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End-User Requirements fOr industrial Process heat Applications with Innovative nuclear Reactors for Sustainable energy supply

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Summary

In the frame of the EUROPAIRS project, the Task 1.2 aims at presenting in the present deliverable D12 the characteristics and limits of the HTR nuclear system to make end-users more familiar with this technology and provide the information needed for assessing the coupling.

The HTR technology has already a long historical background with several HTR programs (experimental reactors and operational demonstrators) over the world and, more generally, good knowledge on gas-cooled reactors. The industrials have gained invaluable experience from these past realizations giving them confidence that the High Temperature Gas-cooled Reactor (HTGR) technology is mature. Today HTR, and more precisely 'modular HTR', is considered as an energy alternative for cogeneration and coupling with the conventional process heat market.

The modular concept of the HTR is directly linked to decay heat removal capabilities and safety features of the reactor. Modular HTR is designed in a way that, also during loss of cooling accidents, decay heat can be removed purely by passive means without exceeding predefined temperature limits for fuel and structures. Such a plant design, however, yields limitations on the power output. Thus the optimal design of a modular HTR is a reactor with maximum power limited to 600 MWth which still obeys the temperature limits of fuel during the accidents, but also to the temperature of structures mainly that of the reactor pressure vessel and components. The key attributes of the modular HTR are its fuel, the helium coolant, no concrete but a steel vessel and limited power level and power density favoring inherent safety characteristics (no need of engineered high reliable and redundant safety system). Such simplification in the safety design leads to investments savings.

HTRs (and VHTRs) are commonly associated to heat temperatures as high as 750°C to 1000°C which correspond to the temperatures reachable at the core outlet of the reactor (ROT) and not to the temperatures available for the heat processes. In fact elevated heat temperatures are mainly limited by the properties of candidate materials used for the heat transfer (high temperature strength and corrosion resistance) regarding reasonable expected lifetime. There are thus still important technological challenges to be able to supply a heat process at 800°C.

Currently three main technologies are foreseen for the heat transfer from the core: steam generators (SG), gas-gas Intermediate Heat eXchanger (IHX), and Molten-salt IHX.

The use of SG is the most mature technology as it is already used in most realizations of LWR and gas-cooled reactors. Furthermore, boilers in general are widely used in the industry which gives a very good knowledge on the wearing mechanisms of materials. The performances already achieved with steam generators, and the expected developments that

will allow reaching higher temperatures for both sub-critical and super-critical technologies are presented in D12. The current limitations are related to the resistance to creep damage and corrosion problems.

The second technology for heat transfer presented here is the gas-gas IHX. The use of gas at secondary side of the heat-exchanger allows reaching higher temperatures than with a SG. Major high temperature design, materials availability, and fabrication issues are addressed for plate and tubular IHX concept. The second one is already mature for applications at temperatures (ROT) around 750°C-800°C. The challenge is to design gas-gas IHX that are sufficiently compact to transfer large thermal power and, at the same time, capable of withstanding creep damage for long lifetimes. Further developments are required to guarantee the performance of the plate technology but it is expected that the technology can gain maturity and become available for HTR realization in the medium term.

A third heat transfer technology is the gas to Molten-salt IHX. Molten-salt would allow a more efficient heat transfer and a much higher thermal density, while staying at low pressure in the secondary loop. However, it has its own drawback, mainly corrosion, resistance to pressure differential and risk of freezing. At present, too many uncertainties still exist to consider the coupling with molten-salt IHX as a potential technology in the near and medium terms.

Other key equipment is the primary blower that produces the force circulation of gas in the reactor. The capacity of the blower limits the power that can be extracted by each primary loop as well as the heat exchanger itself. Currently tubular technology presents the same limits as the blower. It is validated that, for a 600 MWth HTR, a design with 4 blowers and 4 SG is feasible.

Transferring heat from the primary circuit to the secondary circuit is not all. Several configurations are possible to do the coupling. The heat process can be directly coupled with the fluid of the secondary circuit, or indirectly by means of a tertiary circuit requiring additional heat transfer equipment. In parallel to the heat process supply, it is possible to have a turbine that generates electricity. The turbine could be either on the secondary or the tertiary circuit, and depending on the temperature required for the heat process, it could be in a bottoming or a topping cycle configuration. The direct coupling between the secondary circuit and the heat process presents the advantages to be very simple. However, it might not be acceptable from a safety point of view and tritium release which may ‘contaminate’ process heat circuits. The addition of intermediate circuit(s) and a turbine presents advantages, at least for a first realization of a coupling scheme, because it allows more flexibility in the operation of both sides of the coupling (HTR production and heat process consumption). Moreover, the tertiary circuit and a turbine reduce the amplitude of the transients generated by events on the heat process side.

Regarding transients and availability requirements the heat transport system should be designed to ensure enough buffering between nuclear and conventional plants to avoid an important interaction between both during their own transients. HTR is designed to withstand a set of design basis transients, as for instance, loss of cooling fluids, loss of forced circulation, and failure of the reactivity controls, which are more stringent than standard transients that can arise from events occurring on the heat process installations.

In conclusion, the deliverable D12 presents the operating window of the HTR technology, and how it could be expanded in the future, to clarify for heat consuming industries what the constraints for coupling the HTR with their processes are. The laid out performance window of the HTR technology three different configurations regarding coupling potential are proposed: ‘near term’, ‘medium term’ and ‘long term’ applications that may be proposed by nuclear suppliers during an offer without considering the commissioning of the plant. For near term application it is meant 0 to 5 years, for medium term 5 to 15 years, and for long term applications over 15 years.

Revisions

Rev.	Date	Short description	First author	Approved by (optional)
A	2010-01-20	First draft for comments	C. VIALA	
B	2010-04-02	Updated draft for comments	C. VIALA	
0	2010-07-23	First issue	C. VIALA	E. BREUIL
1	2011-05-31	Udpdated version: - economics data presented in separate deliverable D133 - review of technical data by experts - new Appendix 1 dealing with material considerations for supercritical SGs	C. VIALA Viala	E. BREUIL

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ABBREVIATIONS and ACRONYMS

AGR	Advanced Gas-cooled Reactor
ALWR	Advanced Light Water Reactor
ASME	American Society of Mechanical Engineers
AVR	Arbeitsgemeinschaft VersuchsReaktor
ANTARES	AREVA New Technology for Advanced Reactor for Energy Supply
BISO	Bistructural-ISOtropic particle
CCS	Carbon Capture System
CCGT	Combined Cycle Gas Turbine
CEA	Commissariat à l'Energie Atomique
CLF	Capacity Loss Factor
CTL	Coal To Liquid
EAF	Energy Availability Factor
EFP	Effective Full Power
EUROPAIRS	End-User Requirements fOr industrial Process heat Applications with Innovative nuclear Reactors for Sustainable energy supply
FOA	Funding Opportunity Announcement
FP	Fission Product
FSV	Fort Saint-Vrain
GA	General Atomics
GT-MHR	Gas-Turbine – Modular Helium Reactor
HP	High Pressure
HTR	High Temperature Reactor
HTGR	High Temperature Gas-cooled Reactor
HTR-10	Chinese experimental pebble-bed reactor of 10 MWth (Beijing, INET)
HTR-PM	Chinese industrial project of 250 MWth pebble-bed reactor
HTSE	High Temperature Steam Electrolysis
HTTR	High Temperature Test Reactor (Oarai,JAEA)
IAEA	International Atomic Energy Agency
IGV	Inlet Guide Vanes
IHX	Intermediate Heat eXchanger
IWGGCR	International Working Group on Gas-Cooled Reactors
JAERI	Japan Atomic Energy Research Institute (changed name a few years ago to JAEA – Japan Atomic Energy Agency)
KVK	KomponentenVersuchsKreislauf
LP	Low Pressure
LWR	Light Water Reactor
MHR	Modular Helium Reactor
NG	Natural Gas
NGNP	Next Generation of Nuclear Plant (US HTR Project)
NHSS	Nuclear Heat Supply System
NOAK	N^{th} Of A Kind

NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
O&M	Operation and Maintenance
OTSG	Once-Through Steam Generator
PB	Peach Bottom
PBMR	Pebble Bed Modular Reactor (South African HTR Project)
PBMR (Pty) Ltd	Pebble Bed Modular Reactor (Pty) Ltd
PBMR-CG	PBMR reference Cogeneration Plant
PFHE	Plate Fin Heat Exchanger
PMHE	Plate Machined Heat Exchanger
PSHE	Plate Stamped Heat Exchanger
PCS	Power Conversion System
PGS	Power Generation System
PWR	Power Water Reactor
RCCS	Reactor Core Cooling System
ROT	Reactor Outlet Temperature
RPV	Reactor Pressure Vessel
SET Plan	Strategic Energy Technology Plan
SG	Steam Generator
THTR	Thorium High Temperature Reactor
TRISO	TRistructural ISOtropic particle
UNVL	University of Nevada, Las Vegas
URD	User Requirements Document
VHTR	Very High Temperature Reactor

TABLES

FIGURES

1 Introduction

Today non-electricity energy uses represent a segment of the energy market larger than electricity generation, with a continuous growth. This market is presently nearly exclusively fed by fossil fuel burning. Europe has set itself ambitious goals for climate change mitigation and enhancement of security of supply (SET Plan) [1]. In addition, strong price volatility in particular of oil and gas motivates the move towards a more stable energy source.

The Figure 1 below illustrates the forms of energy utilization in 2005.

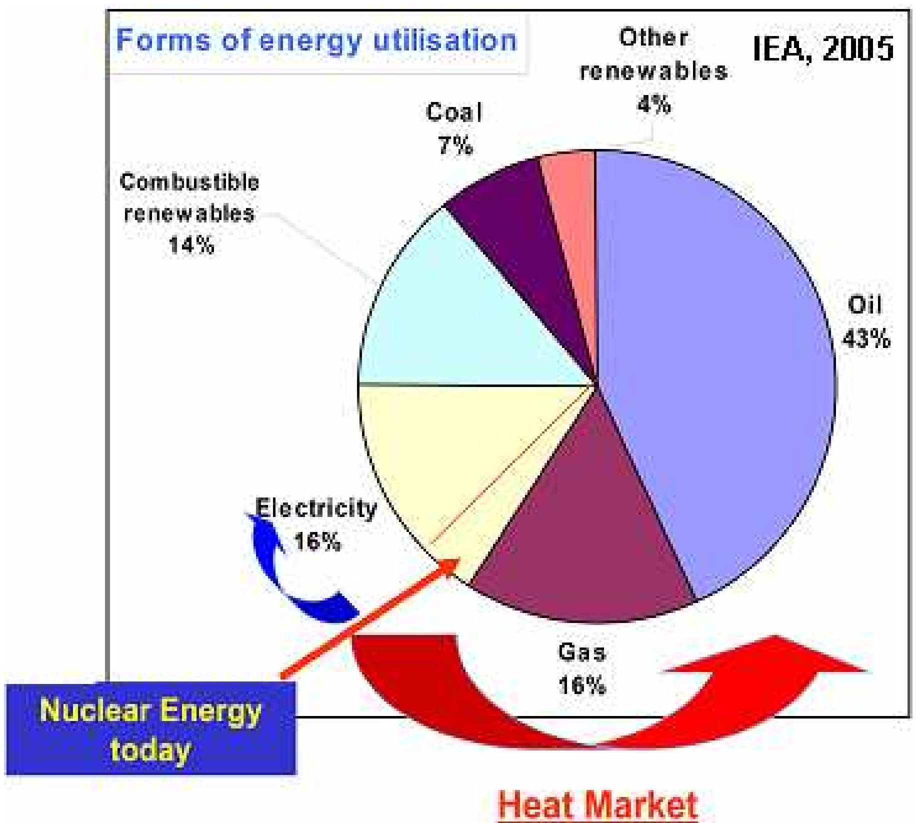

Figure 1: Forms of energy utilization (modified from IEA 2005)

Figure 2 below shows that transport represents about one third of the market, households and services represent about 37% and industry consumes nearly 28% of the market. This last sector is the most likely target for HTR in addition to possible fuel production for transport (e.g. hydrogen or synthetic hydrocarbons).

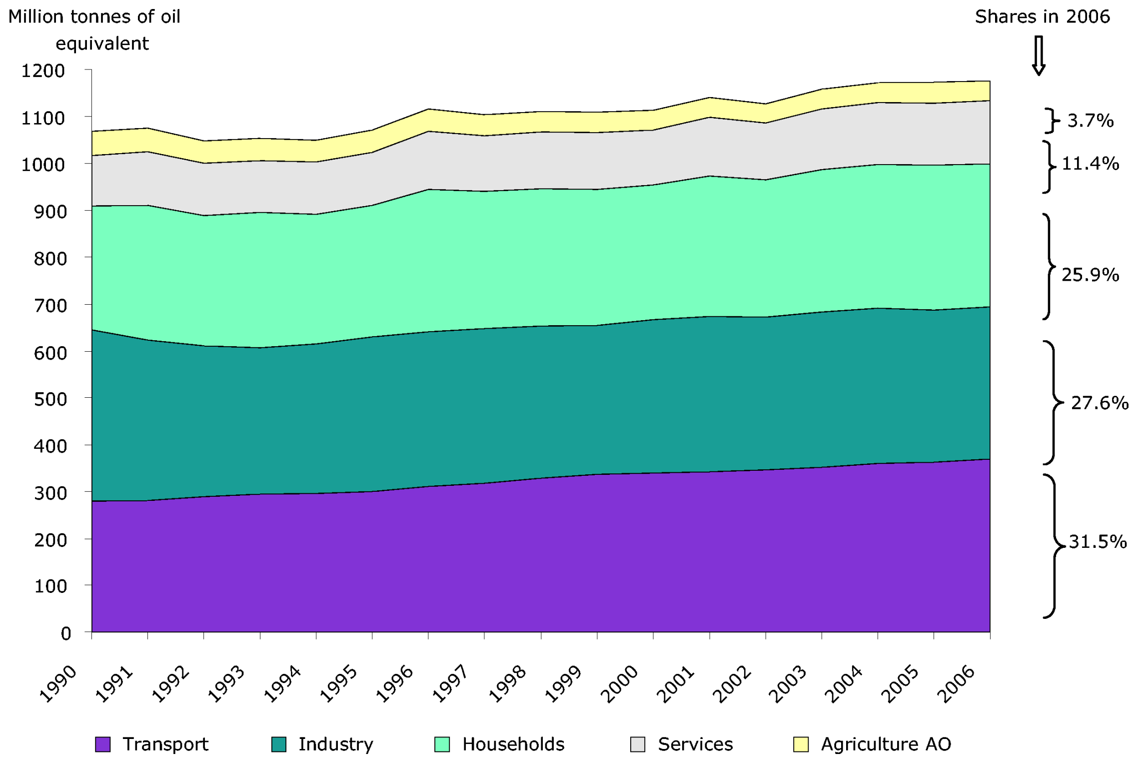

Figure 2: Final energy consumption by sector in EU-27, 1990 - 2006[1]

In this context, nuclear industry is developing a versatile nuclear heat source easily adaptable to different end-user applications by delivering process heat at various temperature levels, with or without electricity cogeneration. This technology is called Nuclear Cogeneration and High Temperature Reactors are a very suitable concept for this purpose.

The FP7 EUROPAIRS project is a joint approach from nuclear industry on the one hand, and process heat / cogeneration end-users and utilities on the other, with the collective goal to evaluate the potential impact of cogeneration of heat and power or heat only coupled to industrial process heat applications. The objective is to identify possible boundary conditions for the technical and economic viability of nuclear cogeneration systems for use in conventional industry processes.

This deliverable addresses the capabilities and limits of such applications. Different basic design options are presented: pebble bed or prismatic core reactor, steam used as secondary fluid produced in a steam generator or a single phase fluid (gas, molten salt, liquid metals) heated in an intermediate heat exchanger (IHX).

Based on needs for industrial process heat and cogeneration applications, the main performance specifications of an industrial HTR will be defined in terms of power, operating temperature, technical maturity and progress, lifetime, transient capability, availability target, preliminary economic assessments etc.

The present document will feed EUROPAIRS Tasks 1.1 and 1.3 to make end-users familiar with HTR technology and to assess a selection of coupling options.

[1] Data issued from European Environment Agency, February 20, 2009 - http://www.eea.europa.eu/data-and-maps/figures/final-energy-consumption-by-sector-in-the-eu-27-1990-2006

2 High Temperature Reactor: some generalities

2.1 What is an HTR?

The HTR, or **H**igh **T**emperature **R**eactor, (or HTGR for **H**igh **T**emperature **G**as-cooled **R**eactor) is a gas-cooled (mostly helium), graphite moderated thermal neutron fission reactor using tri-isotropic coated fuel particles which may come as "pebbles" or "compacts". Various fuel cycle options exist including low-enriched uranium, plutonium burning, deep burn for waste reduction or the thorium fuel cycle. Owing to the high thermal inertia of graphite, HTR exhibit inherent safety features, the fuel is designed to well retain fission products and the reactor comes in modular sizes (typically < 600 MWth) well suited to industrial end user sites and little developed grids. Thanks to its high outlet temperature, it enables use in a number of industrial processes and high efficiency electricity generation. Examples for such processes are depicted in Figure 3.

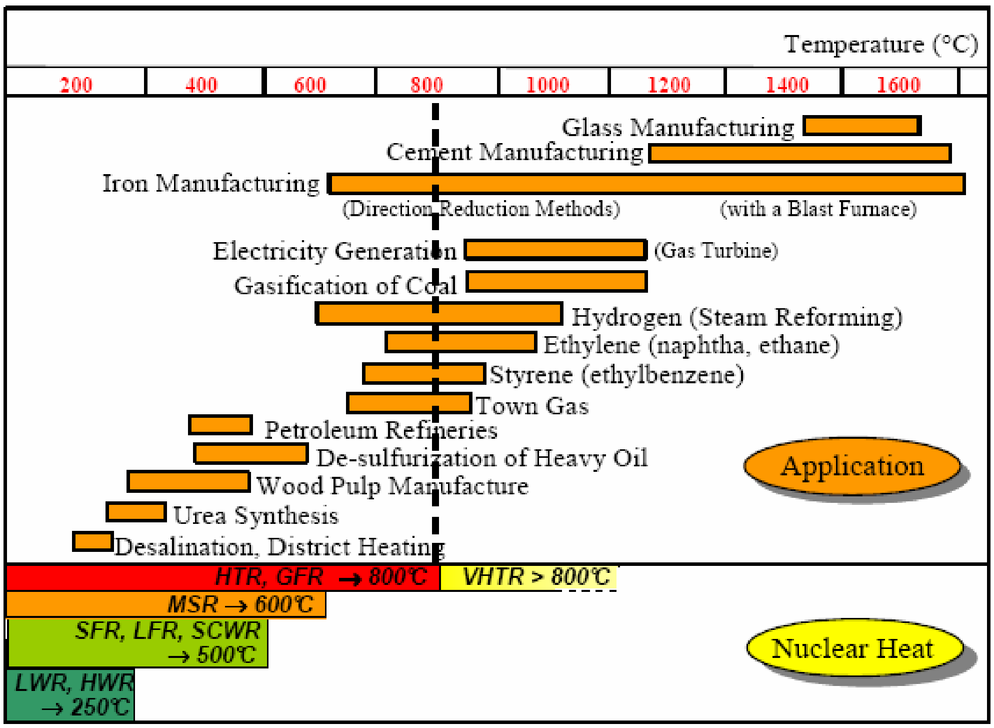

Figure 3: Nuclear heat and industrial heat applications

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Beyond electricity generation a large range of industrial process heat applications are possible. Currently, with the materials presently available for heat exchange components, the reactor could supply process heat up to approx. 700°-750°C, though the reactor itself can operate at significantly higher temperature. Higher temperature heat supply can nevertheless be envisaged in the longer term with advanced materials. The HTR is to large extents a proven technology with concept-specific inherent safety features (catastrophe free), good availability characteristics, modular design and the right size for industrial applications (typically < 600 MWth).

As a mature technology, HTR can address already in the short term the reduction of carbon dioxide (CO_2) emissions and enhanced security of energy supply by substituting fossil fuels for process heat generation.

After presenting the history of HTR and the modular reactor concept, the main capabilities of the HTR in terms of fuel, and inherent safety features are presented. General aspects such as various power conversion options are also described.

2.2 The history of HTRs

This section shortly summarizes historical HTR projects as well as those in currently active HTR programs.

Table 1 presents a list of past and present HTR projects worldwide.

	Dragon	Peach Bottom-1	AVR	FSV	THTR	HTTR	HTR-10
Location	UK	US	Germany	US	Germany	Japan	China
Start Construction	1959	1962	1959	1968	1971	1990	1994
Operation(1)							
Start	1964	1966	1967	1974	1986	1998	2000 /2003(2)
End	1977	1974	1988	1990	1989	NA	NA
Fuel							
Cycle	HEU / Th / LEU	HEU / Th	HEU / Th / LEU	HEU / Th	HEU / Th	LEU	LEU
Type	Carbide / Oxide buffered BISO / TRISO particles	Carbide BISO coated particles	Carbide / Oxide BISO / TRISO coated particles	Carbide TRISO particles	Oxide BISO / TRISO coated particles	Oxide TRISO coated particles	Oxide TRISO coated particles
Form	Hexagonal rods	Long cylinder	Pebble	Hexagonal block	Pebble	Hexagonal block	Pebble
Coolant outlet temp.(°C)	750	728	850 / 950	785	750	850 / 950	700 (900)
Power							
MWth	20	115	46	842	750	30	10
MWe	0	40	15	330	300	0	3
Energy Conversion System	none	Steam turbine	Steam turbine	Steam turbine	Steam turbine	none	Steam turbine
Live steam temp./pressure (°C/MPa)	NA	538/10	505/7.3	538/10	550/18.5	NA	440/4

(1) Start and end dates quoted from [24]
(2) Connection to grid

Table 1: Past and present High Temperature Gas Reactors

2.2.1 Past reactors

Early gas-cooled reactor development started in the United Kingdom and in France. These reactors generally used CO_2 as coolant and metal clad fuel with graphite as moderator. They achieved higher operating temperatures than LWR but the use of metal clad fuel and non-inert coolant limited the ultimate temperature that could be achieved. Such reactors are still successfully operated in the UK (AGR) (primary CO_2 conditions 298/640°C at 4 MPa) combined with a conventional steam cycle (16.6 MPa/541°C live steam conditions) reaching efficiencies of 41-43%. This is already significantly higher than what can be achieved by current Light Water Reactors (32-35%).

Work on high temperature reactors began in the mid-1950s in parallel with CO_2 cooled reactors. The technologies of both types of reactors were closely related and advances in one were typically shared in the other (graphite technology, vessel, blower and heat exchanger design and fabrication).

Forty-five years ago, five reactors over the world began the HTR history from experimental reactors to the demonstration of basic technology.

These reactors demonstrated HTR capabilities and all technology evolutions regarding fuel and operating conditions, design and safety performance. Figure 4 illustrates the first HTR demonstrators.

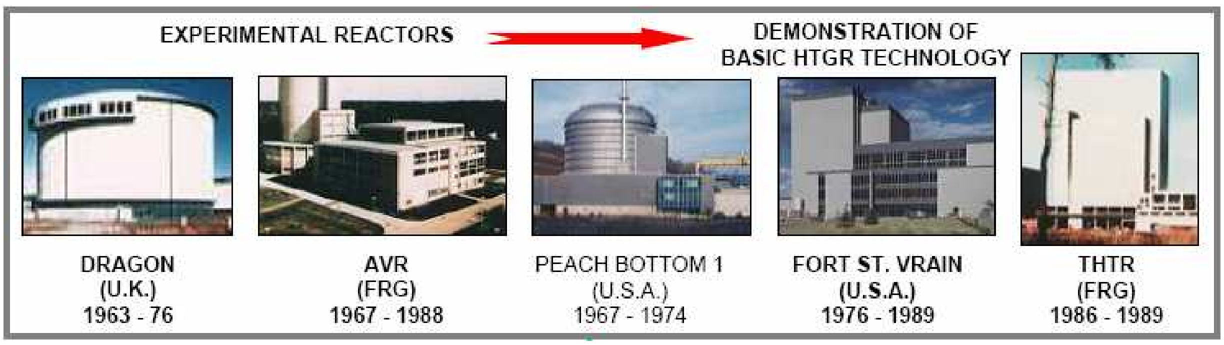

Figure 4: Illustration of the Five First HTRs

2.2.1.1 Dragon reactor

The first major experimental HTR to operate was DRAGON in the United Kingdom. This reactor produced 20 MWth with a helium outlet temperature of 750°C. It first went critical in 1964 and reached full power in 1966. The project ended in 1976.

The fuel was a mixture of 93% enriched uranium and thorium in carbide form sintered into annular compacts contained in hollow graphite hexagonal rods which extended the full height of the core (precursor of block technology). Coated particles were used in DRAGON during later irradiation test campaigns. As a precursor, DRAGON encountered difficulties such as rather elevated fission gas release from the fuel and irradiation effects on graphite. During its operation from 1966 to 1975, the reactor was used to test a wide variety of fuel types and graphite elements; numerous TRISO fuel types were tested. Some problems appeared also on the heat exchangers due to waterside flow instabilities after 3 years of operation, however overall the reactor remained very reliable.

2.2.1.2 Peach Bottom - 1

Peach Bottom Unit 1 (PB-1) was the first 'large' HTR constructed in the United States and it included electricity production (40 MWe). The reactor generated 115 MWth with a helium outlet temperature of 725°C enabling live steam conditions of 538°C at 10 MPa.

It operated from1967 to 1974. While DRAGON was a test reactor, the mission of PB-1 was first to demonstrate electricity production using high quality steam. Availability would have been 85% without planned shutdowns for insertion and removal of test elements (over 1500 fuel samples were irradiated).

A variety of particle fuel was used over the lifetime of the reactor. The fuel in the first core experienced poor performance due to swelling and cracking of the fuel particles which had only a single coating without an inner low density carbon buffer.

The initial core was replaced with fuel using 'BISO' particles which included the low density buffer. Subsequent performance with the new fuel was very good.

2.2.1.3 AVR reactor

AVR, or Arbeitsgemeinschaft VersuchsReaktor GmbH built in Germany, was the third early demonstration HTR and the first using pebble-bed technology which, contrary to a hexagonal block type design, enabled in particular on-line refueling for increased availability. The core contained around 100.000 pebbles of 6 cm diameter. It was a 49 MWth reactor started in 1960 and the steam produced was used to generate 15 MWe since 1967. The reactor was intended to demonstrate high temperature operation with pebble fuel. Initial reactor outlet temperature was 770°C and operation during most of the AVR's early life produced helium at 850°C. In 1974, the outlet temperature was increased to 950°C. The reactor was shut down in the end of 1988 after a successful demonstration of HTR capabilities for 21 years.

During its operation, the AVR employed a number of different fuel forms and numerous tests were carried out to demonstrate key operational and safety characteristics, in particular passive heat removal. Additionally, a leak in the AVR steam generator unintentionally provided additional operational experience. Steam parameters for the AVR were 505°C and 7.3 MPa.

2.2.1.4 Fort St. Vrain reactor

Fort St. Vrain (FSV) was the first large scale HTR power reactor with 842 MWth and a prismatic block core. Fuel compacts containing TRISO fuel particles were loaded in holes in the graphite blocks. With two loops for the primary coolant system, each with six steam generators and two steam-driven circulators, the reactor produced high quality steam which generated 330 MWe (with steam cycle). The outlet steam temperature and pressure in the FSV were similar to the ones at Peach Bottom.

FSV started initial operation in 1976. The reactor encountered several operatinal problems which reduced its availability over its lifetime from 1976 to 1989. The most troublesome was water ingress from the circulator bearing water system into the reactor. This did not have direct adverse operational consequences but induced operating temperatures decreasing or reactor shutdown to avoid unacceptable oxidation. This problem significantly reduced the availability of FSV over its lifetime from 1976 to 1989.

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2.2.1.5 THTR

The German Thorium High Temperature Reactor (THTR) was built in Germany in parallel to AVR operation. As FSV, the THTR was a large reactor which should demonstrate performance of pebble-bed technology. FSV and THTR presented similarities in terms of size and concept, except the fuel element form.

THTR started in 1984. The 700 MWth reactor (14 times the AVR power) produced 300 MWe. The steam conditions were 550°C at 18.5 MPa.

As FSV, the plant experienced teething problems typical for a new prototype plant. Moreover, the Chernobyl accident and the resulting antinuclear political context were significant obstacles to continue reactor operation. Consequently, THTR had to shut down prematurely.

2.2.2 On-going plants

After demonstration of HTR technology in Europe and in the US, Japan and China developed experimental reactors as precursors to modular reactors.

2.2.2.1 HTTR

The High Temperature Test Reactor (HTTR) is a 30 MWth experimental Japanese reactor resulting from a long-term HTR development program with focus on process heat applications. HTTR uses TRISO particles in prismatic fuel.

First criticality of the HTTR was attained in November 1998 with an outlet temperature of 750°C. The full power of 30 MWth and the reactor outlet coolant temperature of 850°C were achieved in December 2001. Rated power operation and safety demonstration tests started in 2002. The maximum reactor outlet coolant temperature of 950°C was first achieved in April 2004 for a short period time.

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The HTTR is exclusively a test reactor without electricity generation. Since the program focuses on heat applications, the plant includes a 10 MWth Intermediate Heat Exchanger (IHX) to demonstrate IHX technology. After an extended outage for replacement of a control rod, the HTTR has successfully performed a 50 day long-term high temperature operation at 950°C in early 2010. Future operation is planned to use heat for hydrogen production with the Sulfur-Iodine process.

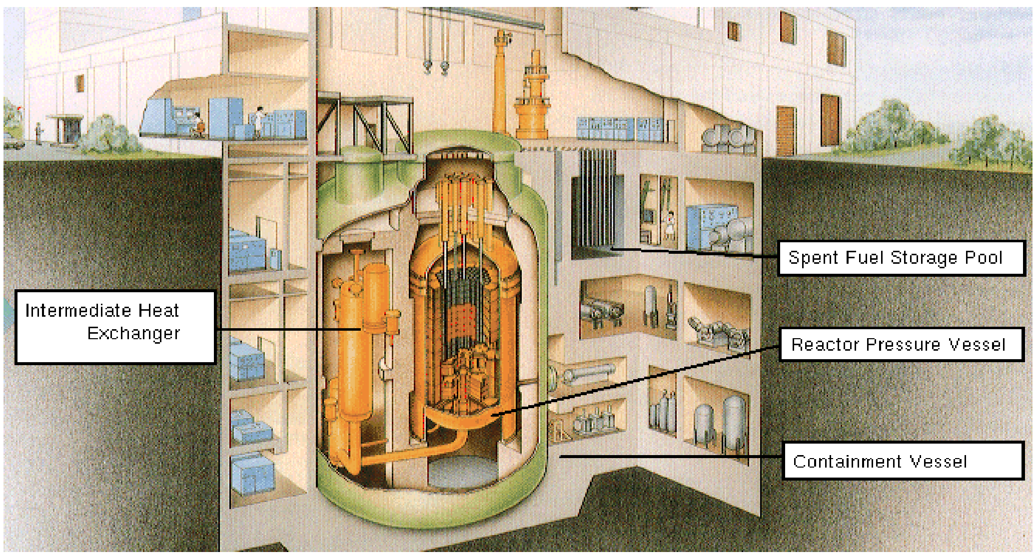

Figure 5: Schematic representation of the Japanese HTTR

2.2.2.2 HTR-10

In China, the HTR was initially designed for electricity generation in areas unsuitable for large LWR. A variety of process heat applications is envisaged for later stages. HTR-10 is an experimental 10 MWth pebble-bed reactor using German technology based on the HTR Modul, designed by Interatom. First operation started in 2000. It currently produces 3 MWe through a steam cycle. Steam parameters for the HTR-10 are 440°C at 4 MPa.

In addition to power generation and technology demonstration, HTR-10 has successfully proven benign behavior during safety transients, similar to those previously demonstrated in AVR. This included loss of flow events and loss of coolant with failure to trip the reactor.

