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Prepared for the U.S. Nuclear Regulatory Commission
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Contract DE-AC07-05ID1451

High Temperature Gas Reactors: Assessment of Applicable Codes and Standards

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BK Mcdowell

October 2015

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Pacific Northwest National Laboratory
Richland, Washington 99352

Summary

This document represents an update to the original version of this report on the subject code and standards for high temperature gas reactor applications. In particular this report addresses the comments provided by INL and documented in TEV-1463 Rev. 0 (INL, 2012). In addition, this report includes updated information on codes and standards for high temperature gas cooled reactors (HTGRs) at the time of writing, as well as additional information on leak-before break (LBB) issues as applied to HTGRs, and foreign codes and standards.

This report pertains to the class of modular reactors with a nominal power rating per unit up to 600 MWe. HTGRs are categorized as Generation IV reactors, characterized by very high temperature operation, and capable of electric power generation, as well as process heat. This last aspect is particular important for this type of plants, since due to their small footprint, HTGRs can be deployed in remote areas where limited access to a power distribution grid, make them very appealing for hydrogen generation and water desalination applications. In addition, this report contains updates from the latest revision of the 2013 revision of the ASME Code Boiler and Pressure Vessel Code Section III Division 5, providing rules for the construction of high temperature reactors, addressing both liquid metal reactors and HTGRs.

Considering these topics in the order they are arranged in the text, first the operational histories of five shut-down and two currently operating HTGR plants are reviewed, leading the authors to conclude that while small, simple prototype HTGR plants operated reliably, but some of the larger plants, particularly Fort St. Vrain, had poor availability. Safety and radiological performance of these plants has been considerably better than LWR plants. Petroleum processing plants provide some applicable experience with materials similar to those proposed for HTGR piping and vessels.

At least one currently operating plant – HTR-10 – has performed and documented a leak before break analysis that appears to be applicable to proposed future US HTGR designs.

Acronyms and Abbreviations

AGR	Advanced Gas-cooled Reactor
ASME	American Society of Mechanical Engineers
ASME BPVC	ASME Boiler and Pressure Vessel Code
ASTM	American Society for Testing and Materials International
ATR	Advanced Test Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor (Working Group Test Reactor)
BISO	bistructural isotropic
DOE	U.S. Department of Energy
FSV	Fort St. Vrain
GT-MHR	Gas Turbine-Modular Helium Reactor
HTGR	high temperature gas reactor
HTR-10	High Temperature Test Reactor (China)
HTTR	High Temperature Test Reactor (Japan)
HTR-NPR	High Temperature Gas Reactor-New Production Reactor
INL	Idaho National Laboratory
ISI	in-service inspection
LBB	leak-before-break
LWR	light water reactor
MHTGR	modular high temperature gas-cooled reactor
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
PBMR	Pebble Bed Modular Reactor
PCRV	prestressed concrete reactor vessel
PNNL	Pacific Northwest National Laboratory
PRA	Probabilistic Risk Assessment
PyC	pyrolitic carbon
RPV	reactor pressure vessel
SSC	structures, systems, and components
THTR	Thorium High Temperature Reactor
TRISO	tristructural-isotropic
VHTR	very high temperature reactor

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1.0 Introduction

This document represents an update to the original version of this report on the subject code and standards for high temperature gas reactor applications. In particular this report addresses the comments provided by INL and documented in TEV-1463 Rev. 0, (INL, 2012). In addition, this report includes updated information on high temperature gas cooled reactors (HTGRs) at the time of writing, as well as additional information on leak-before break (LBB) issues as applied to HTGRs, and foreign codes and standards.

A summary is presented on the existing literature on HTGRs in terms of operational history, LBB analysis, materials, codes and standards, and preferred terminology for the two large vessel configurations proposed for the Gas Turbine-Modular Helium Reactor (GT-MHR) and similar HTGR plants proposed for the near future.

This report pertains to the class of modular reactors with a nominal power rating per unit up to 600 MWe. HTGRs are categorized as Generation IV reactors, characterized by very high temperature operation, and capable of electric power generation as well as process heat. This last aspect is particular important for this type of plants, since due to their small footprint, HTGRs can be deployed in remote areas where limited access to a power distribution grid, make them very appealing for hydrogen generation and water desalination applications.

Current interest expressed by industry in HTGR plants has prompted the U.S. Nuclear Regulatory Commission (NRC) to task Pacific Northwest National Laboratory with assessing the currently available literature related to codes and standards applicable to HTGR plants, summarizing the operating history of past and present HTGR plants, and evaluating the proposed designs of reactor pressure vessel (RPV) and associated piping for future plants.

This document reviews the operational histories of five decommissioned and two currently operating HTGR plants. Safety and radiological performance of these plants is noted. Additionally, this document examines the LBB concept as applied to HTGR plants and next considers the materials that have been used and are proposed for new construction, as well as the codes and standards applicable to HTGR plants. Finally, terminology for the cross-duct component between the RPV and power conversion vessel is discussed, considering the important regulatory requirements that the cross-vessel connection should be designed as a vessel component.

Finally, as a point of comparison, some petroleum refineries provide some applicable experience with materials used in similar high-temperature, high-pressure applications to those proposed for HTGR piping and vessels.

2.0 Operational History of Seven Well-Known HTGR Power Plants

While several smaller, lesser known gas cooled reactors have been built and operated, the seven plants considered here have the most operational history available and are more similar to proposed future units than the few other units mentioned in literature.

Reference 2 states the defining characteristics of HTGRs as ceramic fuel, graphite moderator, and helium coolant. Based on this definition, the British CO_2 cooled reactors, both the lower-temperature Magnox and the higher temperature advanced gas cooled reactors, are not HTGR plants.

Daniels Farrington, a professor of chemistry at the University of Wisconsin, first proposed gas-cooled nuclear reactors in 1942. His first report on the topic was published in 1944 (Daniels 1944). Current design features such as having most of the plant below grade, a direct cycle turbine, and pebble bed cores are all concepts originally proposed by Daniels (Simnad 1991).

Figure 1 shows the approximate dates of construction, operation, and major outages for the HTGR plants. All of these plants except Dragon (Winfrith, England) and HTTR (Tokaimura, Japan) were equipped to generate electricity, although AVR (Julich Research Center, West Germany), Peach Bottom (Delta, near Harrisburg, Pennsylvania), and HTR-10 (Tsinghua University, Beijing, China) were all small enough that they can be considered prototype rather than commercial power plants.

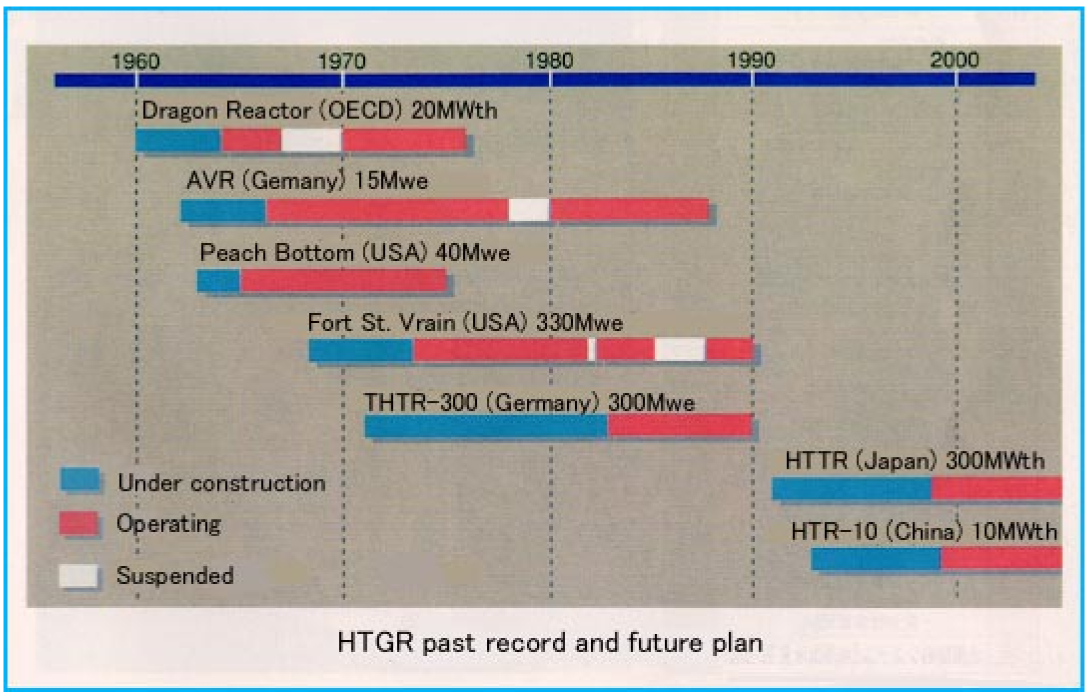

Figure 1. Construction and Operational Time Frames of Seven HTGR Power Plants (http://www.jaea.go.jp/jaeri/english/ff/ff43/randd01.html)

To date, HTGR plant history can be divided into two generations: the first generation, from 1960 to 1990, were American and European (mostly German) plants that were all shut down by 1991; and a second generation consisting of two Asian plants, both of which are still operating as of this writing. The HTGRs categories in the Western hemisphere can be further partitioned into first generation gas cooled reactors (GCR) or Magnox similar in design to Calder Hall, and second generation advanced gas reactor (AGR) similar to Windscale and Hinkley Point B.

The Peach Bottom and Fort St. Vrain (FSV) plants were built and operated in the United States. Paper studies were also performed on several proposed HTGR plants, starting with the 10 large HTGR reactor designs (five stations, each with two reactors) that were sold by General Atomics between 1971 and 1974. Licensing documents, including construction permit safety evaluation reports and letters from the Advisory Committee on Reactor Safeguards, were prepared for two of these (the Fulton and Summit plants), although none of these large plants were ever built. Docket numbers were assigned for Fulton (50464), Summit (50463) (Beck et al. 2010). A U.S. Department of Energy (DOE)-sponsored HTGR program began in the late 1970s, initially focused on an 840 MWth plant with a once-through low enriched uranium/thorium fuel cycle. In the mid-1980s, DOE turned to a smaller modular design (a modular HTGR) with an annular core and steel RPV, and a conceptual design was completed in 1987. *Draft Pre-Application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor* (NUREG-1338) was written in early 1989. During the late 1980s, the modular HTGR was considered by DOE as one of two possible reactor technologies for materials production (Goodjohn 1991). Preliminary design work was performed on this modular HTGR-NPR, but the plant, originally expected to start in 1998, was never built (Hoffman and Mears 1991).

