U.S. DESIGN CERTIFICATION

HIGH-TEMPERATURE MATERIALS - METALLICS

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ABSTRACT

This paper describes Pebble Bed Modular Reactor (Pty) Limited's (PBMR's) approach to addressing selected issues pertaining to the design and qualification of metallic components. The scope of this paper covers those metallic structures and components applied in the PBMR reactor and major supporting systems for which Nuclear Regulatory Commission (NRC) regulatory compliance is not fully clarified in regulations or guidance.

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Approved	Senior GM, US Programs	E G Wallace	See signatures on file

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ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BWR	Boiling Water Reactor
CB	Core Barrel
CBA	Core Barrel Assembly
CBCS	Core Barrel Conditioning System
CBSS	Core Barrel Support Structure
CCS	Core Conditioning System
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CFRC	Carbon Fiber Reinforced Carbon
CIP	Core Inlet Pipe
COP	Core Outlet Pipe
CRDM	Control Rod Drive Mechanism
CS	Core Structures
CSC	Core Structure Ceramics
DBA	Design Basis Accident
DBE	Design Basis Event
DCA	Design Certification Application
DGS	Dry Gas Seal
DIN	Deutsches Institut für Normung
DOE	Department of Energy
DPP	Demonstration Power Plant
DSRS	Dry Gas Seal Supply and Recovery System
ECP	Engineering Change Proposal
EOL	End of Life
FHSS	Fuel Handling and Storage System
FMECA	Failure Modes, Effects and Criticality Analysis
FN	Ferrite Number
GCVS	Gas Cycle Valves System
GDC	General Design Criteria
HAZ	Heat Affected Zone
HAZOP	Hazard and Operability (Study)
HGD	Hot Gas Duct
HGDS	Hot Gas Duct System
HICS	Helium Inventory Control System
HMS	Helium Make-up System

Abbreviation or Acronym	Definition
HP	High Pressure
HPB	Helium Pressure Boundary
HPC	High Pressure Compressor
HPS	Helium Purification System
HTGR	High-temperature Gas-cooled Reactor
HTR	High Temperature Reactor
HX	Heat Exchanger
ICS	Inventory Control System
IGSCC	Intergranular Stress Corrosion Cracking
ISI	In-service Inspection
KTA	Kerntechnische Ausschuβ
LBE	Licensing Basis Event
LERF	Large Early Release Frequency
LPC	Low Pressure Compressor
LWR	Light Water Reactor
MeV	Mega-electronvolt
MHTGR	Modular High-temperature Gas-cooled Reactor
MPS	Main Power System
MSS	Main Support System
n/cm^2	neutrons per square centimetre
NDE	Non-destructive Examination
NDT	Non-destructive Testing
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission Report
PBMR	Pebble Bed Modular Reactor
PCU	Power Conversion Unit
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCCS	Reactor Cavity Cooling System
RCS	Reactivity Control System
RCSS	Reactivity Control and Shutdown System
RG	Regulatory Guide
RIM	Reliability and Integrity Management
RPV	Reactor Pressure Vessel
RSS	Reserve Shutdown System
RT_{NDT}	Reference Temperature for Nil Ductility Transition
RU	Reactor Unit
SAS	Small Absorber Sphere
S_m	Time-Independent Allowable Stress Intensity Value
S_{mt}	Lower of Time-Independent and Time-Dependent Allowable Stress Intensity Values

Abbreviation or Acronym	Definition
SR	Side Reflector
SRP	Standard Review Plan
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
S_t	Time-Dependent Allowable Stress Intensity Value
TGS	Turbogenerator Set
VOPS	Vessel Overpressure Protection System

1. INTRODUCTION

This document is one of a series of white papers which, along with associated workshops, comprise the design certification preapplication program for the Pebble Bed Modular Reactor (PBMR) in the U.S. The objective of the Preapplication Program is to reduce the time required for Design Certification through early identification and, where possible, resolution of issues. This is to be done by addressing issues remaining from Exelon's PBMR preapplication work, by identifying additional issues that require preapplication work or inclusion in the Design Certification Application (DCA), and through the identification of further development and testing required for PBMR certification in the U.S.

1.1 PURPOSE AND SCOPE

The purpose of this paper is to describe and to obtain agreement with Pebble Bed Modular Reactor (Pty) Limited's (PBMR's) approach for addressing selected issues pertaining to the design and qualification of metallic components. The scope of this paper covers those metallic structures and components applied in the PBMR reactor and major supporting systems for which the basis for compliance is not fully clarified in U.S. Nuclear Regulatory Commission (NRC) regulations or guidance.

1.2 STATEMENT OF THE ISSUES

PBMR has established a formal process for the selection of materials used in the PBMR design. A key element of the PBMR design philosophy is, wherever possible, to utilize materials that are included in, and operated within the limits of, a code or standard that the NRC has accepted. Where this is not possible, PBMR will use materials within the limits of a code or standard that has been accepted by a recognized standards body, but has not yet been accepted by the NRC. Because of the innovative high-temperature nature of the PBMR, there are a limited number of components for which neither of these two approaches is possible. In such limited cases, the PBMR approach is to select materials on the basis of their suitability for the functions and requirements of the intended application, and to design from first principles with appropriate supporting qualification programs.

1.3 SUMMARY OF PREAPPLICATION OUTCOME OBJECTIVES

The preapplication objectives of this paper are as follows:

- Obtain NRC agreement in principle that the PBMR approach to selecting and qualifying materials for the components discussed in Section 3 provides a reasonable basis for establishing regulatory compliance.
- Obtain NRC agreement in principle that the application of American Society of Mechanical Engineers (ASME) Code Cases N-499-2 and N-201-5 for the Reactor Pressure Vessel (RPV) and Core Barrel Assembly (CBA), respectively, provides a reasonable basis for establishing regulatory compliance.
- Obtain NRC agreement in principle that the PBMR approach for qualifying non-pressure boundary materials at very high temperatures, as described in Section 3 of this paper, provides a reasonable basis for establishing regulatory compliance.

1.4 RELATIONSHIP TO OTHER PREAPPLICATION FOCUS TOPICS/WHITE PAPERS

This white paper on metallic materials is linked to a series of foundational white papers that provide inputs regarding the role of the metallic components in fulfilling the safety objectives of the PBMR.

These include:

- Probabilistic Risk Assessment Approach [1] – This paper identifies the regulatory issues related to the PBMR Probabilistic Risk Assessment (PRA) approach for which NRC feedback is desired during the preapplication review of the PBMR design.
- Licensing Basis Events [2] – This paper outlines the relevant regulatory policy and guidance for the spectrum of Licensing Basis Events (LBEs) to be considered, defines licensing issues associated with LBE definition, and describes the PBMR approach for the selection of the LBEs.
- Safety Classification of Structures, Systems and Components [3] – This paper describes the PBMR risk-informed approach for establishing the safety classification of Structures, Systems and Components (SSC) for the PBMR design. The processes described in this paper will provide the basis for assigning safety classifications to the metallic SSC addressed herein.
- Defense-in-Depth [4] – This paper identifies the regulatory issues related to the PBMR approach to defense-in-depth for which NRC feedback is desired during the preapplication review of the PBMR design. The regulatory foundation for review of the PBMR approach to defense-in-depth is summarized, compliance with the regulatory criteria is described, and specific issues for which feedback is requested are described.

In addition to the above, a separate white paper will be submitted that addresses ceramic materials used in the PBMR design, including graphite and composites.

2. REGULATORY FOUNDATION

2.1 NRC REGULATIONS

Regulations applicable to the design of primary system metallic components in water-cooled reactors are codified primarily in 10 CFR 50.55a, 'Codes and Standards;' General Design Criteria (GDC) 4, 10, 14, 15, 30, and 31, which are contained in Appendix A to 10 CFR Part 50; and Appendix G to 10 CFR Part 50. Other provisions that apply generally to nuclear components, such as the quality assurance requirements of GDC 1 and 10 CFR Part 50 Appendix B, are not discussed here.

Section 50.55a requires, in part, that components of the reactor coolant pressure boundary be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of Section III, 'Nuclear Power Plant Components,' of the ASME Boiler and Pressure Vessel Code or equivalent quality standards.

The GDC establish minimum requirements for water-cooled nuclear power plants similar in design to existing conventional Light Water Reactor (LWR) plants. The GDC were developed specifically for water-cooled designs, and are not requirements for other types of reactors. Nevertheless, as discussed in the introduction to Appendix A, the GDC are considered to be generally applicable to other types of nuclear power units, and are intended to provide guidance in establishing the principal design criteria for such other units.

Water-cooled nuclear power plant GDC, with potential applicability to PBMR primary system materials, include:

- GDC 4, 'Environmental and Dynamic Effects Design Bases' requires that SSC important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.
- GDC 10, 'Reactor Design,' requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of Anticipated Operational Occurrences (AOOs).
- GDC 14, 'Reactor Coolant System Pressure Boundary,' requires that the reactor coolant system pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- GDC 15, ' Reactor Coolant System Design,' requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.
- GDC 30, 'Quality of Reactor Coolant Pressure Boundary,' requires that components that are part of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards.
- GDC 31, 'Fracture Prevention of Reactor Coolant Pressure Boundary,' requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating

fracture is minimized. It also states that the design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Finally, 10 CFR 50.61, plus Appendices G and H, address fracture toughness and associated surveillance requirements for ferritic materials used in the pressure-retaining components of the reactor coolant pressure boundary. These requirements, which pertain specifically to light water nuclear power reactors, are designed to provide adequate margins of safety during normal operation, including AOOs and system hydrostatic tests. These requirements are based on the ASME Code.

Care must be exercised in the interpretation of these LWR requirements for the PBMR, as the PBMR safety design is not based on the prevention and mitigation of loss of coolant accidents per se. Rather, the safety functions for the Helium Pressure Boundary (HPB) are based on the inherent reactor characteristics and passive core heat removal capabilities of the PBMR. The approach to interpreting these requirements is provided in Section 3.

2.2 NRC POLICY STATEMENTS

NRC policy statements do not address the design, testing, or performance of metallic components in High Temperature Reactor (HTR) primary systems.

2.3 NRC GUIDANCE

NRC's guidance related to the design of metallic components in the primary system generally parallels the NRC regulations discussed in Section 2.1. For example, Regulatory Guide 1.84, *Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,* provides guidance on Section III ASME Code Cases oriented to materials, design, fabrication, examination, and testing that are generically acceptable to the NRC staff, sometimes with additional conditions imposed. The Code Cases referenced in this Regulatory Guide generally apply to LWRs and are not necessarily applicable to High-temperature Gas-cooled Reactors (HTGRs).

Regulatory Guide 1.87, *Guidance for Construction of Class 1 Components in Elevated Temperature Reactors* (Revision 1, June 1975), describes five Code Cases that provide guidance for the construction of components subject to elevated temperature service in HTGRs. Regulatory Guide 1.87 states that the service temperatures and load conditions for HTGRs are such that time-dependent phenomena such as creep and relaxation are important. It further states that Subsection NB of Section III of the ASME Code does not provide adequate guidance for construction of components subject to elevated-temperature service. Therefore, as an interim step, the ASME developed five Code Cases. The referenced Code Cases cover design, fabrication, installation, examination, testing, and protection against overpressure for such components. They reflect both time-independent and time-dependent material properties and structural behavior (elastic and inelastic) by considering the following modes of failure:

- ductile rupture from short-term loadings;
- creep rupture from long-term loadings;
- creep-fatigue failure;

- gross distortion due to incremental collapse and ratcheting;
- loss of function due to excessive deformation;
- buckling due to short-term loadings; and
- creep buckling due to long-term loadings.

Regulatory Guide 1.87 also states that component designs should accommodate the required In-service Inspection (ISI) and surveillance programs for material or component integrity. Finally, it states that the materials evaluations should address representative environmental factors such as compatibility with the coolant (e.g., helium, air, contaminants); irradiation effects that might induce ductility loss; and aging resulting from prolonged exposure to elevated temperature.

The Code Cases referenced within Regulatory Guide 1.87 were superseded by ASME Code Cases N-47 through N-51 (with numerous revisions) and, subsequently, by Section III, Subsection NH. To date, Subsection NH has attained acceptance as a basis for regulatory compliance in one specific application. The current version of 10CFR50.55a – b) 1) vi) states:

> '(vi) Subsection NH. The provisions in Subsection NH, 'Class 1 Components in Elevated Temperature Service,' 1995 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section, may only be used for the design and construction of Type 316 stainless steel pressurizer heater sleeves where service conditions do not cause the component to reach temperatures exceeding 900 °F.'

Additionally, the NRC has established detailed guidance on primary system components in NUREG-0800, Standard Review Plan (SRP), including Sections 5.2.1.1, *Compliance with the Codes and Standards Rule*, 10 CFR 50.55a, 5.2.1.2, *Applicable Code Cases*, and 5.2.3, *Reactor Coolant Pressure Boundary Materials*. These sections of the SRP are not directly applicable to primary systems in HTRs. However, SRP Section 5.2.3 indicates that material selection should include an evaluation of issues such as susceptibility of the material in the reactor coolant pressure boundary to cracking and corrosion, fracture toughness, compatibility of the materials with the reactor coolant (including contaminants in the coolant), and compatibility of the materials in the reactor coolant pressure boundary with the materials in the insulation.

In July 2003, the NRC published NUREG/CR-6824, *Materials Behaviour in HTGR Environments*, which addresses the performance of metallic components in high temperature helium-cooled reactors. As noted in Section 2, 'Metallic Components,' of NUREG/CR-6824, the primary helium coolant in the gas-turbine-based HTGRs is expected to be at temperatures in the range of 850 °C to 900 °C, and the selected materials should have adequate performance over long service life at temperatures in the range of 900 °C to 950 °C. NUREG/CR-6824 includes information on HTGR materials properties and environmental effects on the behaviour of metallic components under these expected conditions.

Regulatory Guide 1.147, *In-service Inspection Code Case Acceptability--ASME Section XI*, provides guidance on Section XI ASME Code Cases oriented to ISI that are generally acceptable to the NRC staff. The Code Cases identified in this Regulatory Guide are incorporated by reference within 10 CFR 55a for application to LWRs. While their use for HTGRs is not specifically addressed, many of the permitted examination and repair activities addressed by the Section XI Code Cases could logically be applied to HTGR components.

More recently, NRC has issued Regulatory Guides 1.174 and 1.178, which provide guidance on how to establish risk-informed ISI programs for LWR components and piping systems. These regulatory guides cannot be directly applied to the PBMR, because they are based on the LWR risk metrics of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), which are specific to LWRs and not directly applicable to the PBMR. In recognition of this, ASME has established a working group under ASME Section XI to develop Reliability and Integrity Management (RIM) programs for passive metallic components in the PBMR. These RIM programs involve the setting of component reliability requirements and the selection of an appropriate combination of design strategies, fabrication and construction strategies, leak detection strategies, and Non-destructive Examination (NDE) strategies to deliver the necessary reliability. The reliability requirements are selected to maintain the frequencies and consequences of LBEs within prescribed regulatory limits. ISI programs for the PBMR Demonstration Power Plant (DPP) and the U.S. design certification will be based on the ASME RIM program requirements currently being developed.

2.4 RECENT NRC PRECEDENTS INVOLVING GAS-COOLED REACTORS

In the late 1980s and early 1990s, the NRC conducted a preapplication review of the Modular High-temperature Gas-cooled Reactor (MHTGR) at the request of the Department of Energy (DOE). The results of that review are published in NUREG-1338, *Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor* (NUREG-1338), which was initially issued by the NRC in 1989. NUREG-1338 notes that lower-level criteria for the design and review of the MHTGR primary system, such as those contained in the SRP, do not exist in a form approaching that for LWR primary systems. NUREG-1338 goes on to note that certain LWR criteria on primary systems are helpful and important in guiding the MHTGR conceptual design, but significant gaps remain, particularly those relating to safety issues. In draft NUREG-1338, the NRC also noted that DOE made a commitment to meet the intent of GDC 1 for items classified as Safety Related and to use suitable versions of the ASME Code in order to meet the requirements of Section 50.55a.

In the final draft version of NUREG-1338 issued in December 1995, the NRC noted that ASME had approved Code Case N-499 for reactor vessels on allowable stresses and design rules for elevated temperatures during Service Level C and D events. The NRC stated that the Code Case and frequency of Service Level C and D events must be approved by the NRC, and that the frequency must meet the values provided in Table 1 to SRP 3.9.3.

In 2001 to 2002, the NRC staff conducted a preapplication review of the PBMR at the request of Exelon. In a letter to Exelon dated May 31, 2002, the NRC staff provided feedback on various technical, safety, and policy issues raised by Exelon during preapplication reviews for the PBMR. With regard to material properties of metallic materials of construction at high temperatures, the staff stated that:

> *'A list should be provided of all materials used for the reactor pressure vessel and its appurtenances, core support structures, primary system boundary, connecting piping, and other components important to safety and the applicable material specifications, design stress and time at temperature and other environmental conditions. The identification of the grade or type and conditions of the materials to be placed in service would also be required. If the code approved material specifications for the intended applications are not available, relevant material specifications should be developed following the format of [ASTM] specifications. The subject specifications should be supported by the data and information*

as identified in ASME Code, Section III, Appendix IV, for approval of the new materials. Additional information unique to the application in the PBMR environment and condition shall also be provided.'

3. PBMR APPROACH

This section identifies and summarizes the approach to regulatory compliance for PBMR metallic materials. The basis for regulatory compliance begins with the processes described in [1] through [4]. In particular, the safety classifications assigned to metallic components are related to their functions and associated requirements during LBEs, supplemented by defense-in-depth considerations. LBEs, in turn, are derived from probabilistic assessments. Additional insights result from deterministic assessments undertaken in support of the PBMR DPP Project in South Africa. Based on these inputs, appropriate design criteria, including codes and standards, are selected. The remainder of this section documents the results of this process and highlights specific areas where NRC feedback would be valuable during the design certification preapplication review.

Section 3.1 provides an overall description of the metallic structures and components applied in the PBMR reactor and major supporting systems. It further identifies, by component, materials requirements that are to be addressed in more detail in this white paper. The remaining sections address these specific materials requirements and PBMR's approach to regulatory compliance.

3.1 OVERVIEW OF METALLIC STRUCTURES AND COMPONENTS

This section starts with an overview of PBMR's approach to the selection and assessment of metallic materials in Section 3.1.1. In Section 3.1.2, key metallic components of the PBMR are identified and described, along with the bases for their design and qualification. Section 3.1.2 further identifies those metallic components for which specific feedback is requested regarding PBMR's approach to regulatory compliance. The identified components are addressed individually in subsequent Sections 3.2 through 3.5.

3.1.1 PBMR Approach to the Selection and Qualification of Materials

The PBMR process for materials selection is summarized in Figure 1. The selection process begins with a specification that is based on the functions, requirements, and design of the SSC of interest. Where appropriate, components are grouped with respect to their operating environment, and enveloping conditions are identified for defining the material requirements. The materials selected for the respective group of components are then evaluated against the limitations of:

- Safety classification and selected design code.
- Material qualification status (material property data, relevant in-service experience feedback).
- Applicable in-service damage mechanisms.
- Manufacturing experience.

The resultant evaluation report serves as justification for the material selection for the component, or group of components.

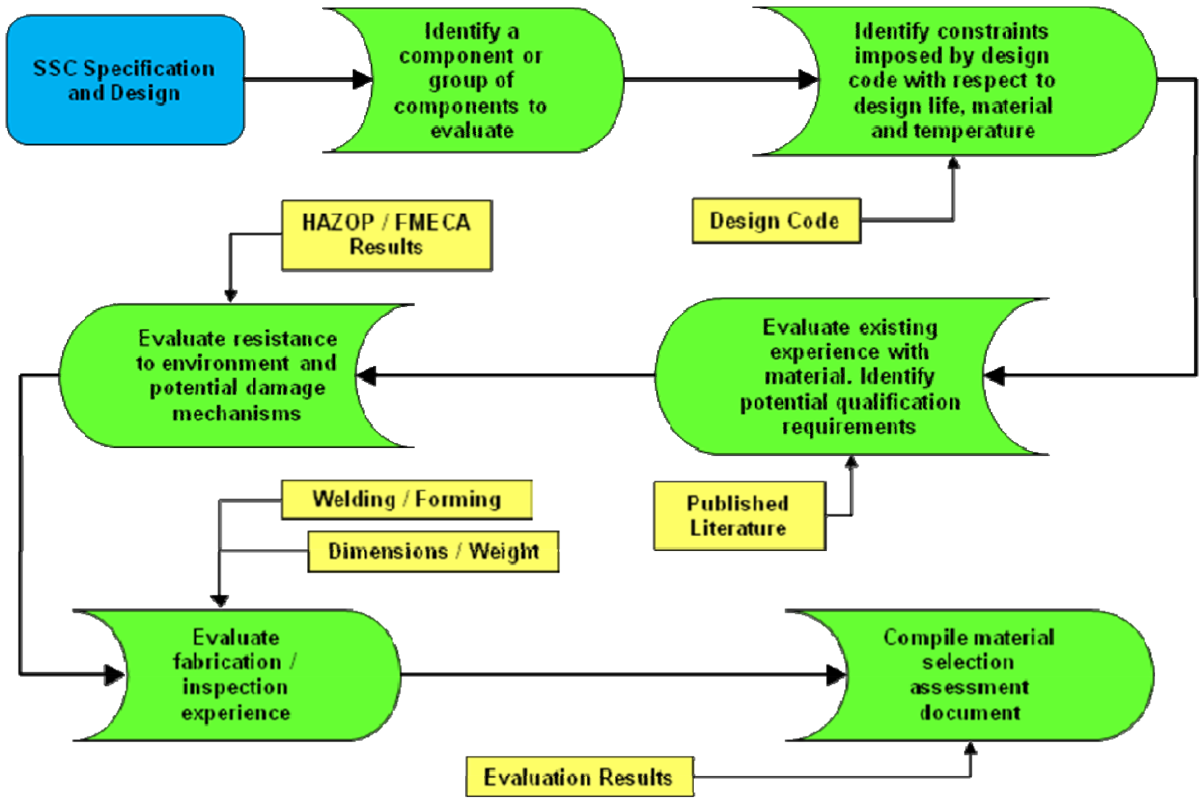

Figure 1: PBMR Materials Selection Process

In both the design and materials selection process, PBMR applies the following general approach, which is summarized in order of preference:

- use materials within the limits of a code or standard that the NRC has accepted; or
- use materials within the limits of a code or standard that has been accepted by a standards body but which the NRC has not yet accepted; or
- use materials that are not incorporated in a code at this time and design from first principles with appropriate supporting qualification programs.

