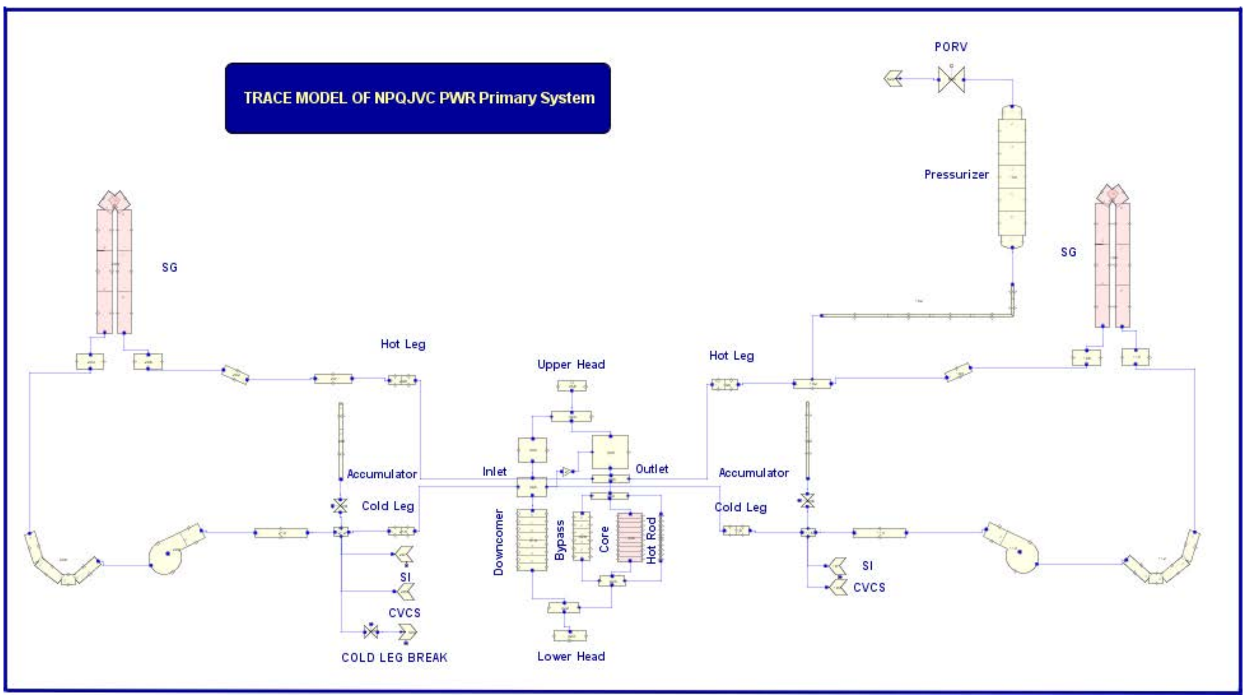

Fig. 7 The TRACE Model of China Domestic PWR

4.2 Steady State Calculation Results

Before transient calculation, steady state calculation must be performed first to make sure each parameter of the system is close enough to the design value. Frictions must be finely adjusted to get the right coolant mass flow distribution. After model refining, key parameters of the system have acceptable difference from the design data. Steady state calculation results are shown in Fig.8 to 14.

