NUREG/IA-0418

International Agreement Report

Application of TRACE V5.0 P2 to Natural Circulation Reactor Safety Analysis

Prepared by:
FENG Jinjun, CHAI Guohan, SHI Junying, ZHOU Kefeng

Nuclear and Radiation Safety Centre, Ministry of Environmental Protection
54, Honglian Nancun, Haidian District, Beijing, 100082, CHINA

A. Calvo, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: February 2012
Date Published: October 2012

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

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:
Address: U.S. Nuclear Regulatory Commission
Office of Administration
Publications Branch
Washington, DC 20555-0001
E-mail: DISTRIBUTION.RESOURCE@NRC.GOV
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—
The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—
American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG-XXXX) or agency contractors (NUREG/CR-XXXX), (2) proceedings of conferences (NUREG/CP-XXXX), (3) reports resulting from international agreements (NUREG/IA-XXXX), (4) brochures (NUREG/BR-XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG-0750).

NUREG/IA-0418

International Agreement Report

Application of TRACE V5.0 P2 to Natural Circulation Reactor Safety Analysis

Prepared by:
FENG Jinjun, CHAI Guohan, SHI Junying, ZHOU Kefeng

Nuclear and Radiation Safety Centre, Ministry of Environmental Protection
54, Honglian Nancun, Haidian District, Beijing, 100082, CHINA

A. Calvo, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: February 2012
Date Published: October 2012

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

ABSTRACT

The purpose of this work is to study the behavior of natural circulation reactor under Accident 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 core and heat exchanger 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 natural circulation reactor.

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 primary side natural circulation mass flow as close as possible. The adjustment work was iterated until all of key parameters were acceptable.

The third part is transient calculation. The Accident scenario “Loss of Main Feed Water” 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 main feed water flow rate and reactor power to primary side natural circulation flow was performed respectively. Special phenomenon of natural circulation flow acceleration due to “loss of feed water” was understood better though the sensitivity analysis.

The last part is accident scenario animation. SNAP was used to create the animation of the “Loss of Main Feed Water” 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 Natural Circulation Reactor, 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.

CONTENTS

FIGURES

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 Natural Circulation Reactor, 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.

Natural Circulation Reactor is a kind of concept design developed by domestic institution in China. Most of the important components are submerged in a twenty-meter deep water pool shown in Fig. 1. The whole system is depressurized. The reactor core is placed at the bottom of the pool and the rated thermal power is 120 MW. Fuel rods are cooled by natural circulation flow. The designed core outlet temperature is 110℃. Hot water out of the core rises upwards to a heat exchanger and transfer heat to secondary side feed water. Primary coolant is cooled down and flows out of heat exchanger to the downcomer, and finally heads to the inlet of the core to complete the whole natural circulation.

The specific natural circulation reactor analyzed in this report has no prototype yet, so no experiment data could be collected to do comparison between calculated results and measured data. In this report, only theoretical analysis was performed and the rationality of calculation results was judged by expert experience. This work indicates that the TRACE code is certainly capable to perform thermal hydraulic analysis on Natural Circulation Reactor and generates reasonable results.

ABBREVIATIONS

CAMP	Code Applications and Maintenance Program
NSC	Nuclear and radiation Safety Center of China
NNSA	China National Nuclear Safety Administration
NCR	Natural Circulation Reactor
LOCA	Loss Of Coolant Accidents
NRC	Nuclear Regulatory Commission
HX	Heat Exchanger
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 Natural Circulation Reactor, 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.

As the mentioned natural circulation reactor is still on design paper on the time being, no real data could be collected to do comparison between calculated and measured data. In this report, only theoretical analysis was performed and the rationality of calculation results was judged by expert experience. Unique phenomenon of natural circulation coolant flow acceleration due to "loss of feed water" was observed in transient analysis. Analysis results indicate that the design is able to keep the reactor safe under assumed accident scenario.

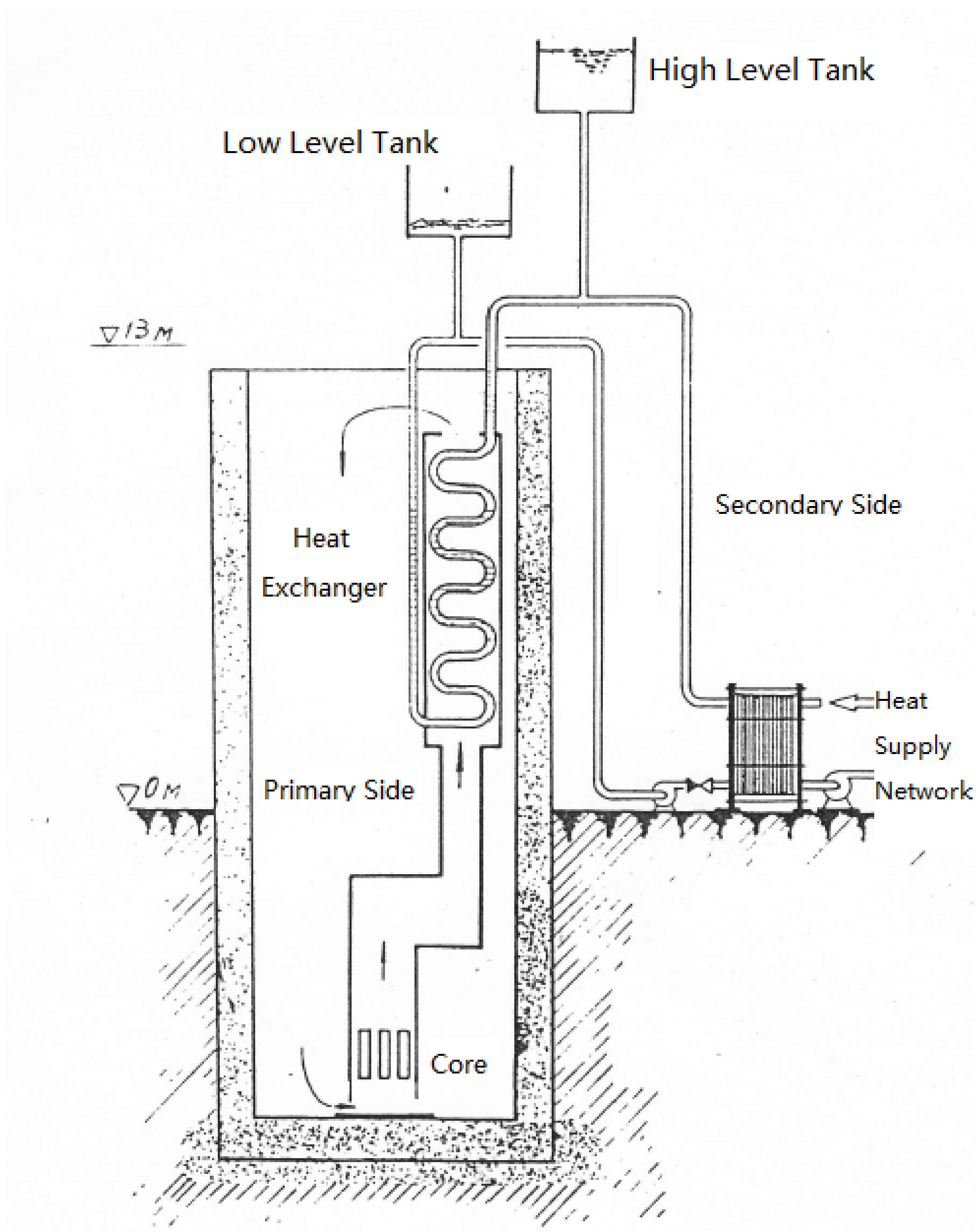


Fig. 1 The Schematic Diagram of Natural Circulation Reactor

2. METHODOLOGY

SNAP v 2.0.8 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 core and heat exchanger 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 primary side natural circulation mass flow as close as possible. Iterated adjustment was performed until all of key parameters were acceptable.