The application of this general approach may be readily seen in the design and materials selections of the PBMR Main Power System (MPS), and particularly in SSC of high safety significance. Specific examples are found in the HPB, where the design allows the use of conventional LWR materials for the vessels and piping, while enabling a reactor outlet gas temperature of 900 °C. Within the HPB, only the RPV requires the application of an ASME Code Case (N-499-2) that has not yet been accepted by the NRC. In this specific instance, the application of the Code Case involves very modest conditions and is only applied for rare Design Basis Events (DBEs). Additional details regarding the design and materials selection for the HPB may be found in Sections 3.1.2.1 and 3.2.

In the few areas in which established materials with adequate codes and standards are not available, existing standards are used to the extent possible. The first example herein is the non-pressure-retaining liner of the Hot Gas Duct (HGD) that channels the flow of helium from the reactor outlet to the turbine inlet. This Alloy 800H component is conservatively designed on a first principles basis. However, maximum advantage is taken of German HTR development, which resulted in a draft standard for that material in the temperature range of interest, and a series of tests of similar components under representative conditions. Additional detail may be found in Section 3.4.

3.1.2 Description

This section provides a summary description of PBMR structures and components, with emphasis on the location and application of metallic components that are evaluated to be of particular significance in the design certification preapplication process. A more detailed technical description of the PBMR may be found in [5]. Specifically, the major areas of focus are those components comprising the primary HPB, the CBA, the Hot Gas Duct System (HGDS), metallic components of the Reactivity Control and Shutdown System (RCSS), the Helium Inventory Control System (HICS), and the Fuel Handling and Storage System (FHSS).

With the exception of the FHSS, these components are all parts of the Main Power System (MPS), which is depicted in Figure 2.

The MPS is a functional grouping of systems and components which consists of the following:

- Reactor Unit (RU)
    - Core
    - Core Structures (CS)
    - Reactivity Control and Shutdown System (RCSS)
- Power Conversion Unit (PCU)
    - Turbogenerator Set (TGS)
    - Recuperators
    - Precooler
    - Intercooler
    - Hot Gas Duct System (HGDS)
    - Gas Cycle Valves System (GCVS)
- HPB System
    - Reactor Pressure Vessel (RPV)
    - Power Conversion Unit (PCU) Component Vessels
    - Helium Pressure Boundary (HPB) Piping
- Main Support Systems (MSS)
    - Core Conditioning System (CCS)
    - Core Barrel Conditioning System (CBCS)
    - Helium Inventory Control System (HICS)

Within the MPS, the RU produces thermal energy and the PCU converts that thermal energy first to mechanical and then to electrical energy. The HPB System encloses the major components of the RU and PCU. The MSS provide for supplemental cooling and conditioning of RU components, plus support for power maneuvering of the PCU via helium inventory control.

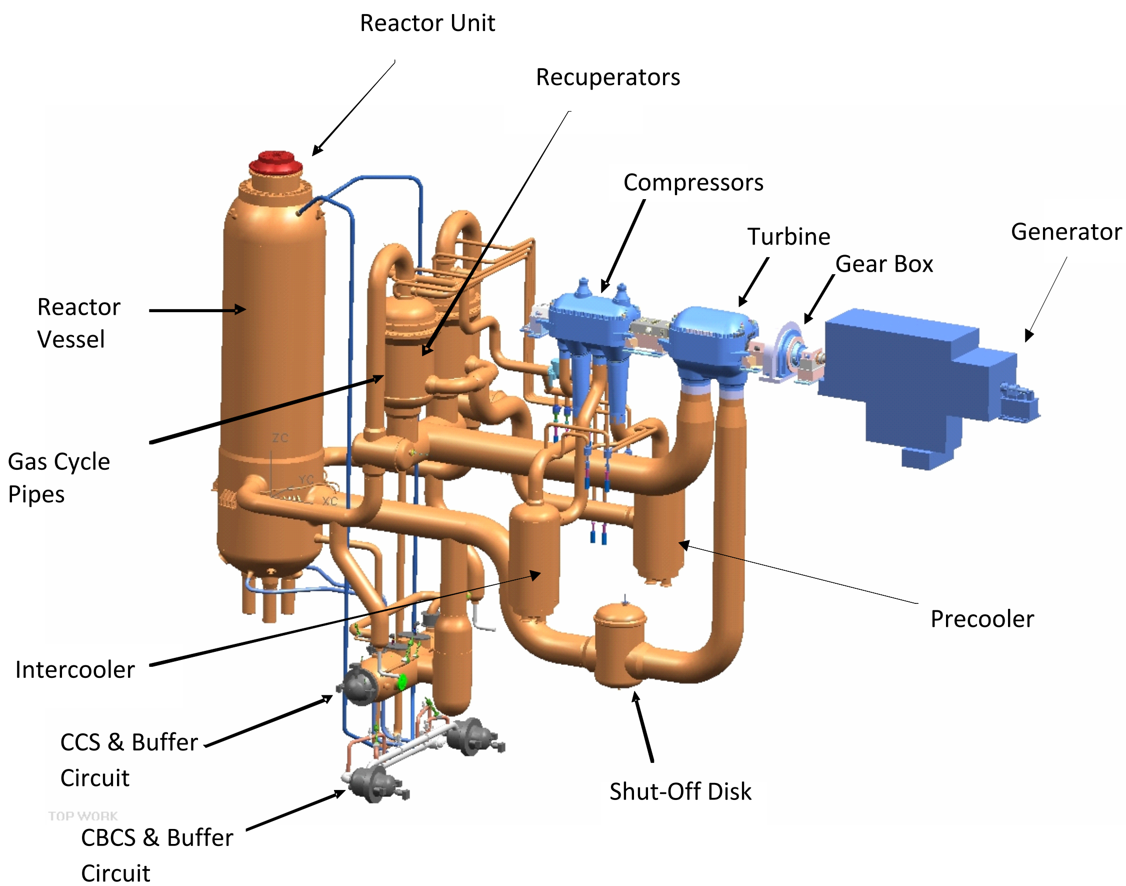

Figure 2: Main Power System Overview

3.1.2.1 Helium pressure boundary

The HPB comprises the vessels and piping of the HPB System (refer to Figure 2), plus the pressure-retaining features of other MPS systems and components. It also includes certain pressure-retaining components of the FHSS and HICS. Within the HPB, the highest pressure and temperature conditions are found (not coincidently) at the High Pressure Compressor (HPC) outlet, where the helium working fluid is compressed to a maximum pressure of 9 MPa at ~110 °C, and at the reactor outlet where the helium, at a somewhat lower pressure, exits at a maximum temperature of 900 °C before being routed to the power turbine. As further described in this section, an important feature of the PBMR design is separation of the high-temperature helium flows from the HPB, both physically and by a pressure gradient that exists under all circumstances in which active coolant flow is present.

The vessels, piping, and other components that collectively form the HPB can be summarized as follows:

a. HPB components enclosing the RU
   - RPV.
   - Vessels enclosing the Reactivity Control System (RCS) Control Rod Drive Mechanism (CRDM) and Drive Motor.
   - Vessels enclosing the Reserve Shutdown System (RSS) (Valve Actuator).

- FHSS isolation valves and FHSS components and piping between the RPV and the isolation valves.

b. HPB components enclosing the PCU
  - Power Turbine outer casing.
  - High- and Low-pressure Compressor (HPC/LPC) outer casings.
  - Intercooler Vessel, plus Intercooler tubes and tube plate.
  - Precooler Vessel, plus Precooler tubes and tube plate.
  - Recuperator Vessels (two).
  - Dry Gas Seals (DGSs) (Turbine and HPC/LPC shaft sections).
  - Vessel Overpressure Protection System (VOPS).
  - Isolating and Control Valve bodies.
  - Connecting piping for instruments.

c. HPB system components enclosing the MSS
  - Vessels enclosing the CCS coolers and blowers, along with their connecting pipework, plus cooler tubes and tube plates (2 x 100%).
  - Vessels enclosing the CBCS coolers and blowers, along with their connecting pipework, plus cooler tubes and tube plates (2 x 100%).

d. HPB piping – high-temperature circuits containing HGDS components

  In these circuits, the outer pipe and internally attached support lugs for the inner pipe are part of the HPB. The annulus between the HPB outer pipe and the inner HGDS pipe (refer to Section 3.1.2.3) contains cool gas flow routed from the HPC outlet:
  - RPV to Turbine Inlet and CCS (main pipe to Turbine with branches to CCS) – Core Outlet Pipe (COP).
  - Turbine outlet to Recuperator.
  - Recuperator and CCS to RPV (two pipes, each with branch from CCS) – Core Inlet Pipe (CIP).

e. HPB piping – low-temperature circuits
  - Recuperator to Precooler.
  - Precooler to LPC.
  - LPC to Intercooler.
  - Intercooler to HPC.
  - HPC to Turbine (for the insulating cool gas flow).
  - Link Line from HPC to RPV (to prevent pressure gradients across the Core Barrel (CB).
  - HPC to HGDS (for the insulating cool gas flow).
  - RPV to CBCS.
  - CBCS to RPV (Upper Lid Closure).
  - CBCS to RPV (Bottom Dome).
  - To the fuel discharge chutes for helium and nitrogen injection.
  - Small Absorber Sphere (SAS) return pipe for removing SASs from the core after their insertion.

f. Connected systems

In addition to the MPS-related HPB components listed above, the following additional systems include HPB components:

- HICS.
- FHSS.
- In-core Delivery System – includes piping into the centre core reflector to provide access for the neutron source and instruments.

The pressure-retaining components of the FHSS and the HICS are elements of the HPB. However, most of these components can be isolated by double isolation valves. The FHSS pipes above the RPV and inside the reactor citadel cannot be isolated. Note that the scope of this paper is limited to the non-isolatable portions of these systems.

Table 1 lists the metallic materials for the various structures/components of the HPB, along with applicable ASME Specifications. Also indicated in the table is the PBMR approach to qualification of these materials for licensing purposes. Note that the RU portions of the HPB (RPV, RCS vessel, and the RSS vessel) utilize SA-508 Grade 3 Class 1 forging material and SA-533 Type B Class 1 plate as the vessel materials. These low-alloy steels are included in ASME Section II, Parts A and D. Their application under the design rules of Section III, Division 1, Subsection NB, Class 1 of the ASME Code, *Rules for Class 1 Components*, is well characterized, understood, and accepted by the NRC for LWR pressure vessels. Bolting materials, where applicable, also fall under Subsection NB. The RPV support structures are designed to ASME Section III, Division 1, Subsection NF. Design and qualification of these structures will rely on the aforementioned ASME Code Sections, except for the RPV during low-probability DBEs, in which heat removal is provided for extended periods by the Reactor Cavity Cooling System (RCCS). During such events, metal temperatures within the RPV are predicted by analysis to exceed 371 °C (the maximum allowed under Section III, Subsection NB) for short periods (< 200 h for unmitigated events). Service conditions during such events fall well within the bounds prescribed for the vessel material under ASME Code Case N-499-2. These events are discussed in greater detail in Section 3.2 (Reactor Pressure Vessel).

The other components of the HPB listed in Table 1 are to be designed to ASME Section III, Subsection NB or NC, using bolting, forging, plate, and valve materials approved in Subsection NB and NC. The design temperature for these components is typically 371 °C; actual operating temperatures are < 150 °C. The design and qualification approach is to use the ASME Section III, Subsection NB and NC specifications, except for support structures, which will be designed to Section III, Subsection NF. These other HPB components, to which accepted specifications apply, will not be discussed further in this white paper.

Table 1: Metallic Materials for PBMR Primary Helium Pressure Boundary Components

Component	Materials	Applicable ASME Design Code (Note 1)	Qualification Approach
RPV	SA-508 Grade 3 Class 1 forgings, SA-533 Type B Class 1 plate, SA-540 Grade B24 bolting	Section III, Subsection NB (to 371 °C) + ASME Code Case N-499-2 (above 371 °C)	Use NRC-accepted ASME Specification + ASME Code Case N-499-2
RPV and PCU Supports	SA-36 and SA-533 Type B Class 2 plate	Section III, Subsection NF	Use NRC-accepted ASME Specification
RCS Vessel	SA-508 Grade 3 Class 1 forgings, SA-533 Type B Class 1 plate, SA-193 B7 bolting	Section III, Subsection NB	Use NRC-accepted ASME Specification
RSS Vessel	SA-508 Grade 3 Class 1 forgings SA-533 Type B Class 1 plate, SA-193 B7 bolting, SA-182-F6NM housing, SA-182-316H housing flange	Section III, Subsection NB	Use NRC-accepted ASME Specification
CCS Vessel	SA-508 Grade 3 Class 2 forgings, SA-533 Type B Class 2 plate, SA-210 Grade A-1 tubes SA-197 B7 bolting,	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
CBCS Vessel	SA-335 Grade P1, Class 2	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
Precooler and Intercooler Vessels and Tubes	SA-508 Grade 3 Class 2 forgings, SA-533 Type B Class 2 plate, SA-193 B7 bolting	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
Recuperator Vessel	SA-533 Type B Class 2 forgings, SA-533 Type B Class 2 plate, SA-193 B7 bolting	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
Turbine Vessel Casing	SA-217 Grade WC9 casting	Section III, Subsection NC (Note 2)	Use NRC-accepted ASME Specification
HP- and LP-Compressor Casings	SA-217 Grade WC9 castings	Section III, Subsection NC	Use NRC-accepted ASME Specification
HPB Pipes Connecting PCU Components	SA-335 Grade P1 Seamless Pipe (to ~600 mm) SA-672 Grade J90 Welded Pipe (made from SA-533 Type B Class 2 plate)	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
VOPS Valves	SA-508 Grade 3 Class 1, SA-533 Type B Class 1, SA-193 B7 bolting	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
FHSS Piping Non-isolatable Components	SA-335 Grade P1	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
HICS Injection and Extraction Lines	SA-335 Grade P1	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification

Notes:

1. The use of ASME Code B31.1 will be considered, where appropriate, for small diameter piping applications.
2. Turbine/Compressor Casing designs are not addressed by ASME Codes and are consequently not stamped.

3.1.2.2 Core barrel assembly

Core Structures (CS) are the structural components within the RU that contain the pebble bed core, and define and maintain the pebble bed geometry. The CS also provide for channeling of helium coolant flow and neutron reflection. The major components of the CS are shown in the layout drawing, Figure 3. The CS consists of two major assemblies – the Core Structure Ceramics (CSC) and the CBA. The CSC, which are addressed in a separate white paper, comprise the graphite and ceramic structures that surround and contain the pebble bed core. The CBA, Figure 4, contains and supports the CSC and the pebble bed core within. The major components of the CBA are the CB, top plate, upper and lower supports, and the upper and lower lateral guides. The materials and design bases of these components are listed in Table 2. The primary safety function of the CS is to maintain core geometry, as needed to support the passive core heat removal and control core heat generation safety functions. Note that two of the components of the CSC, the tie-rod holders of the Top Reflector and the metallic parts of the Lateral Restraint Straps, also involve metallic materials. These are also included in Table 2.

The material selected for the CB sides (cylindrical section of the CB) is SA-240 Type 316H (plate) austenitic stainless steel. The materials selected for the CB top plate, base plate, lateral guides and support rings, and support structure are SA-965 Grade F316H (forging) austenitic stainless steel and/or SA-240 Type 316H plate, as appropriate. These same materials and design standards will be applied for the CSC metallic components (tie-rod holders, metallic strap links and pins). 2.25Cr-1Mo ferritic steel (SA-965 Grade F22 forgings and SA-387 Grade 22 Class 1 plate) will be used as intermediate or transition sections between the RPV and the CB where those occur, so as to reduce the effects of differential thermal expansion.

The design and qualification of the 316 stainless steel CBA components are based on ASME Section III, Division 1, Subsection NG, which is approved for the design and manufacture of core support structure components at design temperatures up to 427 °C (for austenitic materials). However, this design temperature limit is projected to be exceeded in those parts of the CBA that are above the level of the Core Barrel Support Structure (CBSS), even during normal operation. On the basis of present estimates, the design temperature for the 316 stainless steel CBA components has been set at 520 °C.

ASME Code Case N-201-5 provides for the use of Type 316H in-core support structure applications at temperatures up to 816 °C. The time-temperature relationships specified in the Code Case will accommodate the necessary full lifetime of the PBMR CBA. The use of ASME Code Case N-201-5 for the CBA is discussed further in Section 3.3 (Core Barrel Assembly). The interface components to be made from ferritic steel will also be designed to Section III, Division 1, Subsection NG. However, these components will operate within normal Subsection NG temperature restrictions, and are thus not further addressed in this white paper.