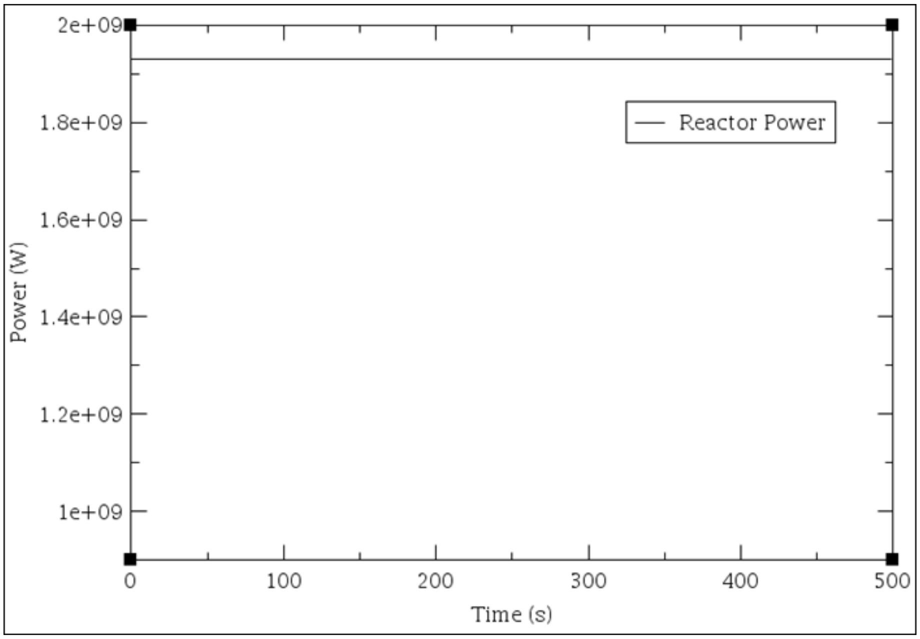

Fig. 8 Reactor Power

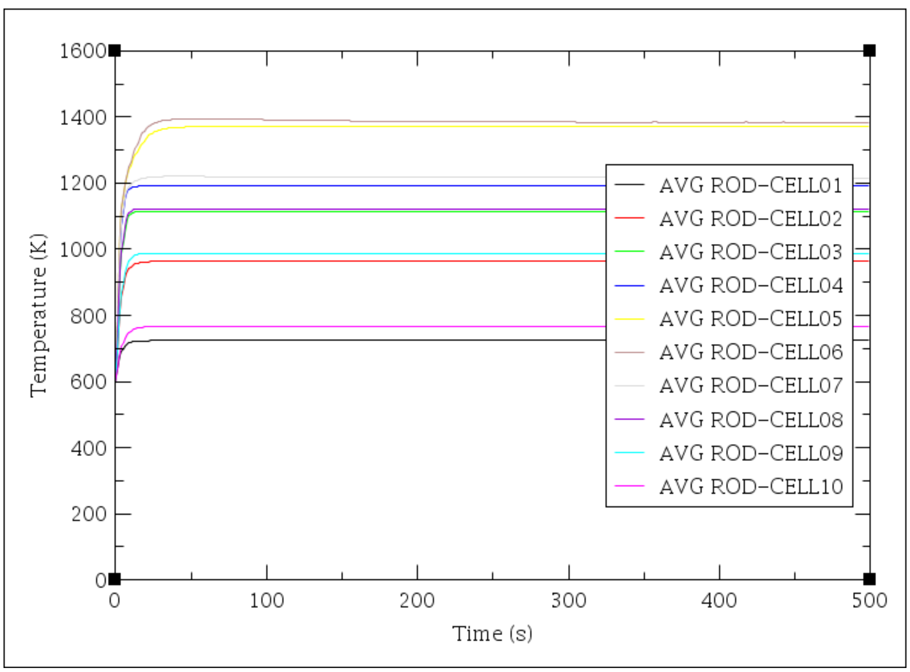

Fig. 9 Average Channel Rod Temperature

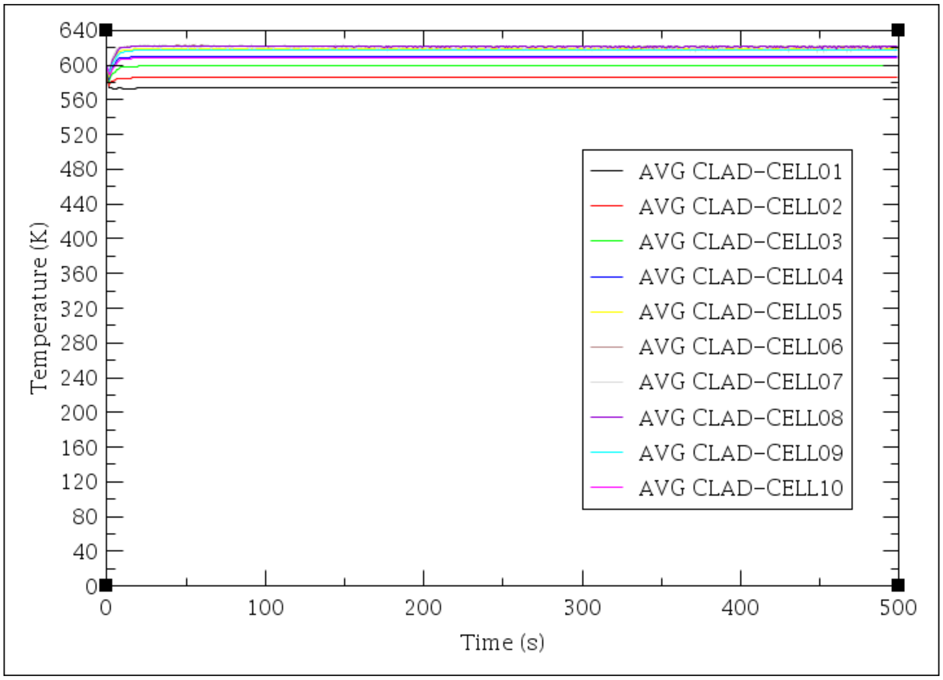

Fig. 10　Average Channel Clad Temperature

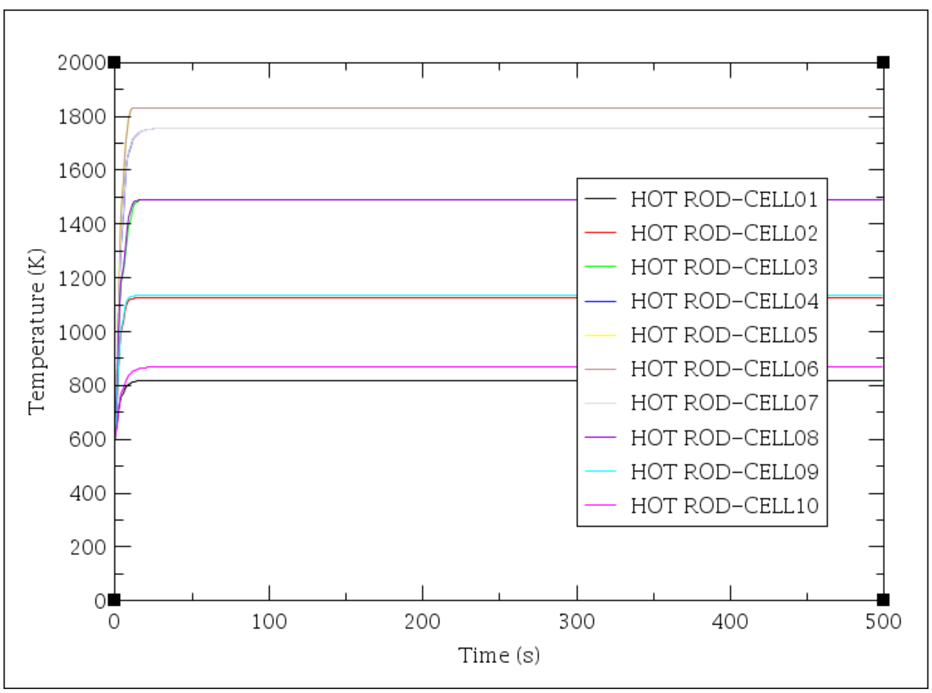

Fig. 11　Hot Channel Rod Temperature

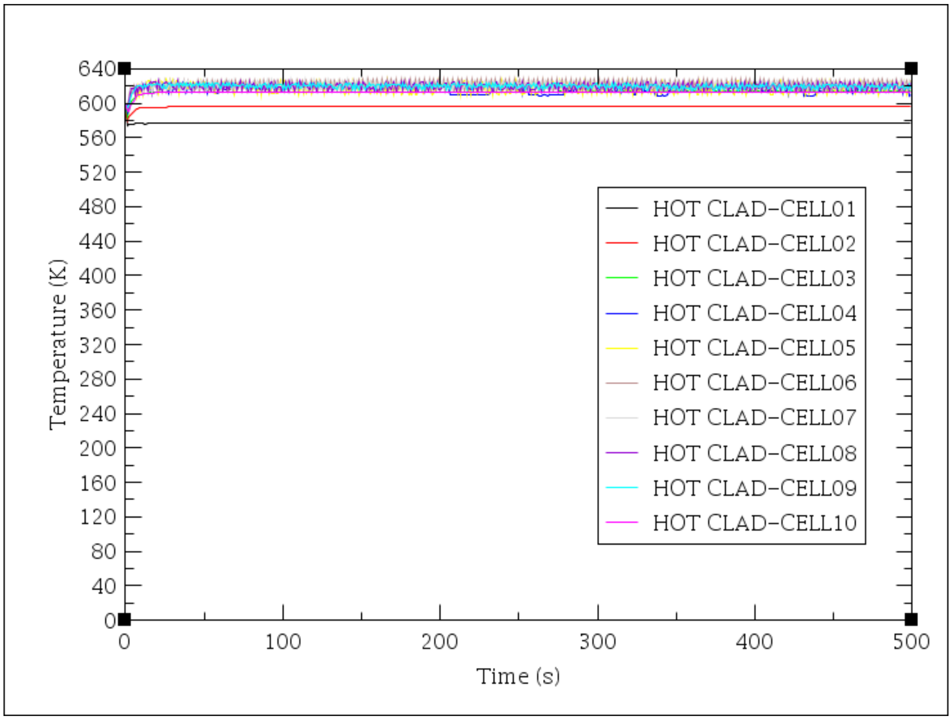

Fig. 12 Hot Channel Clad Temperature

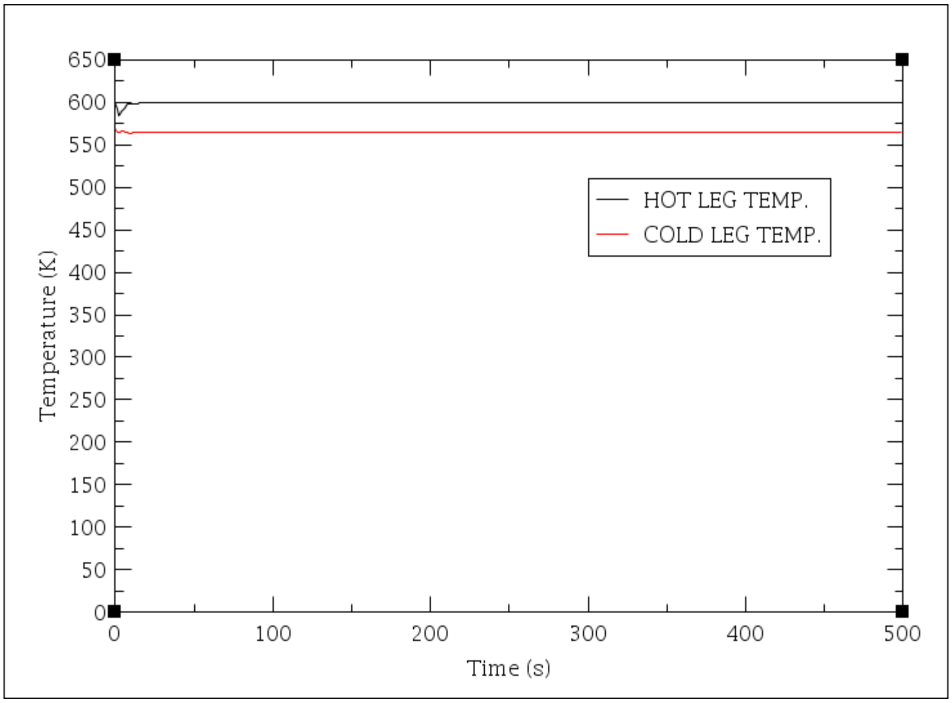

Fig. 13 Hot Leg and Cold Leg Temperature

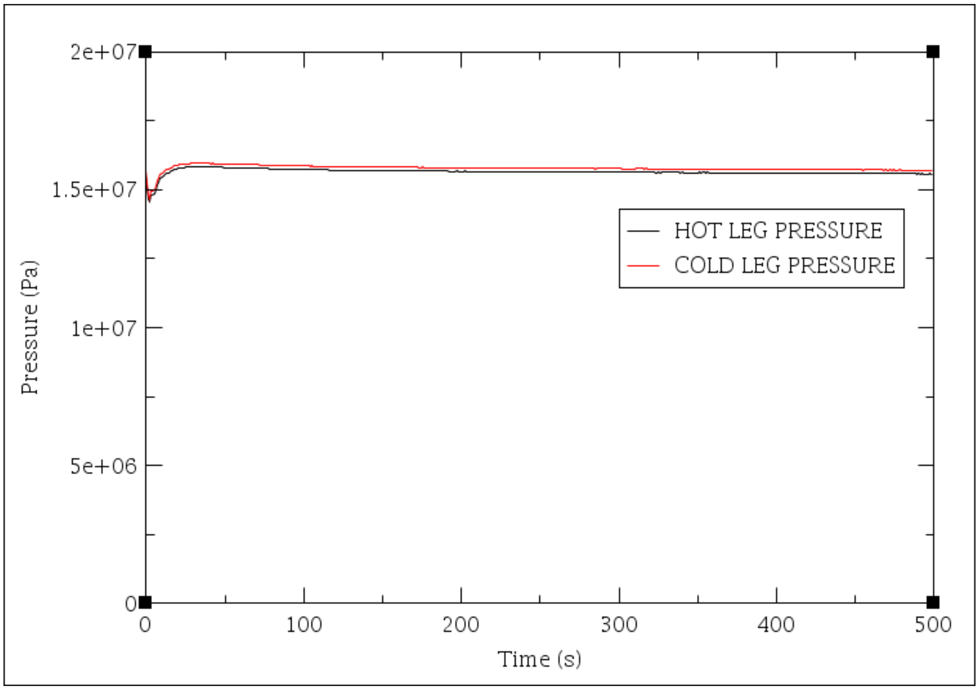

Fig. 14 Hot Leg and Cold Leg Pressure

