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Advanced Reactor for Cogeneration of Heat & Electricity-R&D

Loading and boundary conditions, experience from licensing processes

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Summary

This document provides operating / transient conditions of steel pressure boundaries of recent modular HTR designs (HTR-Modul, PBMR 400, NGNP) and identifies fault conditions relevant for loads to the pressure boundary (e.g. RPV heat-up during pressurized-loss-of-forced-cooling (PLOFC) events, flow stratification in cross duct during natural circulation flow reversal, suppressed motion in supports of the pressure vessel system supports, etc). Furthermore it provides regulators??? views regarding the application of the Basis Safety respectively Incredibility of Failure concept to the primary pressure boundaries of the HTR-Modul and PBMR 400.

Approval

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1 Introduction

The international HTGR development started with the experimental and prototype reactors Dragon, AVR and Peach Bottom. Due to their small thermal power between 20 MW and 115 MW their reactor cores were small enough to fit into steel pressure vessels similar in size to the RPV of the Light Water Reactors at that time.

The next step in the development of HTGR was marked by the demonstration power plants THTR 300 and Fort St. Vrain (FSV) featuring thermal powers of 750 MW to 790 MW. The reactor cores of these power plants, a pebble bed in the case of THTR 300 and hexagonal blocks arranged in columns in the case of FSV, were more than 5 m in diameter, and together with the required graphite reflectors (typically 75 cm) and core shrouds did no longer fit into steel pressure vessels. Both reactors were therefore equipped with pre-stressed concrete reactor pressure vessels (PCRV). The reactors with their reflectors in core shrouds were located together with steam generators in central cavities of the PCRV. Besides the fact that PCRV could be built to the required large sizes, also arguments of incredibility of gross vessel failure were invoked for PCRV since the integrity of such vessels is ensured by a large number of individual pre-stressed steel cables, the status of which can be individually monitored. Provided a sound design of the pre-stressing system, a propagation of initial cracks to a catastrophic vessel failure can be excluded. A negative aspect of PRCV is, however, their long on-site fabrication time resulting in high costs.

Beginning in the early 1980's the HTGR development re-oriented again to small unit powers in order to make maximum use of inherent safety features of the HTGR, especially of their ability to remove the decay heat by passive means and without any mass flow of coolant through the core. Larger plant power output should be achieved by grouping several identical reactor modules together, hence the term "modular HTGR".

The core assemblies (core, reflector and shroud) of modular HTGR fitted again into steel pressure vessels of a diameter similar to that of Boiling Water Reactors. In order to benefit from the experience which had meanwhile been accumulated in design, fabrication and operation of steel RPV for LWR and to reduce erection time and costs by serial shop fabrication, the designers of modular HTGR switched again to steel RPV for their reactors. Only marginal effort was put in the investigation of pre-stressed concrete or cast-iron vessels for modular HTGR.

For steel RPV of LWR, significant work had been performed since the 1970's to rule out the possibility of catastrophic pressure vessel failure by developing Pressure Vessel Integrity Concepts. The "Basis Safety Concept" had been developed in Germany /1/, deterministically ruling out the possibility of catastrophic pressure vessel failure primarily by requirements on design and manufacture, reinforced by four redundancies: multiple party testing, "worst case" principle, in-service monitoring, and validation. A similar concept of "Incredibility of Failure" has been developed in the UK. Incredibility of failure is also implicitly contained in the French Regulations on RPV for LWR.

These concepts have, however, been developed for LWR and are to some degree specific to that type of reactors. It is the aim of ARCHER Work Package (WP) 25 to investigate the potential for application of "Basis Safety" or "Incredibility of Failure" concepts also to the RPV and main pressure piping of modular HTGR.

In the context of WP25 this deliverable (D25.11) shall set the scene by providing information about the operating / transient conditions of steel pressure boundaries of recent modular HTGR designs and identifying fault conditions relevant for loads to the pressure boundary. The experience with application of BSC to HTGR pressure boundaries in licensing processes will also be summarized in D25.11 and open issues will be identified.

The LWR Integrity Concept itself will be described in detail in another deliverable (D25.21). Based on this and on the information from this deliverable, a gap analysis paper will be elaborated in D25.21, which identifies the additional steps that are necessary for the application of the existing LWR-Basis Safety Concept (BSC) to HTR pressure boundaries taking into account the differences in the different pressure boundary designs.

Beginning with a brief discussion of the specific safety features of Modular HTGR in Section 2, this deliverable will describe the relevant types of modular HTGR and justify their choice in Section 3, before the proposed pressure boundary materials are listed in Section 4 and mechanical and thermal loads are discussed in Sections 5 and 6. Potential deterioration mechanisms are addressed in Section 7 and Section 8 provides information concerning the Regulators' position in past licensing processes for modular HTGR.

2 Specific Safety Features of Modular HTGR

In this Section those safety features of modular HTGR shall be discussed which are relevant for the integrity of their primary pressure boundary (PPB).

The most important safety feature in this respect is the ability of modular HTGR to dissipate their decay heat after shutdown out of the RPV into the Reactor Cavity solely by inherent physical mechanisms. In the Reactor Cavity it is removed by a dedicated Reactor Cavity Cooling System. It is worth noting that even this external cooling system appears not necessary to limit the maximum fuel temperatures, but has the purpose to protect the concrete walls from high temperatures. This behaviour has been demonstrated in analyses of Beyond Design Basis Accidents and will later be discussed in detail.

The inherent mechanisms of decay heat transport from the fuel elements in the core into the reactor cavity also work when the modular HTGR is depressurized after a leak or a break anywhere in the PPB. Since the coolant is a gas, a complete loss as in the case of water is not possible, just a reduction to ambient pressure. (Depending on the postulated leak/break sizes and locations, air or air-gas mixtures may enter the reactor and cause corrosion of the fuel elements. Such scenarios must be considered under the aspect of release of radioactivity, the ingress of air (or air-gas mixtures) does, however, not impede the inherent decay heat removal.)

Therefore the requirements on pressure vessel integrity are not the same for LWR and modular HTGR. While for LWR the lower part of the RPV must remain intact to a degree that the reactor core can be kept under coolant with only short term exceptions, this is not necessary for modular HTGR. Here the integrity of the RPV is only required to a degree that the core configuration (for which respective analyses have shown that the inherent heat removal is sufficient) is not impaired.

On the other hand, transferring the decay heat of a modular HTGR across the RPV walls into the reactor cavity usually means that the concerned parts of the RPV wall heat up from their operational temperatures. This will also be the case when the reactor is under operational pressure and significant natural circulation inside the RPV leads to an enhanced transport and redistribution of heat. Depending on the equipment (or not) of the reactor with auxiliary active decay heat removal loops, such RPV heat-up events may be operational transients or low probability design basis accidents.

Due to the high heat capacity of the core assembly and the low heat capacity of the coolant, temperature transients of the pressure boundary are low even at locations where (cold) coolant at operational mass velocity is in direct contact with walls of the pressure boundary. Thermal shocks similar to those that the RPV of PWR experience when in LOCA cases cold water from the dedicated pressurized replacement water tanks is pressed into the RPV, are not conceivable for the pressure boundaries of modular HTGR.

In the core of a modular HGTR, the coolant is heated up to hot gas temperatures which may range from 700 °C up to 950 °C. Streaks or jets of such hot gas could cause damage if impinging the walls of the pressure boundary in case of a leak/break of an internal hot gas duct. Therefore such events are generally excluded by design. The pressure boundary is always exposed to cold gas which is at a higher pressure than the hot gas. Any leak/break of a hot gas duct would lead to a leakage of cold gas into the hot gas stream, not the other direction.