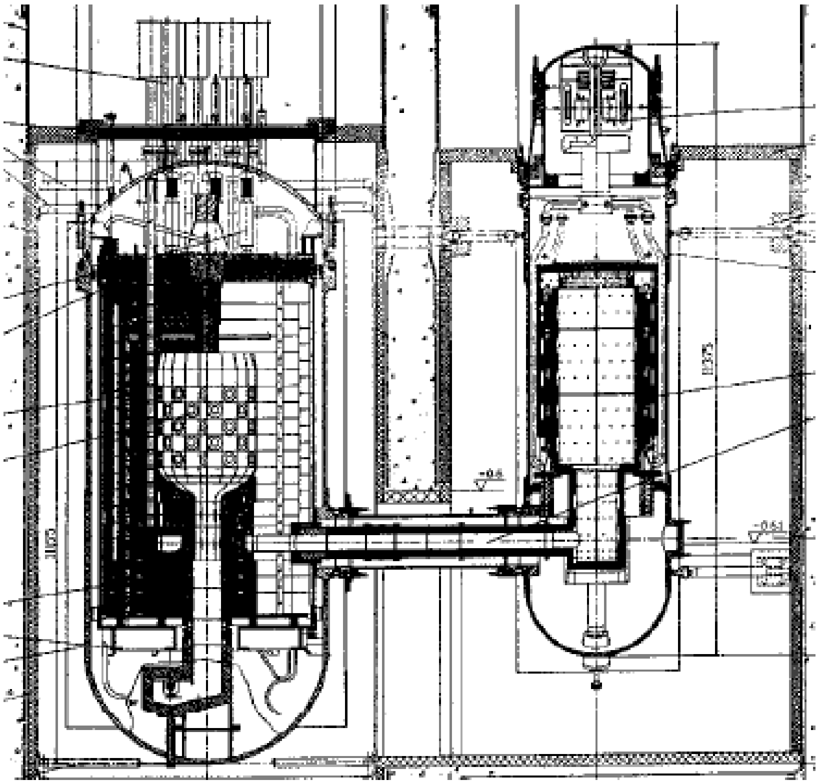

Figure 6: HTR-10 cross section[2]

2.2.3 Modular HTR concept and industrial prototypes

The successful demonstration of HTR capabilities in several countries encountered several obstacles which prevented it from deployment. Amongst others, those were relatively low fossil fuel prices, no restrictions on CO_2 emissions, competition with established nuclear technology and a hostile public opinion and political context in a number of countries following the Chernobyl accident in 1986. In the late 1990s, some evolutions led to renewed interest in HTR technology. To achieve competitiveness with HTR against the economy of size of large LWR, the modular concept (see 2.3) was adopted in the 1980s. HTTR and HTR-10 were built as precursors of this modular concept. Many reasons for that can be evoked. One of them is that building modular nuclear plants by successively adding smaller reactors with high level of factory fabrication enables a substantial time gain compared to field construction, which is a strong economic argument for investors. Continuous expansion of the site can be performed while first modules already produce electricity and cash-flow for new modules. Modular reactors should rely more on "economy of replication" rather than on "economy of size". Smaller modular reactors also take full advantage of inherent safety characteristics of the HTR, including passive decay heat removal from an un-insulated metallic reactor vessel through radiation and convection, allowing a drastic simplification of the safety design and therefore a substantial reduction of costs.

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Rev.1

Internationally, several HTR development and demonstration programs are currently underway, the most important are quoted here:

[2] Z. Daxin, Q. Zhenya, INET, Seminar on HTGR Application and Development, Beijing, 19-21 March 2001

2.2.3.1 PBMR

Pebble-Bed Modular Reactor (PBMR) is a South-African project managed by Pebble Bed Modular Reactor (Pty) Ltd. The initial target was electricity production for the state utility Eskom, but for financial and time reasons Eskom had to find quicker fixes to prevent recurring brown-outs in the country thus focusing on coal-fired generation instead of nuclear power. Therefore the PBMR Company refocused the project towards supply of steam for process heat applications, electricity generation or both (cogeneration).

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PBMR relies on German pebble bed technology. Regarding the thermal power, two times 200 MWth is foreseen. Due to economic crisis, the company had to stop its activity in 2010.

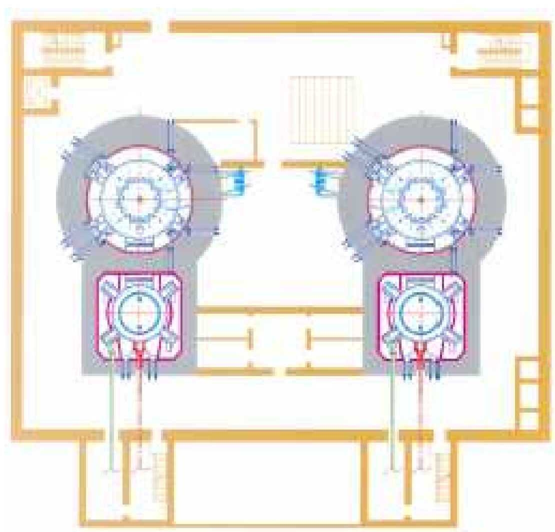
2.2.3.2 HTR-PM [21]

HTR-PM is a Chinese <u>industrial project</u> based on outcome of their experience with HTR-10. The HTR-PM plant will consist of two Nuclear Steam Supply Systems (NSSS), so called modules, each one comprising of a single zone 250 MW_{th} pebble-bed modular reactor and a steam generator. The two NSSS modules feed a common steam turbine and generate an <u>electric power</u> of 210 MWe.

A pilot fuel production line will be built to fabricate 300,000 pebble fuel elements per year. This line is closely based on the technology of the HTR-10 fuel production line which itself is built on European technology.

The design as of 2006 is given in Table 2 below:

Thermal power, MW	*500*
Electrical power, MWe	*212*
Pressure, MPa	*7*
Helium temp., ℃	*250/750*
Steam temp. , ℃	*566*



<u>Table 2: HTR-PM technical characteristics [3]</u>

The main goals of the project are two-fold. Firstly, the economic competitiveness of commercial HTR-PM plants shall be demonstrated.

Secondly, it shall be shown that HTR-PM plants do not need accident management procedures and will not require offsite emergency measures. According to the current project schedule, completion of the demonstration plant will be around 2013. The reactor site has been evaluated and approved; the procurement of long-lead components has already been started. After the successful operation of the demonstration plant, commercial HTR-PM plants are expected to be built at the same site. These plants will comprise several modules and, correspondingly, a larger turbine.

2.2.3.3 GT-MHR:

Work on the Gas-Turbine – Modular Helium Reactor (GT-MHR by General Atomics) began in 1992 following advances in gas turbine technology, modular HTR design and high effectiveness compact heat exchangers. The GT-MHR is a direct cycle concept: helium from the reactor outlet flows directly in the inlet of a gas turbine. The basic configuration of the GT-MHR plant includes four reactor modules, each with a dedicated single loop Power Conversion System and producing 600 MWth. The expected reactor temperature outlet is 850°C.

2.2.3.4 NGNP:

The Next Generation Nuclear Plant (NGNP) was initiated by the US DOE to demonstrate emissions-free nuclear electricity and hydrogen production in a full scale prototype VHTR by 2018. The initial idea was to use the S-I thermochemical process for hydrogen production. This process requires a top cycle temperature of 850°C in the process which was the driver for pushing the primary gas outlet temperature to 950°C.

The project is developing a prismatic core or pebble-bed HTR to run both a primary and a secondary industrial application (hydrogen application of appropriate size). The heat generated by the reactor can be used to feed the hydrogen production process which is then coupled to a bottoming cycle for power conversion.

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A decision was made by the NGNP Project in October 2008 to reduce the nominal reactor outlet temperature from 950°C into the range 750°C to 800°C and to use a steam-generator as heat exchanger, with still an objective of cogeneration of electricity and process heat, but for application to steam supply to existing steam networks accessible on large industrial sites, most particularly in chemical and oil industry. This decision to reduce the reactor outlet helium temperature had a significant impact on the technology development effort required to support the NGNP and on its timing. Most of the technology development required for an NGNP operating with a reactor outlet helium temperature of 950°C will no longer be needed. Examples are development and qualification of high-temperature metal alloys for the IHX and ceramic composites for several reactor internals components, design and verification of a reactor vessel cooling system, etc.

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The NGNP project is being conducted in two phases. Phase 1 comprises research and development, conceptual design and development of licensing requirements. The awardees will support the development of conceptual designs, cost and schedule estimates for demonstration project completion and a business plan for integrating Phase 2 activities. The US DOE will use recommendations from its independent Federal advisory committee, the Nuclear Energy Advisory Committee, recommendations based on information and data gathered in Phase 1, and other factors in determining whether the project should continue to Phase 2. Phase 2 would entail detailed design, license review and construction of a demonstration plant.

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In March 2010, the DOE awarded contracts for Phase 1 to General Atomics (prismatic model, one loop for 350 MWth). No contract was awarded for pebble-bed design, as Westinghouse withdrew. Nevertheless AREVA is making an assessment of the status of pebble bed technology. Phase 1 is expected to finish by the beginning of 2011.

2.3 The modular concept

As presented above (see subparagraph 2.2.3) the modular HTR is a leading technology candidate for the next generation of commercial nuclear reactors, with a specific asset for industrial process heat applications, due to its operating temperature higher than for other reactor types.

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Increasing the power of a unit is usually the most effective way to reduce the energy costs of power plants. But for HTR with a limited power, an innovative safety concept, the modular concept, has been proposed, drastically simplifying the safety design through an optimized use of the unique inherent safety features of this type of reactor detailed in the next chapter (mainly the strongly negative temperature coefficient, low power density, the high thermal inertia and the absence of release of fission products from the TRISO fuel up to at least 1600°C). It has indeed been demonstrated that for such low power, in case of loss of normal cooling, the fuel temperature can be kept lower than 1600°C without activation of any engineered safety system, only through the natural smothering of the nuclear power as soon as temperature increases (due to the strongly negative temperature coefficient) and then through natural decay heat removal by conduction in reactor materials (graphite, steel) and radiative heat transfer between the surfaces of walls (graphite blocks, baffle, reactor vessel) up to panels located outside the reactor vessel, on the sides of the reactor building, cooled by natural convection of air or water. The reactor vessel must be constructed in steel and no more in concrete like in previous HTR design, to be able to withstand higher temperatures in accident conditions.

Rev.1 Depending on the design characteristics (fuel technology (pebble-bed or prismatic), core layout (annular or cylindrical), heat exchanger (SG or IHX), etc.) the power is typically limited to about 600 MWth. But the possibility of doing without very expensive redundant engineered safety systems is a major asset for aiming at competitiveness for such limited power capacity.

3 Inherent safety features

The safety case for modular HTR relies on inherent characteristics that, collectively, limit to extremely low levels the consequences of any faulted condition. Moreover, these characteristics provide a very slow evolution of the accident transients offering additional safety margins for managing post-accident situations. These are:

- the high temperature tolerance of the fuel itself,
- a negative reactivity coefficient (temperature, void, Doppler) in any condition,
- the low power activity,
- large thermal inertia owing to large amounts of graphite in the core,
- a metallic reactor pressure vessel that allows, in combination with a limited power, passive removal of heat from the system,
- helium as primary coolant, chemically inert,
- low radiological consequences.

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Given these inherent safety characteristics, the modular HTR plant design does not require active safety systems in event of subsystem failure and it is immune to major structural failure and operator error. Its design uses thermophysical properties of materials and passive physical processes to achieve an optimum choice of reactor size, geometry and power density. It tolerates a total loss of coolant without any core meltdown risk. This capability enables a simplified design and, consequently, competitiveness of modular HTR.

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In nuclear reactors, accidents are related to mismatches between the generated power and the heat extraction by the primary coolant. In addition in HTR the effects of air or water ingress in the core have to be considered. Mismatches between the power and the cooling are due to untimely reactivity insertions (due to control rod withdrawals or excessive cooling) or to reductions in the coolant flow (circulator trip, loss of coolant due to pipe break). Such mismatches result in temperature increase of the fuel and then, due to the strongly negative reactivity coefficient, the nuclear reaction will quickly choke off. So in any accident situation, only the decay heat (which is decreasing in time, but even in the beginning no more than a few percents of the full power of the reactor) must be removed from the reactor. If the decay heat is sufficiently low (this is done by limiting the reactor power), it can in any situation be removed from the core only through conduction and radiative heat transfer without having the fuel temperature exceeding in any part of the core about 1600°C, which allows, due to the high temperature resistance of the fuel, avoiding any significant radio-contaminant release. Moreover, the large thermal inertia brought by the graphite structures (about 1000 tons in the reactor) makes the thermal transients very slow (it takes days for fuel temperature to reach 1600°C), providing plenty of time for initiating mitigation measures.

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Now if the accident is due to a break in the primary containment, the reactor will be faced with specific phenomena related to air or water ingress (in case of a steam generator pipe break). Graphite and fuel oxidation by oxygen or water is the most important phenomena to be faced. In order to avoid all the graphite to be quickly oxidized and the fuel integrity to be soon threatened by corrosion, the reactor is designed in order to limit the quantity of air or water that can enter in the reactor. For limiting air ingress, special measures are taken in the design and in the construction for the reactor and steam generator vessels (including the cross duct), in order to exclude any possibility of break in these big vessels (as done for LWR vessels); then all other pipes are designed in order to have a limited section (in the German HTR-Modul, the diameter of pipes is limited to DN65). In such conditions it is shown that the quantity of air that can enter in the reactor vessel is such that only a small part of the graphite can be oxidized and in no case the fuel. For limiting the quantity of water in water ingress the steam generator is automatically isolated and dumped. Moreover the steam generator is located below the core level.

As modular reactors obtain their competitiveness rather from the economy of replication than from the economy of size, major system components and assemblies can be fabricated in series in a workshop where optimized production processes, reproducible manufacturing standards, quality control and shorter production times can be achieved. This aspect also adds to improved economics.

3.1 Fuel

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Some of the particular advantages of HTR technology are based on the fuel performance. So-called TRISO[3] fuel is used. All HTR use small (~1mm diameter) fuel particles which consist of a fuel kernel of enriched uranium in the form of an oxide (or oxycarbide) of about 500 µm in diameter. This kernel is subsequently coated with a porous pyrocarbon layer (buffer) capable to absorb gaseous fission products and which can accommodate fuel swelling, a dense pyrolytic carbon layer (PyC) improving the strength of SiC layer and constituting a first barrier against fission product (FP) release, a silicon carbide layer (SiC), main barrier of FP retention, and finally another dense pyrolytic carbon layer (see Figure 7), both to ease thermal bonding with the matrix graphite and still to protect SiC from high stresses. All these coatings surrounding the kernel of TRISO particles produce a very robust fuel form by acting as a first barrier for the radioactive material.

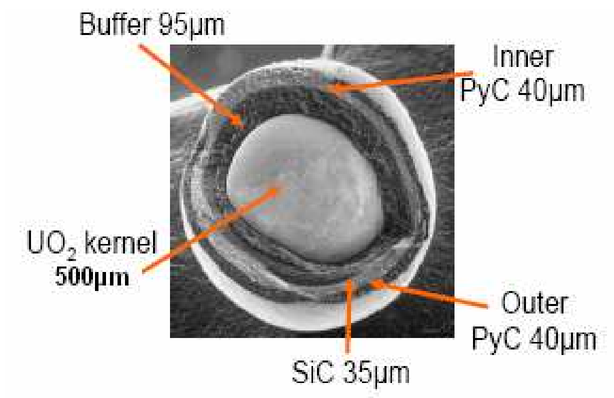

Figure 7: Illustration of a coated fuel particle of HTR reactor TRISO particle

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In a number of post-irradiation heat-up tests, this type of fuel in the form of pebbles or compacts (Figure 8) has demonstrated good fission product retention up to 1600°C (see Figure 13 and Figure 14 below), which is the maximum core temperature expected for the HTR Modul designed by Interatom following an unprotected loss of flow accident. More generally, modular concepts are designed so that most of their fuel does not exceed 1600°C in accident conditions.

The TRISO particle is the common elementary constituent of the fuel in the two main HTR design concepts: the pebble-bed concept and the prismatic block concept (Figure 9).

[3] TRistructural ISOtropic

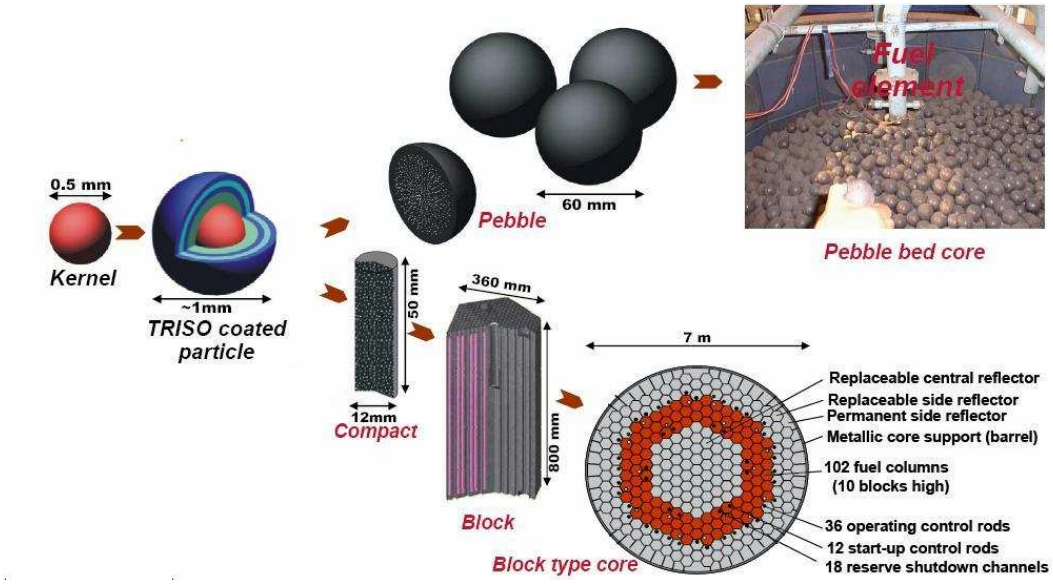

Figure 8: Possible concepts of fuel elements and core

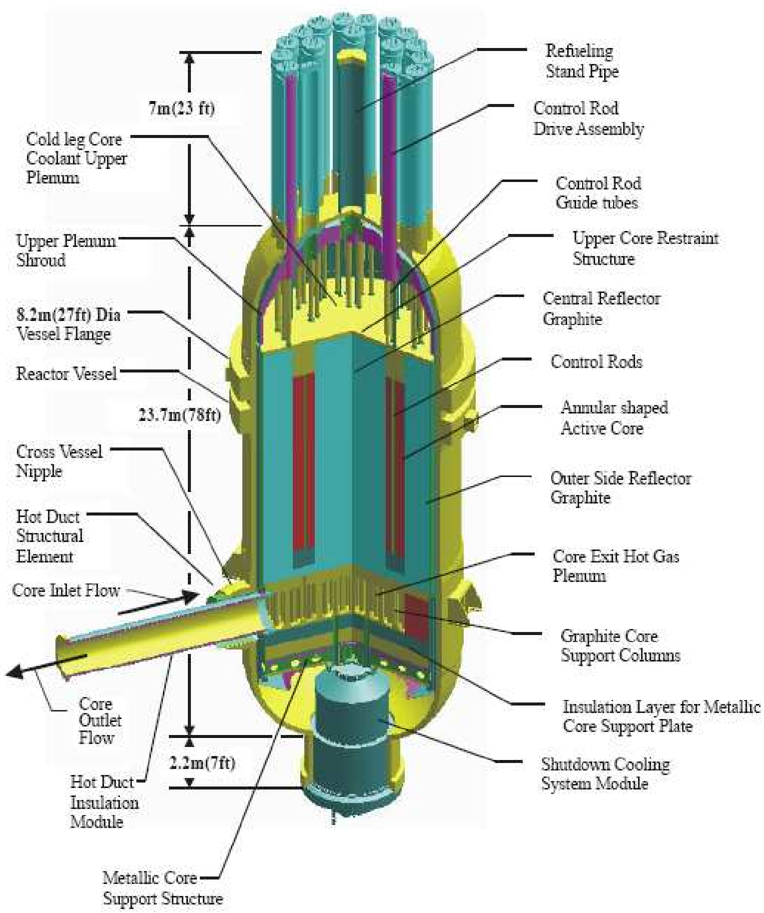

Figure 9: Technology option Prismatic Block[4]

[4] INEEL/EXT-03-00870, Rev. 1, NGNP Point Design – Results of the Initial Neutronics and Thermal-Hydraulic Assessments During FY-03, September, 2003.

As a particularity, 'Pebble-bed' reactors use on-line refueling with pebbles being inserted at the top of the core and removed from the bottom. Pebbles removed from the reactor are checked for burn-up and integrity and those with full burn-up are sent to spent fuel storage while the others are returned to the top of the pebble bed fro recycling. This feature is notably reducing outages for refueling or fuel shuffling. This results in outage for maintenance every 5-6 years (a large part of the maintenance being performed when the reactor is operating), instead of every 18-24 months for a prismatic reactor. A pebble-bed reactor can operate with very low excess reactivity and with the helium flowing between the pebbles. Moreover a lower fuel temperature is achieved compared to a similar prismatic reactor with the same coolant outlet temperature.

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HTR fuel is a safe waste form for permanent geologic disposal, although it may contain up to 96% of carbonaceous matter. Improvements of fuel and graphite waste management are currently being addressed in the FP6 CARBOWASTE project. As stated above and shown in the Figure 7, the TRISO coatings act as miniature containment barriers that are highly resistant to corrosion and pressure build-up over geologic time scales. The very low corrosion rates of silicon carbide and pyrocarbon were confirmed during independent testing at various laboratories. TRISO coatings provide the substantial benefit of long-term containment without having to rely on expensive waste packages. It was shown recently [26], [27], that HTR fuel can be post-processed to reduce the waste streams (liberation of coated particles from the matrix) and then optionally be fragmented for further reprocessing. This is enabling technology for symbiotic fuel cycles, deep-burn of actinides to drastically reduce the long term radio-toxicity of the HTR fuel waste, the U-Pu or the Th-U fuel cycle.

Regarding proliferation resistance, effective safeguards might be more complicated to implement for pebble bed reactors than for the prismatic block option because of the small size of the fuel element and of the multiplicity of fuel flow paths for on line refueling and recycling. Optimum (for diversion) plutonium content may be achieved after the first or second of the multiple passes that it will eventually make through the core. Nevertheless, it should be noted that the fuel handling system remains a closed system for both prismatic and pebble-bed concepts. Moreover a large number of pebbles (~ 100 000) should be diverted to obtain a significant quantity of fissile materials for explosive purpose, which would likely not remain unnoticed from IAEA inspections.

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3.2 Decay Heat

As in all nuclear reactors, and due to decay of fission products, the heating of the nuclear core of an HTR does not stop completely when the power is shut off. The ability to remove the decay heat without excessive fuel temperatures is essential for inherent safety of HTRs.

One particularity of modular HTR is that decay heat can physically not lead to core meltdown even if the coolant is lost. The reactor's low power density and thus low decay heat allow designing the reactor in such a way that decay heat is dissipated passively by conduction and radiative heat transfer without exceeding approximately 1600°C even under the most severe accident conditions. Under such temperature limit, the fuel particles will keep their leaktightness. Unlike in LWR, there is no need for engineered safety systems (with high reliability and high redundancy) for removing decay heat (see Figure 10 and Figure 11).

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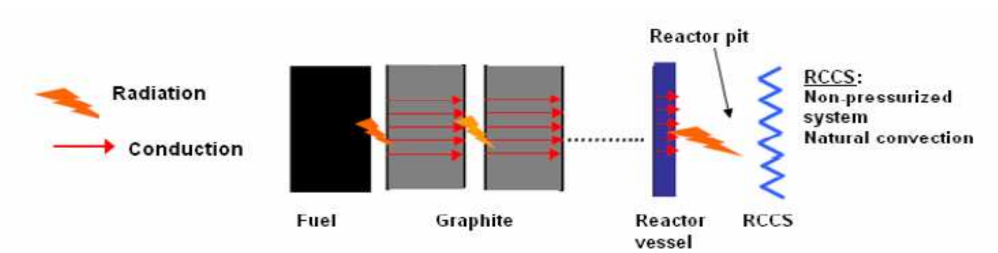

Figure 10: Schematic of passive decay heat removal

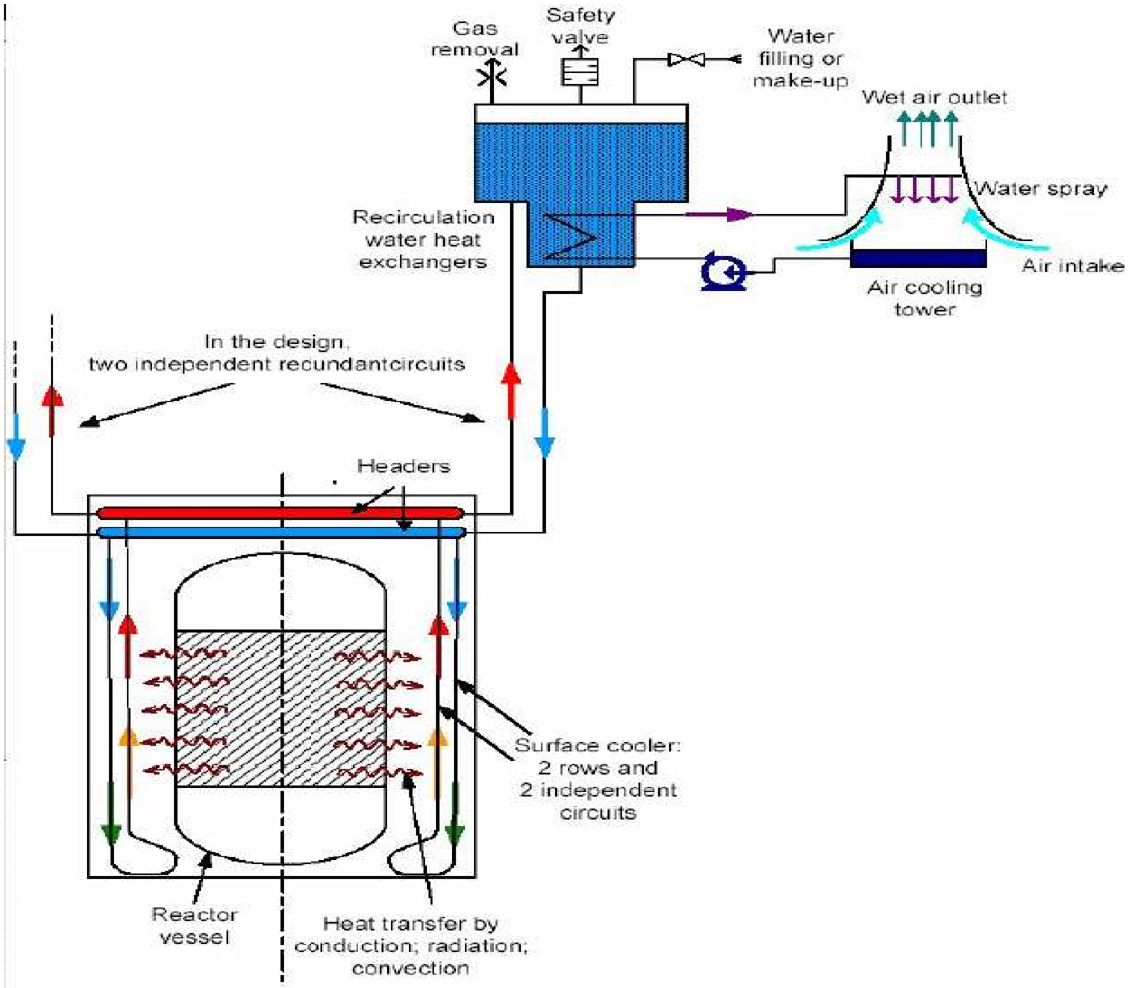

Figure 11: Reactor Cavity Cooling System (RCCS)

The concept of inherent safety features of the modular HTR design with respect to passive decay heat removal through conduction and radiative heat transfer was first introduced in the German HTR modul and in the US MHTGR, e.g. PBMR (pebble fuel), GT-MHR (prismatic fuel), ANTARES (prismatic fuel) and NGNP.

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First demonstration of the potential of passive safety features was achieved on AVR. The main circulators were stopped without allowing the reactor trip. The reactor shut down naturally by negative temperature feedback (see Figure 12 below).

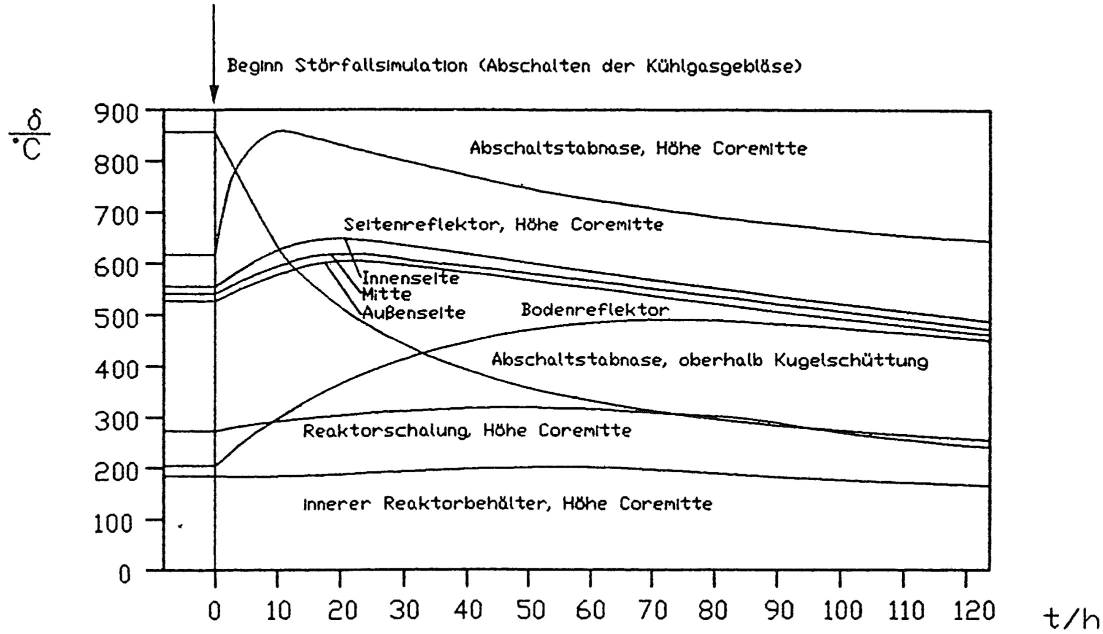

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Figure 12: Evolutions of temperature in AVR following main circulators stop test

Fuel temperatures increased slightly but remained at acceptable levels throughout the transient. Eventually the reactor went critical due to Xenon decay and gradual cooling as decay heat power level fell. The power level was stabilized at a low level until the control rods were finally inserted, terminating the successful demonstration. The Figure 13 below show AVR results obtained: TRISO fuel retains radionuclides beyond maximum conditions. The Figure 14 presents also KÜFA results from the heating test on fuel elements.

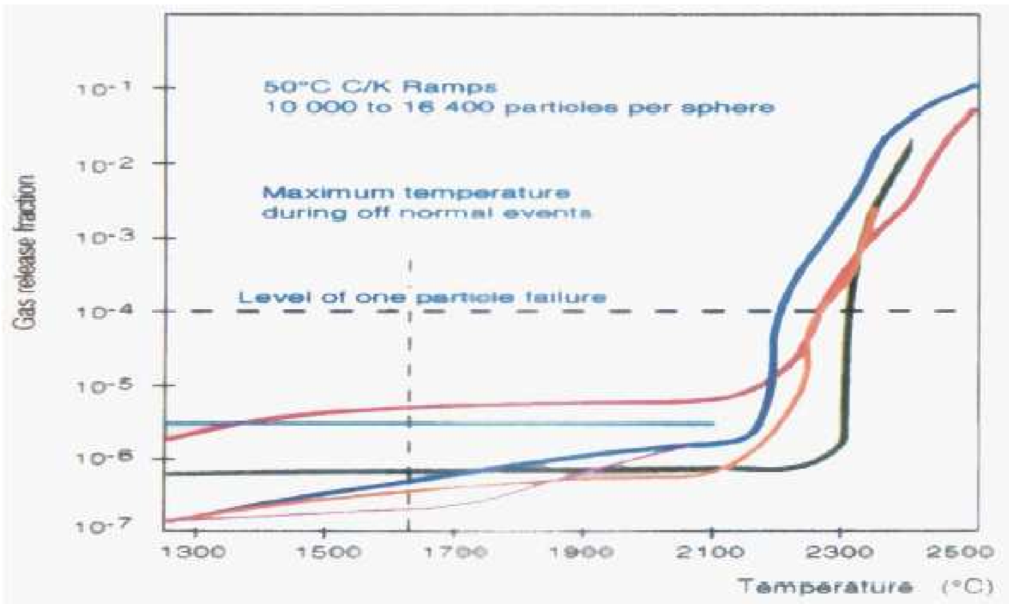

Figure 13: Ramp tests with AVR fuel to 2500°C

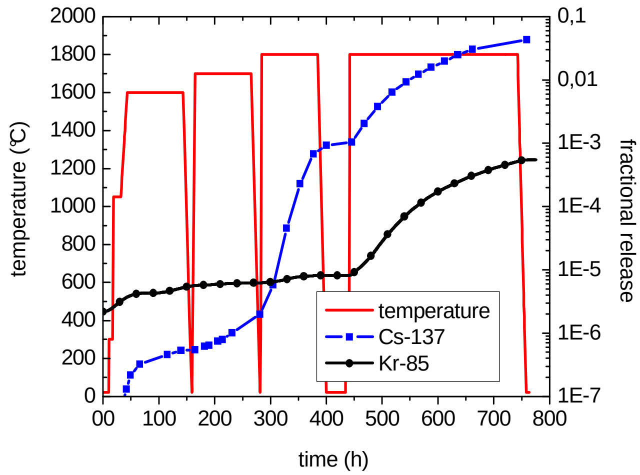

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Figure 14: 137Cs and 85Kr release as a function of time/temperature; KÜFA results from the heating test on fuel element HFR K6/3 [28]

Such inherent safety features, however, yields limitations on the power output. Depending on the reactor design characteristics the power limits range is 200 MWth to 600 MWth. Thus, from the thermal hydraulic point of view a reactor with maximum power which still obeys the temperature limits of fuel and components represents an optimal design of a modular HTR. Optimization of design features is considered with reference not only to the maximum fuel temperature during the accidents, but also to the temperature of structures, mainly that of the reactor pressure vessel. In modular HTR conception, steel vessels replace concrete vessels and can withstand temperature up to 550°C in accident conditions. Though a thermal insulation of the core and the reflector increases the fuel temperature, maximum temperature of the pressure vessel and the core barrel is decreased. Thus, insulating carbon blocks in the periphery of the graphite lateral reflector represent an important element for optimization of the design.

