Thermal-Hydraulic Response Research of Containment Vessel during a Typical LOCA Accident

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Abstract: The containment vessel serves as the final barrier to effectively prevent the release of radioactive materials in the event of accidents at nuclear power plants. Therefore, investigating its pressure and temperature response during accidents is crucial in reactor safety research. The NHR200-II reactor is a new Small Modular Reactor (SMR) with advanced intelligent features. Its designs incorporating integrated arrangement, full-power natural circulation, and self-stabilizing pressure concepts, ensure exceptionally high inherent safety levels. Designed to accommodate various applications, including regional heating, power generation, seawater desalination, and industrial steam supply, its configuration can be tailored to meet diverse usage requirements. In the safety analysis and evaluation on containment design of nuclear power plant, the pressure and temperature response within the containment vessel during Loss-of-Coolant Accidents (LOCA) is one of the critical safety factors. This study integrates the dedicated containment analysis code GOTHIC to conduct an analysis on the thermal-hydraulic response of containment in the event of a typical LOCA accident, specifically addressing the scenario of control rod guide tube rupture in the NHR200-II reactor. Some factors affecting the transient process of the accident, such as initial conditions, boundary conditions, and the arrangement of internal components within the containment vessel, are investigated.

Keywords: NHR200-II, thermal-hydraulic response, containment vessel, LOCA.

1. INTRODUCTION

In recent years, to reduce carbon emissions and mitigate the global warming caused by greenhouse gas emissions, countries around the world have been committed to developing new types of clean energy, including nuclear energy [1-3]. Nuclear energy can not only be used for power generation but also for heating and steam supply, which are the main directions of energy demand in cities and regions with more concentrated energy needs. The demand for heating and steam supply limits the location of thermal power plants, which should not be too far from the city, thus putting forward high safety requirements for nuclear facilities [4].

Since the 21st century, nuclear energy powers represented by the United States and Russia have carried out the research and development of small modular reactors (SMRs) [5] with innovative technical characteristics. Compared with large nuclear power water-cooled reactors, these small modular reactors have the advantages of high inherent safety, diverse uses, flexible deployment, low total investment, and short construction period. The NHR-200 II natural circulation low-temperature reactor designed by the Institute of Nuclear and New Energy Technology of Tsinghua University has inherent safety and well meets the safety needs for heat and steam supply in cities.

This paper mainly studies the pressure and temperature response inside the reactor containment under the accident of control rod guide tube rupture under different boundary conditions and initial conditions.

2. BRIEF DESCRIPTION OF NHR-200Ⅱ

NHR-200II is successfully developed based on the extensive data and experience obtained from the successful design, construction, operation, and testing of the existing NHR-5 [6,7]. Considering the actual needs, the Institute of Nuclear and New Energy Technology at Tsinghua University has made improvements to the NHR-5 and designed a commercially scaled heating reactor, NHR-200I. After the Fukushima accident in 2011, further improvements were made on the basis of NHR-200I in response to new safety requirements, resulting in significant changes to the thermal-hydraulic parameters of NHR-200II. Since the main requirement of NHR-200II is to supply saturated steam, while NHR-5 is primarily for providing hot water to the thermal network, the core inlet/outlet temperatures of NHR-200II have been increased from 146/186°C to 240/286°C, and the main circuit pressure has been increased from 1.5Mpa to 8.0Mpa. Table 1 presents a comparison of the main parameters of the two reactor types.

NHR-200II features many characteristics that are distinct from the current general pressurized water reactors in its design. This article provides a brief introduction to the overall technical solution and safety design philosophy.