3 Relevant Types of Modular HTGR

The development of modular HTGR started with the HTR-Modul (Siemens-Interatom design) in Germany /2/. This reactor was designed for generation of electricity by means of a secondary steam cycle and is chosen here as the first example because it stands for a relatively small unit power of 200 MW_{th} achieved with a cylindrical pebble bed core. This design is also very similar to another German design of the 1980s by HRB Company and to the Chinese HTR-PM /3/ which is currently under construction in China. Its main characteristics concerning the primary pressure boundary (PPB) are a cold gas temperature of 250 °C, a hot gas temperature of 700 °C and the use of 20 MnMoNi 5 5 as a vessel material which is the same material as for German LWR RPV.

Following the establishment of the HTR-Modul, HTGR designers worldwide worked to increase the unit power of a module and to increase the hot gas temperature (note that the AVR had already proven since 1974 that 950 °C were possible with an HTGR) for process heat application or generation of electricity with a gas turbine in a direct cycle. The PBMR 400 designed by PBMR Pty in South Africa is the most developed example for this class of modular HTGR /4/ and is therefore selected for further consideration in this report. It features an annular pebble bed core with a thermal unit power of 400 MW and power conversion to electricity by a gas turbine in a direct Brayton cycle. The hot gas temperature is 900 °C. The “cold” gas temperature returned from the recuperator of the power conversion unit to the reactor inlet is 490 °C. Gas with this temperature is, however, not in contact with the PPB, for details please refer to the successive subsection.

Pursuant to the Energy Policy Act of 2005, the NGNP project was initiated in the United States. The objective of this project is the demonstration of the viability of HTGR technology for the production of process heat, electricity, and hydrogen by the year 2021. The pre-conceptual design phase of the NGNP project was completed in 2007. Three reactor concepts have been evaluated in this phase /5/:

- Westinghouse/PBMR proposed a pebble bed annular core derived from the PBMR 400 with a thermal power of 500 MW and reactor inlet/outlet temperatures of 350/950 °C and a pressure vessel made of SA508/533 steel with the vessel walls exposed to the gas inlet temperature.
- AREVA proposed an annular block type core derived from its ANTARES design with a thermal power of 565 MW and reactor inlet/outlet temperatures of 500/900 °C and a pressure vessel made of 9Cr-1Mo steel with the vessel walls exposed to the gas inlet temperature, albeit with options to insulate the parts of the vessel not needed for passive decay heat removal or to apply an operational vessel cooling system.
- General Atomics (GA) proposed an annular block type core derived from its GT-MHR design with a thermal power of 550 to 600 MW and reactor inlet/outlet temperatures of 490/up to 950 °C and a pressure vessel made of 2-1/4Cr-1Mo (alternatively 9Cr-1Mo) steel with the vessel walls exposed to the gas inlet temperature.

Subsequent to the pre-conceptual phase, interaction with potential industrial end users and completion of trade studies led to a reduction of the aspired hot gas temperature in the NGNP project from 950 °C to 750 °C with an application to generate steam for process heat and electricity generation. The pre-conceptual designs were adjusted accordingly (including a reduction of the cold gas temperature), and both the AREVA and the GA design were converted to a “cold vessel” design with operational vessel cooling systems if necessary to allow for the use of SA508/533 as a vessel material /6/. A “hot vessel” design without active operational vessel cooling and the reactor power and inlet/outlet temperature conditions of the pre-conceptual concepts was only retained as an option for later application. In the further course of the NGNP project concepts with lower unit powers such as SG-MHR with 350 MW_{th} and derivatives of the HTR-Modul with 200 to 250 MW_{th}, both with a hot gas temperature of about 750 °C for steam generation were also discussed /7/. The latest published concept for NGNP was again from AREVA /8/ and featured a 625 MWth annular prismatic block core with reactor inlet/outlet temperatures of 325/750 °C, two steam generators and a “cold vessel” design with SA508/533 as the vessel material. This concept was selected by the NGNP Industry Alliance as their reference design concept /9/.

Since the thermal loads of all NGNP "cold vessel" concepts are limited by the allowable values of the ASME code and are, as far as they are known, covered by the loads of HTR-Modul and PBMR 400, these concepts are not considered in this deliverable. For information, the respective loads applicable to NCNP "hot vessel" concepts with vessels made of 9Cr-1Mo-V steel are provided below, acknowledging that since the definition of ARCHER task 2.5.1 these concepts have become an option for the further future.

3.1 HTR-Modul

The primary circuit of the HTR-Modul and its pressure boundary are shown in Figure 1. The reactor core assembly in the RPV is on the left side, the steam generator (SG) and the blower are visible on the right side. The RPV and the SG pressure vessel are connected by a short Cross Duct Vessel, and the fuel discharge device is connected to the bottom of the RPV. Basis Safety (incredibility of failure) arguments have been invoked in the licensing process for the entire assembly of RPV, SG-PV, Cross Duct Vessel and fuel discharge device.

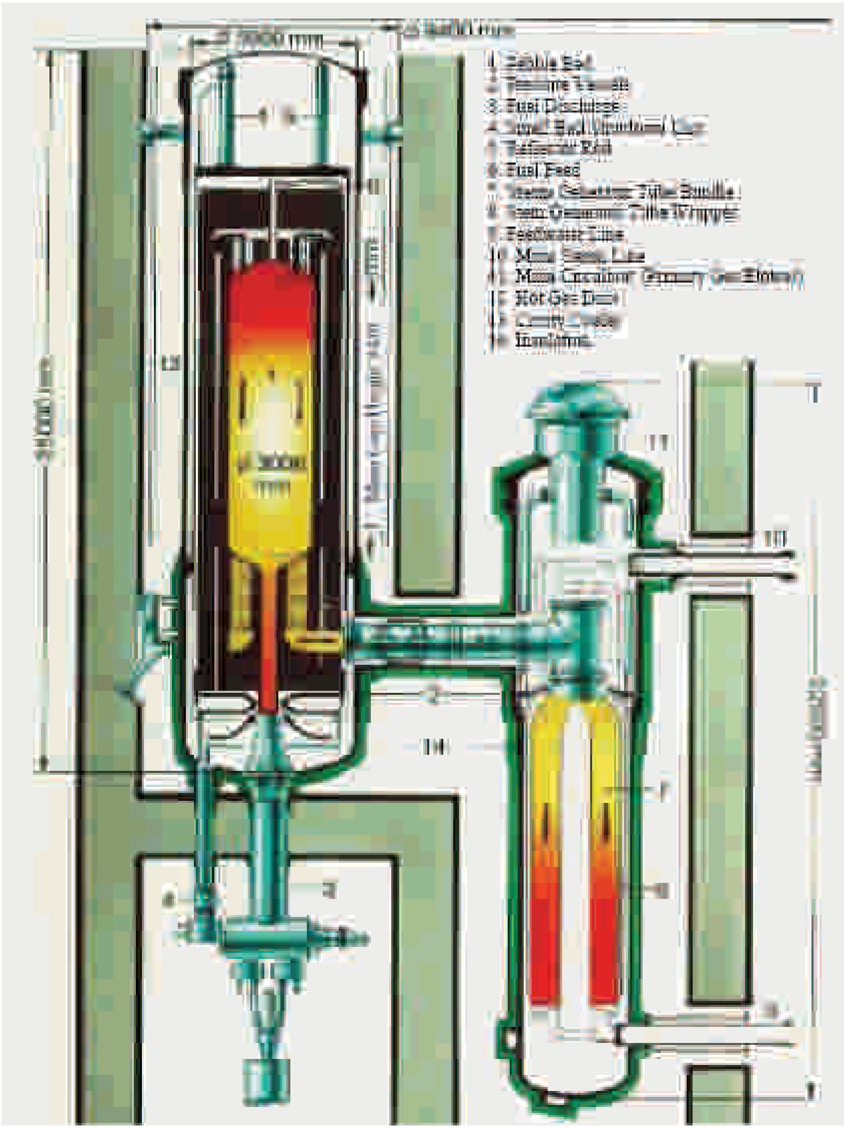

Figure 1: HTR-Modul Primary Circuit (reproduced from /7/)

The design of the PPB was such that the largest break to be assumed was equivalent to a DN65 pipe. Note that to achieve this goal, also the control rod drives were integrated into the RPV.