The third step is transient calculation. The accident scenario “Loss of Main Feed Water” 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 main feed water and reactor power to primary side natural circulation flow was performed respectively.

The last step is accident scenario animation. SNAP was used to create the animation of the “Loss of Main Feed Water 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.

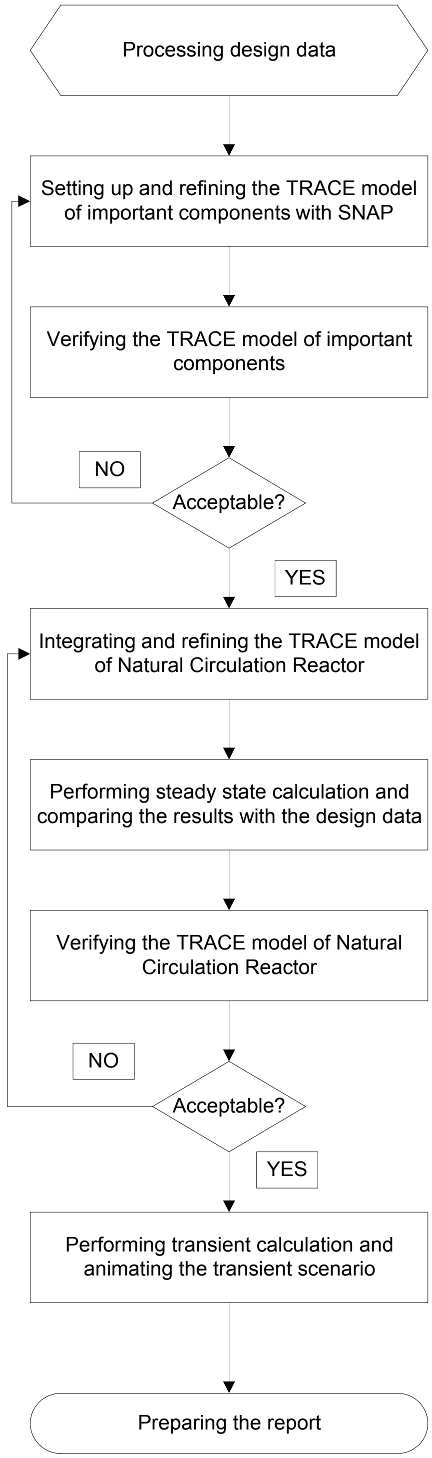


Fig. 2 The Flow Chart of TRACE Model Establishment and Verification

3. ESTABLISHMENT AND VERIFICATION OF NATURAL CIRCULATION REACTOR TRACE MODEL

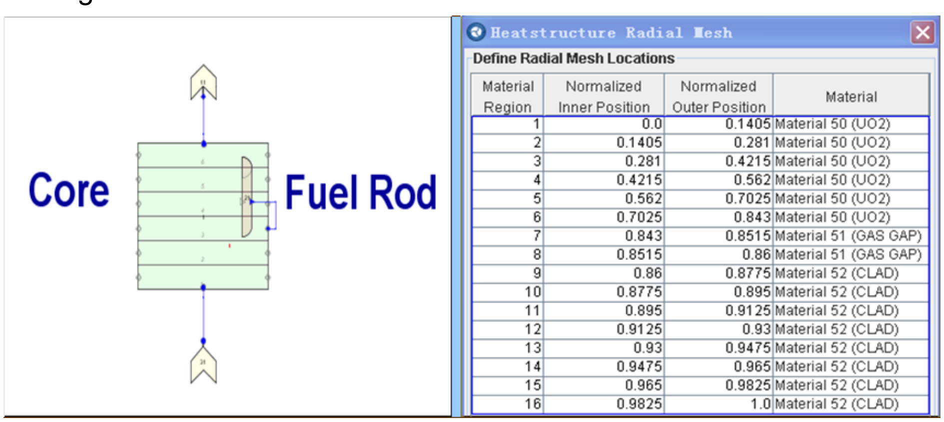
3.1 General Description

Natural Circulation Reactor is a kind of concept design developed by domestic institution in China. Most of the important components are submerged in a twenty-meter deep water pool shown in Fig. 1. The primary coolant loop is depressurized. The reactor core is placed at the bottom of the pool and the rated thermal power is 120 MW. Important components such as core and heat exchanger 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 NCR system.

3.2 Core Component

The core component is placed at the bottom of the primary loop. The rated thermal power is 120 MW. The height of core is 1.1m. It consists of 205 fuel assemblies and each assembly is made up of 60 fuel rods. The arrangement of fuel rods in an assembly is 8 X 8, and the central 4 rod positions are occupied by control rods and water rods. The size of fuel assembly is 10.8cm X 10.8cm. The pitch of fuel rods is 13.4mm. Outer diameter of fuel rod is 10mm. Zr4 clad thickness is 0.7mm. The diameter of UO2 pellet is 8.43mm. The designed core inlet coolant temperature is 80℃ and outlet coolant temperature is 110℃.

Pipe component was used to build up the core and the associated heat structure 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 core component was tested separately and the difference between calculated results and designed data were checked to confirm the accuracy of modeling.



Material Region	Normalized Inner Position	Normalized Outer Position	Material
1	0.0	0.1405	Material 50 (UO2)
2	0.1405	0.281	Material 50 (UO2)
3	0.281	0.4215	Material 50 (UO2)
4	0.4215	0.562	Material 50 (UO2)
5	0.562	0.7025	Material 50 (UO2)
6	0.7025	0.843	Material 50 (UO2)
7	0.843	0.8515	Material 51 (GAS GAP)
8	0.8515	0.86	Material 51 (GAS GAP)
9	0.86	0.8775	Material 52 (CLAD)
10	0.8775	0.895	Material 52 (CLAD)
11	0.895	0.9125	Material 52 (CLAD)
12	0.9125	0.93	Material 52 (CLAD)
13	0.93	0.9475	Material 52 (CLAD)
14	0.9475	0.965	Material 52 (CLAD)
15	0.965	0.9825	Material 52 (CLAD)
16	0.9825	1.0	Material 52 (CLAD)

Fig. 3 The TRACE Model of the Core Component for NCR

3.3 Power Component

Power component was used to generate power in heat structure standing for fuel rods. COSINE distribution was assumed for core axial power. ANS-94 was used to generate decay heat. In transient calculation, Point kinetic method and reactivity feedback of temperature is turned on in the power component. In the steady state, initial power of 120MW was maintained to perform steady state calculation. Accident “Loss of Main Feed Water” was initiated afterwards. But the reactor scram was delayed by 120s. During the delayed 120s, reactivity feedback of fuel temperature and coolant temperature was functional. After reactor scram, decay heat dominated the heat generating of core. The power component is shown in Fig. 4.