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For pebble-bed reactor, the relevant design proposals in the literature can be classified to mainly three options with regard to core layout [6]:

- core with cylinder geometry
- annular core with dynamic inner column of graphite pebbles
- annular core with solid inner graphite column

It is pointed out that annular cores can produce higher power without exceeding fuel temperature limits, especially during depressurization accidents [6]. This is mainly due to geometrical effects. Heat storage effects of the inner column also have an influence on the maximum fuel temperature by increasing the time at which this temperature is reached.

3.3 Graphite

As presented in section 3.1, there are two main HTR design concepts: a pebble-bed concept and a prismatic concept. In all theses designs, the graphite is employed as a moderator and/or for major structural components (see Figure 15) providing channels for the fuel for coolant gas circulation, and for reactivity shutdown devices. Graphite reflectors around the core are also providing thermal and neutron reflection and shielding. In addition, graphite components may act as a heat sink or conduction path during reactor trips and transients [7].

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Graphite has been selected as a structural material in HTR plants because of its favorable characteristics: neutron moderator, low ductility and low tensile strength, its behavior under the impact of fast neutron irradiation and its huge thermal inertia.

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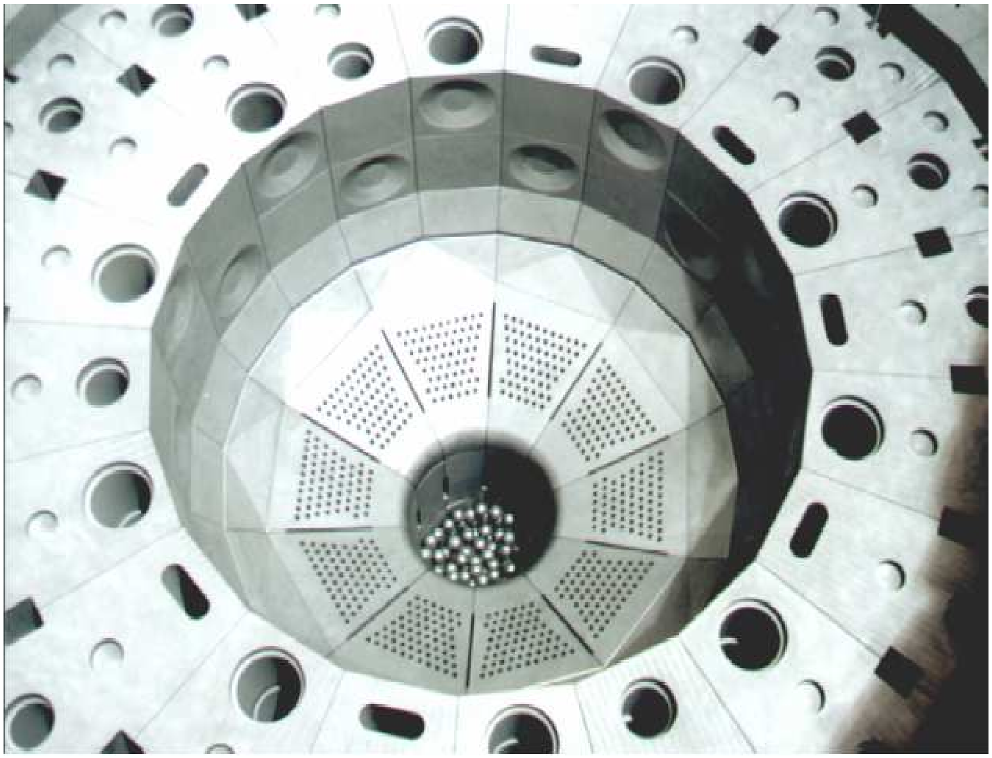

Figure 15: Pebble-bed reactor graphite structure

In HTR designs, some graphite components are replaced at regular intervals along with the fuel, such as the prismactic fuel blocks. Others can easily be changed part way through reactor life, such as replaceable reflector in prismatic concept. However, other graphite components may be more difficult to replace and therefore may have to retain their integrity for the whole life of the reactor. Therefore it's important to understand how the nuclear graphite will behave throughout its life in reactor, through graphite component stress analysis and failure predictions for the graphite core components through R&D programs.

Nuclear graphites are polycrystalline in structure being manufactured from moulded and extruded petroleum cokes and a suitable binder. Numerous tests and calculations have shown that it is virtually impossible to burn high-purity, nuclear-grade graphite [7]. Graphite has been heated to white-hot temperatures (around 1650°C) in air without incurring ignition and self-sustained combustion. After removing the heat source, the graphite cooled naturally to room temperature. Because graphite is so resistant to ignition, it has been identified as a fire extinguishing material for highly reactive metals. In Tchernobyl there was indeed no graphite fire, but after an initial steam explosion, it was the fuel, exposed to water and air at high temperature, that burnt. In fact, at high temperature, graphite oxidizes at a very low pace and the reaction is not exothermic, which explains why it cannot burn (no self-sustained combustion). In air ingress accident the slow oxidation of graphite absorbs all oxygen entering the vessel, protecting during a long period the fuel from any direct oxidation, therefore avoiding any release of its radiological content.

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3.4 Helium

Helium is a monoatomic inert gas with very good thermal conductivity. There is no chemical interaction between pure helium and materials or other gases. Cooling with helium combined with the use of graphite as moderator and core structural component does not allow to reach a power density higher than 10 W/cm3 compared to LWR (100 W/cm3). Regarding radiation protection aspects, helium is a tritium source under neutron irradiation. High purity helium is required to limit corrosion phenomena on metallic alloys and graphite. Therefore, the primary coolant purity is maintained by a Helium Purification System. This system has two purposes: to maintain the primary coolant composition in order to maximize the lifetime of reactor components and to remove radioactive impurities to control circulating activity. The purification system is composed by a series of filters, cold traps and charcoal beds to remove undesired impurities. Main contaminants to remove are:

- particulate and gaseous contaminants (H2O, CO, CO2, N2, H2, CH4) from the primary coolant
- tritium
- other radioactive contaminants from helium such as Xe, Kr, Ar (fission products)

These contaminants are presented hereafter.

3.5 Radiological source term in HTR

As in every nuclear reactor, HTR operation generates fission and corrosion products that are transported and deposited by the coolant (helium) outside the active reactor core. With regard to the core design, HTR differs considerably from LWR due to the fission product retention features of the coated TRISO fuel particle and the ceramic materials used in the core [8].

It has been demonstrated by all operated HTR that the coolant activity is rather low as compared that of water of a LWR and results in lower irradiation doses. However, contamination of end-users plants is still a concern of modular HTR designers. Among fission products produced in HTR, tritium is the main concern regarding HTR coupling with an industrial application because of its easy diffusion through heat exchange walls. Nevertheless, regarding safety and licensing aspects the integrality of the source term should be evaluated. This report addresses both issues on the basis of existing documentation.

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3.5.1 Tritium

In HTRs, tritium is the only radionuclide which can permeate from the primary circuit through heat exchanger walls into secondary circuits and there cause an unwanted contamination [9]. Since first operation of HTR demonstrators, numerous measurements have been made in the helium coolant and on the steam side. Moreover, a series of in-pile and laboratory experiments has been performed for improving the understanding of tritium behaviour in a HTR.

3.5.1.1 Tritium production

In nuclear reactors, tritium is produced by ternary fission and several neutron reactions predominantly with light elements. In HTRs, specific features induce a reduction of great variety of potential tritium forming reactions to only few ones [9]:

- Helium gas (He-3): ^{3}He (n,p) ^{3}H
- Fuel (U-233, U-235): ternary fission
- Core and reflector graphite (Li-6 as impurity): ^{6}Li (n,α) ^{3}H
- Criticality control system (B-10): B^{10} (n,2α) ^{3}H and B^{10} (n,α) Li^{7} (n,nα) ^{3}H

3.5.1.2 Tritium releases

For the indirect cycle concept, in particular for the process heat application, the amount of tritium released through the heat exchanger in the non-nuclear facility using the nuclear heat has to be as low as reasonably achievable. Compared to other releases during normal operation, tritium in this area is not diluted as it is the case for example with the releases at the stack. Depending on specific requirement for each application, the level of acceptance could be very low.

<u>To limit tritium content in process heat installation, consecutive loops with purification system included in each should be envisaged.</u>

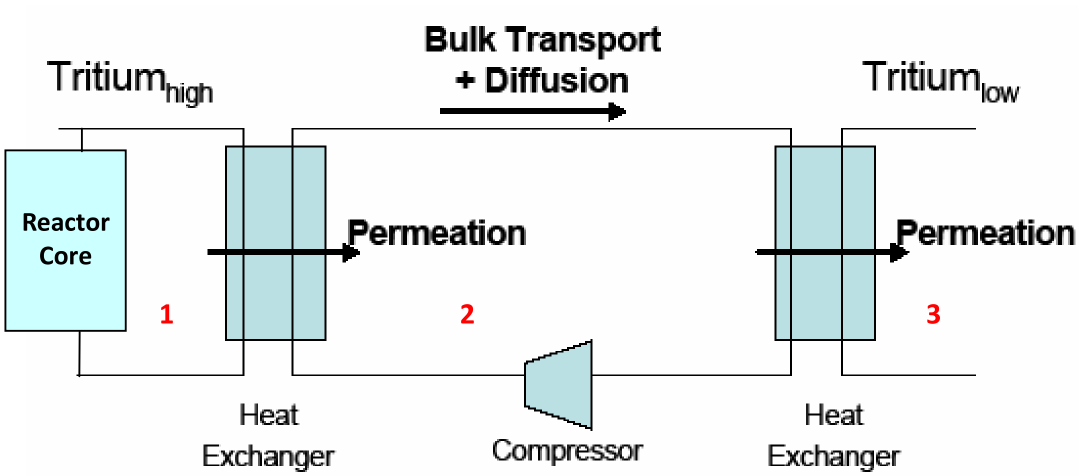

Figure 16: Schematization of 3 loops configuration and tritium permeation

With three loops, the tritium content may be reduced to less than 5% of the core production. Figure 17 below illustrates such results [16]. Steady-state operation was assumed and it was considered no adsorption or absorption of tritium by materials in system. The tritium source term was assumed to scale by reactor power level and tritium generation rate of FSV was used as reference point. According to this example (HTSE plant in NGNP) 0.3% of tritium created in the core is lost to the outside through leaks, 96.2% is removed through the use of dedicated purification systems and 3.4% is lost to process heat plant.

So the tritium material balance must be known in order to understand how tritium contamination in the downstream processes can be controlled because changing the efficacy of any removal mechanism affects the performance of tritium load on all other removal mechanisms.

For example, placing a tritium barrier between the nuclear plant and a downstream plant would eliminate the concern of contaminating downstream processes, but tritium is always being produced by the nuclear reactor, and the tritium must go somewhere if it cannot reach the downstream process. Therefore the concentration of tritium in the fluids upstream from the barrier will increase, as well as consequently, the amounts of tritium lost to leaks or removed by tritium capture. A barrier is a material, coating, or surface layer that is employed in a heat exchanger in order to reduce the permeation of tritium through the heat exchanger. The barrier must still allow efficient heat transfer while reducing the flux of tritium through the heat exchanger surfaces. Then deployment of barriers alone is insufficient to control tritium and effective tritium separation and removal systems are needed to balance tritium generation in nuclear core. Separation and removal systems have an opposite effect to barriers: they use materials that are permeable to tritium, and serve as sinks for it, so that captured tritium is not re-introduced into heat transfer fluid streams [15].

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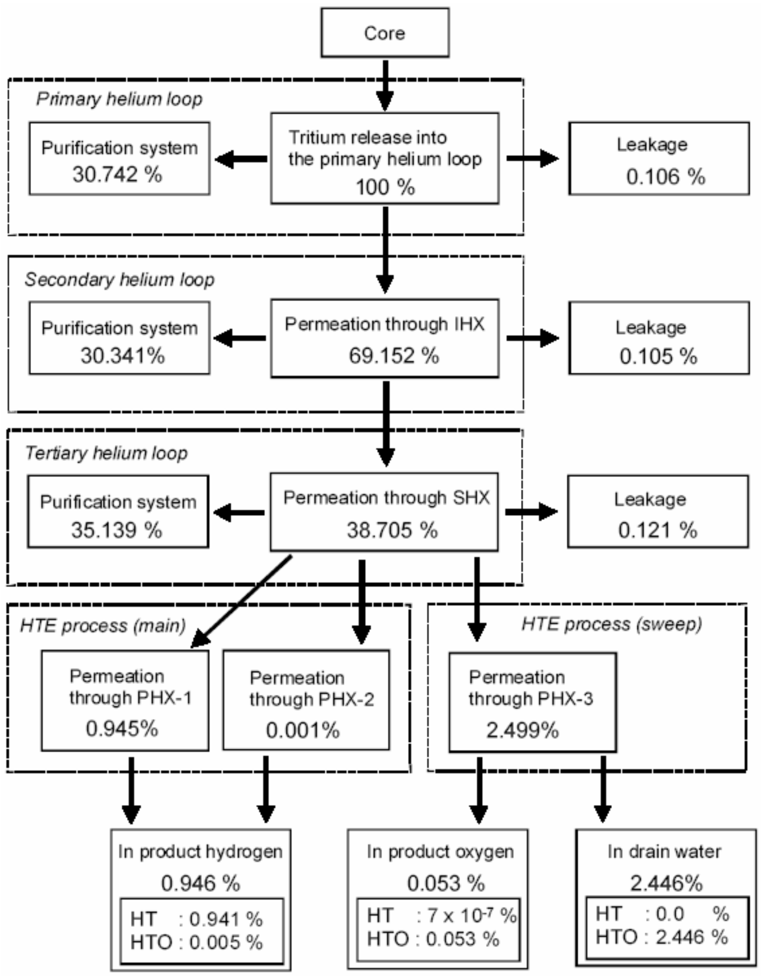

Figure 17: Tritium material balance for NGNP using HTSE process [16]

3.5.2 HTR radiological source term

AVR was an experimental reactor which aimed at testing pebble-bed reactor technology, fission and corrosion products transport, and testing of many different types of pebble-shaped fuel elements. Main statements presented hereafter deal mainly with AVR data collection [22].

Fission Product (FP) transport during normal operation is of main importance in safety examinations. Particularly, the fission products released during reactor normal operation and accumulated in primary coolant are the most relevant contributions to the source terms in accidental conditions. These nuclides may come from different origins:

- From intact[5] and defective coated fuel particles
- from uranium contamination in the graphite by diffusion through kernel, coating layers and graphite (metallic fission products)
- by diffusion out of kernels of defective coated fuel particles without further retention (nonmetals)

Condensable FPs are mainly deposited in the primary circuit. Deposition competes with sorption on graphitic dust. For example a large HTR (600 MWth) presents around 1000 m² of alloy surface: more than 500 m² for reactor vessel and nearly 300 m² for compact IHX.

[5] The intact coated particle is a very efficient barrier for nonmetals, but low melting metallic fission products diffuse through intact coatings particularly at temperatures above 1000℃.

From HTR experience coolant gas inventory presents the following radionuclides: Cs-134, Cs-137, Sr-90, Ag-110m, I-131, Xe-135, Kr-85 and H-3 as presented in 3.5.1.

Concentrations are extremely low as illustrated by AVR:

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Radionuclide	Mass concentration g.m^{-3}
^{3}H	$1.0x10^{-7}$
^{14}C	$1.2x10^{-4}$
^{131}I	$1.1x10^{-13}$
^{137}Cs	$9.4x10^{-11}$
^{90}Sr	$3.9x10^{-11}$
^{110m}Ag	$2.8x10^{-13}$

Transport and deposition in primary circuit [22]:

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Condensable nuclides released from fuel elements may be reabsorbed on fuel elements, absorbed on graphite reflectors or graphitic dust (in particular in pebble-bed reactor) and deposited on metal surfaces (heat exchangers, pipes…).

Dust plays a major role in fission product transport and will be relevant for safety examinations of future reactors too. This is due to the mobility of dust and its large specific activities. Part of the activity is present already in dust during formation from fuel elements, but the major part comes from sorption of molecular activities by the dust. Cesium, Strontium and Silver are strongly sorbed by carbonaceous materials. Maximum specific activity in AVR dust were 100 GBq/kg of Cs-137, 30 GBq/kg of Ag-110m and 400 GBq/kg of Sr-90. These activities, which are not necessarily typical for future reactors, were measured in 1986 on dust sampled from AVR steam generator.

In modern pebble-bed HTRs, friction forces in the active core are about an order of magnitude larger than in AVR (inverse helium flow, greater core height), so dust production in the active core will increase. On the other hand, future reactors will be supplied with a dust filter in the fuel handling system. Conservative design values for dust production rates in modern pebble-bed HTRs (400 MWth) are estimated to about 50 to 100 kg/year. These are overestimated values for safety evaluation of source term. In normal operating conditions minimum 10-20 kg/year are expected.

Dust behaviour is very different between pebble-bed HTRs and prismatic HTRs: the former creates more dust, then more active dust and greater surface coverage. Then distributions are fundamentally different between pebble-bed and prismatic designs due to different carbonaceous dust.

3.5.3 Radiological releases

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Modular HTRs do not need a pressure resistant containment: in case of and accidental depressurization, the radio-contamination of the primary helium dumped in the reactor building is very low and therefore it can be released in the environment, possibly through a filter, in order to avoid pressure build-up. It is only in the long term, when the reactor is fully depressurized, that the reactor building should play a role for confining the additional activity that the core releases when it is heated up to very high temperature by decay heat.

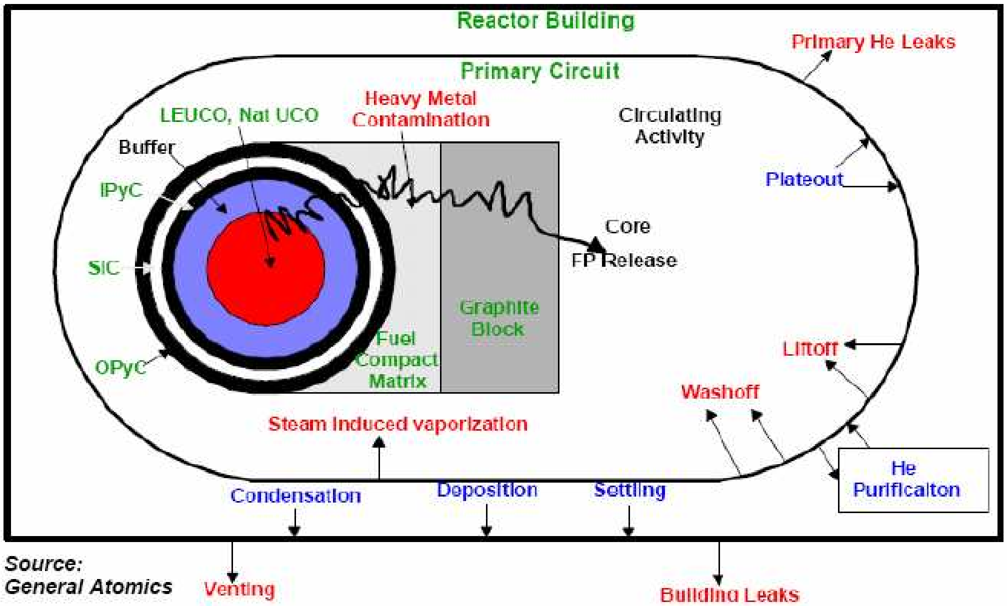


Figure 18: Phenomena governing confinement in HTRs

In case of accident like depressurization, steam / water ingress may lead to a partial release of FP accumulated in the primary coolant, but also non-metals FP release from defected coated particles (as iodine).

Core temperature increase may impact FP releases: slightly for iodine and noble gases, more significantly for metallic FP (Sr-90, Cs-137, Ag-110m) by diffusion. This was observed in AVR first with BISO fuel particles, and later experiments with TRISO particles showed better behavior. Improvements on fuel particles design and lower core temperatures reduce significantly such releases. This is considered in current HTR concepts.

4 Characteristics and technology limits

The feasibility of different HTR concepts is strongly affected by the temperature levels. Fuel performance, selection of reactor materials and design limits, environmental effects on materials, resulting design complexity (first of a kind commercial reactor), and components lifetimes are all strongly affected by the reactor outlet temperature. Moreover for limiting peak fuel temperature increase, rising the outlet temperature typically also involves increasing the reactor inlet temperature. Then increasing the system operating temperatures also tends to increase transient and accident temperatures slightly.

In this chapter are presented the design and main technology limits of some relevant nuclear heat supply system components: steam generator, IHX, blower and supercritical steam generator. For these components, the following points will be addressed: principle, technological maturity, performance and operational limits (current and future), safety aspects.

Other NHSS components may also present significant technical challenges, although it is considered that they do not impose additional limits to the HTR modular reactor feasibility. Rev.1 The availability of forgings of large diameter (e.g. 7.3 m) required for the reactor vessel, which is bigger than in an LWR, has to be confirmed, as well as expected transportation limitations which could impose a final reactor vessel assembly on site (welding processes to be qualified). Particular attention has to be paid also to the fuel handling and storage systems. Contrary to LWRs the reactor vessel remains closed during refueling of a prismatic HTR in order to avoid the oxidation of the graphite components. This requires the implementation of an original and relatively complex system for handling the fuel elements in and out of the vessel. The performances and reliability of this system are of prime importance to the plant availability.

In this part are also presented some power conversion systems and plant configurations.

4.1 Conventional Steam Generator [18]

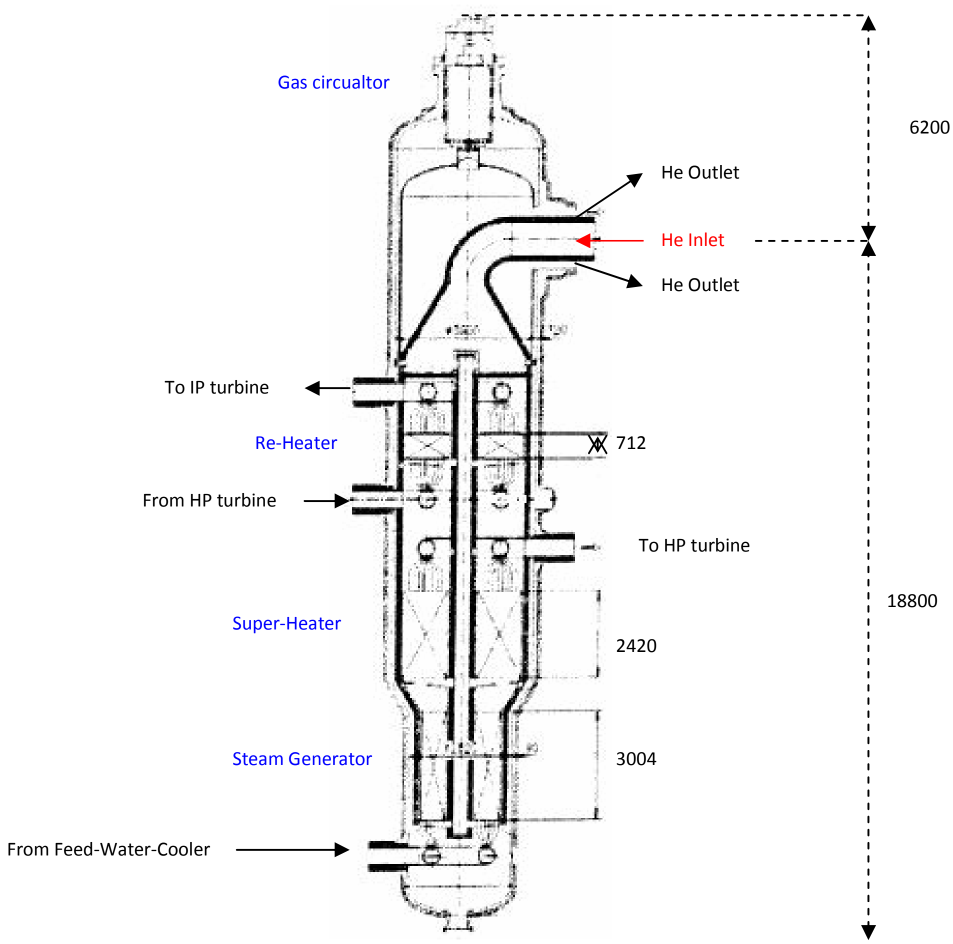
It is understood that the reference to “conventional” in SG terms implies “sub-critical” and consequently this is the context of the information provided here.

In addition, the limits provided hereafter are given for gas to steam nuclear convectional SG and not gas-fired- or pulverized coal fired- radiant SG.

4.1.1 Principle

Rev.1 The SG in a nuclear power plant provides nuclear heat removal from the reactor by converting water into steam. In conventional SG for HTR (generally shell and tube), helium gas is separated by a metal wall from the water. Therefore its structure is similar to water LWR SG structure. The SG tube bundle is the physical separation (barrier) between the primary helium loop and the secondary coolant (water) loop, as well as the heat exchange wall between these two fluids.

First, to deal with SG requirements and constraints, it is instructive to examine the design of a SG. As example the **Erreur ! Source du renvoi introuvable.** below presents the design of a MHI SG.



Rev.1

Figure 19: Schematic cross section through MHI steam genarator

The economizer part is the one in which the feed-water is heated by the primary coolant from its initial temperature to the saturation point. The evaporator is the one in which the water is vaporized by the primary coolant without temperature change until it becomes dry steam (transfer of latent heat). In the super-heater, the dry steam from the evaporator is heated by the primary coolant to increase its temperature until the expected superheated steam temperature. A re-heater can be added in which a steam extracted from an intermediate turbine stage is heated again by the primary coolant, so as to increase the overall thermodynamic cycle efficiency.

As with any functional equipment, there are some requirements which define performance and some other ones which constrain the design. Design of the SG includes hydraulics, heat transfer and stability considerations. Based on the design requirements, it is necessary to ensure sufficient heat removal from the reactor in order to maintain stable operation.

4.1.2 Technological maturity

4.1.2.1 History

The use of SGs within nuclear power plants, both low temperature (e.g. LWR) and intermediate temperature (e.g. AGR) is a common practice and there exists a significant database of performance and integrity data. In the UK, the AGR power stations use the highest steam parameters of nuclear reactors in full commercial operation, although they use CO_2 as a coolant whereas HTR use helium (see Table 3). The earliest plant has recorded 30 years service.

OPERATING PARAMETERS AT 100% CMR LOAD

Gas flow/reactor	4203 kg/s
Gas pressure	41 bar a
Gas temperatures:	
Inlet to reheater	615°C
Inlet to HP boiler	573°C
Outlet from HP boiler	290°C
HP steam flow/reactor	500 kg/s
HP steam outlet	540°C/166 bar a
HP feed temperature	156°C
Hot RH steam after attemperator	538°C/41 bar a
Cold RH steam	342°C/43 bar a

Table 3: AGR SG operating parameters at 100% CMR load[6]

Rev.1 Five out of the seven HTR that have been constructed (see **2.2**), including test reactors, and the grid connected THTR-300, delivered their heat to a SG system. In all cases, the SG was sub-critical. The reference for HTR SG technology comes predominately from the following plants with their associated SG parameters as listed in Table 4 below. Some of them have been presented in paragraph **2.2**.

	AVR	Peach Bottom	Ft. St. Vrain	Fulton	THTR-300	HTR-500	VGM-400	HTR-Module	MHTGR
Country of Origin	Germany	U.S.	U.S.	U.S.	Germany	Germany	Russia	Germany	U.S.
Thermal Power, MWt	46	115	842	2,979	750	1,390	1,060	200	350
Net. Electric Power, MWe	13	40	330	1,160	300	550	Co-Gen.	80	139
Core Outlet Temp., °C	950	725	775	741	750	700	950	700	686
Helium Pressure, MPa	1.1	2.25	4.8	5.0	3.9	5.5	5.0	6.6	6.4
Steam Temp., °C	505	538	538/538	513	530/530	530	535	530	538
Reactor Type	Pebble	Sleeve	Block	Block	Pebble	Pebble	Pebble	Pebble	Block
Vessel Material	Steel	Steel	PCRV*	PCRV	PCRV	PCRV	PCRV	Steel	Steel
Date of Operation	1966 1988	1967 1974	1979 to 1989	No operation	1985 1989	No operation	No Data	No operation	No operation

*Prestressed concrete reactor vessel.

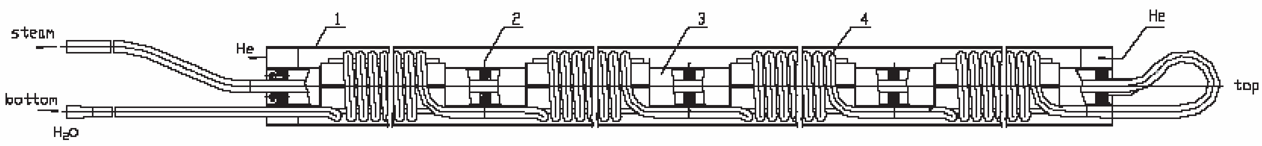
Table 4: Characteristics of steam cycle HTRs[7]

The IAEA has an International Working Group on Gas-Cooled Reactors (IWGGCR) which collaborates in an on-going basis to consider the technology and issues around the design, operation and application of HTRs. In this frame, at the end of the 80s, number of papers have been issued covering the various aspects of the technology of HTR SGs and consolidated in an IAEA proceeding dossier [10].

[6] Heysham and Torness plants

[7] Boyer et al. 1974, Brey 2000, Nickel 1989

The operating experience base has been since expanded with a number of additional years of operation of all the AGRs including the addition of Heysham I and Hartlepool to the fleet. In the realm of helium cooled HTRs, the THTR was since commissioned and decommissioned and added a number of years to the operating plant data base while the AVR continued operating up to 1988. In China the10 MW High-Temperature gas-cooled Reactor (HTR-10) has also added to the database of HTR SG operating experience (see section **2.2.2**). The HTR-10 is equipped with one once-through SG including 30 modular heating helical tube assemblies (Figure 20).



1: Casing tube, 2: Annular cavity, 3: Central tube, 4: Helical tube

Figure 20: Structural scheme of HTR-10 SG modular assembly
Heat transfer tube set structure

4.1.2.2 Performance deterioration and lifetime

Rev.1 The reference for the technology base on SG in HTR's plant comes predominately from the plants listed in Table 4 above. Additionally a quick summary is presented here a quick summary of the feedback from PWRs SG experience, as it shows that, even if SG ageing is a main concern for utilities, they learnt to manage it so that it does not lead to an early decommissioning of the whole plant.

The SG heat transfer tubes are the weakest links in the primary loop of a nuclear power plant. SGs are sensitive to corrosion. Localized corrosion and mechanical wear problems are observed on LWRs SG tubes. To limit leakages from primary to secondary side, failing tubes are plugged. One consequence of tube plugging is heat transfer properties deterioration, so that, beyond a certain plugging rate, the replacement is needed for the plant in order to maintain full power.

Depending on the SG tube material the sensitivity to corrosion differs. Developments over last years on SG tubes material have been oriented to limit cracks and tubes failure. Then for recent SG the expected lifetime approaches 40 years. For older PWR SG, it is expected that during the 40 years of the initial license period, replacement will be needed at least once. As an indication it was stated in [12] that in 2001 average lifetime of SGs was 14 years, with the shortest life being 8 years. Other criteria than material should also be considered, e.g. primary and secondary chemistry as well as plant management (transients, load following…).

Rev.1 Nevertheless even if these considerations are showing that SG early ageing does not lead necessarily to an early reactor decommissioning, the data obtained for PWR SG cannot be directly transposed to HTR SG, as the primary water coolant is replaced by helium. Moreover, pressure and temperature levels are different, so that lifetime expectations.