The blower transports cold gas at 250 °C from the upper plenum of the SG-PV through the annulus formed by the Hot Gas Duct and the Cross Duct Vessel to the bottom plenum of the RPV. After cooling the metallic core support structure, the cold gas flows upwards through channels in the graphitic side reflector to a plenum above the top reflector and turns downward. In the core it is heated up and flows through the bottom reflector and the Hot Gas Duct to the entrance of the steam generator, the hot gas temperature there being 700 °C. Flowing downwards through the SG bundle, it is cooled down to about 245 °C and directed upwards again to the blower.

Those parts of the PPB which are in direct contact with the forced mass flow of cold gas (SG-PV, Cross Duct Vessel, lower part of RPV) are insulated on the outside to reduce heat losses. The central and upper parts of the RPV are not insulated to allow the passive removal of decay heat to the Reactor Cavity Cooling System protecting the concrete walls of the reactor cavity.

The main supports of the pressure vessel unit are located in the mid-plane of the Cross Duct Vessel. Three supports fix the RPV vertically and horizontally with regard to its vertical axis. Two other supports in the same plane fix the SG-PV vertically (not visible in Figure 1), but allow for horizontal expansion of the vessel unit. Additional seismic supports are located at the upper part of the RPV and at the lower part of the SG-PV.

The main dimensions of the pressure vessel unit are:

RPV:	Di = 5.9 m; t = 118/ 250 mm; H = ~25 m
Cross Duct Vessel:	Di = 1.5 m; t = 50 mm; L = 2.9 m
SG-PV:	Di = 3.2/3.6 m ; t = 70/120/140 mm ; H = ~22 m
Live Steam Nozzle:	Di = 550 mm

3.2 PBMR 400

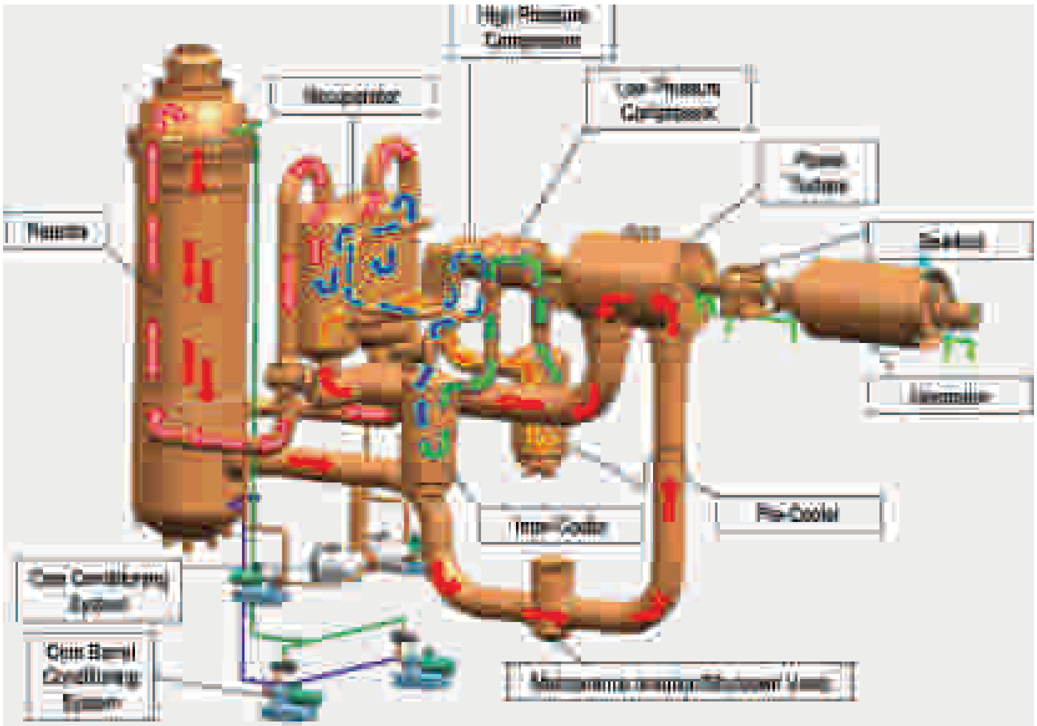

Figure 2: Primary Circuit of the PBMR 400

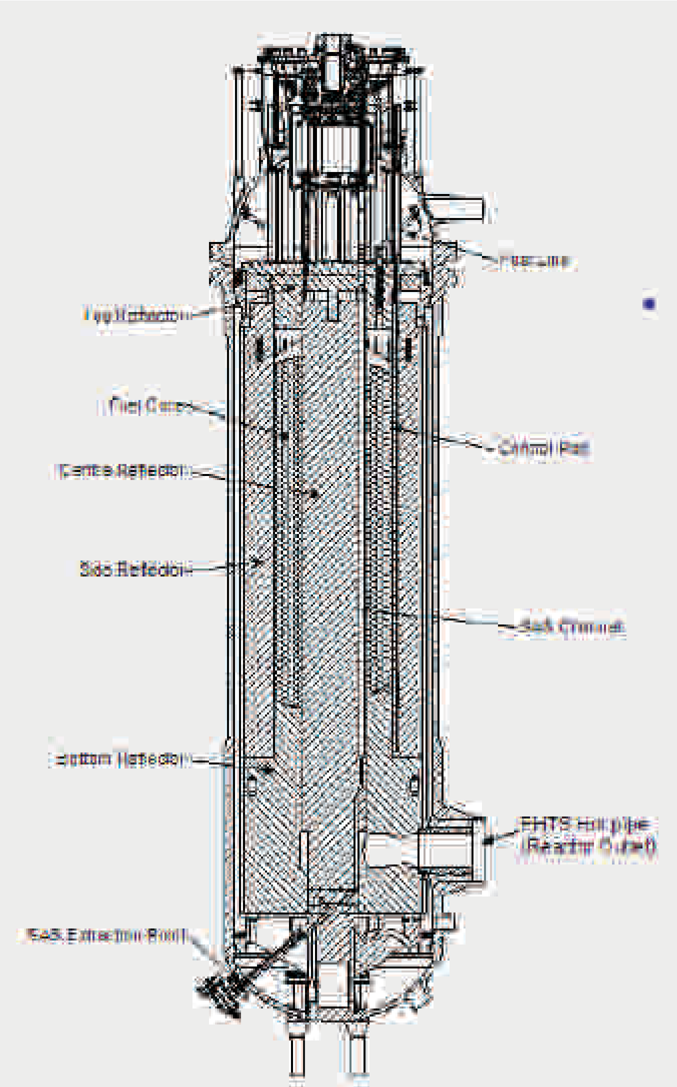

Figure 3: PBMR Core Assembly (reproduced from /5/)

Figure 2 shows a view of the primary circuit of the PBMR 400, Figure 3 allows a look into the core assembly.