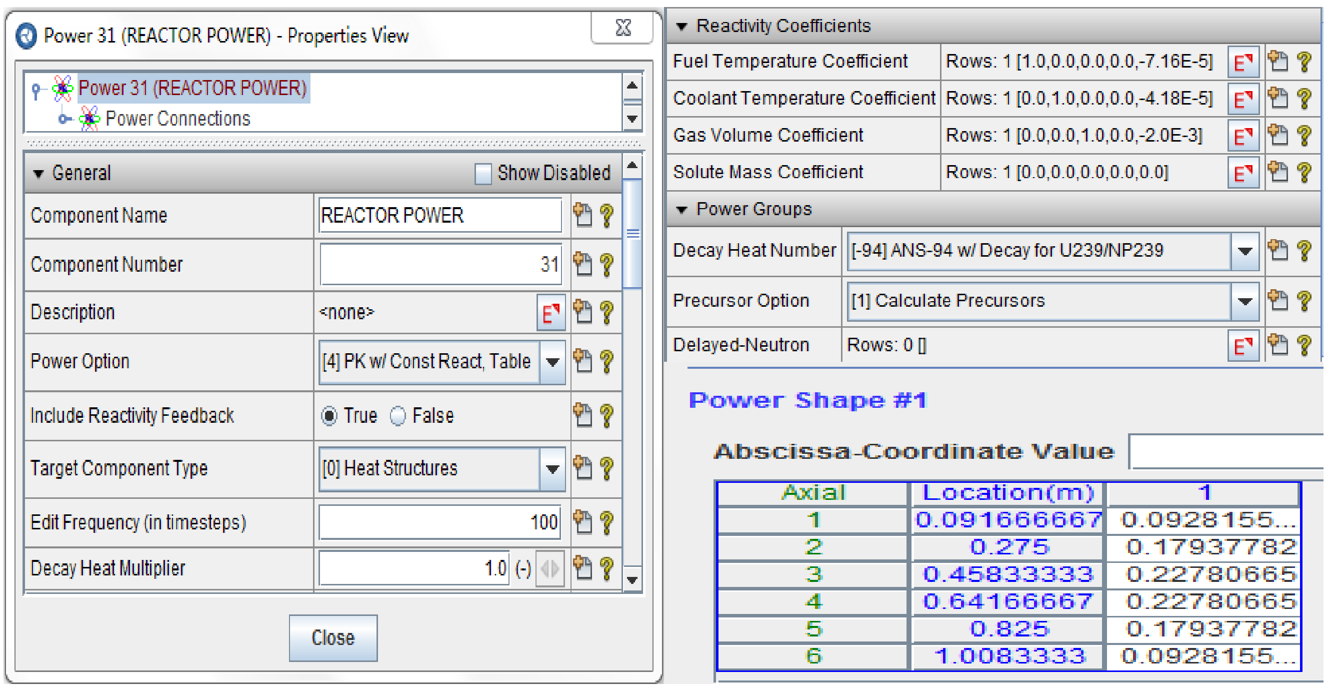


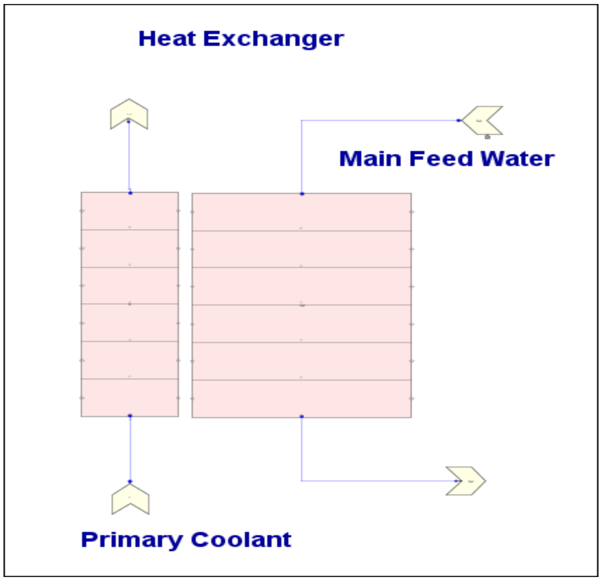
Fig. 4 The TRACE Model of the Power Component for NCR

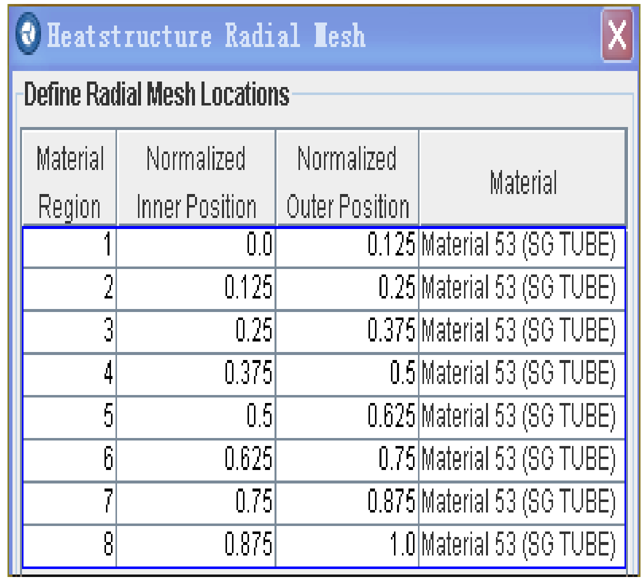
3.4 Heat Exchanger Component

The heat exchanger was modeled to simulate heat transferring from primary loop to secondary loop. The height and volume of heat exchanger was kept as accurate as possible since natural circulation capability and heat capacity could be affected by these two parameters.

Two pipe components were used to model primary side and secondary side of the heat exchanger. A heat structure was built up to simulate tubes to transfer heat. For the exchanger primary side, fill and break components were used to model inlet and outlet boundary condition. For the exchanger secondary side, fill component was used to model main feed water and auxiliary feed water. Break component was used to maintain the secondary side outlet boundary condition. The heat exchanger model is shown in Fig. 5.

Heat exchange surface was carefully adjusted to keep calculated outlet temperatures of both primary side and secondary side as close as possible to the design value.





Material Region	Normalized Inner Position	Normalized Outer Position	Material
1	0.0	0.125	Material 53 (SG TUBE)
2	0.125	0.25	Material 53 (SG TUBE)
3	0.25	0.375	Material 53 (SG TUBE)
4	0.375	0.5	Material 53 (SG TUBE)
5	0.5	0.625	Material 53 (SG TUBE)
6	0.625	0.75	Material 53 (SG TUBE)
7	0.75	0.875	Material 53 (SG TUBE)
8	0.875	1.0	Material 53 (SG TUBE)

Fig. 5 The TRACE Model of the Heat Exchanger Component for NCR

3.5 <u>Other Components</u>

Other components including downcomer and atmosphere were also well modeled with pipe and break components. The atmosphere was modeled by a break component to maintain system boundary pressure. The volume of downcomer is a very important parameter in transient simulation because it affects the heat capacity of primary loop significantly. After all of the components were modeled respectively and well tested, they could be incorporated together to set up the whole picture of the NCR TRACE model.