Table 1. Main design parameters of NHR-5 and NHR-200II

Parameters	Value	
	NHR-5	NHR-200II
Thermal power/MW	5	200
Core inlet/outlet temperature/°C	146/186	230/278
Primary system pressure/MPa	1.5	8.0
Active core height/m	0.69	2.1
Intermediate circuit pressure/MPa	1.7	8.8
Intermediate circuit temperature/°C	102/142	208/248
Heating grid temperature/°C	60/90	145/201

2.1. Overall Technical Plan

The reactor employs an integrated design with full-power natural circulation cooling and a self-pressurizing system. Within the reactor pressure vessel (RPV), the reactor core, primary heat exchanger, internal structures, and built-in control rod drive mechanisms are arranged. The core, located at the bottom of the RPV, is composed of 208 9×9 fuel assemblies. The primary heat exchangers, totaling 14 shell-and-tube units, are positioned around the inner wall of the upper RPV. The built-in control rod drive mechanisms are hydraulically driven and enclosed within the RPV, allowing them to automatically drop into the core under gravity in the event of a power loss, ensuring rapid shutdown during an accident. The upper gas space of the RPV is filled with inert non-condensable gas and saturated steam to maintain primary loop pressure, ensuring the primary loop coolant remains subcooled. The RPV features a double-layered shell structure, which could ensure that the outer shell withstands internal pressure loads and prevents core loss-of-coolant accidents in the extreme case of the inner shell exhibiting cracks [8].

Coolant circulation relies on the density difference between the cold and hot legs which facilitates natural circulation, and a long hydraulic lift section is installed above the core for enhancing the driving force further. The coolant in the primary loop flows through the reactor core and the hydraulic lift section from the lower core plate, absorbing heat before entering the primary heat exchanger. The cooled primary coolant descends through the annular space surrounding the core barrel, reaching the lower plenum of the RPV to complete the natural circulation cycle.

The NHR-200II reactor heat transfer system comprises three loops: The primary loop employs an integrated layout, utilizing natural circulation and a pump-free design to transfer heat. The absorbed heat from reactor core would be transferred to the intermediate loop via the primary heat exchanger. The intermediate loop, acting as an isolation stage between the first loop and the steam loop, is driven by electric pumps and transfers heat from the primary loop to the steam loop via the steam generator. The steam loop generates steam to meet the demands of industrial steam users, power generation, and heating, thereby achieving combined heat and power (CHP) production.

2.2. Safety Design Philosophy

NHR-200II Fully Utilizes the Concepts of Passive and Inherent Safety [9].

Residual heat is removed through natural circulation, with all three loops in the heat removal chain (primary loop, residual heat removal loop, and air cooling tower) driven by natural circulation, eliminating the need for active driving forces.

Both shutdown systems are passively driven. The control rods [10] are managed by a hydraulic drive system, using reactor coolant water as the medium. This water is pressurized by a pump and injected into the hydraulic step cylinders within the reactor, causing the control rods to move through changes in flow rate. The control rods are housed within the pressure vessel, avoiding the severe consequences of rod ejection accidents. In the event of a power failure, the control rods fall into the core by gravity, ensuring a rapid shutdown under accident conditions. The boron injection tank is located above the pressure vessel, utilizing the gravity head between them for boron injection, allowing effective operation even during a complete power outage [11].

The isolation valves of the residual heat removal system are designed to activate upon power loss. During normal operation, the solenoid valve is energized, and the electromagnetic force holds the closure element against the valve seat, keeping the residual heat removal system closed. Under accident conditions, the solenoid valve loses power, the electromagnetic force disappears, and a spring opens the closure element, activating the residual heat removal system.

The integrated primary loop arrangement eliminates the possibility of large break accidents. The double-layer shell design at the bottom of the pressure vessel limits water loss in case of inner shell leakage. The pressure vessel has a sufficiently large volume to support full-power natural circulation. The robust residual heat removal capability quickly reduces pressure and discharge flow after an accident. Multiple automatic isolation valves on small pipes from the primary loop limit water loss. The low operating pressure, temperature, and power density reduce break flow rates and latent heat within the core. These measures ensure that the reactor core's active region remains submerged at all times.