The PBMR 400 was designed for the production of electricity by means of a gas turbine in a direct Brayton cycle. Therefore the primary circuit had to comprise the reactor itself, the gas power turbine connected to the generator, a recuperator to utilize the heat still contained in the primary coolant at the exit of the power turbine, and a two-stage compressor with pre- and intercooling of the primary gas.

The primary coolant is heated up in the pebble bed and collected at about 900 °C and about 85 bar in an annular hot gas plenum below the bottom reflector. From there it is directed through the inner duct of a coaxial, curved hot gas line to the power turbine where it expands to about 30 bar and about 500 °C. This coolant flows further to a twin recuperator where it is cooled down below 150 °C by the high pressure gas on its way to the reactor core. From the recuperator the cold gas is directed to the water cooled pre-cooler and further on to the low pressure compressor which is driven by the power turbine, then to the water cooled intercooler and to the high pressure compressor. The primary coolant leaves the high pressure compressor at about 90 bar and 100 °C and is directed to the high pressure sides of the recuperators. There it is heated up to about 500 °C. From the recuperators it flows to the RPV through two parallel “cold” gas lines and is directed to an annular cold gas room below the side reflector. Through riser channels in the side reflector it flows upwards to the top of the core and is heated up again on its way downward through the core.

The brief description shows that a Brayton cycle has more temperature and pressure levels of the primary coolant than a steam generating Rankine cycle. The “cold” gas flowing back from the

Power Conversion System to the RPV is far away from the lowest temperature in the cycle and would not allow a transport in the same coaxial duct as in the HTR-Modul with the PBMR "cold" gas of about 500 °C (HTR-Modul: 250 °C) in contact with the pressure boundary made of SA508/533 steel. Therefore also the "cold" gas is conducted in the insulated inner duct of a coaxial gas duct. The annular gaps in both the reactor outlet and the reactor inlet coaxial pipes are flushed with high pressure cold gas from the exit of the high pressure compressor. Thus the design principle to have the primary pressure protected from hot gas by cold gas at higher pressure (allowing only in-leakage) is realized. The PPB of the Power Conversion System is therefore at an operational temperature below 150 °C. During normal operation the walls of the RPV are kept at about 300 °C by a dedicated system actively cooling the gap between Core Structure Assembly and RPV.

Basis Safety or Incredibility of Failure arguments have been invoked for the RPV and the other vessels of the PBMR primary circuit. The connecting lines were considered as piping with Leak-before-Break arguments invoked for the larger piping.

The main dimensions of the PBMR 400 pressure vessels and the hot gas resp. "cold" gas pipes are:

RPV:	Di = 6.2 m; t = 180/ 285 mm; H = ~28.5 m Outlet nozzle: Di = 1950 mm (= hot gas pipe) Inlet nozzles: Di = 1120 mm (= cold gas pipe)
Recuperator vessel:	Di = 3.8 m; t = 110 mm
Pre-/ Intercooler vessels :	Di = 3.1 m; t = 100 mm

3.3 HTGR for the NGNP Project

As justified above, the "hot vessel" concepts developed in the pre-conceptual phase of the NGNP project shall be presented in this deliverable because they would impose additional challenges on a vessel integrity concept for HTGR.

Figure 4 shows the primary circuit of the General Atomics (GA) design. The block type annular core is located in the RPV in the middle of the figure. The GA design is based on the GT-MHR. The power conversion system with a direct cycle gas turbine is retained and supplemented by a parallel cycle with an Intermediate (He/He) Heat Exchanger (IHX). Contrary to the multi-vessel design of the PBMR 400, all components of the gas turbine cycle (power turbine, generator, compressors, coolers and recuperator) are integrated in one pressure vessel shown on the left side of Figure 4. A separate pressure vessel houses the plate type IHX and a blower as depicted on the right side of Figure 4. Both vessels of the power conversion system are connected to the RPV by short cross-vessels containing coaxial hot gas ducts. Cold gas was to be returned to the Reactor through the annuli formed by the cross-vessels and the hot gas ducts.

The pre-conceptual design by AREVA was similar in its layout of the PPB with a central RPV and four satellite pressure vessels connected to it by short ducts. The satellite vessels contained IHX (three tubular types similar to those which had been developed in Germany in the 1980s and one compact plate type) and blowers. No direct cycle gas turbine was planned. 9Cr-1Mo steel was foreseen for the RPV.

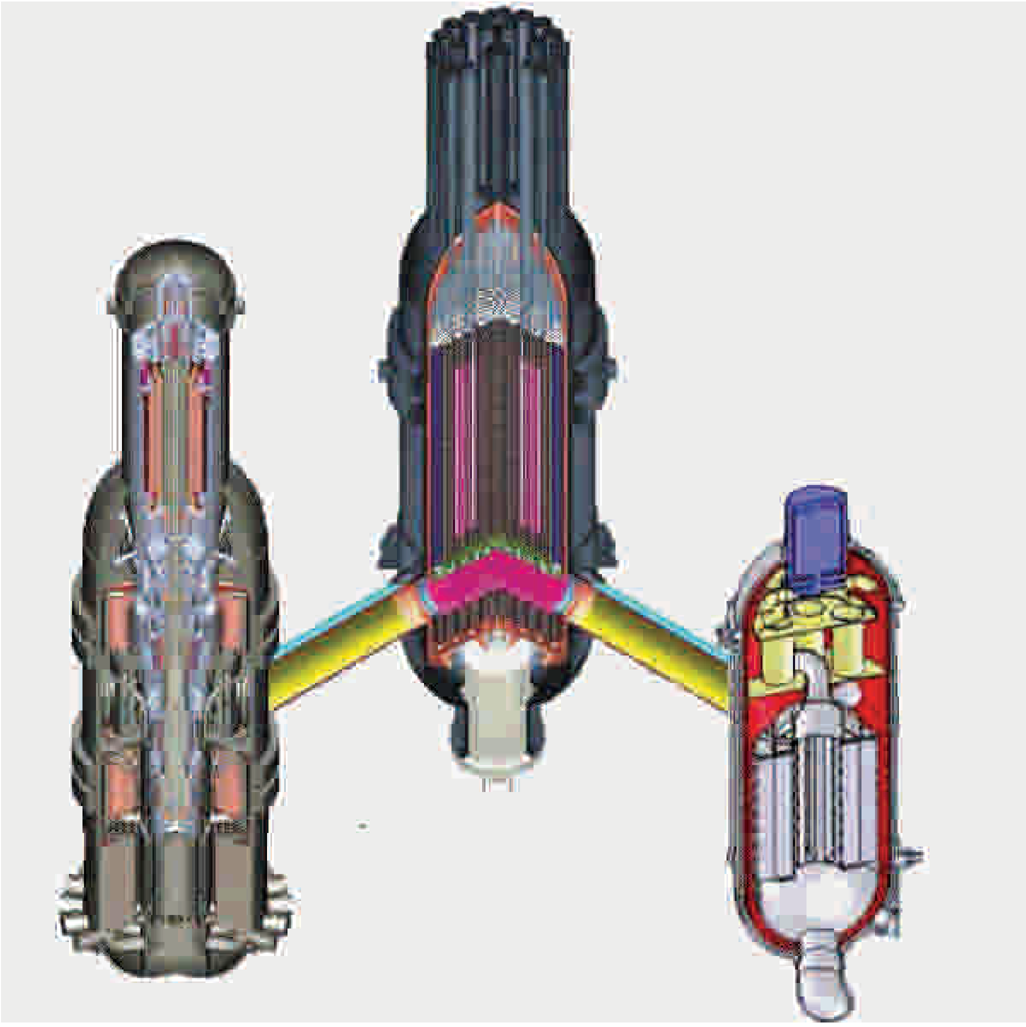

Figure 4 Primary Circuit of the General Atomics pre-conceptual NGNP design (reproduced from /5/)

