Xe-100 Licensing Topical Report
Graphite Core Assembly
Material Qualification and Design Methodologies

Document ID Number	**:**	**009380**
Configuration Classification	**:**	**XE00-N-RX-CGS-STG-GL-L**
Revision	**:**	**1**
Security Classification	**:**	**Non-Proprietary**
Status	**:**	**Approved**
Date Created	**:**	**29-Oct-2024**
Project	**:**	**Xe-100**

E-SIGNATURES

Document Approval Signees

Action	Designation	Name	Signature	Date
Preparer	Manager, Licensing	D. Kalinousky	Maintained in Teamcenter	October 29, 2024
Reviewer	Project Manager, Graphite Program	Z. Burns	Maintained in Teamcenter	October 30, 2024
Reviewer	Materials Engineer, Graphite Engineering	S. Baylis	Maintained in Teamcenter	October 30, 2024
Reviewer	Manager, Mechanical Engineering	J. Roll	Maintained in Teamcenter	October 30, 2024
Reviewer	Mechanical Engineer	FJ van Zanten	Maintained in Teamcenter	October 30, 2024
Approver	Director, Licensing	S. Vaughn	Maintained in Teamcenter	October 30, 2024

SYNOPSIS

The Xe-100 is a Generation IV Advanced Reactor based on pebble bed High-Temperature Gas-Cooled Reactor (HTGR) technology, which utilizes U.S.-developed Uranium Oxycarbide TRISO-coated fuel embedded in spherical graphite fuel elements to form fuel pebbles. The pebble graphite is the primary fuel moderator. The ‘pebbles’ are roughly the size of billiard balls, and the TRISO coating creates a pressure-tight seal around each uranium kernel which helps retain fission products and gases that are produced during operations. The fuel is contained in a Graphite Core Assembly (GCA), which serves as an additional neutron moderator/reflector and maintains the core geometry. The heat produced by the fuel is transferred to the primary heat transport fluid, helium gas, which in turn transfers heat to the secondary coolant (water/steam).

The GCA performance supports the Xe-100 Required Safety Functions (RSFs). Structures, Systems, and Components (SSCs) are designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. The safety functions of the GCA are ensured by maintaining acceptable structural integrity throughout the life of the Graphite Core Components (GCCs) of which it is comprised.

This Licensing Topical Report (LTR) describes the risk-informed and performance-based methodologies used to ensure that the GCA, as a safety-related (SR) system, will be able to perform its RSFs to maintain acceptably low dose consequences in normal operation and in response to Licensing Basis Events (LBEs) by application of the Nuclear Energy Institute (NEI) 18-04 process [1]. These methodologies include:

- The methodology for performing design analysis to ensure that the RSFs associated with the GCA are maintained. This includes both those functions related to assuring the mechanical/structural integrity of the GCA, the focus of application of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (BPVC), Section III, Division 5 [3], and other functions (e.g. control reactivity, control heat removal) not governed by the ASME BPVC. The methodology conveyed herein informs the development of content necessary for a Design Specification and method of demonstrating the acceptable mechanical/structural integrity and performance of the design in a Design Report. This LTR presents the details of augmentations, exceptions, and limitations taken for this methodology as compared to the methodology taken in the 2017 Edition of the ASME BPVC as endorsed with limitations via Regulatory Guide (RG) 1.87, Revision 2 [4].
- The methodology to develop appropriate reliability and capability performance-based targets, as well as the methodology to develop appropriate special treatment requirements as would be included in the Design Specification, and to demonstrate that the performance-based targets intended in the design are met as would be included in a Design Report.
- The material qualification methodology for obtaining material property data for as-manufactured, oxidized, and irradiated graphite to arrive at the set of data needed for the Material Data Sheet to be used for design. This includes, as applicable, the current set of data available for preliminary design analysis, current and future testing that augments the data set currently available for design

and for material specifications to be used in component procurement, as well as known or anticipated gaps or exceptions in the data expected or explicitly required by the ASME BPVC.

10 CFR 810 Export-Controlled Information Disclaimer

This document was reviewed by X-energy and determined to contain information designated as export-controlled per Title 10 of the Code of Federal Regulations (CFR) Part 810 or 10 CFR 110. This information must be withheld from disclosure. Non-export-controlled versions of this report may indicate the redaction of such information through the use of [[ ]]E.

Department of Energy Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0009040.

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

CONFIGURATION CONTROL/DOCUMENT CHANGE HISTORY

Document Change History

Rev.	Date	Preparer	Page/Section Revised	Description
1	10/29/2024	Doug Kalinousky	N/A	Initial Issue.

Table of Contents

Appendices

List of Tables

List of Figures

Abbreviations/Acronyms

Short Form	Phrase
AG	Against-Grain
AGC	Advanced Graphite Creep (INL Experimental Program)
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BDBE	Beyond Design Basis Events
BLH model	Boltzmann-Enhanced Langmuir Hinshelwood Model
BPVC	Boiler and Pressure Vessel Code
CB	Core Barrel
CFR	Code of Federal Regulations
CMTR	Certified Material Test Report
CoA	Certificate of Analysis
CTE	Coefficient of Thermal Expansion
CV	Cross Vessel
DBA	Design Basis Accident
DBE	Design Basis Event
DIN	Deutsches Insitut Normung
DTA	Damage Tolerance Assessment
DYM	Dynamic Young's Modulus
EBC	Equivalent Boron Content
FEA	Finite Element Analysis
FHS	Fuel Handling System

Short Form	Phrase
FSAR	Final Safety Analysis Report
GCA	Graphite Core Assembly
GCC	Graphite Core Component
GDC	General Design Criteria
HDG	High Dose Graphite (INL Experimental Program)
HGD	Hot Gas Duct
HPB	Helium Pressure Boundary
HTGR	High-Temperature Gas-Cooled Reactor
IGNIS	Irradiated Graphite Numerical Iterative Solver
INL	Idaho National Laboratory
ISR	Inner Side Reflector
JIS	Japanese Industrial Standard
LBE	Licensing Basis Event
LTR	Licensing Topical Report
MCR	Maximum Continuous Rating
MDS	Material Data Sheet
MTR	Materials Test Reactor
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NSRST	Non-Safety Related with Special Treatment
NST	Non-Safety Related with No Special Treatments
OE	Operating Experience
OBE	Operating Basis Earthquake

Short Form	Phrase
OL	Operating License
OLA	Operating License Application
ORNL	Oak Ridge National Laboratory
POF	Probability of Failure
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PSF	PRA Safety Function
QA	Quality Assurance
QC	Quality Control
RCCS	Reactor Cavity Cooling System
RFDC	Required Functional Design Criteria
RG	Regulatory Guide
RIM	Reliability and Integrity Management
RPV	Reactor Pressure Vessel
RSF	Required Safety Function
SGTR	Steam Generator Tube Rupture
SR	Safety Related
SRC	Structural Reliability Class
SRDC	Safety Related Design Criteria
SSE	Safe Shutdown Earthquake
SSC	Structure, System, and Component
TBC	To Be Confirmed

Short Form	Phrase
TBD	To Be Determined
TC	Thermal Conductivity
TRISO	Tri-Structural Isotropic
V&V	Verification and Validation
WE	Wigner Energy
WG	With-Grain
X-energy	X Energy, LLC

1. Introduction

1.1 Purpose

The purpose of this Licensing Topical Report (LTR) is to describe the risk-informed and performance-based methodologies, based on the Nuclear Energy Institute (NEI) 18-04 process [1], that ensure the Graphite Core Assembly (GCA) is capable of supporting particular RSFs during certain Licensing Basis Events (LBEs). The Xe-100 nuclear reactor design includes nuclear grade graphite material from graphite billets that are machined into various sized individual Graphite Core Components (GCCs) serving multiple functions in the reactor. Together the GCCs make up the GCA. The GCA performs several Required Safety Functions (RSFs) that must be met to maintain nuclear safety. These risk-informed and performance-based methodologies include a graphite material qualification methodology and a design methodology.

1.2 Scope

This LTR conveys the approaches and methods for the stress analysis of the GCA, including an assessment of degradation mechanisms and deformation limit evaluations relevant to demonstration of performance of the GCA's RSFs. It combines available codes and standards, internal X-energy methodologies, Xe-100 requirements, and operating experience (OE) from other graphite moderated reactors to form a cohesive approach to design and analysis of the GCA and its GCCs.

The material qualification methodology defines the comprehensive set of data required for design and analysis of the GCA based on operating conditions. The design methodology describes the approach to comprehensively evaluate the structural integrity of the GCA based largely on the ASME BPVC III-5, with modifications and additions. The NEI 18-04 methodology, in particular the development and application of special treatments for the GCA and its GCCs, forms the basis for a risk-informed and performance-based approach to design and qualificaiton.

The report also describes the graphite test plan for targeted testing of some specific requirements related to material and design qualification, methodology for use of NEI-18-04, and a high-level discussion of the material model used for design qualification. A specific example pertaining to the use of NEI 18-04 and a summary of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) exceptions are provided in the appendices.

1.3 Interfacing Documents

This LTR interfaces with the following Xe-100 LTR:

- Xe-100 Principal Design Criteria Licensing Topical Report [2]

1.4 Document Layout

This LTR is organized as follows:

- Section 2 provides an overview of applicable regulatory requirements and associated guidance relevant to the design and analysis of the GCA.

- Section 3 provides an overview of the GCA and describes the physical configuration, general design considerations, and environmental conditions.
- Section 4 provides an overview of the graphite material qualification methodology used to collect appropriate graphite grade-dependent material properties to support the design of the GCA. This graphite material qualification methodology is developed based on the requirements of the ASME BPVC III-5 [3].
- Section 5 provides the graphite test plan to address specific requirements identified as part of the graphite material qualification methodology in Section 4.
- Section 6 provides the methodology developed for the design by analysis of the GCA and associated GCCs to demonstrate adequate reliability in performing applicable RSFs. The design by analysis methodology addresses the anticipated loading conditions and degradation mechanisms during normal operations of the plant and during LBEs.
- Section 7 provides a description of the NEI 18-04 process and its application to the GCCs.
- Section 8 provides conclusions and describes the applicable limitations and conditions.
- Section 9 provides references and cross references.
- Appendix A compares the graphite material qualification methodology approach presented in Section 4 and the design methodology presented in Section 6 against the approach in ASME BPVC III-5 (noting exceptions, modifications, and augmentations).
- Appendix B provides an example of how the the NEI 18-04 process described in Section 7 is applied to the Inner Side Reflector.

1.5 Outcome Objectives

X-energy is requesting NRC review and approval of the graphite material qualification and design methodologies as implemented via the NEI 18-04 risk-informed performance-based methodology to support the preliminary design, and subsequent final design, and associated analyses of the GCA.

2. Regulatory Requirements and Industry Guidance

The GCA design, analyses, and associated methodologies comply with the regulatory requirements and guidance described in Section 2.1 and 2.2.

2.1 Regulatory Requirements

The following regulatory requirements are applicable to the GCA:

- Title 10 of the Code of Federal Regulations (10 CFR) Part 50 "Domestic Licensing of Production and Utilization Facilities," including but not limited to, the requirements below:
    - 10 CFR 50.34(a) describes the requirements for a preliminary safety analysis report (PSAR), which includes the identification of principal design criteria (PDC), as part of a Construction Permit Application (CPA) for a proposed facility.
    - 10 CFR 50.34(b) describes the requirements for a final safety analysis report (FSAR) as part of an Operating License Application (OLA) for a proposed facility.
    - 10 CFR 50.35 describes the requirements that would allow the Commission to issue a construction permit while certain safety features that require planned research and development activities are completed satisfactorily prior to the expected construction completion date stared in the application.
    - 10 CFR 50.43(e)(1) describes the requirements to demonstrate safety feature performance for applicants pursuing an operating license for a proposed design that is significantly different from a light-water reactor.
    - 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," establishes quality assurance requirements for the design, manufacture, construction, and operation of those SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to public health and safety.
- PDC specific to the GCA design as described in Xe-100 Principal Design Criteria Licensing Topical Report [2].
    - PDC-RFDC 26: The reactor shall be designed to include movable poisons that can insert and maintain safe shutdown during DBEs and DBAs.
    - PDC-RFDC 34: A passive system shall be designed to ensure that acceptable fuel and radionuclide release limits are not exceeded during DBEs and DBAs.
    - PDC-RFDC 70: The reactor core shall be designed such that the reactor vessel and reactor system geometry is maintained to (1) ensure geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) permit sufficient insertion of neutron absorbers to provide for reactor shutdown.