3. LOSS-OF-COOLANT ACCIDENT (LOCA) PROCESS IN THE NHR-200Ⅱ REACTOR

A loss-of-coolant accident (LOCA) is one of the most critical types of accidents in a pressurized water reactor (PWR)[12]. The integrated pressure vessel design of the NHR-200II reactor eliminates the possibility of large-break LOCAs. Therefore, only credible small-break LOCAs are considered in safety analysis [13]. Upon the occurrence of a small-break LOCA, the primary coolant with high energy releases through the break, causing an increase of the containment pressure and a coolant loss within the RPV.

In the containment design requirements of the NHR-200II, it is essential to ensure that during a LOCA, the entry of coolant into the containment must not cause overpressure damage to the containment whose pressure design limit is 0.25 MPa.

This study examines the scenario of a control rod guide tube rupture combined with the failure of two isolation valves. This scenario is critical as it leads to the maximum loss of primary loop coolant and the highest transient pressure within the containment.

After the rupture occurs, coolant loss results in triggering an emergency reactor shutdown. Consequently, the reactor shuts down, and the residual heat removal system is activated. Due to the failure of the two isolation valves, coolant loss will continue. As the water level drops below the rupture point, the sprayed coolant has a phase transitions from liquid to the gas mixture , accompanying with a significant decrease of coolant loss rate. Following the accident, the containment pressure continue rising due to the inflow of high-temperature coolant.

The coolant loss will cease when the containment pressures and the primary loop pressure reach equilibrium.

4. ANALYSIS MODEL

4.1. Introduction to Computational Software

This study focuses on the pressure and temperature response of containment during a LOCA for the NHR-200II utilizing the GOTHIC software[14]. GOTHIC is a robust, multi-component, multi-phase thermal-hydraulics code that has been endorsed by the Nuclear Regulatory Commission (NRC) of the United States. Serving as a versatile and general-purpose software package for thermal-hydraulics, GOTHIC is widely applied across the spectrum of nuclear power plant operations, encompassing design, licensing, safety assessments, and operational analyses for containment structures and confinement facilities.

4.2. Model Construction

The following sections detail the GOTHIC modeling of the NHR-200II containment and the accident scenario.

Containment Model Construction

This paper presents a two-dimensional model of the overall structure of the NHR-200II containment vessel. The internal structure primarily includes the reactor pool, reactor pit, spent fuel pool, reactor chamber, and other areas such as pipe corridors and equipment rooms. In the analysis model, the containment is totally divided into 45 chambers assigned a unique number for each one. By defining parameters such as the base elevation, height, and volume of each chamber, the position and size of each chamber are determined. Chamber 45 represents the reactor operating hall within the containment, and its volume is significantly larger than that of other chambers, allowing it to be approximated as the free volume of the containment.

The chambers only reflect the volume occupied by various facilities or devices within the containment; the internal pressure of the facilities does not necessarily equal the pressure of the chamber they occupy. For example, Chamber 23 represents the volume occupied by the pressure vessel within the containment, but the pressure, temperature, and other parameters inside the pressure vessel do not correspond to those of Compartment 23.

The interconnected chambers are described by different flow paths which can be precisely defined using parameters such as the coordinates of the start and end points, height, hydraulic diameter, and roughness. The flow paths merely represent the connections between compartments; their geometric shapes are not fixed and could even be as simple as a hole. The model constructed in this study defines a total of 60 flow paths. Among them, flow paths 58, 59, and 60 connect chamber 23 (the reactor chamber and pool) with boundary conditions 1F, 2F, and 3F to simulate coolant loss scenarios in the event of a breach. The remaining flow paths reflect the connectivity between the starting and ending compartments.

The containment shell and its internal concrete structures are modeled as the heat structure components to account for heat absorption and transfer processes during the transient. A total of 164 thermal components are defined, with component 164 dedicated to calculating the heat exchange between the containment and the atmosphere.

GOTHIC offers various steam condensation models, and this study utilizes the DLM-FM, Uchida, and Mod-Uchida models to simulate steam condensation phenomena. By observing the variations in temperature and pressure responses within the containment under each model, the overall patterns and trends are identified.