According to /5/, the GA design has a power level of 550 – 600 $MW_{th,}$ a reactor inlet temperature of 490 °C and a reactor outlet temperature of up to 950 °C. 2-1/4Cr – 1Mo and 9Cr – 1Mo steel are stated as materials for the RPV (and the cross-vessels) in the executive summary of /5/. Parts of the RPV wall and the walls of the cross-vessels are in contact with the primary cold gas of 490 °C. According to /5/ it was not clear at that stage whether the cross-vessels would be allowed to be defined as vessels rather than pipes.

The main dimensions of the RPV as reported in Appendix K of /5/ are:

Di ~ 7,200 mm, t ~ 215/260 mm, Hi ~ 23,800 mm

4 Primary Pressure Boundary (PPB) Materials

The design approach for both the HTR-Modul and the PBMR 400 was to make maximum use of the experience that had been gathered over decades with established RPV materials for LWR, whereas in the NGNP pre-conceptual design phase AREVA and GA chose to cope with the higher cold gas temperatures of their designs by changing over to materials with higher allowable temperatures. As already stated in Section 3, the further development of the NGNP project led back to lower system temperatures and the use of LWR material, however, the choices for the pre-conceptual design phase shall be reported here as an option for the more remote future. In detail the proposed materials were:

<u>HTR-Modul:</u>

For the RPV, the Cross Duct Vessel and the SG-PV including its feed water nozzle the German RPV steel 20 MnMoNi 5 5 according to KTA specifications was chosen.

Other than in a Light Water Reactor, life steam at 530 °C was to be produced in the SG-PV and guided to the turbine. Since this temperature is outside the allowable range of the vessel material, a life steam nozzle made of X20 CrMoV 12 1 and 10 CrMo 9 19 with a thermo-sleeve made of X10 NiCrAlTi 32 20 (Incoloy 800) was foreseen.

<u>PBMR 400:</u>

The entire PPB was planned to be made of ASME SA-508 Grade 3 Cl 1 for forgings and SA-533 Type B Cl 1 for plates. In order to comply with the requirements of the Basis Safety Concept, restrictions to the chemical composition as compared to ASME SA-508/-533 had been considered making the chemical composition very similar to the German 20 MnMoNi 5 5 (or to the French 16 MnD 5). No high temperature resistant nozzles are required in the PBMR 400 design. The only connections to secondary systems are the pipes for water supply of the pre-cooler and the intermediate cooler which work at low temperatures.

<u>NGNP (pre-conceptual design):</u>

In Appendix K of /5/, GA preferred the use of 2¼Cr-1Mo steel instead of 9Cr-1Mo-V which had been proposed for the predecessor GT-MHR. GA argued with welding difficulties to be expected and lack of manufacturing and operating experience with 9Cr-1Mo-V steel. In the main body of /5/, however, 2¼Cr-1Mo was ruled out because of its low allowable stresses at the required temperatures which would lead to large wall thicknesses. Instead 9Cr-1Mo-V steel was proposed for the “hot vessel” option. Notwithstanding this proposal the challenges coming along with the use of 9Cr-1Mo-V steel were identified: lack of experience with forging and welding in the sizes required, issues post weld heat treatment and through thickness properties.

5 Mechanical Loads

5.1 Pressure

5.1.1 Operational Pressure

The primary circuit system pressures in the considered modular HTGR concepts are similar to the pressure in Boiling Water Reactors (BWR). Values of the pressures acting on the primary boundaries are given in Table 5.1-1. For the HTR-Modul, values of the water/steam circuit are also given to support an evaluation of the feed water and live steam nozzles at the primary pressure boundary (PPB).

All pressures in bar	HTR-Modul	PBMR 400	NGNP (GA)
Primary Circuit			
Operating pressure	61	90	~70
Design pressure	70	97	-
Secondary Circuit			
Feed water line			
Operating pressure	210	Water coolers are below 5 bar	-
Design pressure	240		-
Live steam line			
Operating pressure	190	-	-
Design pressure	208	-	-

Table 5.1-1 Operational Pressures of considered HTGR Designs

5.1.2 Pressure Transients

Reactors combined with a steam generator or an intermediate heat exchanger (IHX) would mainly be operated at a constant primary pressure (e.g. HTR Modul). Pressure transients due to deviations from the operational steady state would be low due the high thermal inertia of the system. Depressurisation rates in the course of LOCA events would depend on the maximum break size to be considered. For the HTR-Modul, this was the sudden break of a connecting DN 65 line which would lead to a complete depressurisation within a few minutes.

Reactors with a gas turbine in the primary circuit realising a Brayton cycle (e.g. PBMR 400, NGNP GA proposal) would have two main pressure levels in the primary circuit: high pressure (PBMR: ~90 bar) between compressors and turbine, low pressure (PBMR ~30 bar) between turbine and compressors. Any occasional collapse of the Brayton cycle would result in a rapid pressure equalisation (to ~60 bar in PBMR 400). In addition, the power output of the Brayton cycle is controlled in the PBMR 400 by adjusting the Helium inventory and thus the system pressure with a low rate of change.

As with systems with a steam generator or an IHX, depressurisation rates would depend on the break sizes to be assumed. For the complex primary circuit of the PBMR 400 with pipes of various diameters connecting to the pressure vessels, a sudden break of DN 250 had been assumed which would lead to a depressurisation within about one minute.

No detailed information is available from the NGNP project. The GA pre-conceptual design features both a direct cycle gas turbine loop and an IHX loop, so the pressure transients discussed above for the PBMR 400 would also occur there. Depressurization rates in the aftermath of a leak or break would depend on the definition of “cross-vessels” as vessels or as piping (eventually with leak-before-break arguments).

5.2 Restraints

Thermal restraints may occur when the supports of the PPB are not adequately designed for the thermal expansions occurring during normal operation and accidents. In the 3-vessel-design of the HTR-Modul and the similar NGNP designs of AREVA and GA the cross ducts and their connections to the main vessels may be affected, in the PBMR 400 design thermal expansions have to be taken by the long and generally curved piping between the vessels. Inadmissible thermal restraints of the described kind should be excluded by design.

Differential thermal expansion may also develop along the circumference of the cross duct vessels of the HTR-Modul or the similar NGNP designs during pressurised passive heat removal events if the isolation of the RPV from the power conversion fails (failure to close primary circuit shut off valves). In this case natural circulation may develop not only in the core and the ceramic core structure, but also in the affected primary circuit with gas at elevated temperatures flowing through the annular cold gas path of the cross duct vessel. A thermal stratification of the flow may develop in the annulus leading to different heating over the circumference and consequential bowing of the cross duct vessel (banana effect). In the design of the HTR-Modul, this case has been considered by arranging the steam generator well below the core level. Due to this arrangement, an initially reversed natural circulation through the primary circuit would predictably decrease to negligible values after some time.