2.2 Applicable Guidance

The following regulatory and industry guidance documents are applicable to the GCA:

- RG 1.233, Revision 0, “Guidance for a Technology-Inclusive, Risk-Informed, Performance-Based Methodology to Inform the Licensing Bases and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” [7]” endorsing NEI 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development” [1] as one acceptable method for non-LWR designers to use when performing selection of licensing-basis events (LBEs); classification and special treatments of structures, systems, and components (SSCs); and assessment of defense in depth (DID).
- RG 1.253, Revision 0, “Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” [55] endorsing NEI 21-07 “Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: Content for Applicants Using the NEI 18-04 Methodology” [6] as one acceptable process for use in developing certain portions of the SAR for an application for a non-LWR construction permit or operating license.
- RG 1.87, Revision 2, “Acceptability of ASME Code Section III, Division 5, ‘High Temperature Reactors'” endorses ASME Section III, Division 5 “High Temperature Reactors,” 2017 edition, with limitations and conditions, as an acceptable approach to assure the mechanical/structural integrity of components that operate in elevated temperature environments and that are subject to time-dependent material properties and failure modes.

  ASME BPVC III-5, 2023 Edition [3], with augmentations and modifications as discussed herein, is leveraged as guidance to design the GCA to be capable to support the RSFs through the life of the reactor. The 2023 Edition is leveraged instead of the 2017 Edition that is endorsed by the NRC because it is the most recent edition available and includes several improvements, including some that partially address the exceptions and limitations from NRC’s endorsement. Specifically, ASME BPVC III-5, 2023 Edition [3] is leveraged for the development of the graphite material qualification methodology discussed in Section 4 and the design methodology discussed in Section 6. Note: The Xe-100 GCA will not be designed with full compliance to the ASME BPVC; however, it will be designed to this LTR which describes the extent to which the methodology in ASME BPVC is leveraged, modified, and augmented as appropriate to address the underlying intent of ASME BPVC.

3. Overview of the Xe-100 Graphite Core Assembly

Heat generation in the Xe-100 reactor comes from its spherical fuel elements or pebbles which each contain a large number of tri-structural isotropic (TRISO) particles embedded in matrix graphite. The GCA forms the cylindrical shape of the core geometry consisting of a large number of these fuel pebbles (i.e., “pebble bed”). Pebbles are circulated from the top of the pebble bed to the bottom, extracted, measured for physical integrity and burn-up, and either recirculated for another pass or discharged as used fuel. The GCA is comprised of keyed and dowelled GCCs that provide neutron reflection and moderation. The GCA also provides channels for control rods to enter, moderating the fission reaction, and flow channels directing helium coolant from inlet to outlet of the RPV. Finally, the GCA is one of the systems in the heat transfer pathway towards the passive heat sink used to maintain acceptably low temperatures in off-normal operation (i.e. RCCS).

The GCA supports the Xe-100 RSFs 1.1.2, 1.2.1, and 1.4.1 [8][46]. These RSF’s Required Functional Design Criteria (RFDC) and their related Safety Related Design Criteria (SRDC) are summarized in Table 1.

To support RSF 1.1.2 (Maintain long-term subcriticality), the GCA provides channels for the insertion of control/shutdown rods. This allows for negative reactivity to be inserted within a sufficient time period to ensure safe shutdown during a DBE or DBA. It also allows for sufficient reactivity margin to ensure safe shutdown during a DBE or DBA.

To support RSF 1.2.1 (Maintain core heat removal through passive means), the GCA shall be designed to transfer sufficient heat via conduction, radiation, or convection to the core metallic structures to ensure that fuel and radionuclide release limits are not exceeded during a DBE or DBA. The Xe-100 design establishes GCC geometries and tolerances to ensure adequate heat conduction and radiation across all modes and states throughout the GCA.

To support RSF 1.4.1 (Maintain core geometry), the GCA shall be designed to ensure acceptable geometry is maintained to control reactivity and heat removal during DBEs and DBAs.

In addition, the GCA supports NST PSF 2.3.1 (Retain radionuclides in the reactor system). The GCA is designed in a manner that ensures radionuclides released from the fuel during normal operations are inherently retained and that specified acceptable radionuclides release design limits are met in anticipated operational occurrences (AOOs).

Table 1: *RSFs Applicable to the GCA*

Functional Requirements			System-Level Function Requirements		
RSF #	**RSF**	**Required Functional Design Criteria (RFDC)**	**RSF #**	**RSF**	**Safety Related Design Criteria (SRDC)**
1.1.2	Maintain long-term subcriticality	The reactor shall be designed to include movable poisons that can insert and maintain safe shutdown during DBEs and DBAs. (PDC-RFDC 26)	1.1.2.1b	Provide means to insert negative reactivity to ensure safe shutdown	The reactivity control and shutdown system shall be designed to insert negative reactivity within a sufficient time period to ensure safe shutdown during a DBE or DBA.
1.1.2	Maintain long-term subcriticality	The reactor shall be designed to include movable poisons that can insert and maintain safe shutdown during DBEs and DBAs. (PDC-RFDC 26)	1.1.2.1c	Provide inherent reactivity characteristics to ensure safe shutdown	The reactor core, including the fuel, shall be designed such that, in concert with the reactivity control and shutdown system, there is sufficient reactivity margin to ensure safe shutdown during a DBE or DBA.
1.2.1	Maintain core heat removal through passive means	A passive system shall be designed to ensure that acceptable fuel and radionuclide release limits are not exceeded during DBEs and DBAs. (PDC-RFDC 34)	1.2.1.1c	Transfer heat from graphite core assembly to core metallic structures	The graphite core assembly shall be designed to transfer sufficient heat via conduction, radiation, or convection to the core metallic structures to ensure that fuel and radionuclide release limits are not exceeded during a DBE or DBA.
1.4.1	Maintain core geometry	The reactor core shall be designed such that the reactor vessel and reactor system geometry is maintained to (1) ensure geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) permit sufficient insertion of neutron absorbers to provide for reactor shutdown. (PDC-RFDC 70)	1.4.1.3	Maintain acceptable graphite core assembly geometry	The graphite core assembly shall be designed to ensure acceptable geometry is maintained to control reactivity and heat removal during DBEs and DBAs.

The subsections which follow discuss the configuration and functions of the GCA in more detail. Select design data for the Xe-100 are provided in Table 2.

Table 2: *Xe-100 Design Data*

Parameter	Value/Description
Reactor Type	HTGR
Core Configuration	Spherical fuel in a right cylindrical pebble bed configuration
Pebble Bed Physical Dimensions	Height: 9.2 m, Diameter: 2.4 m
Reactor Thermal Power	200 MWth
Reactor Coolant and Operating Range	Helium, 260 °C to 750 °C
Material for Safety-Related Structures	Graphite, Metal
Note: These values are preliminary.	

3.1 GCA Configuration

As described above, the GCA provides both neutron reflection as well as mechanical support and guidance for the fuel pebbles to move through the core. Cooling gas flow is directed via the GCA and flow that bypasses the core is an important design consideration. The GCCs are stacked according to a single-column principle, with no GCC being supported by more than two other GCCs in parallel. This minimizes the risk of tensile loading on individual GCCs during normal operation. To meet the 60-year design lifetime, the inner wall of the GCA (Inner Reflector GCCs) must be designed for midlife replacement to mitigate the irradiation induced effects on the GCA region exposed to the highest neutron flux. Figure 1 [50] shows a cross section of the GCA, highlighting each of the individual GCC categorizations and splitting the assembly into functional groups.

[[

]]P,E

Figure 1: ***Cross Section View of the Graphite Core Assembly***

The GCA is described by three groupings based on region in the core. Each of the groupings consist of various GCCs and support a range of functions of the overall assembly. The groupings are described as: the side reflector group, the top reflector group, and the bottom reflector group. Each of these groupings is further broken down into its GCCs and described in subsequent sections.

3.1.1 Side Reflector Group

The side reflector group (referred to herein as the 'side reflector') is located between the bottom reflector group and the top reflector group and sits inside the core barrel (CB). It is supported and aligned by rails that are part of the CB. A rendered cross-section of the side reflector showing the inner and outer GCCs, riser sleeves, and control rod penetrations is shown in Figure 2 [50].

The design of the side reflector allows for the replacement of part of the GCA at the middle of the reactor life. Only the inner surface of the inner GCCs is exposed to the highest neutron flux[1], causing the need for replacement. Thus, only this set of GCCs must be replaced at midlife to maintain the integrity of the entire GCA to 60-years. For this replacement, and geometric size limitations, the side reflector is made up of two sizes of GCCs, the inner and outer GCCs. The inner and outer GCCs as well as the control rod sleeves that act as locking keys for the inner GCCs are shown in Figure 3 [50]. The inner GCCs are smaller than the outer GCCs.

The inner GCCs are made of [[ ]]^{P,E} and designed with replacement in mind. The inner GCC is held to the outer GCC by a keyed dovetail fitting, using the control rod sleeve to lock them in place. The inner face of the inner GCC has semi-spherical and irregular indents with the center at the corners of the adjacent inner reflector GCCs. This prevents ordered-structure locking of the fuel pebbles as they move through the core. Mid-life replacement will be performed via robotic manipulation.

The outer GCCs in the side reflector are larger and are made of [[ ]]^{P,E}. The sizing of the GCCs is selected to balance the competing objectives of maximizing billet utilization, minimizing the number of leakage flow paths, and maintaining acceptably low dimensional change-induced stresses. The outer GCCs provide for both gas flow in riser channels, as well as penetrations for the control and shutdown rods. A rendered cut-away showing the inner and outer GCCs as well as helium gas riser sleeves and control rod sleeves is shown in Figure 4 [50].

The larger GCCs provide for better structural stability and are vertically aligned by dowels. Sleeves are inserted in the helium riser channels and control rod channels to minimize helium leakage to the core and outlet plenum. As the GCCs cannot be bonded together, leak prevention keys are used to limit helium leakage flow from the cold high-pressure volume between the CB and the side reflector and into the outlet plenum. The keys are rounded "dog-bone style" which allow for relative movement between the GCCs without wedging or breaking the GCCs. Sealing mechanisms are shown in Figure 5 [50].

[1] See discussion in Section 3.3.

[[

]]P,E

Figure 2: ***Graphite Side Reflector Cross Section View***

[[

]]P,E

Figure 3: ***Representative Side Reflector GCC***

[[

]][P,E]

Figure 4: ***Side Reflector Cut-Away***

[[

]]P,E

Figure 5: ***GCC Leak Prevention Mechanisms***

3.1.2 Top Reflector Group

The top reflector group (referred to herein as the 'top reflector') sits below the CB lid. It rests on top of the side reflector and is aligned by the CB alignment rails. The top reflector contains penetrations for the shutdown and control rods and a removable plug. The features are shown in Figure 6 [50].

[[

]]P,E

Figure 6: *Top Reflector Group*

The thick top reflector provides for thermal insulation for the CB as well as neutron shielding. At the center of the top reflector is a removable plug assembly [[

]]P,E The plug assembly is stepped axially to prevent inadvertent neutron streaming from the reactor. Through the centerline of this plug, and the centerline of the reactor core, a pass-through for the fuel-loading equipment exists. This is connected to an inlet pipe passing through the reactor vessel head, interfacing with the Fuel Handling System (FHS).

The upper GCCs overhang the core and provide helium inlet to the core cavity from above. The remainder of the top reflector rests on the inlet plenum. This subgroup of GCCs overhangs the top of the core. As there are overhangs in the GCA, the GCCs are designed with spacing in the unsupported areas and act as cantilevers. This way, there is no additional tensile stress applied, thus following the single-column principle.

The top two rows of GCCs in the side reflector provide the inlet of the helium for the reactor core. The inlet plenum is designed to balance flow from the risers and spreads the helium into radial inlet slots. These top two rows of GCCs are geometrically similar to the remainder of side reflector GCCs, although they have cutouts to allow for helium flow into the plenum. They also contain holes for the control rod sleeves. The inlet plenum is shown in Figure 7 [50]. The outer surface of the GCCs contains T-grooves to interface with the rails on the CB.

[[

]]P,E

Figure 7: ***Inlet-Plenum in the Side Reflector***

3.1.3 Bottom Reflector Group

The bottom reflector group (referred to herein as the 'bottom reflector') rests directly on the CB support pedestal. The bottom reflector consists of the fuel outlet funnel, the outlet plenum, and an annular region similar to the side reflector. The defuel chute begins at the base of the funnel, passes through the bottom reflector, and ultimately exits the RPV. The side reflector rests on the outer portion of bottom reflector. The bottom reflector also interfaces with the cool helium inlet which enters from the Cross Vessel (CV) as well as the Hot Gas Duct (HGD) at the outlet. The bottom reflector is shown in Figure 8 [50].

The fuel funnel is made up of individual hexagonal-shaped GCCs with through-holes drilled the entire length of the component to allow the cooling gas to leave the core area. These GCCs are stacked on top of a distributing cup where holes are drilled at an angle to meet the flow path from the core. The support columns each support an individual hexagonal and distributor cup pair. In this way, there is some shielding to prevent direct neutron streaming from the core into the outlet plenum. The outlet funnel ends at the defueling chute and is designed with a diameter sufficient to prevent pebble blockage by bridging.