Accident Scenario Input

In this study, we assume that the reactor is operating at its rated power before the accident occurs, and at time 0 seconds, a double-ended rupture of the control rod drive line takes place.

At the rupture point, coolant is discharged into the containment, with parameters such as flow rate and enthalpy already determined at the onset of the accident. These parameters are imported into boundary conditions 1F, 2F, and 3F using the Table function. The variations in flow rate and enthalpy are shown in the following figure:

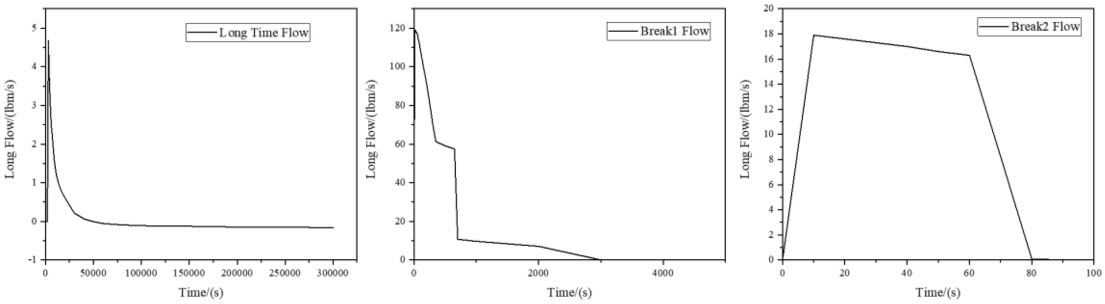

Figure 1. Flow Rate

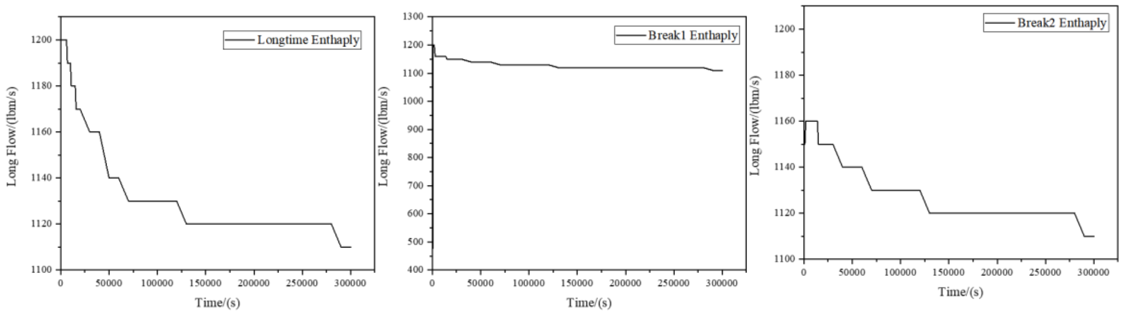

Figure 2. Flow Enthaply

5. CALCULATION RESULTS

This paper establishes a reference case with the following specific parameters:

Table 2. Parameters of reference case

Parameters	Value
Temperature inside the vessel/℃	50
Temperature of atmosphere/℃	25
Humidity inside the vessel/%	0
Shell thickness/cm	80
Initial pressure/KPa	101.4
Vessel volume /m^3	15000

During a small break LOCA, the high-temperature, high-pressure coolant from the reactor enters the containment, causing flash evaporation and a subsequent increase in containment pressure. Initially, the significant pressure difference between the reactor and the containment leads to a large discharge of coolant, resulting in a rapid pressure rise, reaching its peak around 3000 seconds. Subsequently, as the discharge flow rate from the reactor decreases, and with the heat transfer from the internal thermal components of the containment and the heat dissipation to the atmosphere, the pressure gradually decreases, returning to approximately 0.1 to 0.12MPa around 100,000 seconds.