In order to avoid radioactive releases (see 0) leak detection methods should be defined as well as cracks detection to identify leakage before heat transfer tube rupture occurs. These methods strongly depend on the SG design and operating conditions.

4.1.2.3 Expected developments

In on-going R&D projects, developments on SG material are driven by efficiency and economics concerns as well as by the need of understanding tube break mechanisms and propagation of cracks for the selected materials.

4.1.3 Performances and operational limits

4.1.3.1 Steam Generator design issues

As explained above in paragraph 4.1.1 'Principle', SG design should deal with specific requirements and constraints.

The primary requirement of a SG is to produce high pressure superheated steam at convenient temperature and pressure characteristics. The following requirements have to be added:

- High availability,
- High duty cycle over lifetime,
- Broad operating point range,
- Responsive.

Additionally to these functional requirements the following constraints should be considered by designers:

- Low risk of water ingress into the reactor (safety constraint),
- Licensable,
- Long operating life,
- Transportable,
- Inspection,
- Maintainability (including ability to be repaired in case of unacceptable leakage).

Through consideration of the constraints and operational requirements listed above, the following issues are raised:

- Material capabilities: Creep-rupture characteristics – this limits the temperature / pressure combination that can be applied. Typically the high temperature SG tubes would be made of Alloy 800H. Though there are precedents for a SG with a helium temperature of 750°C and above (AVR, THTR), if the tube mean wall design temperature exceeds 650°C, a shorter design life or a higher performance alloyed material would have to be used (see Appendix 1).
- Licensable The ease of licensing of a plant is related to the validity of the application of the materials relative to previous designs that have been operated. Typically the use of materials and designs within the parameters covered by ASME or RCC-M code provides a solid basis for licensing. This imposes temperature and pressure limits on the design as well as design lifetime limits.
- Materials welding: Due to the need to control costs while using materials which are easier to work with during manufacture, materials of differing capabilities might be used within the SG, thus there can be bi-metallic welds. These welds might pose constraints on the temperatures and pressures at which the SG can operate.
- Transportability: The maximum size of the SG will typically be bounded to allow transportability such that the "on-site" pressure vessel assembly can be minimized. This translates to thermal power limits for each SG unit, therefore into the number of SG required for a reactor of a given power. Certain flexibility in this limit can be obtained by increasing the temperature differential between the "hot" and "cold" sides.

Rev.1

- Inspection: As a primary pressure boundary for the nuclear heat supply system, the SG needs to be examined and controlled so as to reduce the probability of failure during operation. This imposes constraints on the technology to be used, implying that the SG will rather be of the shell-and-tube type, rather than of plate (compact) type. In addition, the tubes should rather bare (no heat transfer enhancement) for an easier inspection and ease of manufacture (though there is precedent for heat transfer enhancement features in some AGR SG).
- Availability: This demands high reliability, low capacity loss due to reduced efficiency (fouling) and low leakage. Typically this constrains the design through the application of conservative design decisions and margins as well as the application of simplicity in the design, such as avoiding the complexity introduced through the addition of a reheat cycle within the SG.

Rev.1

- Configuration: HTR SGs are of shell-and-tube type. Mainly due to the higher pressure of the steam side and to the low pressure drop requirement of the helium side (limited by primary blower technology considerations), a configuration with the steam situated inside the tubes is preferred in HTR SGs. In some cases, the SG is located lower than the reactor for improved accident scenario tolerance.

4.1.3.2 Constrained technology limits

In order to meet the performance requirements while satisfying the constraints, the technology to be applied in the design of the SG constrains the performance, while the constraints temper the technology to be applied.

Rev.1

On the basis of past technology optimization a number of parameters may be taken as the current technology limits (see Table 5 below). Specific requirements of process heat (Table 5 shows values typical of electric oriented installations) could also tend to decrease the requirements on certain parameters, allowing reaching more challenging values on others. Generally, future innovation may relax some of the current design limits.

	Parameter	Typical of 750°C ROT[8]
Primary Side	Fluid	Helium
	Inlet Pressure, MPa	~7
	Inlet Temperature, °C	~750
	Outlet Temperature, °C	~230
	Outlet Pressure, MPa	~6.7
Secondary Side	Fluid	Water
	Superheater Outlet Pressure, MPa	~19
	Superheater Outlet Temperature, °C	~570
	Feedwater Temperature, °C	~180
	Feedwater Pressure, MPa	~21
	Reheat	No

Table 5: Typical conservative steam generator performance parameters

A helical-coil type SG is generally used as this provides a high flexibility of the tubes to accommodate thermal expansions in operation.

Overall, available technology limits and past experience tend to define a maximal power of around 300 MWth for today's HTR SGs.

4.1.3.3 Safety considerations

In a typical HTR, SG connects and isolates the primary radioactive helium loop from the secondary non-radioactive water and steam loop. Therefore, particular care shall be taken to control potential entry of water into the reactor primary circuit in case of steam generator leak. Water ingress has occurred in past experience (notably in AVR and FSV). The consequences in terms of graphite corrosion and radiological releases have been shown to be manageable. Future HTR designs shall however carefully take into account the management of water ingress risks.

[8] ROT: Reactor Outlet Temperature

4.2 Supercritical Steam Generator [19]

4.2.1 Principle

Supercritical water is defined by conditions of temperature and pressure in excess of 374°C and 22.1 MPa, respectively. Above the critical point, a fluid exhibits no further phase change and has intermediate properties between a gas and a liquid. Of importance for the nuclear heat coupling application considered here, supercritical water possesses liquid-like neutronic properties and heat transfer characteristics, together with gas-like diffusion rates. In addition, the solubility behavior of water changes dramatically either side of the critical point. As a result, supercritical is a strongly non-polar solvent with high solubility for hydrocarbons and low solubility for ionic compounds. Supercritical water is highly corrosive, particularly when dissolved halogen compounds are present.

The potential for using supercritical water to transfer heat between HTR and industrial processes can be assessed with reference to the previously constructed HTR SG, as well as the boilers employed in modern fossil fuel-fired power stations. In addition to being a candidate heat transfer medium for use in intermediate coolant loops, supercritical steam may be also suitable for cogeneration applications as supercritical steam turbines designs exist. Furthermore, supercritical water may be employed directly in certain industrial processes such as hydrogen production by steam reforming of ethanol.

Rev.1 Supercritical SG are frequently used for the production of electric power in conventional (non nuclear) power plants. In contrast with "subcritical" (conventional) SG, a supercritical SG operates at such a high pressure that 'boiling' ceases to occur, there is no liquid water - steam separation, no generation of steam bubbles within the water, because the pressure is above the critical pressure at which steam bubbles can form. It passes below the critical point as it does work in a high pressure turbine and enters the generator's condenser. This results for non-nuclear plants, in increased thermal efficiency and therefore less fuel use and therefore less greenhouse gas production.

As main interest, in addition to enabling high overall efficiency, supercritical OTSGs are also capable of delivering heat at relatively constant temperature, irrespective of changes in load. Under sliding pressure operation, load changes of the order of 4-6% per minute may be met through adjusting water feed and firing rates with only limited changes in steam temperature.

4.2.2 Technological maturity

4.2.2.1 History

Rev.1 Contemporary supercritical SGs are sometimes referred as Benson boilers[9]. Safety was the main concern behind Benson's concept. Earlier SGs[10] were designed for relatively low pressures of up to about 100 bar, corresponding to the state of the art in steam turbine development at the time. In 1929 the first test boiler operated in subcritical mode in the thermal power plant Gartenfeld in Berlin and the second one began operation in 1930 without a pressurizing valve at pressures between 40 and 180 bar at the Berlin cable factory. This application represented the birth of the modern variable-pressure Benson boiler. Rev.1 The first power plant operated with advanced supercritical steam conditions was Philadelphia Electric Company's Eddystone Unit 1, commissioned in 1960.

During the past 50 years, significant improvements in coal power plant thermal efficiencies have been achieved, principally though increases in steam temperature and pressure – and the associated reduction in heat transfer losses. Progress has generally resulted from the introduction of successively more advanced steel alloys in SGs.

Today, pulverized coal power stations with ultra supercritical steam parameters of the order of 600°C and 31 MPa, with corresponding efficiencies of up to 49%, are in commercial operation. In addition, international efforts to raise the steam conditions of fossil fuel power stations to 700°C have been undertaken.

Rev.1 The coupling of nuclear reactors to supercritical SGs has previously been proposed as an option for lead cooled and sodium cooled fast reactors, and for intermediate-cycle supercritical LWR. However, no prototype nuclear-heat supplied supercritical SG has been constructed and its feasibility in the nuclear field is not yet established. Several high-temperature power generation and heat transfer cycles have been proposed for high temperature gas cooled reactors, including direct Brayton

Rev.1 [9] The term "boiler" should not be used for a supercritical pressure steam generator, as no "boiling" actually occurs in this device.

[10] In 1922, Mark Benson was granted a patent for a boiler designed to convert water into steam at high pressure.

cycles and intermediate transfer loops employing supercritical CO_2. However, all of the HTRs actually constructed have delivered heat to SGs operating at temperatures significantly lower than the reactor's intrinsic capabilities.

4.2.2.2 Performance deterioration and lifetime

Performance deterioration and lifetime of this component are linked to materials limitations. This is discussed in the following sections.

4.2.2.3 Expected developments

Rev.1

The development of HTR SG suitable for steam conditions in excess of 650°C will need to employ high temperature alloys for thick section components. In order to mitigate risk, it is recommended here to build the alloy selection and the SG design on the extensive fossil fuel-focused experience. Additional work is necessary to characterize the long-term stability of strengthening precipitates and phases in the candidate alloy systems. In particular, a better understanding of the effects of changes in microstructure during heat treatment and welding are required. Solid solution strengthened alloys are easily welded; however lack strength at temperatures above 700°C. Steam parameters of 700°C and 38 MPa are conceivable with additional developments.

4.2.3 Performances and operational limits

4.2.3.1 Design issues

The supercritical SG used in advanced fossil fuel fired power plants, are exclusively of once-through configuration. In this design, preheating and superheating are achieved in a single pass through the heating tubes.

A typical coal-fired OTSG (One-Through Steam Generator) shown schematically in Figure 21, consists of a spirally wound or vertical water-wall for removing heat from the furnace walls, superheater and reheater piping, and a superheater header consisting of a thick section pipe with tube penetration. At the outlet end of the superheaters, the tubes connect to thick section pipes or headers which penetrate the vessel. Since heat transfer takes place to a supercritical fluid, forced convection is not impaired by boiling.

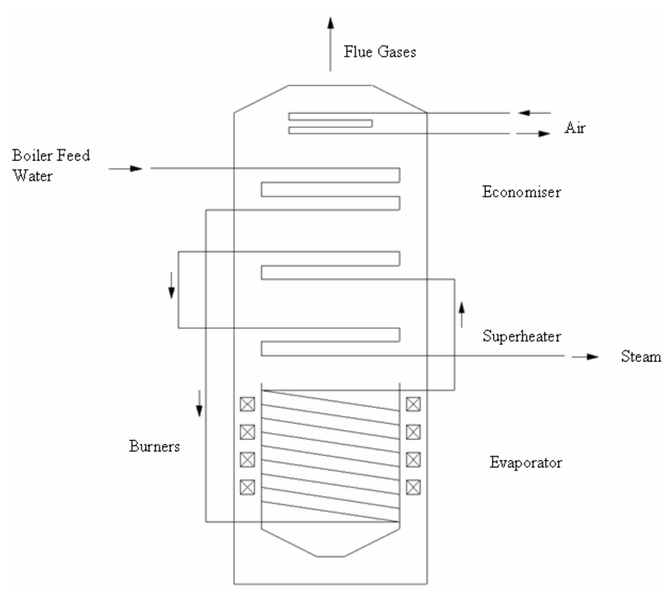


Figure 21 : Schematic cross section through a typical coal-fired Benson boiler

4.2.3.2 Technology limits

Main technology limits for the supercritical SGs concern material selection. In both nuclear and fossil fuel fired plants, the performance limits of SG are principally set by the high temperature creep strength, thermal fatigue resilience and oxidation resistance of the materials used for thick section headers and piping.

The majority of issues relating to high temperature creep performance and steam-side oxidation resistance are shared between HTR SGs and fossil-fuel fired plants SGs, despite important differences arise in terms of requirements for corrosion resistance: the corrosive environment faced in a fossil fuel-fired plant consists of combustion gases. In a HTR, resistance to helium corrosion is important. The **Appendix 1** presents numerous design and material issues that will need to be resolved before deploying a supercritical steam cycle on a HTR.

Rev.1

The opportunity to build on the existing operational experience acquired in fossil fuel plants, as well as the on-going and extensive research programs into 700°C SGs, makes supercritical steam/water a potential heat carrier for HTRs. In terms of material selection, there is sufficient evidence to suggest that supercritical SGs can be considered practical at 600°C within the existing nuclear codes and with relatively low risk. To practically increase the temperatures above this would require higher alloy materials, at greater costs and risk in operating experience.

4.2.4 Safety considerations

Superheated steam presents unique safety concerns however, if there is a leak in the steam piping, steam at such high pressure/temperature can cause serious, instantaneous harm to anyone entering its flow. Since the escaping steam will initially be completely superheated vapor, it is not easy to see the leak, although the intense heat and sound from such a leak clearly indicates its presence.

4.3 Intermediate Heat Exchanger

4.3.1 Principle

As the Steam Generator, the Intermediate Heat Exchanger (IHX) separates the primary loop from the secondary loop or power conversion system. Its main objectives are to transfer energy from primary to secondary loop with high efficiency (very low pressure drop, low heat losses), to operate at high temperature, and to be as compact as possible. It must also prevent water and process gas ingress into the nuclear core and minimize hydrogen and tritium permeation between the primary and secondary loop.

Two kinds of technology should be distinguished: the tubular IHX and the compact IHX.

4.3.2 Technological maturity

4.3.2.1 History

Even if being first of a kind for large scale modular HTR, IHX are already used in conventional industries and benefits from a significant test experience in nuclear industries, e.g. tests of a 10 MWth mock-up (Helium/Helium) in the Komponentenversuchskreislauf (Component test facility KVK) facility in Germany by INTERATOM in the frame of the Prototype Plant Nuclear Process Heat (PNP-project). This mock-up was coupled to a conventional heat source reaching gas outlet temperatures up to 950°C [14].

Rev.1

Another demonstration of tubular IHX technology for HTR was in the Japanese HTTR reactor by the Japan Atomic Energy Research Institute (JAERI) as the program focused on heat applications. Components tests for the IHX were conducted to confirm the structural integrity, but the tests duration was not long enough to conclude regarding the lifetime of such component. This was a 10 MWth Helium/Helium IHX. Overall view of the component is given in the Figure 22 below. It consists in a vertical helically-coiled counter flow type heat exchanger in which primary helium gas flows on the shell side and secondary helium gas in the tube side as shown on the figure below [13].

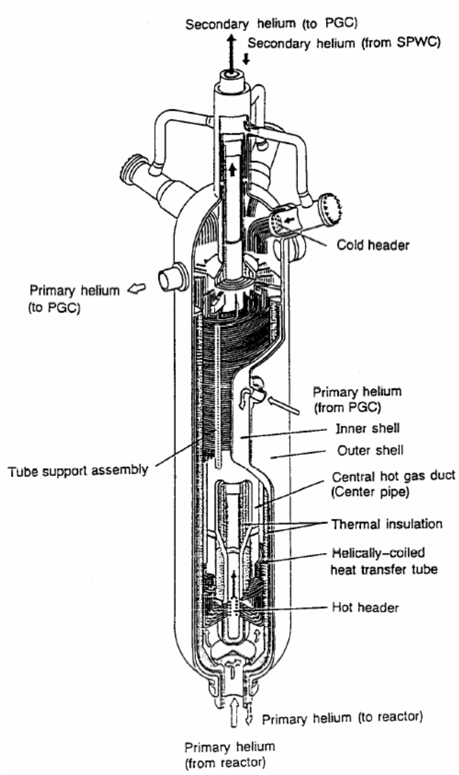


Figure 22: Overall view of He/He IHX Japanese HTTR

Major specifications of this IHX are given in the following Table 6:

Parameter	Rated operation	High temperature test operation
Maximum pressure: - Outer shell - Heat transfer tube	 4.8 MPa 0.3 MPa (differential pressure)	
Maximum temperature: - Outer shell - Heat transfer tube	 430°C 955°C	
Primary flow rate	15 t/h	12 t/h
Inlet temperature of primary helium gas	850°C	950°C
Outlet temperature of primary helium gas	390°C	390°C
Flow rate of secondary helium gas	14 t/h	12 t/h
Inlet temperature of secondary helium gas	300°C	300°C
Outlet temperature of secondary helium gas	775°C	860°C
Heat capacity	10 MWth	
Heat transfer tube: - Number - Outer diameter - Thickness	 96 31.8 mm 3.5 mm	
Outer diameter of shell	2.0 m	
Total height	11.0 m	
Material: - Outer and inner shell - Heat transfer tube - Hot header and hot duct	 2¼Cr-1Mo Hastelloy XR Hastelloy XR	

Table 6: Major specifications of HTTR's intermediate heat exchanger

4.3.2.2 Performance deterioration and lifetime

The temperatures and environmental operational conditions means that materials other than those commonly used in heat exchangers are required. The IHX will be operated in flowing, impure helium on primary and secondary sides at elevated temperatures (up to 750°C in early configuration) which may cause oxidation and carburization / decarburization [20].

There are major high temperature design, materials availability, and fabrication issues that need to be addressed. Currently the prospective materials are Alloys 617, 230, 800H and XR with Alloy 617 being the leading candidate for use above 800°C [14]. Alloy 800H has been used in previous HTR programs for heat exchangers and SG. Alloy 617 has not been used extensively in the programs but benefits from German and US experience. It has been characterized in gas reactor programs in the 1980's.

In the past, integral tests of 10 MWth IHX mockups were made in Germany on U-type and helical tubular concepts, in reactor conditions (including temperature transient cycles untill 40°C/min, and pressure differences of up to 43 bars). Helium temperature was about 900°C at the secondary hot header, and pressure about 40 bars. Operating times totalized 5000 hours for the helical concept and 4000 hours for the U- tube concept, without any failure of the components. It was stated that life expectancy of such designs is more than 10 years (10^5 hours) in reactor conditions for thermal and electricity applications. Service life is expected to increase substantially if the maximum temperature decreases: more than 30 years below 800°C for tubular concepts. The level of temperature has a bigger effect than the occurrences of transients on the lifetime expectancy.

Rev.1 Regarding compact type IHX concepts, the lifetime expectancy appears to be reduced. Besides corrosion effects, stress levels on existing compact IHX concepts are more severe than with tubular IHX. Design improvements for reducing notably thermal stresses seem nevertheless to be possible. Service lifetime is expected to double for each 50°C below 850°C: likely about 10 years at 800°C and 20 years at 750°C (primary temperature).

4.3.2.3 Expected developments

The section 4.3.4.2 below presents the technological limits of the tubular and compact IHX concept. Main expected developments are directed towards improvements regarding materials and lifetime of the plate IHX.

Rev.1 The feasibility of tubular concept has been demonstrated by the Germans, but this type of heat exchanger is very bulky and therefore expensive. In a general way, improvements and developments have to focus on the compactness of such component. For this reason, several concepts of plate heat exchanger used in the industry have been examined (PMHE, PFHE and PSHE – see 4.3.4.2.2) and were analyzed in terms of manufacturability and service behavior, but the calculations about their lifetime have to be confirmed by tests. The real asset of such technologies is their compactness and consequently reduces their prices compared to tubular IHX, but they will certainly have a shorter lifetime. So the choice of the IHX concept will be on the best compromise between cost and lifetime.

To be exhaustive, in the frame of technological developments, particularly innovative concepts of heat exchangers such as foam IHX, capillary IHX and ceramic IHX [14] have been looked at. Among these concepts, none has finally proved acceptable for HTR application, at least in the short and medium terms.

4.3.3 Performances and operational limits

HTR IHXs are subject to several stringent requirements. Among them:

- effectiveness >95%
- compact size
- very low pressure drop
- medium to high temperature capability
- high pressure capability
- high integrity to withstand transients

Among all the components, the IHX appears as representing the highest technical challenge. In this part both technologies of IHX (tubular and compact concepts) will be discussed.

4.3.4 IHX design issues

4.3.4.1.1 Tubular IHX

The tubular IHX is based on a classic heat exchanger design, with either U-type or helical tubes. Today, it is judged feasible to build IHX modules of 150 MWth in Helium (both sides) based on feedback. Maturity seems high enough for extrapolation to 300 MWth. However, it is preferable to use smaller modules to reduce size and pressure losses below 2 bars in each IHX (especially if He/N2 or He/Ar mixture is used on secondary loop). One module of 150 MWth fits in a cylinder of 30 m height and 4 m diameter approximately.

Inspection and repair of this component is possible, and does not represent a technological challenge.

4.3.4.1.2 Compact IHX

Compact heat exchangers provide large performance advantages compared to conventional (tubular) ones in terms of maximizing heat transfer effectiveness and minimizing heat exchanger volume. The reference is a plate-type compact IHX.

Rev.1

From a thermohydraulic point of view, 600 MWth can be housed in one single pressure vessel [14]: around 25 m^3 for the core IHX and 600 m^3 for the global IHX pressure vessel. The core IHX is made of several small modules, each of them made of hundreds plates (see schematic representation on Figure 23 below). The corresponding tubular solution would require 4 tubular IHX modules of 150 MWth (4 IHX vessels, 4 cross vessels and 4 circulators) and a global volume of about 1500 m^3, and will be clearly more expensive.

The Figure 23 below illustrates such configuration:

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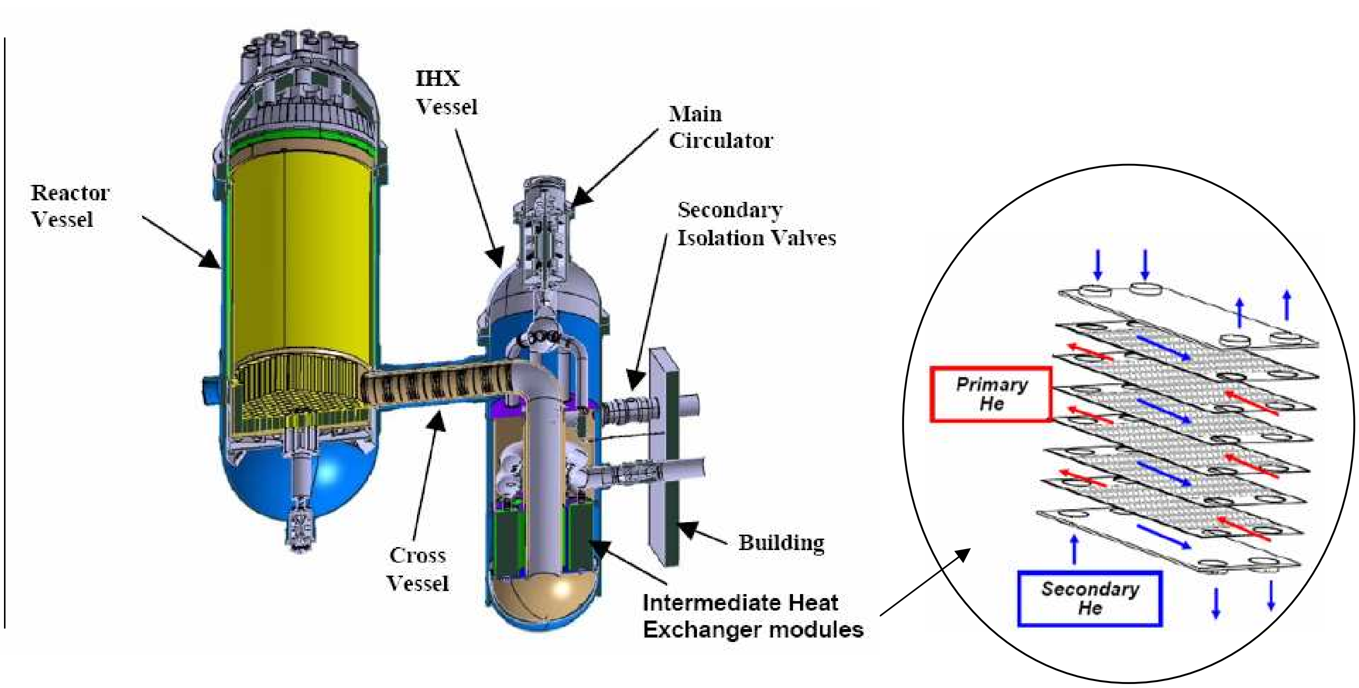

Figure 23: Primary Heat Transport System with IHX
(ANTARES design – 600 MWth)

Metallic plate IHXs are used in many applications but the technology and the materials are not the same as those required for applications at high temperatures. There is no commercial plate IHX for high temperature (≥650°C) in operation today. Above 800°C, such applications require the use of high temperature nickel base alloys that are not used in the conventional industry applications. So their manufacturability and their availability to withstand the pressure and thermal loads as well as corrosion by impure Helium during a significant lifetime at high temperatures have to be demonstrated.

Among compact type IHX, one can mention: Plate Machined Heat Exchanger (PMHE), Plate Fine Heat Exchanger (PFHE) and Plate Stamped Heat Exchanger (PSHE). It is stated in [14] (as AREVA contribution) that PSHE concept is the most promising one, even challenging improvements are still needed. The proposed concept is a modular IHX, plate type compact heat exchanger able to provide around 10 MWth for each module. Each module can fit in a box of 1 m x 1 m x 0.5 m approximately.

PSHE consists of a set of modules, each being composed of stacking of plates stamped with corrugated channels. The plates are stacked in such a way to cross the channels of two consecutive plates and, therefore, to allow the different channels to communicate through the width of the plate as shown on the left below (Figure 24).

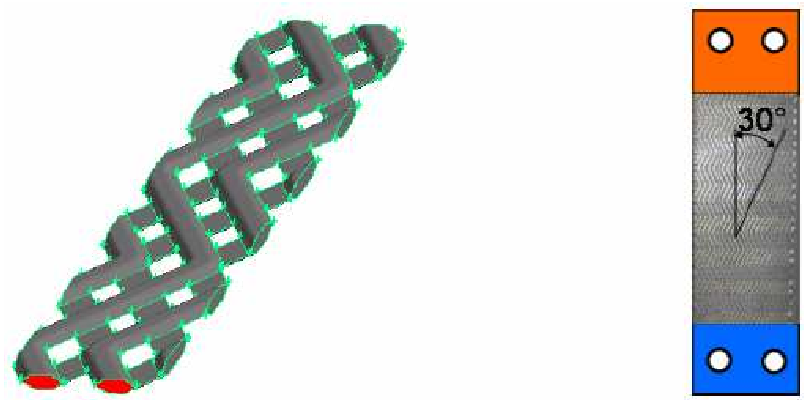

Figure 24: Plate stamped heat exchanger concept

The assembly of the plates together is performed by welding on the edges of the plates only. No joint is performed in the active part of the plates, which gives to the module a relatively good flexibility.

The location of the welded joints is also favorable to inspection, even if this remains a difficult issue.

Lastly the thickness of the PHSE plates is the largest among other metallic plate type IHX (1.5 mm) which allows a better resistance to corrosion. Additionally the PSHE concept presents the best compactness of plate type IHXs.

The table below provides comparison of the different IHX concept alternatives [14].

	Maturity	Stress behavior	Sensitivity to corrosion	Compactness
PMHE	Numerous developments in conventional industry	High stress levels. 5 years lifetime seems very challenging	Sensitive	26 MW/m^3
PFHE	Numerous developments in conventional industry	High stress levels. 5 years lifetime seems very challenging	Very sensitive	24 MW/m^3
PSHE	Numerous developments in conventional industry	Challenging but best stress accommodation among the plate IHXs	Sensitive	35 MW/m^3
Tubular IHX	Industrial components in operation	Limit of state of the art	Better than plates but still sensitive	0.4 MW/m^3

Table 7: Comparison of IHX concept alternatives

4.3.4.2 Technological limits

4.3.4.2.1 Tubular IHX

The main technological limits of this component are creep (long term effect) and thermal fatigue (transients) of welding joints at the tube/central hot header junctions. Helical tubes can sustain higher thermal displacements in normal and transient operations due to the flexibility of the tubes. Flow repartition might be a problem on very large tubular IHX.

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For application above 750°C (secondary side) such IHX concept is foreseen to be made of Nickel base alloys (Alloy 617 or Alloy 230) able to sustain high temperatures and corrosion for the targeted duration of 30 years (replacement at least once during the reactor lifetime). With these alloys it is however considered that a corrosion risk by impure helium exists at primary and secondary sides, for operation at 800°C and should significantly reduce the IHX lifetime above 900-950°C. Nevertheless the tube wall (about 2mm) in IHX tubular design is the thickest as compared to other concepts which makes this concept comparatively less sensitive to corrosion. As regards thermomechanics, the tube bundle design is most favorable to accommodate the thermal expansion and to spread the gradients over long lengths. But with temperatures above 900°C, the limits of material resistance with respect to creep will have to be checked [14].

For a given design (especially number of tubes allowed by manufacturing limits), the maximum power of a tubular IHX is generally given by the maximum secondary flow rate, given by the maximum pressure drop allowed by the secondary loop design and blowers (pressure drops at secondary side are much higher than at primary side). This limit is lower in case of the use of gas mixtures (thermally less efficient) at secondary side. These considerations leads to recommend the use of four tubular IHX of 150 MWth each for a 600 MWth reactor and with secondary gas mixture, while a single 600 MWth plate IHX vessel would be sufficient. When using helium inside the secondary loop for heat applications, the unit-power for one tubular IHX may be enlarged up to 300 MWth.

Despite limits stated above, the tubular IHX remains at the time being and in near term applications the most feasible concept to transfer heat to the PCS [14] compared to compact IHX.

4.3.4.2.2 Compact IHX [20]

Compared to tubular IHX concept, plate IHX concepts remains at the time being challenging and should not be mature enough for near term application. Material candidates are Nickel based alloy (Alloy 617, 800H, or Alloy 230) able to withstand high temperatures and corrosion effects for long term duration.

The main technological limits of this component are creep (long term effect), corrosion and thermal fatigue (smooth transients with low temperature ramp are preferable). This is currently limiting applications to about 850°C.

It is to be noted that with existing designs a plate IHX cannot be fully inspected and repaired. Hence, the concept is in essence modular, which allows easy replacement of a faulty module.

The temperature and environment of the IHX depend on the final application of the HTR, i.e. for process heat generation, and/or for electricity production. For the generation of electricity with an indirect combined cycle (ANTARES design), the turbine is directly coupled to the secondary loop (see Figure 28). As a consequence, turbine transients and incidents affect the IHX (through fluctuations in temperatures and pressures). For process heat applications, transients are expected to be smoothed in time and amplitude.

For all types of applications, helium is the primary coolant. For the generation of electricity, it is interesting to consider a mixture of helium and an inert gas (e.g. argon) as secondary coolant to allow using a conventional turbine technology (adapted to air), at the expense of lower thermal efficiency. For the generation of heat, He/He intermediate heat exchanger is suitable. IHX limits for both applications are presented hereafter.

- Electricity production plant

Typical nominal operating conditions for an electricity production plant are given in

Table 8.

	Primary (Hot Side)	Secondary (Cold Side)
IHX Inlet temperatures	850°C	350°C
IHX Outlet temperatures	392°C	800°C
IHX Inlet Pressure	60 bars	55 bars
Flow rate	256 kg/s	696 kg/s
Maximal pressure loss (75% for IHX – 25% for ducting)	2% (75% for IHX – 25% for ducting)	4% (75% for IHX – 25% for ducting)

Table 8: Nominal operating conditions (600 MWth)

As regards the IHX, these operating conditions lead to a pressure difference between primary and secondary sides of 5 bars and to high temperature gradient between hot and cold ends of each circuit (450°C). This generates some significant deformations in the IHX core and at IHX junctions. These thermal loads are aggravated during thermal hydraulic transients.

Typical reference transients are presented in chapter **5** of this report. Main characteristics are summarized in Table 9 below.

These transients are related to incidents occurring on the gas turbine, and are specific to the production of electricity. Numbers of occurrences are envelope estimations for design.

	Temperature	ΔT/t	Duration	Mass Flows (For information)	Max ΔP (P1 – P2)	Max occurrences /20 years
Reactor Trip	T1: 850°C → 450°C T2: ~ 350°C	- 1°C/s	5 minutes	~ Nominal	40 bars	120
Primary Blower Trip	T1: ~ 850°C T2: 800°C → 400°C	- 10°C/s	40 seconds	F1: Nominal to 0 F2: Nominal	30 bars	20
Loss of Load with turbo machine Trip	T1: 850°C T2: 390°C → 500°C	+ 1.5°C/s	60 seconds	Nominal to 0	40 bars	100

P1/P2: Primary/Secondary Pressures
T1/T2: Primary/Secondary Temperatures
F1/F2: Primary/Secondary Mass flows

Table 9: IHX bounding transients for the generation of electricity

So far there is no technology of large plate IHX for high temperatures (approx. > 600°C) able to sustain variations in temperatures of 10°C/s and differences in pressure of 40 bars. Flexible designs are recommended for large variations in temperature, whereas rigid concepts for large variations in pressure. These requirements are antithetic.