The outlet plenum consists of the void between the hypostyle support columns and the volume where the gas is directed to the HGD connection. The support columns rest in place on the lower portion of the bottom reflector called the plenum floor.

[[

]]P,E

Figure *8*: *Bottom Reflector Group*

3.2 Design Considerations

The design of the GCA to act as a reflector, moderator, radiation shield, helium flow guide, and fuel pebble path in a high temperature, high pressure, and high neutron flux environment must consider several factors. The changes occurring in graphite material with temperature and fast neutron fluence are complex but well characterized as graphite material has been used since the very first reactors. Owing to the mechanical, geometrical, thermal, and neutronic changes of the graphite material, strict design principles must be followed. A summary of the major design principles is contained in the following sections.

3.2.1 Single Column Principle

All GCC stacks are preferentially adherent to the "single column principle." This means that GCCs stacked in the vertical direction can only be supported by a single GCC directly beneath. The intent is to limit the potential for bridging as this could place the GCCs under a tensile stress condition which is a preferential condition for crack initiation and crack growth. In areas where overhangs occur, an analysis will be conducted to ensure proper counter-levering exists without applying excessive tensile stress.

3.2.2 Stacking and Joining

Considering the dimensional changes due to temperature and neutron irradiation, it is not possible to bond GCCs together. In this case, they must only be stacked on top of one another. Guide rails and wedges are used in the CB interface with the GCA to maintain GCC alignment. The GCCs are keyed in such a way as to prevent excessive movement. Rounded and chamfered edges prevent stress concentrators that could lead to chipping.

3.2.3 Sealing

One of the main GCA functions is to provide helium flow channels, acting as the riser for the helium. Additionally, in the core where the helium is in the downflowing region, leakage flows through the GCA and around the pebble bed must be minimized. As it is not possible to bond the GCCs, perfectly tight sealing is impossible. However, using overlapping sleeves, dog-bone keys, rounded edges and keyways, GCC offsetting, and taking advantage of thermal expansion, controlling leakage and bypass flows within acceptable levels is achieved.

3.2.4 GCC Replacement

Graphite material grade is selected, and GCCs are designed to minimize the burden of graphite material waste while achieving an acceptable plant lifetime for a core where neutron flux varies substantially by location. In many graphite-moderated reactor designs, this is achieved by including replaceable components for those exposed to the most onerous conditions (highest fast neutron flux, highest temperature, and/or oxidation rate). The remainder of the GCCs may then be used for the lifetime of the reactor without replacement. For example, in the UK Advanced Gas-Cooled Reactors the fuel assemblies include integral graphite sleeves that are replaced with the fuel. HTGRs similarly include replaceable graphite as part of the fuel elements (component type or pebble type). These reactor designs may also involve the replacement of some or all of the innermost GCCs, which are exposed to the highest fast neutron flux.

The area of the Xe-100 GCA experiencing the highest neutron flux is only about 2 meters in height. The GCA is designed to accommodate replacement of the inner GCCs while leaving the outer GCCs in place for the life of the reactor. Inner GCC replacement is facilitated by a removable plug sub-assembly in the top reflector allowing access for robotic manipulators to operate in the empty core cavity.

3.3 Environmental Conditions

The Xe-100 GCA is exposed to several environmental conditions such as high dose fast neutron irradiation, high temperatures and the presence of low concentrations of oxidizing impurities. The degradation mechanisms associated with these environmental conditions are discussed further in Section 6.6.

[[

]][P,E] Figure 9 [18] represents the bounding temperature and fluence plots at 60 years, considering the following power levels: 100%, 60%, 40%, and 25% Maximum Continuous Rating (MCR). Note that the inner side reflector is planned to be replaced at mid-life of operation which is accounted in this figure.

[[

]]P,E

Figure 9: ***Bounding Fluence and Temperature States at End of Life (60 years) for Operating Power Levels of 100%, 60%, 40%, and 25% MCR***

The side reflector is exposed to the highest neutron flux. Figure 10 [18] shows several 2D top down views of the side reflector stack sliced at key horizontal elevations of the GCA and shows the time that it would take the side reflector to accumulate given amounts of fluence: (1) 0.25 dpa, (2) turnaround (i.e., point at which volumetric dimensional change strain reverses), and (3) nullity (i.e., point at which zero volumetric dimensional change strain occurs after turnaround, also referred to as crossover). The plot is color-coded such that the blue region indicates that the fluence value will be reached in less than 30 years; teal indicates a time between 30 and 60 years; and yellow indicates a time greater than 60 years. The locations of the arbitrary z-slices used are illustrated in Figure 11 [18].[[

]]P,E As discussed in Section 6.5, dimensional change is considered as one element of the IGNIS material model and thus accounted for in the stress and deflection analysis.

[[

]]P,E

Figure *10*: *Side Reflector, Time to Key Fluence Levels 0.25 DPA, Turnaround, Nullity at 100% Power [18], blue < (30 yr) < teal < (60 yr) < yellow*

[[

]]P,E

Figure 11: Slice Plane Locations Used in Table 10

4. Graphite Material Qualification Methodology

4.1 Introduction

[[

]]P,E

This section provides an overview of the material qualification methodology that will be used to collect the appropriate graphite grade-dependent material properties for the design of the GCA. This material qualification methodology, which is documented in detail in Reference [39], is developed based on the requirements from the ASME BPVC III-5, 2023 Edition, Subsection HA, Subpart B (HAB) and Subsection HH, Subpart A (HHA) [3] and covers three aspects: (1) as-manufactured graphite, (2) irradiated graphite, and (3) oxidized graphite.

The material property data generated as a result of the material qualification methodology described herein will be documented in the Material Data Sheet (MDS) as defined in the ASME BPVC III-5. A MDS will be prepared for each of the graphite grades to be used for the Xe-100 reactor. The design material property data contained in these MDS will be used to evaluate the design of the GCA and its GCCs per the design methodology described in Section 6.

Due to limited grade-specific data available to complete the Material Data Sheet, especially for irradiated data, the data set to be used for design analysis is enhanced by addition of data from other grades, leveraging common underlying behavior of graphite and tuning a constitutive material model to the grades selected for construction. The augmented data set and implementation approach in the material model (IGNIS) to be used design analysis is presented in Section 6.

The material qualification methodology identifies needs for material property data that require additional testing. These are discussed further in Section 5 as part of the test plan.

4.2 Approach

ASME BPVC III-5, HHA-2132 (Material Qualification) [3] guides the qualification of a grade of graphite from three aspects:

(1) As-manufactured graphite – see Section 4.4
(2) Irradiated graphite – see Section 4.5
(3) Oxidized graphite – see Section 4.6

The Xe-100 graphite qualification approach will leverage historical data and either plan additional testing to obtain those not available from historical data or justify the use of extrapolated data to qualify [[

]]P,E. Material qualification is an ASME code required activity to holistically evaluate graphite material to qualify its use for the nuclear reactor construction. The scope of major qualification activities is summarized in Table 3 to Table 5.

The qualification plan intends to comply with each code requirement; however, X-energy will take exceptions to various code requirements based on design needs, code achievability, and available data, by applying a performance-based graded approach to quality of design data (Section 4.7). The following preliminary definitions are established to distinguish different categories/use cases of design data:

(1) Inspection data: specimen data from as-manufactured billets which will form the components to be put into the core, which demonstrate that the material meets the graphite specification set by X-energy.
(2) Qualification data: data from specimens from as-manufactured billets (e.g., 12 billets from three graphitization charges) establishing statistical distribution of properties within and between billets and charges, which is used to generate representative data for design use.
(3) Predictive data: data from a range of graphite grades (including as-manufactured, irradiated, and oxidized data), which supports design analysis activities for prediction of future operational performance behavior, accounting for uncertainties, and providing margin to the design.

The treatment of data for each category will be performance-based commensurate with the contribution to safety. Justifications will be provided for each requirement to which an exception will be taken. See Appendix A for a list of code compliance as it pertains to material qualification.

Table 3: *Major Qualification Activities for As-Manufactured Graphite*

Qualification Testing Items	Plan New Testing?	Testing Scope or Approach	Use of Data
Mechanical and Thermal Properties	Yes	Sampling 12 billets from three graphitization charges. 288 specimens for each property testing. Testing matrix, see Table 7, including strengths (tensile, flexural, and compressive), Dynamic moduli; CTE and Thermal conductivity (TC). All the properties test in both WG and AG orientation. Sampling, specimen machining and testing.	Generate representative data for the grade. Input for structural analysis, e.g., tensile strength as input for POF calculation.
Property Variation	No	Utilize supplier's QC (quality control) data accumulated over the years, including local (intra billet) variation and lot-to-lot variation.	Establish specification (used as acceptance criteria in procurement).

Table 4: *Major Qualification Activities for Irradiated Graphite*

Qualification Testing Items	Plan New Testing?	Testing Scope or Approach	Use of Data
Dimensional change, physical properties not available from historical data	Yes	[[ ]]P,E	Structural analysis and safety analysis
Dimensional change, physical properties available from historical data	No	[[ ]]P,E	Structural analysis and safety analysis
Irradiation Creep	TBD	[[ ]]P,E	Structural analysis

Table 5*: Major Qualification Activities for Oxidized Graphite*

Qualification Testing Items	Plan New Testing?	Testing Scope or Approach	Use of Data
Rate of oxidation and oxidation penetration depth	TBD. [[ ]]P,E Note: If the literature data are not justified to use, testing will be performed.	[[ ]]P,E	Determine the severity of oxidation (by local weight loss).
Oxidized Graphite Properties	TBD. [[ ]]P,E Note: If local weight loss is greater than 1%, and no literature data are justified to use, testing will be performed.	[[ ]]P,E	Structural analysis, safety analysis.

4.3 Codes and Standards Involved

The codes and standards involved in the graphite material qualification program are listed below in Table 6.

Table 6*: Selected ASME Code and ASTM Standards for Graphite Material Qualification*

Graphite Properties to be Qualified	ASME BPVC III-5 2023 Edition Articles and ASTM Standards[1]
Mechanical and Thermal Properties	HHA-2121 Application of the Rules of This Subpart HHA-2132 Qualification of Materials HHA-2200 Material Properties for Design HHA-2210 As-Manufactured Materials Properties HHA-2220 Irradiated Material Properties HHA-2230 Oxidized Material Properties HHA-I-1110 Material Specifications HHA-II-1000 Introduction HHA-II-2000 Material Data Sheet Forms HHA-II-4000 Detained Requirements for Derivation of the Material Data Sheet---Irradiated Material Properties HHA-III-3000 Properties to be Determined HHA-III-3100 As-Manufactured Graphite HHA-III-3200 Oxidized Graphite HHA-III-3300 Irradiated Graphite HHA-III-4000 Requirement for Representative Data HHA-III-4100 As-Manufactured Graphite HHA-III-4200 Irradiated or Oxidized Graphite HAB-4557.2 Manufacturing Process Control ASTM C781-08 Standard Practice for Testing Graphite Materials for High-Temperature Gas-Cooled Nuclear Reactor Components[1] ASTM D7219-08 Standard Specification for Isotropic and Near-isotropic Nuclear Graphites Note: Individual ASTM standard for each property test is listed under the corresponding test matrix.
Property Variation	HHA-III-5000 Use of Historical Data
Purity	HHA-III-5000 Use of Historical Data ASTM D7219-08 Standard Specification for Isotropic and Near-Isotropic Nuclear Graphites ASTM C1233-15 Standard Practice for Determining Boron Contents of Nuclear Materials

Graphite Properties to be Qualified	ASME BPVC III-5 2023 Edition Articles and ASTM Standards[1]
	ASTM C561-16 Standard Test Method for Ash in a Graphite Sample[2]
Oxidation	HHA-3141 Oxidation HHA-III-3200 Oxidized Graphite HHA-III-5000 Use of Historical Data

Note(s):

1. If the sample sizes available are smaller than what is specified by each specific ASTM standards herein, a sample size study will be completed in accordance with the guidance provided in ASTM D7775-21, "Standard Guide for Measurements on Small Graphite Specimens."

2. This ASTM standard is required for measurement of ash by ASTM D7219-08.

4.4 As-Manufactured Graphite

4.4.1 Generation of Representative Data

The representative data obtained from the qualification testing will be used to generate Material Data Sheets (HHA-II-2000), which are used for design and safety analysis. For example, the strength data are input for the Probability of Failure (POF) calculation (HHA-II-3000).