5. TRANSIENT CALCULATION

5.1 Transient Calculation Results

The “Large Break LOCA” accident scenario was simulated in this section. LBLOCA assumption is as following: initial reactor power is 100%FP, axial power distribution is chopped cosine, Fq is 2.35, cold leg break size is 0.7 X 2A, coolant pumps coast down after break initiation, trip signal is initiated by low pressurizer pressure, safety injection signal is initiated by low low pressurizer pressure, main feed water isolation is delayed by 7s, safety injection signal is delayed by 30s, auxiliary feed water startup is delayed by 62s. Realistic model plus conservative assumption method is used in this simulation. Normally initial reactor power of 102%FP is assumed in final safety analysis report, but in this report we assumed initial reactor power was 100%FP, so this LBLOCA simulation result is more optimistic. Transient calculation results are shown in Fig.15 to 22.

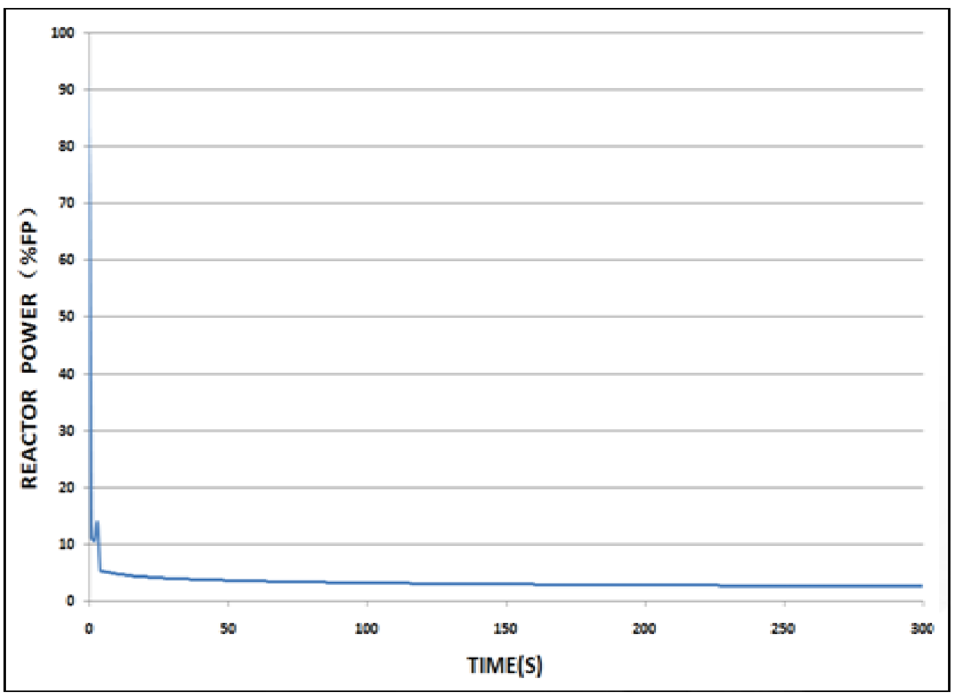

Fig. 15 Reactor Power (Transient)

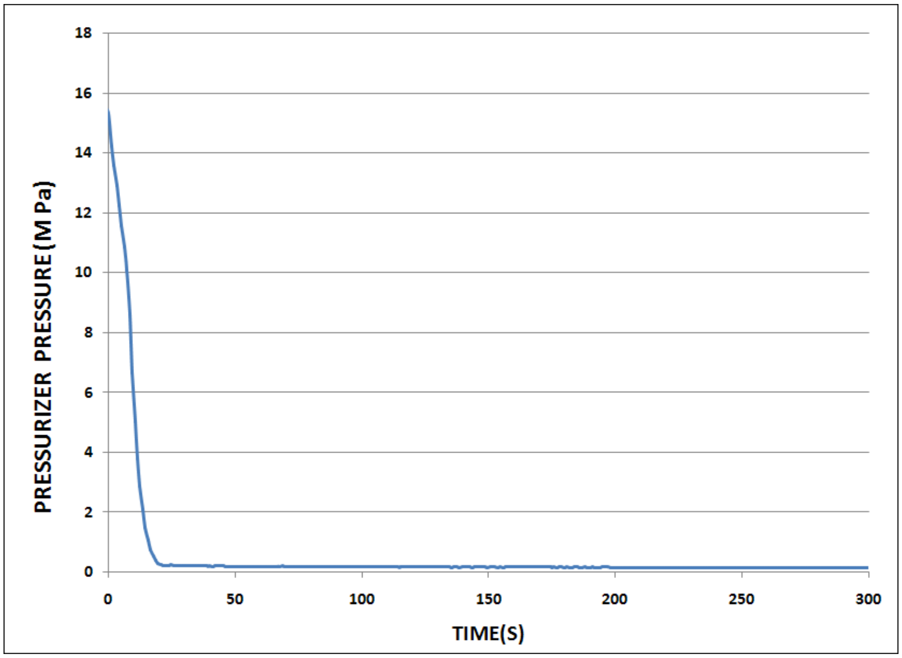

Fig. 16 Pressurizer Pressure (Transient)