Four of the seven plants constructed and operated used prismatic block fuel; the remaining three used pebble bed type fuel. The operational experience considered in this report does not show any obvious advantages to either fuel type.

Sections 2.1 through 2.7 present an overview of the operational experience for seven HTGR power plants. The plants are presented in the chronological order of their initial operation, from Dragon in 1964 to HTR-10 in 2000.

2.1 Dragon Plant Description

Key Specifications (from Beck et al. 2010):

- Thermal Power: 21.5 MW
- Power Density: 14 MW/M^3
- Secondary Coolant: Steam (low quality steam and hot water)
- Primary System Pressure: 2 MPa
- Primary Inlet Temperature: 350 C
- Primary Outlet Temperature: 750 C
- Vessel Material: Carbon Steel
- Core Type: Prismatic Blocks
- Years of Operation: 1964-1975

Dragon was located in Winfrith, England. Dragon was the first HTGR, built in the United Kingdom strictly as a fuel and materials test reactor. It was never intended to generate electric power and did not have any power-conversion equipment. With only one vessel, the RPV, no cross-vessel was used in construction (Simon 2005).

The primary circuit at Dragon was the first one to use "cold" helium returning from the heat exchangers to maintain the vessel at a temperature several hundred degrees Celsius lower than the "hot" helium temperature. This was done to limit creep of the carbon steel reactor pressure vessel (Simon and Capp 2002).

The original design of the Dragon fuel was intended to release some volatile fission products into a purge flow of helium gas (Simnad 1991). The purpose of this feature was to intentionally release xenon and krypton to remove neutron poisons from the core. Experience revealed that, even at temperatures up to 1000 C, the fuel retained most of the xenon and krypton generated. This led to the development of tri-structural-isotropic (TRISO) type coated particle fuel, which releases only 10^{-7} of the fission products generated (Simon and Capp 2002).

Dragon's original fuel load was a highly enriched uranium/thorium fuel, but due to questions about long-term availability of highly enriched uranium in the United Kingdom, this was replaced by low-enriched (3.5%) uranium fuel (Beck et al. 2010).

Dragon rejected heat through six circulator/heat exchanger branch circuits. These were cooled by water at 15.8 bar pressure (lower than the 20 bars primary helium pressure to prevent significant water injection in the event of a cross-vessel duct leak) entering the heat exchanger at 200 C and leaving it with a steam content of 16.4% by weight. This secondary circuit was cooled in turn by a tertiary water circuit pressurized to

13 bars with cold and hot temperatures of 50 C and 187 C, respectively. This rejected heat to the atmosphere via a bank of fin-fan dry coolers (Simon 2005).

2.2 Peach Bottom Plant Description

Key Specifications (from Beck et al. 2010):

- Thermal Power: 115 MW
- Power Density: 8.3 MW/M^3
- Secondary Coolant: Steam (modern fossil steam conditions, no reheat)
- Primary System Pressure: 2.3 MPa
- Primary Inlet Temperature: 327 C
- Primary Outlet Temperature: 700-726 C
- Vessel Material: Carbon Steel
- Core Type: Prismatic Blocks
- Years of Operation: 1966-1974

Peach Bottom Unit 1 was located in Delta, near Harrisburg, Pennsylvania. Peach Bottom Unit 1 was the first HTGR to operate in the United States, and it was first in the world to produce electrical power. It operated successfully for 8 years as a prototype reactor, demonstrating high thermal efficiency power generation and the highly enriched uranium/thorium converter fuel cycle that was subsequently used at FSV (Simnad 1991).

Peach Bottom used a fuel element made of a sleeve of low-permeability graphite with a fuel bearing mid-section in the center, with top and bottom reflectors. The plant featured a purge helium system to remove fission products from the fuel elements. The early fuel designs were expected to release a significant fraction of fission products, the plant featured a purge helium system similar to the one used at Dragon (Simnad 1991).

Because Peach Bottom featured up-flow of helium coolant through the core; decay heat could be removed by natural circulation using the service steam generators. The plant did not have a dedicated decay heat removal system; instead the main circulator was run using a “pony” motor. The plant had a steel containment, which was inserted during operation. Reactivity control was by solid absorber rods with one of three types of actuators: pneumatic accumulators, electric drive, or thermal fuse/gravity inserted (Fisher and Orvis 1981).

One operational problem documented at Peach Bottom was fission product release from the earliest loads of fuel. These were made with a simple fuel particle: the uranium or thorium “kernel” was coated with a single layer of pyrolitic carbon (also known as PyC). This led to replacement fuel being made with a bi-structural isotropic (BISO) fuel particle with an inner buffer layer to accommodate fission product recoil and an outer layer to retain noble gas fission products (Beck et al. 2010). Core 1, the original fuel load, featured the simple single-coated particles and operated from initial criticality until June 1970; Core 2 featured BISO particles and released less fission product inventory than Core 1. Core 1 reached a burn-up of 30,000 MWd/MTU, at which point fuel swelling occurred, leading to cracking of the fuel element sleeves. Core 2 operated from mid-1970 to October 1974 (Simnad 1991).

Another operational problem was oil ingress from the oil-lubricated bearings of the helium compressors (circulators). Near the end of Core 1's life, concerns about oil ingress were raised. Investigation revealed that the demister/filter on the outlet of the compressor had become saturated with oil and that approximately 100 kg of oil entered the reactor. The oil caused carbon deposits on primary circuit metal surfaces and persistent hydrogen and methane impurities in the primary coolant (Beck et al. 2010).

The Peach Bottom plant had an overall capacity factor of 75%, considered high for a prototype plant (Simnad 1991). Exclusive of planned shutdowns for research and development programs, the capacity factor was 88% (Goodjohn 1991). The steam generators were the first nuclear steam generators to use Alloy 800; no leaks or failures of the steam generators were experienced during seven years of plant operation (Simnad 1991). Peach Bottom was the first HTGR plant constructed with a "cross-vessel" connecting the reactor vessel to a steam generator; in fact, it had two such components (Beck et al. 2010).

Peach Bottom had completed its demonstration mission by the end of October 1974. By that time, Core 2 had been expended, and a decision had to be made to apply for a permanent operating license for continued operation. A study conducted by Philadelphia Electric Company (cited in Beck et al. 2010) indicated that the benefits of continued operation beyond the second cycle would not be economically feasible. The plant turned out to be too small to be commercially viable; as a result, it was shut down and decommissioned (Beck et al. 2010).

2.3 AVR Plant Description

Key Specifications (from Beck et al. 2010):

- Thermal Power: 46 MW
- Power Density: 2.6 MW/M^3
- Secondary Coolant: Steam (modern fossil steam conditions, no reheat)
- Primary System Pressure: 1.1 MPa
- Primary Inlet Temperature: 275 C
- Primary Outlet Temperature: 950 C
- Vessel Material: Steel and Concrete Building
- Core Type: Pebble Bed
- Years of Operation: 1967-1988

AVR was located at the Julich Research Center in Germany. AVR was one of the first reactors built in the Federal Republic of Germany. It used a "pebble bed" type core, with the fuel contained in 6 cm diameter graphite spheres. These spheres contain the TRISO and BISO fuel particles in the center 4 cm region of the fueled spheres. The initial core consisted of approximately 30,000 fueled and 70,000 additional non-fueled graphite spheres. During operation, the spheres were circulated out of the reactor, where the fueled spheres were evaluated for burn-up by means of a high-resolution gamma spectrometer. The spheres with high enough burn-up were removed from the reactor; spheres with sufficient remaining fuel were returned (Simnad 1991).

AVR operated successfully for 20 years and reached temperatures of up to 1000 C, the highest of any commercial reactor to date (Simnad 1991). AVR operated with an average availability of 66.4% and generated 1.67 billion KWh of electricity (Goodjohn 1991).

One of the most significant operational occurrences at AVR was a steam generator leak that developed at the beginning of or during a 1978 plant shutdown to repair a safety valve. AVR normally experienced some increase in helium moisture during shutdown; in this case when the reactor was restarted, the moisture levels increased enough to require plant shutdown. Investigation revealed that a 1 to 3 mm^2 leak had formed in one steam generator. Since the steam generators were located above the core, this leak resulted in a significant amount of water in the system. Because AVR did not have a low-point drain to remove bulk water from the vessel, removal of the water and repair of the steam generator required a 15-month shutdown. AVR did not use a cross-vessel type of construction; instead the steam generators were located above the core in a single vessel (Beck et al. 1991).

Toward the end of operation, AVR's availability was increasingly affected by testing programs (although availability was impacted by testing programs from the beginning of operation). Graphite spheres with melt-wires (alloyed to melt at temperatures from 600 to 1280 C) were used in loss of forced cooling tests to show that the pebble bed fuel remained below temperatures that could cause fuel failure (Marnet et al. 1991).

2.4 FSV Plant Description

Key Specifications (from Beck et al. 2010):

- Thermal Power: 842 MW
- Power Density: 6.3 MW/M^3
- Secondary Coolant: Steam
- Primary System Pressure: 4.8 MPa
- Primary Inlet Temperature: 404 C
- Primary Outlet Temperature: 777 C
- Vessel Material: Prestressed Concrete Reactor Vessel (PCRV)
- Core Type: Prismatic Block
- Years of Operation: 1976-1989

FSV was located in Pletteville, Colorado. FSV, the only commercial HTGR to operate in the United States, was built under the Atomic Energy Commission Reactor Demonstration Program (Brey 1991). Unlike pressurized water reactor and boiling water reactor plants, FSV generated steam at typical modern fossil plant single reheat temperatures and pressures: 2400 PSI main steam at 1000 F, with a single reheat to 1000 F, with six stages of regenerative feed-water heating (Benham et al. 1973). The main turbine was essentially a regular single-reheat fossil turbine with the exception that steam flowed to the helium circulator drive turbines before going to the reheat section of the steam generators (Benham et al. 1973).