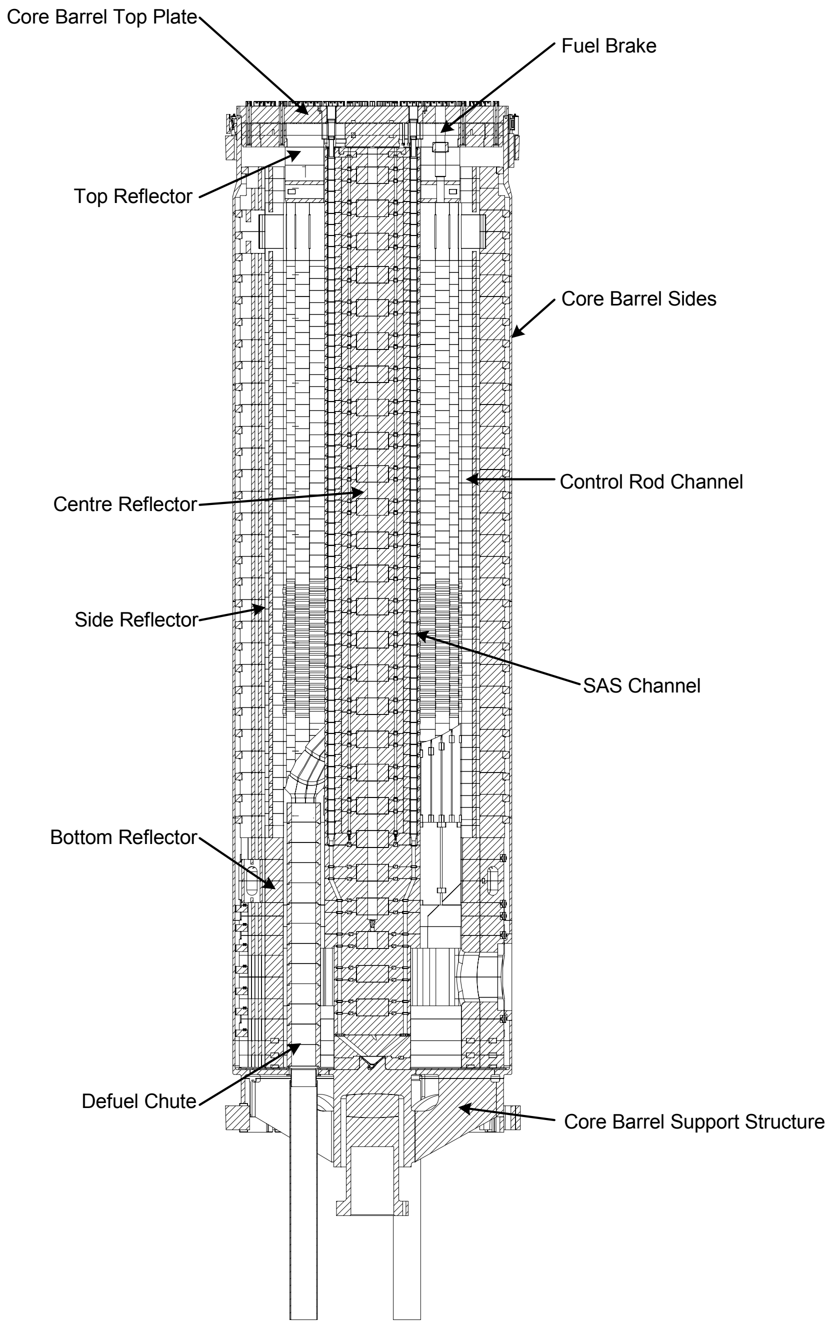

Figure 3: Core Structures General Arrangement

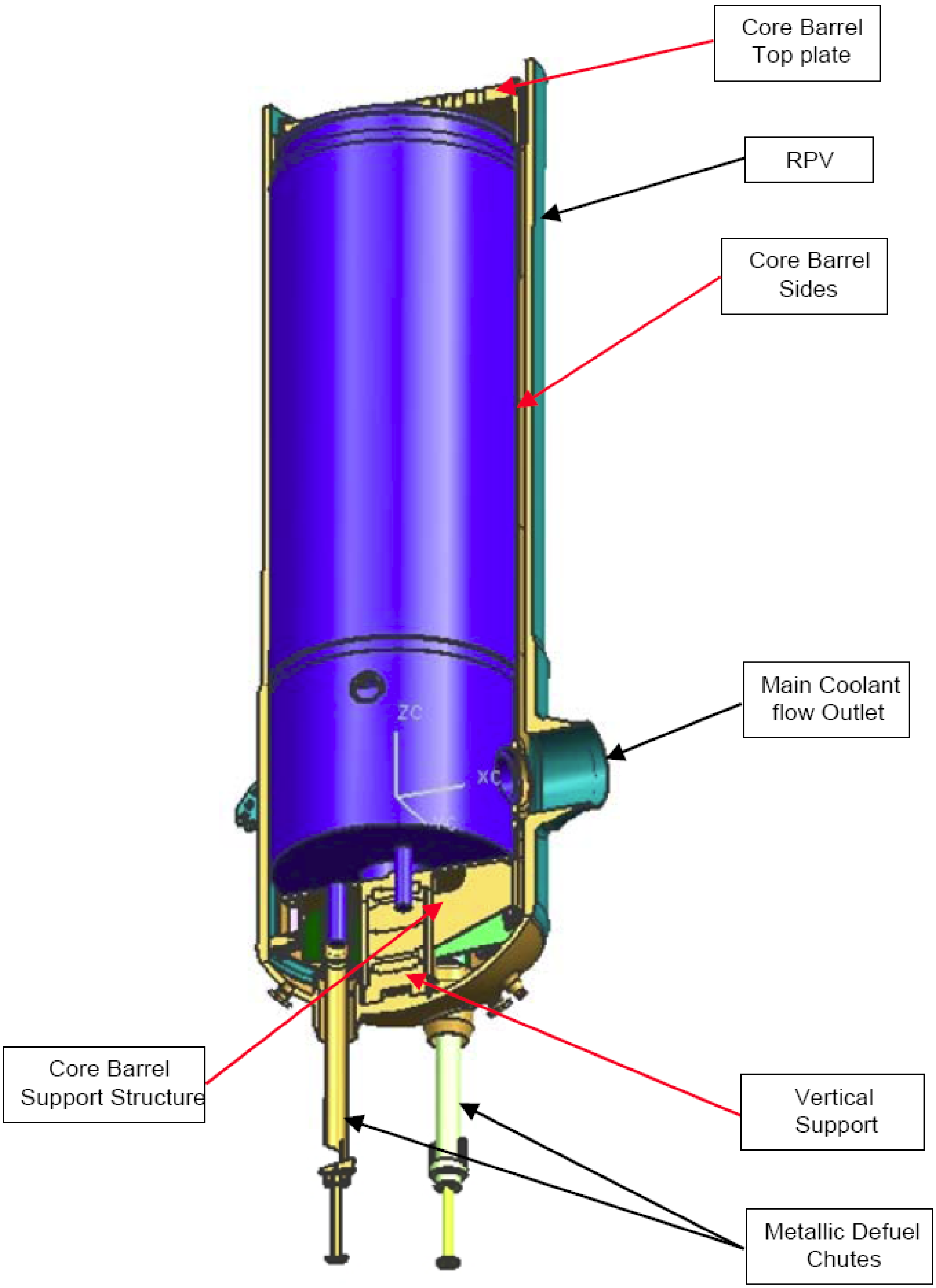

Figure 4: Core Barrel Assembly shown within the Reactor Pressure Vessel

Table 2: Core Barrel Assembly Metallic Materials

Component	Materials	Applicable ASME Design Code	Qualification Approach
CB Sides	SA-240 Type 316H	Section III, Division 1, Subsection NG (to 427 °C) + ASME Code Case N-201-5 (above 427 °C)	Use NRC-accepted ASME Specification + ASME Code Case N-201-5
CB Top Plate Assembly	SA-965 Grade F316H	Section III, Division 1, Subsection NG (to 427 °C) + ASME Code Case N-201-5 (above 427 °C)	Use NRC-accepted ASME Specification + ASME Code Case N-201-5
CB Base Plate	SA-965 Grade F316H	Section III, Division 1, Subsection NG	Use NRC-accepted ASME Specification
CB Lateral Guides and Support Rings	SA-965 Grade F316H	Section III, Division 1, Subsection NG (to 427 °C) + ASME Code Case N-201-5 (above 427 °C)	Use NRC-accepted ASME Specification + ASME Code Case N-201-5
CB Defuel Chute	SA-240 Type 316H	Section III, Division 1, Subsection NG	Use NRC-accepted ASME Specification
CBSS	SA-965 Grade F316H	Section III, Division 1, Subsection NG	Use NRC-accepted ASME Specification
CBA-RPV Interfaces	SA-965 Grade F22 Class 1 forgings and SA-387 Grade22 Class 1 plate (2.25Cr-1Mo)	Section III, Division 1, Subsection NG	Use NRC-accepted ASME Specification
CSC Lateral Restraint Strap Links and Pins	SA-240 Type 316H	Section III, Division 1, Subsection NG (to 427 °C) + ASME Code Case N-201-5 (above 427 °C)	Use NRC-accepted ASME Specification + ASME Code Case N-201-5
CSC Tie-Rod Holders	SA-965 Grade F316H forgings, SA-479 and SA-193 316H bar products	Section III, Division 1, Subsection NG (to 427 °C) + ASME Code Case N-201-5 (above 427 °C)	Use NRC-accepted ASME Specification + ASME Code Case N-201-5
CSC Bolting	SA-193 Grade B8M	Section III, Division 1, Subsection NG (to 427 °C) + ASME Code Case N-201-5 (above 427 °C)	Use NRC-accepted ASME Specification + ASME Code Case N-201-5

3.1.2.3 Hot gas duct system

The internal HGDS pipes (Figure 5) are passive structures within the HPB that provide the path for transport of hot helium gas (> 300 °C) between various components of the MPS. They consist of insulated pipes within the larger diameter pipes of the MPS/HPB. Cool helium from the HPC exits at ~108 °C flows in the annulus between the HGDs and the HPB to ensure that the temperature of the HPB material does not exceed 150 °C during normal operation. When the PCU is not available and the CCS is in operation, cooling gas is provided by diversion from the CCS blower outlet. Given the inherent pressure gradient between the annulus and inside of the HGDs when active flow is present, a failure in the HGD would result in cold gas flowing from the annulus to the HGD interior, away from the HPB. The HGDS is not relied on to perform safety functions during LBEs.

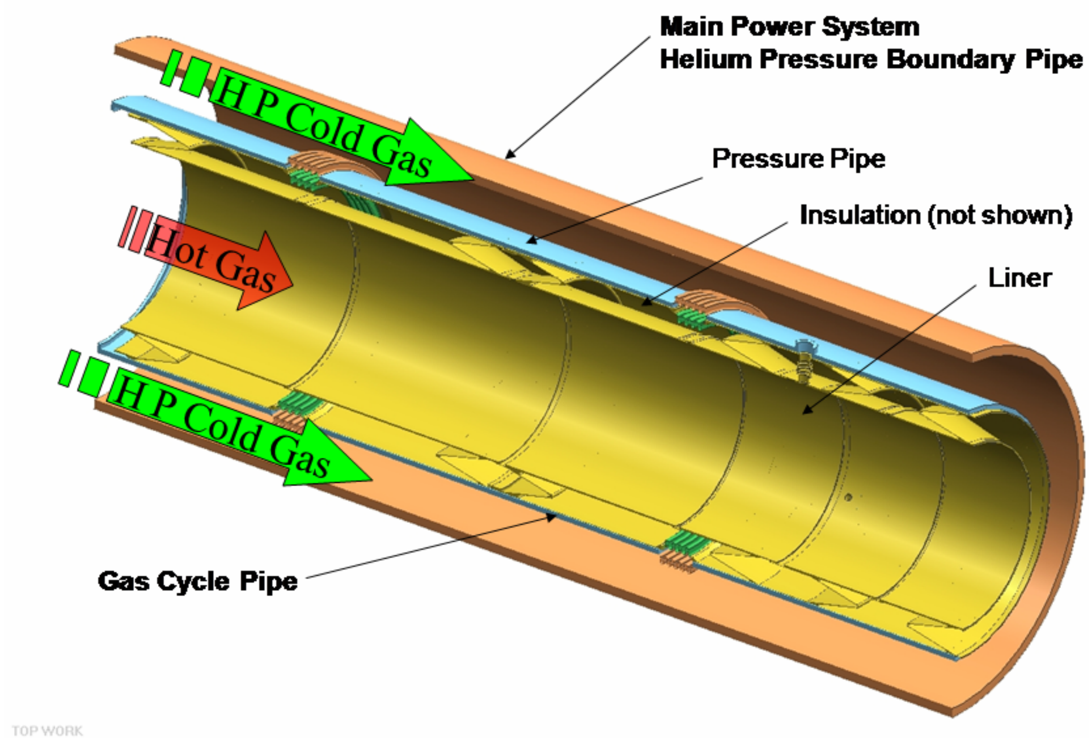

Figure 5: Hot Gas Duct System

The pressure pipes of the HGDS will be fabricated from SA-533, Type B, Class 2 low-alloy steel, which is consistent with the material selected for the outer HPB pipes. The HGD Liners, which face the hot helium gas, will be fabricated from SB-409 Alloy 800H. As can be seen in Table 3, the nominal temperatures to which these liners are exposed vary with the specific component, and range from 900 °C for the COP (Hot Duct)/CCS Inlet Pipe to 500 °C for the Core Inlet/CCS Outlet Pipe. At present, there is no ASME Code that is applicable to the temperature range encompassing all of the HGD Liners. However, there is a potential to utilize the present version of Section III, Subsection NH for all but the Core Outlet Liner. At present, the liner design is being developed on a first principles basis by analysis, using materials data provided in the draft German KTA 3221 standard. As a future consideration, it should be noted that an ongoing cooperative initiative between ASME and DOE is presently developing data to support the use of Alloy 800H within the framework of Section III, Subsection NH to 900 °C. The design of the HGDS is further discussed in Section 3.4 (Hot Gas Duct System).

Table 3: Hot Gas Duct System Metallic Materials

Component	Material	Applicable ASME Design Code	Qualification Approach
Core Outlet/CCS Inlet Pressure Pipe	SA-672 Grade J90 (Made from SA-533 Type B, Cl 2 plate)	Section III, Subsection NC	Use NRC-accepted ASME Specification
Core Outlet/CCS Inlet Liner (900 °C)	SB-409 Alloy 800H	Not applicable	Design by analysis, based on KTA 3221, supported by appropriate test data
Turbine Outlet Pressure Pipe	SA-672 Grade J90 (Made from SA-533 Type B, Cl 2 plate)	Section III, Subsection NC	Use NRC-accepted ASME Specification

Component	Material	Applicable ASME Design Code	Qualification Approach
Turbine Outlet Liner (515 °C)	SB-409 Alloy 800H	Not applicable	Design by analysis, based on KTA 3221, supported by appropriate test data
Core Inlet/CCS Outlet Pressure Pipe	SA-672 Grade J90 (Made from SA-533 Type B, Cl 2 plate)	Section III, Subsection NC	Use NRC-accepted ASME Specification
Core Inlet/CCS Outlet Liner (500 °C)	SB-409 Alloy 800H	Not applicable	Design by analysis, based on KTA 3221, supported by appropriate test data

3.1.2.4 Reactivity control and shutdown system

The RCSS comprises two subsystems: the RCS and the RSS. The RCS provides for the insertion and removal of control rods containing neutron absorber material. The RSS provides an independent and diverse means of reactor shutdown through the introduction of SASs that contain neutron-absorbing material. Safety functions assigned to the RCSS during LBEs relate to control of heat generation.

The primary function of the RCS during normal operation is reactivity control. It is also used for hot shutdown, fine reactor outlet temperature adjustment, and trimming. Twenty-four rods, 12 for control and 12 for shutdown, are employed in the RCS. Each of the 24 rods consists of six segments containing absorber material in the form of sintered B_4C rings between two coaxial Alloy 800H tubes, with the annulus containing the absorber being closed at both ends. Small openings are provided in the containers to avoid any significant pressure differentials. The individual segments are held together by articulated joints and suspended from one another to form a complete rod. All of the rods are freely suspended within graphite Side Reflector (SR) sleeves on Inconel Alloy 625 chains, attached to the control rod drive mechanism. Shock absorbers are provided on the end of each chain to absorb the kinetic energy of the falling rod in the event of a SCRAM. Finally, 316H stainless steel Guide Tubes are provided to guide the control and shutdown rods into the side reflector during all movements of the rods. The metallic materials for components described above are given in Table 4, along with relevant ASME Specifications.

Temperatures of the control rods during normal operation range nominally from 500 °C (top) to 700 °C (bottom), depending on location. Temperatures up to 1 000 °C are projected for certain DBEs/Design Basis Accidents (DBAs) involving passive heat removal. The shock absorber temperature will mimic that of the bottom sections of the control rods. The Alloy 625 chain will operate nominally at 500 °C, but could reach temperatures up to 800 °C in the case of DBEs/DBAs. The approach for qualifying these components is design by analysis, supported by appropriate test data. This is discussed in more detail in Section 3.5.

The primary function of the RSS is to provide cold shutdown capability during the maintenance mode. Dropping the SAS shutdown elements into the SAS channels in the centre reflector of the CSC facilitates maintenance in the cold shutdown state. The SASs contain B_4C as an absorber in a graphite matrix. The system as a whole consists of eight independent storage units that are mounted on a support structure in the RPV upper cavity. The storage containers receive SASs from the SAS feeder bin, and the SASs flow through the guide tubes into the borings inside the

CSC centre reflector. Fail-open actuators control the SAS release valves at the bottoms of the individual storage containers to allow the release of SASs into the centre reflector borings. Cross sections for SAS passage are dimensioned to prevent bridging of the spheres. For reasons of plant availability, the actuator has an uninterruptible emergency power backup.

The SAS guide tubes, which direct the SASs from the storage containers into the borings, are designed to accommodate the expansion difference between the CB and the installation point of the storage containers. A discharge vessel attached to the bottom of the RPV is used to prevent the SAS from leaving the channels after insertion.

A gas transport system is used to return the SASs from the bottom of the SAS channels in the centre reflector of the CSC to the RSS Feeder Bin for distribution to the Storage Vessels. The transfer medium is helium at FHSS transport gas temperature and pressure. The SAS cannot be transported at atmospheric pressure. A feed pipe is used to supply the helium from the FHSS blower outlet manifold to the RSS Discharge Vessel for fluidizing the spheres. After fluidization, the spheres are transported via the SAS Return Pipe to the Feeder Bin and Top Valve Bank for redistribution to the Storage Containers. The SAS return pipe provided for this purpose is part of the HPB. The helium gas is then returned to the FHSS blower inlet manifold.

The metallic materials employed for the various components of the RSS are listed in Table 4. The basic material for the Storage Containers and the associated valving and actuators is Type 316H stainless steel in plate, forging, and pipe forms. SA-193 Grade B7 Section III nuclear bolting is used, as well as a 13%Cr-4%Ni stainless material (SA-182 F6NM). A level probe in the valve assembly is designed to operate at temperatures to 500 °C, but this probe is not a structural component; the valve actuator operates at 300 °C. The Storage Container and the Guide Tubes are designed to ASME Section III, Subsection NG specifications.

Table 4: Metallic Materials for the Reactivity Control and Shutdown System

Component	Materials	Applicable ASME Design Code	Qualification Approach
RCS Chain	SB-446 Alloy 625	Not applicable	Design by analysis, supported by appropriate test data
RCS Absorber Cladding Tubes	SB-407 Alloy 800H	Not applicable	Design by analysis, based on KTA 3221, supported by appropriate test data
RCS Shock Absorber	SB-408 Alloy 800H	Not applicable	Design by analysis, based on KTA 3221, supported by appropriate test data
RCS Guide Tubes	SA-182 F316H SA-312 Gr 316H	Section III, Subsection NG (Tubes)	Use NRC-accepted ASME Specification + EJEMA8 (Bellows)
RSS Valve Block	SA-508 Grade 3 Class 1	Section III, Subsection NB	Use NRC-accepted ASME Specification
RSS Feeder Bin	SA-240 Gr 316H	Section III, Subsection NG	Use NRC-accepted ASME Specification
RSS Storage Container	SA-240 Gr 316H	Section III, Subsection NG	Use NRC-accepted ASME Specification

Component	Materials	Applicable ASME Design Code	Qualification Approach
RSS Valve and Valve Actuator Assembly	SA-312 316H pipe SA-240 316H plate SA-182 316H forging SA-479 316H shaft SA-193 B7 bolts	Section III, Subsection NG	Use NRC-accepted ASME Specification
RSS Guide Tubes	SA-182 F316H/ SA-312 Gr 316H	Section III, Subsection NG	Use NRC-accepted ASME Specification
RSS Storage Container Supports	SA-508 Grade 3 Class 1	Section III, Subsection NG	Use NRC-accepted ASME Specification
RSS Sphere Discharge Pipe	SB-564 Alloy 800H pipe SB-444 Alloy 625 flange Alloy 800H bellows SA-193 B7 bolts	Section III, Subsection NG (Tubes, Flange, Fasteners)	Use NRC-accepted ASME Specification + EJEMA8 (Bellows)
RSS Discharge Vessel	SA-508 Grade 3 Class 1	Section III, Subsection NB	Use NRC-accepted ASME Specification
RSS Sphere Return Pipe	SA-335 P1 pipe SA-182 F11 flange	Section III, Subsection NB	Use NRC-accepted ASME Specification
RSS Bottom Gas Transport System	SA-335 P1 pipe SA-182 F11 flange SA-193 B7 bolts	Section III, Subsection NC	Use NRC-accepted ASME Specification

The piping for the Sphere Discharge Pipes and the Discharge Vessel Inner Assembly is of Alloy 800H material; flanges associated with these two components are of Alloy 625. The temperatures associated with these two components are potentially as high as 900 °C. These components are also discussed in Section 3.5.

All other RSS components (Guide Tubes, Storage Container Supports, Discharge Vessel, Sphere Return Pipe, Discharge Valve Block, Feeder Bin, and Gas Transport System), of materials as indicated in Table 6, are designed to accepted ASME Section III specifications.

3.1.2.5 Helium inventory control system

The HICS is made up of four subsystems:

- Inventory Control System (ICS).
- Helium Purification System (HPS).
- Helium Make-up System (HMS).
- Dry Gas Seal Supply and Recovery System (DSRS).

The role of the ICS is to control helium mass flow within the MPS and to store helium associated with the FHSS and the MPS during maintenance outages. The HPS is used to remove gaseous contaminants (H_2O, O_2, CO, CO_2, H_2, and CH_4) from the helium gas stream and to purify the primary system after inspections and maintenance. The HMS is used to replenish the daily MPS helium leakage and to provide for initial filling of the MPS with helium. Finally, the DSRS supplies dust-free helium to the DGS of the turbomachine, recovers helium from the DGS of the turbomachine, and supplies dust-free helium to the blowers of the CCS, the FHSS, and the DSRS. The HICS SSC are not classified as Safety Related, as they are not relied on during DBEs.

Metallic materials for HICS components are listed in Table 5. Note that these components are located beyond the double isolation valves, and are thus not considered part of the main HPB. HICS-related piping that cannot be so isolated is considered part of the HPB and is accounted for in Table 1. Various grades of carbon steel (plate, piping forgings, forged and rolled flanges and fittings) are used for the helium storage tank pressure vessels) as well as an 8% to 9% Ni steel for the very low-temperature components. Temperatures involved with the nozzles can range from -40 °C to -196 °C. The helium tanks are designed to ASME Section VIII, Division 2.

An austenitic stainless steel, Type 316L, is used for the main piping, most of the components of the HPS, the DSRS filter vessel, and the helium storage extraction buffer manifold vessel. Pipe, plate, and forging product forms are used, in accordance with the materials specifications noted in Table 5. Lastly, more corrosion-resistant Type 321 stainless steel is used for the CuO/Cu reaction vessel and the catalytic recombiner of the HPS. The maximum temperature seen by these components will be nominally 200 °C. Design temperatures range from –200 °C to +250 °C, depending on the component. There are no apparent HICS materials issues, therefore the HICS will not be addressed further in this document.

Table 5: Metallic Materials for the Helium Inventory Control System

Component	Materials	Applicable ASME Design Code	Qualification Approach
Helium Storage Tanks	SA-516 Grade 70 plate, SA-105 vessel nozzles, SA-350 nozzles, SA-522 Type 1 nozzles	Section VIII, Division 2	Use NRC-accepted ASME Specification
Main Piping (beyond isolation valves)	SA-312 316L pipe	TBD	Use NRC-accepted ASME Specification
HPS	SA-312 316L pipe, SA-240 316L plate, SA-182 316L forgings	TBD	Use NRC-accepted ASME Specification
DSRS Filter Vessel	SA-312 316L pipe, SA-240 316L plate, SA-182 316L forgings	TBD	Use NRC-accepted ASME Specification
Helium Storage Extraction Buffer Manifold	SA-312 316L pipe, SA-240 316L plate, SA-182 316L forgings	TBD	Use NRC-accepted ASME Specification
Recombiner and Reaction Vessel	SA-312 321 pipe, SA-240 321 plate, SA-182 321 forgings	TBD	Use NRC-accepted ASME Specification

3.1.2.6 Fuel handling and storage system

The FHSS provides for the following functions:

- Initial loading of the reactor core with graphite spheres.
- Replacement of the graphite spheres with fresh fuel spheres intermixed with graphite spheres during initial start-up.

- Change over from the start-up core composition to a fuel only composition and thence to an equilibrium core.
- Loading and unloading of the fuel into and from the reactor core while the reactor is operating at power.
- Unloading and reloading of fuel from/to the used fuel tanks for certain maintenance activities.
- Discharge of spent fuel to the spent fuel tanks.
- Detecting and removing physically damaged spheres.
- Burn-up and activity measurement of spheres removed from the core.
- Storage of spent fuel and temporary storage of used fuel.
- Safeguards, monitoring, and inventory control.

The isolatable portions of the FHSS SSC are not classified as Safety Related as they are not relied on during DBEs.

The metallic materials choices for the various FHSS components shown in Table 6 are based on a design pressure of 9.7 MPa maximum and normal operating temperatures from 5 °C up to 250 °C. The choices also reflect compatibility with the chemical environment. The valve blocks are a combination of Cr-Mo SA-965 Grade F22 and alloy steel SA-182 Grade 11. The valve block seals will be replaced during the six-year maintenance cycle.

SA-335 Grade P1 ferritic seamless piping is used for the ribbed pipe sections that transport the fuel. During events in which forced circulation is lost, the fuel loading pipes (between the charge locks and the RPV) could reach elevated temperatures that remain below 371 °C for short periods. This is still within the design limits of ASME Section III, Subsection NB or NC.

Type 316L stainless steel (SA-240) will be used for the spent fuel tanks. Whether ASME Section III, Subsection NB or NC, or ASME Section VIII, Division 2 design specifications will be invoked for the spent fuel tanks has not been decided/confirmed.

No significant material issues have been identified with the FHSS, therefore, the FHSS will not be addressed further in this document.