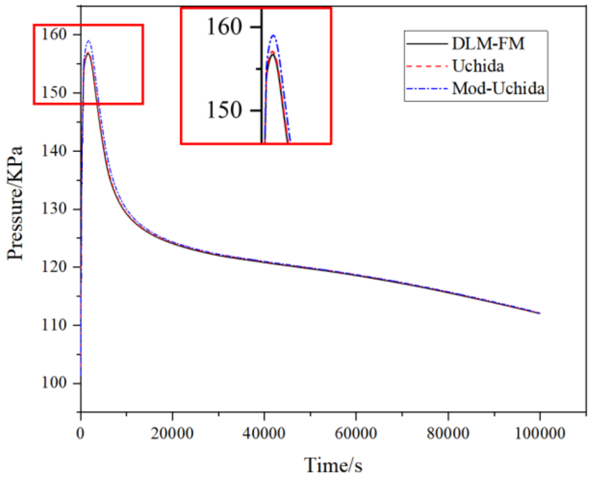

Figure 3. Reference Case(pressure)

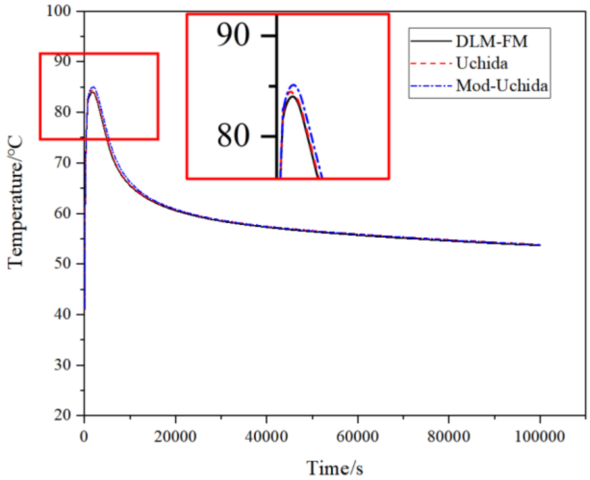

Figure 4. Reference Case(temperature)

The results of the three models are generally similar under the reference case, with only relative deviations observed near the peak values. The subsequent analysis will focus on the calculation results using the DLM-FM model.

5.1. Impact of Initial Temperature in Containment

The initial temperature in containment refers to the average temperature of the air in containment in normal steady operation. In this analysis, the initial temperatures of the internal structures in containment are assumed to be same with the initial air temperature in containment because energy balance has reached under long-term steady condition.

The initial containment temperatures were varied at 40℃, 50℃and 60℃ to analyze the changes in pressure and temperature within the containment, as illustrated in Figure 5 and Figure 6. It can be observed that as the initial temperature inside the containment increases, the peak pressure increases, and the final equilibrium pressure between the containment and the pressure vessel also increases correspondingly, because of the internal heat sinks with lower temperature which has larger heat capacity.

For the temperature response, the peak temperature difference narrows to around 5°C due to the larger heat capacity of the internal structures at lower initial temperatures. However, because the initial temperature difference is greater, the peak temperature is higher at higher initial temperatures.

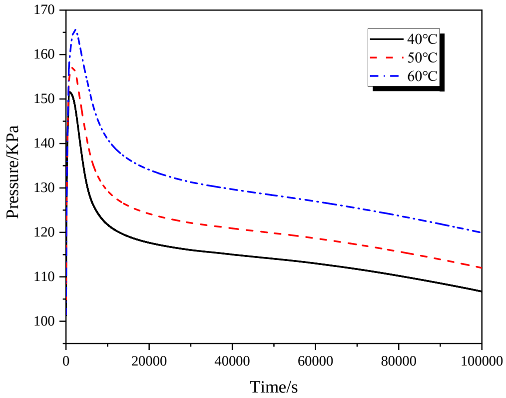

Figure 5. Variation in pressure within containment vessel with alteration in initial temperature