The construction of the plant featured the entire primary circuit inside the PCRV, including the reactor core, four steam driven helium circulators, and twelve once-through steam generators (two loops of six each) with reheat. No cross-vessel type structures were used in the FSV design (Benham et al. 1973). As such, the reactor vessel at FSV was completely different in construction and in terms of the codes and standards it met for nuclear use when compared to U.S. LWR plants and to the Peach Bottom Unit 1 prototype, which all have or had steel reactor vessels.

The overall layout of the reactor core was similar to the proposed design of the GT-MHR. The FSV core featured 37 individually regulated flow regions with adjustable orifices at the inlet of each region. These were adjusted by control room operators to equalize outlet temperatures in the regions (Brey 1991).

FSV was seldom run at 100% rated power for several reasons. Initial rise to power testing revealed core fluctuations above about 70%, apparently due to combined thermal/hydraulic effects, which continued for the next 2.5 years. The plant required its first refueling before full-power operation could be achieved. The fluctuations were stopped by installing constraining devices connecting the upper fuel elements of adjacent refueling regions of the core in a separate shutdown in late 1979. Following this, FSV operated at rated power for the first time on November 6, 1981. From 1981 to 1983, the plant operated at up to 100% power, with major shutdowns for the second refueling and a modification of the circulator auxiliary systems. The plant shut down at the end of 1983 for the third (and as it turned out, final) refueling (Brey 1991).

Regulatory changes associated with Appendix R (to 10 CFR Part 50) Fire Protection imposed power level limits on FSV. After a major shutdown was completed in 1987 (to make modifications for environmental qualification of electrical components to comply with Subpart 49 of 10 CFR 50), NRC limited maximum allowable reactor power to 35%. This limitation stemmed from concerns about heat removal capability using safe shutdown cooling per the plant Appendix R Fire Protection program. These concerns were based on initial reassessments of the ability to remove decay heat by flooding either the reheat module or economizer-evaporator-superheat module of the main steam system. For the case of the reheat module, the fire water system could only supply enough water volume to flood one module; a single module would not provide enough heat removal capacity to deal with 100% decay heat. FSV management decided to remove this method of heat removal from the final safety analysis report rather than modify the plant to provide adequate heat removal capability. For the economizer-evaporator-superheat module, it was determined that steam binding could occur above 83% of maximum decay heat; this led to an administrative limit on reactor power that continued to the end of plant operation (Brey 1991).

FSV was plagued by moisture intrusion and limited ability to remove moisture once it got into the primary system. Moisture in the helium coolant was a problem for three main reasons:

- It could attack the graphite fuel assemblies and more importantly the permanent core support blocks, which were made from PGX graphite, which exhibited a higher oxidation rate than the H-327 and H-451 graphite used in the fuel elements (FSV final safety analysis report)
- It could cause corrosion of carbon steel components, particularly the control rod drive mechanisms

- Significant moisture in the helium coolant added positive reactivity. For example, during a moisture intrusion event on June 22, 1984, after control room operators reduced reactor power manually to 30%, moisture ingress raised power to 40%, which contributed to a scram on high vessel pressure at 0029 hours on the June 23. Six control rod pairs failed to insert on this scram. The cause was eventually traced to corrosion of the control rod drive mechanisms (Brey 1991)

Moisture ingress from the steam generators and reheaters was less problematic than moisture from the circulator auxiliary systems. A leak occurring in late 1977 in one reheater tube was identified and plugged after a manual reactor scram. The identification of the leak and its plugging were carried out without significant exposure of personnel to radiation (Beck 2010).

There were several important sources of moisture intrusion into the primary system at FSV. Most important were the liner cooling system and the circulators' water-lubricated bearings. The ability of the helium purification system to remove moisture once it got into the system was limited, in part because it was designed to remove small volumes of moisture (NUREG/CR-6839).

Part of the corrective actions associated with LER 50-267/84-008 (NUREG/CR-6839) was a test of the reserve shutdown system; this test was performed in November 1984 and revealed that several hoppers failed to discharge the reserve shutdown balls. This indicated that the reserve shutdown control system, consisting of hoppers with graphite rupture diaphragms and borated graphite balls, was also subject to degradation by moisture intrusion. Low-level moisture intrusion over an extended period of time had the potential to compromise both the control rods and the reserve shutdown system (NUREG/CR-6839).

On Friday, October 2, 1987, at 2359 hours a fire was identified in the turbine building; the fire was apparently caused by hydraulic oil leaking from a hydraulically controlled valve (HV-2292; Loop II Main Steam Bypass Valve) onto hot steam piping. The fire damaged several pieces of equipment and many electrical cables. Cable damage led to loss of power to the control room radiation monitor, causing an automatic transfer of ventilation to a recirculation mode. This in turn led to some smoke entering the control room and to a trip of one reheat steam radiation monitor, leading to a shutdown of Loop I. Since Loop II had been shut down manually due to the hydraulic valve failure, this caused a loss of forced cooling at 0008 hours on October 3. By 0020 hours, the reheat steam radiation monitors had been bypassed and helium circulation and Loop I secondary flow were restored. This fire is unusual because a steam pipe ignited it; LWR reactors do not operate at high enough temperatures to ignite most hydraulic oils. A more typical effect was that fire-induced cable damage caused unexpected control system responses generally similar to the Brown's Ferry Unit 1 fire. The final root cause determination found that a restrictor orifice was left out of a thermal relief for hydraulic valve HV-2292. This caused the relief to open on the normal surge of pressure when the valve was actuated, leading to an increased rate of hydraulic oil discharge and prevented reseating of the valve. The higher-than-normal volumetric rate of hydraulic oil discharge overwhelmed the hydraulic oil containment system for the valve, resulting in the oil spilling onto the hot surfaces of the valve and igniting (LER 50-267/87-023).

FSV suffered from operational problems as noted above and by the mid 1980s was confronted with rising maintenance, operations, and fuel costs. On December 5, 1988 the Board of Directors of the Public Service Company of Colorado decided to shut the plant down on or before June 30, 1990 (Brey 1991). In late August 1989, during a shutdown to repair a faulty control rod drive, a steam leak was discovered in the main steam ring header, and further inspections revealed more cracking. Faced with the considerable expense required to repair the ring header, the company decided to permanently shut FSV down (as a nuclear power plant) on August 29, 1989 (Goodjohn 1991).

FSV generated over 4.8 billion KWh of electricity over its operating life (Goodjohn 1991).

2.5 THTR Plant Description

Key Specifications (from Beck et al. 2010):

- Thermal Power: 750 MW
- Power Density: 6 MW/M^3
- Secondary Coolant: Steam (modern fossil steam conditions, with reheat)
- Primary System Pressure: 4 MPa
- Primary Inlet Temperature: 404 C
- Primary Outlet Temperature: 777 C
- Vessel Material: PCRV with Liner
- Core Type: Pebble Bed
- Years of Operation: 1985-1991

The Thorium High Temperature Reactor (THTR) was built by an industrial consortium near the city of Hamm Uentrop, in the German state of North Rhine Westphalia. The commissioning program was completed in May 1987, and the plant was turned over to the operating utility on June 1, 1987 (Goodjohn 1991).

Initial criticality was on September 13, 1983, the generator was synchronized to the grid for the first time on November 16, 1985, and full power reached on September 23, 1986 (Baumer and Kalinowski 1991).

In general, the technical performance of THTR was good partly due to the high reliability of the electric drive circulators. These six circulators, 2300 KWe each, never required a reactor shutdown. The PCRV type construction of THTR did not use any cross-vessel type structure, although it did have a metallic hot duct internal to the PCRV (Beck et al. 2010).

The hot duct of THTR was made of Alloy 617, which contains cobalt. The THTR experience was that the cobalt was not incorporated into an oxide scale, so the cobalt did not enter the hot gas circuit (Beck et al. 2010).

The THTR shut down for planned maintenance in October 1988, and maintenance inspections found that 35 out of about 2600 head hold-down bolts were defective. Technical evaluations of these findings indicated that the plant could continue operating safely, but the plant was not restarted by the utility pending renegotiation of the risk-sharing contract between the members of the consortium. Political considerations of risk-sharing among the Federal, state, and industrial (members of the consortium) stakeholders resulted in a decision not to restart the plant (Goodjohn 1991). Specific issues contributing to the decision not to restart the plant were the increase of financial operating losses to be borne by the utility from 10% for the first 3 years of operation to 30% thereafter; increases in the estimated cost of eventual decommissioning; the original fuel manufacturer ceasing to manufacture the fuel pebbles; failure to secure a permanent spent fuel repository agreement; and uncertainties about issuance of a “permanent” operating license after the initial provisional license expired after the first 1100 full-power operating days (Baumer and Kalinowsk 1991).

2.6 HTTR Plant Description

Key Specifications (from Beck et al. 2010):

- Thermal Power: 30 MW
- Power Density: 2.5 MW/M^3
- Secondary Coolant: He/Pressurized Water
- Primary System Pressure: 4 MPa
- Primary Inlet Temperature: 395 C
- Primary Outlet Temperature: 850-950 C
- Vessel Material: 2-1/4Cr-1Mo Steel
- Core Type: Prismatic Blocks
- Years of Operation: Startup in 1998, still operating as of date of report (2015)

HTTR is located on the Japanese Atomic Energy Agency campus in the Ibaraki Prefecture, Tokaimura, Japan. HTTR is a prismatic fuel type HTGR reactor featuring a steel RPV connected to heat exchangers that cool the outlet helium (there is no power conversion equipment at this strictly-research reactor) by means of a secondary helium loop. This transfers heat to pressurized water heat exchangers, and these in turn reject heat to the atmosphere. The RPV is connected to the heat-exchanger vessel by a cross-vessel, which in English-language literature refers to as a cross-duct (Shiozawa et al. 2004).