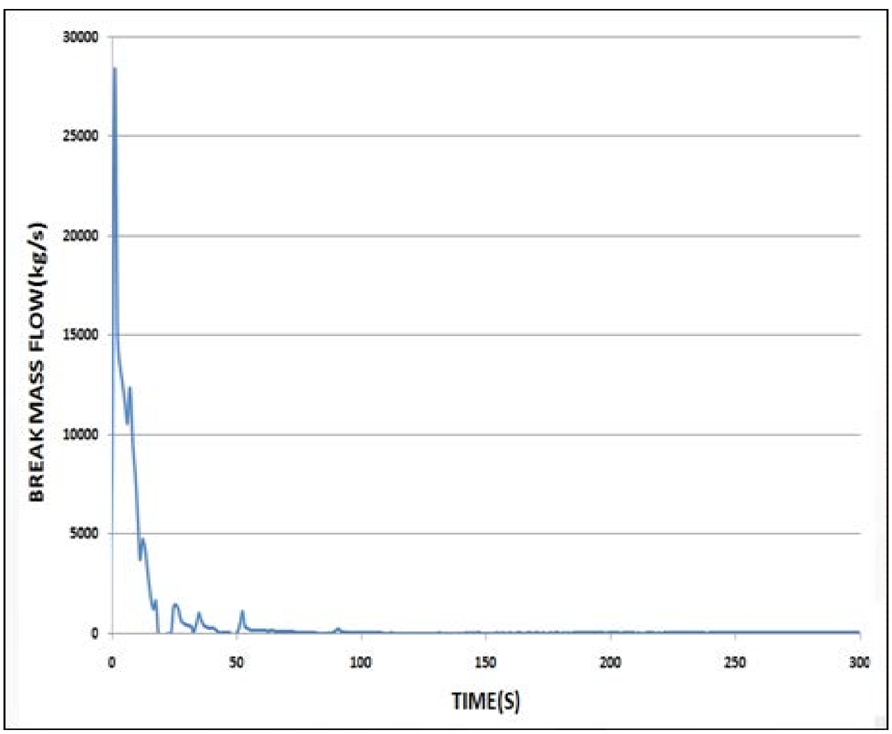

Fig. 17 Break Flow (Transient)

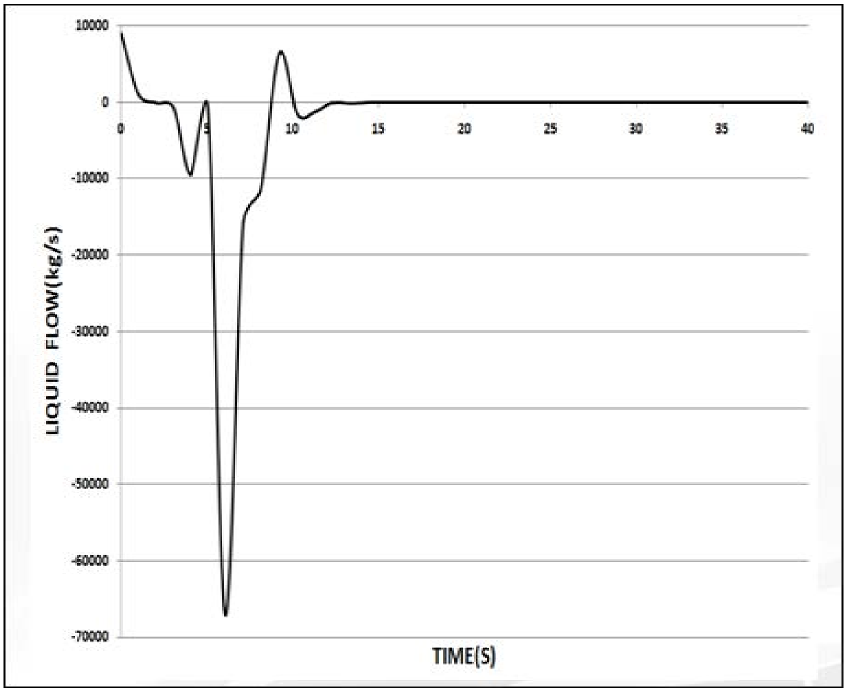

Fig. 18 Core Flow (Transient)

