Study on Safety of Advanced Integrated Natural Circulation Reactor of NHR200-II under LOCA Condition

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Abstract: The 200 MWth nuclear heating reactor of NHR200-II is a small modular reactor with high passive safety, which would serve as a safe and economic energy source for combined heat and power of city or other industrial applications. A number of advanced designs have been applied in NHR200-II including integrated arrangement, natural circulation, self-pressurized performance, innovative hydraulic control rod drive and passive residual heat removal system. The thermal-hydraulic transient response of the reactor system during the postulated loss of coolant accident (LOCA) which is significantly different with that of a general pressurized water reactor, has a significant impact on the containment design and the evaluation on reactor passive safety. In this paper, based on the NHR200-II design at the current stage, several typical LOCA scenarios were analyzed by using the PCNHR code, which is a validated transient analysis program developed by Institute of Nuclear and New Energy Technology of Tsinghua University. The calculation results indicate that decay heat could be removed away to ensure the safety of reactor system and the NHR200-II reactor core is always coved by residual coolant in the reactor pressure vessel without any special emergency core cooling system, even during the most severe postulated LOCA transient, which validates good safety feature of the NHR200-II design.

Keywords: Nuclear Heating Reactor, Natural Circulation, LOCA, Passive Safety

1. INTRODUCTION

In modern word, the demands of energy grow very fast. Energy shortage resulting from uneven distribution of energy supply limits the development of regional industry and population centers, meanwhile, humankind have been suffering from environmental pollution caused by massive conventional coal burning. Therefore, a potential market for space heating and electricity favors the development of nuclear heating reactors (NHRs) as a clean energy source in substitution for coal-fired power plants. Various types of NHR were researched and designed, including vessel-type reactor (Zhang and Wang, 2003) and pool-type reactor (Ke et al., 2020), etc. On the basis of the heating grid conditions in China, a 5Mth vessel-type experimental reactor NHR-5 (INET, 1990) was designed by Institute of Nuclear and New Energy Technology of Tsinghua University (INET), which was originally conducted as one of national key projects in science and technology since the 1980s. The reactor construction was completed in 1989 and has been operated successfully for space heating since then. Some important safety experiments, such as passive safety test of NHR-5 for loss of main heat sink accident without reactor shutdown, residual heat removal under the interruption of natural circulation in primary loop, were performed to demonstrate the safety of NHR-5 (Wang et al., 1993a).

Given the excellent performance of NHR-5 which shows great potential for the application in combined heat and power and seawater desalination (Wu et al., 2000), a larger 200 MWth demonstration plant of NHR200-I based on the experience gained from the design, construction and operation of NHR-5 was designed by INET for seawater desalination. The preliminary safety analysis report of NHR200-I was finished and approved by the respective authorities in 1996, which demonstrates its feasibility for seawater desalination from the perspectives of technology, economic and environmental protection (Wang et al., 1993b; INET, 1996).
Due to the intrinsic safety of NHR, the location of nuclear power plant could be very close to industrial park, but the thermal parameters of NHR-5 and NHR200-I are too low for industrial steam supply. With continuous

research and improvement, an improved NHR200-II reactor is proposed to satisfy distinct application demands by INET in 2006. The key parameters of NHR200-II are listed in Table 1.

Table 1. Design parameters of different NHR200-II

	NHR200-II
Thermal Power (MW)	200
Primary Pressure (MPa)	8.0
Core inlet/outlet temperature (℃)	230/278
Number of fuel assemblies	208
Number of fuel rods per bundle	77
Intermediate loop pressure (MPa)	8.8
Intermediate loop temperature (℃)	208/248
Heating grid pressure (MPa)	1.6
Heating grid temperature (℃)	145/201.4

NHR200-II has been designed with a number of advanced and innovative features, which distinguish it fundamentally from the present-day nuclear power plants. For a better knowledge of NHR200-II, its technical details are described in the following section.

