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Discuss on the accident behavior and accident management of the HTGR

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Abstract – *The Chinese 200 MWe High Temperature gas-cooled Reactor Pebble-bed Module (HTR-PM) plays an important role in the world-wide development of Generation-IV nuclear energy technology. The first concrete for the HTR-PM reactor building was poured in December 2012, and the connection to the electric grid is expected in 2018.*
In this paper, based on the design of the HTR-PM, the reactor behaviors during typical design basis accidents (DBAs) and beyond design basis accidents (BDBAs) have been studied and summarized. It can be proved, that the design of the HTR-PM guarantees the inherent safety feature. In DBAs, the maximum fuel element temperatures will never surpass its design limit temperature and the release of the fission products will also below the limitation. Even in the typical BDBAs with extremely low probability, there is enough time, e.g. several days, to adopt appropriate measures to mitigate the consequence, so that the large release of the radioactive materials would not happen.
Accident management is important for the nuclear power plant. Besides, After the Fukushima Dai-ichi nuclear accident, in the worldwide, people are more concerned about the severe accident management of the nuclear power plant, and some standards and guidelines are established. Based on the accident behaviors of the HTR-PM, its accident management is preliminary discussed in this paper. Due to the inherent safety design, the accident management procedures could be simplified to a great extent and no offsite emergency measures are needed. Compared to the other types of the nuclear power plant, e.g. the Pressurized-Water Reactor (PWR) power plant, the Severe Accident Management Guideline (SAMG) can be simplified or even unnecessary for the HTR-PM.
Above work also can provide reference for the further study on the safety standards or guidelines for the design, operation and accident management of the high temperature gas-cooled reactor (HTGR).
Abstract – *High Temperature gas-cooled Reactor Pebble-bed Module (HTR-PM), inherent safety, accident analysis, accident management, SAMG*

I. INTRODUCTION

Accident management is important for all the nuclear power plant. Besides, after the Fukushima Dai-ichi nuclear accident, the severe accident gains more and more attention in the nuclear industry and the regulatory body. For example, the Severe Accident Management Guideline (SAMG) is required according to the policy of the regulatory body in China.

High Temperature Gas-cooled Reactor (HGTR) with well-known safety features has been selected as one of the candidates for the Generation IV nuclear energy system and can be widely used for power generation, heat supply and technology heat utilization. The commercial-scale 200 MWe High Temperature gas-cooled Reactor Pebble-bed Module project (HTR-PM) has been designed and is now under construction in China [1, 2]. With the adoption of the TRISO-coated particle fuel and the reasonable core design, the core meltdown and large release of radioactive materials can be minimized or truly eliminated for the HTGR. It can be expected, that the accident management of the HTGR with the inherent safety design could be significantly simplified compared to the current generation II and generation III nuclear power plants.

In this paper, based on the design of HTR-PM, the typical Design Basis Accident (DBA) scenarios and Beyond Design Basis Accident (BDBA) scenarios have been analyzed. Based on the accident behaviors of HTR-PM, its accident management is preliminary discussed.

II. HTR-PM

The HTR-PM nuclear power plant consists of two pebble-bed modular reactors connected to a single steam-turbine generator for a combined power of 500 MWt and an electrical generating efficiency of about 42%. The primary circuit of the HTR-PM is shown in Fig.1, while the general design parameters are listed in Table 1.