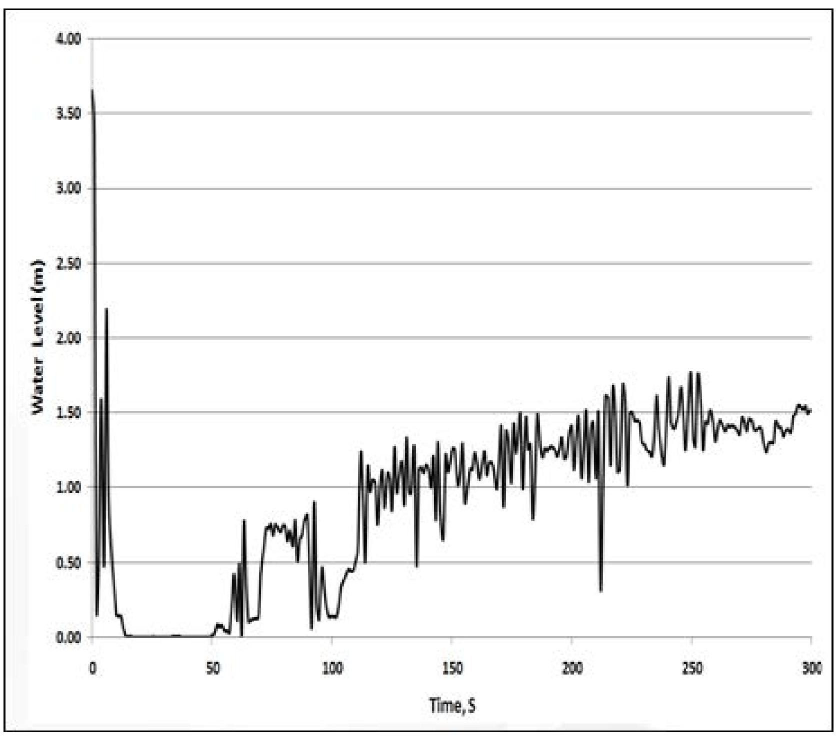

Fig. 19 Core Collapsed Water Level

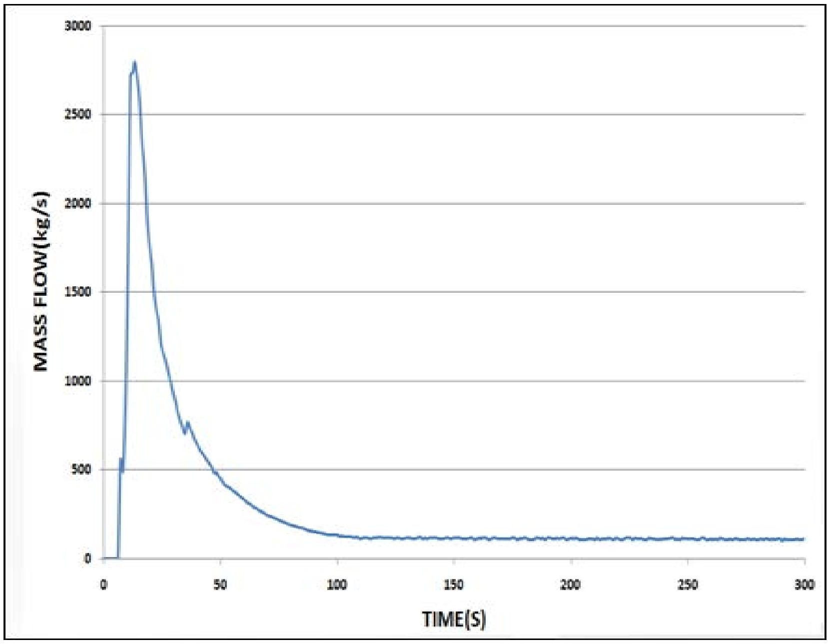

Fig. 20 Safety Injection Flow

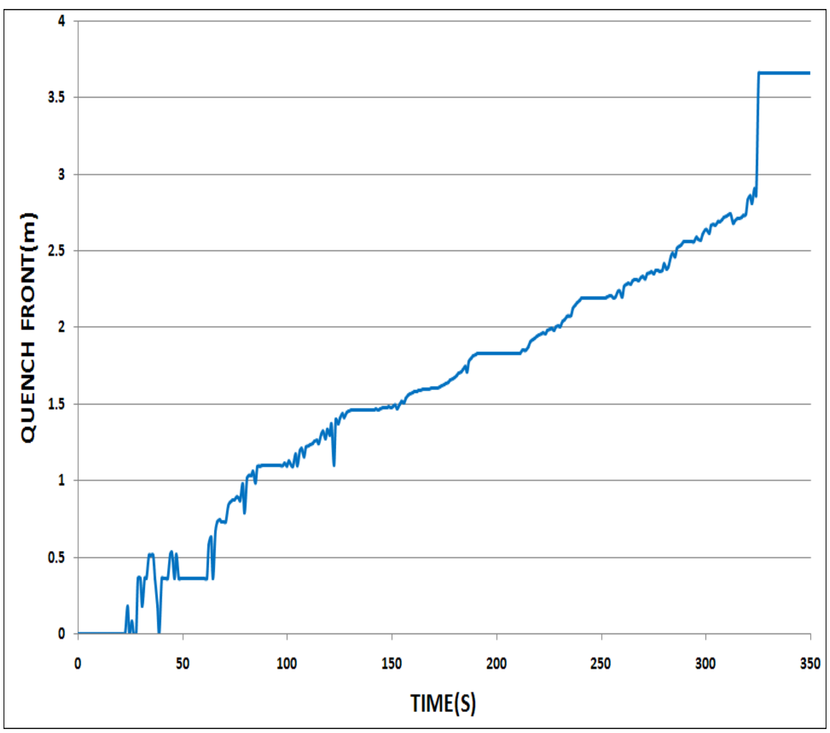

Fig. 21 Quench Front

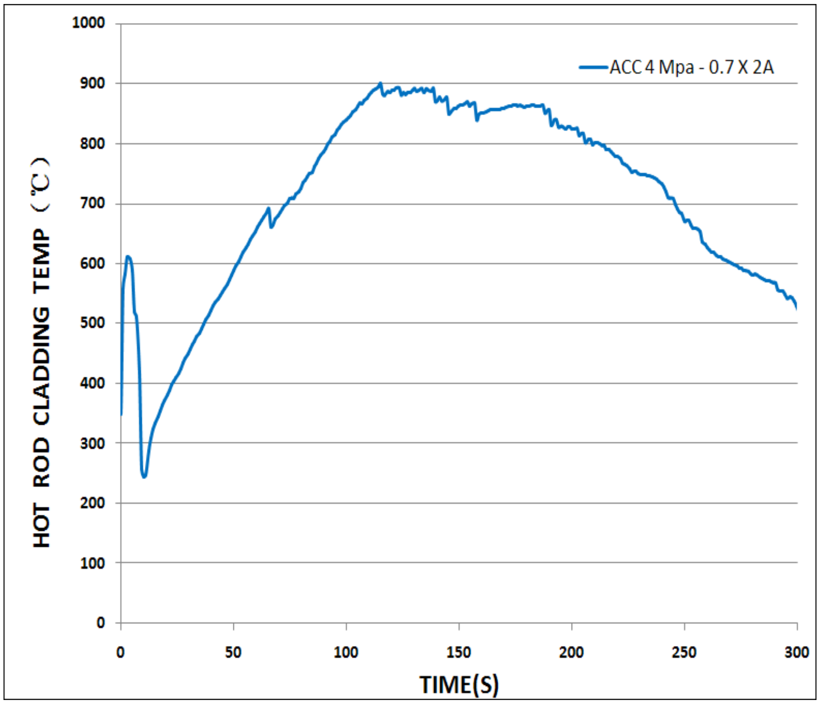

Fig. 22 Hot Rod Clad Temperature

5.2 Sensitivity Analysis

Lots of studies show guillotine break of cold leg is not the most limiting case in large break LOCA accident. Break size must be searched to determine which one is the most limiting case. This is so called break spectrum analysis. In this report, break size of 0.7 X 2A, 0.6 X 2A, 0.5 X 2A were analyzed, sensitivity analysis result shows that break size of 0.6 X 2A is the most limiting case in our simulation. Break spectrum analysis results are shown in Fig.23.