Table 6: Metallic Materials for the Fuel Handling and Storage System

Component	Materials	Applicable ASME Design Code	Qualification Approach
Valve Blocks	SA-965 Grade F22 SA-182 Grade F11	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
Ribbed Pipes	SA-335 Grade P1	Section III, Subsection NB or NC	Use NRC-accepted ASME Specification
Graphite Tank	SA-240 316L [To be confirmed]	Section III, Subsection NB or NC or Section VIII, Division 2	Use NRC-accepted ASME Specification
Used Fuel Tank	SA-240 316L [To be confirmed]	Section III, Subsection NB or NC or Section VIII, Division 2	Use NRC-accepted ASME Specification
Spent Fuel Tanks	SA-240 316L [To be confirmed]	Section III, Subsection NB or NC or Section VIII, Division 2	Use NRC-accepted ASME Specification

3.2 REACTOR PRESSURE VESSEL

The RPV encloses the RU and is an integral part of the HPB. It is relied upon for important safety functions during DBEs and their corresponding deterministic DBAs [2]. The RPV is therefore designated as Safety Related in accordance with 10 CFR 50.2 and [3]. Design and fabrication of the RPV are in accordance with ASME Code Section III, Division 1, Subsection NB, supplemented with ASME Code Case N-499-2.

3.2.1 Key Functions and Requirements

The functions of the RPV include the following:

- To provide support for the CBA containing the CSC (graphite structures) and Core.
- To maintain Core and CSC geometry.
- To provide a high-integrity pressure boundary to contain the reactor heat transfer fluid (helium), radionuclides, and the reactor core.
- To support and locate the pipework carrying the heat transfer fluid to and from the PCU.
- To provide a conductive heat transfer path to the RCCS in the event of loss of forced cooling or normal heat transfer by helium.
- To provide a boundary against the release of radionuclides in addition to that provided by the ceramic fuel spheres.

During normal operation, the RPV operates at a maximum temperature of 300 °C and 9 MPa. For the temperature-limiting DBA, which postulates heat removal via the RCCS for an extended period of time, the maximum projected temperature is 432 °C and occurs approximately 52 h into the unmitigated transient.

3.2.2 Design Overview

The RPV, Figure 6, consists of a vessel closure head, a hemispherical upper dome with a flanged cylindrical section for mounting the vessel closure head, a main cylindrical section, and a lower torispherical head. The part-spherical upper dome, containing the closure head, is located at the top of the vessel, and is joined to the vessel by welding in the field after installation of permanent CS. The closure head incorporates eight forged nozzle penetrations for mounting the RSS release valve actuators.

The partial spherical dome in the closure head also incorporates a forged ring. The forged ring includes nozzle penetrations for the 24 RCS mechanisms. Additional forged nozzle penetrations are provided for the three fuel recirculation return pipes, the return line of the CBCS, the return line of the RSS, and a common entry for in-core instrumentation.

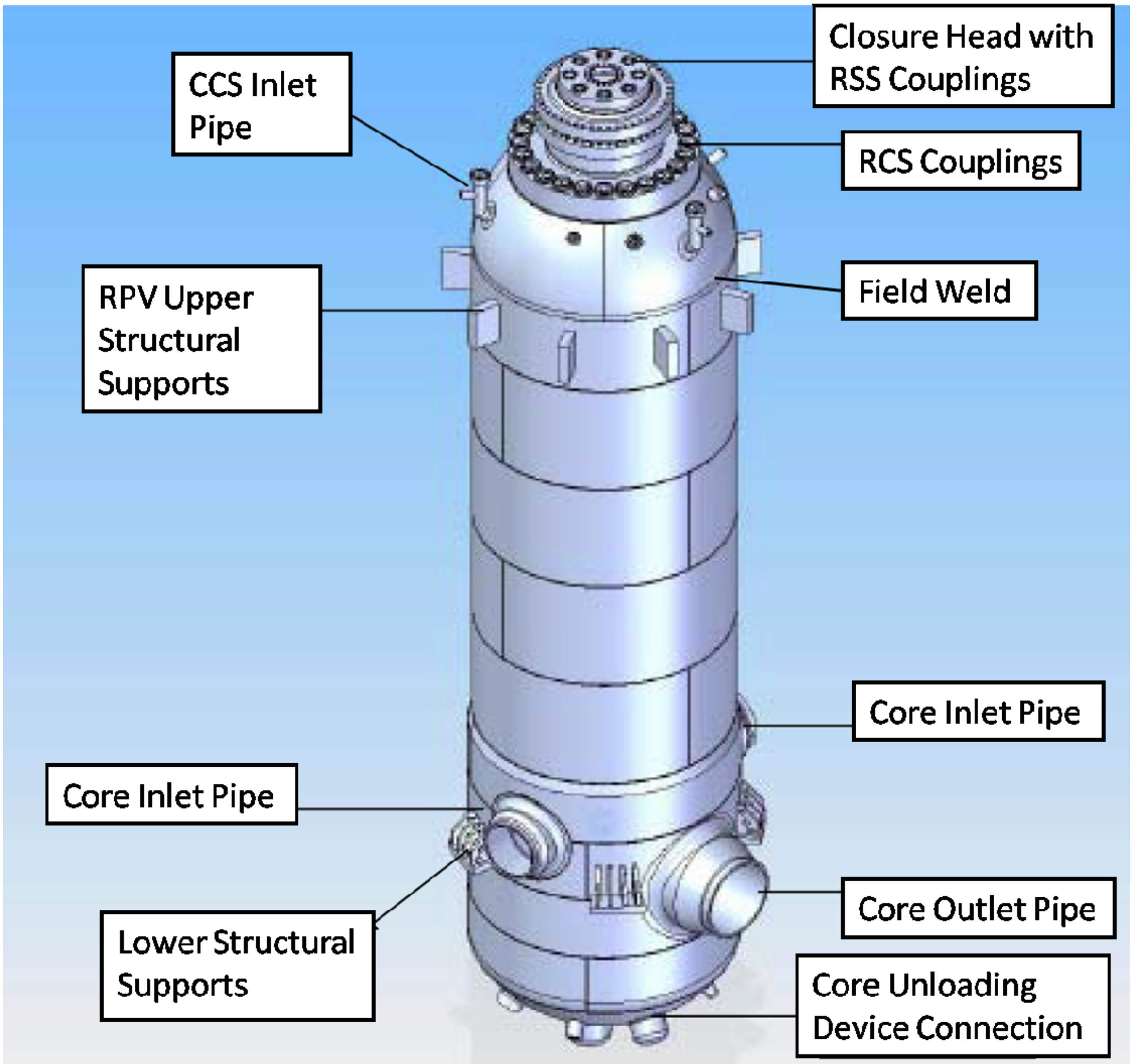

Figure 6: Overall View of the Reactor Pressure Vessel

The main cylindrical section of the vessel comprises seam-welded rolled sections, which are welded together. The lower cylindrical section of the vessel is reinforced, has two nozzle forgings for the reactor helium gas inlets, one larger nozzle for the reactor helium gas outlet, and an outlet for the pressure balancing Link Line. The vessel is supported on five support lugs, spaced circumferentially, and welded to the lower cylindrical section at the elevation of the COP centerline. This reinforced shell section is designed to withstand reactor vessel support loads and manifold nozzle loads. The support lugs allow radial growth, while providing vertical as well as horizontal support for the bottom of the RPV. Radial keys are provided near the upper end of the main cylindrical section for additional lateral support.

The lower head is welded to the main cylindrical section and has three openings for the discharge chutes to which the FHSS is connected, a nozzle forging for the discharge line of the RSS, a nozzle forging for the discharge line of the CBCS, and three instrument penetration openings. There is also an opening for access to the bottom CS. This access opening is intended for use only during initial installation. On completion of installation, a cover is welded to the access opening. Finally, the bottom head also incorporates an array of nine radial keys that provide for internal vertical support of the CS and the pebble core contained within the RPV.

The welding of the upper partial spherical dome to the main cylindrical section will be performed on-site to enable the installation of the CS (CB, graphite reflector, etc.) into the cylindrical and lower head section. The reinforced cylindrical CB is mounted within the reactor vessel and is supported and located by the lower and upper keys and the vertical support (refer to Section 3.3). There is an annular space between the CB and the inner wall of the RPV for flow of cool helium from the CBCS. This flow is at a lower temperature and higher pressure than the inlet gas entering the CBA and provides a temperature separation path, thereby maintaining the RPV material temperature below 300 °C.

The nuts and washers of the closure head hold-down bolts have convex mating surfaces in order to ensure a good load distribution. The flanges are joined using a double seal arrangement with an option to add a welded lip seal.

The key dimensions of the RPV are as follows:

- Height from bottom of lower dome to upper dome mounting flange section: 27 228 mm.
- Height of closure head from the formed mounting flange section: 935 mm.
- Internal diameter: 6 200 mm.
- Nominal wall thickness: 180 mm.
- Top and reinforced parts thickness: 285 mm.
- Vessel head internal radius: 3 100 mm.
- Vessel head nominal thickness: 165 mm.
- Vessel bottom dome internal radius: 3 750 mm.
- Vessel bottom dome nominal thickness: 165 mm.
- RPV maximum external diameter: 6 770 mm.
- Mass of the RPV vessel, including its closure head but excluding the contents of the vessel: approximately 942 t.

The materials selected for the RPV are approved grades for reactor vessel components under ASME Section III, Division 1, Subsection NB, Class 1. They are:

- SA-508 Grade 3 Class 1 in accordance with the ASME Specification for Quenched and Tempered Vacuum-Treated Carbon and Alloy Steel Forgings for Pressure Vessels.
- SA-533 Type B Class 1 in accordance with the ASME Specification for Pressure Vessel Plates, Alloy Steels, Quenched and Tempered, Manganese-Molybdenum and Manganese-Molybdenum-Nickel.
- SA-540, Grade B24, Class 3, in accordance with the ASME Specification for Alloy Steel Bolting Materials for Special Applications.

Information on manufacturing and fabrication processes, Non-destructive Testing (NDT), fracture toughness, pressure/temperature limits is covered in subsequent RPV sections.

3.2.3 Bases for Structural Design

3.2.3.1 Specifications

The RPV will be designed, fabricated, tested, installed and inspected to the standards required by ASME Section III, Subsection NB, Class 1, supplemented by ASME Code Case N-499-2. The major materials of construction are SA-533 Grade B Class 1 plate and SA-508 Grade 3 Class 1 forgings. These materials are the same as those which have been approved and used in the design, fabrication, and operation of LWR pressure boundaries worldwide for over 30 years.

The metal temperatures of the PBMR RPV during normal operation are in the range of 260 °C to 300 °C, which is comparable to the service conditions seen in LWRs. ASME Section III, Division 1, Subsection NB (Class 1 Components) allows the use of these materials to a maximum temperature of 371 °C.

The limiting DBAs postulated for the RPV are those in which active cooling paths are assumed to be lost and heat removal is via the RCCS. For these events, although the average temperature of the RPV is only approximately 300 °C, the maximum temperature may reach 432 °C, based on conservative estimates, at about 50 h into the event. Total time at temperatures > 371 °C will be < 200 h. This is, however, outside the range of conditions permitted under ASME Section III, Division 1, Subsection NB.

ASME Code Case N-499-2 (applicable to the use of SA-533 Grade B, Class 1 Plate and SA-508 Class 3 Forgings and their Weldments for Limited Elevated Temperature Service above the limits of Section III, Division 1, Subsection NB) was developed to gain ASME approval for temperature transients similar to those noted above (relevant to the design of the MHTGR). In particular, the inquiry to the ASME Committee asked whether the SA-533 and SA-508 steels could '... be used for Section III, Division 1, Class 1 construction at temperatures exceeding 371 °C up to 427 °C during Service Level B events but not exceeding 538 °C during service Level C and D events for limited time of exposure not to exceed 1 000 h'. The ASME Committee replied in the affirmative with the following guidance:

- Metal temperatures shall not exceed 427 °C for Level B events or 538 °C during Level C or D events.
- Maximum cumulative time between 371 °C and 427 °C shall not exceed 3 000 h; cumulative time between 427 °C and 538 °C shall not exceed 1 000 h.
- The number of events with metal temperatures > 427 °C shall be limited to a total of three.
- The rules for materials in Section III, Division 1, NB-2000 and NH-2000 for Class 1 Components in Elevated-Temperature Service shall apply to the materials in this case.
- The design rules of NB-3000 shall be satisfied for all design and operating conditions where metal temperature does not exceed 371 °C.
- The design rules of NH-3000 shall apply for all events where metal temperature exceeds 371 °C.

Further, Code Case N-499-2 specifies the creep-fatigue interaction damage envelope to be used as well as physical and mechanical properties. Finally, additional rules of the following articles related to Section III, Division 1 are stated to apply:

- NH-4000, Fabrication and Installation of Elevated-Temperature Components.
- NH-5000, Examination of Elevated-Temperature Nuclear Components.

- NH-6000, Testing of Elevated-Temperature Components.
- NH-7000, Protection Against Over-pressure of Elevated-Temperature Components.

The appropriateness of applying ASME Code Case N-499-2 to the PBMR RPV for DBAs and the degree of conservatism that N-499-2 includes can be examined in light of the allowable stress intensity values that N-499-2 prescribes for various times and temperatures. Consider the allowable stress intensity values (S_{mt}) shown in Table 7 for temperatures from 371 °C to 538 °C and times from 1 h to 3 000 h. First, S_{mt} is the lower of S_m (the time-independent stress intensity) and S_t (the time-dependent stress intensity). S_m is derived from yield strength and ultimate tensile strength values and S_t is based on various creep parameters. Values of both contain significant conservatisms.

Note from the table that the value of S_{mt} is constant at 26.7 ksi for all times to 3 000 h at 371 °C through 427 °C. (This is also the allowable stress intensity value at all temperatures down to 24 °C.) The value of $S_{mt} = S_m$, indicates time-independence of the allowable stress. It is within this range of conditions that the maximum temperature portion of the RPV will remain for the vast majority of the time during the relevant DBAs. As indicated earlier, a maximum temperature of 432 °C will be achieved at approximately 50 h, and the time above 371 °C will be limited to < 200 h. Three such excursions, the limit imposed by ASME Code Case N-499-2 for temperatures > 427 °C, could be accommodated and no significant creep deformation would be expected. Even allowing for a 20 °C overshoot of the predicted RPV material maximum temperature, the 1 000 h stress intensity value is 95% of that allowed for an excursion up to 427 °C.

Table 7: Allowable Stress Intensity Values, S_{mt}

Temperature (°C)	Time at Temperature (h)						
	1	10	30	100	300	1 000	3 000
	Allowable Stress Intensity, S_{mt}, Values (ksi) for SA-533 Grade B Class 1 and SA-508 Class 3 RPV Steels						
371	26.7	26.7	26.7	26.7	26.7	26.7	26.7
399	26.7	26.7	26.7	26.7	26.7	26.7	26.7
427	26.7	26.7	26.7	26.7	26.7	26.7	26.7
454	25.5	25.5	25.5	25.5	25.5	25.5	-
482	24.3	24.3	24.3	24.3	24.3	24.0	-
510	22.5	22.5	22.5	22.5	22.0	16.0	-
538	20.7	20.7	20.7	18.0	14.0	9.5	-

3.2.3.2 Fabrication and examination considerations

Fabrication of the RPV will conform to the requirements of ASME Section III, Subsection NB, covering special processes for fabrication to ensure vessel integrity. The requirements associated with controls on welding, material composition, and heat treatment will be comparable to those in NRC Regulatory Guides 1.31, 1.34, 1.43, 1.44, 1.50, 1.71, and 1.99. The NDE program to be applied during fabrication of the RPV will conform to the requirements of ASME Section III. The NDE program includes appropriate application of ultrasonic, radiographic, liquid penetrant, and magnetic particle testing.

Since final assembly of the RPV will occur in the field, the RPV design includes provisions to manufacture and hydraulically test the vessel as two component parts, using temporary closures. These two components, the upper part spherical dome, and the lower main sections of the vessel, will be joined by welding after installation of the permanent CS in the field. The on-site welding is to be performed using machinery specially designed for that purpose. A 100% radiographic inspection will be performed on the weld. A pneumatic pressure test will be performed on the MPS pressure boundary system, including the complete RPV.

3.2.3.3 Emissivity considerations

The RPV is an essential link in the passive decay heat removal path. For this link to function as designed, it is necessary that the surfaces of the RPV (internal and external) maintain an emissivity level of at least 0.85. Although this is not typically a difficult value to obtain on steels, it does require that a thermal or chemical treatment be applied to produce an oxidized surface with a high emissivity level.

The PBMR has an ongoing experimental program aimed at ensuring RPV surfaces with high emissivity. Remaining issues are associated with demonstrating a practical surface treatment consistent with vessel size, dimensional stability constraints, hydro testing and cleaning requirements, NDE inspections, transport, and installation. Current efforts are concentrating on a low temperature oxidation after hydro testing or a chemical treatment before shipment.

3.2.4 Reliability and Integrity Management

PBMR will implement a RIM Program for all passive SSC including the RPV, the remainder of the primary HPB, the RPV supports and the RCCS piping. The PBMR program will be based on the recommendations detailed in the *Reliability and Integrity Management Pilot Study Report*. This program is being developed with the cooperation of the ASME. The purpose of the RIM program is to select the combination of design, fabrication, inspection, surveillance, operation, and maintenance requirements that meet the reliability goals in an efficient and cost-effective manner. The RIM process includes: (1) defining the safety and economic reliability goals; (2) defining candidate combinations of design, fabrication, inspection, surveillance, operation, and maintenance procedures that may be able to achieve and maintain the reliability goals; (3) assessing the reliability of these combinations to determine those that provide an efficient and cost-effective means to achieve the reliability goals; (4) selecting, implementing, and updating the final RIM strategies that will provide the desired level of reliability in an efficient and cost-effective manner and provide assurance that the reliability performance will be maintained throughout the lifetime of the plant; and (5) long-term monitoring of SSC performance, feeding back to periodic updates of the RIM program.

An area to be given particular attention in the RPV RIM program is assuring adequate resistance to brittle fracture over the plant operating lifetime. The predicted shift in RT_{NDT} at the end of the PBMR RPV life is comparable to that of Boiling Water Reactor (BWR) vessels and significantly lower than that of Pressurized Water Reactor (PWR) vessels currently in service. This is primarily due to the lower end-of-life fluence (estimated to be ~1.2 x 10^{18} n/cm^2 (E > 1MeV)), which is 10 to 40 times less for the PBMR RPV than for a typical PWR vessel. The low neutron fluence of the PMBR design results in a low level of embrittlement over the life cycle. The RCS and RSS housings that are fixed to the top of the RPV dome are at a significant distance from the RPV beltline, resulting in even less exposure to irradiation.

Due to the specific design characteristics of the PBMR, a 'standard' surveillance program to monitor vessel materials irradiation response (i.e., that typical of a LWR) is not possible. There is no physically feasible location for surveillance specimens within the PBMR RPV where the fast neutron flux is appreciably higher than that at the inside surface of the RPV while, at the same time, having the same nominal temperature as the RPV (to provide a lead factor > 1, which is common for LWR surveillance programs). The neutron flux at the outside of the CB is essentially identical to that at the inside of the RPV, but the metal temperatures on the CB are much higher. While higher fluxes are found within the CB, temperatures are even higher still. To summarize, surveillance specimens would provide, at best, a real-time indication as to the state of the RPV material. Obtaining results from such specimens typically takes up to two years. Further, the RPV closure head is not designed to permit its removal during normal maintenance outages. Access to surveillance capsules within the RPV would need to be through openings/penetrations in the RPV.

From the above, it is clear that a 'standard' surveillance program, suitable for LWRs, will not be practical for the PBMR. However, the primary objective of a surveillance program (i.e., assuring adequate material fracture toughness) can be achieved through other methods. First, it has already been noted that the fluence seen by the PBMR RPV will be a small fraction of that seen by a typical LWR RPV. This suggests that the historic irradiation database for these materials can be applied in a conservative manner. There are several options to supplement the application of the historic database. For the PBMR DPP, an irradiation and testing program is planned that will incorporate the use of specimens from the RPV manufacturing coupons. Irradiation conditions in the Materials Test Reactor will envelope the RPV design conditions.

3.2.5 Assessment of Licensing Basis

3.2.5.1 Licensing basis events and design basis accidents

The RPV represents an important element of the overall PBMR safety concept. In addition to serving as a secondary boundary to radionuclide release (the fuel particles themselves being the first and most important), it also forms a link in the passive decay heat removal path between the core and ultimate heat sink (Figure 7) that is relied upon when active forced cooling systems (the PCU and CCS) are not available.

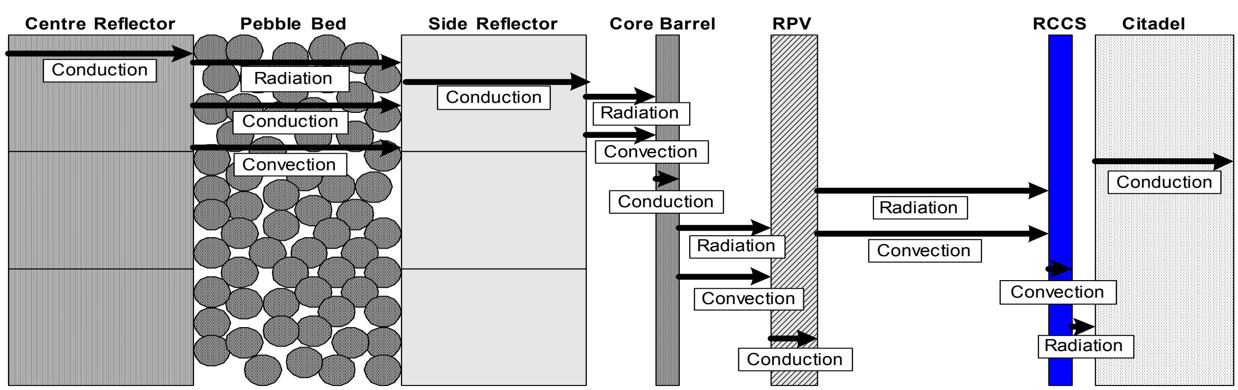

Figure 7: Passive Heat Transport Path from Core to Heat Sink

LBEs that are significant for the RPV are those involving implementation of the passive heat transport path for decay heat removal, plus the Safe Shutdown Earthquake (SSE). These events are summarized in Table 8 [2], which lists DBEs for the PBMR in the second column. As further described in [2], DBAs are derived from DBEs, by assuming that only SSC classified as Safety Related [3] are available to fulfill their respective functions. Since the PCU and CCS are not classified as Safety Related, only the passive heat removal path via the RCCS is assumed to be available in the case of the DBAs, listed in the third column of Table 8.