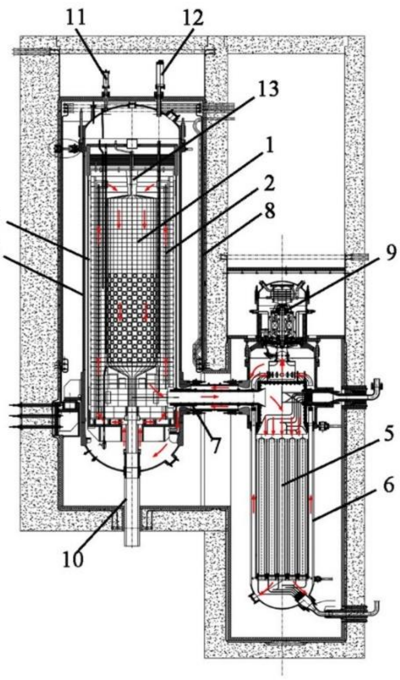


Fig. 1: Cross-section of the HTR-PM reactor.
1 reactor core; 2 side reflector and carbon shield; 3 core barrel; 4 reactor pressure vessel; 5 steam generator; 6 steam generator vessel; 7 coaxial hot-gas duct; 8 water-cooling panel; 9 blower; 10 fuel discharging tube; 11 control rod driving system; 12 small absorber sphere unit; 13 fuel charging tube

Table 1: Main design parameters of the HTR-PM.

Parameter	Value
Reactor total thermal power, MWt	2×250
Rated electrical power, MWe	210
Average core power density, MW/m^3	3.22
Net electrical efficiency, %	42
Primary helium pressure, MPa	7
He temperature at reactor inlet/outlet, °C	250/750
Primary helium flow rate, kg/s	96
Core main flow rate, kg/s	≥86.4
Heavy metal loading per fuel element, g	7
Enrichment of fresh fuel element (for the equilibrium core), %	8.5
Active core diameter, m	3
Equivalent active core height, m	11
Diameter of the RPV, m	~6.0
Number of fuel elements in one module	420,000
Number of fuel passages through the core	15
Average burn-up, GWd/tU	90
Main life steam pressure, MPa	13.9
Main life steam temperature, °C	571
Main feed water temperature, °C	205

The main material of the fuel element, reflector and carbon brick is the graphite, with the fusion point about 3600 ºC. The thermal decomposition of the SiC layer of the TRISO particle can be obviously detected only when its temperature reaches 2100 ºC. According to the enrichment of fresh fuel element used in the HTR-PM, as well as the design burn-up, the temperature limitation of the HTR-PM fuel elements during DBAs is set at 1620 ºC, below which the fission-product-retention capability of the TRISO particles can be well guaranteed [3, 4, 5]. Due to the excellent fission-product-retention capability of the TRISO particles, during the course of normal operation, the radioactivity in the primary circuit is maintained at very low levels.

Two independent shutdown systems, the control rod system and the Small Absorber Sphere (SAS) system, are installed and placed in the side graphite reflectors.

II.A. Engineered safety system

Some important engineered safety systems are introduced below.

(1) Primary pressure release system

In order to effectively protect the reactor pressure boundary from overpressure, the primary pressure release system is designed as a two-fold redundancy. The set-points of the two safety valves are 7.9 MPa and 8.4 MPa respectively, while the diameters are 6.5 mm and 52 mm respectively. Both safety valves are expected to close again once the primary pressure has reached the value of 6.9 MPa.

(2) Reactor Cavity Cooling System (RCCS)

In HTR-PM, the RCCS [5] is designed as a passive system to remove the heat from the RPV and reactor cavity to the final heat sink – the atmosphere in normal and accident conditions, ensuring the thermal integrity of the RPV and the cavity concrete. The water-cooling panel, placed on the inner surface of the reactor cavity concrete, mainly consists of a cylindrical plate and vertical cooling tubes uniformly welded on the plate. After the reactor shutdown, the decay heat of the core will be transferred first to the RPV, and then to the water-cooling panel by radiation and natural convection. The heated water in the tubes flows upwards to the air-water heat exchangers in the air-cooling tower and finally transfers the heat to the atmosphere.

(3) Reactor Pressure Vessel (RPV) support cooling system

Besides, the RPV support cooling system is also designed as a passive system, which can guarantee the temperature of the concrete near the RPV support below the limitation, even in the BDBA of the complete failure of the RCCS.

(4) Secondary circuit isolation system

Two feed-water isolation valves, as well as two live-steam isolation valves, are installed on the feed-water line and the live-steam line respectively. In accident condition, these valves will be closed soon after the accident is detected and the protective actions are triggered.