HTTR’s website shows that the overall layout of the reactor vessel is similar to the proposed design of the GT-MHR. The RPV is connected to secondary heat exchange vessels by a hot duct inside a cold duct pipe. As shown in Figure 3 below, the bulk of the plant is built below grade.

The plant achieved initial criticality in November 1998 and reached rated power of 30 MW for the first time in December 2001 (Beck et al. 2010).

2-1/4Cr-1Mo Steel is used for the HTTR vessel because it normally operates at about 400 C. This alloy has higher creep rupture strength at elevated temperatures than the usual Mn-Mo steels used for light water RPVs. The RPV is built up from three types of components: forgings meeting Japan Industrial Standard (JIS) SFVAF22B, which is equivalent to ASTM A336/A336M – 10a (2010); plates meeting JIS SCMV4-2, equivalent to ATSM A387 / A387M – 11 (2011); and seamless pipes meeting JIS STPA24, equivalent to ASTM A355 – 89 (2006). The standards JIS SFVAF22B, JIS SCMV4-2, and JIS STPA24 are found in Tachibana (1997).

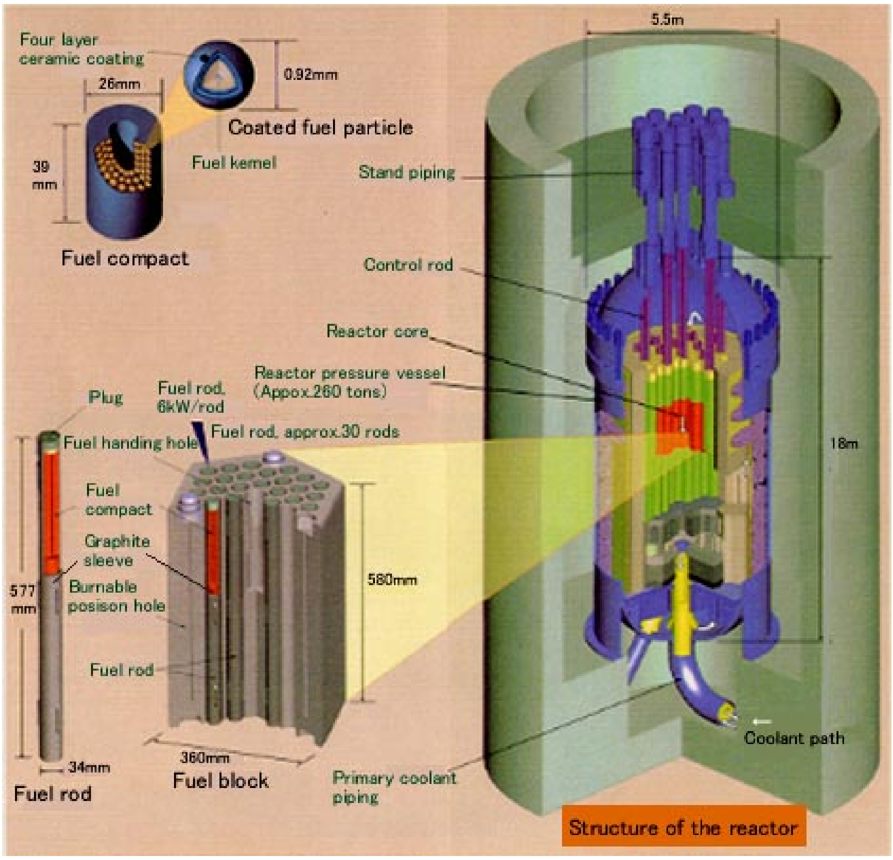

Figure 2. HTTR Fuel and Reactor Construction Details (http://www.jaea.go.jp/jaeri/english/ff/ff43/randd01.html)

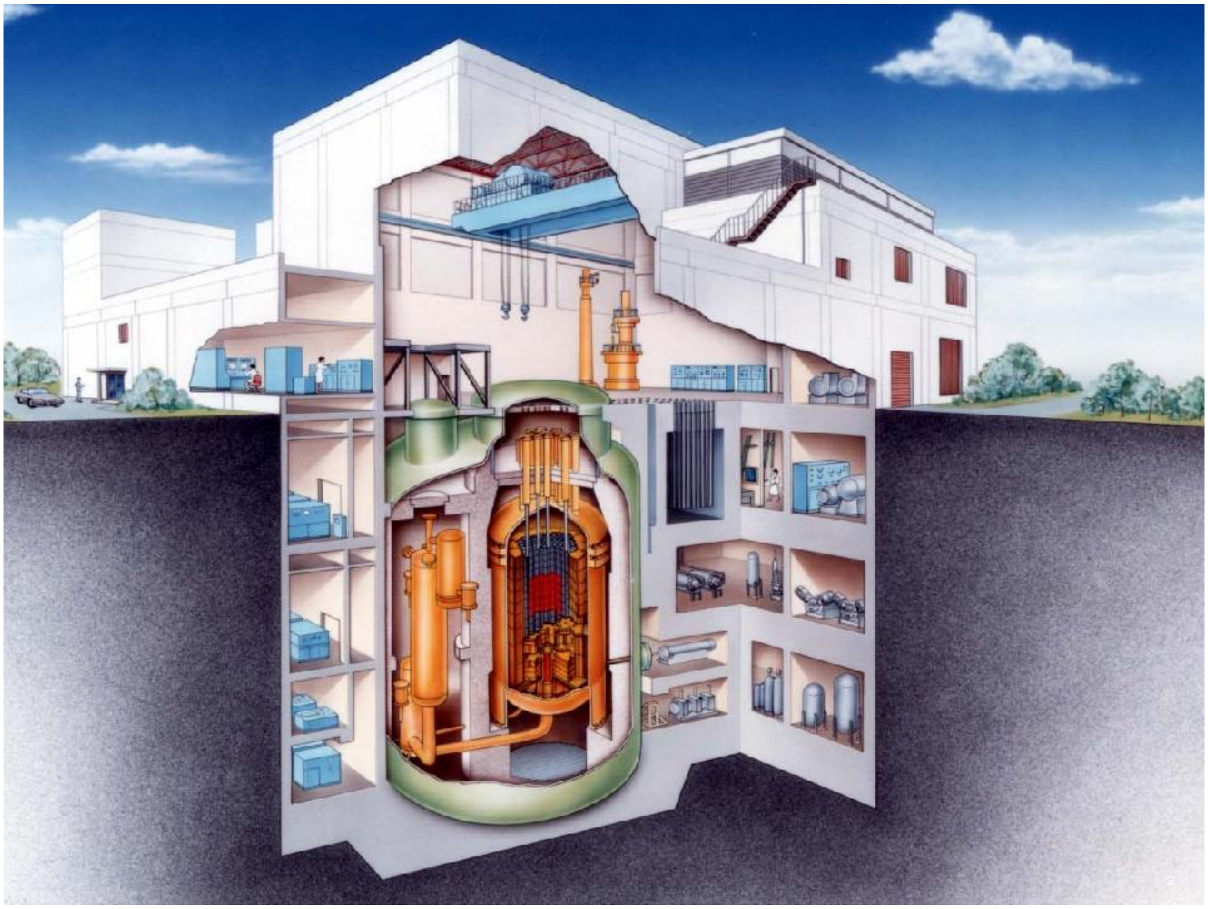

Figure 3. HTTR Plant Layout (http://httr.jaea.go.jp/eng/index.html)

2.7 HTR-10 Plant Description

Key Specifications (from Beck et al. 2010):

- Thermal Power: 10 MW
- Power Density: 2 MW/M^3
- Secondary Coolant: Steam
- Primary System Pressure: 3 MPa
- Primary Inlet Temperature: 250 C
- Primary Outlet Temperature: 700 C
- Vessel Material: C-Mn-Si Steel
- Core Type: Pebble Bed
- Years of Operation: Start-up in 2000, still operating as of date of report (2015)

HTR-10 is a helium-cooled, pebble-bed reactor, located at Tsinghua University in Beijing, China. HTR-10 is very similar in construction to the Siemens HTR design (Wu et al. 2002). The overall layout of a reactor vessel, a power conversion vessel, and a “cross-vessel” with a hot gas duct inside the cooler gas duct, is essentially the same design that has been used on several steel-vessel HTGR plants starting with Peach Bottom Unit 1. Some sources state that China does not have a specific licensing basis for HTGR reactors (Beck et al. 2010).

The overall layout of the plant is also similar to the proposed design of the GT-MHR. The three-vessel type design, with a hot duct inside a cross-vessel connecting the reactor and power conversion vessels, was first seen on Peach Bottom Unit 1 (which operated from 1966 to 1974). HTR-10 personnel refer to the connection between the reactor and power conversion vessels as a “vessel”; however, as noted below, they performed an LBB analysis on it as if it were considered a pipe.

A paper presenting an LBB analysis for HTR-10, which should be largely applicable to GT-MHR and PBMR plants, was presented at the Second International Topical Meeting on High Temperature Reactor Technology in Beijing, September 2004 (Shiozawa et al. 2004). This topic is discussed in more detail in Section 3 of this report.

The HTR-10 served as an initial concept to drive the development of a larger PBMR, namely the High Temperature Reactor-Pebble-bed Modules (HTR-PM). The first two 250 MWth HTR-PM will be installed at the Shidaowan plant, near the city of Rongcheng in Shandong Province. The two modules will drive a steam turbine generating approximately 200 MWe. The HTR-PM is expected to begin commercial operation in 2017. A comparison of the plant parameters for the HTR-PM and HTR-10 is presented in Table 1; while Figure 4 shows details of the HTR-PM reactor and building. The specific Chinese codes and standards that govern the HTR-10 or HTR-PM RPV were not found in the literature when this report was written.