4.4.2 Graphite Billets Selection

The graphite billets selected for qualification shall be produced to the same specifications as those graphite billets supplied for irradiation tests. This is to be accomplished by using raw materials with the same specifications and the same graphite manufacturing processes. [[

]]^{P,E} The graphite properties shall comply with the general requirement for high purity nuclear graphite specified in American Society for Testing and Materials (ASTM) D7219-08, cited by ASME BPVC III-5 HHA-I-1110.

The qualification tests, to generate qualification data, will select 12 total graphite billets per grade of graphite from three graphitization charges, four billets per graphitization charge, as per HHA-III-4100.

The inspection testing report, e.g., Certificate of Analysis (CoA), of each billet selected for qualification testing shall be provided to X-energy for review and approval before cutting the billets for sampling. CoA shall have the same content as Certified Material Test Reports (CMTRs) listed in HHA-2121. Note: Once the graphite is qualified, the term CMTR shall be used in billet purchasing to demonstrate that the graphite meets the specifications set by X-energy.

4.4.3 Specimen Sampling

Specimen sampling will reflect the property pattern of graphite inherited from the forming process.

For [[ ]]P,E, the sampling plan is illustrated in Figure 12. Iso-molded graphite typically has property gradients in the mold filling direction, which is the against-grain (AG) direction. The specimen will be taken from three slabs, i.e., top, middle, and bottom, along the AG direction. On each slab, specimens will be taken from center and periphery (shorter side) locations. At each location, both with-grain (WG) and AG specimens will be taken for each testing item. Therefore, four WG and four AG specimens will be taken from each slab. A total of 12 WG and 12 AG specimen for each testing item will be taken from each billet.

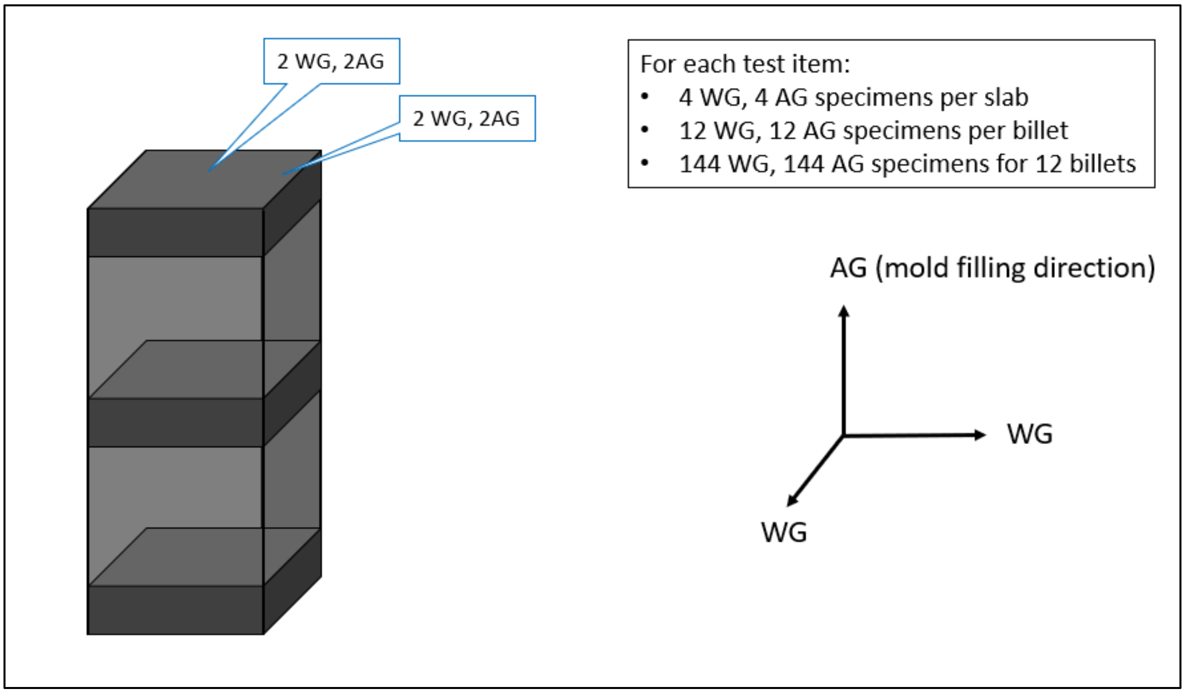

Figure 12: ***Specimen Sampling Plan for [[ ]]P,E***

For [[ ]]P,E, the specimen sampling plan is proposed in Figure *13*. It will be finalized with the supplier regarding the property pattern they observed. On each slab, specimens will be taken from center and periphery (longer side) locations. At each location, both WG and AG specimens will be taken for each testing item. Therefore, four WG and four AG specimens will be taken from each slab, total 12 WG and 12 AG specimens for each testing item will be taken from each billet.

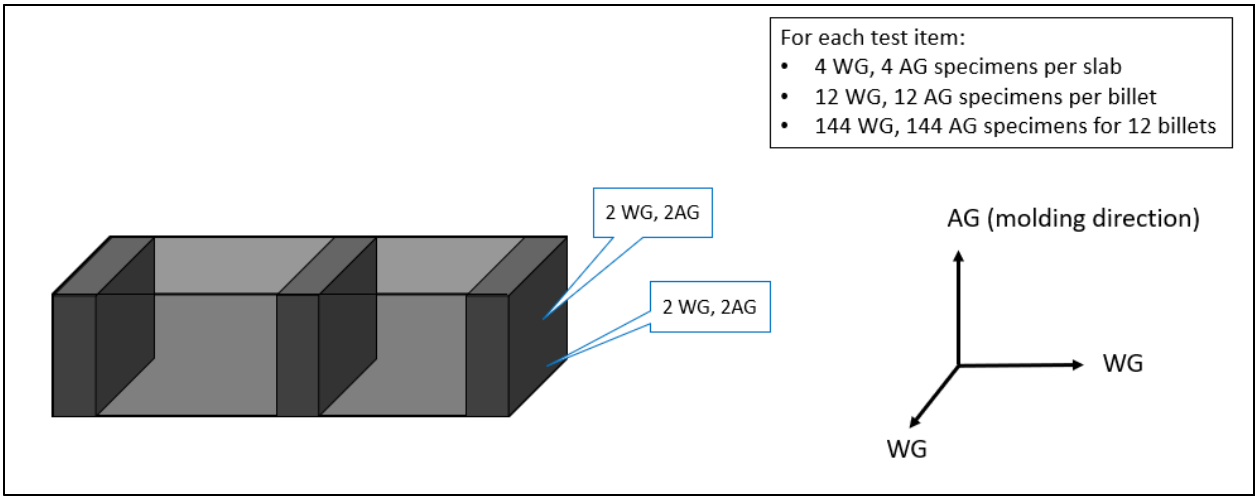

Figure 13: ***Specimen Sampling Plan for [[ ]]^{P}***

Test specimen machining will follow the requirement provided in the corresponding ASTM standard of each test. X-energy will review and approve the machining drawing before specimen machining starts.

4.4.4 Test Matrix

The test matrix presented in Table 7 is developed based on the requirements from HHA-III-3100 (Properties to be determined for as-manufactured graphite) and HHA-II-2000 (Material Data Sheet Forms).

The test specimen size will be finalized with testing labs, e.g., [[ ]]^{PE}. X-energy will ensure the specimen size meets the corresponding ASTM requirement. When [[ ]]^{PE} is the only option to use, the equivalency of the test results will be compared and correlated with those measured by ASTM to justify the proper use of testing methods for material qualification purpose.

The test temperature range for the Coefficient of Thermal Expansion (CTE) and thermal conductivity envelops the Xe-100 reactor normal operation temperature.

Strength tests (Tensile, Flexural, and Compressive) will be at room temperature, as X-energy does not take credit for the increased strength at elevated temperature in the current set of methodologies. Room temperature strength is conservative. If methods are changed to leverage increased strength in the evaluation of GCC integrity, this testing requirement will be reviewed to ensure data needs are met. This approach complies with HHA-II-2000 (2023 Edition).

Poisson’s ratio and anisotropy factor (ratio) are calculated values.

- Poisson’s ratio is calculated from the measured Dynamic Young’s modulus and Dynamic Shear modulus, per ASTM C747 (“Standard Test Method for Moduli of Elasticity and Fundamental Frequencies of Carbon and Graphite Materials by Sonic Resonance”).

- Anisotropy factor (ratio) is calculated from measured CTE, i.e., CTE_{AG}/CTE_{WG}.

The Xe-100 graphite material qualification approach will take exception from the Code for the following tests with the justification given below.

1. Flexural strength: three-point flexural strength will be measured instead of four-point flexural strength as required per HHA-II-2000 (Material Data Sheet). The justification is listed below.
    a. Graphite suppliers routinely measure the flexural strength in three-point mode. Based on discussions with each graphite supplier on historical flexural strength data, it is advantageous to be consistent with the measurement techniques previously used to assess and track manufacturing variability.
    b. It is expected that the full assessment method will primarily be used to determine the graphite's reliability curve parameters, where only tensile and compressive strengths are used as inputs.
    c. The option of using simplified assessment method may also be considered to determine the material's reliability curve parameters, where flexural strength will be used as well. To satisfy this need, a small group of flexural strength test in both three-point and four-point mode will be performed to generate a correlation factor between the two tests for the specific grade of graphite. Then use such correlation factor to convert the three-point flexural strength to four-point flexural strength, if needed. To generate a reasonable correlation, the two types of specimens will be machined to the same cross-section dimension. The detail of test matrix and four-point loading fixture configuration will be further discussed with graphite suppliers.
2. Fracture Toughness: historical data will be leveraged for this parameter. The Crack Propagation Analysis Methodology [29] identifies suitable literature values for fracture toughness of graphite. Based on the current assessment of the risk significance of graphite fracture, it is judged that no additional testing is required.
3. Elastic Modulus at elevated temperature: new elastic modulus measurement data at elevated temperature is not required for the performance-based design approach. Instead, published data will be used in the case such data are needed, see Reference [39] for the justification.

Table 7: *As-Manufactured Graphite Qualification Test Matrix*

Property	Orientation	Number of Specimens Per Billet	Total Number of Specimens	Test Standard	Temperature
Bulk Density [1]	WG & AG	24	288	ASTM C559-16	Room Temperature (RT)
Tensile Strength	WG AG	12 12	144 144	ASTM C749-15	RT
Flexural Strength [2]	WG AG	12 12	144 144	ASTM D7972-20(3-pts) ASTM C651-20(4-pts)	RT
Compressive Strength	WG AG	12 12	144 144	ASTM C695-21	RT
Dynamic Young's Modulus and Shear Modulus	WG AG	12 12	144 144	ASTM C747-16	RT
Coefficient of Thermal Expansion (CTE)	WG AG	12 12	144 144	ASTM E228-22	RT, 200, 400, 600, 800, 1000°C
Thermal Conductivity [3]	WG AG	12 12	144 144	ASTM E1461-22 ASTM C781-20	RT, 200, 400, 600, 800, 1000°C

[1] Bulk density specimen can be shared with other tests, e.g., CTE, flexural strength specimen, etc., to be finalized with [[ ]]P.

[2] Three-point flexural strength will be performed for all the specimens; a small group of four-point flexural strength test will be added to testing program to generate a correlation factor between three-point and four-point flexural strength. The cross-section dimension of the two types of specimens shall be the same.

[3] Thermal conductivity value is calculated from thermal diffusivity (measured per ASTM E1461-22) and specific heat capacity of graphite provided in ASTM C781-20, A4.

4.4.5 Evaluation of Graphite Property Variation

Graphite is known to be less uniform than metallic materials. The local (or intra-billet) variation is the characteristics of graphite determined by specific forming process. It is not a quality issue. However, lot-to-lot variation reflects the quality control ability. It shows how consistently the graphite manufacturer can produce a grade of graphite over the years. Raw material consistency and processing control ability affect the lot-to-lot variation.

Note: The property variation data from manufacturers are typically measured per manufacturer's internal standards and sampling plan. The practice fulfills the purpose evaluating the graphite property variation, but the data are not intended to substitute the qualification testing data.

4.4.5.1 Local Variation

Local (or intra-billet) variation will be evaluated by using the data from (1) manufacturer's historical data on billet uniformity study and (2) data from this qualification testing program as described below.

1. Data from manufacturer's historical data: Performing billet uniformity study is a common practice in a well-established graphite company, especially in the research and development stage of this grade. The uniformity study typically involves comprehensive sampling throughout the billet but limited to a few billets and test items. The data provides important information regarding property pattern and local (or intra-billet) variation.
2. Data from this qualification testing program: The data are collected from this qualification program will include 12 billets from three graphitization charges. Compared to graphite manufacturer's historical data, the qualification program will test more billets with all the properties required by ASME code, but limited number of specimens are sampled per billet (12 specimens each orientation per billet). The data from qualification testing program are the confirmation to the graphite manufacture's historical data.