(5) Steam Generator (SG) emergency drainage system

The SG emergency drainage system is specially designed for the water ingress accident. High attention has been paid to the water ingress phenomenon because it is one of the most particular and important accident types for the HTGR [6]. An airtight drainage tank, installed inside the reactor building below the steam generator, is connected to the feed-water line via two parallel and independent relief lines. Each relief line has two relief valves. The large content of the steam and water between the feed-water isolation valves and the live-steam isolation valves can be discharged to the tank within 30 s if the high humidity in the primary circuit is detected, so as to reduce the water ingress amount to the primary circuit.

(6) Ventilated low pressure containment

Compared to the air-tight confinement, the ventilated low pressure containment is a special structure on technical basis of inherent safety design of the HTR-PM.

In normal operation, negative pressure is kept in the containment and the air is released through stack after being filtered. An accident negative pressure ventilation system is also designed, for example, to be used after a Depressurized Loss of Forced Cooling (DLOFC) accident.

II.B. Protection System

During the normal operation, if any accident is detected by the protection system, protective actions will be executed immediately, including:

▪ Dropping of control rods
▪ Shutdown of blower and close of blower flap
▪ Isolation of SG on both sides

Besides, if a DLOFC accident occurs and is detected by high pressure sliding rate (absolute value) of the primary circuit, the isolation of the primary circuit will also be executed. The water ingress accident can be detected by the highly reliable and redundant humidity sensors in the primary circuit and the SG emergency drainage system will be started immediately. The relief valves will close again when the pressures of two sides of the SG reach balance. For the BDBA of Anticipated Transient Without Scram (ATWS), the SASs will be dropped in case of the drop of control rods fails.

II.C. Typical Accident Scenarios

Typical accident scenarios of HTR-PM include:

(1) Reactivity accident
(2) Main heat transfer system malfunction
(3) Primary circuit depressurization
(4) Water ingress
(5) Air ingress
(6) ATWS

Based on the Probability Safety Analysis (PSA) results, typical DBAs and BDBAs of HTR-PM have been analyzed [7, 8]. Some are listed as below:

▪ DBAs

(1) Inadvertent withdrawal of one control rod
(2) Inadvertent acceleration of blower
(3) Loss of off-site power
(4) Loss of feed water
(5) Break of one primary tube with a diameter of 65 mm
(6) Double-ended guillotine break of a SG heating tube

▪ BDBAs

(1) ATWS
(2) Loss of feed water together with failure of closing blower flap
(3) Double-ended guillotine break of a SG heating tube together with failure of SG emergency drainage system
(4) Break of one primary tube together with total failure of RCCS
(5) Air ingress caused by the simultaneous rupture of the fuel charging tube and fuel discharging tube

III. ACCIDENT BEHAVIOR OF THE HTR-PM

The features of the HTR-PM accident behavior can be summarized based on a series of in-depth analyses.

III.A. DBA - DLOFC accident

DLOFC accident is a typical DBA and receives high attention, because it results in the higher maximum fuel temperature compared to other DBAs. Fig. 2 shows the maximal and average fuel temperatures during a DLOFC accident caused by break of a primary tube with the diameter of 65 mm.

With such large a break, the reactor will depressurize quickly (as shown in Fig. 3). Since the relatively low density of the coolant at about 1 bar atmosphere, the natural convection in the core can be neglected, which means, after reactor shutdown, the decay heat in the core will be transferred from the fuel to the RPV mainly via radiation and heat conductivity. With the development of the accident, the reactor starts to heat up and the fuel temperature continuously increase because the decay heat production rate is initially stronger than the heat removal rate. About 20~30 h later, the maximal fuel temperature is predicted to reach a peak value around 1500 ºC. Considering the uncertainty, there is a high degree of confidence that this peak value would not exceed the fuel temperature limitation of 1620 ºC and additional particle failure would not happen. The average fuel temperature will reach the peak value more lately.