2. The NHR200-II reactor

2.1 General description

Figure 1 sketches the main reactor structure of NHR200-II. It is a vessel type light water reactor with the integrated arrangement, where the reactor core is located at the bottom of reactor pressure vessel (RPV). The reactor core of NHR200-II consists of 208 9 × 9 fuel assemblies. Cruciform BC control rods are arranged among the adjacent fuel assemblies, and Gadolinium Oxide as a burnable absorber is mixed along with the BC control rods used for reactivity control. Different with most of the current nuclear power plants that need boron acid for power regulation of reactor, the reactor coolant does not contain boron acid during normal operation. 14 double-pipe type primary heat exchangers (PHEs) are arranged on the periphery of the upper part of RPV, which provide enough capacity to transfer heat from reactor core to intermediate loop.

Under pressurized condition which is maintained by inert non-condensable gas and steam in gas space located in the top part of RPV, NHR200-II reactor could achieve a full power operation with natural circulation driven by coolant density difference. The coolant in low plenum flows up into the core and is heated there from 230 ℃ to 278 ℃. The heated coolant out from reactor core channels passes through a long riser which enlarges natural circulation capacity and converges in the upper plenum of RPV. Then, the coolant flow turns over and flow downwards into the circumferentially arranged PHEs. After heat exchange in the PHEs, the cooled water goes along with down comer and is collected in the low plenum of RPV.

The nuclear heat supply system of NHR200-II contains triple loops: primary loop, intermediate loop and steam loop, whose schematic diagram is given in Figure 2. There are two intermediate loop driven by pumps, and each steam loop is separately connected with one turbine system. Primary coolant absorbs heat from reactor core and transfers heat through the PHEs to the intermediate loops, and the heat is subsequently transferred by the SGs between the intermediate loop and the steam loop. Then, the saturated steam generated in the SGs is delivered to turbine or heating grid. The pressure of intermediate loop is designed to be higher than that of primary loop, which could keep heating grid free of radioactive contamination under some postulated accidents.

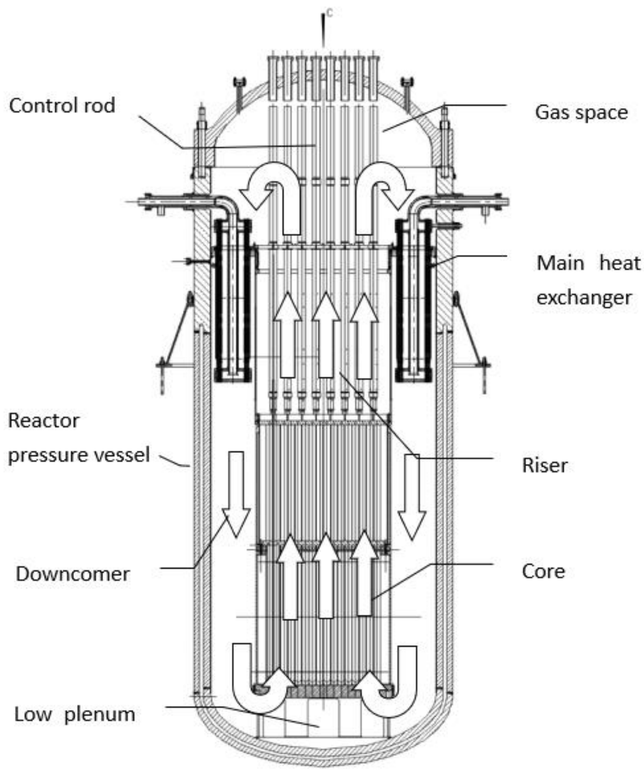

Figure 1. Reactor structure of NHR200-II

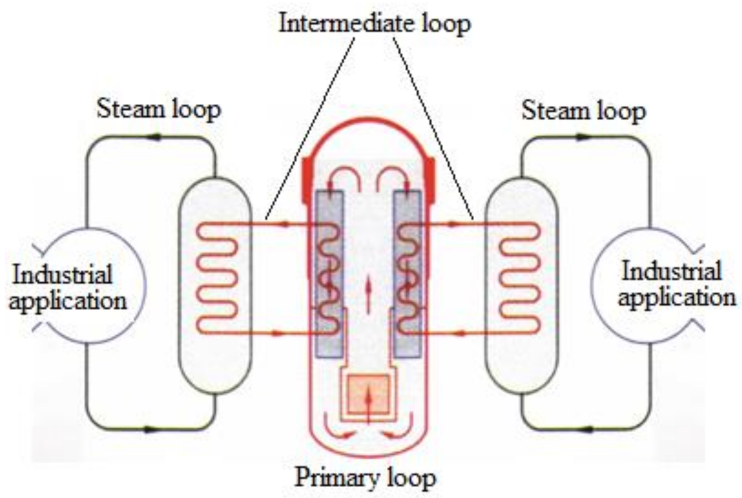

Figure 2. Schematic diagram of NHR200-II

2.2 Main Safety concepts and features in NHR200-II

The safety concepts of NHR200-II are fundamentally based on the following inherent characteristics and excellent passive safety.