For the RPV, the DBEs and corresponding DBAs listed in Table 8 are enveloped by two decay heat removal thermal transients, differentiated by whether the HPB remains intact, plus the structural loads imposed by the SSE. In all cases, for the corresponding DBAs, heat removal is assumed to be via the RCCS. The time/temperature profile for the highest temperature locations within the RPV during a representative depressurized transient event is shown in Figure 8. As can be seen, the peak temperature (432 °C) occurs at approximately 56 h into the transient.

Table 8: PBMR Design Basis Events and Related Design Basis Accidents*

DBE Designation	Design Basis Event	Deterministic DBA Designation	Deterministic Design Basis Accident
DBE-1c	Loss of PCU with core conduction cooling to RCCS	Deterministic DBA-1	Loss of PCU with core conduction cooling to passive mode of RCCS
DBE-2b	Control rod withdrawal with CCS forced cooling	Deterministic DBA-2	Control rod withdrawal with core conduction cooling to passive mode of RCCS with unfiltered release
DBE-3a	Small, auto-isolated HPB break with PCU forced cooling	Deterministic DBA-3	Small, unisolated HPB break with core conduction cooling to passive mode of RCCS with unfiltered release
DBE-3b	Small, manually isolated HPB break with CCS cooling		
DBE-4a	Small, unisolated HPB break with pumpdown with RCCS cooling	Deterministic DBA-4	Small, unisolated HPB break with core conduction cooling to passive mode of RCCS with unfiltered release
DBE-4b	Small, unisolated HPB break with without pumpdown with RCCS cooling		
DBE-5b	HX tube break, manually isolated with RCCS cooling	Deterministic DBA-6	Heat Exchanger (HX) tube break, unisolated with core conduction cooling to passive mode of RCCS with unfiltered release
DBE-6a	HX tube break unisolated with pumpdown with RCCS cooling with filtered release		
DBE-6b	HX tube break unisolated with pumpdown with RCCS cooling with unfiltered release		
DBE-6c	HX tube break unisolated without pumpdown with RCCS cooling with filtered release		
DBE-6d	HX tube break unisolated without pumpdown with RCCS cooling with unfiltered release		
DBE-7a	Medium, auto-isolated HPB break with PCU cooling	Deterministic DBA-7	Medium, unisolated HPB break with core conduction cooling to passive mode of RCCS
DBE-7b	Medium, isolated HPB break with CCS cooling		
DBE-11a	Safe shutdown earthquake with PCU cooling	Deterministic DBA-11	Safe shutdown earthquake with core conduction cooling to passive mode of RCCS

DBE Designation	Design Basis Event	Deterministic DBA Designation	Deterministic Design Basis Accident
DBE-11b	Safe shutdown earthquake with CCS cooling		

Note: * DBEs and DBAs in this table were identified for an earlier 268 MWt PBMR design. They are considered representative for the present 400 MWt design, but are to be confirmed.

The maximum temperature vs time profile associated with a representative pressurized transient event is depicted in Figure 9. Note that the maximum RPV temperature (~400 °C) occurs somewhat earlier in the transient (between approximately 30 h and 40 h) and is somewhat lower than that of the depressurized event. While there is no significant pressure within the RPV during the depressurized event, the maximum nominal pressure during the limiting pressurized event is ~6 MPa vs the design value of ~9 MPa. This pressure corresponds to the equilibrium pressure of the collapsed Brayton cycle when shut down from the 100% power operating condition.

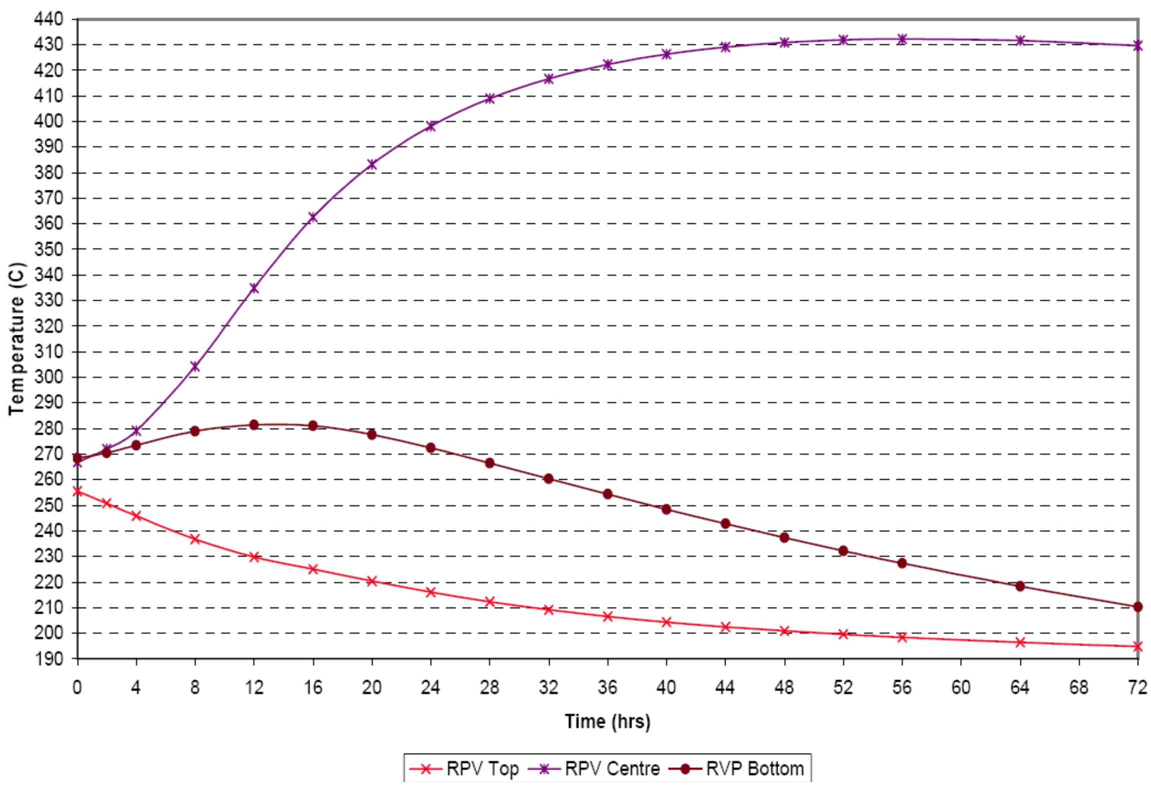

Figure 8: Reactor Pressure Vessel Temperatures for Representative Depressurized Transient

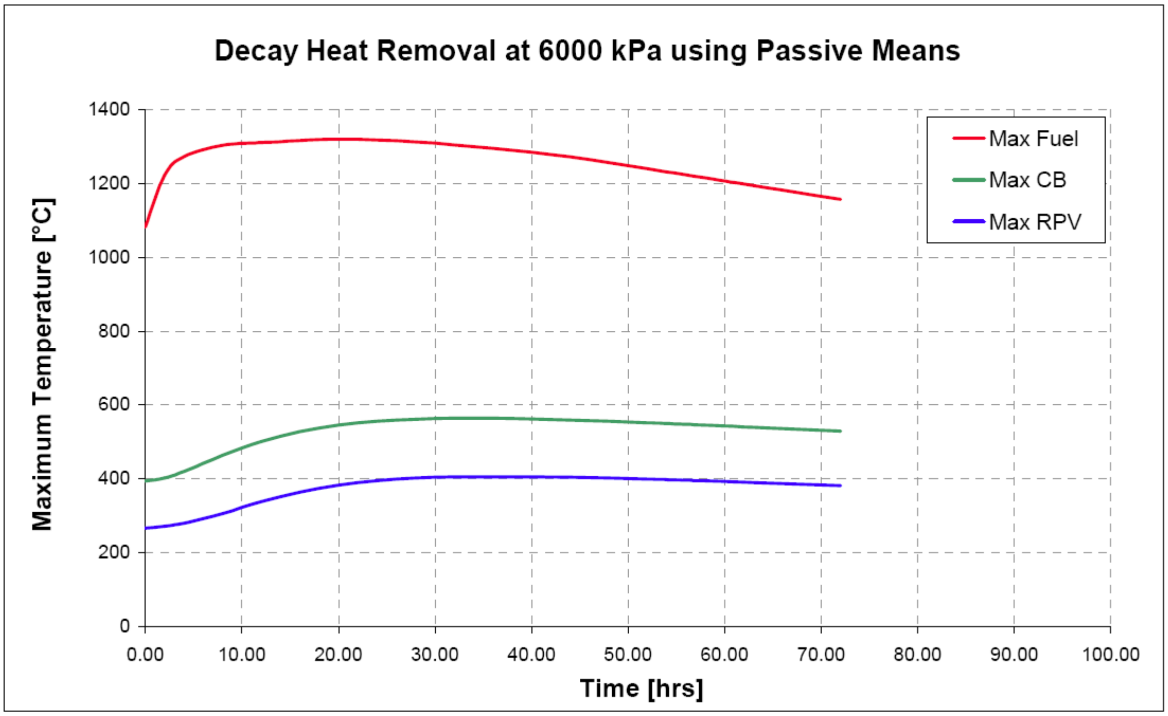

Figure 9: Component Temperatures for Representative Pressurized Transient

For the depressurized transient, which results in higher vessel temperatures, the time associated with the event depends upon the elapsed time for reestablishing active cooling. In the limit, where it is assumed that active cooling is not reestablished, the long-term trend would be as shown in Figure 10.

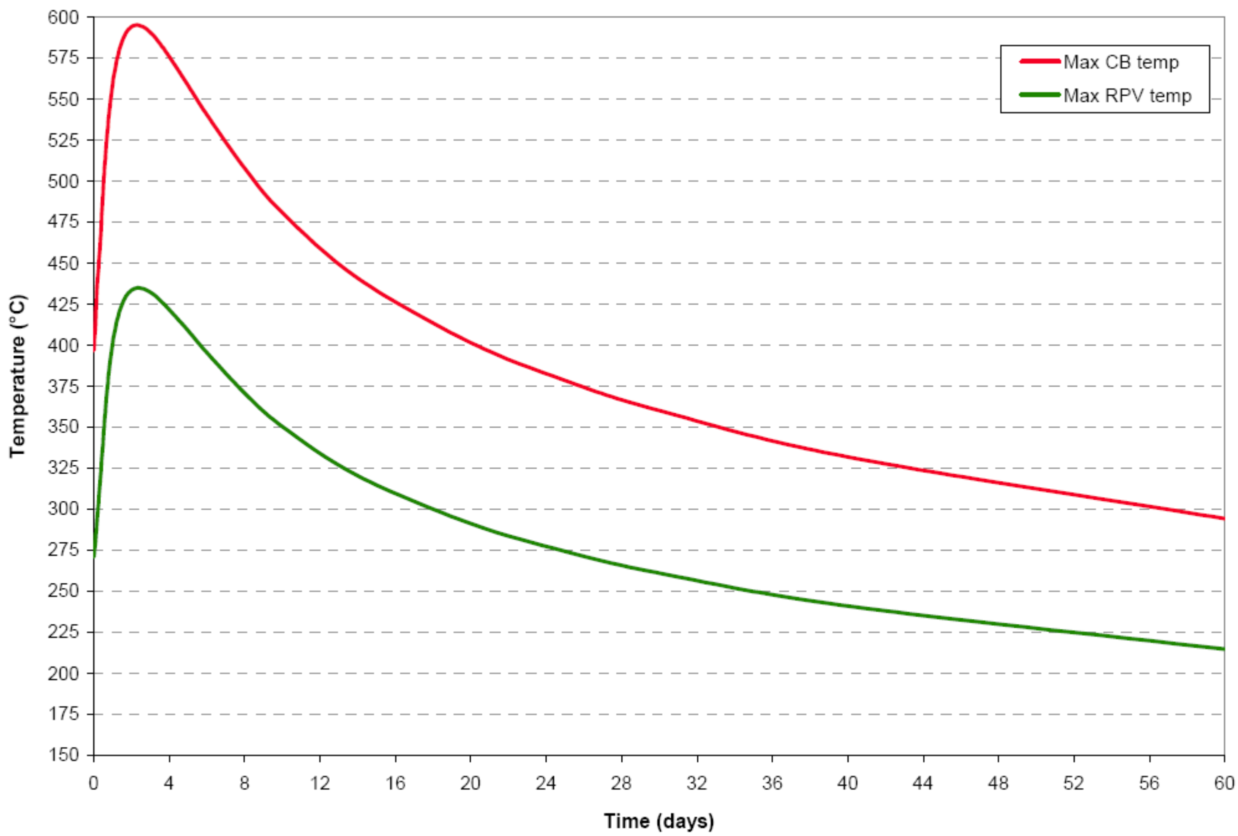

Figure 10: Depressurized Transient Long-term Trends

3.2.5.2 Assessment

During normal operation and AOOs, the RPV is designed for and operated within the limits of Section III of the ASME Code for Class 1 components. Seismic loads corresponding to the SSE are accounted for in the design. In the case of DBEs and their corresponding mechanistic DBAs, the design of the RPV relies upon ASME Code Case N-499-2, which has not yet been approved by the NRC. For the limiting depressurized event, which assumes no restoration of forced cooling, the longest time over which the maximum temperature in the RPV exceeds 371 °C (700 °F) is approximately eight days, or somewhat < 200 h. Present calculations predict that the peak RPV temperature (432 °C) will be > 427 °C (the temperature above which the allowed cumulative time at temperature reduces from 3 000 h to 1 000 h) for a short period of time (< 72 h).

On this basis, the PBMR RPV design represents an appropriate and conservative application of Code Case N-499-2.

3.3 CORE BARREL ASSEMBLY

The CBA contains and supports the CSC and the pebble fuel core. It is relied upon for important safety functions during DBEs and their corresponding mechanistic DBAs. The CBA is therefore designated as Safety Related in accordance with 10 CFR 50.2 and [3].

3.3.1 Key Functions and Requirements

The CBA, together with the CSC, has the following important safety functions:

- To provide and maintain the geometry of the reactor core. This is essential to both reactivity control (control of heat generation) and heat removal safety functions. The specific role of the CBA is to support and maintain the geometry of the CSC.
- To provide and maintain channels for the Control Rods. This is part of the reactivity control (control of heat generation) function. The channels themselves are in the CSC (inner SR). The specific role of the CBA is to support and maintain the geometry of the CSC.
- To provide and maintain channels for the SASs. This is part of the shutdown (control of heat generation) function. The specific role of the CBA is to support and maintain the geometry of the CSC.
- To provide and maintain a path for passive heat transfer. This is part of the passive heat removal safety function. Designing the CSC to allow heat to be transferred from the core to the CBA and designing the CBA to allow heat to be transferred from the CSC to the RPV provides this function.

Finally, the CBA serves to separate the incoming gas adjacent to the RPV from the gas volume passing through the core. In this respect, the CBA functions to prevent the impingement of hot gas from the core on the RPV. Higher pressure in the RPV volume ensures that any leaks through the CBA will be from the cold gas region adjacent to the RPV, to the hot gas of the reactor core.

3.3.2 Design Overview

The CBA (Figure 4 and Figure 11) comprises three major components: the CBSS, located below the CSC; the CB Sides (cylindrical portion of the CB); and the CB Top Plate. The CBSS and CB sides are welded together, while the CB top plate is attached by bolting. In addition to these major components, there are additional components that include the CSC Circumferential Restraints, the Defuel Pipes, the Upper and Lower Supports, the Upper and Lower Lateral Guides, and the CB Vertical Support Assembly. The CBA support is shown in Figure 12.

The CBA Vertical Support Assembly supports the CBA vertically from a single central point on the lower head of the RPV. This support structure allows for a small amount of unconstrained angular rotation of the CBSS around any horizontal axis to compensate for possible temperature-induced bowing of the CB sides. This is accomplished by the spherical contact surfaces of the Vertical Support Bearing within the CB Vertical Support Assembly (refer to Figure 12). The spherical contact surfaces are hard coated to prevent self-welding of the contact surfaces. During a seismic event, the Vertical Support Assembly allows the CBA to move laterally until the gaps to the upper and lower seismic restraints close and the loads from the CBA are transmitted to the RPV.

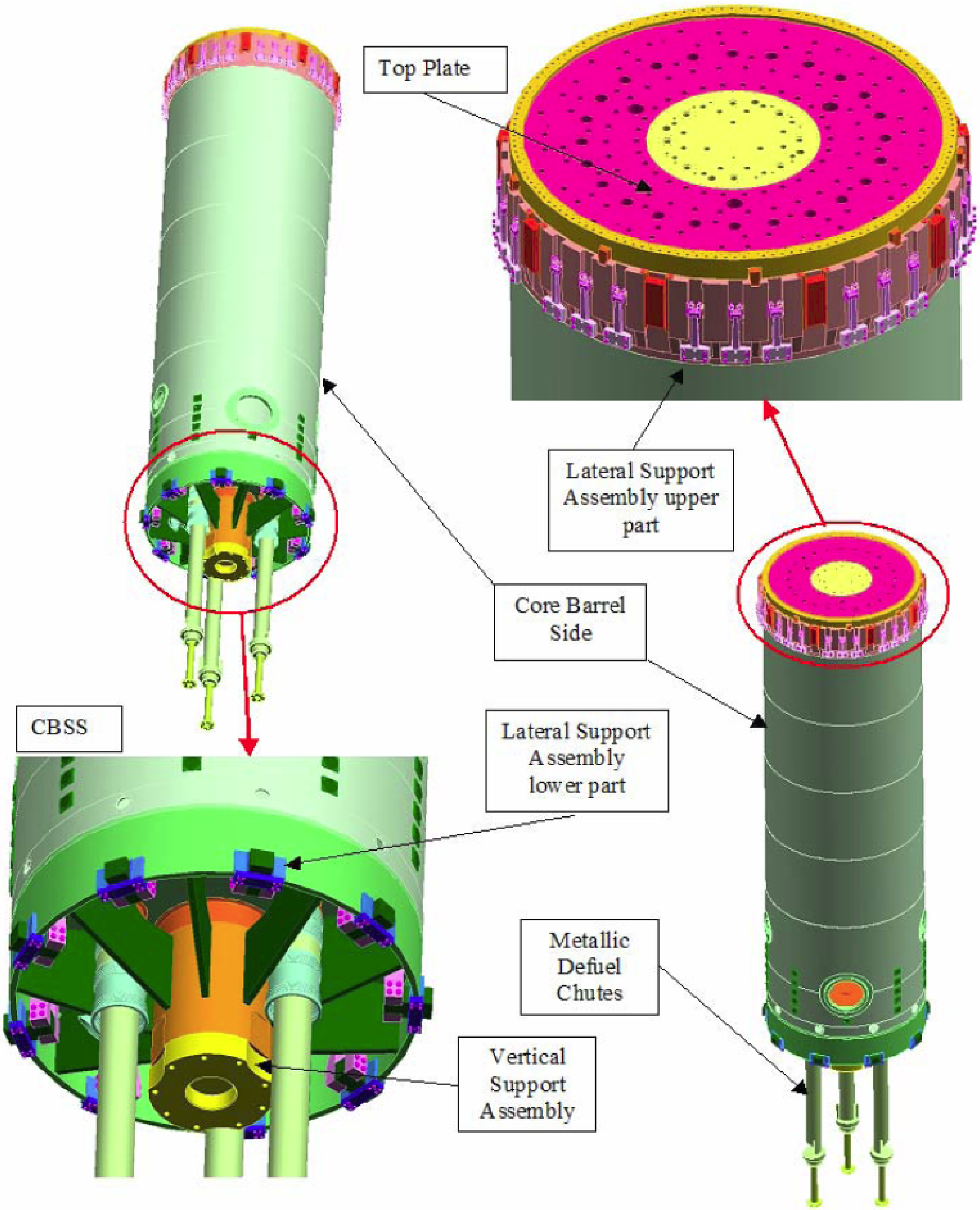

Figure 11: Core Barrel Assembly

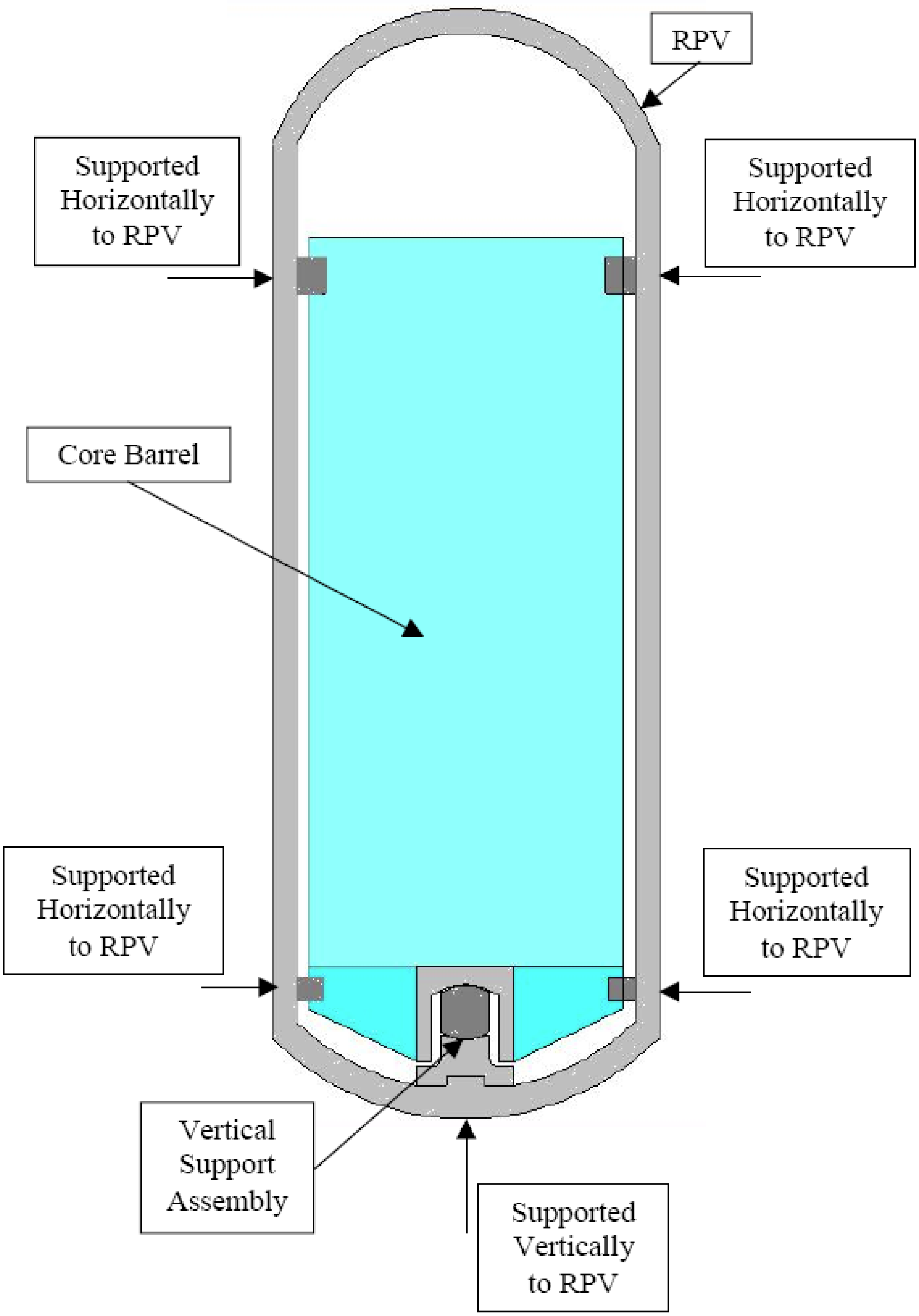

Figure 12: Core Barrel Assembly Support Concept

The CBSS is comprised of a horizontal plate (referred to as the CB bottom plate) that is supported by several radial webs (called the CBSS Beams). A shell called the CBSS Skirt circumferentially surrounds the outside of the webs, providing additional stiffness and channeling for the CBCS flow. The primary design requirement for the CBSS is stiffness. This stiffness provides a stable base upon which the CSC can be assembled. The Lower Lateral Guide laterally centers the CBSS and transfers lateral loads between the CBSS and the RPV during normal operation. However, this guide is sufficiently compliant to allow for CB bowing.