Due to the low radioactivity level in the primary circuit, even all the coolant is discharged, the release of radioactive materials can be maintained below the limitation even if the accident negative pressure ventilation system fails to work [8].

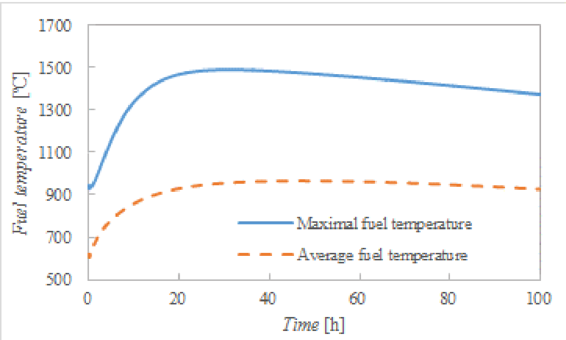


Fig. 2: Fuel temperature during DLOFC accident.

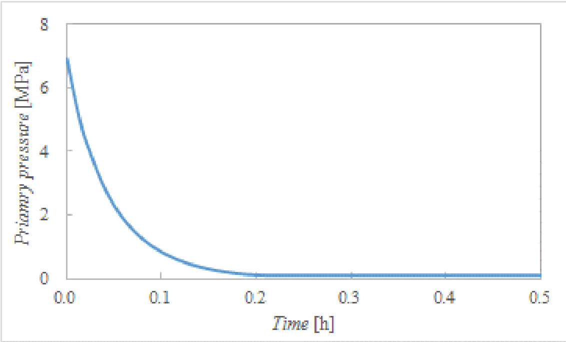


Fig. 3: Primary pressure during DLOFC accident.

III.B. DBA - PLOFC accident

Due to the inherent safety design features of the HTR-PM (e.g. large negative temperature feedback coefficient, large temperature margin between the operation temperature and temperature limitation, low power density, large heat capacity, and so on), the accident phenomena, especially the long-term change of the fuel temperature and the primary pressure, in most of the DBAs except the DLOFC accident and water ingress accident, is very similar and can be described as Pressurized Loss of Forced Cooling (PLOFC) accident.

Fig. 4 shows the maximal and average fuel temperatures during a typical PLOFC accident (inadvertent withdrawal of one control rod), while Fig. 5 shows the primary pressure during the accident.

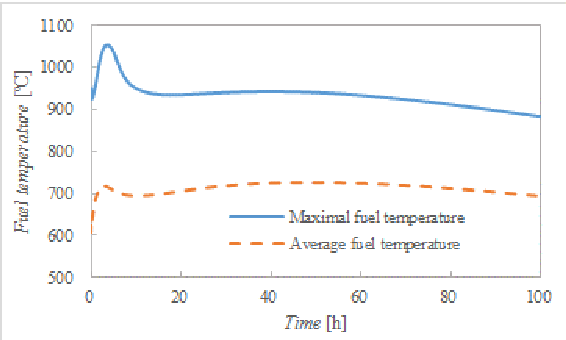


Fig. 4: Fuel temperature during PLOFC accident.

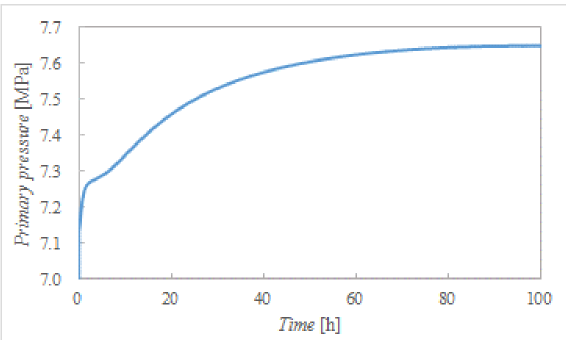


Fig. 5: Primary pressure during PLOFC accident.