Table 1 – Comparison of Plant Parameters for HTR-PM and HTR-10

	HTR-PM	**HTR-10**
Electrical Power (MWe)	211	2.5
Core Thermal Power (MWth)	250	10
Core Diameter (m)	3	1.8
Core Height (m)	11	1.97
Core Outlet Temperature (°C)	750	700
Core Inlet Temperature (°C)	250	250
Fuel Enrichment (%)	8.5	17
Steam Pressure (Mpa)	13.25	3.5
Steam Temperature (C)	567	435

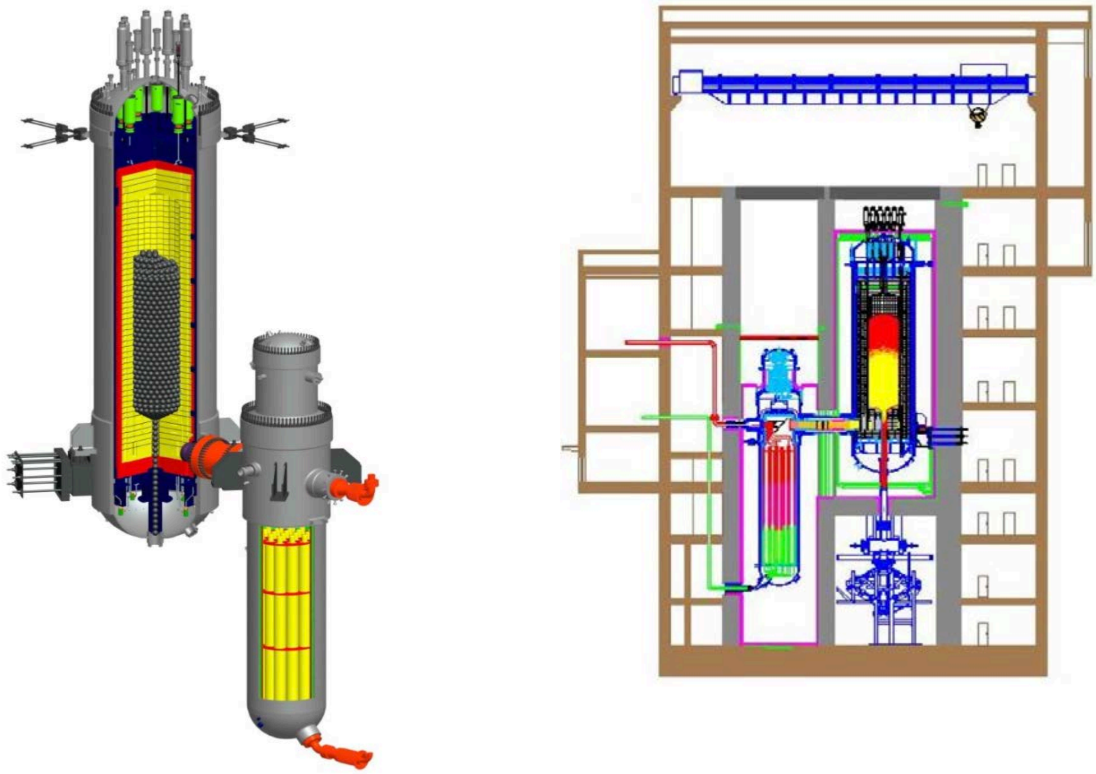

Figure 4. HTR-PM Reactor and Steam Generator (Left) – Reactor Building (Right)

2.8 Non-Nuclear Operating Experience Applicable to HTGR

HTGR RPV conditions, in terms of pressure and temperature, are within the experience base for petrochemical pressure vessels (maximum normal temperature of 500 C (930 F) or less and upset temperatures of 565 C (1050 F) or less). Petrochemical vessels in this type of service have useful working lives of 50 years or more, comparable to nuclear plant RPVs. Furnace tubes, operating at 950 C (1740 F) to 1150 C (2100 F), typically have a service life of 5 to 10 years. An ASME document (Sims 2012) includes a table summarizing the alloys used, vessel thicknesses, service conditions, and damage mechanisms experienced.

2.9 Conclusion

Considering the history of the seven HTGR plants reviewed in this report, only two plants—FSV and THTR—were built with thermal power greater than 150 MW. Additionally, these two plants are reasonably regarded as the only two plants that had any expectation of commercial viability. The third-most powerful plant, Peach Bottom, was shut down and decommissioned at the end of its life as a prototype primarily because utility analysis indicated that it was too small to be commercially viable. (It ran reliably, however and the authors found no documented technical reasons that it could not have continued to run many more years had the utility chosen to do so.)

Of these two plants, FSV did not operate reliably enough to generate a profit for Public Service of Colorado—or even to cover its own expenses. FSV also had a significant financial disadvantage in that the fuel assemblies used were unique to that plant and were supplied only by General Atomics. Even if the plant had operated reliably the cost of fuel would have been higher per MWe than the LWR plants in the United States. The large HTGR plants designs that General Atomics sold in the early 1970s were to use the same fuel as FSV and would have provided greater fuel manufacturing volume, which could have provided some economies of scale, but these plants were cancelled.

THTR operated reliably and had few technical issues. As the trial period of operation came to a close, with its reduced liability on the utility for decommissioning and other costs, a political decision was made not to apply for a permanent operating license but to decommission under the more favorable terms of the existing license.

The experience with these two plants may be viewed quite unfavorably by financial professionals who will be involved in any new HTGR plant construction. To date, all HTGR projects have been financially unsuccessful, including the two larger-than-prototype plants that were intended to be profitable.

Experience with inherent safety and radiological issues has been much more positive. FSV typically had personnel exposure in terms of man-rem far below contemporary U.S. LWR experience, even in years when refueling and/or circulator maintenance were performed. For example in 1985, the year of the major control rod drive refurbishment, FSV was responsible for 35 person-rem of exposure. During that year pressurized water reactor plants averaged 425 and boiling water reactor plants averaged 800 person-rem. For all the other years (from 1980 to 1988), FSV never exceeded three person-rem of personnel exposure (Brey 1991). In terms of operational safety, a definite conclusion is harder to reach because the number of reactors and reactor years of operation is small, but the greater thermal mass of the graphite moderator and much higher margin from normal operating temperature to threshold fuel failure temperature support the idea that HTGR plants feature improved inherent safety characteristics compared to LWR plants. Experiments conducted at AVR confirmed reactor shutdown due to thermal feedback even without scram and without fuel failure (Marnet et al. 1991).

Operational reliability was good for the small, simple plants like Peach Bottom and AVR but not good for FSV (Goodjohn 1991). THTR exhibited fair reliability for the brief operating life it had: 61% in 1987 and 52% in 1988 (Baumer and Kalinowski 1991). The limited information available indicates that both HTTR and HTR-10 have had good availability since they were put into operation.

Non-nuclear experience in the petroleum refining industry indicates that vessels made from materials similar to those being proposed for HTGR RPV construction have useful service lives of 50 years or more when vessel temperature is maintained at or below 500 C (930 F), but metallic components routinely exposed to temperatures near 1000 C (1830F) have useful working lives of only about 5 to 10 years (Sims 2010).

Three of the seven plants reviewed, Peach Bottom in the United States, HTTR in Japan, and HTR-10 in China have a hot gas duct inside a cooler cross-vessel that connections the RPV to the power conversion vessel, similar to the GT-MHR design. Peach Bottom had two “cross-vessels” connecting the reactor vessel to the two steam generators. The HTTR refers to the “duct” connecting the RPV and the secondary heat exchange vessels. The HTR-10 refers to the connection between the reactor and the power conversion system as a vessel; however their LBB analysis was performed as if it is a pipe. The HTTR and HTR-10 literature reviewed appears to have been translated from Japanese and Chinese respectively. In the author's opinion the difference in terminology may be an artifact of translation. Based on this review, the authors have formed the opinion that the component connecting the RPV and power conversion vessel has not been consistently defined in operating plants.

3.0 Discussion of LBB

3.1 Leak-Before-Break Introduction and Requirements

The primary purpose of performing a leak-before-break (LBB) analysis is to demonstrate that a leak can be detected prior to a piping break. If the LBB requirements cannot be satisfied then a piping break must be postulated and appropriate protection against the dynamic effects of the break must be provided for the affected safety-related structures, systems and components. The LBB analyses can provide the plant design basis to justify eliminating the dynamic effects of pipe rupture. GDC 4 of Appendix A to 10 CFR 50 allows for the exclusion of dynamic effects associated with pipe rupture from the design basis, when analyses demonstrate that the probability of pipe rupture is extremely low for the applied loads resulting from normal conditions, anticipated transients and the safe shutdown earthquake (SSE). Typically only high energy, ASME Code Class 1 or 2 piping or the equivalent should have LBB methodology applied, however, other high energy piping can be considered. LBB analyses allow for eliminating the use of pipe whip restraints, jet impingement barriers, sub-compartment pressurization and other protective measures against the dynamic effects of pipe rupture.

Only the dynamic effects of pipe ruptures can be eliminated when using a LBB methodology. The non-mechanistic pipe rupture requirements for containment design, emergency core cooling system performance and environmental qualification (certain exceptions permitted for EQ) remain unchanged. The LBB methodology cannot be applied to piping that can fail in service due to degradation mechanisms such as water hammer, creep, erosion, stress corrosion cracking, fatigue, thermal aging and brittle cleavage-type conditions ensuring crack growth is governed by known design and loading conditions rather than uncertain degradation.

Following review of the degradation mechanisms the piping undergoes a rigorous stress and fracture mechanics analysis to show crack stability and any resulting leakage can be detected by the plants leak detection systems should a flaw develop.