Fig. 6 The TRACE Model of the Atmosphere and Downcomer for NCR

4. STEADY STATE CALCULATION

4.1 Model Description

There is only one primary loop and one secondary loop for the whole NCR system as shown in Fig. 7. Fuel rods generate heat in the core which is cooled by natural circulation coolant flow. Hot water leaves the outlet of the core, rises through pipes and enters the inlet of heat exchanger. Heat transfers from the primary side to secondary side and primary coolant temperature drops before getting to outlet of the heat exchanger. Cooled water flows through a horizontal pipe to the downcomer and finally heads to the bottom of the core. The NCR TRACE model is shown in Fig. 7.

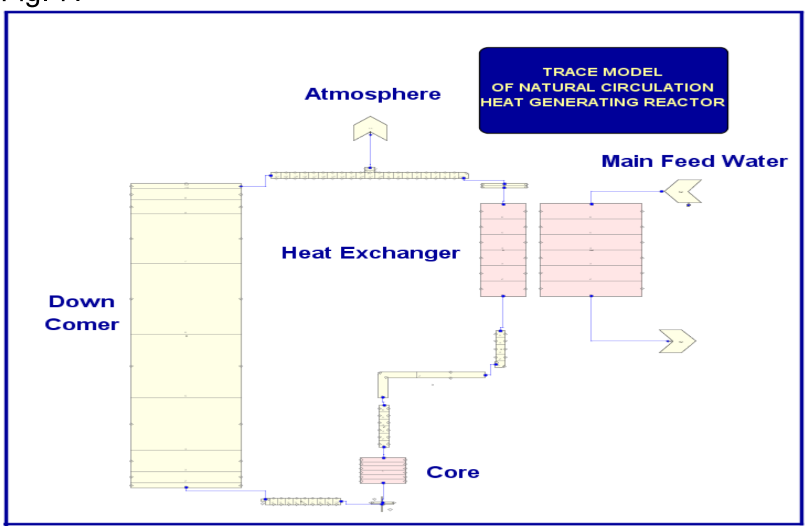


Fig. 7 The TRACE Model of NCR System

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 natural circulation coolant mass flow. After model refining, key parameters of the system have acceptable difference from the design data. The difference between steady state calculation results and the design data are shown in Fig. 8. More steady state calculation results are shown in Fig. 9 to 14.

Parameter	Prim. Side Mass Flow (kg/s)	Max. Rod Temp. (℃)	Max. Clad Temp. (℃)	Core		Heat Exchanger Temp. (℃)			
				Inlet Temp. (℃)	Outlet Temp. (℃)	Primary Side		Secondary Side	
						Inlet	Outlet	Inlet	Outlet
Design	949	800	150	80	110	110	80	70	100
TRACE Calc.	962	673	146	82	111	111	82	70	101
Error	13	-127	-4	2	1	1	2	0	1

Fig. 8 Difference Between Steady State Calculation Results and Design Data

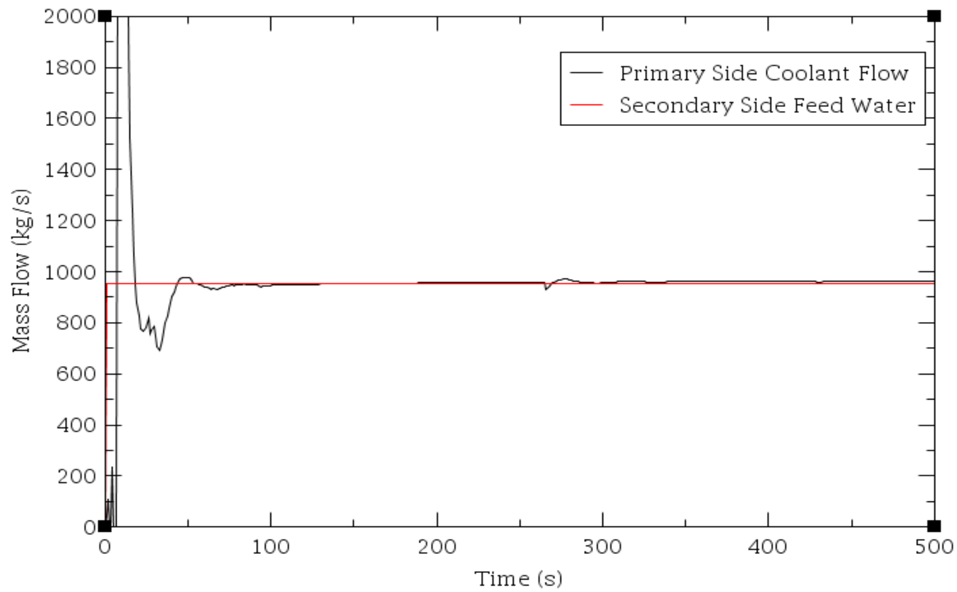


Fig. 9　Mass Flow of Primary and Secondary Side (SS)