Mechanical restraints might also occur in case of a mechanical failure of seismic shock absorbers in the support system. Such events need to be analysed and include in the load collective of the PPB.

5.3 Hydraulic Loads

The PPB of LWR may be exposed to hydraulic loads such as water hammers. Owing to the single phase, gaseous nature of the primary coolant Helium, such loads can be excluded for HTGR.

5.4 Other Mechanical Loads

Other mechanical loads from internal or external events should be similar for LWR and HTGR. An exception may be the existence of fast rotating machinery in direct cycle gas turbine applications. The consequences of mechanical disintegration of these components must be contained. This issue exists to some degree also in LWR (main coolant pumps) and other HTGR applications (primary circuit blowers), but appears to be more serious in direct gas turbine applications due to higher rotating masses at higher speeds.

6 Thermal Loads

6.1 Design Temperatures, operating Temperatures

For the HTR-Modul, the design temperature quoted below has been determined in the concept licensing process. According to the German rules and standards for RPV (KTA-3101.1 and VdTÜV-Werkstoffblatt 401/3) the allowable temperature for 20 MnMoNi 5 5 is 375 °C.

The design temperature for the PBMR 400 material (ASME SA-508/SA-533) corresponds to the allowable temperature of that material for unlimited operation.

The design temperature for the NGNP "hot vessel" option with 9Cr-1Mo-V appears to be under discussion within the project. According to ASME III, Subsection NH (as quoted in Appendix B of /6/) allowable stress values exist up to 649 °C and fatigue curves up to 538 °C. However, it is stated in appendix B-2.1 of /6/ that "the maximum design temperature will be determined by the need to avoid the thermal creep regime, which is about 425 °C for the Cr-Mo steels in the hot vessel version". This would mean that the pre-conceptual "hot vessel" design would either have to work in the thermal creep range or would have to undergo design changes to lower the operational temperatures of the PPB. At least direct contact with 490 °C coolant would have to be avoided.

Table 6.1-1 shows the design temperatures and operating temperatures of the considered HTGR designs.

All Temperatures in °C	HTR-Modul	PBMR 400	NGNP (9Cr-1Mo-V)
Primary Pressure Boundary (PPB):			
Design temperature	**350**	**371**	**see above**
Operating temperatures:			
In contact with forced primary mass flow (main gas flow or dedicated bypass)	~ 250 (RPV bottom, SG-PV, Cross Duct Vessel)	< 150 (PCU) < 25 (piping between coolers & compress.)	490 (author's estimate: cold gas temperature)
RPV at position of max. irradiation	> 150 < 200 (annular gap with stagnant Helium inside RPV)	280-300 (achieved by dedicated cooling system, CBCS)	440 (/6/, Appendix B-1.2.1)
RPV top	< 150	280-300	-

Live steam line:			
Design temperature	**540**	-	-
Operating temperature	530	-	-

Table 6.1-1 Design Temperatures and Operating Temperatures of considered HTGR Designs

Operational temperature gradients parallel to the axis of vessels or pipes may have to be considered. The most prominent location in the HTR-Modul is the transition from the live steam line to the SG-PV where a thermo-sleeve had been foreseen as a design measure. Another location is the transition between the insulated lower part of the RPV and the cylindrical part above. The lower part is at cold gas temperature of 250 °C, the part above, in contact with stagnant helium and facing the Reactor Cavity Cooling System is at about 200 °C.

Somewhat higher temperature differences will be observed in the PBMR 400 at the connections of the Hot Gas Pipe and the “Cold” Gas Pipes to the RPV. The Gas Pipes are cooled to about 150 °C by cold gas from the High Pressure Compressor outlet, the RPV is at about 300 °C.

There is no information available about comparable locations in the NGNP designs.

6.2 Temperature Transients

In this section firstly operational temperature transients are presented, followed by temperature transients in the course of Design Basis Accidents (DBA). As already discussed above, HTGR when compared to LWR are generally characterized by slower transients in the PPB walls for two reasons: the heat transfer from the coolant to the PPB walls is lower for gaseous coolants than for water, and those parts of the PPB close to the active reactor core are in contact either with stagnant Helium (HTR-Modul) or with a small dedicated mass flow of Helium (PBMR 400). On the other hand, the gas temperatures in an HTR may fluctuate more than the water temperatures in a LWR. As an example, the reactor scram limit value of the HTR-Modul cold gas temperature was set to 280 °C i.e. 30 K above the nominal cold gas temperature. Occurrences like a loss of feed water preheating may cause cold gas temperature decreases of similar magnitude.

The HTR-Modul was deliberately designed without any active decay heat removal systems. Decay heat should primarily be removed via the operational start-up / shut-down system. If this system was not available, the decay heat would be removed in passive mode to the cooling system protecting the reactor cavity wall. The pressurized loss of forced cooling was therefore considered as an anticipated operational occurrence.

The PBMR 400 was designed with a two-fold redundant active decay heat removal system. Therefore passive decay heat removal is a design basis event for the PBMR 400. Similarly, the NGNP “hot vessel” designs were equipped with one active decay heat removal system.

As regards design basis events, an analogue to the cold water shock during PWR LOCAs, caused by the insertion of cold water by the Emergency Core Cooling System, does not exist with HTGR. The most significant temperature transients are observed in the passive decay heat removal mode. In such cases, owing to its small unit power, the maximum RPV temperature of the HTR-Modul remains slightly below the design temperature of 350 °C. The PBMR 400 with its double unit power, but similar geometry of the RPV and the reactor cavity will experience significantly higher RPV temperatures above the design limit of the ASME material SA-508 resp. SA-533 (371 °C). The designer has therefore invoked Code Case N-499 which allows for higher RPV temperatures for limited time periods (up to 538 °C for a maximum of three events and a cumulative period less than 1000 h). In addition, the designer has categorized thermal loads from passive heat removal events as ASME service level D loads. The non-availability on demand of the redundant active decay heat removal system was targeted to 10^{-4}/a, making events with passive decay heat removal low frequency DBA under South African regulation (range 10^{-4}/a - 10^{-6}/a for low frequency DBA).

For the NGNP "hot vessel" designs a maximum admissible temperature in passive decay heat removal modes of 650 °C is given in /10/ which coincides with the maximum temperature for which allowable stress values are available.

Pressure boundary temperature transients are generally low in the passive decay heat removal mode. Starting from operational conditions, the temperatures may even decrease slightly in the first hours of the event, and then reach the maximum values reported below in about three to four days before they start to decrease again even more slowly. As long as the Reactor Cavity Cooling System is in operation, the RPV wall will remain at the maximum temperatures reported below only for limited periods of time because the decay heat will be continuously decreasing.

Table 6.2-1 provides the maximum transient PPB temperatures and upper limits for temperature change rates.