During a seismic event, the CBSS Skirt augments the Lower Lateral Guide by providing a direct load path between the CBA and the RPV. This is by means of radial keying to the CBSS Skirt. An engineered gap between the radial keys and CBSS restricts the functioning of this mechanism to seismic events. The CBSS provides interfaces for the systems that access the CSC through the bottom plate, including the defuel chutes, the RSS channels, and the instrumentation system.

The CB Sides are fabricated from several individual rolled steel courses (welded rings). Overall, the CB Sides are 22.9 m in height with an inner diameter of 5.75 m and a wall thickness of 60 mm. The CSC Circumferential Restraints are welded to the CB Sides. These restraints provide for location and support of the CSC Bottom Reflector during normal operation as well as during seismic events. The CB Sides provide lateral support for the CSC in the region of the core during seismic events.

The top of the CB Sides is laterally supported and centered by the Upper Lateral Guide. The Upper Lateral Guide transfers lateral loads between the CBA and the RPV while allowing for differential thermal expansion between the RPV and the CBA in the vertical direction. The Upper Lateral Guide is sufficiently compliant to allow for CB bowing and is augmented during a seismic event by the Upper Support Ring. The functioning of these components is analogous to the functioning of the Lower Lateral Restraint and the Lower Support Ring. The seismic restraint scheme for the CBA, described above, ensures that the relative lateral motion between the CS and the RPV during an SSE is limited. All moving contact surfaces are coated to prevent in-service self-welding.

The CB Top Plate is bolted to the top of the CB Sides. The Top Plate provides for interfaces between the CBA and the following systems: the RCS, the RSS, the FHSS, and the In-core Delivery System. In addition, the Top Plate is provided with a removable plug for access to the CSC and the core. The CSC Top Reflector is suspended from the Top Plate.

The Defuel Chutes are also part of the CBA. These chutes connect the Defuel Chutes within the CSC to the FHSS Core Unloading Device. The Defuel Chutes are designed to accommodate the differential motion between the CBA and the RPV. The Defuel Chutes are constrained such that fuel cannot escape, even in the event of failure of the tube attachments.

3.3.3 Bases for Structural Design

3.3.3.1 Specifications

The materials utilized in the CBA are summarized in Table 9. The primary material used in the CBA components is Type 316H austenitic stainless steel. Where there is an interface between the austenitic stainless steel CBA and the low-alloy steel RPV, 2.25Cr-1Mo ferritic steel is used as a bridge material. This is primarily to reduce secondary stress effects due to differential expansion at the interface with the RPV. The components made from Type 316H stainless steel are either of bar, plate or forged material; the components made from 2.25Cr-1Mo are either plate or forgings. All of the CBA materials are approved for use in ASME Section III, Division 1, Subsection NG.

Table 9: Core Barrel Assembly Materials

Material	Product Form	Specification
Type 316H Stainless Steel	Bar	SA-479 316H SA-193 Grade B8M Class 1
Type 316H Stainless Steel	Forging	SA-965 Grade F316H
Type 316H Stainless Steel	Plate	SA-240 Type 316H
2.25Cr-1Mo Ferritic Steel	Forging	SA-965 Grade F22 Class 1
2.25Cr-1Mo Ferritic Steel	Plate	SA-387 Grade 22 Class 1

ASME Section III, Subsection NG provides design rules for construction of core support structures to temperatures of up to 371 °C for most ferritic steels (e.g., 2.25Cr-1Mo) and to 427 °C for all of the austenitic alloys, including Type 316 stainless steel. The normal maximum operating temperature for the CBA is ~480 °C on a conservative basis. However, as shown in Figure 13, peak temperatures approaching 600 °C can be projected for limited periods of time during the enveloping depressurized passive heat removal transient. As with the RPV, the temperatures associated with a pressurized passive heat removal transient are less severe (Figure 9). Long-term trends for an unmitigated transient are shown in Figure 10.

The higher temperatures in the CBA would be associated with the CB Sides at elevations corresponding to the location of the core. This means that, with conservative assumptions, portions of the CB Sides fabricated from Type 316H stainless steels may operate above the limits of Subsection NG during normal operation, as well as during DBEs.

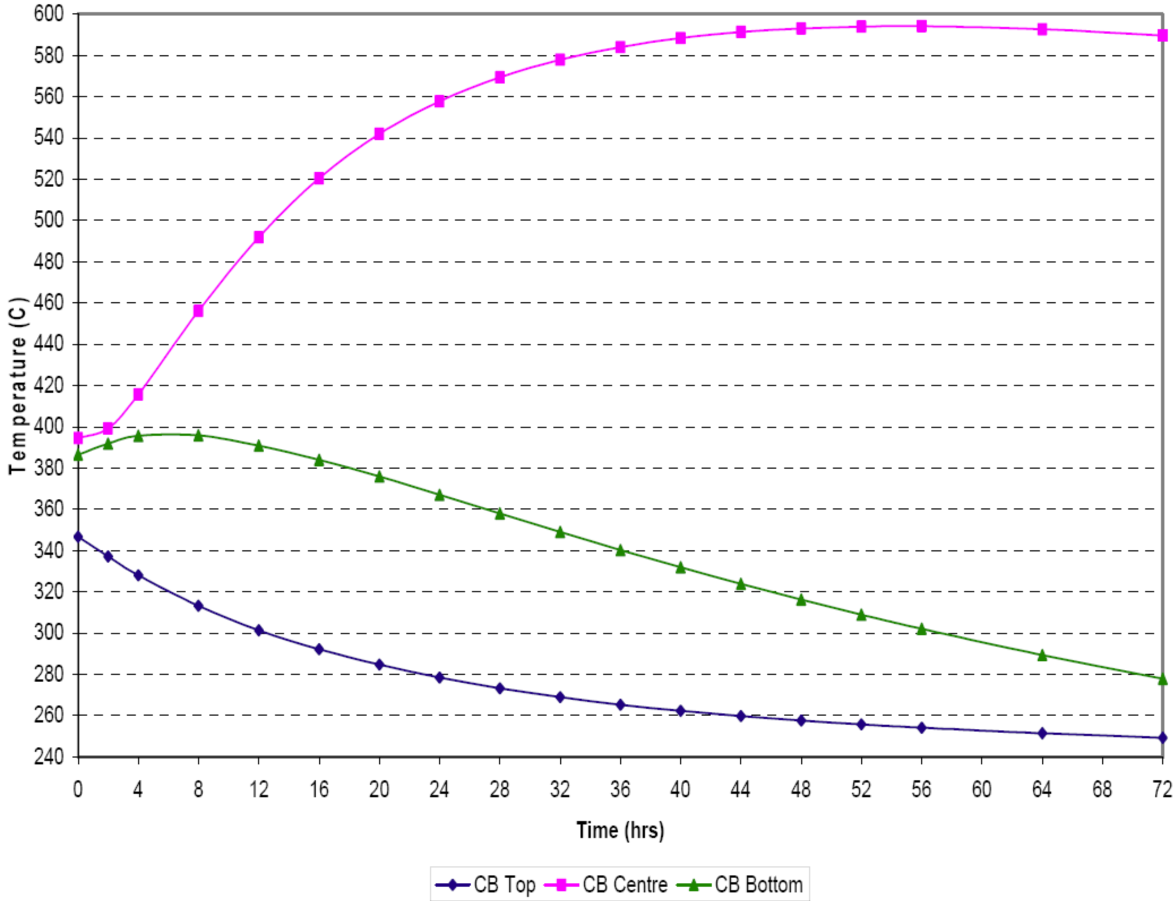

Figure 13: Core Barrel Assembly Temperatures for Representative Depressurized Transient

Code Case N-201-5 (Class CS Components in Elevated Temperature Service, Section III Division 1) was developed to allow operation at temperatures that exceed the normal time-independent limits of Subsection NG. The original inquiry to the ASME Standards Committee was 'Under what rules shall Section III Core Support Structures be constructed when metal temperatures exceed those for which allowable stress values are given by Section II, Part D.' The reply (i.e., Code Case N-201-5, approved September 18, 2006) was that these structures may be constructed in accordance with the following:

- rules for construction in Subsection NG as modified by Part A or Part B of this Case (N-201-5), where the time/temperature requirements of Appendix XIX of Part A are satisfied; or
- rules for construction in Subsection NG as modified by Part B of this Case, where the time/temperature requirements of Appendix XIX are exceeded.

Parts A and B contain modifications and additions to Subsection NG necessary for applications at temperatures above the upper limit for Subsection NG.

In particular:

- Part A (Rules for Construction of Subsection NG Extended for Restricted Service at Elevated Temperatures without Consideration of Creep or stress-rupture) permits use of S_m (time-independent allowable stress intensity) values for service within specified ranges of time and temperature. For Type 316H stainless steel, for example, design can be based on

S_m values from 427 °C to 593 °C, but with limitations on time. A full 300 000 h of life is permitted at temperatures in the range of 427 °C to 482 °C, but by 593 °C, the time limitation is approximately 20 h. The normal operating temperatures of the CBA are evaluated to remain < 482 °C on a conservative basis.

At the upper end of the CBA operating temperature range (500 °C), the life limitation of Part A would be < 50 000 h. This would not be sufficient for design to full lifetime, but would potentially accommodate DBEs. For DBAs that involve extended periods of RCCS cooling, however, the alternate rules of Part B provide additional flexibility for operation at the higher temperatures.

- Part B (Rules for Construction of Subsection NG Altered for Service at Elevated Temperatures to Suitably Account for Creep and Stress-Rupture) provides for explicit consideration of time-dependent phenomena (creep and stress-rupture). Allowable stress values, S_{mt}, are the lower of S_m (time-independent) and S_t (time-dependent) values and are provided to 300 000 h from 427 °C to 816 °C for Type 316H stainless steel. Part B S_{mt} values are provided for 2.25Cr-1Mo to 300 000 h at 371 °C through 593 °C (1 100 °F) and for lesser times at temperatures to a maximum of 649 °C. Note that the rules of Part B can also be applied for time/temperature conditions that do not exceed those of Appendix XIX.

A Design Specification will be produced and certified for the CBA in accordance with Section III, Division 1, Subsection NG and Code Case N-201-5. A design report, also produced in accordance with the requirements defined in the ASME Code, Section III, will be provided to demonstrate that the fabricated component meets the requirements of the relevant Design Specification and the applicable ASME Code.

3.3.3.2 Fabrication and processing factors

The fabrication of the CBA will be done in a clean area dedicated to austenitic stainless steel component manufacturing. This will preclude contamination of the austenitic stainless steel surfaces with low-alloy steel particles. Normal cleanliness practices will apply to ensure contaminant-free surfaces during and after fabrication. Cleanliness standards provided for by the American Society for Testing and Materials (ASTM) will be invoked.

Due to the carbon content of the Type 316H stainless steel, some level of sensitization is expected to occur during fabrication and/or in service. Since helium is non-corrosive, no Intergranular Stress Corrosion Cracking (IGSCC) is to be expected during operation.

The CBA is primarily manufactured from hot-rolled and solution-annealed plate products (the use of cold-worked stainless steel components is not foreseen). These plates are joined (welded) together through full penetration butt-welds and/or fillet welds. Austenitic stainless steels are readily weldable, using established welding processes. The optimum welding process for a particular weld is dependent on the joint design and size. The welding procedures will ensure compliance with the stipulations of NG-2433 on delta ferrite contents of the welds, which will be between 3 FN and10 FN.

The 2.25Cr-1Mo-material is also a common material used for high-temperature structural applications, and is readily weldable, using most welding processes. The current CBA design does not require welding of this material.

3.3.3.3 Non-destructive examination

NDE of the materials and welds for the CBA will conform to the requirements of ASME Section III, Subsection NG and the supplementary requirements of Code Case N-201-5. The components of Type 316H stainless steel will be subjected to radiographic testing, ultrasonic testing, dye penetrant testing, and visual inspections. The welding procedures utilized for joining the heavier plate sections will control the resultant grain sizes in the Heat Affected Zone (HAZ) and weld metal. This will allow acceptable sensitivity of the NDE techniques applied. In addition to the NDE techniques mentioned above, magnetic particle inspections could also be utilized on the components of 2.25Cr-1Mo-material.

The inspection methods will be applied in accordance with the stipulations of the relevant articles in ASME Section V. Qualified personnel will perform the inspections and evaluate the results, as prescribed in Subsection NG-5220. The acceptance standards applied to the welds will be at least equivalent to the relevant provisions in Subsection NG.

3.3.3.4 Emissivity considerations

One of the safety functions of the CBA is to ‘provide and maintain a path for passive heat transfer’. The property of the CBA that is most important is the emissivity of its surface. The design of the CBA assumes a total normal surface emissivity of 0.85 for both the CSC and the RPV sides of the CB. Preliminary testing has indicated that this is achievable through certain treatments or coatings applied to the CBA surfaces. There are various ways of conditioning the surface to attain a stable, adherent surface layer with the required emissivity. The selected process must be compatible with the fabrication processes of the CBA.

3.3.3.5 Radiation damage

There is a significant amount of data available on radiation effects in austenitic stainless steels. The maximum end-of-life dose on the CB will be ~5 x 10^{18} n/cm^2 (E > 1 MeV). This is well below the fluence level of 1 x 10^{21} n/cm^2 (E > 1 MeV), where irradiation studies have shown that the mechanical properties of austenitic stainless steels start to change due to irradiation-induced mechanisms.

3.3.4 Assessment of Licensing Basis

3.3.4.1 Licensing basis events and design basis accidents

Like the RPV, the CBA represents an important element of the overall PBMR safety concept. In addition to containing and supporting the CSC and pebble core, the CBA also forms a link between the core and ultimate heat sink (Figure 7) that is relied upon when active systems (the PCU and CCS) are not available.

As with the RPV, the DBEs and corresponding DBAs that are significant for the CBA are listed in Table 8, and are enveloped by the transient events involving heat removal via the RCCS under pressurized and depressurized conditions (Figure 9 and Figure 10), plus the SSE. Like the RPV, the peak temperature (somewhat < 600 °C) is associated with the depressurized transient event at approximately 56 h into the transient.

For the limiting depressurized transient, the time associated with the event depends upon the elapsed time for reestablishing active cooling. In the limit, where it is assumed that active cooling is not reestablished, the time during which the normal operating limit (482 °C) of the CBA would be exceeded is something < 240 h (Figure 10).

3.3.4.2 Assessment

During normal operation and AOOs, the more highly loaded components of the CBA (at and below the level of the CBSS) are designed and operated within the limits of ASME Section III, Subsection NG for core support structures, taking full account of seismic loadings. However, the more lightly loaded, but higher-temperature regions of the CBA, the CB Sides and Top Plate, may exceed the time-independent 427 °C allowable under Subsection NG (without Code Case N-201-5) during normal operation. During DBEs and their corresponding DBAs, temperatures could be considerably higher (conservatively estimated up to a maximum of somewhat < 600 °C) with times above the Code Case N-201-5 normal operations benchmark of 482 °C up to ~240 h. These conditions are covered conservatively under Part B of ASME Code Case N-201-5. Overall, the design of the CBA and the application of ASME Code Case N-201-5 are evaluated to provide a conservative basis for regulatory compliance.

3.4 HOT GAS DUCT SYSTEM

The HGDS channels the flow of high-temperature helium gas from and to the reactor, to and from the CCS, and within the high-temperature sections of the PCU. In addition to normal operation, the HGDS supports decay heat removal during certain DBEs, but is not required to function during DBAs.

3.4.1 Key Functions and Requirements

The functions of HGDS are related to heat transport, decay heat removal, and the retention of radionuclides. Specifically, these functions are to:

- Channel high-temperature helium from the reactor to the PCU. In addition to normal plant operation, this function supports decay heat removal via the PCU during certain AOOs and DBEs.
- Channel high-temperature helium to and from the CCS. This function supports maintenance operations and supports a diverse means of decay heat removal during certain AOOs and DBEs.
- Limit heat losses from the high-temperature helium streams to the remainder of the system. This function is important for power conversion efficiency.
- Prevent high-temperature helium from coming in contact with the HPB. In this respect, the HGDS ensures that the HPB remains at temperatures below the design temperature and supports the HPB functions to retain the helium and to serve as a barrier to the release of radionuclides.

3.4.2 Design Overview

The sections of pipe providing for transport of high-temperature helium gas within the PBMR MPS are dual circuit pipes. Their design is based on developments within the German HTR program. In the dual circuit design (Figure 5), the high-temperature helium is contained within an inner circuit (the HGDs) and isolated from the HPB. Active cooling is provided in the annulus between the HPB and HGDs by diverting flow from the HPC outlet (PCU in operation) or from the CCS circulator outlet (CCS in operation). This design limits temperatures seen by the HPB and permits the use of a conventional material for the HPB (SA-533, Grade B, Class 1 or 2) that is maintained within established ASME Code conditions (refer to Section 3.1.2.1).

The HGDs consist of the following (Figure 14):

- Core Outlet/CCS Inlet HGD – connects the Reactor Outlet Nozzle Liner to the inlet of the Turbine. A branch of this pipe connects to the CCS Heat Exchanger (HX) inlet. (**Note:** The Reactor Outlet Nozzle Liner is a Carbon Fiber Reinforced Carbon (CFRC) component that is separately addressed in a companion white paper on CSC.)
- Turbine Outlet HGD – connects the Turbine outlet to the low-pressure inlets of the two Recuperators.
- Core Inlet/CCS Outlet HGDs – connect the high-pressure outlets of the Recuperators to the CSC inlet plenum within the reactor. Branches of these pipes return cooled helium from the CCS circulator, when the CCS is in operation.

As shown in Figure 5, each HGD consists of three layers. The inner layer is a non-pressure-retaining liner that actually channels the high-temperature helium flow. The outer layer is a Pressure Pipe. The Pressure Pipes, which operate at temperatures < 200 °C, are designed for the maximum differential pressures that will be seen during normal operation and transients, including DBEs. In all cases where flow is present, the higher pressure is on the outside of the Pressure Pipes, tending to leak from the low-temperature to the high-temperature regions. Insulation is provided between the inner liners and outer Pressure Pipes. The ‘V’-shaped supports between the Pressure Pipe and the inner liner are designed to reduce stresses and to restrict convective flow of helium within the insulating layer.

The core outlet to turbine HGD has an outer diameter of 2 070 mm, inner diameter of 1 950 mm, and is fabricated in formed sections. The turbine outlet to recuperator HGD has an outer diameter of 2 430 mm, inner diameter of 2 300 mm, and is fabricated to include a dual header for connection to the two recuperators. The two core inlet HGDs from the recuperators have an outer diameter of 1 184 mm and inner diameter of 1 120 mm.