Operation with Natural Circulation

Natural circulation is adopted in the primary loop of NHR200-II to remove heat from reactor core, which avoids the layout of pumps in primary loop to drive the coolant flow. Consequently, the possibility of loss-of-flow accident (LOFA) caused by the failure of reactor coolant pumps is eliminated and the passive safety of reactor is enhanced.

Integrated Arrangement of Reactor

The NHR200-II reactor adopts a vessel-type RPV design with integrated arrangement. It is noteworthy that there is no large pipe penetration through the RPV, and thus the possibility of large break LOCA is eliminated. The design that important in-vessel penetration pipelines are equipped with isolation valves and most of the in-vessel penetrations are located on the head of RPV will limit coolant loss and mitigate the consequence seriousness of small break LOCA. Additionally, the dual RPV design of NHR200-II also could avoid loss of coolant in case of the inside RPV break.

Accordance to the preliminary safety analysis of NHR200-II, the integrated NHR200-II operating at low pressure, low temperature, low power density has a low probability of LOCA occurrence and its reactor core could be always covered by coolant under any design basis accident (DBA) and the credible beyond basis accident (BDBA) without any special emergency core cooling system (ECCS), such as safety injection system in general PWR.

Large water inventory and Low pressure/power density
The water inventory in the RPV of NHR200-II is quite large. The subcooled water with large thermal inertia could both decrease the neutron influence on the RPV and contribute to reactor core cooling under the postulated accidents. In comparison with general PWRs, the primary system pressure (8 MPa) and the power density (~ 60 W/cm) of NHR200-II are much lower, which will mitigate coolant loss during the postulated LOCA and decrease the damage probability of fuel element claddings.

Effective Reactivity Control
Two shutdown systems are designed in the NHR200-II reactor: One is an innovative built-in hydraulic control rod drive system (HCDRS) (Qin et al., 2018), where the control rods driven by a hydraulic driving system are enclosed in the RPV and thus the control rod ejection event is excluded. The control rod drive mechanism designed on passive safety that control rods will fall into reactor core automatically by gravity after loss of power supply, could ensure that the reactor is shutdown rapidly under the postulated accidents; A boron injection system designed as the reserve shutdown system will be activated in the ATWS event to ensure the reactor activity could be controlled effectively under the unexpected conditions. In addition, a large negative temperature reactivity coefficient is achieved in the core physics design, which increases the inherent safety of NHR200-II.

Passive residual Heat Removal
The passive residual heat removal system (PRHRS) serving as an important safety system of NHR200-II provides heat removal capacity under the postulated accidents, which utilizes natural circulation to cool down the reactor. Based on the design considerations of redundancy and single failure criterion, two independent trains of PRHRS are provided, and each train has the capability to remove 100% decay heat load to the atmosphere.

3. LOCA of the NHR200-II Reactor

LOCA is one kind of important accidents for light-water reactor, and the amount of coolant-loss is closely related to the severity of accident consequences. In LOCA scenarios, high-energy coolant will blowout rapidly from the break into the containment, which will result in the reduction of coolant in the RPV and the increase of containment pressure. A larger amount of coolant loss would lead to higher pressure of containment. Therefore, the core will be exposed to potential damage risk when it cannot be covered by the residual water in the RPV, and the containment will also be faced with potential overpressure risk for an excessive amount of coolant entering into the containment after the LOCA occurs.

Due to the integrated NHR200-II design, large break LOCA will be eliminated completely, and only the break of in-vessel penetration pipelines with small diameter will be considered in the safety analysis and evaluation. In this paper, two typical postulated small break LOCA (SBLOCA) scenarios are selected to be analyzed:

1) Break of the pipeline in HCRDS.