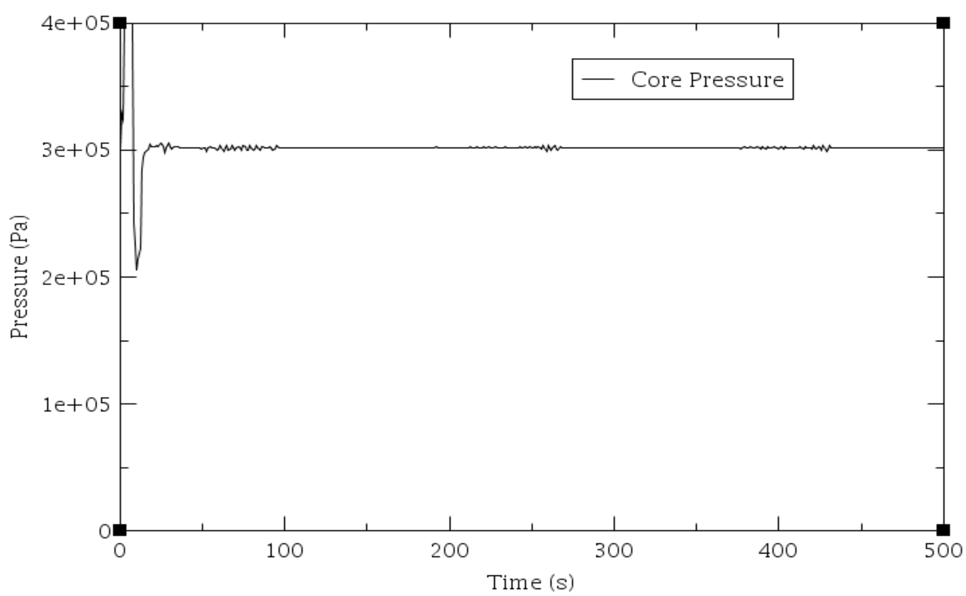


Fig. 10　Pressure of Core

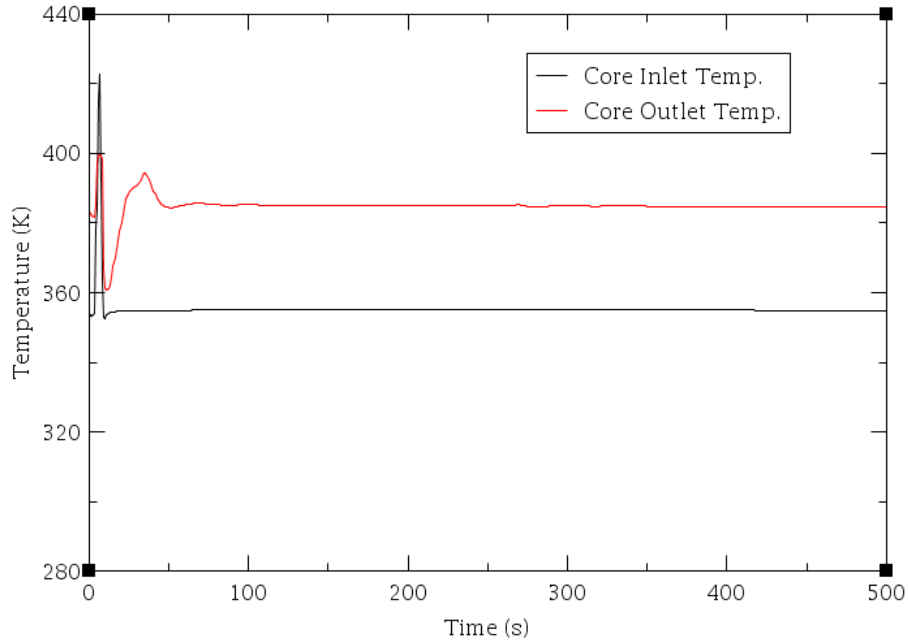


Fig. 11 Core Inlet and Outlet Temperature (SS)

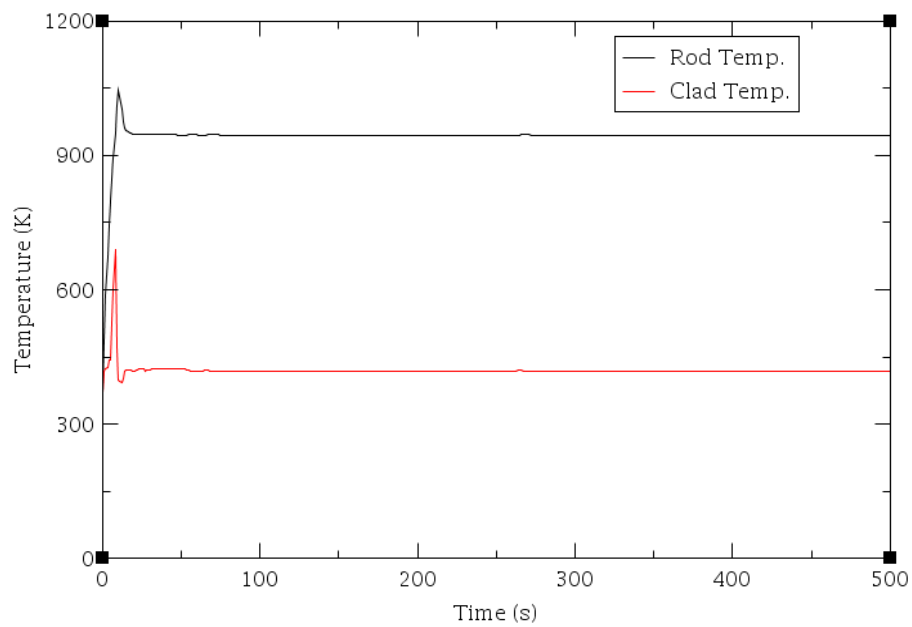


Fig. 12 Rod and Clad Temperature (SS)

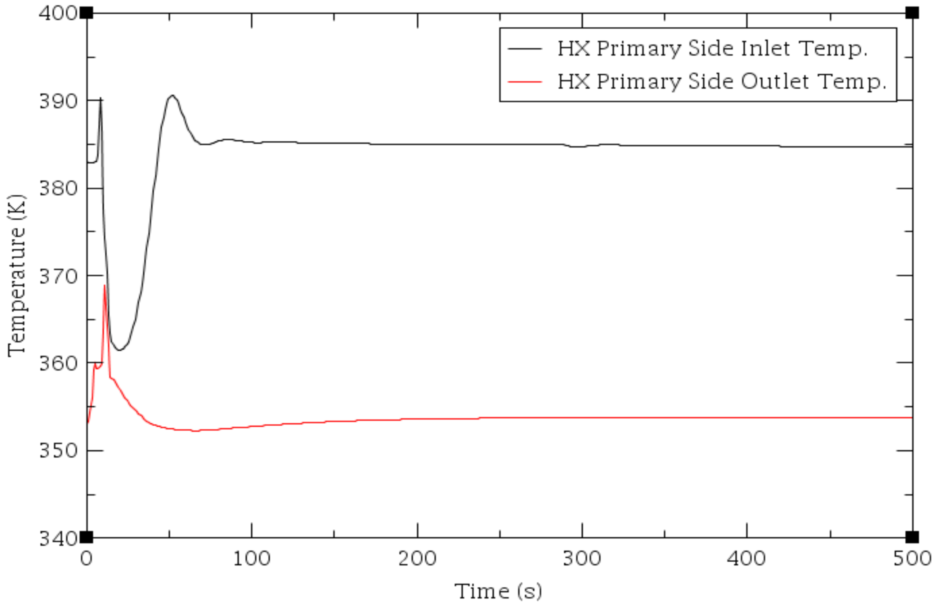


Fig. 13　HX Primary Side Inlet and Outlet Temperature (SS)

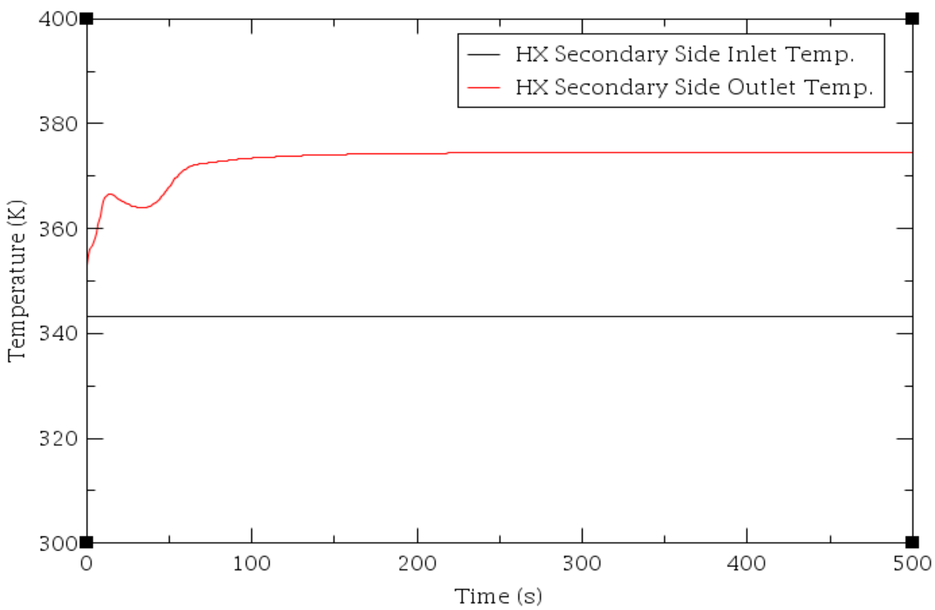


Fig. 14　HX Secondary Side Inlet and Outlet temperature (SS)