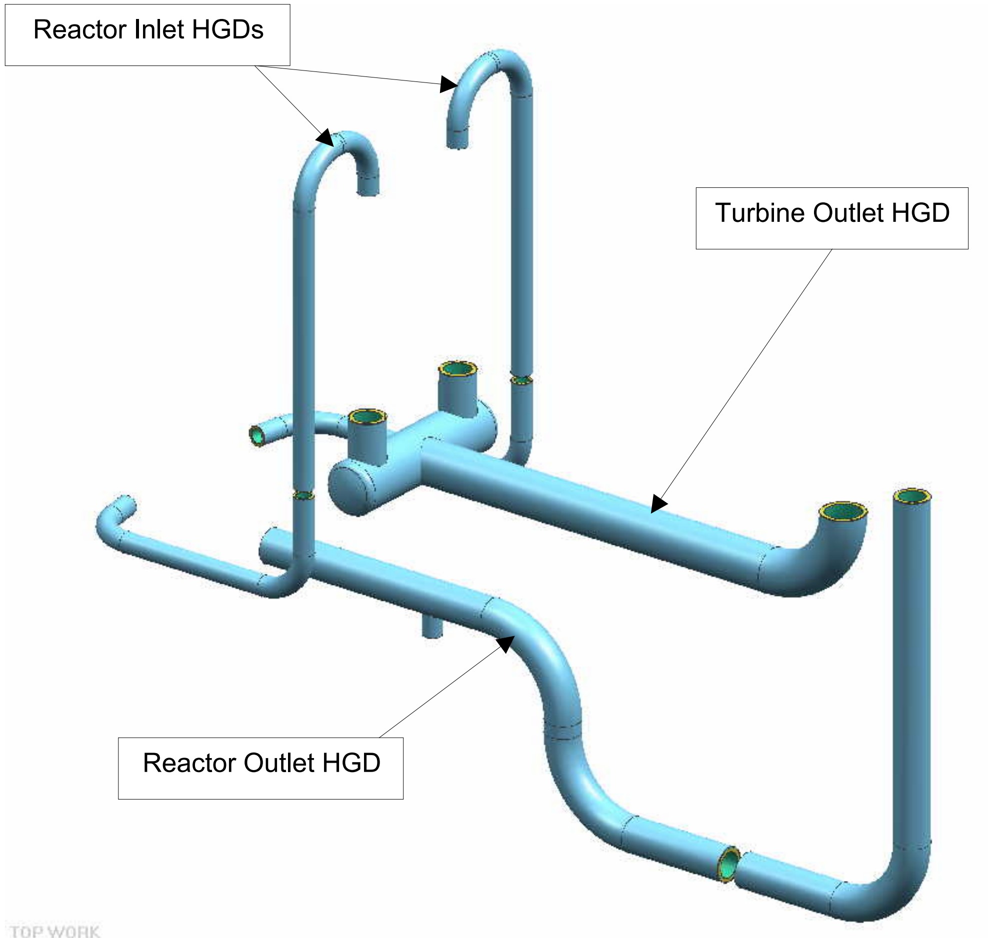

Figure 14: Schematic Layout of Hot Gas Ducts

Nominal operating conditions are summarized in Table 10, which provides the temperatures and flows within the respective HGDs, plus the differential pressure across the outer Pressure Pipe. With the PCU in operation, cool helium gas diverted from the HPC outlet (nominally at 108 °C) provides for flow in the annular interspaces between the Pressure Pipes and the HPB. This ensures that the outer Pressure Pipe material is operated within the design temperature limits of ASME Section III, Subsection NC. When the CCS is in operation, cooled helium is diverted from the CCS circulator outlet.

The materials utilized in the HGDS are shown in Table 11. The HGD metallic parts are fabricated from SA-533, Grade B, Class 2 steel (Pressure Pipe) and Alloy 800H (Liner and Liner Supports). The insulation is fabricated from Al_2O_3 and SiO_2 (Saffil).

Table 10: Nominal Hot Gas Duct Operating Conditions

Pipe	Differential Pressure (kPa)	Nominal Temperature (°C)	Averaged Helium Flow Velocity (m/s)
Reactor Outlet Pipe	400	900	64
Turbine Outlet Pipe	6 000	515	73
Reactor Inlet Pipes	100	500	69
CCS – Cold Pipe	100	350	~ 60
CCS – Hot Pipe	400	900	~ 60

Table 11: Hot Gas Duct System Materials

Item	Material
Pressure Pipe	SA533, Grade B, Class 2
Liner	SB-409 Alloy 800H
Insulation	Al_2O_3 and SiO_2 (Saffil)

3.4.3 Bases for Structural Design

3.4.3.1 Design Basis

The HGDs are completely enclosed within the piping of the HPB. The design is such that the driving force for the cooling flow in the annulus between the Pressure Pipe and the HPB (the PCU compressor section or, alternatively, the CCS circulator) is the same as that driving the flow of hot helium within the liner. Since the resistance of the cooling flow path is lower, cooling flow occurs preferentially and the Pressure Pipe remains under external pressure for all conditions in which there is forced circulation within the HGD. With the exception of the Turbine Outlet HGD, the differential pressures across the Pressure Pipes are modest (Table 10). During events in which forced circulation is lost (pressurized or depressurized conditions), the pressure external to the Pressure Pipe will rapidly decline to equilibrium with the internal pressure in the liner.

The design of the HGDs is such that even large failures in the Pressure Pipes would not result in exceeding the design basis of the outer HPB. A large failure in the Turbine Outlet Pressure Pipe would result in collapse of the Brayton cycle. In the case of the remaining HGDs, the limiting condition resulting from a large Pressure Pipe break would be diversion of the annular cooling flow to the main helium transport path and loss of annular flow downstream. In this case, the downstream insulation alone is sufficient to protect the HPB.

The pressure-retaining Pressure Pipes are designed in accordance with ASME Section III, Subsection NC. As is the case with the HPB pipes, the temperatures seen by the Pressure Pipes are modest, and the ASME Section III, Subsection NC Code directly applies. The modest temperatures also result in low differential expansion and associated thermally induced loads. Transient conditions within the licensing basis typically result in reduced flow and collapse of the differential pressure across the Pressure Pipes, thus reducing loads. Since the Pressure Pipes are the load-carrying element of the HGDs, they must be designed to accommodate the SSE. Spacers are attached to the external surface of the HGDs for this purpose. An exception is the

Reactor Outlet HGD, which has one internal support and is evaluated to have adequate stiffness without the requirement for additional seismic spacers.

As noted in Table 10, the highest nominal temperature seen by the HGDs is 900 °C at the reactor outlet. Preliminary estimates of the peak temperatures during transients indicate a somewhat higher peak temperature of ~920 °C. However, this must be confirmed through further analysis. The design of the HGD Liners and the V-shaped spacers between the liners and the Pressure Pipes has been developed on a first principles basis through design and analysis. The design process utilizes the data provided for Alloy 800H in draft KTA Standard 3221.1, issued by the Kerntechnische Ausschuβ (the Nuclear Technical Committee of the German Nuclear Safety Standards Commission). It is noted that a potential alternative is ASME Section III, Subsection NH for the intermediate temperature pipes (all but the Reactor Outlet/CCS Inlet HGD).

While Alloy 800H is incorporated into a number of ASME Code and German DIN standards, the ASME Code (Section III, Subsection NH or Section VIII in this case) does not presently provide for its application at temperatures above 760 °C. However, there is sufficient data available to justify raising this temperature to 900 °C or higher, and a review of Alloy 800H stress allowables in ASME Section III, Subsection NH and an extension of time-dependent allowable stresses to 900 °C is presently being pursued as one task in a cooperative agreement between DOE and ASME. Alloy 800H is presently covered in draft KTA 3221.1 at temperatures up to 1 100 °C; hence the selection of this German standard as the basis of the analyses of the HGD Liner.

As noted earlier, the design and materials of choice for the HGDs are based on prior developments within the German HTR program. As part of that development, the HGD design underwent successful qualification testing in Germany at 900 °C to 950 °C for periods up to 15 000 h. The velocity limit applied in the PBMR design is consistent with that earlier testing. Wind tunnel testing was also used in the course of design verification to evaluate flow-induced loads on the liner.

Alloy 800H may suffer some degradation in ductility during exposure at 900 °C. However, this ductility loss should be relatively small and acceptable, especially as there is essentially zero loading across the liner during operation.

Finally, there is no exposure to high radiation of any of the components (outer and inner pressure pipes, liner, or insulation) of the HGDS. Therefore, limit curves for irradiation are not applicable for these components.

3.4.3.2 Fabrication and inspection considerations

Fabrication of the HGDs will be in accordance with ASME Section III, Subsection NC (Pressure Pipes) and design by analysis (liners). Quality control provisions will be employed during the manufacturing, installation, and commissioning processes to ensure that the HGDS functions as per the design.

3.4.3.3 Reliability and integrity management

The following operational considerations relevant to Reliability and Integrity Management are noted:

1. The Pressure Pipes operate at modest temperatures and are externally loaded. Additional internal support is provided by the internal HGD Liner and Liner Supports. The likelihood of gross Metallic Support Pipe rupture or collapse will, therefore, be very low.
2. Leaks resulting from failures of the Pressure Pipes will result in flow from the cooler helium in the external annulus into the high-temperature helium stream within the liner.
3. Temperatures and pressures will be monitored in the hot ducts and annular cooling passages during operation as a means of identifying significant failures of the HGDS pressure boundary (the Pressure Pipes).
4. Inspection of high-temperature liner components during scheduled maintenance outages is to be evaluated as part of the overall RIM strategy.
5. Since the pressure gradient is from the outside to the inside of the HGDs, impingement of high-temperature helium on the HPB would require both complete circumferential failure and offset of the Pressure Pipe. Offset of the Pressure Pipe would, in turn, require coincident failure of the liner and insulation.

3.4.4 Incremental Development and Qualification

Efforts to extend appropriate ASME Code sections to include stress allowables for Alloy 800H are presently underway as a cooperative effort between ASME and DOE. It would be desirable to extend the present temperature range to at least 950 °C. This would benefit both the design and licensing processes.

The Reactor Outlet HGD (hot duct) will be subjected to qualification testing using representative test sections at appropriate conditions. This is in addition to the qualification testing of the hot duct performed in the German HTR program (refer to Section 3.4.3).

3.4.5 Assessment of Licensing Basis

3.4.5.1 Licensing basis events and design basis accidents

Referring to Table 8, the DBEs that are significant for the HGDS are those which rely on active cooling for decay heat removal, either via the PCU or CCS. These include DBEs 2b, 3a, 3b, 7a, 7b, 11a and 11b [2]. Since neither the PCU nor CCS is assumed to be available in the case of the corresponding DBAs, the HGDS plays no role in those postulated events. As indicated above, failures of the HGDS resulting in consequent failure of the HPB are evaluated to be below the threshold for DBEs and DBAs.

3.4.5.2 Assessment

The HGDS design applies the general PBMR philosophy of utilizing established design and materials codes, where available, and applying conservative first principles design and analysis, plus appropriate testing, where existing codes and standards are inadequate or non-existent. In particular, the HGDS design with active cooling allows the HPB piping to be designed and operated within the normal limits of ASME Section III, Subsection NB. It further allows use of the conventional ASME Section III, Subsection NC Code within its normal limits as the basis for the internal Pressure Pipes. The use of these codes has been previously accepted by NRC in other nuclear installations.

At the temperatures seen by the Reactor Outlet/CCS Inlet Liner (up ~920 °C), there is no present coverage by the ASME Code. For this reason, the PBMR approach is to use conservative first principles design and analysis, with Alloy 800H materials properties based on the German KTA Code 3221.1. The design will be supported by appropriate testing. As with the other HGD Liners, there is no pressure-retention function, and the loads on this structure are very small with the exception of the short duration loads of the SSE, for which it is designed. The resulting design is a scale up of the design developed and tested in Germany for high-temperature process heat applications.

The HGDS is comprised of passive components with no active functions. Only the Pressure Pipes are subjected to external pressure under modest temperature conditions, and rupture of these pipes in a manner that would result in the temperature limits of the HPB being exceeded is not expected under any condition within the design basis. The relative resistances of the flow paths, which drive the flow of cooler helium gas within the annulus between the Pressure Pipe and the HPB, plus the design of the HGD insulation, ensure that both the Pressure Pipes and the surrounding HPB piping remain within the design and code allowable temperature limit for the selected materials. Significant failures of the Pressure Pipes will be detected by monitoring of the temperatures and flows within those circuits. The HGDS is not credited in DBA conditions, and, therefore, its integrity is not required for safe shutdown of the plant during such postulated accident conditions.

On the basis of the information summarized in this section, PBMR asserts that the design bases, materials selections, and applied codes and standards and/or first principles analyses and testing provide an adequate basis for demonstrating regulatory compliance in the case of the materials aspects of the HGDS design.

3.5 REACTIVITY CONTROL AND SHUTDOWN SYSTEM

3.5.1 Reactivity Control System

3.5.1.1 Key functions

The primary function of the RCS during normal operation is reactivity control. The RCS is used for hot shutdown, reactor outlet temperature adjustment, and trimming. Inserting or withdrawing the absorber (control) rods within channels in the SR of the CSC at a designated speed performs this function. The capability to hold the control rods steady in any position in their entire range of travel also exists. Another function of the RCS is to limit the potential rate of reactivity increase by limiting the extraction speed of the control rods. Specific RCS safety functions, in addition to ensuring hot shutdown and limiting the withdrawal speed of the control rods, are to provide the capability to identify a stuck control rod, and to ensure control rod insertion on power failure. The RCS is assumed to function during DBAs and is, thus, designated to be Safety Related.

3.5.1.2 Design overview

Each control rod consists of six segments containing sintered B_4C absorber material in the form of rings between two coaxial cladding tubes with an outer diameter of 100 mm. There is sufficient gap between the cladding tubes and the B_4C rings to prevent constraint forces from arising due to radiation-induced swelling of the B_4C. Pressure-equalizing openings expose the B_4C to the coolant gas to avoid any pressure buildup. The total length of each control rod is 6.95 m with an absorber length of 6.5 m.

The individual segments are held together by articulated joints and suspended from one another to form a complete rod. This minimizes torsion caused by asymmetric temperature profiles. Each segment joint is held in place by mechanical stops.

The RCS consists of both control rods and shutdown rods. The configuration for both the control rods and the shutdown rods is identical, except with regard to chain length. The control rods form a bank consisting of 12 identical rods and the shutdown rods form a second bank of 12 identical rods. These two banks are arranged with alternating rods above the control rod channels in the SR. The layout of the construction of the RCS is illustrated in Figure 15.

The rods are freely suspended in the SR channels by chains. Guides are not necessary between the rods and the SR channels, which are sleeved with graphite. A large annular gap (25 mm) exists between the control rods and the sleeves to avoid jamming of the rods. The rods are cooled inside and outside by a stream of gas diverted from the reactor inlet stream to remove the heat generated in the absorber.

The chains link the control rods to the CRDMs that are used to raise and lower the control rods in the control rod channels and to hold them at any position in their travel range (refer to Figure 16). The CRDMs are installed above the core and are integrated into the RPV head.

The essential parts of the drive mechanisms are:

- A link chain connecting the drive mechanism and absorber. When the rod is raised, the chain is stored in a loose pile in a container, and it is drawn out of this container when the rod is lowered.
- An electric motor drive to hold the control rod in position or move the rods up or down.
- A gearbox, with a reducing bevel gear and spur gear between the chain sprocket and drive motor.
- An eddy current brake with permanent magnets to limit drop velocity in the case of a SCRAM.
- A shock absorber to absorb the kinetic energy of the falling rod and in the case of a SCRAM.
- A rod position indicator, plus proximity sensors for the upper and lower limit positions to indicate rod full-in and full-out positions.

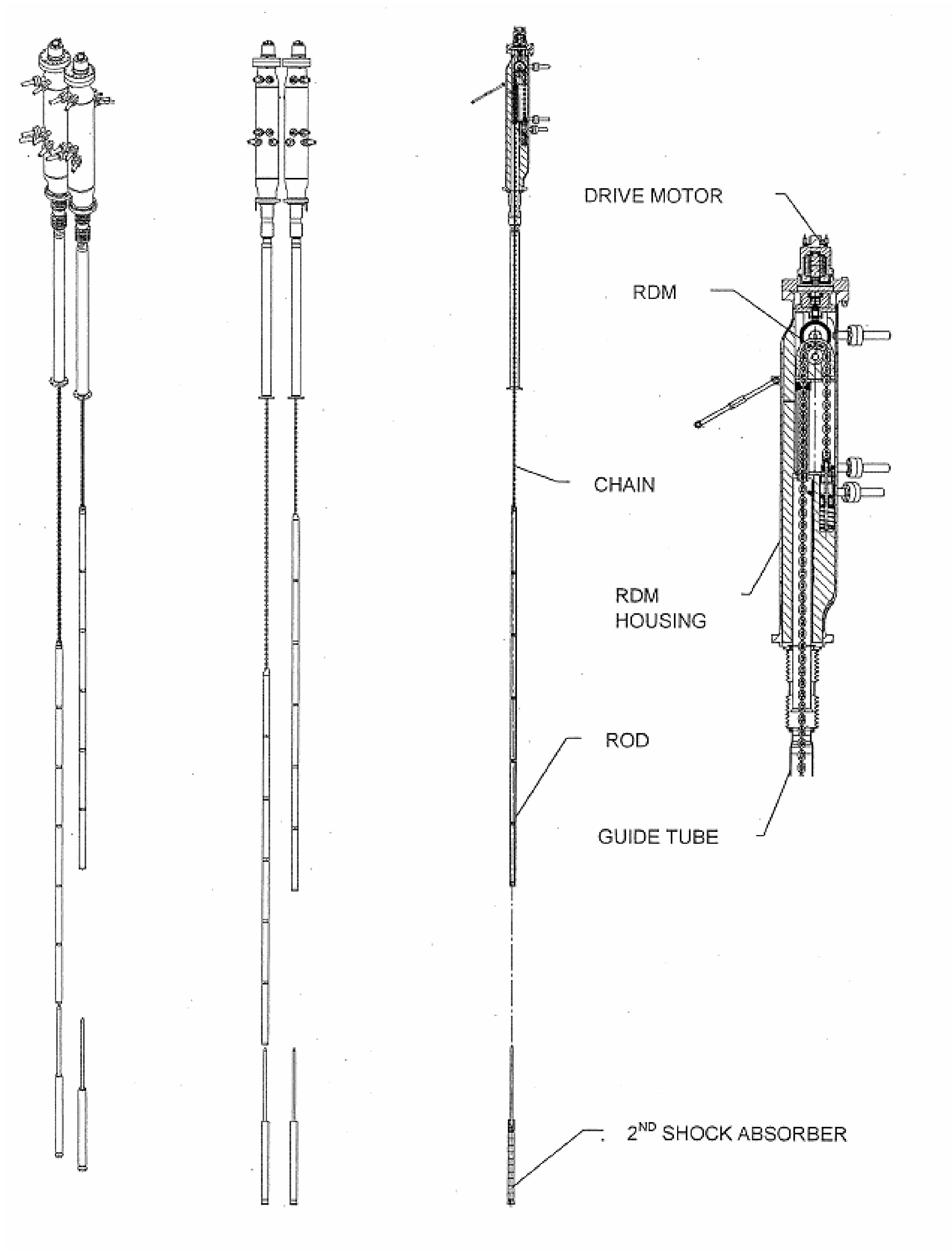

Figure 15: Construction Layout of the Reactivity Control System

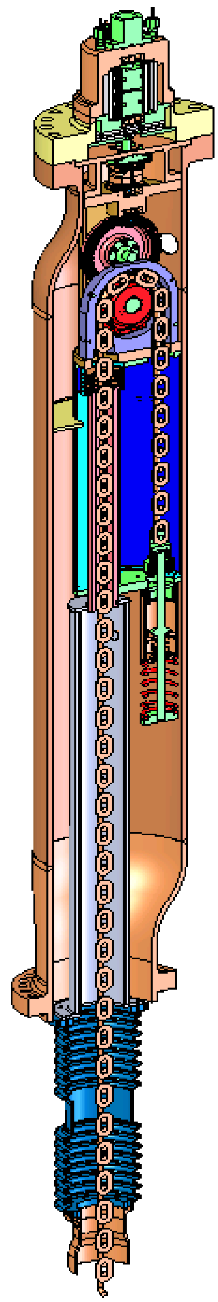

Figure 16: Reactivity Control System Control Rod Drive Mechanism

The control rod is moved up and down in the SR columns by the electric motor through the intervening gearbox, chain sprocket and link chain. The speed of insertion or withdrawal is limited by the design of the motor. The withdrawal speed is limited to 10 mm/s.

Parameters and materials related to the RCS are shown in Table 12. The chains are constructed of Alloy 625, a Ni-base material; the absorber clad tubes are of FeNi-base material Alloy 800H.

Table 12: Control Rod Parameters and Materials

Parameter/Material	Value
Control rod absorber length	6 500 mm
Control rod absorber diameter	100 mm
Control rod maximum travel	Approximately 6 500 mm (control rods) Approximately 10 200 mm (shutdown rods)
Control rod normal speed	10 mm/s
Chain material	SB-446 Alloy 625
Rod cladding tubes	SB-409 Alloy 800H
Rod absorber material	B_4C rings
Secondary shock absorber	Alloy 800H

3.5.1.3 Bases for structural design

Temperatures seen by the Alloy 800H cladding tubes of the control rods during normal operation will range from nominally 500 °C (top) to 700 °C (bottom). However, cladding temperatures could reach in excess of 1 000 °C under the long-term passive heat removal conditions assumed for DBAs. The Alloy 800H secondary shock absorber temperature will mimic that of the bottom sections of the control rods. As was discussed in Section 3.4 (Hot Gas Duct System), although Alloy 800H is incorporated into a number of ASME Code and German DIN standards, there is no ASME Code basis for design with Alloy 800H at temperatures > 760 °C. This is so even though sufficient data exists to extend ASME Code usage to higher temperatures. Further, Alloy 800H is covered in the draft KTA 3221.1 code at temperatures to 1 100 °C.

The Alloy 800H cladding tubes of the control rods are to be designed by analysis from first principles, supported by appropriate test data. Design criteria to be adhered to include < 0.4% creep strain during full-life under normal operation and < 0.1% strain during all off-normal events (taking into account the effects of irradiation on tensile ductility). Materials data utilized will come from KTA standards, suppliers' data, extrapolations from ASME data, and additional testing, if necessary.

Although Alloy 800H may, as is common with most FeNi-base and Ni-base alloys, suffer some degradation in ductility by thermal aging during its service life as the absorber rod cladding, its ductility loss should be relatively small and acceptable. Also, Alloy 800H in this application will see End-of-Life (EOL) fluences of up to 7×10^{21} n/cm^2 (E > 0.1 MeV). A significant amount of irradiation data for Alloy 800H has been collected in the German HTR program. Some of their early results were published in [6]. Considerably more irradiation data on Alloy 800H has been generated by Forschungszentrum Jülich in Germany. Examination of the existing radiation hardening (strength increase and ductility decrease) data for Alloy 800H gives confidence that its EOL ductility will be sufficient to meet the design criteria. Further, these Alloy 800H components can be replaced, if necessary.