	HTR-Modul	PBMR 400	NGNP
Operational transients:			
Increase of cold gas temperature	< 280 °C (Criterion for interruption of forced flow by Protection System)	-	-
Decrease of cold gas temperature	-	-	-
Heating/cool-down rates during cold start-up/cool-down:			
RPV, axial middle of cyl. part	< 15 K/h	-	-
RPV bottom, CD-PV, SG-PV	< 65 K/h	-	-
any part of pressure boundary during any transient	-	< 100 K/h	-
Passive decay heat removal from pressurised reactor Location: cylindrical part of RPV, in height of upper core	< 320 °C	not applicable here, DBA	not applicable here, DBA
Design Basis Accidents:			
Allowable Temperatures (see above)	**375 °C**	**538 °C**	**650 °C**
Passive decay heat removal after LOCA, reactor depressurised Location: cylindrical part of RPV, in height of core middle	< 350 °C	~435 °C	582 °C (Appendix K to /5/)

Passive decay heat removal from pressurised reactor after loss of active decay heat removal systems Location: cylindrical part of RPV, in height of upper core	n/a, passive DHR is operational transient, see above	~515 °C	~565 °C (Appendix K to /5/, extrapolated)

Table 6.2-1 Maximum Transient Temperatures and Temperature Change Rates of considered HTGR Designs

The levels of conservatism in the temperatures reported above are different. The results for the HTR-Modul are explicitly characterized as conservative, whereas the results for the PBMR are best estimate and for NGNP the status is undetermined.

From parameter variations performed in the past it is known that the most significant parameters for the RPV temperatures in passive heat removal mode (DLOFC) are the decay heat to be removed and the material properties of components adjacent to the RPV in the heat transfer path, especially the emissivity coefficient for thermal radiation.

Effect of margins on the decay heat production:

The results for the HTR-Modul contain a 5% margin on the decay heat production rate. Parameter studies for NGNP designs have been published in /12/ and /13/. The results indicate that a 5% margin on the decay heat production rate would increase the maximum RPV temperature of a 600 MWth reactor under DLOFC conditions by about 12 K.

Effect of margins on the emissivity for thermal radiation:

At the RPV temperatures which are presented above for DLOFC conditions, the total heat transfer between the core assembly, the RPV and the cavity cooling system is dominated by thermal radiation. This is the reason why the emissivity coefficient of the involved metallic surfaces is influential. The heat transfer rate increase with increasing emissivity coefficient. For the HTR-Modul, an emissivity coefficient of 0.6 had been assumed as a conservatively low value for oxidized steel surfaces. For the PBMR 400 and the NGNP designs a value of 0.8 has usually been used. In /12/ the emissivity coefficient has been varied from 0.8 to 0.6 resulting in an increase of the maximum RPV temperature by about 50 K, albeit at a level well below “official” values reported in /5/. It is also not clear whether the emissivity coefficient has been varied only at the surfaces outside the primary circuit (RPV and cavity cooling system, as assumed by the author of this deliverable) or also at the surfaces inside the RPV (RPV and Core Barrel). Since the surfaces inside the RPV are in a reducing Helium atmosphere, it should be more difficult to maintain a high emissivity coefficient there than outside the RPV. Should the emissivity coefficient decrease over lifetime inside the RPV and outside not, this would lead to lower maximum RPV temperatures. Therefore the author of this deliverable tends not to add margins for eventually lower emissivity coefficients.

Effect of the surface temperature of the cavity cooling system:

For the NGNP designs different cavity cooling systems have been proposed: water cooled in forced convection or natural convection mode, and air cooled with direct connection to the environs. Water cooled systems are reported with surface temperatures between 40 °C and 65 °C and 140 °C in boiling off mode. Air cooled systems are also supposed to operate at about 140 °C /13/. While /12/ and /13/ find only a limited increase of the maximum RPV temperature of about 10 K when increasing the cavity cooling system wall to 140 °C, this effect is as large as 60 K in /14/. Based on his own experience the author would favour the low effect on the RPV temperature because radiative heat transfer is dominated by the upper temperature, not by the lower. In addition, air cooled cavity cooling systems using ambient appear not licensable in Europe because

Argon in the air is activated when conducted as closely to the operating reactor. Such systems are not likely to comply with the ALARA (As Low As Reasonably Achievable) principle.

Conclusive proposal for margins on the maximum RPV temperatures:

Looking at the discussion above and at the fact that the temperatures reported in the table above for PBMR 400 and NGNP are the highest that he found, the author proposes to use the tabulated values and add no additional margins to them.

Maximum RPV temperatures in hypothetical events

The decrease of the decay heat with time results in some limitation of the RPV temperatures even in the Beyond Design Basis Event of a long-term loss of all reactor cavity cooling. Scoping analysis of such an event has been performed for the HTR-Modul and for the PBMR 400 /11/. In such a hypothetical event the maximum fuel temperature is not influenced significantly because it is reached well before the heat-up of the outer side reflector, the RPV and the concrete of the reactor cavity impairs the conditions for heat transfer. However, the subsequent decrease of the fuel temperatures is much slower than in the DBA case with operational Reactor Cavity Cooling System. With the heat capacity of the concrete walls being the only heat sink for a long time, the RPV will heat up to about 600 °C within two weeks at the height of the active core before it cools down again very slowly with the decreasing decay heat. Also the concrete will reach similar temperatures at the same elevation, but it will stay at 300 °C to 450 °C at the location of the RPV supports. The scoping analysis of the past should be taken as a first indication that even in such hypothetical cases the integrity of the RPV is not necessarily lost. The available results for PBMR and HTR-Modul which show a slightly higher maximum RPV temperature for the HTR-Modul despite its lower decay heat raise, however, questions regarding potentially different levels of conservatism in the particular analyses. More detailed analyses should be performed when relevant design detail is available for future modular HTGR applications. In any case it appears recommendable to equip the primary circuit such that it can be depressurized as an Accident Management Measure to eliminate the pressure load in such a hypothetical case.

7 Potential Deterioration Mechanisms

7.1 Irradiation with Fast Neutrons

Irradiation with fast neutrons (E > 1 MeV) can cause embrittlement of the pressure boundary material. Usually it is assumed that irradiation damage in pressure vessel steel needs to be considered for fast neutron fluences above 10^{18} neutrons/cm². German guidelines for RPV of PWR require a limitation of the fast neutron fluence to 10^{19} neutrons/cm².

Besides the level of the neutron fluence, the material temperature at which the fluence is applied, i.e. the irradiation temperature, influences the level of irradiation damage. The irradiation damage due to a given neutron fluence decreases with increasing irradiation temperature. Vast experience exists for LWR RPV steel at about 300 °C.

In the design of the core structures of the HTR-Modul, a boration of the carbon brick layers surrounding the graphite reflectors was implemented in order to limit the irradiation effects on the metallic core structures and the RPV. As a result, the maximum fast neutron (E > 1 MeV) flux at the inner surface of the RPV was calculated to about 10^{9} neutrons/cm²/s at rated reactor power. Over 32 full power years (equivalent to 40 calendar year) this would lead to a fluence of about 10^{18} neutrons/cm². The irradiation temperature at this range of the RPV would be above 150 °C and below 200 °C.

The PBMR 400 has no explicitly boronized outer layer of its ceramic core structure and has a higher unit power. Both effects lead to a higher fluence of fast neutrons at the RPV wall. According to preliminary analyses a value of 10^{19} neutrons/cm² will not be exceeded after 40 calendar years. The corresponding irradiation temperature would be controlled by the Core Barrel Conditioning System to lie between 280 °C and 300 °C.

No values are available from the NGNP project.