5. TRANSIENT CALCULATION

5.1 Transient Calculation Results

The “Loss of Main Feed Water” accident scenario was assumed in this section. Before 500s it was steady state. Main feed water was cut off at 500s. Auxiliary feed water was delayed by 1000s. Reactor scrammed 120s after the initiation of accident. Before reactor trip, primary side coolant flow increased and core outlet temperature dropped obviously. It resulted in reactor power increasing about 1%FP due to reactivity feedback of coolant temperature. After reactor trip, the balance of decay heat generating and the coolant heat capacity in the pool dominated the accident process.

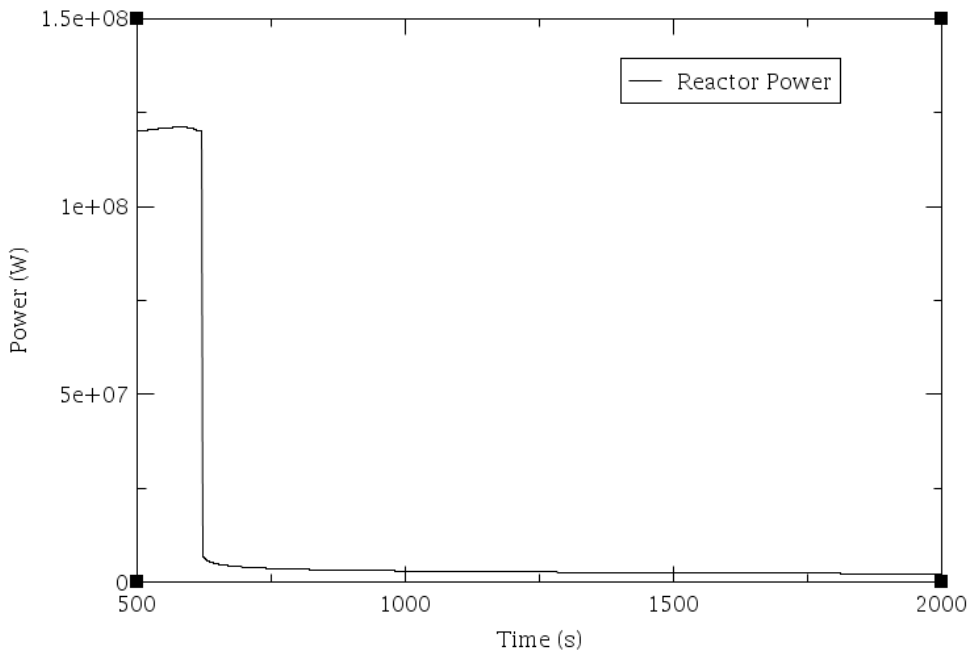


Fig. 15 Reactor Power (Transient)

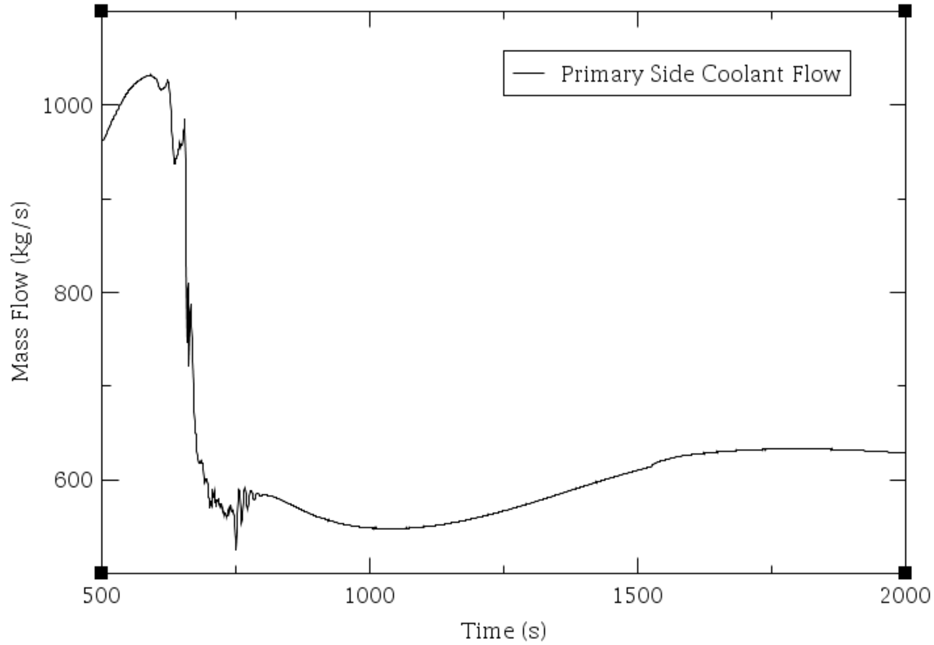


Fig. 16 Primary Coolant Flow (Transient)

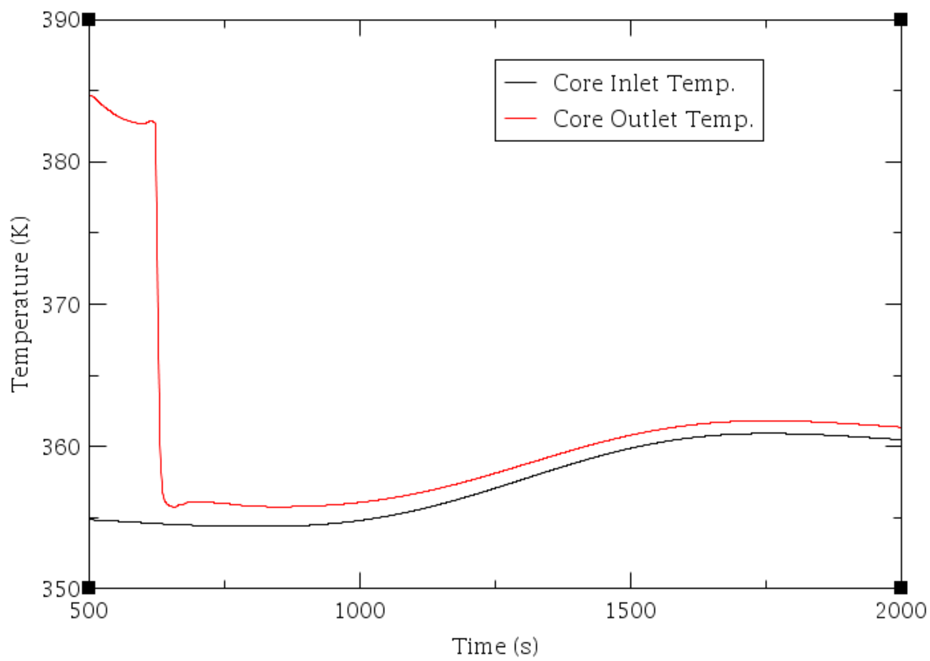


Fig. 17 Core Inlet and Outlet Temperature (Transient)