As one of the in-vessel penetration pipelines, the penetration location of the HCRDS pipeline is lower than the others, which may lead to larger amount of coolant loss during the accident transient, although the accident occurrence has very low possibility.

2) Break of the in-vessel penetration on the RPV head.

According to previous descriptions, most of in-vessel penetrations are located on the head of the NHR200-II RPV. The amount of coolant loss in this scenario is less than that in the break of the HCRDS pipeline, but its occurrence has higher probability.

4. Analysis model

A code of the PCNHR developed by Tsinghua University for transient thermal-hydraulic analysis on flow systems is utilized to simulate the LOCA transient of NHR200-II. The code PCNHR (Jie et al., 2000) had been benchmarked by comparison with the famous general analysis code of RETRAN-02 (EPRI, 1981), which is widely applied in the plant design for PWR or BWR.

Based on some reasonable approximations and simplifications, the model nodalization on the main system of NHR200-II is illustrated in Figure 3, where the details on the PRHRS and the containment including the heat structures inside it are not depicted. The primary loop with natural circulation consists of 7 main control volume groups in the model: reactor core, riser, upper plenum, PHE, downcomer, and low plenum. The intermediate loops are composed of cold leg, pump, hot leg, and volume compensator. The steam loop mainly includes feed water pipeline, SG, and steam pipeline. The PHE tubes and the SG tubes are modelled as heat structure in the model, which could transfer heat between the primary side and the secondary side. The controlled 'valve' component on the junction connecting the RPV and the containment is used to simulate break or isolation of the penetration pipeline on the RPV.

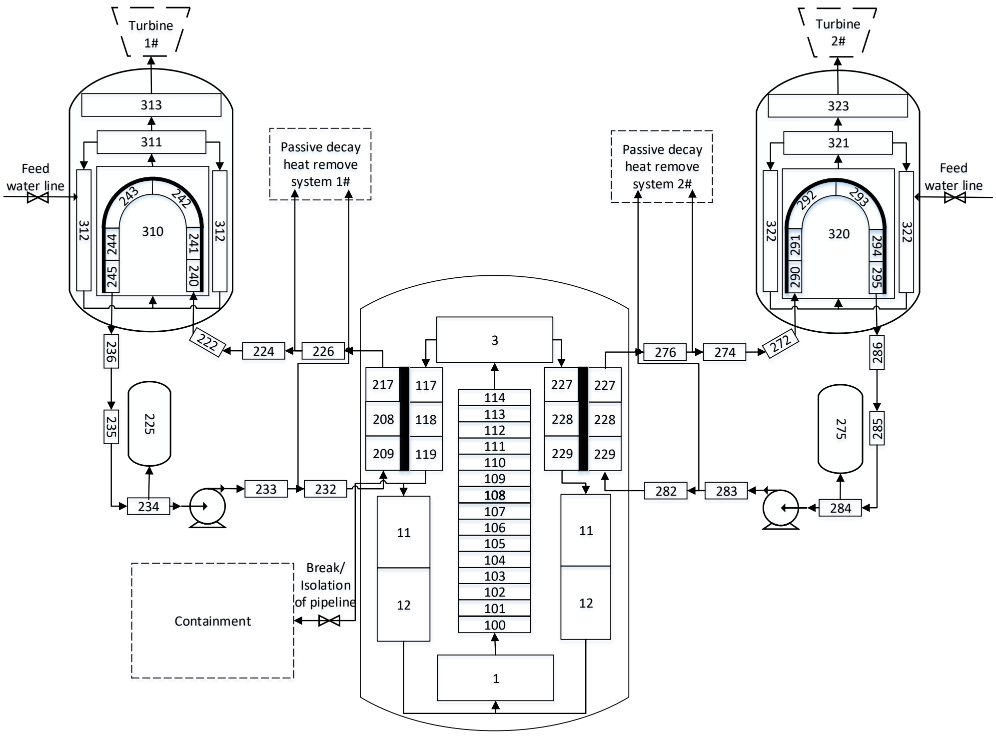

Figure 3. Nodalization of analysis model

5. Results and discussion

Based on conservative consideration, some unfavorable conditions for the LOCA transient are assumed in the analysis:

1) The initial thermal power of reactor is 105% rated power and the initial primary pressure is 101% design pressure, which are considered according to the operating limits imposed by measuring error and derivation of the reactor control system.