- Process heat applications

Bounding steady state conditions that can be technically sustained by the IHX are similar to

Table 8. The IHX is a He/He heat exchanger to achieve better thermal exchange with reduced flow rate on the secondary side.

Applications for heat generation are characterized by 'smooth' transients (no turbine). Typical bounding transients for heat generating applications are given by Table 10**.**

During transients, it is assumed that no change in primary and secondary pressures (respectively 55 and 60 bars) and flow rates occurs. Secondary temperature is unchanged.

	Initial Temperature Primary Inlet	DT/t	Duration	Occurrences/20 years
Hot Shock	650°C	+ 40°C/min	5 minutes	1000 to 1500
Cold Shock	850°C	- 40°C/min	5 minutes	1000 to 1500

Table 10: IHX bounding transients for the generation of heat

The Figure 25 below shows the typical IHX transient preliminary considered for heat applications.

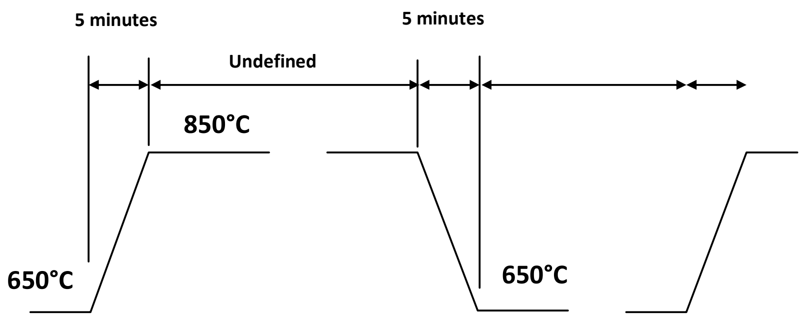

Figure 25 : Typical IHX transient cycles for heat applications

Thus, transients caused by the coupling with industrial processes are expected to be less stringent than those caused by turbine incident in case of electricity production in the secondary loop.

4.3.5 Safety considerations

Compared to steam generator, no water ingress in primary loop is expected with an IHX concept as secondary coolant is either pure helium (for coupling with process heat applications) or preferably a mixture of helium with an inert gas like Argon for electricity production.

The tritium permeation through the walls of the heat exchanger remains a point to consider.

4.4 IHX for molten salts

4.4.1 Principle

As a 'standard' IHX, an IHX with molten salt would deliver the heat to an intermediate loop between the reactor and the process plant. The primary coolant remains helium and the secondary one becomes a molten salt.

4.4.2 Technological maturity

The use of liquid salts provides the potential for improved heat transfer and reduced pumping powers, but also introduces materials compatibility issues [17].

It is very difficult to anticipate life expectancy of a tubular design as effects like corrosion and pressure differences would play an important role. It can reasonably not exceed the life expectancy of similar Helium-Helium IHX. He/molten salt heat exchangers need substantial R&D.

4.4.3 Performance and operational limits [20]

The main characteristics of the heat delivered by such IHX are:

- Primary loop: 50 bars, 850°C
- Secondary loop: a few bars, max delivered temperature of 800°C

The main technological limit is the mechanical resistance of the IHX to the difference in pressure between the primary and secondary loops at high temperatures (more than 750°C). Current plate IHX designs can not be stiff enough to sustain pressure differences > 40 bars, and flexible enough to accommodate temperature gradients at the same time. In reference [17], it is recommended to place the plate IHX in a low pressure loop (20 bars), despite increased complexity (primary loop in Helium can not be low pressure). If the design must be kept simple, it is recommended to use a tubular concept that will sustain larger pressure differences than plate IHX. The pressure losses may be another reason for that choice.

Despite engineering advantages like low pressure of the secondary loop and enhanced heat transfer, the use of a liquid salt also results in several challenges compared to helium or water [17]. Some of them are:

- Material compatibility (corrosion): two mechanisms need to be addressed, metal dissolution and oxidation of metal to ions. Molten salts are both electronic and ionic conductors, and as such are potentially very harmful for structures. Hastelloy or other nickel base alloys seem to be the best candidates.
- Freezing issues: molten salts have a relatively high freezing temperature (above 400°C). A liquid salt has the potential to freeze during transients (time to freeze lies between 5 and 65 minutes), especially after reactor trip. This places constraints on the IHX design (flow must be as uniform in temperature as possible), and on process side.

4.5 Primary Blower [20]

4.5.1 Principle

The primary blower ensures the flow of helium in the primary loop. It is made of:

- One shaft supported by bearings,
- One electrical motor (with appropriate electronics to change speed),
- A fan driven by the motor.

4.5.2 Technological maturity

The motor is always overhung to benefit from the blower thrust (decrease of load on bearings). The rotational frequency of the fan ranges from 3000 to 6000 tr/min when the shaft is driven by an electric motor. Smaller blowers have higher rotational frequency. The fan is immersed in helium at 390°C.
Two solutions have been considered for the motor of the gas circulators:

- A design with the motor outside the primary / secondary containment, in air, connected with the impeller through a leak tight rotating seal, the support of the circulator being obtained at least partially by conventional bearings,
- A fully immersed design with magnetic bearings.

The first solution has been used in the past and its feasibility is demonstrated. Despite a simpler design of the motor cavity, this solution has several drawbacks, like risks of air ingress in the circuit and limited efficiency of conventional bearings.

The second solution has been investigated by AREVA for ANTARES. In spite of a limited access to the motor is more limited this solution should be technically feasible, though requiring a substantial R&D effort. The motor is laid under helium. Since the motor can not sustain the temperature of the fan, it is enclosed in a pressurized cavity cooled below 100°C by a forced convection heat exchanger. The motor casing and vessel are separated by an insulated wall. The shaft crosses the vessel wall, and a special rotating seal limits Helium leaks outside or inside the vessel. The shaft is supported by magnetic bearings (axial and radial) that provide very high efficiency (first attempt for the MHTGR Project, USA).

Minimum lifetime of 10 years with maintenance (blower, bearings, motor) is achievable. Target of 40 years is not impossible.

4.5.3 Performances and operational limits

Based on past HTR experience (in helium) the feasibility of a circulator up to 4 MW is demonstrated with the flow rate of 250 kg/s.Increasing the power should be possible before meeting a technological limit, but there is no industrial experience beyond 4 MW. The main technological limit should be essentially related to the high conductibility of helium that may lead to arc discharge (Corona effect) if too many cables with high current density are packed together, especially at penetrations.

Another challenge is the development of tougher catcher bearings, but new hybrid systems (passive and active magnetic bearings) can relieve this constraint a bit. Technological feasibility of magnetic bearings is admitted, even for heavier motors (16 MW).

Integral testing of this component is preferably done on nuclear site during the commissioning.

4.5.4 Safety considerations

The blower ensures the flow of helium in the primary loop and is a key component for reactor heat removal. Main safety considerations concern the loss of forced cooling by trip of one or more helium blower. ‘Blower trip’ accident is studied in the following chapter 5 ‘Transients management’.

4.6 Power Conversion System

The development of HTR has been first focused on electricity generation, with core outlet at elevated temperature (around 850°C – 900°C) using a direct Brayton cycle. But recent market studies have shown a huge interest in the high-temperature process heat or cogeneration applications (coal-to-liquid, oil sands). Then current on-going works are oriented towards a versatile heat source design easily adaptable to cogeneration applications delivering heat at various temperature levels.

Studies have been conducted to assess the capability of the nuclear heat supply system (NHSS) to serve various process heat applications. They consist in defining plant architectures meeting the objective of being adapted to specific process heat delivery, selecting an adapted secondary fluid, selecting operating parameters of the primary and secondary heat transport systems depending on the core outlet temperature, investigating the ways of transporting heat at high temperature over long distances and also analyzing specific plant operating conditions, especially those involving strong interaction between the NHSS and process heat application.

4.6.1 Direct cycle

Initially companies focused on a direct Brayton cycle for plant electricity generation and low temperature cogeneration applications such as desalination.

In a direct cycle (see Figure 26), the driving force is generated by the expansion of hot helium heated in the reactor in a non-combustion gas turbine. The expanded gas is then cooled down, recompressed and returned as cold gas to the reactor. Because the hot gas expansion produces more power than the cold gas compression absorbs, the balance is positive and can be converted by a turbogenerator into electricity. The efficiency of the conversion is driven by the temperature differences. A recuperator is added to the cycle to heat the compressed gas by the expanded gas, recycling the corresponding energy. An intermediate cooler added between the LP compressor and the HP compressor significantly improves the recovered power and the cycle efficiency. The turbine exhaust pressure has an impact on the exhaust gas temperature, but is largely driven by the need to keep reasonable sizing of the equipment.

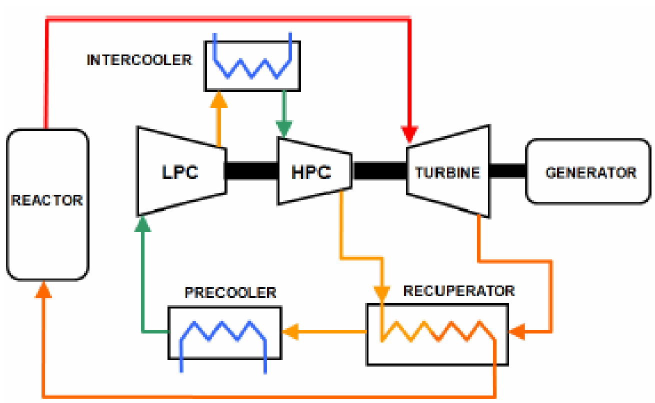

Figure 26: Direct Brayton cycle

With the high gas outlet temperature achieved in HTR reactors, a direct cycle using such a Brayton cycle can reach a relatively high thermal efficiency, with figures as high as 45% being quoted with 750°C/ 850 °C hot gas temperature. Depending upon the compression ratio, exhaust gas temperature in the range of 500 °C can be expected. Process Heat applications at mild or low temperature using a secondary loop on the pre-cooler or intercooler for instance can be considered.

Here should be mentioned the German program to test the performance of high helium turbo-machinery which was significant to the development of the gas turbine. This program involved two experimental facilities. The first was an experimental closed-cycle helium turbine cogeneration power plant (OBERHAUSEN II – 50 MWe) constructed and operated by the municipal utility, Energieversorgung Oberhausen (EVO) from end 1975 to 1988. This facility consisted of a fossil fired heater, helium turbines, compressors and related equipments. The second facility was the High Temperature Helium Test plant (HHV) for developing turbomachinery and components at FZJ-Juelich. The heat source for this plant was derived from an electric motor driven helium compressor.

The experience gained from these facilities is both positive and negative. At initial commissioning operation difficulties were encountered with the EVO facility including failure to meet fully the design power output of 50 MWe. The reasons for these difficulties were identified and as far as economically feasible the problems were corrected. Other difficulties dealt with vibration problems of the rotor of High-Pressure set, sealing and cooling gas losses and noise levels. At the HHV some start-up problems also occurred, but were soon corrected. The research and development programs at both facilities can be judged successful and fully supportive of the feasibility of the use of high temperature helium as a Brayton cycle working fluid for direct power conversion from a helium cooled nuclear reactor [25]. Nevertheless for safety requirements this solution is dismissed today as a design with a turbine directly in the primary helium circuit is considered to be a high risk solution.

4.6.2 Indirect cycle

For heat applications, heat cannot be extracted directly from primary active cycle: an additional level of separation is needed to considerably diminish radiological problems for the heat application plant. As this would also be beneficial for the design feasibility of the turbine and for its maintenance, indirect cycle is also considered now.

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Then indirect cycle is a flexible design adapted to versatile cogeneration needs. For process heat application (T<750°C) indirect steam cycle is preferred. Steam cycle HTR concepts couple the reactor to a Rankine power generating system through a steam generator. Theses concepts have the fewest feasibility issues, because their operating conditions tend to be less demanding. More importantly, several such systems have been built and operated around the world, and the basic technology is established. As a general concept, HTR steam cycle has no feasibility issues. There are only developmental details unique to each specific application.

With standard Rankine cycle, efficiency around 38 - 40% can be reached and up to 45% with supercritical water cycle. Nevertheless there are two real issues regarding the power cycle when looking at a cogeneration plant employing a supercritical Rankine cycle: firstly, the need for a reheat cycle and secondly the power / efficiency relationship. These considerations will carry differing weight when one considers the heat supply – process plant coupling options, such as a topping Rankine, extraction- or back-pressure turbine, but that will not be debated here. The use of supercritical steam in a Rankine cycle necessitates the use of a reheat cycle to ensure acceptable wetness at the outlet of the low-pressure turbine (see 4.2.3). Including a reheat segment in a HTR SG adds significant complexity to the SG through additional penetrations, tube-sheets, flow guides, etc. In particular, the size of the SG is increased significantly, if not disproportionately given that the HTR SG is not a radiant boiler but a convection boiler with a constrained primary side temperature. Through the addition of the re-heater, the available temperature differential for heat transfer is decreased not only for the re-heater but also the initial superheater portion.

For temperature above 750°C, gas cycle may become advantageous with efficiency around 45%, but today the feasibility of designs for such temperatures is highly uncertain due in particular to material strength limits.

In an indirect cycle, the heat from the reactor (primary circuit) is transferred to a secondary system via a SG (see Figure 27 and Figure 29) or an Intermediate Heat Exchanger (IHX - Figure 28) or both (Figure 30). A circulator (or blower) is used to circulate the primary coolant (helium) through the reactor and heat exchanger overcoming the circuit pressure losses.

Process heat can be extracted relatively easily at several temperature levels. The choice of the primary working fluid is limited to helium; however the choice of working fluid on the secondary side depends on the choice of the power conversion system and on the process heat requirements. The heat transferred to the secondary system can be used to generate electricity either in a combined cycle as represented in Figure 28 or in a conventional Rankine steam cycle in subcritical or supercritical conditions.

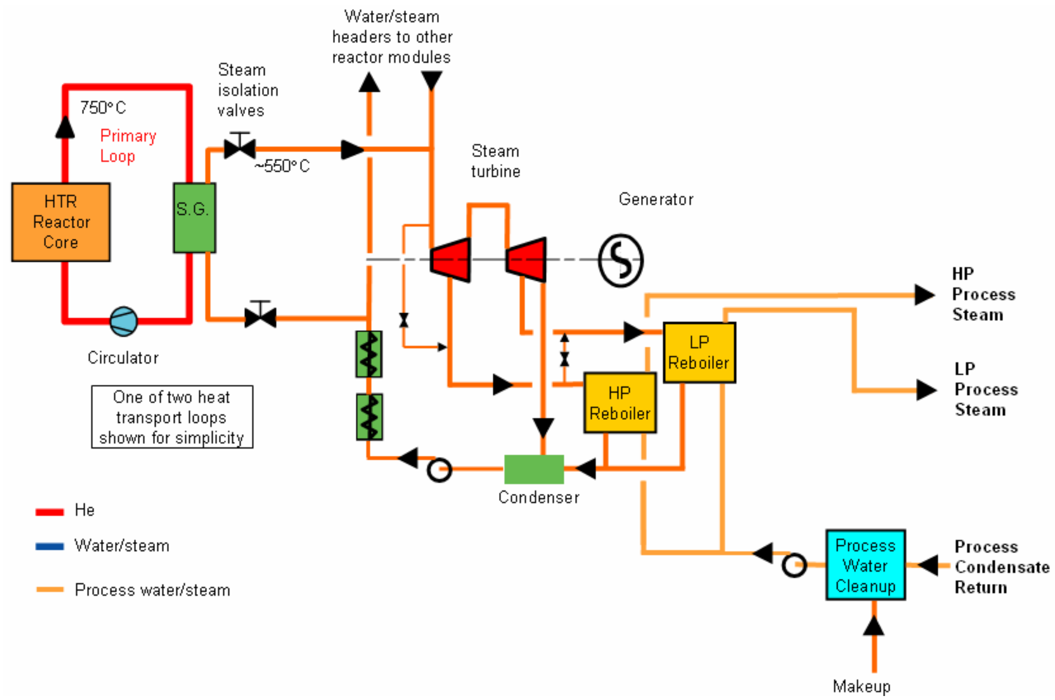


Figure 27: Commercial process heat cogeneration facility – Basic configuration (AREVA proposal for NGNP)

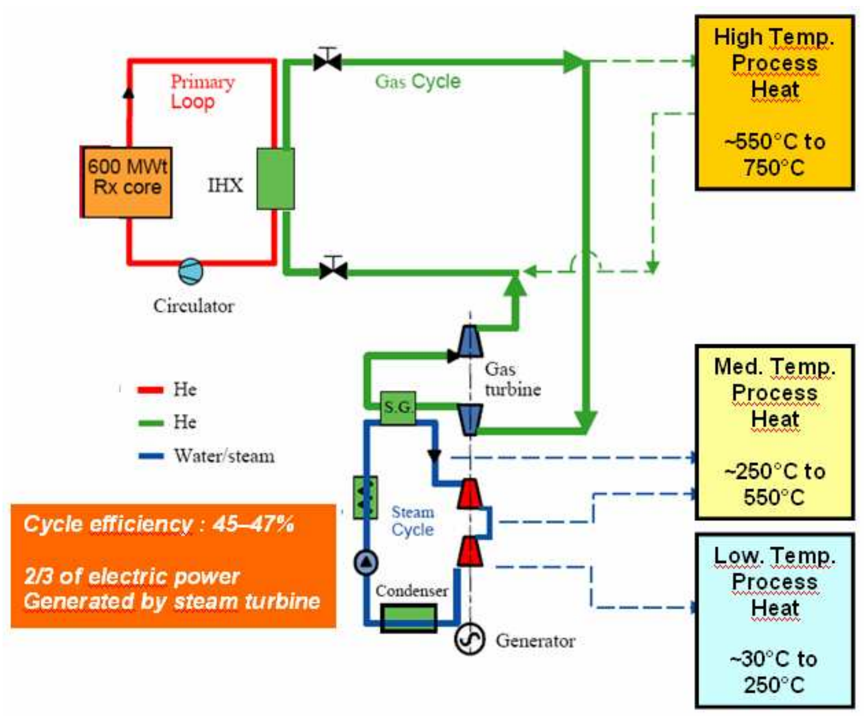


Figure 28: Indirect combined steam cycle[11]

[11] ANTARES data (2006)

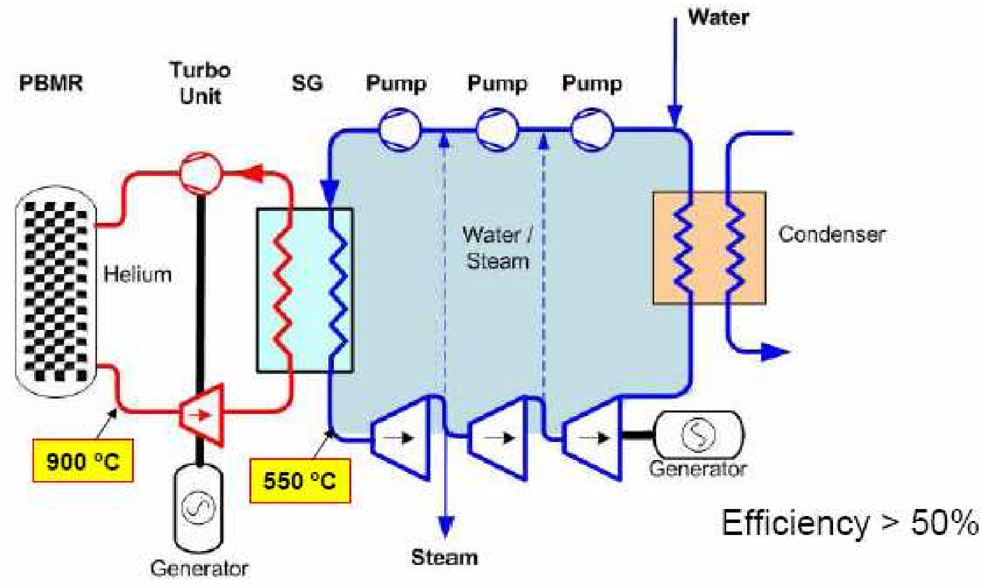


Figure 29: High efficiency combined cycle cogeneration plant[12]

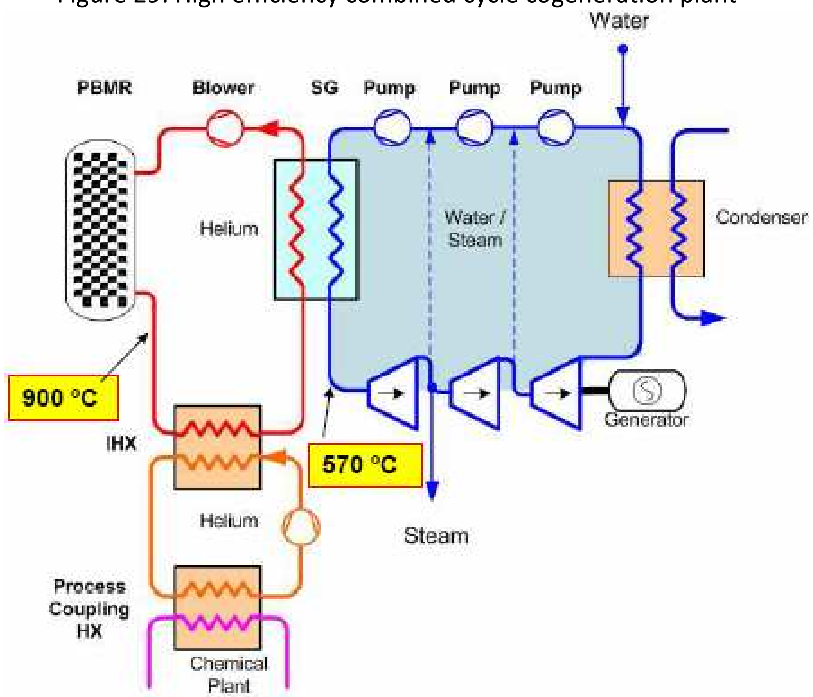


Figure 30: High-temperature process heat cogeneration plant[13]

[12] PBMR data (2009)

[13] PBMR data (2009)

4.6.3 Direct cycle versus indirect cycle

When comparing, the direct cycle has the advantage of the conceptual simplicity for electricity generation: a heat source directly coupled to the power conversion system. This will result in less equipment; however for HTR it means that helium must be used as working fluid. This is a disadvantage in that requires the development of specific helium turbo-machinery and the development of an efficient He/He recuperator. Brayton cycle efficiency is dropping rapidly as the inlet temperature is decreasing. Additionally, in the direct cycle, it becomes more difficult to extract process heat from the system without affecting system performance.

From a safety point of view, the direct cycle has disadvantage for managing accident due to mechanical failure of the turbo-machinery. Additionally for high temperature process heat or cogeneration applications there is no advantage to have a direct cycle configuration (safety requirements, technological developments…).

On the other hand the indirect cycle presents a flexible design adapted to steam generation plants and versatile cogeneration needs. It allows physically decoupling the reactor or primary system from the PCS and requires the addition of two major elements: a helium circulator and a heat exchanger (IHX or SG). It also permits the use of conventional power conversion or heat removal technologies because the secondary working fluid is not limited to helium. Nevertheless regarding safety criteria an additional 'separating circuit' is required to limit at the most tritium on end-users side as stated above in 3.5.1.2.

Regarding thermal efficiency, the combined cycle offers the most efficient one at high temperatures.

4.7 Steam and heat transport

It should also be noted that the feasibility of a modular HTR concept as process heat producer could also depend on the feasibility of transporting heat over long distances at high temperature and pressure.

From a pure technical point of view, very high pressures and temperatures are possible; the available materials for the line itself dictate the limits. It is however economics that determine the optimum transport conditions. Much depends on the pressure and temperature (saturated or superheated) needed at the end of the pipe. Condensation during transport should be avoided. The higher the steam pressure is the smaller the pipe diameter needs to be and the lower the heat losses are. However the pressure drop will be higher by the increased transport velocity. Each application/context requires a specific economical consideration.

For example at Chemelot site the maximum pressure for steam transport was decided on the basis of the optimum pressure for the NAFTA crackers (the steam turbines). High-pressure steam (end of pipe temperature between 490°C and 525°C and pressure between 110 bar and 140 bar) is transported over a distance of 4 km. This results in a pressure drop of about 15 bar and temperature drop of 30°C. Today the temperature mentioned as a maximum is around 570°C at reasonable cost.

On the other hand first investigations show that transport by gas like helium would result in large circulating flows and then require large insulated pipes and a high pumping power. A liquid transport medium like molten salt has the potential to reduce these constraints, but will probably require significant technological developments.

4.8 Plant lifetime

Expected and targeted plant lifetime of modular HTR is in range 40 – 60 years as already designed recent LWRs and other new advanced reactors. Nevertheless main components such as blower, SG(s) or IHX(s) will be replaced (maybe several times) during this period (see 4.1 and 4.3).

4.9 Plant configuration – Reactor building

The plant layouts are developed with consideration to many requirements including plant safety, plant operations and maintenance, plant construction schedule and costs and investment protection. The systems, structures and components are designed so that facilities are easily and economically maintained:

- incorporation of easily maintained features and durable materials into the facility design
- ease of replacement of installed equipment
- accessibility of installed equipment and building systems for performance of maintenance and testing
- personnel safety

The buildings are designed with an objective of a 60 years operating life. Any component parts that may require replacement to achieve this design objective will be readily accessible for replacement. Potentially hazardous materials will be located with care to minimize the potential for contamination or interaction with other hazards.

The following Figure 31 and Figure 32 show two configurations for reactor buildings as examples.

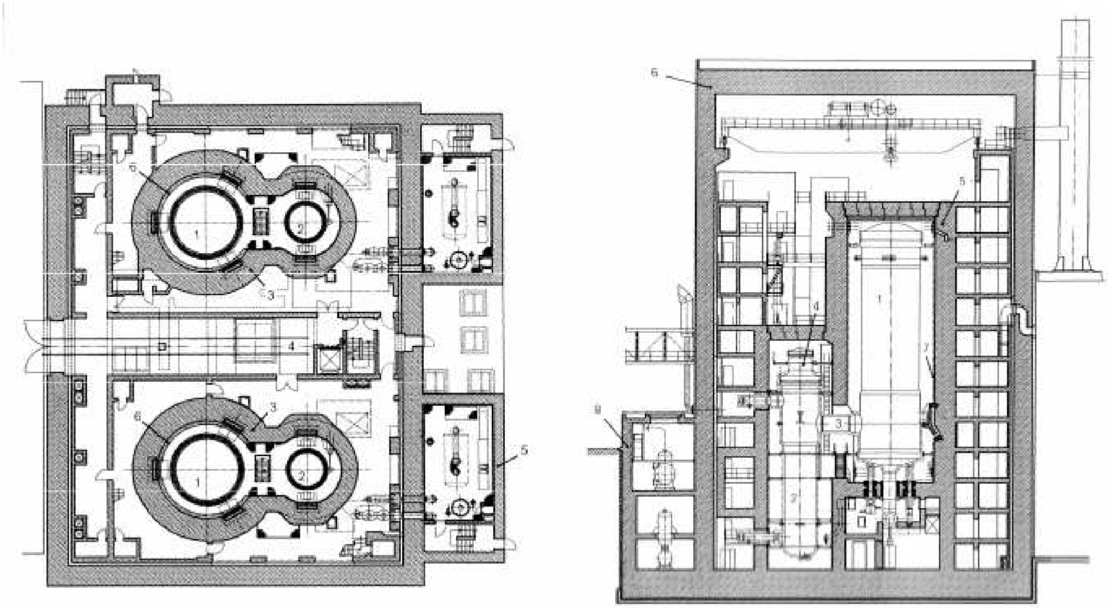


Figure 31: PBMR – Two modules reactor building

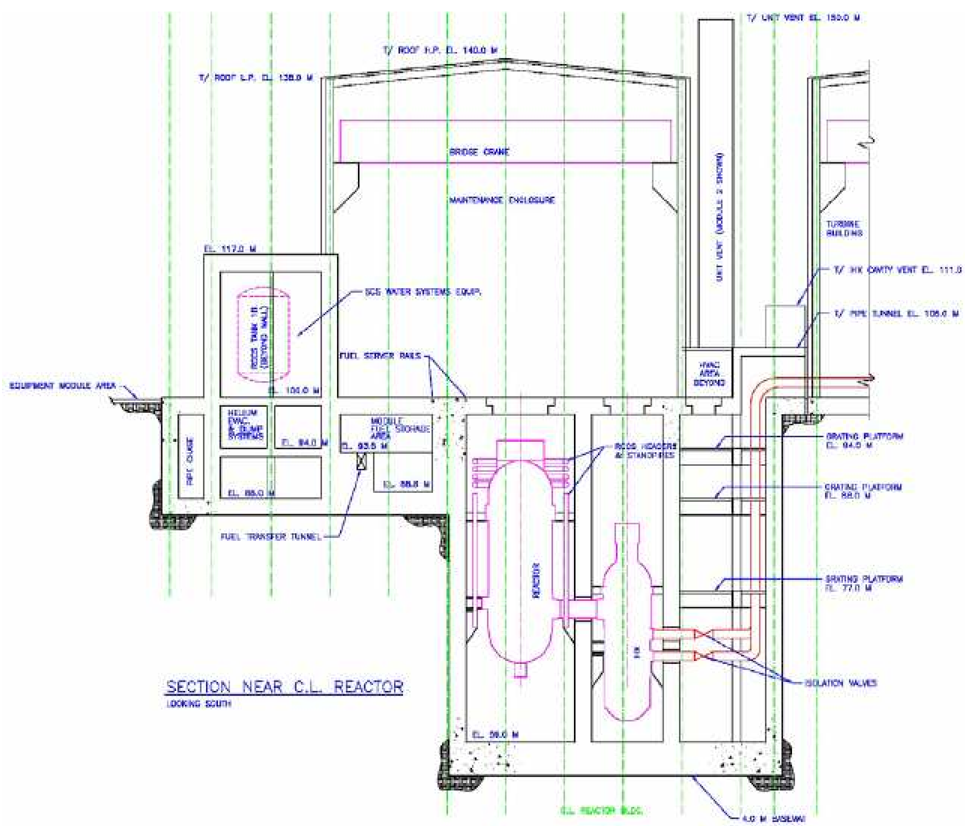


Figure 32: ANTARES design - Reactor building section view

4.10 Summary of precedent statements

The following Table 11 proposes a summary of HTR characteristics presented in the precedent sections:

Rev.1

Main characteristics		
Core technology	**Prismatic**	**Pebble-bed**
Power (maximal)	600 MWth	250 MWth
Refueling frequency	18 – 24 months	5 – 6 years
Reactor Outlet Temperature (ROT)	Max 950°C (safety considerations) Near term application: 750°C	
Heat transport system		
Component	**SG**	**IHX**
Characteristics	Superheated steam outlet temperature: max 570°C Superheated steam outlet pressure: max 17 MPa	Max inlet T°C (primary side): 850°C Outlet T°C (secondary side): ~800°C Outlet pressure (secondary side): ~5 MPa
Thermal rating	~ 250 MWth for one conventional SG	150 to 300 MWth for tubular design 300 to 600 MWth for compact design

Table 11: Summary of some HTR characteristics

Based on precedent statements the most appropriate process heat plant design should be an indirect cycle (steam cycle under 750°C and gas cycle for temperature above) with 3 separated circuits (in series). The advantage of the indirectly coupled heat supply system is that it provides the ability to keep the reactor module running at optimal steady state conditions, isolated from the process steam demand fluctuations. Steam cycle HTR concepts couple the reactor to a Rankine power generating system through a steam generator.

Rev.1 It is necessary for most of the applications to have an intermediate circuit (two heat exchangers) separating the HTR plant and the process heat plant, because the heat exchanger at the limit of the primary circuit should not be in contact with corrosive species, dust, etc… coming from the application plant on the one hand, because the process heat end users will not accept the risk of having his facility and his products contaminated with radioactive species.

The baseline option for short term applications in terms of working fluid to transport heat over significant distances is steam, currently used in heat networks existing on many industrial sites. With usual materials used for steam pipes, the temperature of steam is limited to 550°C. For higher temperatures, the main option is helium operated at relatively high pressure to minimize pumping losses. Other inert gases or liquid salts may be considered. A molten salt loop can operate at any desired pressure by installation of a pressurizer with an inert gas. The salts are essentially incompressible; thus the gas pressure in the pressurizer determines the system‘s pressure.

The needs of the process heat plant will in many cases determine the suitable pressures. In some cases the strength of corrosion resistant materials will require lower pressures in intermediate heat transport loops. In other cases higher pressures will be desired to maximize heat transfer. In some chemical reactors the rate of chemical reactions and the size of the chemical reactor are determined by the rate of heat transfer through the process heat exchanger. High-pressure helium is less efficient in heat transfer than low-pressure liquid salts; consequently, in some chemical systems the use of helium heat-transfer systems increases the size of process equipment in the process heat plant. Consequently the choice of intermediate loop fluid may be determined partly by process heat plant requirements [15].