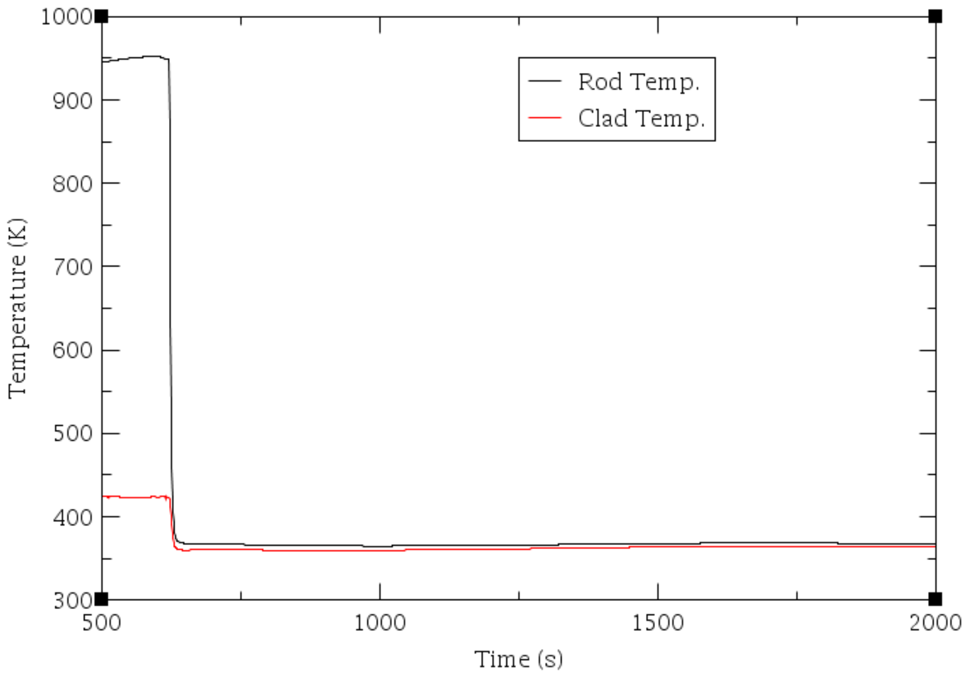


Fig. 18 Rod and Clad temperature (Transient)

5.2 Sensitivity Analysis

In a short period after the loss of main feed water and before reactor trip, primary side natural circulation coolant flow increased significantly. As core inlet coolant temperature didn't increase much at that time, fuel cooling capability became better even than normal operation condition, core outlet temperature dropped consequently. It is very interesting to find that reactor power increased, main feed water got lost, but core outlet temperature dropped. In order to get better understanding of this specific phenomenon, sensitivity analysis was performed in two cases. One is feed water sensitivity to natural circulation flow; the other is reactor power sensitivity to natural circulation flow.

Sensitivity analysis results indicate that primary natural circulation flow increases significantly as feed water flow decreases as well as reactor power increases. As core inlet coolant temperature stayed almost constant, core cooling capability temporarily increased shortly after the loss of main feed water. This is an important feature special for coolant flow of natural circulation system.