2) The released decay heat from the core after reactor shutdown is multiplied with a conservative factor of 1.2.

3) Only one PRHRS is available during the postulated accident.

5.1 Break of the pipeline in HCRDS

1) Assumption and claim
It is assumed that the reactor is in normal condition and then a double-ended guillotine break of the HCRDS pipeline occurs. The coolant spouts out rapidly from the RPV into the containment owing to high pressure of the primary loop. With the continuous coolant loss, the decreasing water level in the RPV reaches its low water-level limitation, and the emergency shutdown is triggered to start the following protection actions in the NHR200-II design:

- Control rods fall to shut down the reactor
- Turbine stops and feed water is closed
- Three loops (primary loop/intermediate loop/steam loop) are isolated
- The PRHRS starts up

Especially mentioned, the coolant loss would be stopped by successful isolation of the primary loop in a short time after the LOCA occurs (DBA). If all isolation valves on the broken HCRDS pipeline are in failure, although which has a very low probability as a BDBA, the coolant loss will not stop until the pressure balance between the RPV and the containment is realized.

2) Analysis results
After the double-ended guillotine break of the HCRDS pipeline occurs at 0s, the water level in the RPV decreases with the ongoing coolant loss, and the 's' signal is triggered when the water level reaches the limitation of low water-level at about 76 s. Subsequently, emergency reactor shutdown starts with control rods falling, and the reactor power decreases quickly, as shown in Figure 4.

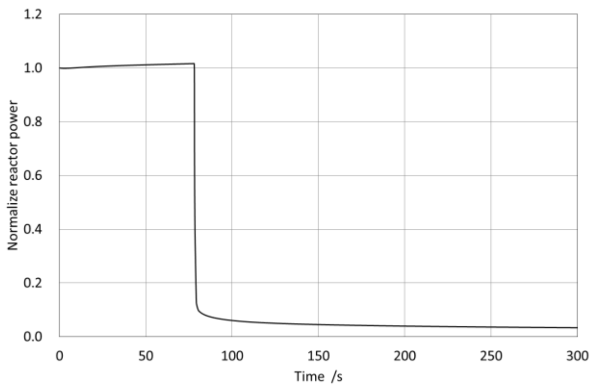

Figure 4. Normalized Power under accident of break of the HCRDS pipeline

(1) Successful isolation
As for the HCRDS pipeline break with successful isolation, the coolant loss will cease after the isolation finishes. It can be seen from Figure 5 that the coolant flow discharged from the break gets to about 55kg/s at the beginning of LOCA and decreases to 0kg/s at 96s, where it is conservatively assumed that the complete isolation of the broken pipe needs 20s. In Figure 6, the normalized amount of residual water in the RPV in the current scenario is 94% (The original inventory of coolant water in the RPV is defined as 100%), which is adequate to cover the reactor core and ensure the reactor safety after the LOCA.

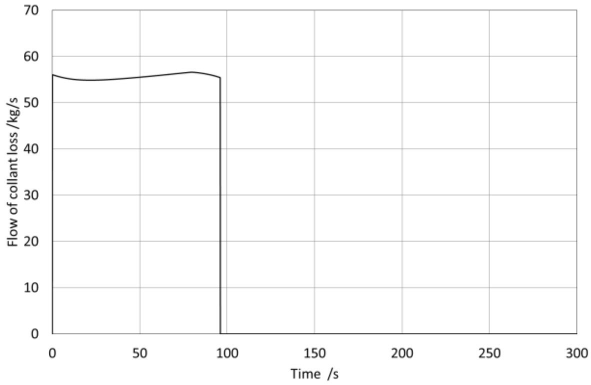

Figure 5. Flowrate of coolant loss under accident of break of the HCRDS pipeline (Isolation success)

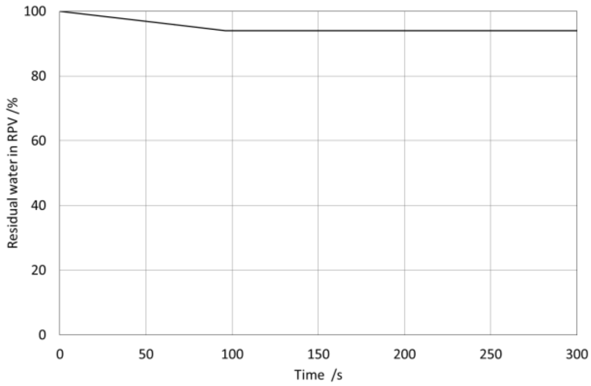

Figure 6. Residual coolant in RPV under accident of break of the HCRDS pipeline (Isolation success)