After the shutdown of the reactor and close of the blower, the fuel temperature will increase due to the loss of forced convection. But in the PLOFC accident, the high temperature difference in the core and the high pressure helium in the primary circuit will cause a strong natural convection, which can effectively enhance the heat transfer and cooling of the core. Maximal fuel temperatures and average fuel temperatures during the accident are much lower compared to those in the DLOFC accidents. The peak value of the maximal fuel temperature is less than 1100 ºC, still within the limit for which the fuel is designed.

The heat accumulation in the core will also result in the temperature increase, as well as the pressure increase of the helium coolant. As shown in Fig. 5, during the accident, the primary pressure remains below 7.9 MPa, and the safety valve would not open.

III.C. DBA – water ingress accident

Water ingress accident is special for the HTGR, because it may cause the introduction of the reactivity, the corrosion of the graphite, and the increase of the primary pressure.

The most serious water ingress DBA is the Double-ended guillotine break of a SG heating tube. After the break of a SG heating tube, at the first stage, less than 200 kg steam may flow into the primary circuit due to the much higher pressure in the secondary circuit. After the pressures of the primary circuit and secondary circuit reach balance, there is about 400 kg steam remained in the steam generator, as well as the feed-water line and live-steam line between the isolation valves, and more than 3000 kg water/steam is discharged into the drainage tank [8, 9].

Conservatively assumed that 600 kg steam enter the primary circuit with a mass flow of 5 kg/s, the introduction of positive reactivity by water ingress into the reactor core leads to a nuclear power increase. But the power will ultimately decrease via the negative temperature feedback and drop of the control rods, as shown in Fig. 6. As shown in Fig. 7, the fuel temperature curves are similar as those in other PLOFC accidents. The water ingress together with the resultants of the chemical reaction will cause the faster pressure increase compared to the other PLOFC accidents and the safety valve may open if the helium purification and pressure regulation are neglected (Fig. 8).

With the open of the safety valve, a part of the helium will be discharged and flow into the atmosphere after being filtered. Even conservatively assumed that the helium flows out of the containment without being filtered, the release of the radioactive materials is still below the limitation.

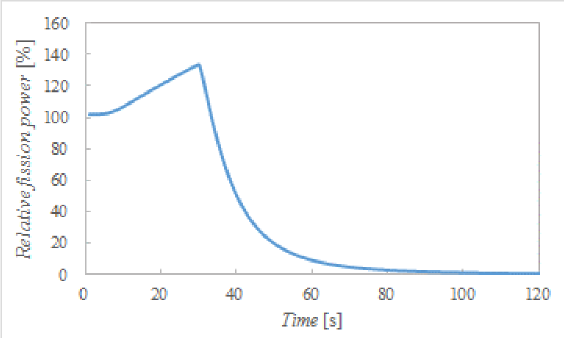


Fig. 6: Relative fission power during water ingress accident.

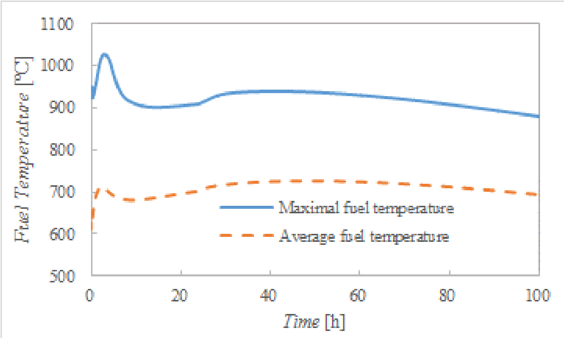


Fig. 7: Fuel temperature during water ingress accident.

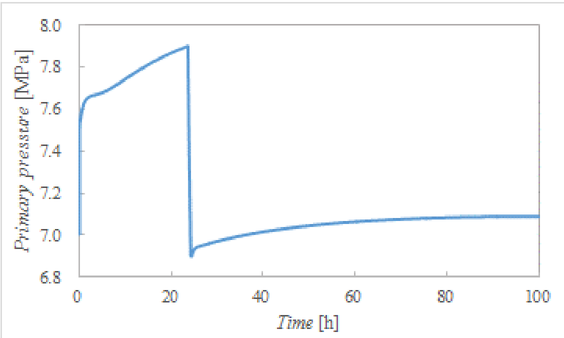


Fig. 8: Primary pressure during water ingress accident.