4.4.5.2 Lot-to-Lot Variation

The evaluation of lot-to-lot variation relies on graphite manufacturer's historical data on a grade of graphite. Only such data are meaningful to evaluate a grade of graphite produced over the years. Use of historical data complies with HHA-III-5000. X-energy will first work with [[

]]^{P}, then determine the variation (average value and standard deviation) of critical properties, e.g., strength, modulus, CTE, electrical resistivity, and purity, etc. The lot-to-lot variation will also help to determine the graphite specification, i.e., value range of each property.

4.4.6 Purity

The purity of graphite billets selected for the qualification test will comply with ASTM D7219-08 on purity requirement for high purity nuclear graphite, i.e., Ash content =< 300 ppmw (Parts Per Million by Weight) and Equivalent Boron Content (EBC) =< 2 ppmw.

In addition, the upper limit of oxidation-promoting impurities and other impurities, e.g., Lithium (a source of Tritium), will be determined through the qualification program. The process involves analyzing the trace impurities content in regularly produced [[ ]]P to determine if the level of such impurities satisfies the oxidation resistance and Tritium generation requirements. If such impurities content is too high, even if the total ash content complies with ASTM D7219-08 (=<300 ppmw), a lower ash content limit will be specified to ensure such impurities content is below their upper limit.

Since the purity inspection test at the graphite manufacturer site is typically performed on the specimens taken at near surface location, the impurity distribution throughout the billet will be assessed to ensure the purity measured at the near surface represents the whole billet purity. In the case that the impurity distribution shows a pattern, such pattern will be assessed. The assessment of purity and its distribution within the billet relies on the data from graphite manufacturer accumulated over the years. The use of such data complies with HHA-III-5000 (Use of Historical Data).

4.4.7 Marking, Tracking of Billets and Test Specimens

The billets selected for qualification testing shall be able to be traced back to the raw material lot and processing lot for each processing step. The test specimen tracking method shall be discussed and finalized between X-energy and graphite manufacturers. The details can be found in Reference [39].

The test data shall be reported in spreadsheet format and can be sorted by the billet identification, graphitization charge number, sampling locations, orientation for each property tested, including individual data point, average value, and standard deviation. A set of documents shall be included in data package to demonstrate that the billet selection, sampling plan and testing procedures are approved by X-energy. The details can be found in Reference [39].

4.5 Irradiated Graphite

4.5.1 Introduction

The objective of irradiation qualification is to support a risk-informed, performance-based safety case for the Xe-100 within the framework of NEI 18-04 [1]. In this context, data requirements depend on the risk significance of uncertainty in graphite properties and the resulting effects on GCCs. While the irradiated data qualification plan is based on the requirements of the ASME Code, full compliance with the code is not believed to be achievable. This plan highlights anticipated gaps to Code compliance. Gaps in data contribute to uncertainty in graphite properties, the consequences of which will be evaluated separately.

When subject to fast neutron irradiation, graphite undergoes complex changes in dimensions and material properties. Stress and distortion in the components result. This may lead to cracking, effects on the shape, load-bearing capacity, thermal resistance, and other properties of potential significance to safety and other functions of GCCs. In general, properties need to be known as a function of fast neutron dose, irradiation temperature, instantaneous temperature, and stress history to allow predictions of the irradiation response of a GCC in an inert environment. Some important practical limitations apply to the collection of irradiation data.

4.5.2 Limitations

Ideally, reactor designers would have a complete understanding of irradiated graphite behavior. This would be supported by high quality, fully representative, grade-specific irradiated data for all properties. Experimental dose, temperature and weight loss would envelope the anticipated reactor conditions. Material variability would be well characterized and the requirements of the ASME code would be met in full.

In reality, there are significant barriers to obtaining this data and a full theoretical understanding of irradiated graphite remains elusive. Materials Test Reactor (MTR) capacity is limited, experiments are complex and time-consuming, and only a small volume of material can be irradiated. Recent irradiation experiments have struggled to achieve the planned irradiation temperatures, while creep experiments typically cannot investigate all aspects of the phenomenon. Even within the ASME Code requirements, only a small fraction of one graphite billet must be irradiated and it is assumed that accelerated irradiation in a high flux, water-cooled reactor is representative. Variability in the irradiation response of graphite, the effects of specimen size, neutron flux, coolant chemistry and load history must all be addressed through analysis and extrapolation.

These challenges are particularly acute for a pebble-bed type gas-cooled reactor such as the Xe-100, where the pebble bed wall is formed from GCCs that must withstand irradiation for decades and where the range of irradiation temperatures is broad.

The ASME Code offers one route to material qualification; however, as a generic code it cannot be aligned with the safety requirements of every reactor design. Some onerous Code requirements may have little benefit to reactor safety, while there are other requirements that are not well defined in the Code.

Given these constraints, some reference to historic data for the same grade and other grades is always required. The purpose of irradiation qualification must be to show that the specific grade conforms to the expected behavior of nuclear graphite in general and that uncertainty in the irradiation response can be characterized.

4.5.3 Approach Taken for Irradiated Graphite Qualification

While it would be preferable to comply fully with the ASME Code, the challenges noted above mean that this may not be feasible. It is also neither necessary nor sufficient for the design of a safe reactor. The approach to irradiated graphite qualification taken for the Xe-100 GCA starts from the RSFS of the GCA.

The approach taken to irradiated graphite qualification is as follows:

- Develop a generic, semi-empirical model for irradiated graphite based on historic data, theoretical understanding of graphite irradiation damage, and available grade-specific irradiated and unirradiated data. This should include structural and thermal changes to the graphite material, GCCs and GCA, including the effects of oxidation and other degradation phenomena if required. See further discussion in Section 6.5 and Reference [21].

- Solve these models numerically, for example via a finite element analysis (FEA) package to allow the modelling of GCCs under realistic reactor conditions [21], including applying the ASME POF methods.
- Assess the uncertainty in irradiated graphite behavior and evaluate the consequences of this uncertainty on the outputs of interest (e.g., prevalence of cracking, thermal conductivity of GCCs), see Reference [42].
- Evaluate performance of the GCA against the RSFs, including in the presence of irradiation damage, cracking, GCA distortion and other damage (see Damage Tolerance Assessment Methodology [43]).
- Assess the significance of graphite damage within a Probabilistic Safety Analysis framework consistent with NEI 18-04 (see [13]).
- Prioritize obtaining new data based on the assessed safety margins. An evaluation of the gaps to meeting the ASME Code requirements should contribute to testing plans, but obtaining the full range of data specified by the Code may ultimately not be the best use of resources. The primary goal of new testing is to reduce uncertainty in areas of graphite performance that have significant consequences for safety.

The overall approach relies on a modelling methodology which starts by recognizing that irradiation damage in all grades of graphite is driven by common underlying processes. This means that data for one property or one grade of graphite provides useful information for other properties and grades. As an example, in the case of irradiation creep there is not enough data to treat different grades differently (once the effect of elastic modulus has been accounted for), and a generic model should be used for all grades. The uncertainty in irradiation creep behavior is significant and should be addressed through sensitivity studies in creep model inputs.

This section (Section 4.5) summarizes the available data and current testing efforts relevant to the material qualification of the Xe-100 GCA. It also outlines the gaps to ASME BPVC compliance as a first step to identifying potential areas for future testing. A broader range of testing which X-energy may pursue if it is justified by safety analysis will be summarized in a subsequent comprehensive graphite testing plan.

Overall plans for testing, including material qualification tests as well as tests not required to support material qualification, are summarized in Section 5. Material qualification testing is included in the test plan as part of Table 12 and discussed in Section 5.2. It is intended that this qualification methodology (Section 4) stands on its own while Section 5 acts as a summary of all testing, hence the same testing is referenced in multiple places.

4.5.4 Quality Requirements

As is further discussed in Section 4.7, the ASME BPVC (2023) [3] as currently written appears to unintentionally assign that all design data should be generated in compliance with ASME NQA-1, using material manufactured in accordance with HAB-4000 or HAB-3800. These requirements are not met by any currently available irradiated data, nor is it believed that this is the intent of the clauses associated with representative or historic data (see additional discussion in Section 4.7). Historic data has been generated over decades in a wide variety of settings, using a range of test standards and with varied

Quality Assurance (QA) records available. While newer data from national laboratories often comes close to meeting the ASME requirements, there may still be significant benefits to using older data, data from experiments using foreign standards, data from academic laboratories, etc. The value of data depends on both the quality and the relevance, including environmental conditions and graphite grade.

As is further discussed in Section 4.7, the quality of each source of graphite data will be assessed and the use justified, or additional work will be specified based on the risk significance of uncertainty due to quality. The irradiated data forms an important element of the predictive data set used for design analysis.

4.5.5 Required Material Properties for ASME Code Compliance

The material properties required for Code qualification of nuclear graphite according to the ASME Code are discussed below.

4.5.5.1 Dimensional Change

When irradiated in typical HTGR conditions, nuclear graphite initially shrinks then stops shrinking (dimensional change turnaround) before swelling, passing the initial dimension (dimensional change crossover) and continuing to expand up to high dose. The initial shrinkage rate, turnaround dose and crossover dose depend on the graphite grade, irradiation temperature and orientation relative to the forming direction. Dimensions are typically measured using a micrometer.

4.5.5.2 Creep Coefficient

Irradiation creep is defined as the difference in inelastic strain, between a stressed and unstressed region of equivalent graphite subject to the same irradiation conditions, in the irradiation conditions. Practical irradiation creep models include a large number of coefficients. In the Xe-100 GCA case, the implemented model coefficients include creep coefficients for primary, secondary, and recoverable creep, dose scale coefficients for primary and recoverable creep, and lateral strain ratios (‘creep Poisson’s ratio’) for each. See further discussion in Section 6.5 and Reference [21].

Several of these may be dependent on the stress sign (tension/compression), irradiation temperature and/or fast neutron dose. Creep model coefficients cannot be measured directly but must be inferred by fitting a model to experimental data.

In the ideal graphite irradiation creep experiment, creep (loaded) and control (free) specimens would be fabricated from identical material and irradiated in identical conditions except for the applied load. In practice, there is typically some variation in material properties and environmental conditions that complicates analysis of these experiments.

It is not possible to determine all of these properties from one experiment. A creep model must be formulated largely using historic data, with assessments accounting for uncertainty in the model and data. Some modern, grade-specific data should be obtained for model validation purposes, but it should be recognized that this may not span the full range of properties and irradiation conditions. The design analysis for the Xe-100 GCA will use a generic creep model, backed by extensive data from multiple nuclear graphite grades over a wide range of dose and temperature conditions, plus limited grade-specific data [[

]]^{P,E} that does not span the full reactor irradiation envelope.

High dose creep data is available [[

]]^{P,E}

As a deviation from the ASME Code requirements, the use of a generic creep model without fully enveloping grade-specific creep data will be justified by demonstrating that the effects of uncertainty can be tolerated.

4.5.5.3 Coefficient of Thermal Expansion

The Coefficient of Thermal Expansion (CTE) varies as a function of fast neutron dose, irradiation temperature and instantaneous temperature. The CTE is also affected by irradiation creep. There is initially some variation between orientations, although the change with irradiation can be treated as isotropic. [[

]]^{P,E}

Creep strain leads to CTE anisotropy in irradiated GCCs. Measurement temperature may be limited to below the irradiation temperature to avoid thermal annealing of irradiation damage (as allowed by the ASME Code). The CTE is typically measured using a pushrod dilatometer.

4.5.5.4 Strength

In the absence of oxidation, strength (tensile, bend and compressive) initially increases with irradiation and remains above the unirradiated value until beyond dimensional change crossover [19]. This pattern is seen for tensile, bend and compressive strength in both orientations. The POF methods in the ASME BPVC [3] do not currently credit the increase in strength with irradiation. Strength also increases with measurement temperature and is higher in a dry, inert atmosphere than typical laboratory conditions. In the context of material qualification, the primary purpose of irradiated strength data is therefore to demonstrate that the material does not fall below the initial strength within the reactor irradiation envelope. Otherwise, irradiated or high-temperature strength measurements are only required by the Code if the designer wishes to take advantage of the strength increase at low-to-intermediate doses. The effect of instantaneous temperature on strength is conservatively neglected. Unless otherwise specified, strength is measured at room temperature in air. As a destructive test, fewer strength measurements are typically performed than other measurements. Full-sized ASTM-compliant test specimens are generally too large for irradiation in high flux test reactors, so techniques such as the split disk test are often used instead.