3.2 Leak-Before-Break Evaluation Methodology

Satisfying the LBB criteria ensures that if a crack grows to a through-wall leak, the leak will be detected at load conditions that are well below those that would cause pipe rupture. The LBB analyses, using conditions consistent with the design basis, as-built configurations and sufficient margins to failure, will demonstrate that the probability of pipe rupture is extremely low. The following approach is used to ensure successful application of the LBB methodology:

- The plant leak detection systems must meet the requirements of Regulatory Guide 1.45 and are sufficiently reliable, redundant, diverse and sensitive

- The size of the leakage crack postulated should be large enough that leakage during normal operation is 10 times greater than the minimum leak rate threshold that the leak detection system can sense

- Pre-service inspection of all welds is performed to ensure no pre-existing weld defects are present that could accelerate crack growth rates

- Materials of adequate toughness and tensile properties are used in the specifications. Plant specific material data with lower bound compatible sets of material tensile and toughness properties are to be used

- For each pipe size the locations with the least favorable combination of stress and material properties are analyzed. The analysis uses design basis loads and the as-built piping configurations

- The analysis must verify that the maximum vibrational stresses are well within the applicable fatigue endurance limits

- Using a fracture mechanics stability analysis or a limit load analysis, the normal plus safe shutdown seismic loads are used to determine a critical crack size for a postulated through-wall crack. The critical crack size is compared to the size of a leakage crack for which detection is certain. If the crack size is significantly smaller than the critical crack size with a margin of 2.0 then the LBB requirement is satisfied

The goal of an LBB analysis is to provide assurance of an extremely low probability of piping rupture

3.3 Leak-Before-Break Application to High Temperature Reactors

GDC 4 is a broad scope rule that can be applied to different reactor types including high temperature reactors with appropriate adjustment and accommodation for the different design basis including but not limited to considerations for material types, helium coolant, leak detection systems, operating and accident loading conditions and as-built configurations. USNRC Regulatory Guide 1.45 recommends, but does not require, the use of at least three different methods to detect system leakage (IAEA 2008). This document applies to current LWR plants and recommends leak detection methods such as measurement of sump levels, humidity, and airborne radioactivity levels, which are not applicable to HTGR plants. In a General Atomics report from December 1980, the opinion is expressed that this regulatory guide is applicable to HTGR reactors; monitoring of reactor coolant pressure, containment pressure, temperature, or radioactivity; and PCRV penetration seal system flow for leak detection (Lewis 1980).

Section III, Division 5 of the ASME Code, which provides rules for the construction of high temperature reactors, does not include or specifically address leak-before-break methodologies. ASME Code Section III, Division 1 rules use different terminology than Division 5; Class A in Division 5 is similar to Class 1 and Class B is similar to Class 2. Section III, Division 5 states, “Reflecting the design approach of high temperature reactors, Class A rules address those items deemed to be “safety-related,” and the Class B rules address those items deemed to be “nonsafety related with special treatment.” These safety classifications reflect the risk-based approach derived from safety criteria established for high temperature reactor plants.” Leak-before-break methodology was developed separate from the ASME Code to address the elimination of the dynamic effects of cross-vessel duct rupture from the design basis.

Zhengming and Shuyuan of Tsinghua University presented an LBB analysis at the Second International Topical Meeting on High Temperature Reactor Technology in Beijing, held in September 2004 (Zhengming and Shuyan 2004). The HTR-10 reactor features a cross-vessel type design, similar to Peach

Bottom and to the proposed GT-MHR. The cross-vessel was selected as the most likely part of the primary system to fail for the purpose of the analysis, which was done in accordance with NRC recommended methods. (The methods are described by NUREG/CR-4572, which is referenced by Zhenming and Shuyan 2004). Zhenming and Shuyan concluded that detectable coolant leakage would occur from a postulated circumferential through-wall crack well before it propagated far enough to cause sudden failure. The details of how leakage would be detected, beyond greater than normal makeup flow required to maintain main coolant pressure, and the overall cost-benefit, were not discussed in this paper (Zhengming and Shuyan 2004).

4.0 Materials, Codes, and Standards

Several candidate materials are under consideration for most of the structures, systems, and components at modular HTGR plants. At the time of this writing, more attention is being paid to SA-508/SA-533 and to modified 9Cr-1Mo steels for the RPV, power conversion vessel, and cross-vessel. For pebble bed modular reactor plants, as of 2009 the design called for SA-508 Grade 3 steel, with a normal operating temperature of 300 C (572 F), with design basis events bringing the vessel to as high as 430 C (806 F). The GT-MHR design calls for modified 9Cr-1Mo ASME Grade 91 steel, with normal temperature of 400 to 460 C (752 to 860 F) and with temperature excursions as high as 570 C (1058 F).

Review and Assessment of Codes and Procedures for HTGR Components (NUREG/CR-6816) and *Materials Behavior in HTGR Environments* (NUREG/CR-6824) were written in 2003. NUREG/CR-6816 reviews and evaluates currently available national and international codes and procedures for use in the design of high-temperature gas-cooled reactors (HTGRs). NUREG/CR-6824 reviews and evaluates data currently available on materials performance and long-term behavior in High-Temperature Gas-Cooled Reactors (HTGRs).

A detailed study of potential materials for HTGR applications is extensively covered in the NGNP High Temperature Materials White Paper (INL 2010). In particular, HTGR materials will be selected using a comprehensive approach that will consider component functional and performance requirements, safety classification, and code and regulatory compliance. A detailed description of the classification process is presented. The main material issues for HTGR components design include material application, operating environment, choice of material technology (metallic, composite…), material qualification and codification, e.g., ASME B&PV Code Section III Division 5, as well as material performance.

The regulatory base for existing materials is primarily based on the experience on LWR technology; for example 10 CFR 50.55a requires reactor coolant pressure boundary components to be designed, fabricated, erected, and tested in accordance with the requirements of Class 1 components in Section III of the ASME B&PV Code, or Class A components per the 2013 release of Section III Division 5 (ASME BPVC-III-NB 2013). The main goal for material selection is to ensure that the materials are commercially available and a near-term codification path is in place. The HTGR technology issues that drive material selection and qualification are passive heat removal, slow accident progression, no core meltdown conditions, and fission product retention capability provided by the coated particle fuel. The codification and regulatory acceptance is mainly driven by the identification of failure mechanisms, with potentially the development of a PIRT analysis, performance limits, as well as design and safety margins.

The NRC has published NUREG/CR-6944, where a PIRT technique was used to identify safety-relevant/safety-significant phenomena and assess the importance and related knowledge base of high-temperature structural materials issues for the Next Generation Nuclear Plant (NGNP), a very high temperature gas-cooled reactor (VHTR).

The NGNP high-temperature-materials PIRT maps the key phenomena relevant for normal operations, anticipated transients, and postulated accidents (design basis and beyond). All structural materials other than the graphite to be used in the core and core support structures were addressed in the PIRT analysis.

The results of the PIRT evaluation can be used as a tool for identifying and prioritizing research needs and material selection for HTGRs.

The classes of high temperature materials can be defined as follows

- Ferritic steels T < 450 C
- Ferritic/martensitic steels T < 650 C
- Austenitic stainless steels T < 800 C
- Nickel-based superalloys T < 1050 C
- Inter-metallics T < 1250 C
- Refractory Alloys T < 1400 C
- Ceramics / Composites T < 1600 C

Currently, the candidate materials for the main components of the primary loop of the HTGR, such as the RPV, cross-vessel, steam generator and reactor internals include

- SA-508/SA-533
- Alloy 800H
- Grade 91 steel
- Alloy XR

Other candidate materials such as Alloy X were actively investigated for earlier programs, but are not currently under investigation for VHTR applications. Two paths can be followed for material qualification. The ASME B&PV Code bases one path on codification or equivalent, where the 2013 release of the Code has significantly progressed in this direction. A secondary path involves the ad-hoc qualification by testing and analysis for specific HTGR applications.

In 2013, ASME issued a new Division 5 of Section III of the Boiler and Pressure Vessel Code for components exceeding the temperatures in ASME BPVC Section III, Division 1, Subsection NH. Specifically, ferritic materials up to 370 C (700F), austenitic/high nickel materials up to 425 C (800F), and graphite based materials are addressed in this revision of the code.

4.1 Review of the 2013 ASME Boiler and Pressure Vessel Code as Applicable to HTGRs

This section contains a review of the most recent version of the ASME B&PV Section III of the Code. ASME formed a "Working Group on Nuclear High Temperature Gas–Cooled Reactors" within the framework of the B&PV Committee on Construction of Nuclear Facility Components (Section III). The charter of the Working Group is given as follows:

The Working Group shall develop rules for the construction of Nuclear High Temperature Gas-Cooled Reactors (HTGR) within Section III Division 5 Part 1. The rules of Part 1 shall constitute the requirements for the construction of the nuclear HTGR facility components such as pressure vessels, piping, pressure retaining portions of rotating equipment including pumps, blowers, turbines and compressors, valves, heat exchangers and for core support structures, both metallic and nonmetallic, and for containment or confinement structures. The rules shall contain requirements for materials, design, fabrication, testing, examination, inspection, certification, and the preparation of reports. The Working Group shall identify research and development efforts required to support the technical development of these code rules. Coordination with BPV XI on in-service inspection (ISI) issues shall be maintained.

At the time of this writing, a revision from 2013 exists for the Code for Division 1 Subsection NH "Class 1 Components in Elevated Temperature Service", as well as for Division 5 "High Temperature Reactors". These two sections are directly applicable to HTGR design and construction. An important document in guiding their mission is the Roadmap for the Development of ASME Code Rules for High Temperature Gas Reactors that was developed by an ASME Project Team for HTGR Code Development. The effort to develop this roadmap was originally sponsored by the NRC.