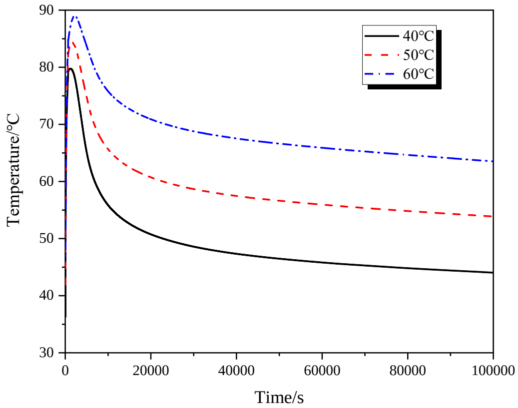

Figure 6. Variation in temperature within containment vessel with alteration in initial temperature

5.2. Impact of Containment Volume

When the containment volume changes, the peak values of internal pressure and temperature also vary, as shown in 错误!未找到引用源。 and Figure 8. The smaller the volume, the higher the peak pressure and temperature. As the volume of the containment structure decreases, its internal space becomes smaller, which means that under the same heat input, the pressure rises higher. This is because the reduced space limits the capacity of the gas to expand, leading to an increased peak pressure. During coolant leakage and heat release, the heat is more concentrated, causing the temperature to rise more quickly. The smaller space restricts the dispersion of heat, allowing the temperature to reach a higher peak in a shorter period of time.

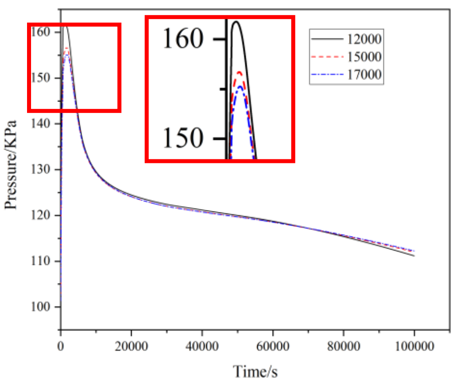

Figure 7. Variations in internal pressure with alteration of containment vessel volume

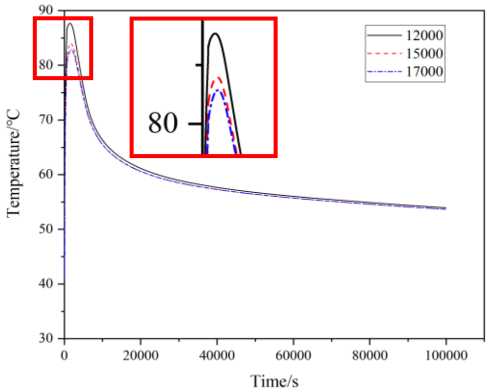

Figure 8. Variations in internal temperature with alteration of containment vessel volume

5.3. Impact of Ambient Temperature

By varying the atmospheric temperature to 5℃, 25℃, and 40℃, which largely covers the temperature range in regions where nuclear power plants are located, the pressure and temperature changes within the containment are shown in 错误!未找到引用源。 and Figure 10. The changes in pressure and temperature within the containment are relatively consistent. However, upon reaching equilibrium, due to heat exchange between the environment and the containment, the pressure and temperature inside the containment corresponding to an atmospheric temperature of 40℃ are relatively higher.

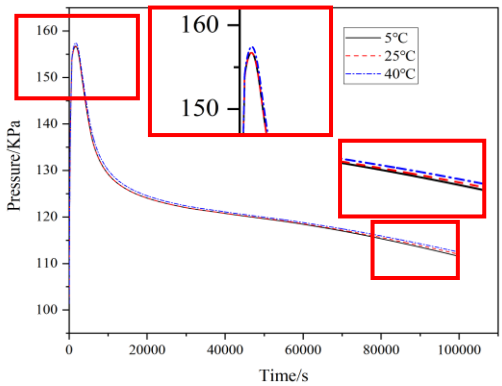

Figure 9. Variations in internal pressure with alteration of atmospheric temperature