III.D. BDBAs

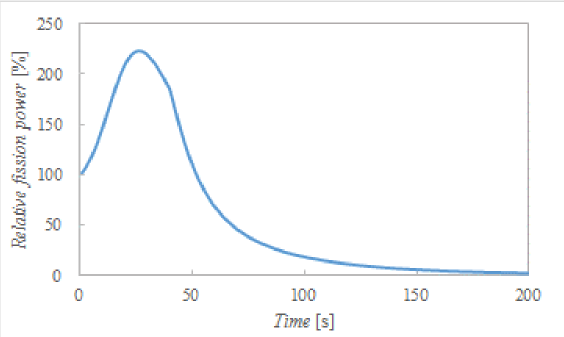
All the BDBAs with relatively higher occurrence probability (e.g. > 10^{-8} per reactor year) or being considered as important according to the PSA results are studied. Some typical accidents are introduced below.

(1) ATWS accident

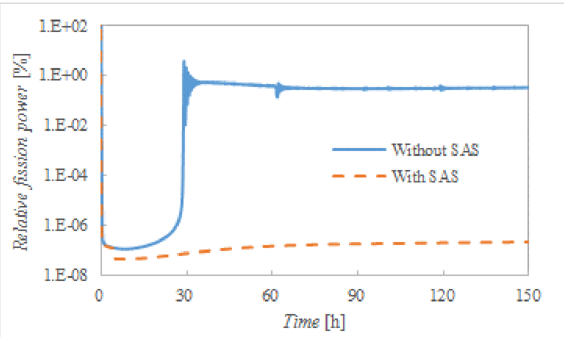
Fig. 9 to Fig. 11 show the analysis results of one control rod withdrawal ATWS accident. In HTGR, after an accident has been detected, the most important and only necessary protective action is to close the blower. After loss of forced cooling, the reactor core can shut itself down via the temperature increase and as well the consequent negative temperature feedback, even all the control rods fail to drop, as shown in Fig. 9a).

Even if the secondary shutdown system (SAS system) fails, as a result of the xenon concentration decrease, the reactor will be critical again at approximately 30h later. Despite of re-criticality, the reactor power maintains a much lower level below 0.5% rated power after several damped oscillations. The fuel temperature will increase again and the primary pressure will reach the set-point of the first safety valve. But the maximal fuel temperatures keep below the limitation of 1620 ºC and the release of the radioactive materials is limited.

It can be seen from the analysis results, that unlike the ATWS accident in a typical Pressurized-Water Reactor (PWR), in HTGR, the slow transient ensures operators enough time to repair the control rod system or launch the SAS system. ATWS accident would not cause any unacceptable consequence.

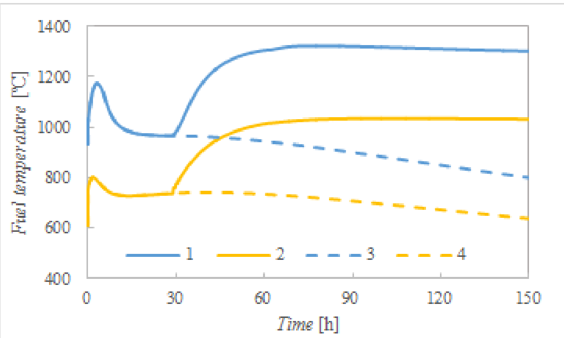


a) Short-term



b) Long-term

Fig. 9: Relative fission power during ATWS accident.



1: Maximal fuel temperature (without SAS)
2: Average fuel temperature (without SAS)
3: Maximal fuel temperature (with SAS)
4: Average fuel temperature (with SAS)
Fig. 10: Fuel temperature during ATWS accident.