4.5.5.5 Elastic Modulus

The change in elastic modulus (Dynamic Young's Modulus, DYM) with irradiation is required as a function of fast neutron dose and irradiation temperature. The effect of instantaneous temperature on DYM is neglected on the basis that strength increases by more than DYM with temperature, hence ignoring both remains conservative (irradiation-induced stress increases with DYM) [39]. Unless otherwise specified, DYM is measured at room temperature in air. To determine the change in DYM with irradiation, either ASTM C747 (resonant frequency) or ASTM C769 (sonic velocity) methods may be used. DYM measurements are required for both orientations unless it can be shown that the graphite is isotropic. [[ ]]P,E the effect of irradiation on elastic modulus is isotropic and data from both orientations can be used for model fitting.

4.5.5.6 Thermal Conductivity

The thermal conductivity of graphite decreases rapidly with irradiation, saturating before falling further beyond dimensional change turnaround. The initial rate of decrease in conductivity and the pre-turnaround saturated conductivity both depend strongly on the irradiation temperature, with lower irradiation temperatures being associated with a greater loss in conductivity. In unirradiated graphite, the thermal conductivity decreases with instantaneous temperature. The temperature/conductivity curve becomes flatter, or reverses, in irradiated graphite material. Measurement temperatures may be limited to the irradiation temperature to avoid annealing of irradiation damage. Thermal conductivity should be determined in both orientations unless it can be shown that the graphite is isotropic. For [[ ]]P,E the effect of irradiation on thermal conductivity is treated as isotropic and data from both orientations can be used for model fitting. Thermal conductivity is typically determined by measuring the thermal diffusivity using the laser flash method. Along with density and specific heat capacity (assumed not to change with irradiation) this is used to calculate conductivity.

4.5.6 Graphite Grades

Two graphite grades have been selected for the Xe-100 GCA: [[ ]]P,E These will experience different irradiation conditions in the reactor and the existing coverage of irradiation data is different. Table 8 below summarizes the data from external experimental programs, [[ ]]P,E

Table 8: *External Sources of Irradiated Graphite Data (No Direct X-energy Involvement)*

[[

]][P,E]

4.5.7 New Testing: X-energy/ORNL Experiments

[[

]]P,E The following subsections describe the proprietary testing that X-energy is pursuing at ORNL to obtain this additional irradiation data.

4.5.7.1 Material Properties and ASTM Standards

Table 9, below, shows the properties that will be measured, along with the ASTM standard (where relevant), ORNL procedure reference and planned irradiation temperature range. Note that the detailed requirements of the ASTM standards cannot always be met due to the small specimen size. A size effects study will be used to demonstrate that under-sized specimens give valid results.

Table 9: *Summary of ORNL Irradiation Program [[ ]]P,E*

Property	ASTM Reference	ORNL Standard Operating Guideline	Irradiation Temperature Range (°C)
Dimensions, bulk density	C559-16	MST-LAMDA- SOG-009	[[ ]]P,E
Coefficient of Thermal Expansion	E228-17 (C781-20 modification)	MST-LAMDA-SOG-005	[[ ]]P,E
Tensile strength by diametral compression	D8289-20	MST-LAMDA-SOG-008	[[ ]]P,E
Compressive strength	C695-21	MST-LAMDA-SOG-010	[[ ]]P,E
Specific heat capacity and Wigner energy (differential scanning calorimetry)	E1269-11, C1470-20	MST-LAMDA-SOG-006	[[ ]]P,E
Thermal diffusivity	E1461-13 (C781-20 modification)	MST-TPPL-SOG-001	[[ ]]P,E
Dynamic Young's Modulus (sonic velocity)	C769-15	MET-NMST-001	[[ ]]P,E

4.5.7.2 QA Arrangements

ORNL's quality controls for this irradiation program are described in their Quality Assurance Plan (Reviewed and approved by X-energy). As a national laboratory, ORNL cannot provide a certificate of conformance to ASME NQA-1 [23][23] and cannot be audited, but the work is planned to be conducted in accordance with the requirements established in their Quality Assurance Plan. X-energy is planning on conducting surveillance and survey activities to ensure compliance to the approved quality assurance program document.

4.5.7.3 Size Effects Study

While the specimens used for irradiation testing would ideally conform to the minimum sizes specified by the relevant ASTM standards, space limitations in the HFIR flux trap mean that smaller specimens must be used. To demonstrate that acceptable results can be obtained from sub-sized specimens, a size effects study is required. [[ ]]^{P,E}

4.5.8 Dose-Temperature Matrix and Data Coverage

As discussed in Section 3.3, the Xe-100 GCA experiences a wide range of irradiation temperatures, [[ ]]^{P,E}

The approach taken for obtaining data from across the full range of irradiation temperatures for the Xe-100 GCA is to use a combination of existing results from [[ ]]^{P,E} Even after these tests, there will be some gaps in the data compared to the expected operating conditions of the reactor.

The currently available and planned irradiation data (dose and irradiation temperature) for each orientation and each property are plotted in Reference [44] and summarized in Table 10 and Table 11 for [[ ]]^{P,E} respectively. The figures in Reference [44] show the dose-temperature envelope encompassing normal operating conditions at 25-100% power. The irradiation temperature range will be narrower during full-power operation. A summary of the gaps in available/proposed data compared to the full qualification envelope required for ASME Code compliance is provided in Table 10 and Table 11 for [[ ]]^{P,E} respectively. These also include possible routes to addressing these gaps, if it is determined that this is required to make a safety case. The referenced figures can be found in Reference [44]. Example plots (showing all properties combined) are reproduced in Figure *14* [[ ]]^{P,E} and Figure *15* [[ ]]^{P,E}.

Where the data does not fully envelope the reactor environment, limited extrapolation will be performed through analysis, using methods based on those described in Reference [21]. This relies on the general similarity of different grades of graphite and an extensive database of historical measurements. Uncertainty will be evaluated and used to investigate the range of possible results, which will then be used to investigate the effects of graphite properties uncertainty on RSFs. If analysis shows that uncertainty in irradiated graphite properties significantly affects the RSFs, further testing or limits on operation may be required.

[[

]]P,E

Figure 14: ***Overview of Data Coverage for [[ ]]P,E (Combined Properties)***

[[

]]P,E

Figure 15: ***Overview of Data Coverage for [[ ]]P,E (Combined Properties)***

Table 10: ***Review of ASME Code Compliance in Current and Planned Irradiated Data***

Property and References to Figures in [44]	Irradiation Temperature/Dose Range Not Covered	Possible Routes to Resolve Gap
Thermal Conductivity	[[ ]]P,E	[[ ]]P,E
Dimensional Change	[[ ]]P,E	[[ ]]P,E
Coefficient of Thermal Expansion Figure 5-16	[[ ]]P,E	[[ ]]P,E
Dynamic Young's Modulus Figure 5-21	[[ ]]P,E	[[ ]]P,E
Irradiation creep Figure 5-15 (creep) Figure 5-17 (creep-CTE)	[[ ]]P,E	[[ ]]P,E
Strength Figure 5-18 (compressive) Figure 5-19 (split disk, see note 2)	[[ ]]P,E	[[ ]]P,E
Notes: 1. [[2.]]P,E		

Table 11: ***Gaps to ASME Code Compliance in Irradiated Data***

Property and References to Figures in [44]	Irradiation Temperature/Dose Range Not Covered	Possible Routes to Resolve Gap
Dimensional change Figure 5-1 (AG) Figure 5-2 (WG)	[[ ]]^{P,E}	[[ ]]^{P,E}
Dynamic Young's Modulus Figure 5-10	[[ ]]^{P,E}	[[ ]]^{P,E}
Coefficient of Thermal Expansion Figure 5-4	[[ ]]^{P,E}	[[ ]]^{P,E}
Irradiation creep Figure 5-3 (creep) Figure 5-5 (creep-CTE)	[[ ]]^{P,E}	[[ ]]^{P,E}
Thermal conductivity Figure 5-9	[[ ]]^{P,E}	[[ ]]^{P,E}
Strength Figure 5-6 (compressive) Figure 5-7 (flexural) Figure 5-8 (split disk)	[[ ]]^{P,E}	[[ ]]^{P,E}

4.6 Oxidized Graphite

The effect of oxidation on structural graphite properties will be assessed in accordance with ASME BPVC III-5, HHA-3141 [3], for two scenarios, i.e., (1) chronic oxidation caused by trace moisture and air in the reactor system; and (2) acute oxidation caused by steam, e.g., in the postulated event of steam generator tube rupture. If the local oxidation weight loss is less than 1%, the effect of oxidation will not be considered based on the guidance of HHA-3141. If the local oxidation weight loss is greater than 1%, the effect of oxidation on graphite's strength, modulus, and thermal conductivity will be assessed. The local oxidation weight loss refers to the weight loss of the region of the graphite component affected by the oxidation rather the entire component.

The steam and air oxidation of [[ ]]^{P,E} graphite grades have been well studied and have data available. According to ASME BPVC III-5, HHA-III-5000 [3], use of historical data (published data) is allowed if the graphite is the same grade. The literature data will be utilized in the assessment if the reported oxidation conditions are conservative or equivalent to the oxidation condition in the Xe-100 reactor. Additional testing will be performed if such literature data are not sufficient.

Rate of oxidation and oxidation penetration depth are two important parameters to assess local oxidation weight loss. They are mainly determined by the temperature and concentration of oxidant supplied, e.g., O_2 and H_2O, for a given grade. As shown in Figure *16* taken from Reference [30], [[ ]]^{P,E} Therefore, the impact of oxidation to graphite integrity at different region of the reflector is not the same. The assessment of oxidation will be performed region-by-region based on the temperature profile as described in the Xe-100 GCA Oxidation Analysis Plan [30].

[[

]]^{P,E}

Figure 16: ***100% Power Temperature Profile of the Graphite Structures in the Xe-100 Reactor Assessment of Local Oxidation Weight Loss***

4.6.1 Assessment of Local Oxidation Weight Loss

The local weight loss will be assessed based on the rate of oxidation and oxidation penetration depth for [[ ]]^{P,E} at given oxidation conditions, e.g., temperature and oxidant concentration. Steam oxidation and air oxidation are assessed separately as they have different reaction mechanism.

For chronic steam oxidation, the analysis methodology is described in Reference [31]. The rate of oxidation of [[ ]]^{P,E} is predicted by using a Boltzmann-enhanced Langmuir Hinshelwood (BLH) model [[

]]^{P,E} This model is used to simulate oxidation due to the impurity (moisture) in the Xe-100 reactor He coolant gas. The input data are based on literature reported data for the same or similar grade of graphite [[ ]]^{P,E} The oxidation penetration depth is estimated based on the literature data and considering the scenario of Xe-100 reactor. Reference [34] confirmed that Boltzmann-Enhanced Langmuir Model (BLH model) can also be used to analyze the acute steam oxidation. The difference between chronic and acute oxidation is mainly the concentration of oxidant (low oxidant concentration for chronic oxidation). Therefore, the oxidant concentration and its flow velocities will be incorporated into the acute oxidation assessment [30].

For air oxidation, the analysis methodology is described in Reference [32]. The air oxidation rate and oxidation penetration depth are assessed based on literature data (mainly activation energy and pre-exponential factor) as summarized in Reference [32], [[ ]]^{P,E} Due to the very low oxidation rate at temperatures of 400°C and below, oxidation below 400°C is considered negligible.

Concerning the effect of irradiation on graphite oxidation, the research is in progress but there is lack of data, especially at high doses and for the graphite grades of interest. Based on the best available data, a methodology for analyzing integrated irradiation and oxidization effects [33] was developed by using [[ ]]^{P,E}, with an unverified assumption that the irradiation effects on oxidation remains the same across graphite grades. The rate of oxidation of irradiated [[ ]]^{P,E} can then be obtained based on their unirradiated oxidation rate and a rate multiplier accounting for the irradiation effect generated from [[ ]]^{P,E} However, the proposed methodology has some limitations, e.g., the rate multipliers extrapolated beyond 6.5 dpa is unverified, as it does not consider the effect of graphite structure change after turnaround on the oxidation rate [33]. The significance of these limitations will be evaluated when the oxidation methodology is applied to the specific oxidation and irradiation conditions relevant to the Xe-100.

4.6.2 Assessment of Property Change of Oxidized Graphite

The change of graphite properties will be assessed as needed, only if the local oxidation weight loss is found to be greater than 1%, which is the threshold over which the graphite material is considered oxidized per ASME BPVC III-5 HHA-3141 [3]. Further discussion regarding the design methodology to evaluate oxidation is available in Section 6.6.2. Per HHA-2230, the affected properties are strength, elastic modulus, and thermal conductivity. The changes in strength, elastic modulus and thermal conductivity are discussed further in Section 5.2 of Reference [39].

4.7 Graded Approach to Quality for Graphite Design Data

This section provides a summary of X-energy's planned approach for assessment of quality of design data of the GCA and the GCCs within the GCA of the Xe-100 reactor, using a performance-based graded approach to qualification and use of such data commensurate with the contribution to safety.