Two top-level temperature requirements emerge for the interface between the nuclear and the process heat plants. The requirements are defined by the outlet temperature of the HTR core reactor and the maximum temperature delivered to process heat plant. At this stage of development these maximum temperatures are under investigations.

Regarding the maturity of the different heat exchangers technologies, three reference plants may be proposed for the near, medium and long term. For near term applications, the outlet temperature of the HTR reactor core is reasonably limited to 750°C, consistent with current capabilities. For mid-term and long-term applications higher outlet temperatures up to 950°C (maximal ROT regarding safety considerations) may be considered but their effects on component performance should be assessed more in depths. This leads to the three following proposals for near, medium and long term coupling applications. For near term application it is meant 0-5 years, for medium term 5-15 years, and for long term applications after 15 years. It should be noticed that from a nuclear supplier point of view this does not take into account plant commissioning but only the period for an offer starting on today.

- **<u>Near term proposal:</u>**

Rev.1 ROT = 700°C (HTR)

Steam production around <u>500°C – 520°C</u> with one conventional SG and additional reboiler or heat exchanger in series on tertiary circuit.

- **<u>Medium term proposed solution:</u>**

ROT = 750°C (HTR)

Steam production until <u>570°C</u> with IHX and conventional SG or reboiler in series on tertiary circuit.

Rev.1 <u>NB:</u> In a medium term, the ROT may reach 850°C for IHX demonstration at more elevated temperatures.

- **<u>Long term propects:</u>**

ROT = 950°C (VHTR)

Heat production until <u>850°C</u> with 2 consecutive IHX (secondary and tertiary circuits) and possible ***complementary steam production*** until <u>570°C</u> (similar to medium term solution).

Intermediate heat temperatures (between 570°C and 850°C) may also be provided.

Figure 33 below proposes a schematic plant configuration for an alternative solution between medium and long term solution coming from ANTARES design.

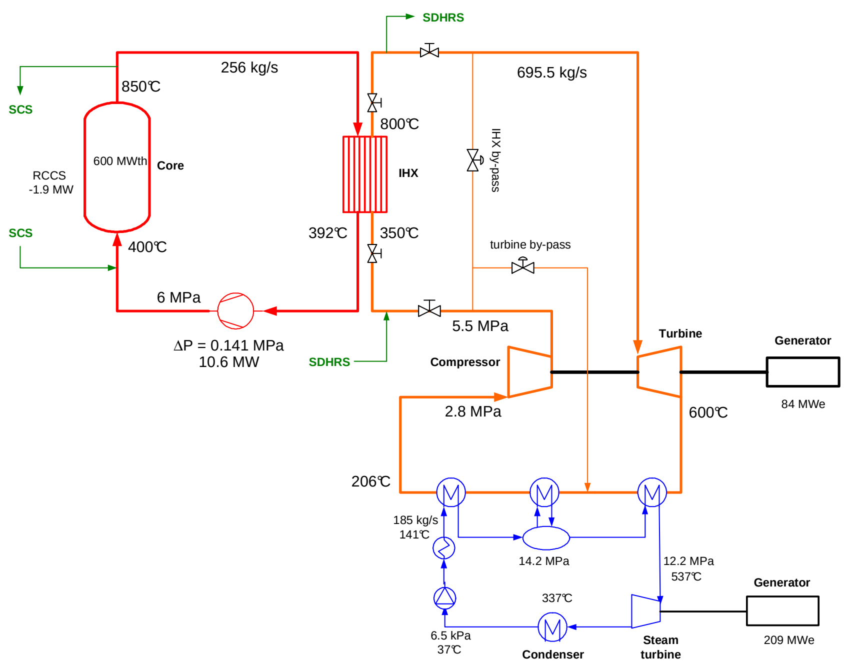


Figure 33: ANTARES nominal parameters

5 Transients management

5.1 Nuclear heat supply system transients management

In this part data from different HTR options are presented: modular pebble-bed core reactor with SG (PBMR-CG) and modular prismatic core reactor with IHX (ANTARES pre-conceptual design).

5.1.1 Pebble-bed core HTR with 'conventional' SG

5.1.1.1 Plant operational states and transitions

In order to describe the transient management capability of the Nuclear Heat Supply System (NHSS) it is necessary to describe the operational states and state transitions of the NHSS.

The diagram in ***Appendix 2*** shows the state transition diagram for the PBMR-CG NHSS. It shows the operational states as well as the allowed state transitions.

The high level characteristics for each of the operating states are listed in ***Appendix 3***, and for allowed state transitions in ***Appendix 4***.

5.1.2 Prismatic core HTR with IHX

5.1.2.1 Normal transients

5.1.2.1.1 Startup transient

The objective is to start the plant from a cold shutdown (<120°C) to power state (technical minimum, i.e. 15 to 30% of Nominal Power) within 24 hours. The sequencing consists of four stages (from A to D) from a departure state 1 to a final state 7.

Stage A: *Preheating of the primary circuit (including IHX) to obtain about 400°C at the core outlet (helium temperature)*:
Duration: about 13 hours
- **State 1**: Primary circulator start-up
- **State 2**: Core divergence. Nuclear power maintained to some percents of Nominal Power

Stage B: Coupling Nuclear Heat Source with Power Conversion System
Duration: about 4 hours
- **State 3**: Gas turbine start-up
- **State 4**: Secondary isolation valves gradual opening

Stage C: Rise in temperature to technical minimum
Duration: about 7 hours
- **State 5**: Nuclear power rise by core rods withdrawal.
- **State 6**: The technical minimum (to be specified, between 15 and 30% NP) is reached.

During all the start-up phase, helium makeup's are performed in the primary loop while the secondary loop mass is held constant. The helium makeup does enable to maintain the IHX pressure difference below 12 bars.

Stage D: Rise to Nominal Power (Power rate: about 5%/min)
From the State 6, the primary temperatures remain constant and the blower speed is controlled to transfer the required thermal power (from technical minimum to nominal power). At the secondary side the by-pass valve is progressively closed. At about 60% of nominal power, the by-pass is totally closed and the compressor inlet guide vanes (IGV) are closed. From 60 to 100% nominal power, the IGV is progressively opened.
Duration: less than 1 hour
- State 7: Nominal Power

For each state, corresponding pressures and temperatures are presented in Table 12 below.

State	Time (hour)	Primary pressure (MPa)	IHX secondary pressure (MPa)	IHX pressure difference (MPa)	Compressor outlet pressure (MPa)	Turbine outlet pressure (Mpa)	IHX temperatures (°C)			
							Primary inlet	Primary outlet	Secondary inlet	Secondary outlet
1	0	1.21	0.77	0.44	0.77	0.77	20	20	20	20
2	1	1.33	0.84	0.49	0.77	0.77	50	32	*	*
3	13	3.14	1.99	1.15	0.77	0.77	400	175	*	*
4		3.14	1.99	1.15	1.99	0.77	400	175	*	*
5	17	3.14	2.69	0.45	2.69	0.77	400	175	173	398
6	24	5.40	4.76	0.64	4.76	1.84	850	390	340	800
7	25	6.00	5.50	0.50	5.50	2.60	850	390	340	800

* IHX secondary side isolated by isolation valves

Table 12: Pressure and temperature variations (rough values)

5.1.2.1.2 Shutdown transient

Before the plant is shut down, it brought down to the technical minimum (i.e. 15 to 30% of Nominal Power). The objective is that the plant is ready for refueling within 24 hours. The shutdown procedure is roughly reversal to the startup one.

5.1.2.2 Abnormal transients

The following main bounding events are presented:

- Reactor trip
- Primary circulator trip (blower trip)
- Loss of offsite power

5.1.2.2.1 Reactor trip

5.1.2.2.1.1 Description of the transient

This transient can be split up into the main following stages:

- At t=0 s a reactor trip is set off.
- Grid disconnection and bypass opening are set off when electrical power is lower than 10% of nominal electrical power.
- The by-pass is kept fully open.
- When the pressure at the secondary side of the IHX reaches the lower pressure of the secondary circuit (2.8 MPa), isolation valves are closed and secondary IHX volume is blown up till the nominal value (5.5 MPa).
- The Decay Heat Removal System is started-up.

An envelope estimation of occurrence number is 3600 times over plant lifetime (60 years).

5.1.2.2.1.2 Parameters versus time

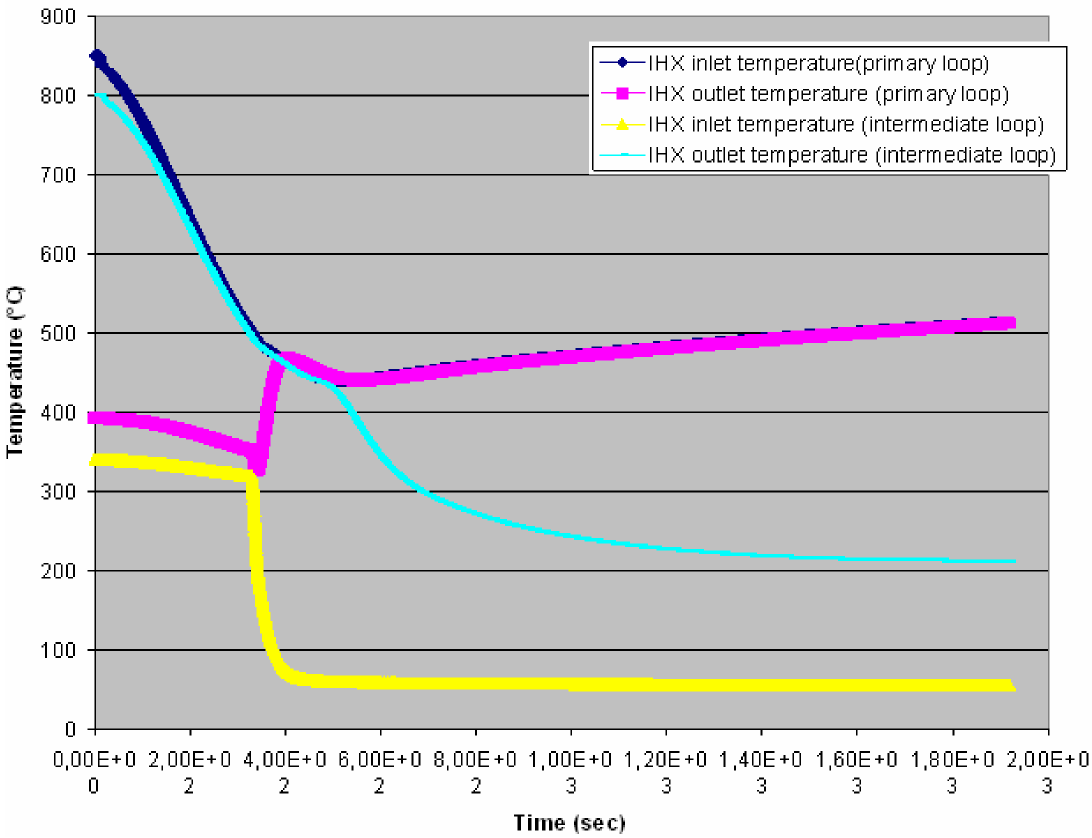
<u>Step 1:</u> Until by-pass opening (about t = 350 s),

- The core outlet (IHX primary inlet) temperature decreases at the rate of about 67°C/min during the first minutes.
- The primary pressure follows the primary mean temperature decrease (about 1.2 bar/min).
- The IHX secondary temperatures and pressures follow the primary ones decrease.

<u>Step 2:</u> From by-pass opening:

- The core inlet temperature rapidly increases and reaches the core outlet one because the core power is not removed by the IHX anymore. The primary pressure follows the temperature increase.
- The secondary flow rate is suddenly lost and the High Pressure (HP) decreases immediately to a pressure close to the Low Pressure (LP) because the LP volume is very big in comparison to the HP one. The pressure decrease is about 2.9 bar/s.
- At about t = 500 s, the secondary IHX pressure rises and reaches 5;5 MPa at t = 800 s.

Figure 34, Figure 35 and Figure 36 give the temperature, pressure and flow-rate evolutions at each side of the IHX.



<u>Figure 34: IHX temperature during reactor trip</u>

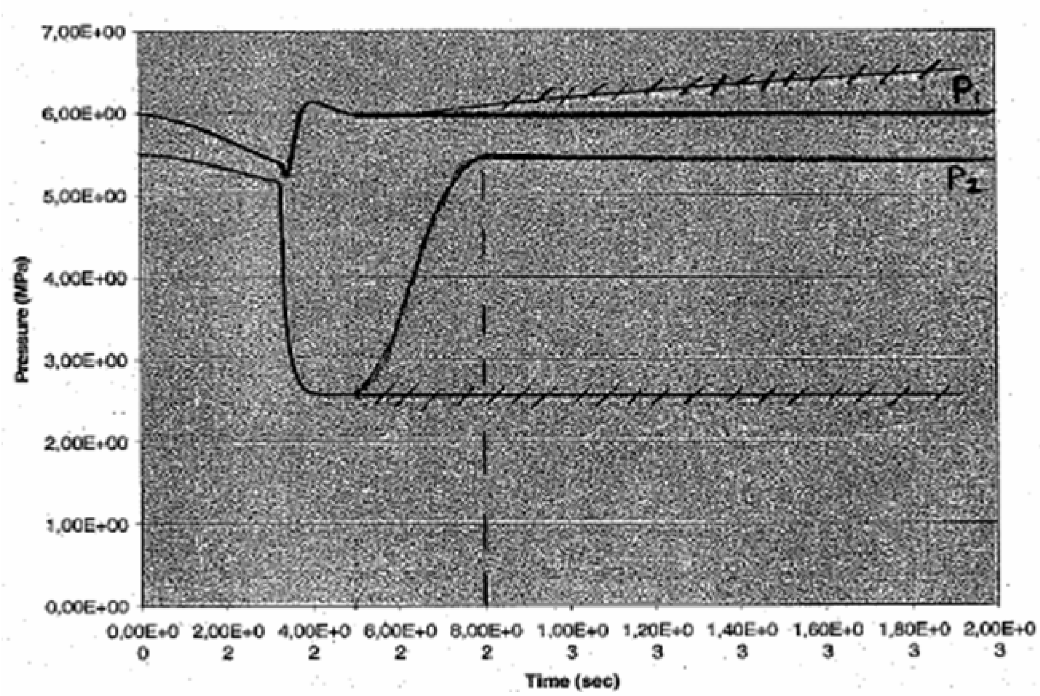


Figure 35: IHX Pressure during reactor trip

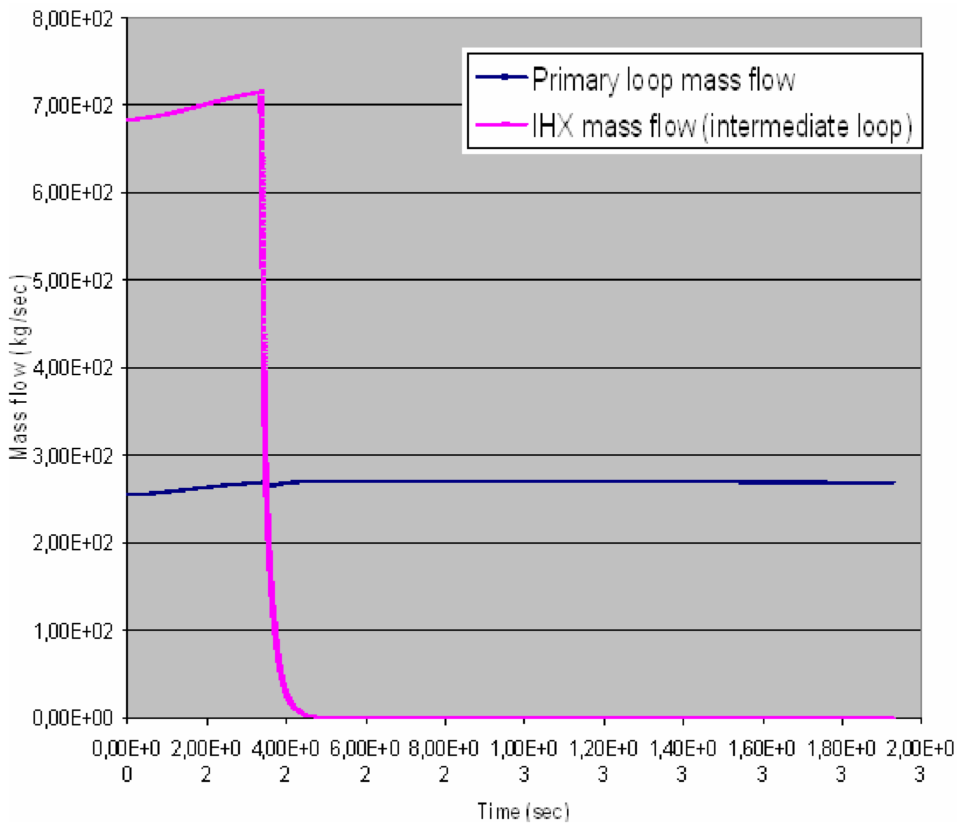


Figure 36: IHX flowrate during reactor trip

5.1.2.2.2 Primary circulator trip

5.1.2.2.2.1 Description of transient:

This transient can be split up into the main following stages:

- At t=0 s, the blower stops.
- Grid disconnection and bypass opening are set off when electrical power is < 10% of nominal electrical power.
- Reactor trips when P/Q is > 1.2, where P is the neutron power and Q the core mass flow rate.
- When the pressure at the secondary side of the IHX reaches the lower pressure of the secondary circuit (2.8 MPa), to protect the IHX isolation valves are closed and secondary IHX volume is blown up in 5 minutes till the primary pressure value minus 0.5 MPa.

An envelope estimation of occurrence number is 600 times over plant lifetime (60 years).

5.1.2.2.2.2 Parameters versus time

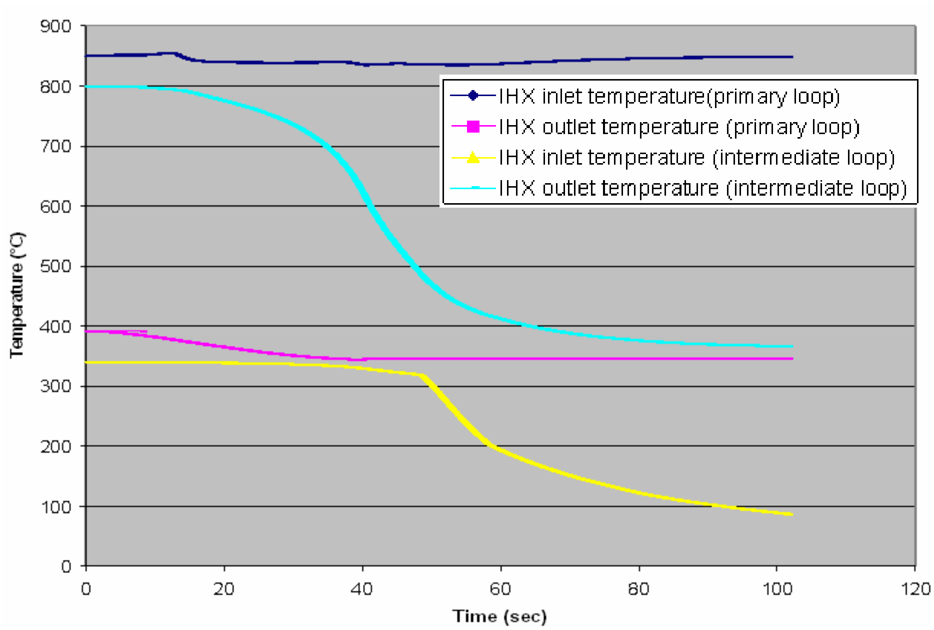
<u>At the primary side:</u>

Until the blower stops, the core outlet temperature is more or less constant and the inlet temperature decreases due to flow rate decrease. The pressure slightly decreases due to thermal effect.

<u>At the secondary side:</u>

The primary blower trip induces an IHX outlet temperature decrease (no heat transfer anymore) with a rate of about 16°C/s. Then the by-pass opening induces a fast pressure decrease of about 2 bar/s. The secondary IHX volume is blown-up till about 5.2 MPa (5.7-0.5 MPa).

Figure 37, Figure 38 and Figure 39 give the temperature, pressure and flow-rate evolutions at each side of the IHX.



<u>Figure 37: IHX temperature during blower trip</u>

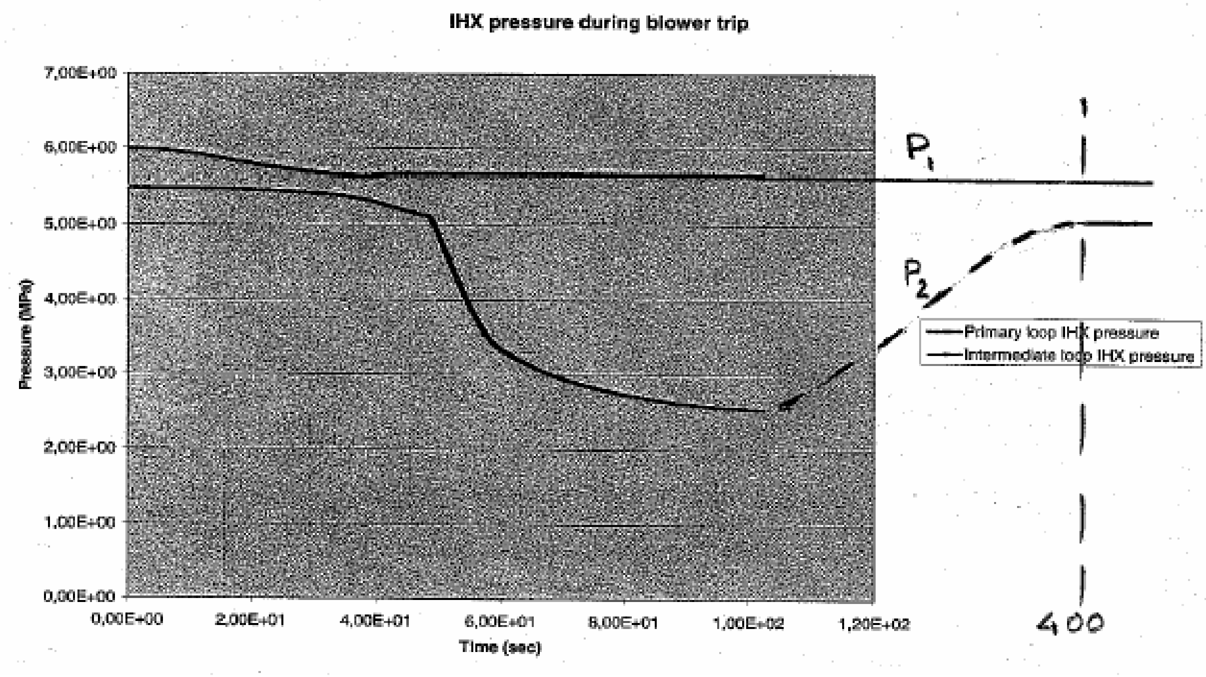


Figure 38: IHX pressure during blower trip

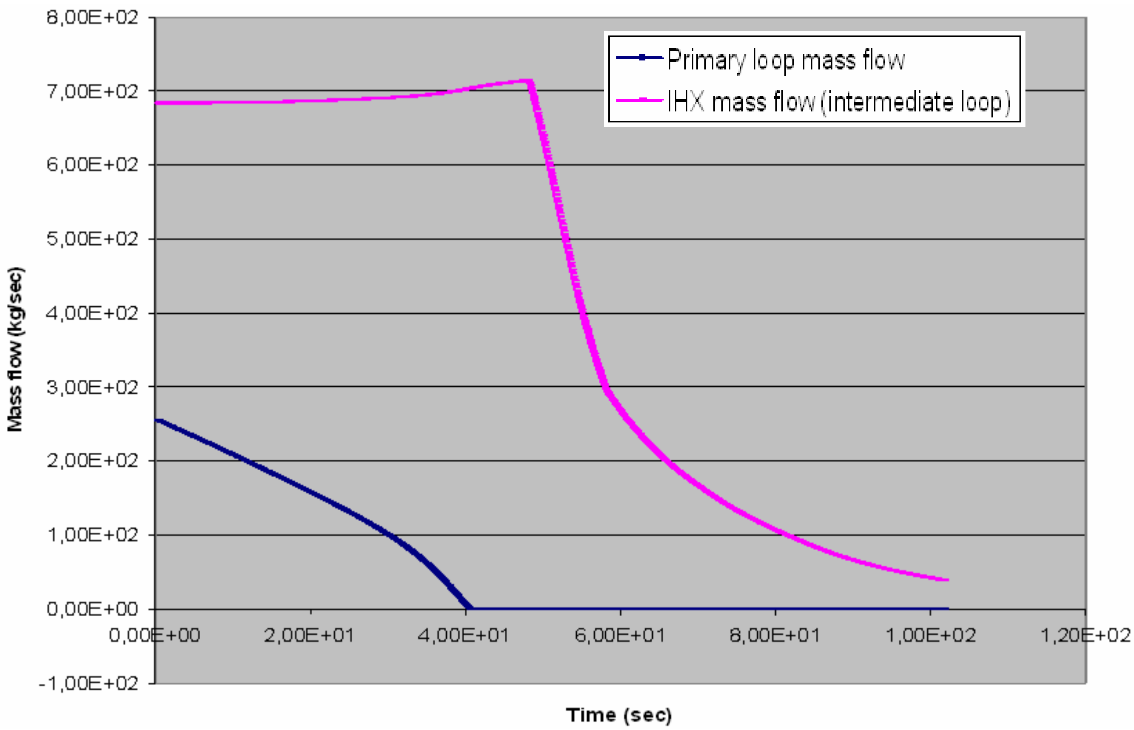


Figure 39: IHX flowrate during blower trip

5.1.2.2.3 Loss of offsite power

5.1.2.2.3.1 Description of transient

This transient can be described as follows:

- Loss of load associated to turbine by-pass opening. The by-pass is fully open after 10 seconds and remains open after.
- Reactor trip occurs with the loss of load at t =10 s.
- Blower shuts down when temperature at blower outlet is higher that a limit set to 50°C above the nominal value. This occurs latest after 60 s.
- When the pressure at the secondary side of the IHX reaches the lower pressure of the secondary circuit (2.8 MPa), isolation valves are closed and secondary IHX volume is blown up till the primary pressure value minus 0.5 MPa.

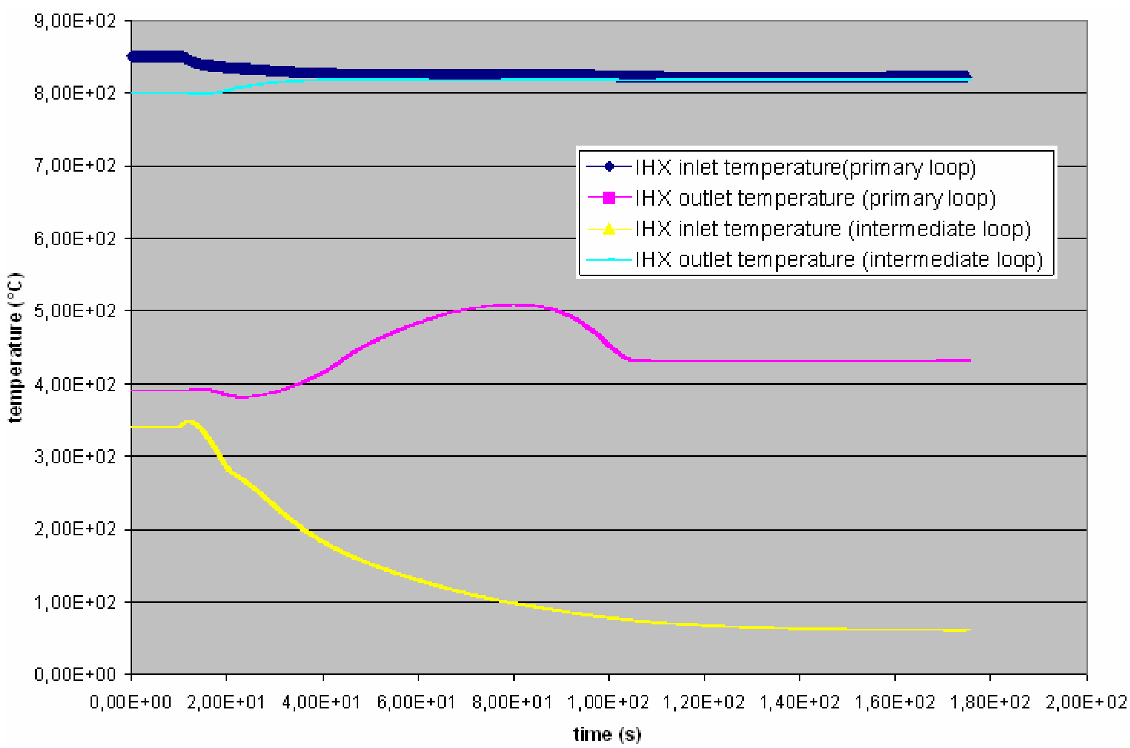
An envelope estimation of occurrence number is 300 times over plant lifetime (60 years).

5.1.2.2.3.2 Parameters versus time

The evolution of temperature, pressure and flow-rate is described hereafter:

- The primary IHX outlet temperature increases till about 500°C and decreases later due to blower shutdown.
- The primary pressure slightly increases due to thermal effect.
- The secondary IHX pressure decreases till 2.8 MPa at t = 100 s, then increases till about 6 MPa at t = 400 sec

Figure 40, Figure 41 and Figure 42 give the temperature, pressure and flow-rate evolutions at each side of the IHX.



<u>**Figure 40: IHX temperature during loss of load with turbo-machine trip**</u>

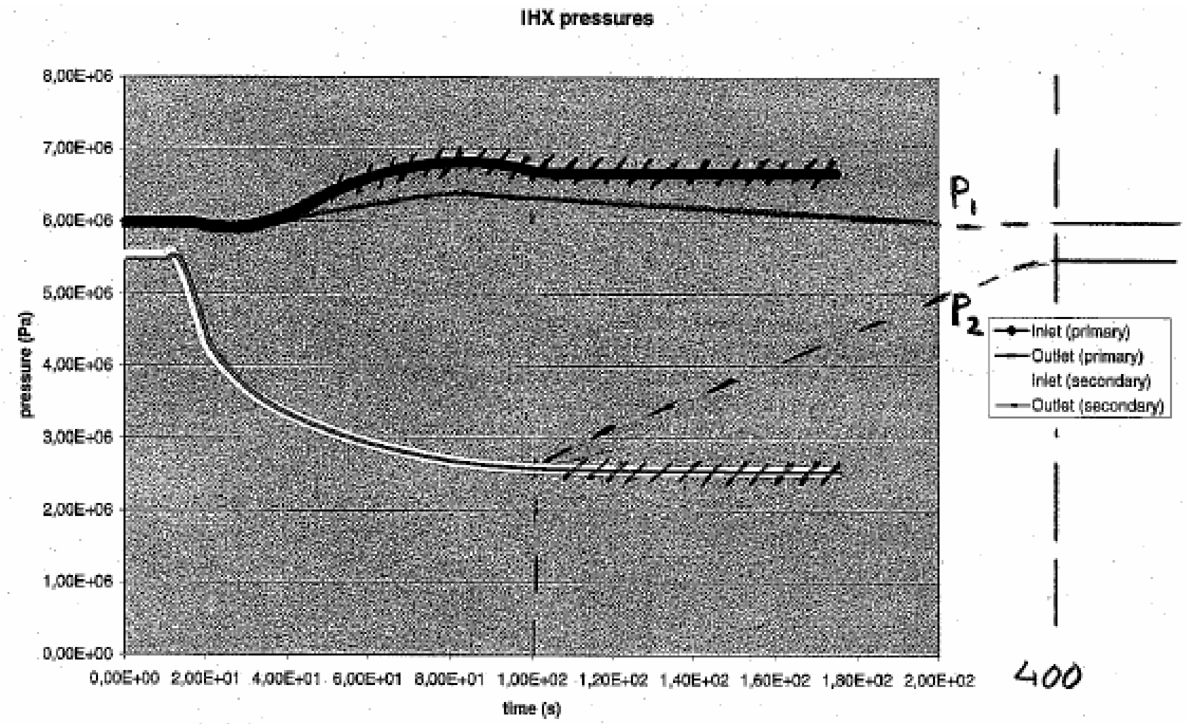


Figure 41: IHX pressure during loss of load with turbo-machine trip

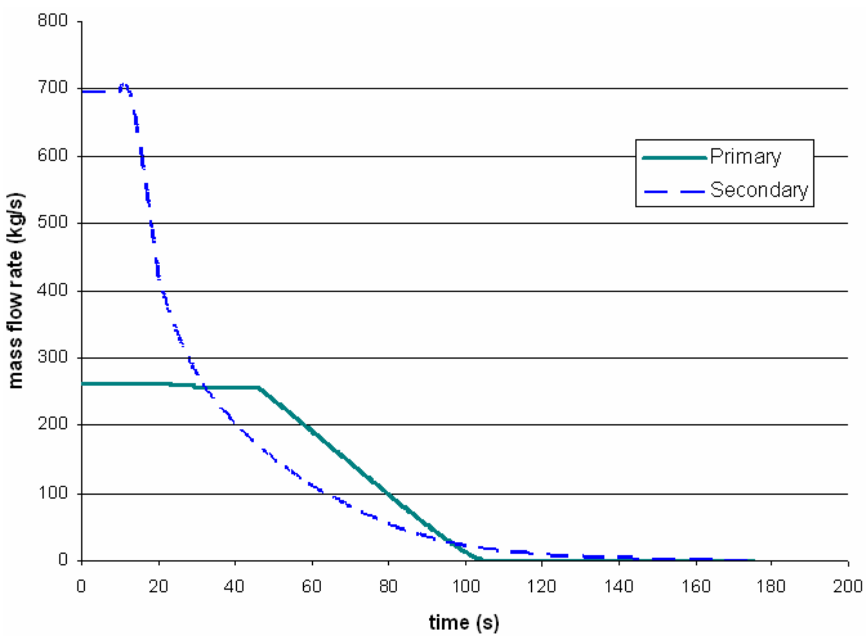


Figure 42: IHX flow rate during loss of load with turbo-machine trip

5.2 Process Heat Plant Transients Management

Regarding the process heat plant in comparison to the electricity production one, the NHSS arrangement is not modified. The power conversion system (PCS) of a process heat plant is adapted to heat transfer to process unit and to electricity production local needs (nuclear plant and process unit).