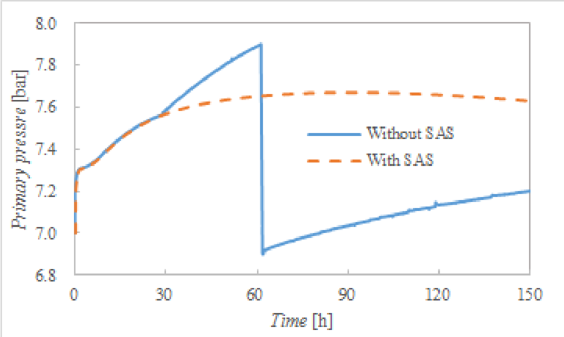


Fig. 11: Primary pressure during ATWS accident.

(2) Air ingress accident

If an upper pipe and a lower pipe (for example, the fuel charging tube and discharging tube both with diameter of 65 mm), which are both connected to the primary circuit, simultaneously rupture in front of the isolation valves, after a quick depressurization, the colder air in the containment will be sucked into the hotter core through the lower pipe and then flow out from the upper pipe, due to the so-called chimney effect. The probability of this kind of air ingress accident is below 1×10^{-10} per reactor year [8].

If it is assumed that after the rupture of two pipes and the consequent quick depressurization, the gas mixture in the reactor cavity consists of 20% air and 80% helium, with the temperature of 200ºC, the calculation result of the mass flow rate at the inlet is shown in Fig. 12, among which the mass flow rate of the oxygen is about 0.02 mol/s. Fig. 13 shows the total mass of the graphite being corroded during the accident [10].

In each modular reactor of the HTR-PM, there exist more than 80 t fuel elements and 250 t graphite reflectors. Besides, for each fuel element with the diameter of 6 cm, there exists an outer protective graphite shell with a thickness of 0.5 cm, also called as fuel-free zone. So, the calculation result indicates that in several days, there is only very little part of the graphite being corroded.

The slow air ingress and slow corrosion mean that the exposure of the coated particles will not occur in a relatively long time of about several days. Therefore, there is enough time to execute some mitigation measures, for example, plugging the broken pipe or injecting gases which would not react with the graphite, so as to deter the air from continuously entering the core.

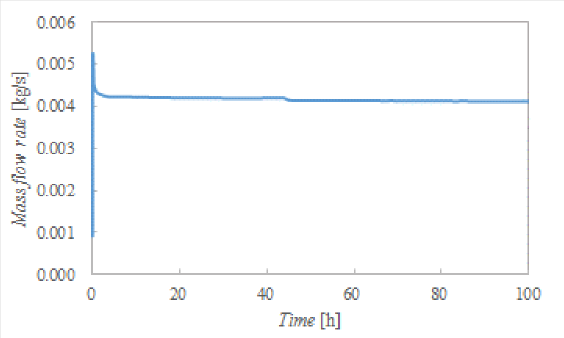


Fig. 12: Natural convection flow rate during air ingress accident.

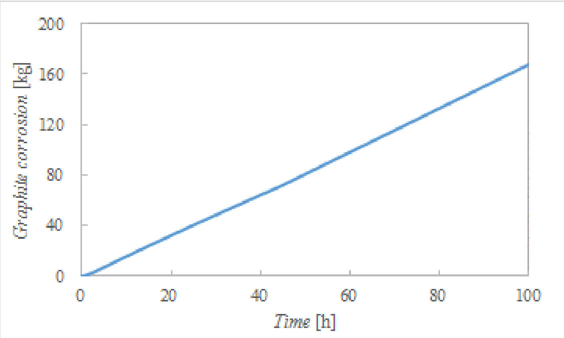


Fig. 13: Total graphite corrosion during air ingress accident.

(3) Break of one primary tube together with complete failure of RCCS

As mentioned above, in HTR-PM, a passive RCCS is designed to remove the heat from the RPV and reactor cavity, ensuring the thermal integrity of the RPV. The RCCS of HTR-PM is designed as 3 independent series with 3×50% heat removal capability.