4.7.1 Industry Consensus Opinion & Basis for Exception

ASME BPVC Section III, Division 5 is a nascent consensus code and standard supported by volunteer experts in their field. Discussions with industry experts have confirmed broad agreement across industry stakeholders that the current language in the ASME BPVC is ambiguous with respect to the quality requirements for design data (including as-manufactured, irradiated, and oxidized data) and may have unintended constraints on its users. The ambiguity stems from reference to general requirements from HHA-2000, invoking HHA-III-2000 "Design data" which appears to require this data to be "generated under a QA program that complies with HAB-4000 or HAB-3800." HAB-4000 is focused on quality requirements for design and construction activities while HAB-3800 is focused on quality requirements for the material organization.

Discussions with industry experts confirm agreement that the underlying intent of the HAB section is to control manufacturing activities associated with data for the graphite that will be used in a GCA, not all data used to inform the design (BPV III-5-24-04, See Reference [54]). There is broad agreement that when environmental deterioration factors such as irradiation, oxidation and/or chemical attack type conditions are considered, there should be more liberty in the ASME BPVC on ability to use a broader set of quality controls than implied in HHA-2000. This supports X-energy's performance-based exception from a literal interpretation of the ambiguous language in HHA-III-2000, and plan to develop a graded approach to quality of design data based on significance to safety, discussed further herein.

4.7.2 NEI 18-04 Application to Assessing Quality of Design Data

As noted elsewhere in this document, the GCA supports performance of certain RSFs, which are variously supported by individual GCCs, some of which support the SR functions. As part of this development work, it is expected that the quality requirements associated with the data used for design will be graded commensurate with their significance to the GCA performing the RSFs (i.e., a performance-based approach). Such a graded approach is supported by the NEI 18-04 framework, with relevant excerpts as follows:

- *"10 CFR 50 Appendix B Quality Assurance Program – For SR SSCs, QA requirements consistent with 10 CFR 50 Appendix B <u>should be risk-informed and performance-based and not compliance-based</u>; guidance in SRP 17.5 Quality Assurance for safety-related SSCs, 10 CFR 50.69, SRP 1.201."*
- *"User provided Quality Assurance (QA) Program for non-safety SSCs – For NSRST SSCs, QA requirements consistent with SRP 17.4 (Reliability Assurance Program) for non-safety-related, safety significant SSCs <u>should be risk-informed and performance-based and not compliance based</u>; guidance in SRP 17.5 Quality Assurance for non-safety-related SSCs, 10 CFR 50.69, SRP 1.201."*

The reliable performance of the components in the core is principally expected to be ensured by rigorous (e.g., NQA-1) testing of inspection data (specimens from as-manufactured billets which will form the components that will be put into the core). Structural reliability is largely supported by preventing substandard material from being present inside the reactor. This is distinguished as unique from predictive data (from a range of graphite grades including as-manufactured, irradiated, and oxidized which supports the prediction of future operational performance behavior).

ASME NQA-1 presents several methods for Commercial Grade Dedication. However, applying these acceptance methods might not be possible or practical for some of the available historical and representative data which is believed to be useful in design analysis. Depending on their use case and the data available, it is expected that a range of consensus quality standards can provide a sufficient level of quality appropriate for use of the data in performance-based design. An approach to a graded application of qualification will be developed which examines the consensus standards and processes available and applies the philosophy of NEI 18-04 to select appropriate measures for qualification and application of the data based on the source, type, use, and significance of the data. Additionally, data uncertainty will be accounted for as one element of uncertainty analysis (see Section 6.4.1).

5. Graphite Test Plan

5.1 Introduction

5.1.1 Background

As previously discussed in Section 3.3, the environmental conditions in the Xe-100 reactor include high dose fast neutron irradiation, high temperatures and the presence of low concentrations of oxidizing impurities, all of which may affect the properties of nuclear graphite. The GCA may be subject to a range of loading mechanisms in normal operation and during LBEs, including loads due to fuel and gas pressure, thermal transients, and seismic events.

X-energy has adopted a risk-informed approach to GCA design, in which the strength of evidence required to support a claimed function should be commensurate with the risk significance of that function. A broad range of experimental data already exists for nuclear graphite, which has been used in conceptual design activities. The need for new data depends on the quality and relevance of existing data, as well as the potential safety consequences of errors that could result from inadequate data.

5.1.2 Purpose of Testing Plan

The purpose of the testing plan is to summarize the comprehensive test needs for the design and materials of the GCA and any applicable GCCs. There are likely to be many instances where testing is not required; however, the needs for each individual test may be influenced by design analysis results which are not yet available. In those instances, the testing plan will provide criteria against which testing needs will be weighed.

5.1.3 Scope

The scope of the graphite test plan documented in Reference [38] is limited to testing needs of the GCA, individual GCCs, and the material of which both are comprised. This may include material property, design Verification and Validation (V&V) (to confirm that design analysis has been implemented correctly and models the correct phenomena), design development, and component functional testing.

5.1.4 ASME Code Requirements

While the testing requirements for the Xe-100 GCA are derived from the need to demonstrate safety and reliability of the GCA, many of the planned material qualification activities are based on the requirements of the ASME BPVC (The ASME code requirements are discussed in more detail in Reference [39], while the differences between the (NRC-endorsed) 2017 ASME code version and the (current) 2023 version are reviewed in Reference [15]).

5.1.5 Limitations

The results presented herein are preliminary. They are not intended to commit to performing specific tests and the conclusions presented herein are subject to change as the design of the GCA develops.

5.1.6 Summary of Testing Plans

An extensive list of tests for the GCA, the GCCs of which it is comprised, and the material from which it is manufactured is discussed herein. Performing all of these tests is likely not required to produce a safe and reliable nuclear power plant. Table 12 provides a comprehensive summary of the GCA-related tests, a status of the testing or plans for testing, and reference to source documentation for the testing in question.

The status for each test is summarized as follows:

- ***In Progress*** – Testing is currently in progress.
- ***Planned*** – It has been decided that testing is required, but the testing either has not started yet or is in the planning process.
- ***To Be Confirmed (TBC)*** – The potential for testing is acknowledged, but the requirement for testing is not yet understood. This test item may or may not be required as more information is available through the course of design development.
- ***Not Required*** – This testing was identified as an item which may contribute to the overall design and qualification of the GCA or its GCCs; however, through engineering justification, the testing was deemed to not be required for design development of qualification of the GCA.

Table 12: *Graphite Test Summary*

#	Test Name	Test Details	Test Category	Status	Source
1	[[ ]]^{P,E} Irradiated Material Testing	[[ ]]^{P,E}	Material Property Testing	In Progress (Planned to conclude in 2025)	[35] [39]
		[[ ]]^{P,E}	Material Property Testing	In Progress (Planned to conclude in 2029)	[35] [39]
		[[ ]]^{P,E}	Material Property Testing	In Progress (Results anticipated through mid-late 2020s)	[35] [39]
		[[ ]]^{P,E}	Material Property Testing	Preliminary Discussions in Progress, TBC	[39]
		[[ ]]^{P,E}	Material Property Testing	Preliminary Discussions in Progress, TBC	[39]
		[[ ]]^{P,E}	Material Property Testing	Preliminary Discussions in Progress, TBC	[39]
2	[[ ]]^{P,E} As-Manufactured Material Testing	Mechanical and thermal properties:	Material Property Testing	Planned	[39]

#	Test Name	Test Details	Test Category	Status	Source
		• Sampling 12 billets from three different graphitization runs. 288 specimens for each property testing. See Figure *13* for details. Representative sampling of billet with specimens in With Grain (WG) and Against Grain (AG) orientations. • Testing matrix, see • Table 7, including strengths (tensile, flexural, and compressive), DYM, CTE and thermal conductivity.			
3	[[ ]]^{P,E} Oxidation Material Testing	Rate of oxidation and oxidation penetration depth (conditions relevant to chronic and acute oxidation by steam and air) – as required, depending on results of conservative predictions of oxidation. Oxidized material properties: effect of weight loss on thermal conductivity of [[ ]]^{P,E} Plan to use historical data for other properties.	Material Property Testing	Planned	[39]
4	[[ ]]^{P,E} Oxidation Testing: Irradiation Effects	Rate of oxidation of irradiated graphite in steam and air. Rate of oxidation of irradiated graphite in steam and air with simultaneous irradiation.	Material Property Testing	Not Planned	[33]
5	[[ ]]^{P,E} Irradiated Material Testing	In the reactor operating envelope, it is planned to use historical data without additional testing. This will be supplemented by data from higher temperature tests (beyond the reactor envelope and not required for qualification), [[ ]]^{P,E}	Material Property Testing	In Progress	[39]

#	Test Name	Test Details	Test Category	Status	Source
		[[ ]]^{P,E} irradiation testing to resolve gaps in qualification data (see 5.2.3), particularly at low irradiation temperatures and highest doses.	Material Property Testing	Not planned, need TBC	[39]
6	[[ ]]^{P,E} As-Manufactured Material Testing	Mechanical and thermal properties: • Sampling 12 billets from three different graphitization runs. 288 specimens for each property testing. See Figure *12* for details. Representative sampling of billet with specimens in WG and AG orientations. • Testing matrix, see • Table 7 including strengths (tensile, flexural, and compressive), DYM, CTE and thermal conductivity.	Material Property Testing	Planned	[39]
7	[[ ]]^{P,E} Oxidation Material Testing	Plan to use historical data only.	Material Property Testing	Not planned	[39]
8	[[ ]]^{P,E} Oxidation Testing: Irradiation Effects	Rate of oxidation of irradiated graphite in steam and air. Rate of oxidation of irradiated graphite in steam and air with simultaneous irradiation.	Material Property Testing	Not Planned	[33]
9	Abrasion/Friction Testing	Graphite hardness testing to inform abrasion test plans. Graphite abrasion testing: [[ ]]^{P,E}	Material Property Testing	Planned	[36] Appendix C

#	Test Name	Test Details	Test Category	Status	Source
		Measure friction coefficient and wear rate in representative conditions (temperature, pressure, gas composition).			
10	Seismic Assembly Benchmarking Testing	Dynamic testing of assemblies of multiple GCCs to validate seismic modelling methodology. May use scale models, not necessarily fabricated from graphite.	GCA Testing	TBC	[37]
11	Fatigue Testing	Based on analysis of historic data in [19], it is concluded that fatigue testing is not required.	Material Property Testing	TBC	[19]
12	Fracture Testing	The Crack Propagation Analysis Methodology [29] identifies suitable literature values for fracture toughness of graphite. Based on the current assessment of the risk significance of graphite fracture, it is judged that no additional testing is required.	Material Property Testing	TBC	[29]
13	Erosion Testing	Data being evaluated to determine testing need.	Material Property Testing	TBC	N/A
14	Installation Testing	Test-assembly of all or part of the GCA may be performed to for commercial reasons but it is not required for nuclear safety.	GCA Testing	Not required for safety	N/A
15	GCA Control Rod Entry Testing	Test insertion of control rod into representative control rod channel model to validate rod entry assessment in distorted/irradiated GCA (Required Safety Function 1.1.2).	GCA Testing	Not required, plan to use simulation	N/A
16	GCA Heat Transfer Testing	Test transfer of heat through representative GCA sub-assembly to validate heat removal assessments (Required Safety Function 1.2.1)	GCA Testing	Not required, plan to use simulation	N/A
17	Leakage and Bypass Flow Testing	Test of gas flow through representative GCA sub-assemblies or component interface features to validate	GCA Testing	TBC	[24]

#	Test Name	Test Details	Test Category	Status	Source
		leakage and bypass flow predictions outlined in reference [24] [24].			
18	GCC/GCC Feature Destructive Testing	Fracture of full-sized GCCs or GCC sub-features to determine load-bearing capacity and/or validate failure criteria for cracking assessments.	GCC Testing	Not required (design by analysis is conservative)	[29]

5.2 Discussion

Graphite testing is split into multiple categories including the following: (1) Material Property Testing, (2) GCA Testing, and (3) GCC testing. These categories of test scope are associated with each of the tests identified in Table 12, above. The discussion herein provides more context and background for each of the various tests identified in Table 12.

5.2.1 Material Property Testing

Material property testing is planned to provide inputs to the reactor design and analysis and to verify that manufactured material conforms to requirements. Material property tests are summarized in Sections 5.2.2, 5.2.3 and 5.2.4 and discussed in the references in the associated sections.

5.2.2 As-Manufactured Graphite Properties

‘As-manufactured properties’ refers to the unirradiated properties of the graphite grade intended for use in the reactors. As-manufactured qualification testing is required to define the acceptable range of properties for the manufacturing graphite billet material specification [40]. Inspection testing is carried out on a sample of material from each batch to confirm that the material meets the specification. Requirements for as-manufactured qualification and inspection testing may be found in Reference [39] and Reference [40].