Nevertheless the event of sudden loss of process has to be examined. The unit should be capable of sustaining sudden loss of the process heat sink and continue to produce electricity. If the heat transfer to process is installed on the secondary circuit, the addition of a thermal load absorber on the secondary circuit could be necessary. On the other hand specific procedures could be necessary, according to heat application, especially startup and shutdown phases.

Preliminary analysis of the typical process industry steam consumer needs indicates that the typical installation will required a Nuclear Steam Supply System (Steam Utilities Block) consisting of multi-module Nuclear Heat Supply Systems (NHSS). This implies that the effect of the internally initiated transients will be damped (or limited) to a proportional percentage change in output of the total NSSS. Likewise, the externally initiated transients will require only a damped (or limited) proportional response from the individual modules.

The steam production of the NSSS can be direct from the NHSS, or indirect via a secondary heat exchanger, such as a steam reboiler (SR). This is illustrated in Figure 43 and Figure 44 below[14].

It should also be noticed that greatest economic benefit is expected when the process steam is extracted after partial expansion in the turbine, depending upon the end-user steam requirements.

5.2.1.1.1 Directly Coupled Steam Supply

The advantage of the directly coupled system is that higher steam temperatures can be supplied, the plant is smaller with lower capital costs and the control system is simpler.

This coupling is best suited to process steam consumers where all the steam is returned as condensate to the NSSS. This allows the use of a smaller water purification plant for the make-up feedwater supply where there is a need to ensure that there is a high quality water supply for the SG in order to ensure that the mean time between scheduled outages for SG cleaning is as long as possible.

There are however disadvantages, the primary one being that the NHSS is more directly affected by events within the process steam consumer's plant. The primary SG making superheated steam has a limited ability to operate at variable steam temperatures and fluctuations in demand will require a rapid response from the electrical power generation system for load balancing. The steam supply transients are thus constrained to a large degree by the NHSS transient response constraints as listed in ***Appendix 3*** and ***Appendix 4***, except for turn-downs. The NSSS can manage rapid turn-downs through either steam dumping or through routing the excess steam through the various dump condensers or shut-down coolers. Ramp-up can be rapid if there is adequate pre-notification of the intended increase in steam demand and the system can be pre-set using the dump condensers as an interim buffer prior to ramp-up, otherwise the constraints as per ***Appendix 4*** above apply.

The other disadvantage of this system is that there is only a single barrier to contamination both into- and out of- the primary NHSS coolant, both for accidents and normal operations. Accident conditions should not result in contamination of the off-taker's process or product due to the pressure differentials and the ability to isolate the steam / feedwater supply. The constraint may be introduced by the radionuclides, such as Tritium, which can migrate through metallic barriers under normal operation. When the steam is used within a product and there are stringent limitations on the contamination levels of the product. When the steam is used in such a way that there can be a build-up of contamination over time, this configuration may place sever requirements on the purification system.

[14] Note that the figures are simplified illustrations and do not include the detail of secondary shut-down system heat exchangers, process steam recovery condensers, bypass flows, feedwater pre-heating, make-up water supplies and treatment, Steam supply atemperation, etc.

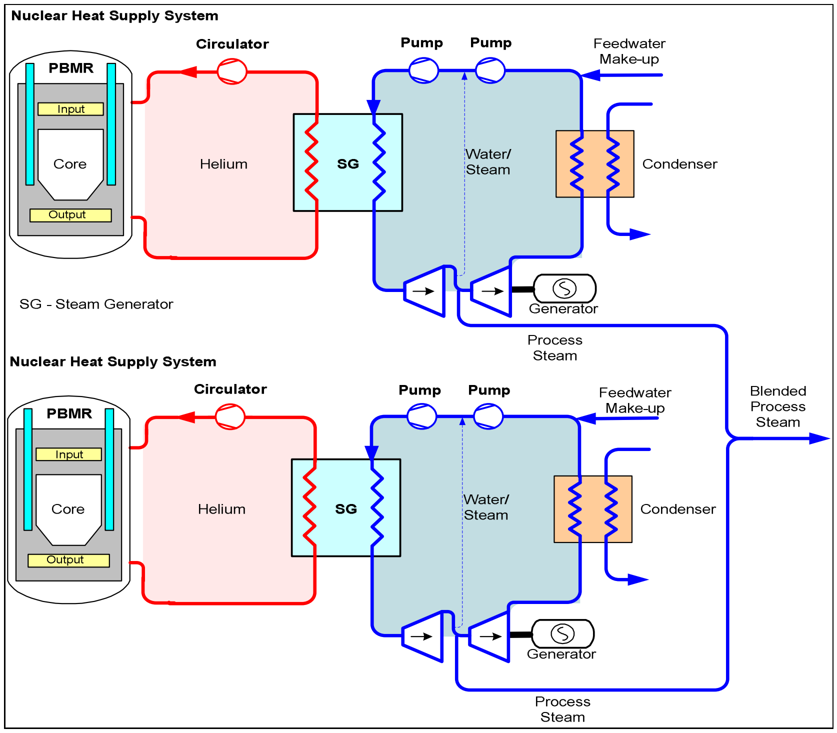


Figure 43: Two modules configuration with direct process steam coupling

5.2.1.1.2 Indirectly Coupled Steam Supply

The advantage of the indirectly coupled steam supply system is that it provides the ability to keep the reactor module running at optimal steady state conditions, isolated from the process steam demand fluctuations.

The control system is slightly more complex, but the nuclear island is buffered. Essentially the electrical power generation system also does not need to react immediately to act as a load balancer in order to dampen the transient effect on the reactor module. This configuration also allows (within a limited range as pre-determined by design) the process steam to be supplied at a variable temperature and pressure without requiring a change in reactor set point. The water steam loop can use high quality demineralized water with minimal make-up processing while the process steam loop can use water of a quality commensurate with the minimum needs of the process and the steam reboiler. The additional cost of acquiring and operating the steam reboiler (SR) system is offset by the savings in feedwater treatment, particularly if the required process steam quality does not need evaporative feedwater treatment, allowing a higher temperature feedwater supply. Additionally the decoupled system provides an additional barrier to contamination, both into- and out of- the nuclear primary coolant loop. This is for both accident conditions as well as migratory products during normal operation.

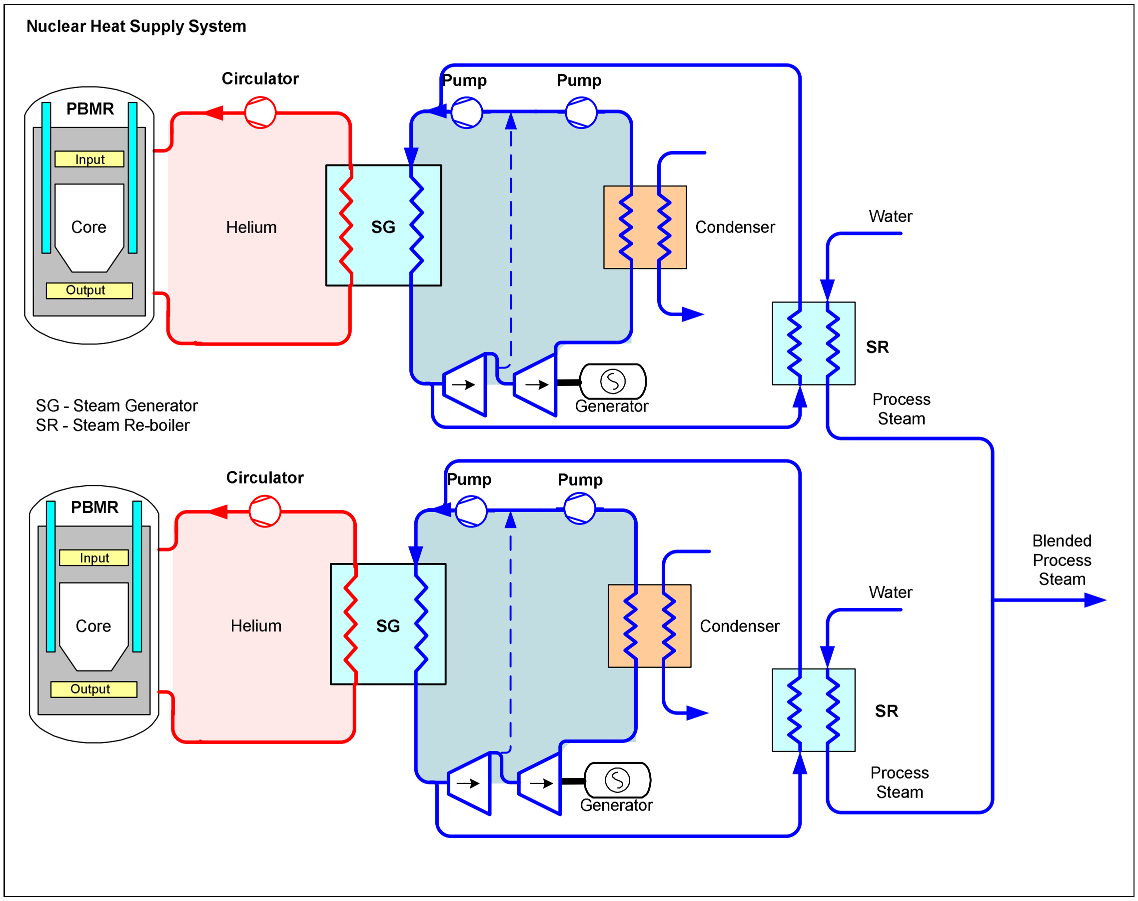


Figure 44: Two modules configuration with indirect process steam coupling

5.2.1.1.3 Steam Plant Transient Management Summary

From the illustrations and text above, it can be derived that:

- Externally initiated turn-down transients are not a limiting factor and can be managed at plant level through steam dumping or by-pass.
- Externally initiated ramp-up transients are either limited by the NHSS ramp rates, or if managed through pre-set NHSS conditions, only limited by the consumer ramp rates.
- Process steam temperature variations are more limiting than demand variations.
- A decoupled process steam supply is more resilient and flexible than a direct coupled process steam supply.
- A multi-module approach to steam supply is more flexible and more resilient to events as the internal and external initiated transients are proportionately scaled by the number of modules.

6 Availability – Allocation of reliability performance

6.1 Introduction and definitions

The availability is one important feature that has to be addressed in this project. This chapter will deal with the overall nuclear plant availability and not only the reactor one. Data concerning prismatic core reactor (ANTARES studies) and pebble-bed core reactor (PBMR studies) will be presented. The availability assessment results from a review of relevant availability analyses. The values presented here have been compared to existing PWR plants or under construction reactors such as EPR™ reactor. Some information regarding feasibility to switch on in case of industrial site transient or incident should also be broached.

First are defined availability and reliability performance indicators that are used to express the plant availability and production oriented performance.

The energy availability factor (EAF) is defined as the expected fraction of time at which the plant is capable of producing net power output over a specific time interval of interest. In this report the availability factor is assessed by the ratio of duration during which the plant is not shut down, with the operational phase duration. The duration of the shutdown phase during the plant life has to include:

- Normal planned shutdown for fuel handling operation and maintenance
- Expected shutdown, not planned, due to abnormal events, including the delay for restoring the normal plant conditions.
- The shutdown possibly necessitated by particular event occurring on another power unit in case several modules are implemented (e.g., unavailability of common equipment, particular operation during the construction, the commissioning of another power unit).

NB: This deliverable doesn't address the coupling with process heat plant. This point will be addressed in task 1.3 of the project.

During operation the plant availability may be assessed using the capacity factor that is defined as the ratio of the actual energy produced in a given time to the maximum amount of energy which could have been produced. Conservatively it is assumed that the capacity factor should be very high considering that when the power unit operates (not in shutdown state) the generated power is the nominal power; no consideration of power evolution during start-up / shutdown transients and load-follow.

Rev.1 The reliability is the probability that the plant will operate (regardless of the power required) over the entire mission time.

The plant trip frequency is the frequency of unexpected plant trips, usually expressed as the number of events per calendar year or per operating year. The plant trip can be an event or event sequence that leads first to a reactor trip and a consequential power conversion system trip. The plant trip frequency is expected not to exceed one per year.

The forced outage time is the fraction of time that a plant is down in a forced outage when it was scheduled to produce net electrical power.

The capacity loss factor (CLF) is the fraction of energy production not realized due to planned or unplanned power reductions and shutdowns, as compared to that for 100% power operation for a specific period of time, usually expressed as a percentage. The capacity loss factor is sometimes broken down into contributions from planned and unplanned losses in capacity.

For unplanned outages, capacity loss factor averaged over the plant lifetime is less than 5%. Unplanned outages of 6 months or greater contribute to less than 10% of this percentage. Unplanned outages of 2 years or greater contribute to less than 10% of the unplanned outages of 6 months or greater and less than 1% of the unplanned outages (lower than 6 months). These data were ANTARES assumptions.

6.2 Nuclear heat supply system availability

6.2.1 New nuclear power plant design basis availability

As background, it is necessary to consider the standards being set at the moment by the nuclear industry in general.

The nuclear industry is continually investigating new methods to improve the return on investment to the investor. For commercial reasons, one area the industry is improving on is overall unit availability. Many operating nuclear power plants (NPP) that were designed with technology for the 1970's and early 1980's are now operating with availabilities in the low
Rev.1 90% range. In 2007 the energy availability factor was 81% in average (see Figure 45) and half of nuclear reactors operated with EAF above 85%. The market leader for achieving high unit availability is Olkiluoto NPP in Finland. They have achieved sustained and outstanding availabilities of 94.9% for the last several years. The graphic below shows the averaged EAF's evolution over the twenty last years in the nuclear power plants in the world.

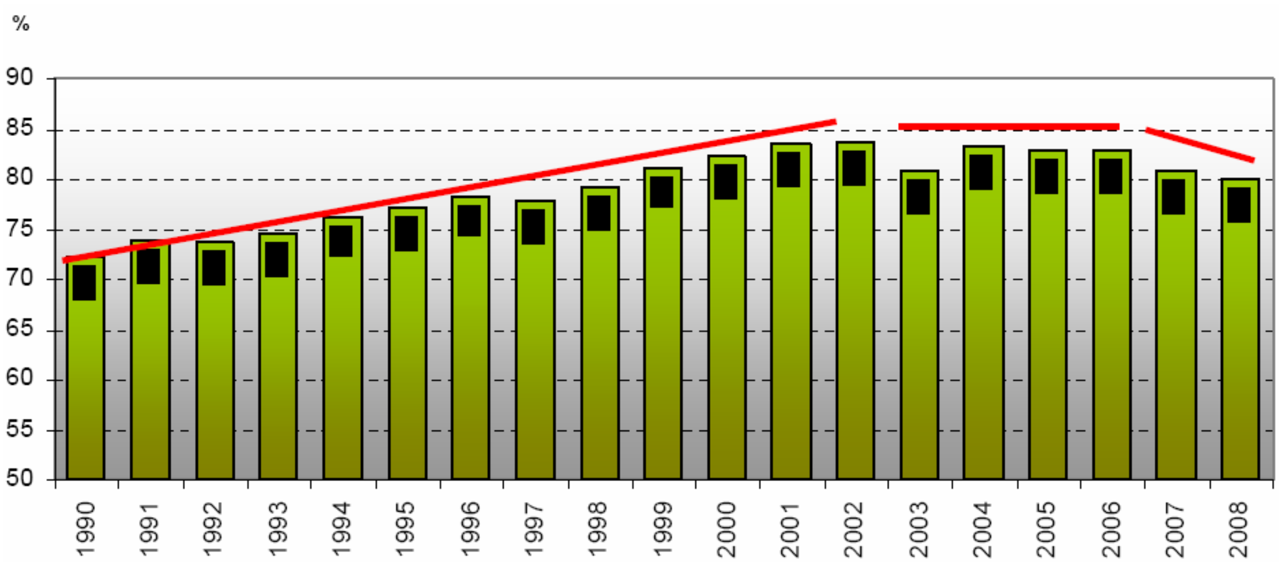


Figure 45 : Nuclear Power Plants energy availability factor worldwide weighted averaged 1991 – 2008 (SOURCE: AIEA 2009)

Allocations of availability performance requirements to major plant systems for the HTR are derived from available user requirements and design specifications. Both, EPRI and the EU User Requirements Document (URD) recommend designers incorporate lessons learned and design into the new nuclear power plants a forced outage rate of less than 5 days per year over the design life of the new NPP. Today, most NSSS designers are establishing design goals for a unit availability of 90% or higher. This includes planned, unplanned and forced outages.

6.2.1.1 HTR pebble-bed technology design basis availability

Nota: This part is based on PBMR-CG design studies.

The pebble-bed reactor has an inherent availability due to the on-line refueling capability of the design. This eliminates the need to shut down at regular intervals to refuel and re-arrange the fuel within the reactor. This also allows the maintenance down-times (for preventative maintenance) to be scheduled to coincide with those of the process steam users' down-times. The durations of these down-times are however, constrained by the transient capabilities of the NHSS (see Figure 46) meaning that the maintenance work which requires a reactor cold shutdown and restart will require the shutdown and restart durations to be added to the maintenance task duration (outside of the breaker-to-breaker duration).

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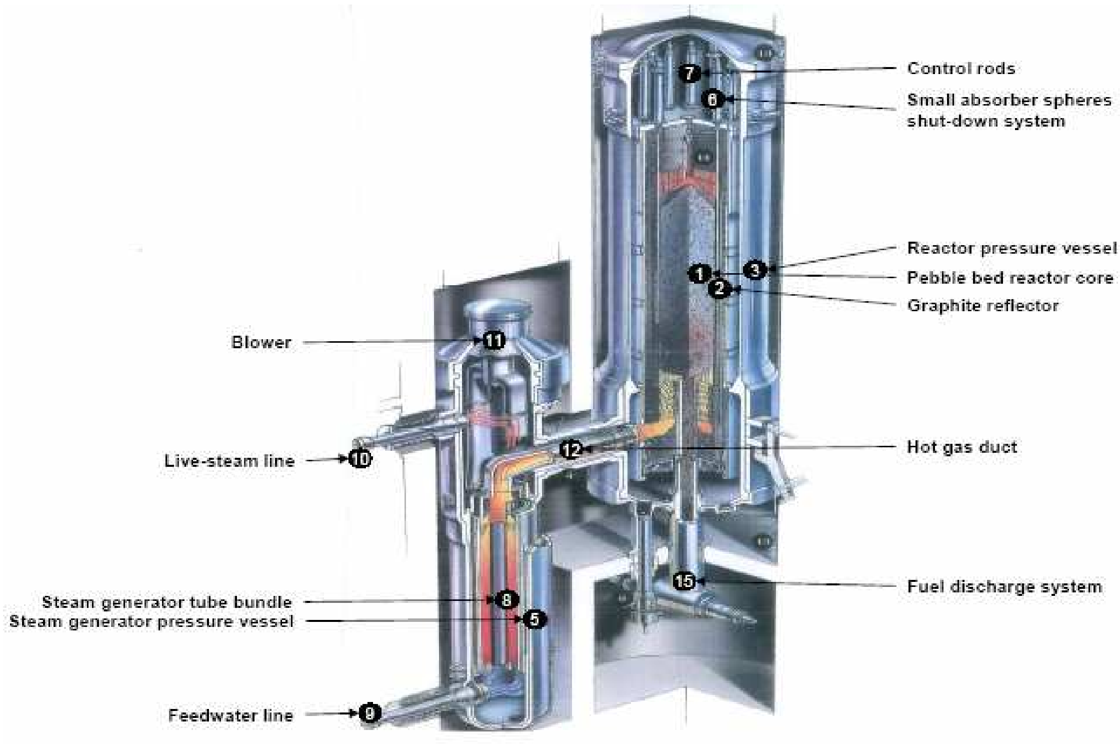

Figure 46: Nuclear Heat Supply System (NHSS)

The primary components driving maintenance shutdowns on the NHSS are:

- Primary Steam Generator
- The Fuel Handling and Storage System
- Main Helium Circulator
- Regulatory Pressure Vessel Inspections

Rev.1

On the basis of the experience of the nuclear industry, through the application of sound design for reliability, availability and maintainability principles, the maintenance events requiring shut-down or turn-down / loss of output can be scheduled for intervals as much as 5 years apart. There is nevertheless some uncertainty on components with no equivalent on other types of reactors, such as the fuel handling system and the helium circulator, but solutions have been developed to address the main deficiencies shown by these systems in the first HTRS and minimize related unavailability.

However when scheduling outages only at 5 year intervals, the probability of unplanned outages increases and the capacity factor de-rate due to SG fouling, etc becomes more relevant. For the purpose of this study of operational availability, the NHSS maintenance interval has been analyzed with a shorter planned outage interval of 2½ years, primarily for accomplishing ASME Section XI requirements (inspection part) and which should map to the process plant's outage schedule.

The PBMR-CG availability model has been derived from the available data of the operating AGR's as well as the current LWR's and the expectations of the EU and EPRI for Advanced LWR's (ALWR's) [11], as applied to the PBMR-CG design. These references provide both reliability data of existing designs and components as well as event occurrence probabilities which should be considered in conducting a Probabilistic Risk Assessment (PRA). By using the reliability and occurrence data, the allocation for forced outages was calculated. Consequently an availability roll-up for the NHSS provides the following for a 40 year NHSS life[15] for a single module cogeneration plant.

[15] The analyzed plant life of 40 years is for the current licensing basis, excluding license extensions. The design life could be for as long as 60 years, but there is no benefit to analyzing availability for the license extension at this point.

	Specified Design Target	Current Forecast Availability
Total days in 40 years	14610	14610
Total outage days	525	199.5
Planned Outage Time	3.6%	1.4%
Typical forced outage time	1.4%	1.4%
Total forecasted unavailability	5.0%	2.8%
Forecasted Availability	95.0%	97.2%

Table 13: PBMR-CG availability target

6.2.1.2 HTR prismatic core reactor design basis availability

Nota: This part is based on ANTARES design studies.

The allocation of reliability performance targets for a single reactor module is shown in the following table. The work has been done for electricity production plant and should be envelope for a plant dedicated to process heat application for which the secondary circuit is not more complex.

The following data are assumed for the Nth of a kind, mature reactor after several years of reactor experience. The values of Effective Full Power days (EFP) are given per year, this explains the decimal values.

Contribution to Capacity Loss Factor (CLF)	Nuclear Heat Supply System (NHSS)		Power Generation System (PGS)	
	Target EFP	CLF	Target EFP	CLF
Refueling	< 13.7	3.8%	/	/
Other planned outages	< 2.4	0.7%	< 2.3	0.6%
Unplanned outages – short duration (<6 months)	< 5.0	1.4%	< 10	2.7%
Unplanned outages – long duration (>6 months)	< 0.5	0.14%	< 1.0	0.3%
Reduced power operation	< 2.0	0.6%	< 4.0	1.1%

Table 14: Reliability performance targets for NOAK[16] ANTARES module

Contribution to Capacity Loss Factor (CLF)	Nuclear Heat Supply System (NHSS)	Power Generation System (PGS)	Overall reactor module
Unplanned CLF	2.0%	4.1%	6.0%
Forced outage time	1.5%	3.0%	4.5%
Capacity factor	93.6%	95.3%	89%
Availability factor	**94.2%**	**96.4%**	**91%**

Table 15: Availability for overall ANTARES module

This first allocating assessment for the NOAK plant nearly leads to a 90% capacity factor and availability factor. As for pebble-bed technology such assessment is deduced from a review of service experience with currently licensed LWR's and specified requirements for advanced reactors.

[16] NOAK = Nth Of A Kind

6.2.1.3 Comparison

The comparison in Table 16 below indicates that the performance goals above may be somewhat ambitious, but yet are in reasonable agreement with comparable goals that have been set for other nuclear reactors.

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For NHSS	Pebble-bed core (NHSS only)	Prismatic core (overall reactor module)	EPRI ALWR (overall reactor module)	EPR™ reactor (overall reactor module)
Expected lifetime	60 yr	60 yr		60 yr
Total planned outage days per year (averaged on lifetime)	13.1	18.4		10.4
Planned Outage Time	3.6%	5% Incl. refueling	11.8%	2.8%
Typical forced outage time	1.4%	1.5%		
Total forecasted unavailability	5.0%	6.4%		
Forecasted Availability	95.0%	91%	89%	91%

Table 16: Availability targets

6.3 Process heat plant Availability

When the NHSS is integrated into a multi-module NSSS plant, the loss of availability in one module while others are still operating, results in a capacity reduction and not a complete loss of availability. The more modules used to supply the bulk steam, the impact of a single module unavailability is proportionately less (n-1)/n.

The correct measure for a complete steam plant would then be the capacity factor of the plant. This figure of merit would then need to take into account the degradation systems between maintenance intervals meaning that the capacity factor will be slightly less than the availability (in percentage), however where the off-taker has a minimum capacity factor necessary to keep the process plant operational, small reductions in capacity are preferable to complete losses of capacity, thus the maintenance of the modules can be staggered to coincide with regular process plant turn-downs or maintenance shut downs.

In summary, the availability of a steam plant containing more than one NHSS will be greater than 95% while the capacity factor will be less than 95%, but better than 90%.

7 Conclusion

Based on current estimates, the HTR technology used for electricity production only, would be substantially more expensive than electricity produced from today's fleet of large LWR's.

But the modular HTR appears as a leading technology candidate for the next generation of commercial nuclear reactors, dedicated to providing both electricity and process heat to industrial facilities. Indeed, the high operating temperatures of HTR not only offers high efficiency electrical production but also covers of a broad spectrum of industrial process for process heat applications.

Beyond operating temperatures, key attributes of the modular HTR that distinguish it from the LWR are its unique safety design taking full benefit of inherent safety features of HTR (fuel, Helium coolant, huge thermal inertia, etc.). These attributes offer robust safety and simple protection systems.

Furthermore, because of extremely low radiological consequences levels, the emergency planning zone may be limited to the footprint of the plant site (approximately 450 m radius from each module). This is a crucial feature for a nuclear heat production unit that allows the plant to be sited near to the industrial process facility and not in remote locations as most of the LWR today.

Another benefit inherent in the HTR is that it has the potential to accommodate new waste reduction strategies such as actinide burning and the more effective consumption of plutonium (e.g. surplus weapons grade plutonium or plutonium recycled from reprocessed LWR spent fuel).

Its competitiveness over conventional heat supply systems is not proven yet but it is expected to rely on economics advantages of modular designs, on anticipated fossil fuel prices increase and on postulated future carbon taxes.

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APPENDIXES

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APPENDIX 1
Material issues for supercritical steam generators
(Reference [19])

As stated in 4.2.3.2, main technology limits for the SGs concern material selection. This paragraph aims to present material feedback from past HTR demonstrators, advanced gas-cooled reactors, power plant operated with advanced supercritical steam conditions, but also specific material experiments in the frame of R&D projects.
As a result, based on these data, candidate materials are emphasized depending on the SG part and the operational temperature foreseen.

A. Feedback from existing plants (supercritical steam) and past demonstrators

The first power plant operated with advanced supercritical steam conditions was Philadelphia Electric Company's Eddystone Unit 1, commissioned in 1960. As a consequence of the particularly high values of steam temperature and pressure applied, 649°C and 34 MPa, respectively, austenitic steel (TP316), was used for the header, pipes, and valves. The primary role of the station was base load generation, however serious thermal fatigue damage was incurred in thick section components during start-up and shutdown cycles. Steam conditions had to be reduced during subsequent operation of the plant.
More moderate, but significant improvements in steam temperature and pressure have been successfully achieved during the past decades. This progress has principally resulted from of the development of high-strength heat resistant ferritic steels, which exhibit lower thermal expansion coefficients and higher thermal conductivities than austenitic phases. Since the failures at Eddystone Unit 1, ferritic steels have been used exclusively for high pressure steam piping and headers.

The materials used for the pressure parts in the THTR steam generators were as follows. The economiser tubing, and feed water and reheat inlet headers, were constructed from 15 Mo 3 steel (0.15% C 0.3% Mo). For the evaporator tubing and lower stages of the super heaters 2.25 Cr 1 Mo steel was used. The finishing Superheater tubing and main and reheat outlet headers were fabricated from Alloy 800 (X 10 Ni Cr Al Ti).

As a starting point for examining the performance limits of steam generators for gas cooled reactors, it is also useful to consider the materials and service record of the steam generators coupled to the UK's Advanced Gas-cooled Reactors (AGRs). AGR power stations use the highest steam parameters of reactors in full commercial operation.

The earliest plant has recorded 30 years service. The AGR was designed to achieve the same steam conditions as a conventional *subcritical* coal-fired plant, namely 538°C and 16 MPa. The steam generators at Hinkley B and Hunterston B power stations are vertically-oriented and of once-through configuration, shown schematically in Figure **47** below. A CO_2 circulator is situated at the bottom of the steam generator, close to where coolant enters the vessel. Situated above this are the economiser, evaporator, first and second superheaters, and reheater. These subunits are constructed from serpentine tubing. (Helical coils are used in other AGRs). The economiser section is made from mild steel and 9Cr 1Mo steel is used for the evaporator and primary superheater. The secondary superheater and reheater are 316 stainless steel.

The subcritical performance limits imposed by the choice of materials used in the THTR and AGRs are discussed below.

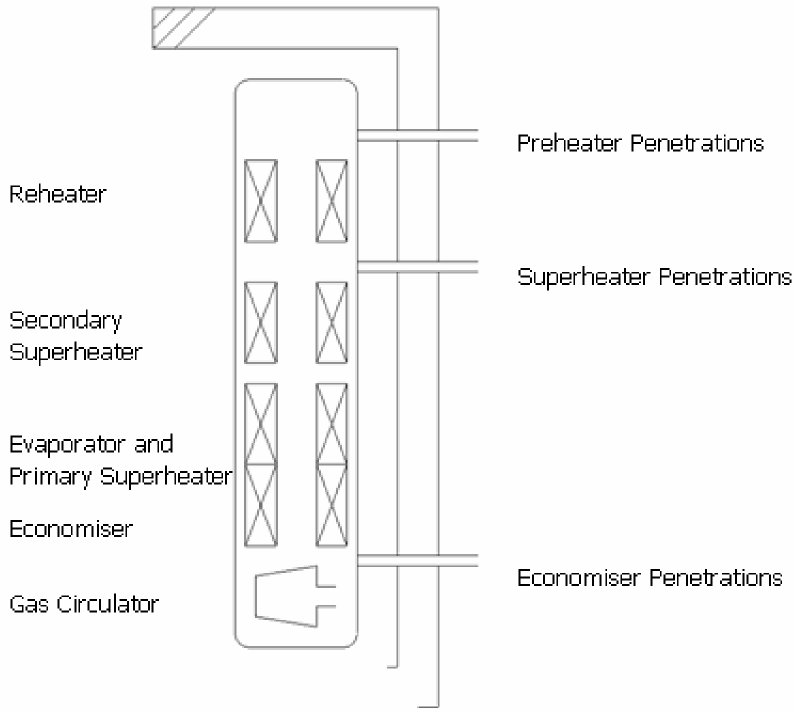

Figure 47 : Schematic cross section through a AGR Steam generator

B. Material selection for HTR SGs

As said before austenitic and ferritic steels were first selected for SG parts. In this section it will be outlined how variants of these steels will remain the dominant material for thick section SG components until 650°C, with Ni superalloys. Also alloy design and selection for headers, steam piping, and superheater and reheater tubing for use in 600°C, 650°C, 700°C, and 760°C class power stations is discussed with reference to the following criteria:

- Forging and welding capabilities
- Creep rupture strength
- Toughness
- Oxidation and corrosion resistance
- Microstructural stability over the service lifetime
- Economic considerations

B.1. Materials for 600°C Class Steam Generators

Beginning in the late 1970s, a series of coordinated R&D programs bringing together boiler manufacturers, steel producers, and research institutes, were initiated in the United States, Japan, and Europe. The aim of the respective EPDC, EPRI, and COST programs was to develop steam generators with ultra supercritical parameters in excess of 26.2 MPa and 593.3°C, which would form the basis for 600°C class power plants. The basic approach followed in these activities was to develop new alloys suitable for three categories of boiler components:

- headers and steam piping,
- superheater and reheater tubing,
- water wall tubing or steam tubing.