Different initial accumulator pressure will also affect peak clad temperature during large break LOCA accident. If initial accumulator pressure is too high, more accumulator injection mass flow will bypass from the downcomer to the cold leg break, fewer water will be left to inject into the active core, the core cooling condition will be worse. If initial accumulator pressure is too low, high pressure steam out of the core can easily prevent accumulator injection mass flow penetrating downcomer to enter active core. So it is necessary to perform initial accumulator pressure sensitivity analysis to search proper pressure actuating setpoint for accumulators. In this report, initial accumulator pressures of 2.0 MPa, 2.5 MPa, 3.0MPa, 4.0MPa were analyzed separately. Simulation result shows initial accumulator pressures of 2.5 MPa will significantly decrease peak clad temperature during large break LOCA scenario compared to other pressure setpoints. Analysis results are shown in Fig.24.

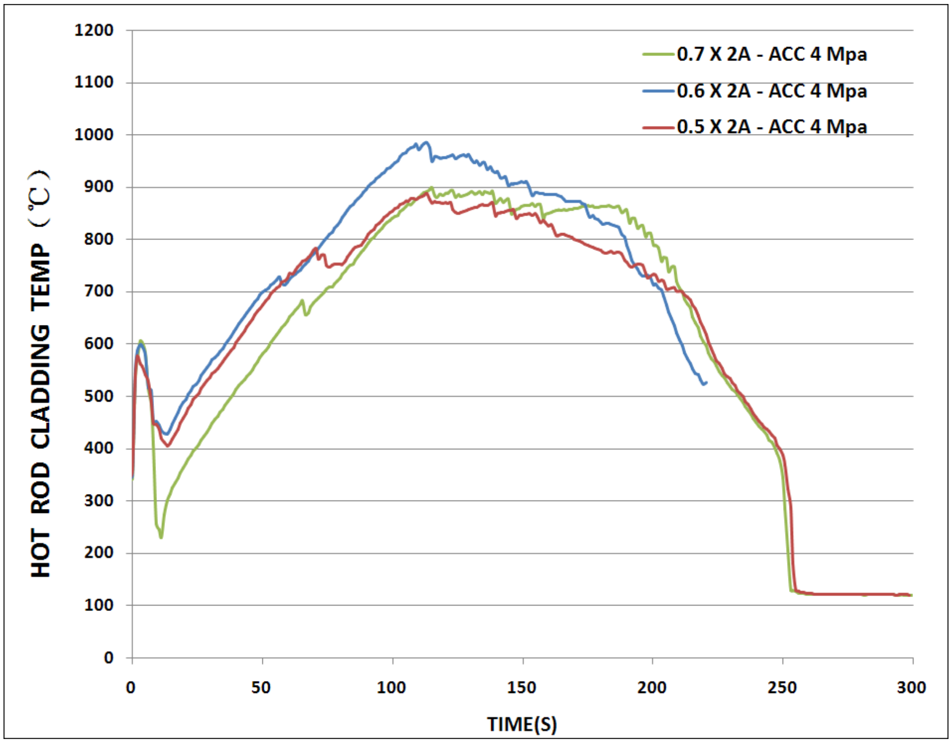

Fig. 23 Break Spectrum Analysis

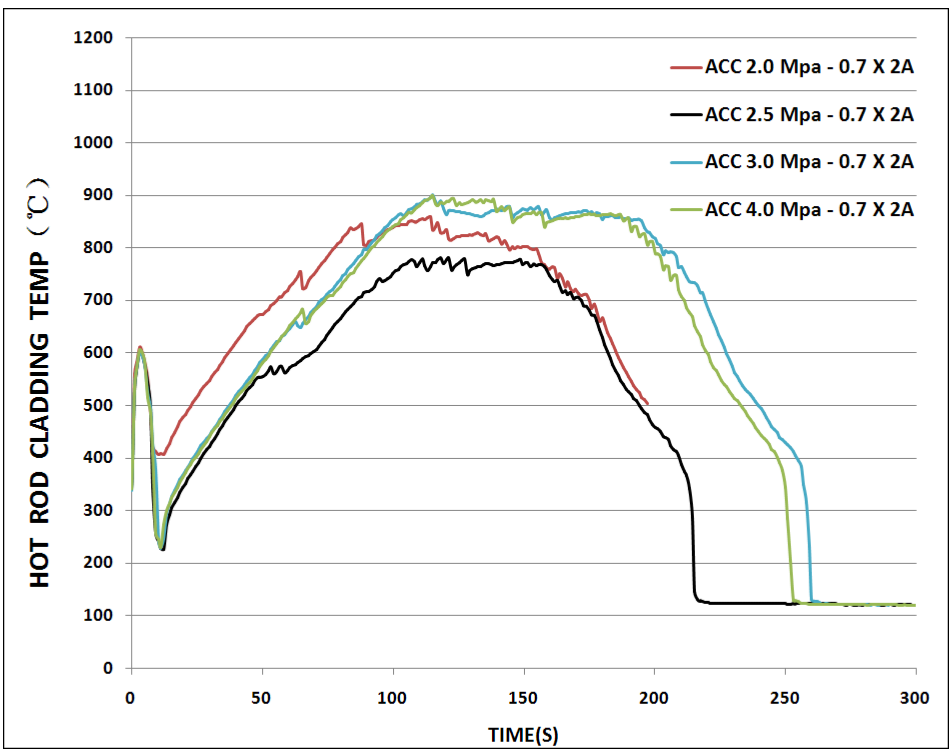

Fig. 24 Initial Accumulator Pressure Sensitivity Analysis

6. TRANSIENT ANIMATION

The “Large Break LOCA” accident scenario was animated in this section. SNAP 2.1.0 was used to animate the accident scenario. Fluid conditions and key parameters such as temperature and mass flow direction can be observed clearly for each part of system during the accident progress. Better understanding of the calculated physical phenomena and transient process was obtained via animation demonstration.