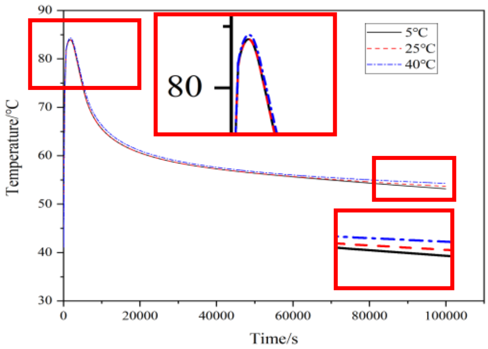

Figure 10. Variations in internal temperature with alteration of atmospheric temperature

5.4. The Impact of Shell Thickness and Internal Thermal Conductors

The thickness of the containment shell also affects the internal pressure and temperature response. By varying the shell thickness to 60 cm, 80 cm, and 100 cm, as shown in Figure 11 and Figure 12.The wall thickness can affect the heat dissipation between the containment structure and the atmosphere; as the wall thickness increases, the heat transfer efficiency deteriorates. it is observed that an increase in wall thickness leads to higher peak pressure and temperature within the containment.

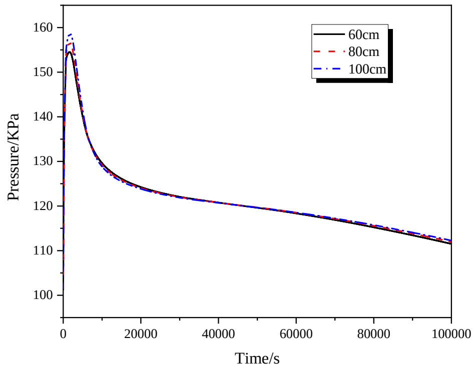

Figure 11. Variations in internal pressure with alteration of shell thickness

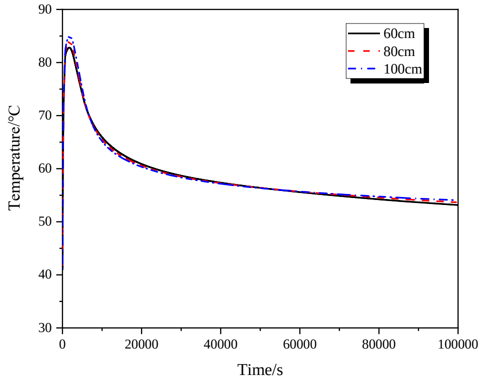

Figure 12. Variations in internal temperature with alteration of shell thickness

The internal thermal components of the containment play a crucial role in heat transfer during accident conditions. This paper also considers the containment shell as an internal heat structure. Assuming the internal thermal components are adiabatic under accident conditions, the pressure and temperature inside the containment will significantly increase compared to when the internal thermal components are intact. As shown in Figure 13, The pressure and temperature rapidly reach their peak as the high-temperature, high-pressure coolant is released from the pressure vessel, with the peak pressure already exceeding the design basis pressure of 0.25 MPa. Subsequently, equilibrium is reached with the internal pressure of the pressure vessel, and due to heat absorption by the water inside the containment, both pressure and temperature slightly decrease.

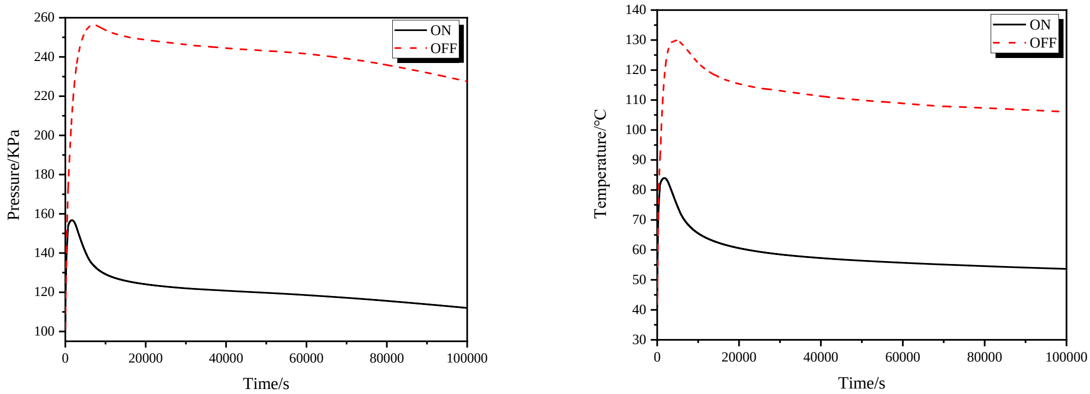

Figure 13. Variations in internal pressure and temperature with the impact of internal thermal conductor