Despite differences in the actual SG configurations, these broad grouping correspond closely to the component requirements necessary for the construction of supercritical SGs for HTRs. From this viewpoint, water wall tubing in the coal fired boiler faces similar demands to the non-superheater tubing in a HTR SG – aside from the different corrosive environments faced.

B.1.1. Headers and steam piping

The key developments underpinning the introduction of steam generators for advanced supercritical steam conditions have been in the area of ferritic steels for headers and steam piping. Despite their good high temperature strength, the relatively high thermal expansion coefficient and low thermal conductivity exhibited by austenitic steels makes these alloys unsuitable for use in thick section components.

Consequently, the primary objective of early ultra supercritical power plant research was the development of a ferritic alloy to replace the X20 steel commonly used for headers and piping. Prior to these activities, a maximum metal-use-temperature of 565°C had been enforced. The starting point for new alloy design was the 9Cr-1MoV steel (P91), commonly used in the petrochemical industry. The choice of Cr content was governed by requirements of oxidation resistance, and also from the perspective of maximizing strength. Despite it was stated that Cr does not directly contribute to strengthening in these systems, Cr compositions of around 2% and 9-12% have been empirically found to be optimum from this perspective.

Research focused on the development of precipitation strengthening modifications, achieved through the formation of Nb and V carbides and nitrides. The result was the second generation 9-12%Cr steel, 9Cr-1Mo-V, designated P91. As with the earlier generation of ferritic steels, Mo provided a degree of solid solution strengthening. In common with other 9-12% Cr steels, the microstructure of P91 following normalizing and tempering consists of tempered martensite, ferrite sub-grains, and carbide and nitride precipitates at sites of prior austenite grain boundaries. The obstruction of dislocations by finely dispersed precipitates is the principal strengthening mechanism in this alloy system. The introduction of P91 steel directly led to the construction of the ultra supercritical power plants during the 1980s.

Further modification of the base 9Cr-1Mo-V alloy, through the addition of W in place of a fraction of the Mo content, resulted in the development of the third generation ferritic alloy P92. This steel has typically been used for the thick section components of steam generators in advanced power plants constructed since 1995, and represents today’s best available (operational) technology. Precipitate strengthening in P92 steel is achieved through the formation of a dispersion of MX carbonitrides (where M= Nb, V, Ta and X= C, N), Nb(C,N), VN plates, and Nb(C,N).

The progress achieved in ferritic alloy development over this period is highlighted by the increase of creep rupture stress, 10^5 h/ 600°C, from 35 MPa to 180 MPa between generation 1 and generation 3 alloys, respectively.

It is instructive to examine the choices made during the design of the high pressure steam headers for the THTR. It was recognized that this component required special attention during the design phase due to the wall thickness used and, therefore, its sensitivity to thermal transients. The issues were compounded in this case, due to the fact that the material used, Alloy 800, had “no service experience from fossil steam generators”. The lack of available data meant that the designer had to rely on the results of simple experiments and thermal and structural analysis, together with an effort to limit the consequences should rupture occur through the provision of heavy whip restraints.

In summary, from the perspective of steam headers and thick section components, an outlet steam temperature of 600°C is readily achievable today using service-proven P92 steel.

B.1.2. Superheater and reheater tubing

For applications in coal-fired plant, the oxidation and corrosion resistance of the advanced ferritic steels is insufficient above a metal temperature of 593°C. As a result, austenitic alloys are typically used in thin-shell superheater and reheater components. With an operating temperature of approximately 50°C above the final steam conditions, this tubing is subject to severe sulphur and chlorine-induced fire-side corrosion.

In comparison, shell-side corrosion requirements may be relaxed for HTR superheaters. In this case, the corrosive environment faced is due to the presence of impurities in the Helium coolant itself being effectively inert. Major primary side contaminants include H_2O, H_2, CO, CH_4, and N_2. Sources of these impurities include out-gassing from core and primary loop materials, water and oil ingress, and reactions with core graphite. Typical impurity concentrations are of the order of parts per million, with corresponding partial pressures between 0.1 to 10 Pa. Under the high gas flow rates imposed in a HTR, reactions between impurities and primary-side materials are not in thermodynamic equilibrium. Instead, the coolant composition at a particular temperature is dictated by kinetic considerations.

Depending on the temperature, rates of reaction, and partial pressures of the relevant molecular species, oxygen and carbon may be exchanged between the gas coolant and metal surface. Despite the low overall impurity levels, the addition or removal of carbon from the alloy may result in a significant modification of mechanical properties. Generally, carburization, (the transfer of carbon to the metal) is associated with enhanced low temperature embrittlement, whereas material transfer in the opposite direction, decarburization, results in a reduction in creep rupture strength.

Corrosion resistance is generally associated with the presence of a continuous surface oxide layer. In high-alloy steel (and superalloy) systems, the promotion of an adherent and passive thin film of Cr_2O_3 is necessary to create a barrier to corrosion and further oxidation. The oxidizing potential of the coolant mixture is dictated by the partial pressure of molecular oxygen – typically very low – and, more importantly, the ratio of the partial pressures of water and hydrogen, $P(H_2O)/P(H_2)$. In HTRs, typical values for this fraction vary between 0.01 and 1. A necessary condition for the formation of a stable oxide layer is that $P(H_2O)/P(H_2)$ is required to be $> 10^{-4}$.

Decarburization of structural materials may take place, as, for a given CO partial pressure, above a critical temperature; C is more stable in the gaseous CO than in the metal carbides. In order to discourage this reaction, it is important to maintain the partial pressure of CO in excess of 1.5×10^{-5} Pa [1]. In addition, it is also important to avoid excessive carburisation and reduction of the oxide layer through reactions with methane. For this reason, it is necessary to maintain the ratio $P(CH_4)/P(H_2)) < 0.01$.

The coolant composition in previously operated HTRs has typically exhibited stable oxidising and moderate carburising characteristics [1]. As a result, no significant primary-side corrosion has been reported. Therefore, the use of austenitic steels is not anticipated to limit the performance of 600°C class steam generators from the perspective of primary-side corrosion. The effects of impure coolant corrosion at higher temperatures are discussed in subsequent sections.

A high degree of steam-side oxidation resistance is also essential in steam generators. High water purity is typically necessary for use in once-through boilers. In addition, fossil fuel-fired supercritical power stations have commonly employed oxygenated treatment (OT) of feedwater. In this process, oxygen is injected into the boiler water in order to promote and maintain the oxide passive layer on metal surfaces. Despite concerns that OT may facilitate stress corrosion cracking, austenitic steels have a good operational record when used for superheater tubing this class of power plants.

It is worth noting that the operational environment faced in supercritical steam generators is significantly more benign than is expected to be the case in the nuclear super-critical water reactor. For reasons of neutron economy, fuel, control, and water rods in the SCWR will need to be fabricated with tube thicknesses of the order of 0.4-0.6 mm, as opposed to the 6-12 mm used in superheat tubing. With oxide thicknesses of up to 300 μm possible, considerable reductions in structural integrity are conceivable in this reactor type. In addition, radiolysis of supercritical water is expected to heighten the level of corrosive attack.

Austenitic steels are commonly categorized by Cr content. Alloys with less than 20% Cr include the 18Cr-8Ni family of steels. Austenitic steels with this basic composition have been used in superheater and reheater tubing for several decades.

During this period, alloying elements Ti and Nb were added to austenitic steels in order to promote stabilization and corrosion resistance [2]. In subsequent generations of these steels, the Ti and Nb content was reduced in order to enhance creep strength. More complex alloy compositions include W for solid solution strengthening, Cu to promote precipitation strengthening, and N to aid stabilisation [3].

Higher alloy 20Cr-25Ni steels have more recently been successfully employed in superheater and reheater tubing [4]. Cr content in excess of 20% provides excellent high temperature corrosion resistance and strength at 600° [5]. The 25% Ni content suppresses the precipitation of the σ-phase, thus avoiding embrittlement [6]. Higher Cr and Ni alloyed steels, such as 30Cr-50Ni, exhibit excellent high temperature oxidation and corrosion resistance. However, the cost of these alloys is often prohibitive.

A major concern of using austenitic superheater piping relates to the welding of these components to the ferritic steel header. Typically, a post-weld heat treatment will be necessary for the ferritic steel which can lead to sensitisation of the austenitic alloy [7]. In the AGRs, the transition joint weld between the ferritic evaporator tubing and austenitic superheater tubing has an upper temperature limit of 500°C (a lower temperature limit exists for the reheaters due to the requirement to avoid saturated steam conditions and the possibility of stress corrosion cracking) [8].

In summary, a large number of austenitic steels commercially available and economic suitable for super-heater tubing in 600°C class supercritical SGs for HTRs are available.

B.1.3. Water wall tubing

Low alloy steels 1.25Cr-0.5Mo, 2.25Cr-1.0Mo, and 15Mo3 are commonly used for the water wall piping of 600°C class boilers in fossil-fuel fired plant. The maximum mid-wall temperature under these operating conditions is of the order of 500-520°C. This temperature is close to the maximum achievable using low alloy steels for easily welding. For example, the design limit for 1.25Cr-0.5Mo corresponds to steam conditions of 600°C and 28 MPa [9]. Variants of high strength 2-1/2%Cr steels, including T23, are likely to be used in the future [10].

B.2. 650°C Class Steam Generators

As outlined in the preceding sections, 600°C class steam generators represent the current best available technology, and steam parameters of the order of 625°C and 32 MPa are likely achievable following established approaches.

The natural progression from this point would be to develop materials suitable for operation in 650°C class plant. It was initially hoped that fourth generation 12%Cr ferritic steels, exhibiting enhanced corrosion resistance over 9%Cr alloys, would be suitable for the construction of thick section components in this class of power plant [3]. However, the route to developing ferritic steels with further enhanced performance parameters has proved to be more difficult than initially thought. As a result, large-scale research projects, described below, were instigated to span the gap between 600°C class and 700°C class plant.

B.3. 700°C Class Steam Generators

Research programs aimed at developing boiler and turbine materials suitable for steam conditions up to 700°C and 38 MPa, corresponding to a plant thermal efficiency of 52-55%, have been organized in Europe and the United States. The European effort, THERMIE Advanced Power Plant (AD700), proceeded between 1998 and 2005 [11]. The equivalent US DOE program, termed “Vision 21 Power Plants”, was undertaken between 2001 to 2006 [2].

The research component of both programs initially focused on the development of new ferritic and austenitic alloys with creep rupture performance of the order of 100 MPa/ 10^5 h at 650°C and 700°C, respectively [12]. In addition, it was recognized from the outset that components exposed to the highest operational temperatures in this class of power plants would require a level of creep rupture strength of the order of 10^5 h/ 100 MPa at 700°C, necessitating the use of Ni-based superalloys.

B.3.1. Thick section components and steam piping

The original aim of the research activities relating to thick section components was to develop variants of existing 9-12% Cr ferritic steel suitable for 650°C operation [13]. However, no ferritic alloy with superior long-term creep performance in comparison to P92 was successfully produced [14]. For example, the advanced 12% Cr steels NF12 and SAVE12 were initially identified to fill this role, however further work has suggested that oxidation resistance may be significant degraded above 620°C [15].

Research into the use of Ni-based superalloys for thick section components within the THERMIE program focused on Alloy 263, a high-strength variant of Alloy 617 [18]. The feasibility of manufacturing large components using this alloy system were investigated by processing a 2000 kg ingot of Alloy 263, using conventional pipe production methods. The resultant pipe was 4.5 m in length, had an outside diameter of 310 mm and was 66 mm thick. Welding trials of thick sections of Alloy 263 were successful using gas metal arc welding. Creep strength requirements were exceeded for 700°C class operation [16].

In addition to the demands of developing new manufacturing processes for superalloy fabricated thick section components, the basic economics also appears challenging [17]

B.3.2. Superheater and reheater tubing

Austenitic steels capable of satisfactory performance above 620°C were identified in the section concerned with superheater tubing for 600°C class boilers. New austenitic steel, alloy 174, characterized by creep rupture strength of 150 MPa under 650°/ 10^5 h conditions, was developed as part of the AD700 program.

This represents a substantial improvement over existing high-performance austenitic steels, such as 304h, where creep rupture strength of 125 MPa (650°/ 10^5 h) represents best in class performance [19].

It is recognized that steam conditions above 650°C will necessitate the use of Ni-based superalloys for superheater tubing [20]. While existing superalloys met the desired creep requirements, a new Ni-alloy with superior corrosion resistance was developed by the THERMIE partners. The starting point for this work was NIMONIC Alloy 263. Over 30 melts representing variants of this alloy were produced. Creep rupture strengths of up to 175 MPa at 700°C/ 10^5 h were recorded. A similar level of performance was also observed in the case of the commercial alloy INCONEL 740.

B.3.2.1. Steam tubing

Significantly more oxidation resistant and stronger steam tubing will be necessary under 700°C conditions than has been used previously in supercritical SGs [36]. 9-12% Cr steels, in particular 12Cr-1Mo-1W-V-Nb HCM12, are candidate alloy systems for this role. The creep strength and oxidation resistance of Alloy 617 meets the requirements of steam tubing at these temperatures; however the cost would likely to be of the order of 10 times higher than for advanced ferritic steels.

Investigations into the corrosion resistance of Ni-based superalloys in impure Helium environments at temperatures up to 950°C have been undertaken (in the frame of NGNP project). Hastelloy X was found to be the most corrosion resistant alloy of those studied, followed by Alloy 230, Alloy 617, and then Alloy 800H. Other research has identified the high corrosion resistance of Haynes 230. Since some level of corrosive attack has been observed in all these alloy systems, it will be important to identify and maintain a level of coolant chemistry which is slightly oxidizing at the operational temperature [1].

Preliminary studies of the steam-side oxidation behaviour of similar Ni-based alloy systems have been carried at steam temperatures between 700 and 800°C. In this case, the most oxidation resistant materials were the high Cr nickel alloys: Alloy 617, Alloy 230, Alloy 740, and HR6W [21].

Synthesizing these two studies, it may be concluded that at least Alloys 617 and 230 exhibit the necessary primary and steam-side corrosion and oxidation resistance to enable 700°C steam conditions.

B.4. 760°C Class

The US DOE's program for the development of materials and components for use in ultra supercritical steam generators has particularly ambitious final target steam conditions of 760°C and 37.8 MPa [22]. In this work, Ni-based superalloys (Haynes 230, Inconel 740, and CCA 617) have been proposed for use in thick section components and the highest temperature tubes. Creep strength requirements are of the order of 100 MPa under 750°C/ 10^5 h conditions.

The long-term stability of the carbide and Cu precipitates, γ' and η phases, and the intermetallic laves phases, responsible for the excellent creep rupture strength of the superalloy systems considered, requires additional investigation. For example, in preliminary studies, Haynes 230, an alloy system which does not employ γ' strengthening, has been found to suffer less strength degradation than Inconel 740, CCA617 and Inconel 617, in which a high precipitate coarsening rate of the γ' phase has been observed [23]. Preliminary studies regarding welding and fabrication with superalloys have also been undertaken as part of the program.

Austenitic steels (HR 6W and SUPER 304H) are also being investigated for use in superheater and reheater tubing. Advanced ferritic steels SAVE 12, T92, T23 and HCM 12 may retain a role in lower temperature tubing applications, even under such demanding operational conditions. In addition to the work on alloy design and selection for bulk components, a number of coating and cladding options have been investigated [24]. Diffusion and sprayed chromium, chromium-silicon, and chromium-aluminum coating were found to aid oxidation resistance in the case of ferritic steels.

C. Conclusion

All seven of the high temperature nuclear reactors constructed have been coupled to steam generators. In all cases, subcritical steam conditions were used (they were also planned for the MHTGR). Of the reactors, the THTR used the highest steam temperature and pressure, 550°C and 18.5 MPa, respectively. The reference designs of HTRs under recent development have generally used direct Brayton cycles or gas/ gas intermediate heat exchangers. As a consequence, relatively little progress in steam generators for HTRs has occurred. The coupling of nuclear reactors to supercritical steam generators has previously been proposed for lead cooled and sodium cooled fast reactors, and for intermediate-cycle supercritical light water reactors [25–27]. However, no prototype nuclear-heat supplied supercritical SG has been constructed.

In contrast, fossil fuel fired boilers with supercritical steam conditions of 600°C and 30 MPa are commercially mature products, with 10-15 year operational records. Furthermore, it is generally accepted that steam parameters of 625°C and 34 MPa are achievable using close variants of the currently used, and code approved, alloys. While the configuration of steam generators for high temperature nuclear reactors differs considerably from those used in fossil-fuel fired plants, the majority of issues relating to high temperature creep performance and steam-side oxidation resistance are shared [58]. Important differences arise in terms of requirements for corrosion resistance. The corrosive environment faced in a fossil fuel-fired plant consists of combustion gases. While in a HTR, resistance to Helium corrosion is important.

In both nuclear and fossil fuel fired plants, the performance limits of steam generators are principally set by the high temperature creep strength, thermal fatigue resilience, and oxidation resistance of the materials used for thick section headers and piping. Currently, the best performing alloy in this role is P22 steel – a product of several collaborate efforts in the 1980s to attain advanced supercritical steam conditions. Despite intensive research, no ferritic steel with higher long-term creep strength has been developed.

The steam header in the THTR was fabricated from Alloy 800. As was recognised during the design stage, the use of this alloy-type carried risk due to lack of information available on the effects of long term service. The potential consequence of using new alloy-systems in these critical components is highlighted by the fatigue failure of the austenitic steel header in Eddystone Unit 1.

The development of HTR steam generators suitable for steam conditions in excess of 650°C will need to employ Ni-based superalloys for thick section components. In order to mitigate risk, it is recommended here that alloy design and selection should build on the extensive fossil fuel-focused research activities of the Thermie AD700 and Vision 21 programs. Additional work is necessary to characterize the long-term stability of strengthening precipitates and phases in the candidate alloy systems. In particular, a better understanding of the effects of changes in microstructure during heat treatment and welding are required. Solid solution strengthened alloys present good welding capabilities, however lack strength at temperatures above 700°C. Steam parameters of 700°C and 38 MPa are conceivable with additional developments.

Austenitic steels suitable for superheaters and reheaters with operating parameters in excess of the current best-performing power plants are available. The performance limits for these components is likely to correspond to steam conditions of 650°C. Once again, the use Ni-based superalloys will likely be necessary for superheater tubing in 700°C class steam generators.

Achieving a balance between performance and cost is particularly important for steam tubing. Lower creep strength and less oxidation resistant materials are appropriate in this role than for superheater and reheater tubing. The need to construct long heating tube lengths, however, necessitates the use of alloys with good manufacturability and welding capabilities. More highly alloyed ferritic steels are the leading candidate for this role.

The **Table 17** the below synthesizes the precedent paragraphs and shows steam conditions, thermal efficiencies, and steam generator component materials, of representative subcritical and supercritical fossil-fuel and nuclear power plants.

Power Plant or Project	**Pressure (MPa)**	**Temperature: Superheat / Reheat (°C)**	**Plant Efficiency LHV (%)**	**Header & Steam Piping**	**Superheater & Reheater Finishing**	**Steam Tubing**
THTR	18.5	550	41	Alloy 800	10 Cr Mo 9 10	15 Mo 3
AGR	16	538	42	9Cr 1Mo	SS 316	C-Steel
Typical Coal Subcritical	16.8	538 / 538	38	9Cr 1Mo	T22	T11
Typical Coal Supercritical	24.5	565 / 565 / 565	44	9Cr 1Mo	T22	T11
Nordjyland 3	28.9	580 / 580 / 580	47	9Cr 1Mo	T22	T11
Best Available Technology	31.5	593 / 593 / 593	49	T91, 304H	T91, 304H, 347	C-Steel, T11, T12, T22
COST-522 Programme	34.5	650 / 650 / 650	50	P92	TP347 HFG Super 304 H/P-12	T11, T12, T22 T23 (HCM12)
THERMIE AD700	38	700 / 720 / 720	50-55	SAVE12 + NF12	NF 709 Inconel 617	T11, T12, T22 T23 (HCM12)
EPRI/ Parsons Corp.	37.8	700 / 700 / 700	52 (Estim.)	Alloy C263 (Modified Alloy 617)	Alloy 263 Inconel 740	HCM12 Alloy 617
DOE USC	38.5	760 / 760 / 760	55 (Estim.)	Haynes 230, CCA617, Inco 740	Super 304H HR6W	

Table 17: Summary of materials used in existing plants or tested in projects

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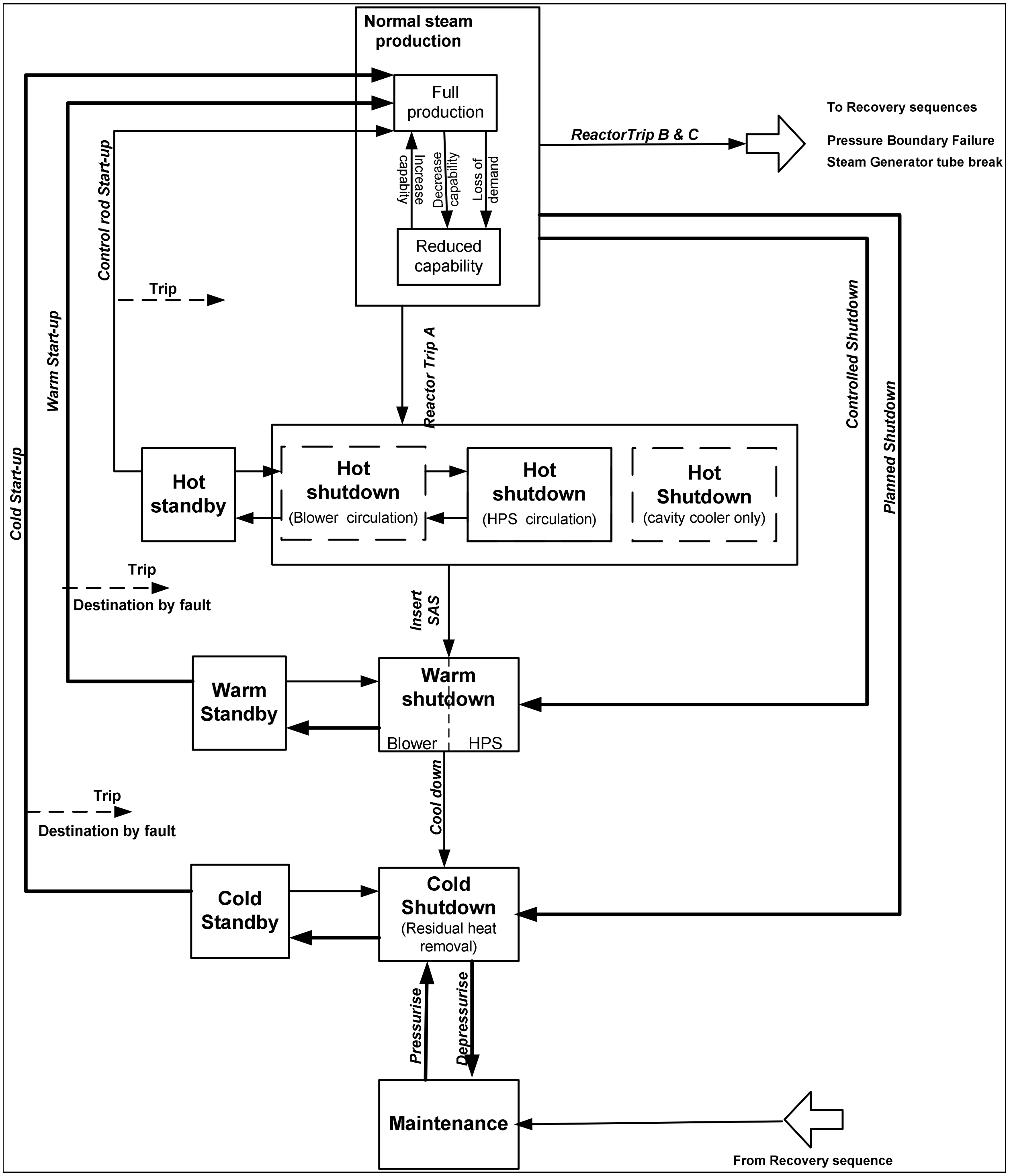
APPENDIX 2

PBMR-CG state transition diagram for normal operation of a single module



APPENDIX 3

PBMR-CG operating states

State	Description	Defining parameters
Normal Steam Production	Deliver the required quantity of superheated steam to the external user and to generate and export power to the grid within the constraints imposed by the system capacity and the external user steam requirement	Normal operating range lies between 50% and 100% of rated capacity. Cavity cooler is operational.
Hot Shutdown	Plant enters this state when it is no longer possible to continue with normal operation due to operational or safety reasons, which are expected to be of relatively short duration. (less than 48 hours).	Reactor – sub critical. Control rods inserted. Small Absorber Spheres (SAS) not inserted. Depending on the situation, The reactor is cooled by means of the blower and Start Up/Shutdown circuit (SU/SD) or the HPS cooler. Cavity cooler is operational in both cases. With loss of normal power, the reactor is cooled by means of the cavity cooler running on emergency power.
Hot Standby	The process function of this state is the preparation of the plant for Control Rod Start Up. All supporting systems are initialized and confirmed available, and there is a confirmed demand. The plant is in this state for a short duration, with the intent of progressing with the Control Rod Start Up.	Reactor – sub critical. Control rods inserted. Small Absorber Spheres (SAS) not inserted. The reactor is cooled by means of the blower and Start Up/Shutdown circuit (SU/SD). Cavity cooler is operational.
Warm Shutdown	The process function of this state is to cool reactor down to lower temperatures and maintain sub-criticality with the SAS inserted. Shutdown margin is such that reactor can be cooled to maintenance state.	Reactor – sub critical. Control rods inserted. SAS inserted. The reactor is cooled by means of the blower and SU/SD in water only mode. Cavity cooler is operational.
Warm standby	The process function of this state is to prepare the plant for Warm Start Up. All supporting systems are initialized and confirmed available. This plant is in this state for a short duration, with the intent of progressing with the Warm Start Up.	Reactor – sub critical. Control rods inserted. SAS inserted. The reactor is cooled by means of the blower and SU/SD circuit, maintaining hot gas temperature at 250°C (TBC). Cavity cooler is operational.
Cold shutdown	The process function of this state is to maintain the reactor sub critical at a temperature of 50 - 150°C (TBC).	Reactor – sub critical. Control rods inserted. SAS inserted. Long term heat removal via SU/SD circuit for the first 175 hours after the onset of the trip, then the blower is stopped as a prelude to entering maintenance state. Cavity cooler is operational.
Cold standby	The process function of this state is to maintain the reactor sub critical at a temperature of 50-150°C (TBC) and ready to enter the Cold Start-up Transition. The primary characteristic is that all post-maintenance checks must have been done and all work permits signed off, all support systems and the steam systems must be operational but not necessarily at operating temperature.	Reactor – sub critical. Control rods inserted. Small Absorber Spheres inserted. Long term heat removal via SU/SD circuit. Cavity Cooler is operational. All protection systems must have passed pre-start checks.

State	Description	Defining parameters
Maintenance (fuelled)	The process function of this state is to provide access for inspection and maintenance.	Reactor – sub critical. Control rods inserted. Small Absorber Spheres inserted. Primary pressure boundary depressurized Long term heat removal via Cavity Cooler. Blower and SU/SD circuit may be operational.
Maintenance (de-fuelled)	In the event that access to the reactor internals is required, the design makes provision to transfer the used fuel from the reactor to storage and to reload it after completion of the work. The process function of this state is to provide access for replacement of the reactor internals.	Reactor – sub critical. Control rods inserted. Small Absorber Spheres inserted. Primary pressure boundary depressurized Long term heat removal via the Cavity Cooler.

APPENDIX 4

PBMR-CG allowed operating state transitions

Transition	Description	Defining parameters
Increase or decrease of capacity	Ramp up or ramp down of capacity, as demand changes or necessitated by operating constraints. Normal operating range lies between 50% and 100% of rated capacity.	Normal power ramp rate of 1% per minute. Design can accommodate faster rates.
Loss of Demand	Loss of demand of electricity or steam demand or both, without tripping the reactor.	Designed for complete loss of steam or electricity demand. In the case of a complete loss electricity demand, the design caters for unit islanding to house load, all without tripping the reactor.
Reactor Trip A	In the event of this type of reactor trip, Rods are inserted, and in certain cases the blower is tripped and the Steam Generator (SG) is isolated. If the blower is tripped, Residual Heat Removal (RHR) is done via the Helium Purification System (HPS) cooler and the cavity cooler. If the blower is not tripped, Residual Heat Removal (RHR) is done via the blower and the SU/SD circuit and the cavity cooler.	Reactor is tripped for: • Reactivity increase/decrease and reactivity accidents • Blower failure and inadvertent blower damper actuation • Feed-water line break • Main steam line break • Normal power failure (Emergency power operation) • Other less important cases
Reactor Trip B	In the event of this reactor trip, Rods are inserted, blower is tripped, Steam Generator (SG) is isolated and the primary system is isolated. Residual Heat Removal (RHR) is done via the cavity cooler.	Reactor is tripped for: • Primary system depressurization
Reactor Trip C	In the event of a reactor trip, Rods are inserted, blower is tripped, Steam Generator (SG) is isolated and the SG is purged and re-isolated. Residual Heat Removal (RHR) is done via the HPS cooler and the cavity cooler.	Reactor is tripped for: • Steam generator tube break
Controlled shutdown	Shutdown from Normal Steam Production to Warm Shutdown. Reactor Outlet Temperature (ROT) and reactor power are reduced in a controlled fashion, and the SAS are inserted towards the end of the transition to maintain shutdown reserve.	Transition takes 4-8 hours (TBC). Limited by a 2℃ /min ROT rate to protect graphite.
Planned shutdown	Shutdown from Normal Steam Production to Cold Shutdown. Reactor Outlet Temperature (ROT) and reactor power are reduced in a controlled fashion, and the SAS are inserted towards the end of the transition to maintain shutdown reserve.	Transition takes 6-12 hours (TBC). Limited by a 2℃ /min ROT rate to protect graphite.
Insert SAS	Regression from Hot Shutdown to Warm Shutdown. Usually done when it is clear that the plant must be taken to Warm Standby state before it is re-started. SAS are inserted to maintain shutdown margin.	SAS insertion takes <10 minutes (TBC)
Cooldown	Regression from Warm Shutdown to Cold Shutdown. Usually done when it is clear that the plant must be taken to Cold Standby state for maintenance or other operating reasons before it is re-started.	Cool-down takes 2-4 hours (TBC). Limited by a 2℃ /min ROT rate to protect graphite.

Transition	Description	Defining parameters
De-Pressurize	Regression from cold shutdown to Maintenance. Primary coolant is extracted to the Helium Services System (HSS) via the HPS.	Depressurization takes 20-24 hours (limited by the HPS).
Pressurize	Progression from Maintenance to Cold Shutdown. Primary coolant is inserted from the Helium Services System (HSS) via the HPS. Plant must be taken to cold standby state before it may be re-started.	Pressurization takes 20-24 hours (limited by the HPS).
Cold start-up	Progression from cold standby state to normal steam production state. Reactor is first heated by running the blower. Reactor is made critical by removing SAS in a controlled manner. Reactor Outlet Temperature (ROT) and reactor power are increased in a controlled fashion.	Transition takes 12-24 hours (TBC). Limited by a 2℃ /min ROT rate to protect graphite.
Warm start-up	Progression from warm standby state to normal steam production state. Reactor is made critical by removing SAS in a controlled manner. Reactor Outlet Temperature (ROT) and reactor power are increased in a controlled fashion.	Transition takes 10-18 hours (TBC). Limited by a 2℃ /min ROT rate to protect graphite.
Control rod start-up	Progression from hot standby state to normal steam production state. Reactor is made critical by removing control rods in a controlled manner. Reactor Outlet Temperature (ROT) and reactor power are increased in a controlled fashion.	Transition takes 1-3 hours (TBC). Limited by a 2℃ /min ROT rate to protect graphite.