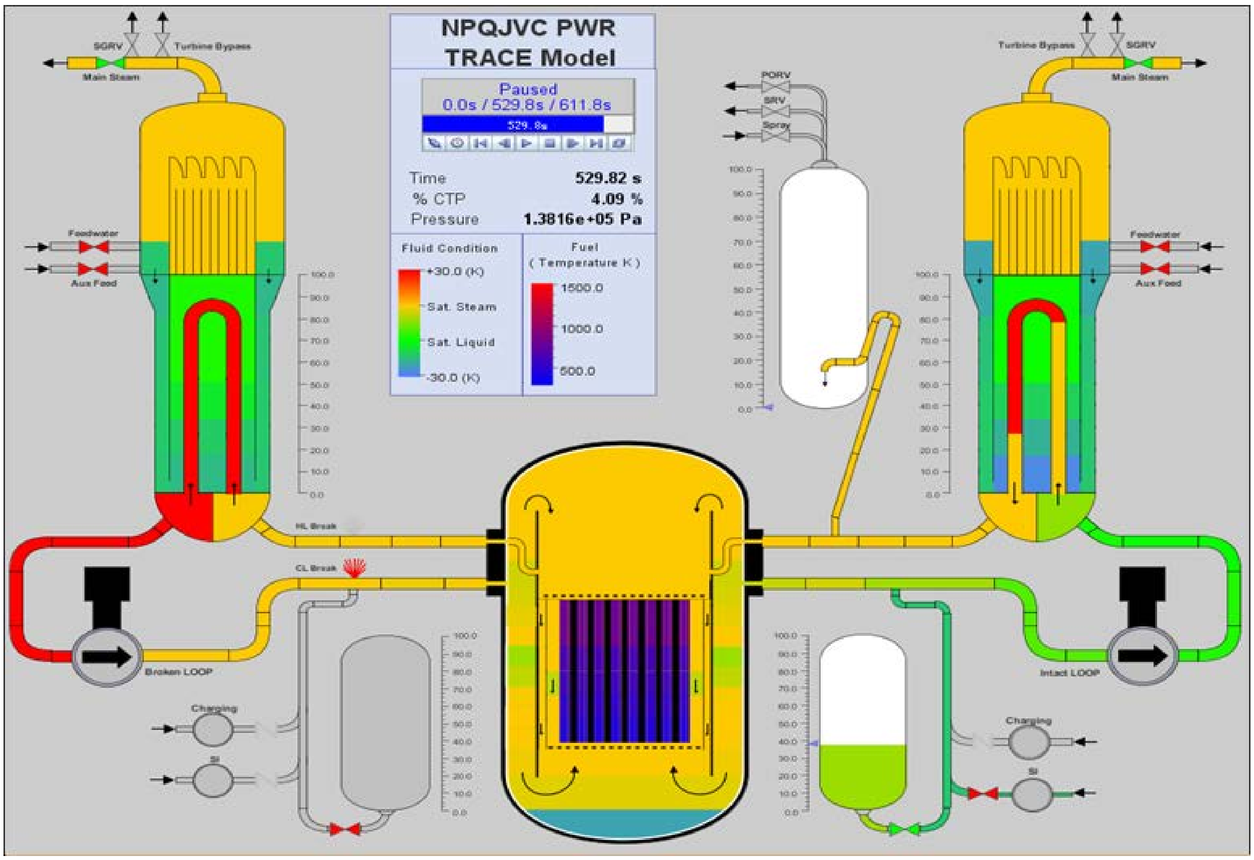

Fig. 25 Transient Scenario Animation

7. CONCLUSIONS

In this report, China domestic PWR was successfully modeled by SNAP and TRACE code. The model was mainly made up of pipe, fill, break, prizer, separator, valve and power component. Key parameters of each component were finely modeled to match the design data. Steady state calculation was performed with the integrated TRACE model. Errors between steady state calculation results and design data were negligible. Restart file was generated by steady state calculation. Based on the steady state restart file, transient case was introduced and transient calculation results were generated consequently. After transient calculation, animation work was done with SNAP code. Fluid conditions and key parameters could be observed simultaneously as the transient scenario went on. The animation function gave more interactive information and better understanding of accident progress and physical phenomena to analyst. This work indicates that SNAP/TRACE code works well to simulate China domestic PWR and the calculation results are reasonable.

In China, NSC has the responsibility to apply TRACE code in thermal hydraulic analysis. The TRACE application work carried out by NSC is still on the beginning. The work in this report is very preliminary, but we are always ready to report our work to NRC and share our experience of TRACE code application with other TRACE users according to CAMP agreement. NSC has made a plan to model AP1000 with TRACE code next step. The future work will be also reported to NRC after completion and we hope it more contributable.

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11. ABSTRACT *(200 words or less)*

The purpose of this work is to study the behavior of China domestic PWR under LBLOCA scenario using the TRACE (TRAC/RELAP Advanced Computational Engine) code. The work is divided into five parts:

The first part is TRACE model establishment. SNAP (Symbolic Nuclear Analysis Program) program was used to facilitate system modeling work. Important components such as active core, pressurizer, accumulator and steam generator were modeled respectively. These components were tested separately and results were compared with design data to check the accuracy. Key parameters were indentified and properly adjusted to refine the model further. All of the components were incorporated together to build up the integrated TRACE model of China domestic PWR.

The second part is steady state calculation. Steady state of full power operation was simulated by TRACE code and calculation results were compared with design data. Hydraulic frictions were adjusted to keep calculated and designed flow distribution as close as possible. The adjustment work was iterated until all of key parameters were acceptable.

The third part is transient calculation. The LBLOCA scenario was simulated in this part. Restart case of accident scenario was prepared based on the steady state TRACE model established previously.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*
TRACE (TRAC/RELAP Advanced Computational Engine)
CAMP (Code Applications and Maintenance Program)
SNAP (Symbolic Nuclear Analysis Program)
Nuclear and radiation Safety Center of China (NSC)
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