7.2 Corrosion

As described in the previous chapters, the PPB is in direct contact with the reactor coolant Helium. No cladding or liner was planned in the various reactor concepts. Depending on the reactor concept and the location inside the PPB, the Helium may touch the walls of the PPB either in forced convection mode, in natural convection mode or as a stagnant fluid. Similar to the irradiation with fast neutrons, potential corrosion is considered a long term effect occurring during normal operation, if at all. Corrosion would then be induced by the operational impurities of the coolant.

During normal operation, the impurities of the reactor coolant are limited by the Helium purification system. For the HTR-Modul impurity values similar to those observed in the AVR and in the Peach Bottom reactor have been assumed in the Concept Safety Analysis Report.

The impurity values assumed in the PBMR project were similar in general, albeit lower for some species.

No values are available from the NGNP project.

In addition to gaseous impurities, the reactor coolant will be loaded with carbonaceous dust. For this impurity, values have been reported for the AVR in /15/.

Table 7.2-1 provides an overview of the reactor coolant impurities assumed for the HTR-Modul.

Impurity	Limit value
H_2O	< 0.1 vpm
H_2	< 5 vpm

CO	< 2 vpm
CO_2	< 0.5 vpm
CH_4	< 1vpm
N_2	< 1 vpm
carb. dust /15/	5 – 50 µg/m³

Table 7.2-1 Specified Coolant Impurities of considered HTGR Designs

In the NGNP project, as a state of the art in 2010, data for oxidation and corrosion of RPV material in a Helium atmosphere as characterized above are assumed to be very limited /6/ and further investigations are deemed necessary.

During design basis accidents (LOCA, SG tube rupture) these concentrations may significantly increase for a limited period of time, until the Helium purification system has reduced them again to the specified values. In case of a large break LOCA which was categorized as a beyond design basis event for the HTR-Module, part or all of the primary coolant might be replaced by air or by flue gas from the reaction of the reactor graphite with air. However, this gas composition is only acting for the limited period of the accident and the system is depressurised in this case. The issue of potential deterioration of the material impeding the restart of the reactor after recovery from accidents is beyond the scope of this work package.

7.3 Erosion/Abrasion

Erosion and abrasion can be caused by the dust laden Helium coolant at surfaces which are in direct contact with the primary mass flow at velocities in the range of 50 m/s. As with corrosion due to impure Helium, no data base is known regarding the abrasive effect of Helium with carbonaceous dust in the micro-meter range and the concentrations quoted above.

8 Regulators’ Position in past Licensing Processes

Licensing processes of different purposes have been initiated and terminated for the HTR-Modul and the PBMR 400. The experience from that processes relevant for the topic of this deliverable is summarised below. No respective information was found for the NGNP project. The Regulator in charge, the US NRC is currently in the process of preparing for a future NGNP licensing process.

8.1 HTR-Modul

For the HTR-Modul a site-independent Concept Licensing Process was launched in the state of Lower Saxony in 1987. The German TSO TÜV Hannover (with TÜV Rheinland as a subcontractor) was commissioned with the evaluation of the safety case. The licensing process was stopped two years later for political reasons. Since the evaluation process was well advanced at that time, the Federal Ministry of Research and Technology funded its completion as a R&D project. This project was successfully completed in 1989.

With regard to the application of the Basis Safety Concept to the pressure vessels of the HTR-Modul the safety case evaluation yielded the following results:

- The Basis Safety Concept was assessed to be applicable to the HTR-Modul in general
- For the Cross Duct Vessel this assessment could only be finalised after a specific site with specific seismic loads would be defined
- An open issue was seen in the thermal sleeve nozzle connecting the live steam line to the SG vessel: regarding material and temperature this component is outside the LWR experience
- The reviewers also noted a lack of details regarding blower rotor failure

8.2 PBMR 400

A combined Site and Construction Licensing Process was launched for the PBMR 400 in South Africa in 2004. The South African Regulator contracted the British company AMEC and the German TÜV Rheinland as consultants for this process. The licensing process was stopped in 2010 for financial reasons before the safety case and the collective load case of the primary pressure boundary was finalised. The production of the RPV had, however, already been started at the risk of the license applicant as a Long Lead Item activity.

With regard to the application of the Basis Safety or Incredibility of Failure Concept to the primary pressure vessels of the PBMR 400 licensing process yielded the following results:

- Basis Safety or Incredibility of Failure was assessed as generally achievable for the RPV and other pressure vessels
- The design of the RPV demanded a site welding of one big circumferential weld. The consequences of this feature for applicability of Basis Safety have not been finally assessed.
- Depending on its size, leak-before-break arguments rather than Incredibility of Failure were deemed applicable for the connecting piping. For small size piping the consideration of sudden breaks was deemed inevitable.
- A final evaluation was not possible because both Safety Case collective load cases of the pressure vessels were not finalised

- The Basis Safety requirements of the German KTA rules on material testing could not be fulfilled literally due to a lack of sample material as a consequence of premature material orders.

The complex rotating machinery in the primary circuit was seen as a major obstacle to assign Basis Safety. Satisfying analysis of bounding loads due to disintegration of turbine or compressors was not presented during the licensing process.

9 Conclusion

In support of ARCHER Task 2.5.2 to identify gaps to be closed for the application of the existing LWR Basis Safety Concept to HTGR steel pressure boundaries, the operating / transient conditions of pressure boundaries of recent modular HTR designs (HTR-Modul, PBMR 400, NGNP pre-conceptual designs) have been provided. Fault conditions relevant for loads to the pressure boundary (e.g. RPV heat-up during pressurized-loss-of-forced-cooling (PLOFC) events, flow stratification in cross duct during natural circulation flow reversal, suppressed motion in supports of the pressure vessel system supports) have been discussed. Furthermore, regulators' views regarding the application of the Basis Safety respectively Incredibility of Failure concepts to the primary pressure boundaries of the HTR-Modul and PBMR 400 have been provided.

The conditions reported for the HTR-Modul (200 MW_{th}) will similarly apply to the Chinese HTR-PM currently under construction and to a European HTR Demonstrator still to be defined. Compared to existing LWR (BWR) experience, these conditions are generally characterized by the choice of the same material, similar temperature and pressure loads, less severe thermal transients and different coolant, i.e. Helium with very low impurities of oxidizing or carburizing gases and carbonaceous dust. An HTR specific design feature as the live steam thermal sleeve nozzle at the Steam Generator Pressure Vessel may deserve particular attention.

Modular HTGR designs of higher unit power such as the PBMR 400 or recent NGNP designs are characterized by the same material as LWR RPV, but significantly higher maximum RPV temperatures in the passive heat removal modes. These temperatures exceed 500 °C, but are covered by the ASME Code Case N-499. The other characteristics described above for the HTR-Modul apply correspondingly.

Some pre-conceptual NGNP proposals featured a "Hot Vessel" design, i. e. with operational RPV temperatures of 425 °C or above, in the creep range. This design would demand choosing a different vessel material (9Cr-1Mo-V steel or 2 1/4Cr-1Mo steel) for which no appropriate LWR experience exists. Compared to the PBMR 400 even higher RPV temperatures in the passive decay heat removal modes have been reported. In the course of the NGNP project the focus for the first installation shifted from process heat for hydrogen production to process steam. The hot gas temperature was lowered accordingly from 950 °C to 750 °C and "Cold Vessel" designs were established with characteristics similar to the PBMR 400. The "Hot Vessel" option has nevertheless been treated in this deliverable as an example for the potential further development of HTGR steel pressure vessels.

10 Literature

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