As seen in Figure 7, the RPV pressure will drop quickly at the early stage of the accident due to the coolant loss, and then maintain a slower decrease after successful isolation due to the PRHRS starting up to remove the decay heat. Because of the coolant with high energy entering into the containment, the containment pressure will increase. However, the peak containment pressure is no more than 0.14 MPa on account of the limited amount of coolant loss in this case.

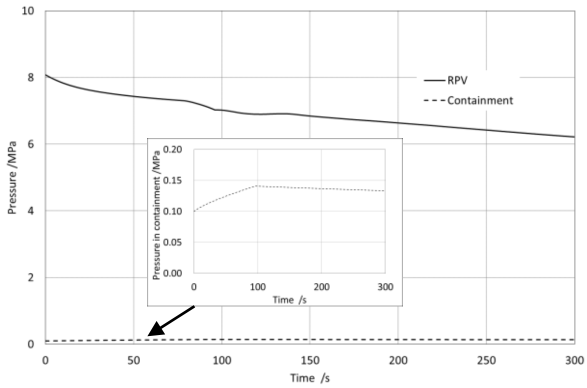

Figure 7. Containment pressure under accident of break of the HCRDS pipeline (Isolation success)

After emergency shutdown, the PRHRS will provide a steady long-term cooling to ensure the reactor safety. The normalized power removed by the PRHRS in this scenario is shown in Figure 8, where the rated thermal power of NHR200-II is defined as 100%.

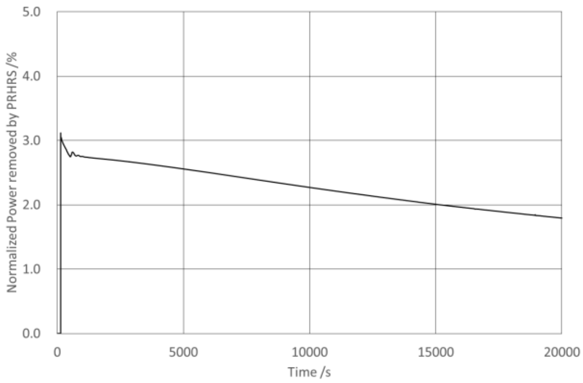

Figure 8. Normalized power removed by the PRHRS (Isolation success)

(2) Isolation failure

In the case of LOCA with isolation failure, the coolant loss will continue till the pressure balance between the RPV and the containment is reached. Figure 9 and 10 show respectively the flowrate of coolant loss and the normalized amount of residual water in RPV. Obviously, the amount of coolant loss due to isolation failure is much greater than that owing to successful isolation. However, in this case, even though the minimum standardized residual water in RPV is about 45%, it still provides a large margin for the reactor core to be covered by coolant.

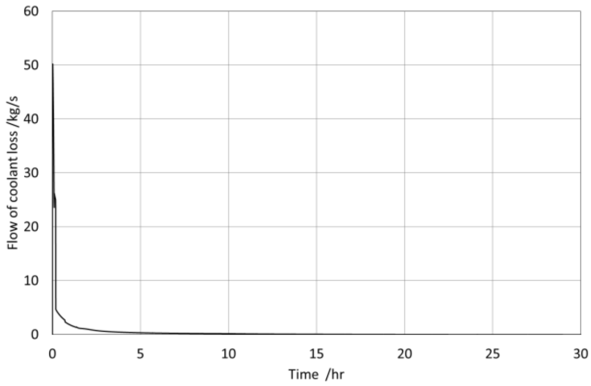

Figure 9. Flowrate of coolant loss under accident of break of the HCRDS pipeline (Isolation failure)