The Alloy 625 chain will nominally operate at 500 °C, but may reach temperatures up to 800 °C for DBAs. It was chosen as the chain material rather than Alloy 800H, because of its greater high temperature strength. Alloy 625 is incorporated in ASME Section II, Parts B and D. The situation of Alloy 625 with regard to thermal and radiation effects is also very similar to that for Alloy 800H, but irradiation levels should be slightly lower. The chains can be replaced, but the likelihood of replacement being required is even less than for the control rods. As with the Alloy 800H components, above, the qualification approach for the Alloy 625 chains is design by analysis, supported by appropriate test data.

3.5.2 Reserve Shutdown System

3.5.2.1 Key functions

The primary function of the Reserve Shutdown System (RSS) during normal operation is to provide cold shutdown capability during maintenance operations. Dropping thousands of SASs into channels in the centre reflector of the CSC accomplishes this function.

3.5.2.2 Design overview

The RSS as a whole consists of eight independent units. A schematic of the system is shown in Figure 17. Eight storage container units are mounted on a support structure in the RPV upper cavity. The storage containers are used to store the SASs (B_4C in a graphite matrix) in a ready-to-be-inserted state. The storage containers receive SASs from the SAS feeder bin. From the storage containers, the SASs flow through the guide tubes into the borings inside the CSC centre reflector. Fail-open valve actuators open the valves at the bottom of the storage containers to allow the release of SASs from the storage containers into the centre reflector borings. Cross sections for SAS passage are dimensioned to prevent bridging of the spheres. For reasons of plant availability, the actuators have an uninterruptible emergency power backup.

The SAS Guide Tubes are used to direct the SASs from the storage containers into the borings inside the CSC centre reflector. The Guide Tubes are designed to accommodate the expansion difference between the container support structures, which are attached to the RPV, and the CBA, which fixes the location of the CSC. Prior to taking the reactor back to power, the SASs are removed from the centre reflector through a Sphere Discharge Pipe to a Discharge Vessel, and thence to the Gas Transport System.

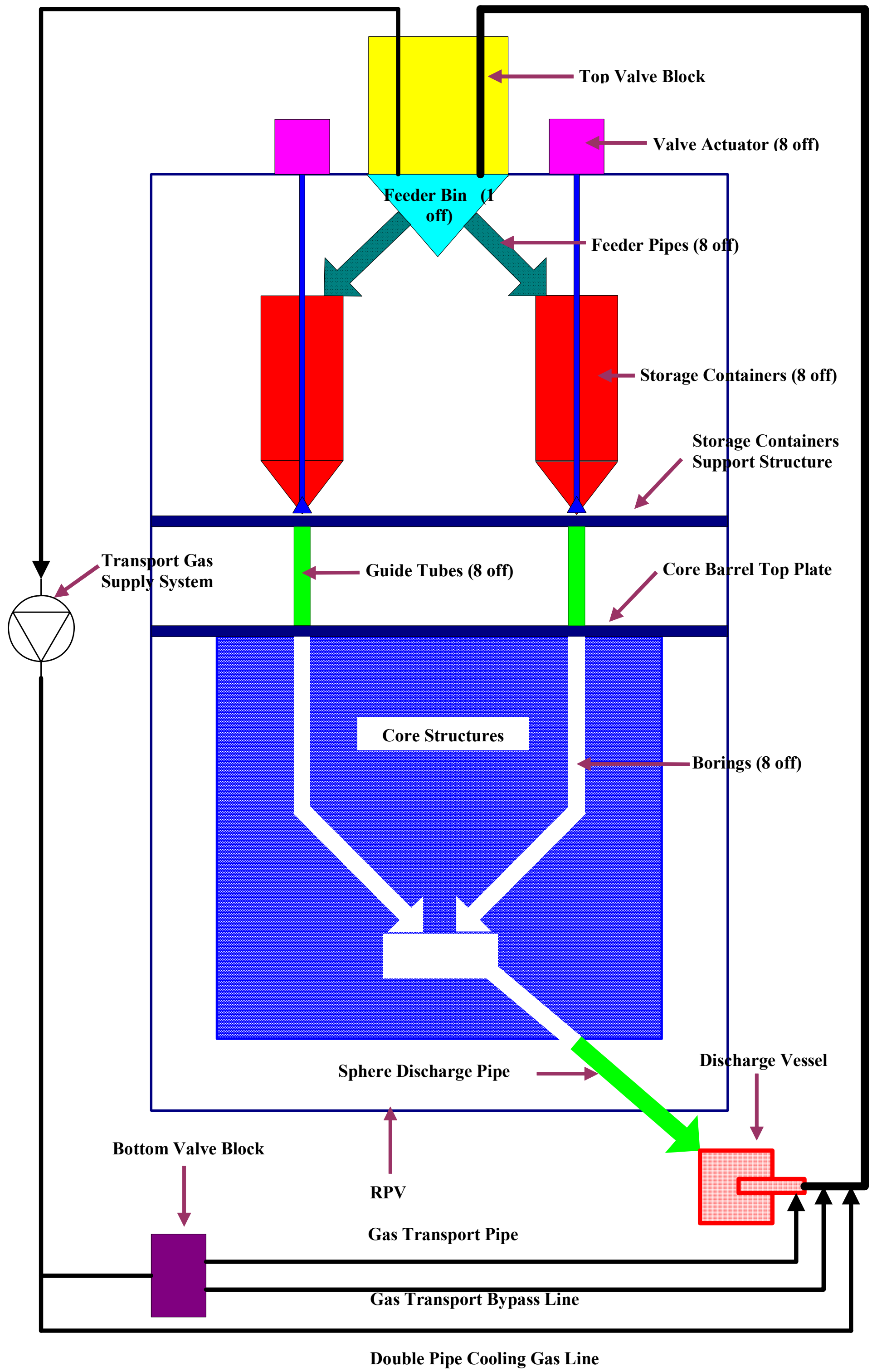

Figure 17: Schematic of the Reserve Shutdown System

As noted in Section 3.1.2.4, only the very high-temperature portions of the RSS will be discussed here. These are the Sphere Discharge Pipes shown in green at lower right in Figure 17, and the Discharge Vessel Inner Assembly shown outside the RPV at lower right in the same figure. The design of all of the other components falls under Subsections NB, NC or NG of ASME Section III, and meets the Section III requirements relative to materials and maximum use temperatures.

3.5.2.3 Basis for structural design

3.5.2.3.1 Sphere discharge pipe

The Sphere Discharge Pipe provides for movement of the SAS absorber spheres from the borings in the CS to the Discharge Vessel. At its upper end, it is attached to the CS and, at the lower end, to a nozzle on the RPV via flanges of Alloy 625. The component consists of concentric tubes of Alloy 800H, the inner tube serving as a liner to transport the SASs and the outer tube serving as a pressure boundary, which sees the differential pressure between the reactor inlet and outlet. The annulus between the tubes is insulated and both tubes contain Alloy 800H bellows to allow for differential thermal expansion between the CS and the RPV. The design allows for contact with SAS absorber tubes at 900 °C.

As was discussed in Section 3.5.1.3, although Alloy 800H is incorporated into a number of ASME Code and German DIN standards, there is no ASME Code basis for design with Alloy 800H at temperatures > 760 °C. This is so even though sufficient data exists to extend ASME Code usage to higher temperatures. Further, Alloy 800H is covered in the draft KTA 3221.1 Code at temperatures to 1 100 °C.

The Alloy 800H pipes and bellows for the Sphere Discharge Pipe structure are to be designed by analysis from first principles, supported by appropriate test data. Materials data utilized will come from KTA standards, suppliers' data, extrapolations from ASME data, and additional testing, if necessary.

Although the Alloy 800H pipes may suffer some degradation in ductility by thermal aging during their service life, ductility loss should be relatively small and acceptable. This, as well as radiation damage considerations, is discussed in Section 3.5.1.3. Some portions of the Alloy 800H pipes may experience an EOL fluence of up to 5×10^{18} n/cm^2 (E > 0.1 MeV), which is considered insignificant for austenitic materials.

The flanges will also operate at a maximum temperature of 900 °C. Alloy 625 was chosen as the flange material as opposed to Alloy 800H, because of its higher strength. The ASME Code recognizes Alloy 625 in Section II on materials and in other sections up to 427 °C. The qualification approach for the flanges is design by analysis, supported by appropriate test data, as discussed above. The situation of Alloy 625 with regard to thermal and radiation effects is very similar to that for Alloy 800H. Irradiation levels for the flanges should be a maximum of 5×10^{18} n/cm^2 (E > 0.1 MeV).

3.5.2.3.2 Discharge vessel

The SAS Discharge Vessel consists of the vessel itself plus a fluidizing gas pipe assembly, a level indicator, and insulation. The primary purposes of the Discharge Vessel are to fluidize the SAS bed in a controlled manner, and to allow transport of the absorber spheres to the Feeder Bin serving the Storage Containers. The vessel itself is designed to ASME Section III Subsection NB or NC within normal temperature limits. The vessel, level indicator, and insulation are not considered further in this document.

The remaining component is the fluidizing gas pipe (also called the inner vessel assembly) and consists of plate (Alloy 800H), pipe (Alloy 625), and tube (Alloy 625) components. Under some circumstances these materials could be exposed to helium gas at 900 °C, but there is considerable uncertainty as to the metal temperatures that would result. The Alloy 800H and Alloy 625 materials were chosen because of the possibilities for such exposure, and the fluidizing gas pipe will be designed by analysis, supported by appropriate test data. The suitability of these materials and the bases for their qualification at the temperatures of interest are discussed above.

3.5.3 Incremental Development and Qualification

As previously discussed in Section 3.4.5, it would be desirable to initiate efforts to extend appropriate ASME Code sections to include stress allowables for Alloy 800H at temperatures up to nominally 950 °C. This could benefit both the design and licensing processes for the Sphere Discharge Pipe. As was also noted earlier, a joint ASME-DOE task force has been considering such an option. Also, efforts to obtain the full suite of Alloy 800H data from Forschungszentrum Jülich would provide additional data for design and licensing.

Collection and analysis of thermal stability, compatibility, and radiation effects data for Alloy 625 would also be beneficial.

3.5.4 Reliability and Integrity Management

The nominal design basis is to utilize the RCS control rod absorber assemblies and chains for the full lifetime of the PBMR. However, these components are replaceable during normal maintenance outages.

3.5.5 Assessment of Licensing Basis

3.5.5.1 Licensing basis events and design basis accidents

The DBEs and associated DBAs listed in Table 8 are applicable to the RCSS. Since the RCSS is relied upon for DBAs, it has been designated as Safety Related in accordance with [3].

3.5.5.2 Assessment

Temperatures seen by the Alloy 800H cladding tubes and links of the RCS control rods during normal operation will range from nominally 500 °C (top) to 700 °C (bottom). However, cladding temperatures could exceed 1 000 °C under the long-term passive heat removal conditions assumed for DBAs. The Alloy 625 chains that support the control rods will nominally operate in the range of 500 °C, but like the control rod components, may see temperatures in excess of 800 °C during DBAs.

The Sphere Discharge Pipe Liner and flange and the Discharge Vessel internal (non-structural) components will see temperatures up to 900 °C when in operation to remove the SASs. The materials used for these components are also Alloy 800H and Alloy 625.

As noted above, these components will be designed by analysis on a first principles basis, using the draft German KTA Standard 3221.1 for Alloy 800H, comparable data for Alloy 625, and supported by test data, as appropriate. Conservative analysis, supplemented by testing, as required, will provide the basis for demonstrating the reliability necessary for regulatory compliance.

4. ISSUES FOR PREAPPLICATION RESOLUTION

4.1 PBMR BASIS FOR MATERIALS SELECTIONS

As noted in Section 3.1.1, PBMR applies the following general approach for design and material selection, which is summarized in order of preference:

- use materials within the limits of a code or standard that the NRC has accepted; or
- use materials within the limits of a code or standard that has been accepted by a standards body but which the NRC has not yet accepted; or
- use materials that are not incorporated in a code at this time and design from first principles with appropriate supporting qualification programs.

In the following tables, the Qualification Approach for a number of materials is noted to be 'Use NRC Accepted ASME Specification' or 'Use industry-accepted ASME Specification'.

Table 1: Metallic Materials for PBMR Primary Helium Pressure Boundary Components

Table 2: Core Barrel Assembly Metallic Materials

Table 3: Hot Gas Duct System Metallic Materials

Table 4: Metallic Materials for the Reactivity Control and Shutdown System

Table 5: Metallic Materials for the Helium Inventory Control System

Table 6: Metallic Materials for the Fuel Handling and Storage System

The question for which a response is requested is the following:

Does the NRC agree that the cited specifications provide a reasonable basis for PBMR to develop the design certification application for the components in question?

4.2 APPLICATION OF CODE CASE N-499-2 FOR THE REACTOR PRESSURE VESSEL

Code Case N-499-2 is an ASME-approved addition/modification to Section III, Subsection NB that provides for limited operation above 371 °C during Service Level B, C, and D events. Based on the discussions provided in Section 3.2, PBMR believes that ASME Section III, Subsection NB, together with Code Case N-499-2, is an appropriate and conservative design basis for the PBMR RPV.

The question for which a response is requested is the following:

- Does the NRC agree that ASME Section III, Subsection NB, supplemented by Code Case N-499-2, provides a reasonable basis for PBMR to develop the design certification application for the RPV?

4.3 APPLICATION OF CODE CASE N-201-5 FOR THE CORE BARREL ASSEMBLY

Code Case N-201-5 is an ASME-approved addition/modification to Section III, Subsection NG that provides for the design and construction of core support structures for temperatures above 371 °C during both normal operation and duty cycle events. The scope of N-201-5 includes Type 316 Stainless Steel, which has been selected for the CBA. Based on the discussion of Section 3.3 and Section 3.4, PBMR believes that the application of ASME Section III, Subsection NG, together with Code Case N-201-5, is an appropriate and conservative basis for the design of the CBA.

The question for which a response is requested is the following:

- Does the NRC agree that ASME Section III, Subsection NG, supplemented by ASME Code Case N-201-5, provides a reasonable basis for PBMR to develop the design certification application for the CBA?

4.4 SELECTION AND QUALIFICATION OF MATERIALS FOR HIGH TEMPERATURES

The bases for selection and qualification of the materials for the Reactor Outlet HGD Liner and the high-temperature components of the RCSS were described earlier in Sections 3.4.3 and 3.5.3, respectively. This section will briefly review these bases and provide PBMR's perspective regarding their adequacy. In all cases, the question for which a response is requested is the following:

- Does the NRC agree that the approach outlined herein provides a reasonable basis for PBMR to develop the design certification application? Note that the issues to be resolved are limited to selection and qualification of materials, as opposed to justifying the specific designs.

4.4.1 Hot Gas Duct Liners

The issue to be addressed by PBMR is the adequacy of the liner material for the high-temperature HGDs, which channel helium at temperatures up to 900 °C during normal operation (preliminary estimates up to ~920 °C during transients), with the highest temperatures being from the Reactor Outlet to the Turbine or CCS inlet. The outer Pressure Pipes, which are insulated from the liners, operate at low temperatures and are designed in accordance with ASME Section III, Subsection NC. The specific design and materials of choice for the HGDs are based on a system developed in the German HTR program. Test sections of this design underwent successful qualification testing at 900 °C to 950 °C for periods up to 15 000 h.

The HGD Liner material (SB-409 Alloy 800H) selected in the German design is incorporated into a number of ASME Code and German DIN standards; the ASME Codes (e.g., Section III Subsection NH and Section III Code Case N-201-5) typically do not provide for application of the material at temperatures > 760 °C. However, Alloy 800H is covered in the draft German KTA 3221.1 Code at temperatures to 1 100 °C. Sufficient data is available to justify raising the allowable temperature to at least 900 °C in relation to ASME Section III, and a joint DOE/ASME task has been ongoing to collect and assess this data. Extending the appropriate ASME Code sections to these higher temperatures would benefit both the design and licensing processes.

PBMR considered materials other than Alloy 800H for use as the HGD Liner. Among these were Ni-base alloys Hastelloy X and Alloy 625 and CFRC. However, Alloy 800H was the choice based on a combination of factors, including lower cost, sufficient strength and environmental compatibility, real life service experience, and the German HTR design and testing work.

The Reactor Outlet HGD Liner will be designed by analysis from first principles and will be subject to qualification testing. The loads imposed on the liner during normal service are negligible, and the structural design is largely based on thermal considerations, plus the SSE. Materials data utilized will come from KTA standards, suppliers' data, extrapolations from ASME data, and additional testing, if necessary. The hot duct will be subject to qualification testing using representative test sections at appropriate conditions. Temperatures and pressures will be monitored in the hot ducts during operation.

4.4.2 Reactivity Control System Control Rod Cladding Tubes and Chains

The issue to be addressed by PBMR is the adequacy of the materials for the high-temperature components of the RCS. Temperatures seen by the Alloy 800H cladding tubes of the control rods during normal operation will range from nominally 500 °C (top) to 700 °C (bottom). However, cladding temperatures could reach in excess of 1 000 °C for DBAs. The temperature of the Alloy 800H secondary shock absorber will be the same as that of the bottom sections of the control rods. The off-normal events above will result in a maximum temperature exceeding that covered in the ASME Code. However, as noted in Section 4.4.1, Alloy 800H is covered in the draft KTA 3221.1 Code at temperatures up to 1 100 °C.

The Alloy 800H cladding tubes are to be designed by analysis from first principles, supported by appropriate test data. Design criteria to be adhered to include < 0.4% creep strain during full-life under normal operation and < 0.1% strain during all off-normal events. Materials data utilized will come from KTA standards, suppliers' data, extrapolations from ASME data, and additional testing, if necessary.

Although Alloy 800H may, as is common with such alloys, suffer some degradation in ductility by thermal aging during its service life as the absorber rod cladding, its ductility loss should be relatively small and acceptable. Also, Alloy 800H in this application will see EOL fluences of up to ~7×10^{21} n/cm^2 (E > 0.1 MeV). A significant amount of irradiation data for Alloy 800H has been collected in the German HTR program and elsewhere. Examination of the existing radiation hardening (strength increase and ductility decrease) data for Alloy 800H provides confidence that its EOL ductility will be sufficient to meet the design criteria. Further, these Alloy 800H components could be replaced if necessary.

The Alloy 625 chain will operate at nominally 500 °C, but may exceed 800 °C for DBAs. It was chosen as the chain material as opposed to Alloy 800H, because its strength is significantly greater. Materials such as Alloy 617 and Hastelloy X were considered for this application, but offered no clear advantage. The ASME Code recognizes Alloy 625 only in Section II on materials. The qualification approach for the chains is design by analysis, supported by appropriate test data. As for the Alloy 800H components, the design criteria are < 0.4% creep strain during full lifetime and < 0.1% strain total during off-normal events. The situation of Alloy 625 with regard to thermal and radiation effects is also very similar to that for Alloy 800H, but irradiation levels should be an order of magnitude lower. The chains can be replaced, but the likelihood of replacement is even less than for the control rods.

In summary, the high-temperature alloys selected for the control rod cladding and shock absorbers and the control rod chains are suitable to their application in terms of strength and environmental resistance. The design approach selected should provide effective components of appropriate conservatism.

4.4.3 Reserve Shutdown System Materials

The issue to be addressed by PBMR is the adequacy of the materials for two of the components that make up the RSS not covered under Section III, Subsection NB or Section III, Subsection NG. These are the Sphere Discharge Pipe and the Discharge Vessel Inner Assembly. Both have the potential for contact with helium at 900 °C, and are therefore designed with high-temperature FeNi-base (Alloy 800H) and Ni-base (Alloy 625) materials. The situation of these materials with respect to ASME Code acceptance and thermal and irradiation performance has been discussed in Sections 4.4.1 and 4.4.2 for Alloy 800H and in Section 4.4.2 for Alloy 625. The design approach selected for the high-temperature RSS components is essentially that described in Section 4.4.1 for the HGD Liner and in Section 4.4.2 for RCS control rod cladding tubes and chains.

5. PREAPPLICATION OUTCOME OBJECTIVES

The objective of this paper is to obtain NRC agreement, in principle, with PBMR's approach to the selection and qualification of metallic materials. Specifically, PBMR would like the NRC to agree with the following statements, or to provide an alternate set of statements with which they agree:

1. The PBMR approach to the selection and qualification of materials that is outlined in Section 3.1.1 provides a reasonable basis for developing the design certification application.
2. For those materials listed in Table 1 through Table 6, and for which the 'Qualification Approach' is 'Use NRC-accepted ASME Specification', the cited specifications provide a reasonable basis for developing the design certification application.
3. ASME Section III, Subsection NB, supplemented by Code Case N-499-2 provides a reasonable basis for PBMR to develop the design certification application for the RPV.
4. ASME Section III, Subsection NG, supplemented by Code Case N-201-5 provides a reasonable basis for PBMR to develop the design certification application for the CBA.
5. The use of first principles analysis, based on the draft KTA 3221 standard and supplemented by appropriate testing, provides a reasonable basis for developing the design certification application for the Alloy 800H HGD Liners.
6. The use of first principles analysis, based on the draft KTA 3221 standard and supplemented by appropriate testing, provides a reasonable basis for developing the design certification application for the Alloy 800H RCSS components.
7. The use of first principles analysis, based on existing data and supplemented by appropriate testing, provides a reasonable basis for developing the design certification application for the Alloy 625 RCSS components.

6. REFERENCES

[1] ‘Probabilistic Risk Assessment Approach for the Pebble Bed Modular Reactor’, document number 039144, Rev. 1, PBMR, 13 June 2006.

[2] ‘Licensing Basis Event Selection for the Pebble Bed Modular Reactor’, document number 040251, Rev. 1, PBMR, 30 June 2006.

[3] ‘Safety Classification of Structures, Systems, and Components for the Pebble Bed Modular Reactor’, document number 043553, Rev. 1, PBMR, 24 August 2006.

[4] ‘Defense-in-Depth Approach for the Pebble Bed Modular Reactor’, document number 043593, Rev. 1, PBMR, 11 December 2006.

[5] ‘Technical Description of the PBMR Demonstration Power Plant’, document number 016956, Rev. 4, 14 February 2006.

[6] ‘Irradiation Behavior of High-Temperature Alloys for High-Temperature Gas-Cooled Reactor Service’, **Nuclear Technology,** Vol. 66, No. 3, September 1984.