Additional groups have been organized within the ASME Board on Nuclear Codes and Standards infrastructure that support the development needs of HTGRs in areas relevant to materials. These support groups are the following

- Subgroup on High Temperature Reactors
- Subgroup on Elevated Temperature Design
- Subgroup on Graphite Core Components
- Special Working Group, High Temperature Gas Cooled Reactors

4.2 2013 ASME BP&V Section III Division 5 – High Temperature Reactors

This section contains an overview of Section III Division 5, which represents the most up-to-date development within the ASME BP&V code for the design and construction of HTGRs. Division 5 is broken down into the following subsections

- Division 5 — High Temperature Reactors
  - Subsection HA — General Requirements
    - Subpart A — Metallic Materials
    - Subpart B — Graphite Materials
    - Subpart C — Composite Materials
  - Subsection HB — Class A Metallic Pressure Boundary Components
    - Subpart A — Low Temperature Service
    - Subpart B — Elevated Temperature Service
  - Subsection HC — Class B Metallic Pressure Boundary Components
    - Subpart A — Low Temperature Service
    - Subpart B — Elevated Temperature Service
  - Subsection HF — Class A and B Metallic Supports
    - Subpart A — Low Temperature Service
  - Subsection HG — Class A Metallic Core Support Structures
    - Subpart A — Low Temperature Service
    - Subpart B — Elevated Temperature Service
  - Subsection HH — Class A Nonmetallic Core Support Structures
    - Subpart A — Graphite Materials
    - Subpart B — Composite Materials

4.2.1 Metallic Materials

Subsection HA, Subpart A contains the general requirements associated with metallic components used in the construction of high temperature gascooled reactor and liquid metal reactor systems and their supporting systems. The rules of this Division for metallic materials provide requirements for new construction and include consideration of mechanical and thermal stresses due to cyclic operation and high temperature creep. They do not cover deterioration that may occur in service as a result of radiation effects, corrosion, erosion, thermal embrittlement, or instability of the material. The changes in properties of materials subjected to neutron radiation may be checked periodically by means of material surveillance programs. This last requirement has been common practice in the nuclear industry with many reactor-years of experience in this area.

SA-508/SA-533 steel has been used for LWR RPVs in the United States and abroad, and more than 40 years of experience has been accumulated using this alloy in the nuclear industry. The weldability, toughness, and thermal aging properties of this material have been improved over the years by limiting the allowable carbon content and by reducing the levels of sulfur and phosphorus impurities. This alloy is ASME Code Section III approved for Class 1 nuclear components. Subsection NB rules are applicable up to 371 C (700 F) (Simon 2010).

Modified 9Cr-1Mo steel is being considered for GT-MHR vessel applications (Beck et al. 2010) and is under consideration for core support structures in the NGNP HTGR (http://www.ngnpalliance.org/index.php/htgr).

This alloy is ASME BPVC-III approved for Class 1 nuclear components; ASME BPVC-III-NB rules are applicable up to 371C (700 F) and ASME BPVC-III-NH rules are applicable to 649 C (1200F). For the NGNP application, temperatures are not expected to exceed 350 C (662 F). The ASME Code covers this alloy in terms of high-temperature strength, creep, and stress rupture but does not cover emissivity, corrosion, thermal aging, and irradiation effects. These latter characteristics will require additional qualification before this material can be used in HTGR plants (Simon 2010).

A similar alloy, 2.25Cr-1Mo steel, was used to build the RPV for HTTR in Japan (Beck et al. 2010). It was also used for the economizer-evaporator sections of the FSV once-through steam generators (below the bimetallic weld with Alloy 800 superheaters). This alloy is under consideration for similar use in the NGNP steam generators and for possible use in the construction of the steam generator vessel. The allowable stress for 2.25Cr-1Mo is similar to modified 9Cr-1Mo up to about 430 C, (806 F) but its strength drops off significantly above this temperature when compared to modified 9Cr-1Mo (Simon 2010).

Alloy 800H is one candidate for the hot duct (within the cross vessel) and other metallic components in contact with helium exiting the core. High temperature strength and creep/stress rupture resistance for this alloy are covered by ASME Code Case N-201-5 (cited in Simon 2010) for core support structures, and ASME BPVC-III-NH covers Class 1 components. Up to about 427C (800F), there are no significant time dependent effects on allowable stress for component life up to 300,000 hours. Since some non-structural components could see temperatures of 800 C (1472 F) or higher, allowable stress limits will need to be extended. Studies sponsored by ASME Standards and Technology indicate that current information can support an extending maximum use temperature to 850 C (1562 F) and maximum lifetime to 500,000 hours (Simon 2010).

Alloy X and alloy XR are being considered for core and core support structures that can see temperatures above 750 C (1382 F) during normal operation. Alloy X is a nickel-chromium-iron-molybdenum alloy that has favorable high temperature strength, corrosion resistance, and ease of fabrication. Alloy X is the older version, and is covered by ASME codes, but contains enough cobalt to raise activation concerns. Alloy XR is a proprietary version developed by the Japan Atomic Energy Agency for HTTR, with lower limits on cobalt. For nuclear applications of Alloy X, ASME Section II Part D (Properties) lists a maximum usage temperature of 427 C (800 F) (ASME Section III rules). For non-nuclear applications (ASME Section VIII rules), Section II Part D lists a maximum usage temperature of 899 C (1650 F) with correspondingly much lower allowable stress values (Simon 2010).

Type 316 H stainless steel is being considered for core barrel assemblies and other metallic reactor internal components that would experience temperatures up to 620 C (1148 F). This alloy is a cost-saving substitute for alloy 800 and alloy X/XR where steam-side corrosion potential can be controlled (stress corrosion cracking). Type 316 H stainless steel is covered by ASME BPVC-III-NB (Class 1 components and core support structures) up to 427 C (800 F). Type 316 and similar type 304 stainless have been commonly used for LWR reactor internal components. ASME BPVC-III-NH and Code Case N-201-5, extensions to Subsections NB and NG respectively, (cited in Simon 2010) extend temperatures up to 816 C (1500 F) (Simon 2010).

Alloy 617 has superior strength and creep resistance compared to alloy 800 or alloy X/XR above 800 C (1472 F), and is often used in aircraft type gas turbines. Alloy 617 has 10 to 15% cobalt content by weight, making activation a concern. This alloy was studied for HTGR applications in the United States and Germany during the 1970s and 1980s. It has been studied extensively by Huntington Alloys, Oak Ridge National Laboratory, and General Electric, and the Huntington data have been used to develop ASME Code applications in Section I, Section VIII Division 1, and a Draft Code Case for Section III of the ASME BPVC. Alloy 617 is not qualified for use in ASME Code Section III, and work on the Draft Code Case was terminated when very high temperature reactor programs were terminated. (Simon 2010)

CLASSIFICATION OF COMPONENTS AND SUPPORTS

ASME Code Section III, Division 5 provides design rules for Class A “safety-related” and Class B “non-safety related” components and support structures. These safety classifications reflect the risk-based approach derived from safety criteria established for high temperature reactor plants. Remaining items not in these two classifications listed above in Section 4.2 shall satisfy the requirements of other appropriate non-nuclear codes and standards.

4.2.2 Graphite Materials

Subsection HH, Subpart A constitute requirements for the design, construction, examination, and testing of Graphite Core Components and Graphite Core Assemblies used within the reactor pressure vessels of nuclear power plants. The requirements for the selection and qualification of graphite materials are given in Article HHA-2000. This includes material identification and certification, material properties for design, and material property deterioration during service (due to fast neutron irradiation and oxidation).

The requirements for the design of Graphite Core Components and Graphite Core Assemblies are given in Article HHA-3000. For the design of Graphite Core Components, the within-billet and billet-to-billet variability in material properties shall be taken into account. Due account shall also be taken of the effects of fast neutron irradiation, irradiation temperature, and oxidation on the appropriate mechanical and thermal properties, and on dimensional change behavior, as well as the design and service loadings. Both probabilistic and deterministic design methodologies are acceptable and therefore covered. In addition, for the design of the Graphite Core Assembly, account shall be taken of the fast neutron irradiation induced changes in component geometries, which could significantly affect its stability and geometry. These in turn could significantly affect the coolant flow paths, the freedom of movement of fuel and control devices, and the interaction with interfacing metallic components or structures.

(a) The rules shall apply to Graphite Core Components utilized in a high temperature, graphite moderated, gas-cooled fission reactor. Graphite Core Components include fuel blocks, reflector blocks, shielding blocks, and any keys or dowels used to interconnect them.

(b) The rules shall also apply to the arrangement of Graphite Core Components that form the Graphite Core Assembly.

(c) The rules shall not apply to fuel compacts, bushings, bearings, seals, blanket materials, instrumentation, or components internal to the reactor other than those defined above.

The applicable standards to graphite materials are listed in Table 2.

Table 2 – Applicable Standards for Graphite Materials

Standard ID	Published Title	Referenced Edition
	The American Society of Mechanical Engineers (ASME)	
ASME NQA-1a	Quality Assurance Requirements for Nuclear Facility Applications	2008, 2009a
ASME QAI-1	Qualification for Authorized Inspection	Latest
	American Society for Testing and Materials (ASTM)	
ASTM C625	Standard Practice for Reporting Irradiation Results on Graphite	2005
ASTM C781	Standard Practice for Testing Graphite and Boronated Graphite Materials for High-Temperature Gas-Cooled Nuclear Reactor Components	2008
ASTM D7219	Standard Specification for Isotropic and Near Isotropic Nuclear Graphites	2008
ASTM D7301	Standard Specification Nuclear Graphite Suitable for Components Subjected to Low Neutron Irradiation Dose	2008

4.3 Classification of Components, Nameplates, Stampings, and Reports

The rules for certificates, nameplates, the Certification Mark, and Data Reports for metallic components, metallic supports, and metallic core support structures under Division 5 shall be the same as that established for Division 1 metallic components and metallic core support structures. The only change in Division 5 is the terminology used to address components, e.g., Class A and Class B rather than Class 1 and Class 2 as specified for Division 1.

ANS-53.1-2011, *Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants* defines the process for specifying criteria to assure that modular helium-cooled reactor plants are designed to be constructed and operated safely without undue risk to public health and safety. This purpose is achieved through the identification of applicable safety requirements from the national nuclear regulator, industrial codes and standards, and other published guidance and professional engineering practices.

ANS-53.1-2011 provides a process for establishing top-level safety criteria; safety functions; top-level design criteria; licensing basis events; design basis accidents; safety classification of systems, structures, and components; safety analyses; defense-in-depth; and special treatment requirements.

ASME BPVC-III-NCA, Article 1210 (“Components”) requires each component of a nuclear power plant to have data reports and stamping as required in ASME BPVC-III-NCA Article 8000.