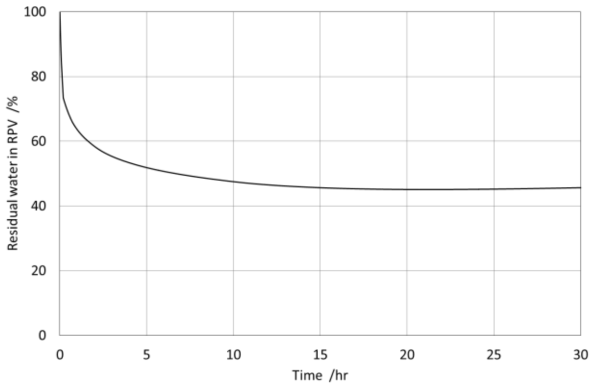

Figure 10. Residual water in RPV under accident of break of the HCRDS pipeline (Isolation failure)

Figure 11 gives the RPV pressure and the containment pressure in the case with isolation failure. Because the coolant loss rate during the small break LOCA of NHR200-II is rather slower than that during the LOCA of a general PWR, the containment pressure increases gradually, which will be mitigated by heat absorption of the structures in the containment with large heat capacity and heat transfer with the ambient environment outside the containment shell. The peak pressure is about 0.21 MPa during the LOCA transient, which is below the pressure limitation of the NHR200-II containment. About 20 hours after the accident, the RPV pressure and the containment pressure get to a balance.

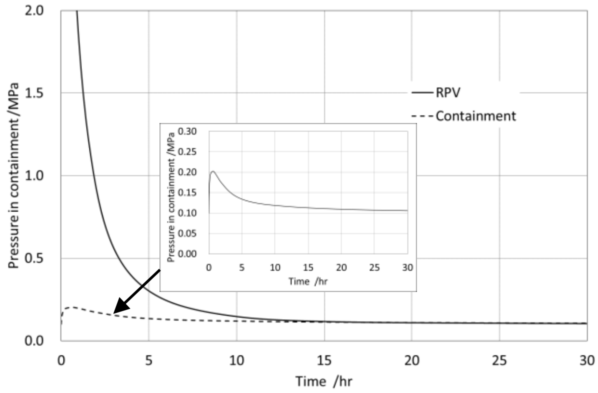

Figure 11. Containment pressure under accident of break of the HCRDS pipeline (Isolation failure)

5.2 Break of in-vessel penetration on RPV head

1) Assumption and Claim

Since most of the in-vessel penetrations are located on the RPV head of NHR200-II, a double-ended guillotine break of one in-vessel penetration as one typical DBA scenario is analyzed. Considering that the space in the RPV head of NHR200-II is filled with a mixture of non-condensable nitrogen and steam, the mixed gas will spout out from the break into the containment at the beginning. As the blowing goes on, the low water-level signal is triggered to start emergency shutdown. The coolant loss will not stop until the pressure is balanced, if no special isolation valves on the broken pipeline.

2) Analysis results

It is assumed that a double-ended guillotine break of one in-vessel penetration on the RPV head suddenly occurs at 0 s during normal operation of the reactor, the mixture of non-condensable gas and steam is discharged from the break of in-vessel penetration. The water level in the RPV drops along with the coolant loss, and the emergency shutdown is triggered when the water level reaches the limitation at about 512 s. As shown in Figure 12, with the fall of control rods, the reactor power decreases quickly.

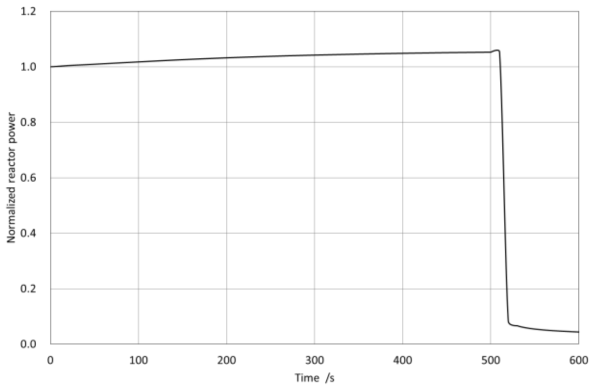

Figure 12. Normalized Power under accident of break of in-vessel penetration on the RPV head

The fluid flow through the break is presented in Figure 13. The coolant loss will cease when the pressure balance between the RPV and the containment is achieved, and the minimum of the normalized amount of the residual water in the RPV shown in Figure 14 is about 50%, which could cover the reactor core with great margin. As is obvious in analysis results, the amount of coolant loss under the in-vessel penetration break on the RPV head is much less than that under the HCRDS pipeline break. It can be explained by that the location of the broken HCRDS pipeline is lower than that of the broken in-vessel penetration on the RPV head. The discharged coolant from the in-vessel penetration break is the mixture of gases, while the discharged coolant from the HCRDS pipeline break is liquid before the water level in the RPV decreases to the break location, which result in a wide difference on the amounts of coolant loss between the two LOCA scenarios.