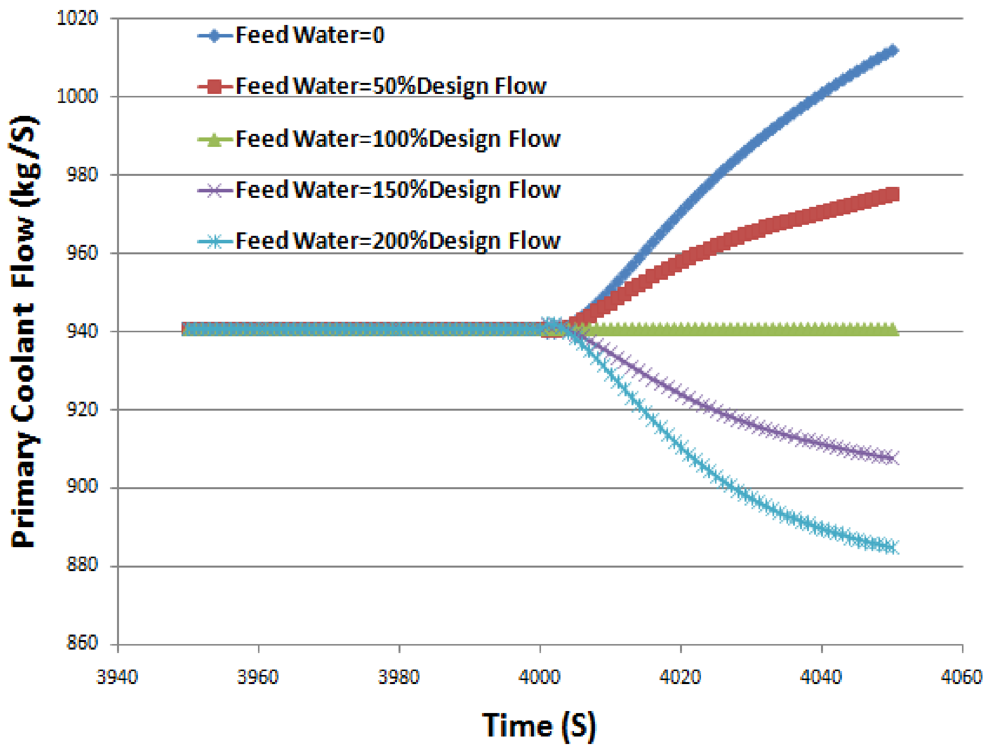


Fig. 19 Feed Water Sensitivity to Natural Circulation Flow

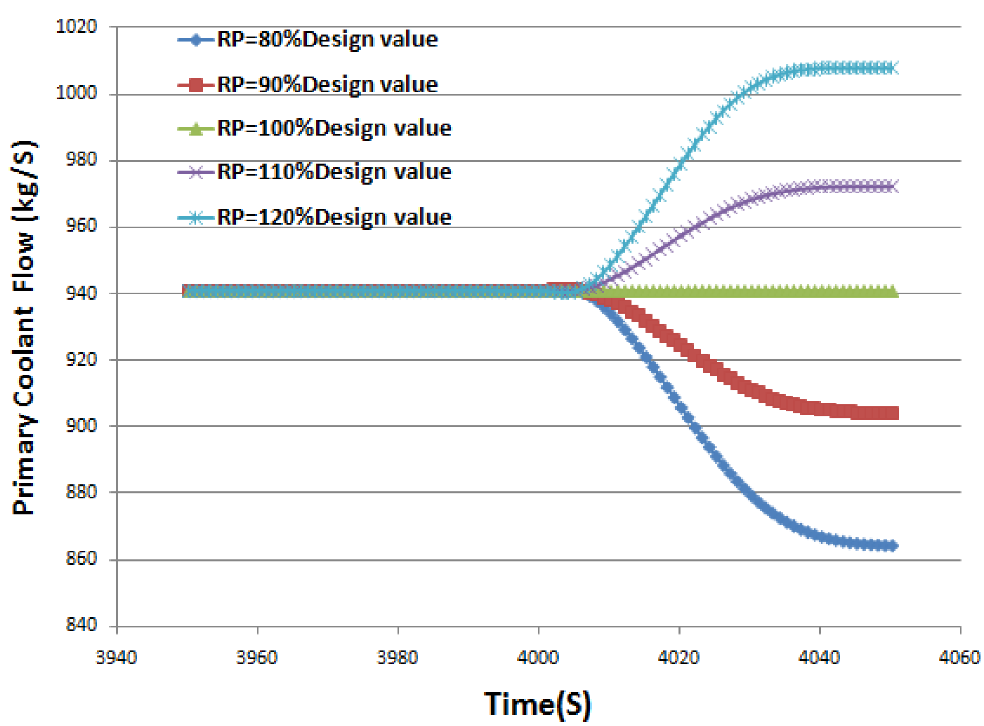


Fig. 20 Reactor Power Sensitivity to Natural Circulation Flow

6. TRANSIENT ANIMATION

The “Loss of Main Feed Water” accident scenario was animated in this section. SNAP 2.0.8 was used to animate the accident scenario. Fluid conditions and key parameter trends such as temperature and mass flow 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.

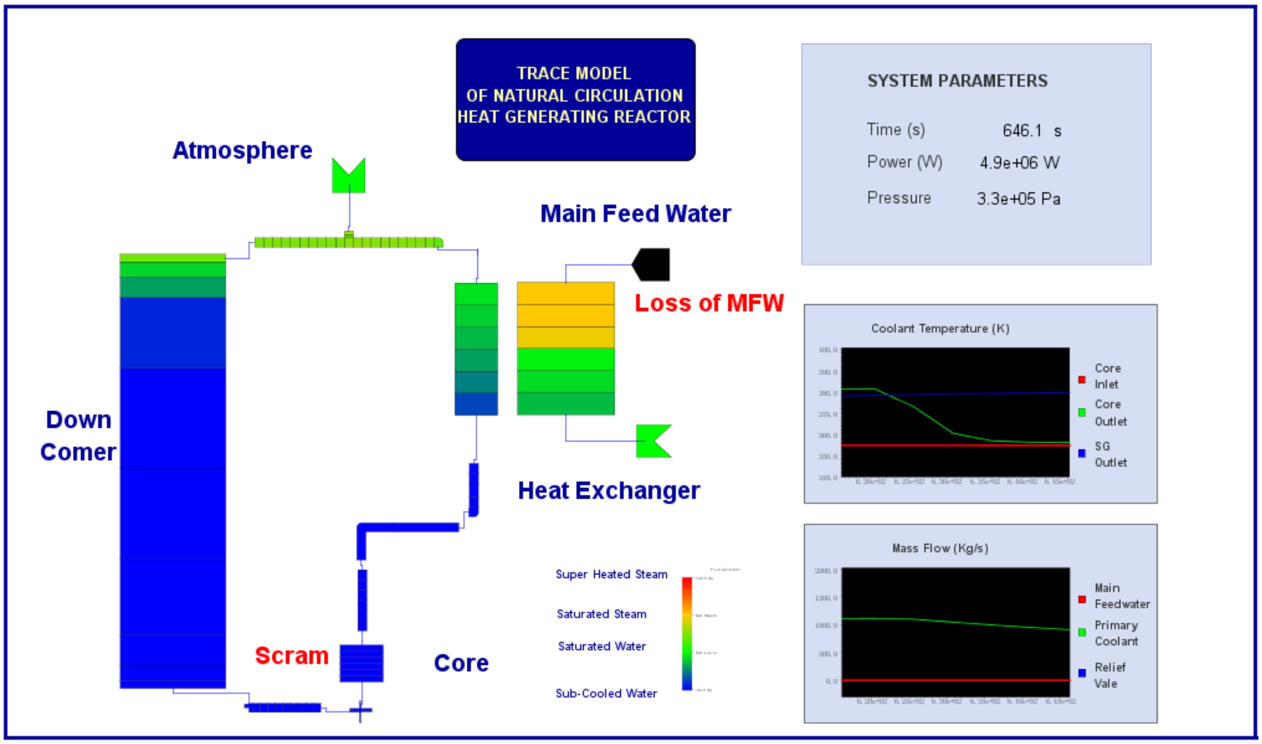


Fig. 21 Transient Scenario Animation

7. CONCLUSIONS

In this report, Natural Circulation Reactor was successfully modeled by SNAP and TRACE code. The model was mainly made up of pipe, fill, break 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 parameter trends 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 natural circulation reactor system and the calculation results are reasonable based on engineering judgment.

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 domestic PWR NPP with TRACE code next step. The future work will be also reported to NRC after completion and we hope it more contributable.

8. REFERENCES

1. NRC, “TRACE V5.0 User’s Manual”, volume 2, Office of Nuclear Regulatory Research, 2007

2. Jong-Rong Wang, Hao-Tzu Lin, Chunkuan Shih, “Assessment of the TRACE Code Using Transient Data from Maanshan PWR Nuclear Power Plant”, International Agreement Report, NUREG/IA-0241, Institute of Nuclear Energy Research, Atomic Energy Council, R.O.C., 2010.

3. A. Prošek, B. Mavko, “IJS Animation Model for Krško NPP”, International Agreement Report, NUREG/IA-0227, Jožef Stefan Institute, 2010.

4. E. Takasuo, “An Assessment of TRACE V4.160 Code Against PACTEL ATWS-10 – 13 and ATWS-20 – 21 Pressurizer Experiments”, International Agreement Report, UREG/IA-0231, VTT Processes, 2010.

NRC FORM 335 (9-2004) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **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-0418	
2. TITLE AND SUBTITLE Application of TRACE V5.0 P2 to Natural Circulation Reactor Safety Analysis	3. DATE REPORT PUBLISHED MONTH: October	YEAR: 2012
	4. FIN OR GRANT NUMBER	
5. AUTHOR(S) FENG Jinjun, CHAI Guohan, SHI Junying, ZHOU Kefeng	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.)*

Nuclear and Radiation Safety Centre
Ministry of Environmental Protection
54, Honglian Nancun, Haidian District, Beijing, 100082
China

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

A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The purpose of this work is to study the behavior of natural circulation reactor under Accident 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 program was used to facilitate system modeling work. Important components such as core heat exchanger were modeled respectively. These components were tested separately and results were compared with design data to check the accuracy. Key parameters were identified and properly adjusted to refine the model further. All of the components were incorporated together to build up the integrated TRACE model of natural circulation reactor.

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 primary side natural circulation mass flow as close as possible. The adjustment work was iterated until all of key parameters were acceptable.

The third part is transient calculation. The Accident scenario 'Loss of Main Feed Water' was simulated in the this part.

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) Nuclear and radiation Safety Center of China (NSC) National Nuclear Safety Administration (NNSA) PWR BWR LOCA Loss of Main Feed Water	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

NUREG/IA-0418

Application of TRACE V5.0 P2 to Natural Circulation Reactor Safety Analysis

October 2012