6. CONCLUSION

This study calculates the temperature and pressure responses within the containment under various initial conditions, boundary conditions, and the impact of internal thermal components.

The calculation results indicate that, compared to the initial and boundary conditions such as the initial temperature inside the containment and the ambient temperature, the intrinsic parameters of the containment (such as containment volume, shell thickness, and internal heat structures) have a greater impact on the containment's pressure and temperature response, particularly the internal heat structures, which play a significant role in heat absorption within the containment. Furthermore, the initial temperature inside the containment has a more significant effect on the temperature and pressure response than the ambient temperature.

Due to the integrated design of the NHR-200II, which eliminates the possibility of large break accidents, this study focuses on the most severe credible small break accident: the control rod drive line rupture combined with the failure of isolation valves at both ends. Even under this scenario, the peak temperature and pressure within the NHR-200II containment remain below the design limits, demonstrating the reactor's high level of safety.

Acknowledgements

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Reference

[1] Dong, Duo et al. Study of 200 MW NHR nuclear seawater desalination and its interface plan. Nuclear Power Engineering16, 4, 377-384, 1995.
[2] IAEA (Corporate Author), and IAEA (Corporate Editor). Advances in Nuclear Power Process Heat Applications. International Atomic Energy Agency, 2012.
[3] Jianqiang, W et al. Research status and prospect of comprehensive utilization of nuclear energy. Bulletin of Chinese Academy of Sciences (Chinese Version) 34, 4, 460-468, 2019.
[4] Dong, Duo et al. "Study of 200 MW NHR nuclear seawater desalination and its interface plan." Nuclear Power Engineering 16, 4, 377-384, 1995.
[5] Subki and Hadid. Advances in small modular reactor technology developments. 2020.
[6] Dazhong, Wang et al. Technical design features and safety analysis of the 200 MWt nuclear heating reactor. Nuclear engineering and design 143, 1, 1-7, 1993.
[7] Wang, Dazhong et al. The integrated natural circulation district heating reactors. Nuclear Power Engineering 9, 1, 1-5, 1988.
[8] Wentao, H. A. O et al. Characteristics and heating market applications of NHR200-II, a small, modular integrated full-power natural circulation reactor. Journal of Tsinghua University (Science and Technology) 61, 4, 322-328, 2021.
[9] WANG, Yan et al. "Preliminary study on LOCA of natural circulation reactor NHR-200II."Atomic Energy Science and Technology 54, 5, 769, 2020.
[10] Bo, H. L. et al. Hydraulic control rod drive technology for nuclear reactors. J. Tsinghua Univ.: Sci. Technol. 45, 3, 424-427, 2005.
[11] Heng, X. I. E. et al. "Safety characteristic of NHR-200II." Atomic Energy Science and Technology 53, 10 1961-1968, 2019.
[12] Bae, Hwang et al. Comparison of three small-break loss-of-coolant accident tests with different break locations using the system-integrated modular advanced reactor-integral test loop facility to estimate the safety of the smart design. Nuclear Engineering and Technology 49, 5, 968-978, 2017.
[13] Papini, Davide et al. Analysis of different containment models for IRIS small break LOCA, using GOTHIC and RELAP5 codes. Nuclear engineering and design 241, 4, 1152-1164, 2011.
[14] Wang, Guodong et al. Application of GOTHIC 3D model to simulate thermal-hydraulic response of containment safety verification via integral test. Atomic Energy Science and Technology 51, 11, 1960-1967, 2017.