5.0 Foreign Codes and Standards

5.1 French Code: Design and Construction Rules for Mechanical Components of FBR Nuclear Islands (RCC-MR)

The French high-temperature code RCC-MR is an extension of the low-temperature code RCC-M. Both codes have rules that are similar to the ASME Code rules. However, the RCC- MR rules are organized according to the damages that are possible at high temperature, which is somewhat different from how the ASME Code is organized. RCC-MR distinguishes between two broad types of possible damages, P type and S type. The P type damages result from the application of a steadily increasing load or constant load. The S type damages occur due to repeated application of loading. The P type damages include immediate excessive deformation, immediate plastic instability, time-dependent excessive deformation, time-dependent plastic instability, time-dependent fracture, and elastic or elastoplastic instability. The S type damages include progressive deformation and fatigue or progressive cracking. Most of the design rules contained in RCC-MR are very similar to those in the ASME Code. As in the ASME Code, both elastic analysis rules and elastoplastic analysis rules are provided. Therefore, we will concentrate our discussions mainly in those areas where there are differences.

As in the ASME Code, RCC-MR contains criteria for Service Load Levels A, C, and D, but Service Load Level B is absent in RCC-MR The classification of stresses into primary and secondary, and into membrane, bending, and peak is identical to the ASME Code. To handle multi-axial stresses, RCC-MR allows the use of either the maximum shear theory (Tresca) or octahedral shear theory to compute stress intensities or stress range intensities.

The primary membrane and membrane-plus-bending stress allowable at low and high temperatures in RCC-MR are basically the same as those in the ASME Code, Section 111, Subsection NH. However, the rules are cast in terms of creep and creep rupture usage fractions.

5.2 Evaluation of RCC-MR Code

RCC-MR was developed in France as a high-temperature extension to RCC-M, for the French breeder reactor program. The basic rules in RCC-MR are very similar to those in ASME Code Subsection NH. RCC-MR provides more detailed instructions on how to carry out fatigue and creep-fatigue design analysis than the ASME Codes. It also uses a somewhat different approach to analyzing creep ratcheting without the use of isochronous stress-strain curves.

5.3 British Procedure: Procedure R5 Assessment Procedure for the High Temperature Response of Structures

Procedure R5 involves a comprehensive assessment of the high-temperature response of structures. The procedure addresses the following aspects of high-temperature behavior

- Simplified methods of stress analysis
- Creep-fatigue crack initiation
- Creep crack growth
- Creep-fatigue crack growth
- Behavior of dissimilar metal welds
- Behavior of similar metal welds

5.4 Evaluation of the R5 Procedure

Creep cracking is more of an issue in residual life assessment than in the design phase. Neither the ASME Code nor the RCC-MR Code addresses the subject at all. Only R5, which is a guideline, not a code, considers creep cracking explicitly. Therefore, the use of this procedure in the design of HTGR components is limited. However, this procedure presents the use of ductility exhaustion as an alternative to the life fraction rule for calculating the creep component of damage. The use of ductility exhaustion for estimating creep damage had been proposed as an element of the rules for a possible extension of ASME VIII, Division 2, to elevated temperatures.

6.0 Cross-Vessel vs. Cross-Duct Issue

Review of the current edition of the 2010 ASME Section III, Rules for Construction of Nuclear Facility Components (2011 Addenda), provides the following rationale for the belief that the ASME would not define the cross-duct a vessel or a pipe, but rather leave it to the owner to provide the design criteria (i.e., vessel or pipe) within the Design Specification.

The forward in ASME BPVC-III-NCA, states:

> The Code Committee does not rule on whether a component shall or shall not be constructed to the provisions of the Code. The Scope of each Section has been established to identify the components and parameters considered by the Committee in formulating the Code rules.

Figure 5 shows a typical arrangement for the hot-duct and cross-vessel connections.

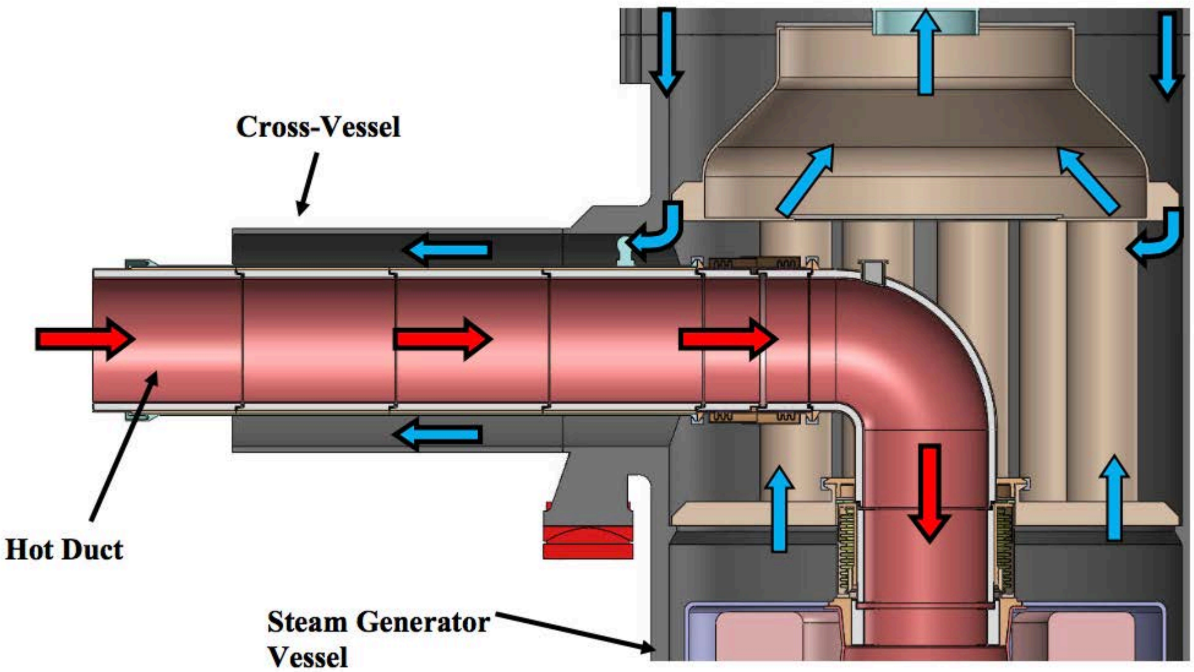

Figure 5. Typical Hot Gas Duct Arrangement within Cross-Vessel

The cross-duct would be defined as a component by ASME BPVC-III-NCA, Article 9000 ("Glossary"), i.e., "a vessel, concrete containment, pump, pressure relief valve, line valve, storage tank, piping system, or core support structure that is designed, constructed, and stamped in accordance with the rules of this Section."

ASME BPVC-III-NCA, NCA-1210 ("Components"), requires each component of a nuclear power plant to have a Design Specification (ASME BPVC-III-NCA, Subsubarticle 3250), a Design Report (ASME

BPVC-III-NCA, Subsubarticles 3350 and 3550), and other design documents specified in ASME BPVC-III-NCA, Article 3000.

ASME BPVC-III-NCA, NCA-3251 ("Provision and Correlation"), states it is the responsibility of the owner to provide, or cause to be provided, Design Specifications for components, supports, and appurtenances. ASME BPVC-III-NCA, NCA 3252(a) ("Contents of Design Specifications"), states that the design specifications must contain sufficient detail to provide a complete basis for Division 1 construction.

In the mid-1980s, the NRC began to accept the concept of LBB for large-diameter, high-quality piping systems as a means of enhancing the safety of nuclear power plants. In *Roadmap to Develop ASME Code Rules for the Construction of High Temperature Gas Cooled Reactors (HTGRS)*, ASME suggests that LBB criteria be included in ASME BPVC-III-5-2011 (Sims 2012).

An approach based on design-by-analysis should be employed for the cross-vessel connection. This statement is based on the geometric characteristics of the cross-vessel connection, relatively short and straight. Subsections NB-3200 and NH-3200 for Class 1 components provide guidance on this aspect.

7.0 Overpressure Protection

ASME BPVC-III-NCA, NCA-3271, Responsibility and Content, requires the owner to provide, or cause to be provided, an overpressure protection report for each component or system (ASME BPVC-III-NB, NB-7200; ASME BPVC-III-NC, NC-7200; or ASME BPVC-III-NE, NE-7200).

ASME BPVC-III-NH, NH-3137.2, Design Consideration for Overpressure Protection of the System, requires each component and the system into which it will be installed to be protected against overpressure events as required by the rules on overpressure protection of Class 1 components and systems exposed to elevated temperature service. It is assumed that ASME BPVC-III-5-2011 will require overpressure protection requirements similar to ASME BPVC-III-NH, NH-3137.2.

8.0 Conclusion

This document reviews the operating history of past and present HTGR plants, assesses the currently available literature related to codes and standards applicable to HTGR plants and evaluates the proposed designs of RPV and associated piping for future plants.

The operational histories of five decommissioned and two currently operating HTGR plants were reviewed, leading the authors to conclude that while small, simple prototype HTGR plants operated reliably, while some of the larger plants (particularly FSV) had poor availability. Safety and radiological performance of these plants has been considerably better than LWR plants. The full potential of HTGR plants for electrical generation has not yet been demonstrated in practice. Pebble-bed and prismatic block HTGR reactor cores have both been built, and both designs appear to be practical. Of the seven reactors, three were built with cross-vessels (Peach Bottom, HTTR, and HTR-10).

Petroleum refineries plants provide some applicable experience with high-temperature materials similar to those proposed for HTGR piping and vessels.

At least one currently operating plant—HTR-10—has performed and documented an LBB analysis that appears to be applicable to proposed future U.S. HTGR designs. This analysis was performed in accordance with NUREG/CR-4572 and concluded that the HTR-10 cross-vessel would exhibit detectable leakage before any through-wall crack grew to critical size.

The question of terminology for the cross-vessel connection between the RPV and power conversion vessel was examined. It is concluded that this connection should be evaluated using the vessel rules, and an approach based on design-by-analysis should be utilized. This approach is identified in Subsections NB-3200 and NH-3200 for Class 1 components.

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