Fig. 14 shows the maximal RPV temperature during an accident of one primary tube break together with complete failure of RCCS. The complete failure of the RCCS would cause the continuous increase of the RPV temperature in a long time. But, it can be seen that the heat up of the RPV is also a very slow process and there is enough time to execute some mitigation measures, for example, repairing the RCCS or re-establishing the forced cooling in the primary circuit. Besides, in unpressurized condition, the RPV can keep integrity under higher temperature.

Due to the inherent safety design of the HTR-PM, the decay heat can be transferred from the core to the RPV via the heat conductivity, radiation and natural convection. So, even the RCCS fails, the fuel temperature still can keep below the 1620 °C.

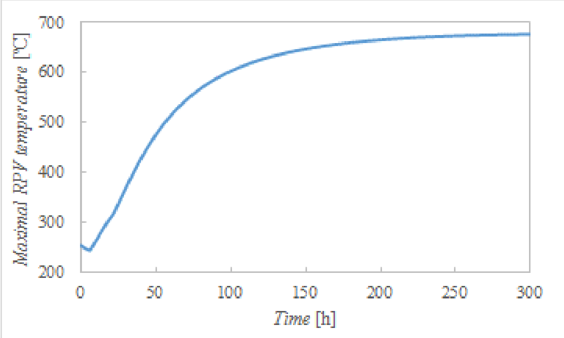


Fig. 14: Maximal RPV temperature during a DLOFC together with complete failure of RCCS accident.

IV. DISCUSS ON THE ACCIDENT MANAGEMENT OF HTGR

Based on the design and accident analysis, accident management procedures and guidelines have been developed for the HTR-PM. But as the first commercial-scale modular HTGR, further study on its accident management is also necessary and important.

As mentioned above, the use of TRISO particle fuel element ensures no core meltdown in HTGR, and a reasonable design can guarantee the inherent safety of the reactor. The analysis results also prove, that in all the DBAs, as well in the selected BDBAs which are with relatively higher occurrence probability or considered as important for HTR-PM, the fuel temperature keeps below the limitation of 1620 ºC and the release of the radioactive materials is very limited.

Besides, the systems and apparatuses involved in the HTGR is much less than what are used in the PWR, which can significantly minify the accident management actions. The simpler structure of HTGR also makes it easy for the operators to determine the accident scenarios. Once an accident has been detected and reactor is shut down, only several parameters need to be monitored to confirm the safety of the reactor, including the reactor power, the primary pressure, the RPV temperature, the working parameters of the RCCS, and so on.

In the HTGR, the system, equipment and instrumentation, which are designed for the normal operation and DBAs, also can be used for the BDBAs, and no additional equipment and instrumentation need to be designed for the BDBAs.

Because there exists no core meltdown in the HTGR, and the heating-up process of the core is very slow, the operators have enough time to take measures mitigating the accident consequence. Therefore, it would be possible for a HTGR power plant that the SAMG can be simplified to a great extent or even unnecessary.

V. SUMMARY AND FURTHER WORK

Based on the design of the HTR-PM, the study of its typical DBAs and BDBAs has been carried out and introduced in this paper. The accident behaviors are summarized and the accident management of HTGR is discussed based on it:

(1) Adoption of the TRISO particle fuel element and reasonable design can guarantee the inherent safety feature of the HTGR.

(2) Accident analyses of the HTR-PM prove that the maximum fuel temperature and release of radioactive materials in all the DBAs can keep below the limitation.

(3) Even in the BDBAs with extremely low probability, there is enough time (more than several days) for the operators to execute mitigation measures.

(4) According to the inherent safety of the HTGR, it might be unnecessary to constitute a SAMG or only a simple SAMG is enough.

As one of the candidates for the Generation IV nuclear energy system, the study and discuss on the accident behavior, and especially the accident management are important and necessary for the further development of the HTGR. The work introduced in this paper can provide preliminary reference for the further study on the safety standards or guidelines for the design, operation and accident management of the HTGR. Further work, including more discussion and cooperation would also be expected.

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