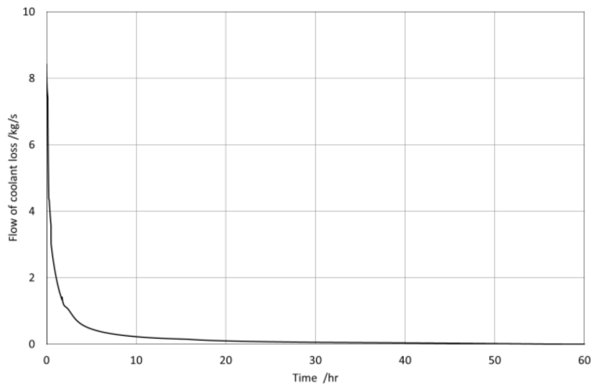

Figure 13. Flowrate of coolant loss under accident of break of in-vessel penetration on the RPV head

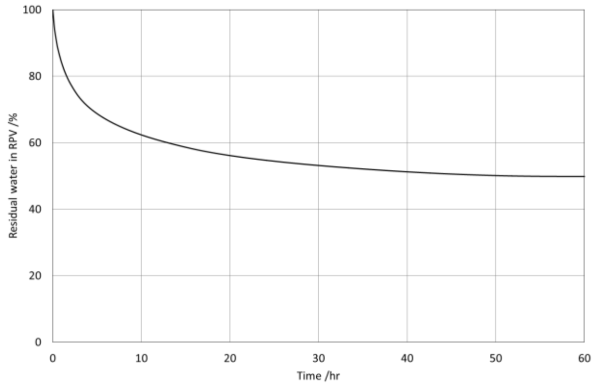

Figure 14. Residual water in RPV under accident of break of in-vessel penetration on the RPV head

As shown in Figure 15, about 40 hours after the accident, the pressure balance between the RPV and the containment are achieved. Furthermore, the peak value of the containment pressure in this scenario is 0.18 MPa, which is obviously lower than that under the HCRDS pipeline break.

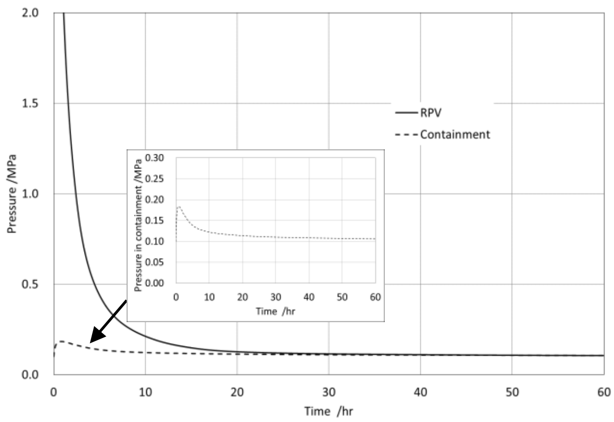

Figure 15. Containment pressure under accident of break of in-vessel penetration on the RPV head

6. Conclusion

The NHR200-II serving as an advanced natural circulation reactor with high passive safety could provide combined heat and power for city or other industrial applications. The integrated reactor pressure vessel is adopted in the NHR200-II design, which eliminates the possibility of large break LOCA. Several typical small break LOCA scenarios have been envisaged and analyzed to confirm the NHR200-II safety. In the postulated LOCAs including DBA scenario and BDBA scenario, the reactor core of NHR200-II could always be coved by the residual water in the RPV and cooled down effectively by the PRHRS to ensure reactor safety without special emergency core cooling system, which greatly simplifies the safety system design of NHR200-II. The peak pressure of containment is much lower than that in general PWR, which benefits for the containment design of NHR200-II. The analysis results validate the excellent passive safety of NHR200-II under LOCA condition.

Acknowledgements

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