5.2.3 Irradiated Graphite Properties

Fast neutron irradiation in reactor conditions causes dimensional change and changes to many material properties of nuclear graphite. A wide range of generically applicable and grade-specific data is already available, but some additional data must be obtained for material qualification and to reduce uncertainty in certain areas. The qualification requirements for irradiated properties and details of the current test program are set out in Reference [39]. Additional testing needs beyond those currently underway are summarized below:

[[

]]^{P,E}

Generic Irradiated Properties (Not Grade-Specific)

Wigner energy at intermediate irradiation temperatures: the accumulation of Wigner energy during irradiation at low temperatures (<200°C) is recognized as a potential issue in the ASME BPVC. A broader review of the phenomenon [45] suggests that Wigner energy may be significant even at lower levels that could accumulate during irradiation at up to around 300°C. X-energy and ORNL's current irradiation testing program [35] plans to measure Wigner energy in low-temperature specimens only (nominal irradiation temperature ~200°C). Depending on the results of preliminary safety analysis, testing at higher temperatures might be beneficial. See similar discussion in Sections 4.5.7 and 6.6.1.

5.2.4 Graphite Oxidation

Oxidation Rate and Penetration Depth in Steam and Air

While conservative, high-level predictions of the total oxidation can be obtained by considering the amount of available oxidant, detailed predictions of acute and chronic oxidation would need test data for the oxidation of graphite by steam or oxygen in relevant conditions. Both the rate of oxidation and the spatial distribution (penetration depth) need to be considered. There is some evidence that irradiation affects oxidation (see next section), which may need to be accounted for in predictions. Oxidation testing plans are discussed in Reference [39] and analysis methods are discussed in References [31], [33], [34], and [41][41]. The need for new testing will depend on expected concentrations of oxidizing species in the reactor coolant and the assessed safety consequences of graphite oxidation.

Effect of Irradiation on Oxidation Rate

There is some evidence that prior irradiation may increase the rate of graphite oxidation (in the absence of simultaneous irradiation), summarized in Reference [33]. Simultaneous irradiation may also increase the rate of graphite weight loss (radiolytic oxidation). The risk-significance of these phenomena depends on the spatial distribution of irradiation dose and temperature. With the highest dose occurring in relatively low temperature regions of the core, it may be that the effects of irradiation on peak graphite weight loss and total weight loss are not detrimental to the performance of the GCA. Currently X-energy does not plan to test oxidation of irradiated graphite or simultaneous oxidation and irradiation. The potential effects of the relevant phenomena should be considered in analysis and the need for testing should be reviewed as safety analysis results are obtained.

[[

]]^{P,E}

5.2.5 Additional Testing Programs

Further discussion of testing is marked as “TBC” in the table above is outlined in detail in Reference [38]. Future revisions of this LTR will incorporate a comprehensive set of testing programs which are established for the GCA in the XE-100.

5.3 Conclusions

Potential areas for GCA-related testing have been summarized, based on the assumption that full compliance with the requirements of ASME BPVC Section III Division 5 will not be achieved and will not be required to justify the safety of the Xe-100 reactor. A combination of new testing and historical data will be used to substantiate the design and safety analysis.

Unless otherwise stated in the Graphite Material Qualification Plan [39], ASME Code requirements will be met for the as-manufactured material.

6. Design Methodology

6.1 Introduction

The design by analysis methodology developed for the Xe-100 GCA and its GCCs intends to design the GCA to be able to support the RSFs and associated RFDCs through the life of the reactor. This design methodology is documented in detail in Reference [14].

The methodology for designing the GCA and its GCCs addresses:

- All known material degradation mechanisms that the GCA and its GCCs experience during normal operation, DBEs, and DBAs: irradiation effects, oxidation, abrasion, erosion, and fatigue. These degradation mechanisms are discussed further in Section 6.6.
- All known loading conditions that the GCCs are subjected to during normal operations of the plant and all LBEs (i.e., AOOs, DBEs, and DBAs).

As discussed in earlier sections, RFDC 1.4.1 associated with "maintain core geometry" serves to "(1) ensure geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) permit sufficient insertion of neutron absorbers and inherent reactivity feedback to provide safe shutdown of the reactor during DBEs and DBAs." The RFDC "maintain core geometry" thus deconvolves to the GCA's RSF 1.4.1.3, "Maintain acceptable GCA geometry" which principally supports demonstrating the other RSFs of "control reactivity" and "control heat removal." Therefore, the current stage in design is mainly focused on implementing a structural design methodology which ensures that the GCA and its GCCs are designed to withstand stresses developed during normal operation and applicable LBEs and ensure acceptable geometry is maintained, as the first and most fundamental step towards demonstration that RSFs associated with "control reactivity" and "heat removal" are adequately performed. The design methodology to address RSF 1.1.2.1c (directly related to "control reactivity") and RSF 1.2.1.1c (directly related to "control heat removal") and associated acceptance criteria will be formally developed and documented in the detailed stage of design.

The structural design methodology developed for the Xe-100 leverages as guidance the structural design methodology defined in Subsection HH, Subpart A (HHA) of ASME BPVC III-5, 2023 Edition [3], with augmentations and modifications discussed in Section 6.3.

While the design methodology documented in Reference [14] provides a roadmap of all appropriate documents applicable to the design of the GCA and its GCCs, the Design Specification for the GCA is intended to be the document that enforces and controls the design process for the GCA and its GCCs.

The following subsections are organized as follows:

- Section 6.2 provides a brief introduction of the ASME BPVC III-5 design methodology (HHA-3000) that is leveraged in the Xe-100 GCA design and provides the background needed to discuss the differences between the Xe-100 GCA structural design methodology and the ASME BPVC III-5 design methodology.

- Section 6.3 summarizes (1) the differences between the Xe-100 GCA structural design methodology and the ASME BPVC III-5, 2023 Edition design methodology, (2) the changes in design methodology made between ASME BPVC III-5, 2017 Edition and ASME BPVC III-5, 2023 Edition, and (3) exceptions and limitations from NRC's endorsement of ASME BPVC III-5, 2017 Edition.
- Section 6.4 discusses the stress analysis methodology that is part of the overall design methodology documented in Reference [14] which will be used to demonstrate that stresses are limited to acceptable levels.
- Section 6.5 discusses a material model named "Irradiated Graphite Numerical Iterative Solver" (IGNIS) that is being developed in phases to take the material data available and apply it for use in the stress analysis to support the Xe-100 GCA/GCC design.
- Section 6.6 presents all known material degradation mechanisms that the GCA and its GCCs experience during normal operation, DBEs, and DBAs: irradiation effects, oxidation, abrasion, erosion, and fatigue.
- Section 6.7 summarizes the preliminary acceptance criteria applicable to the GCA, and GCCs that are defined to ensure that the GCA successfully supports the applicable RSFs.

6.2 Overview of the ASME BPVC III-5 Design Methodology (HHA-3000)

This subsection provides a brief introduction of the ASME BPVC III-5 design methodology (HHA-3000), which is discussed further in Reference [14].

Article HHA-3000[2] of the ASME BPVC III-5 provides requirements for:

(1) the design of the GCA as a whole (Subarticle HHA-3300), and

(2) the design of the individual GCCs comprising the GCA (Subarticle HHA-3200).

Subarticle HHA-3200 defines the requirements for the design of GCCs and allows for the following three alternative design approaches:

a. Simplified Assessment (HHA-3220) – Design of GCCs to meet the structural reliability targets based on stress limits derived from the material reliability curve.

b. Full Assessment (HHA-3230) – Design of GCCs to meet the structural reliability targets based on calculated reliability values derived from the distribution of stresses in the GCC and the material reliability curve.

c. Design by Test (HHA-3240) – Design of GCCs that meet the structural reliability targets based on experimental proof of the component performance.

The first two approaches are design-by-analysis approaches while the third approach is a design-by-test approach. Only the two design-by-analysis approaches are planned to be used to design the GCCs for the

[2] All mentions of HHA-XXXX refer to the ASME BPVC III-5 in Reference [3] unless specifically noted.

Xe-100. It is anticipated that the full assessment will be used for the majority of the GCCs. However, where appropriate, the simplified assessment may be used.

These two design-by-analysis approaches require the use of stress analysis to characterize the stress states under the various loading conditions and degradation mechanisms that the GCCs experience. Discussion of the stress analysis methodology is provided in Section 6.4.

An overview of the design-by-analysis methodologies is provided in Figure 17.

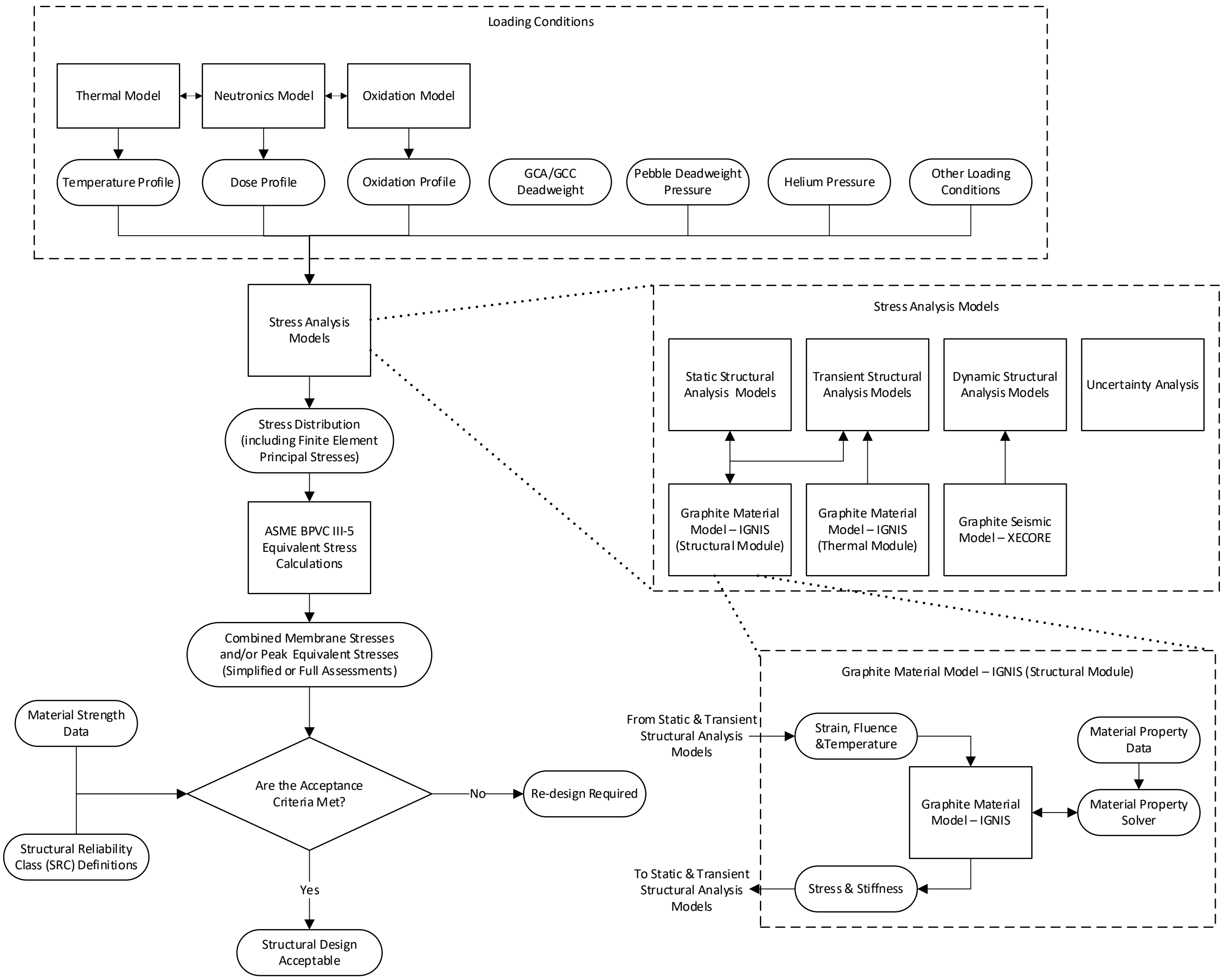

Figure 17: ***Overview of the Design-by-Analysis Methodology***

6.2.1 Overview of Full Assessment (HHA-3230) and Calculation of Probability of Failure

The full assessment (HHA-3230) requires the calculation of POF using the procedure described in HHA-3217 and uses a three-parameter Weibull distribution to describe the material reliability curve (i.e., distribution of tensile strength measurements). With the three-parameter Weibull distribution, the typical equation for the POF is as follows (HHA-II-3200).

$$POF(S) = 1 - exp\left[-\left(\frac{S-S_0}{S_c-S_0}\right)^m\right]$$

Where:

S = material stress.

S_0 = threshold stress (defined as such in the 2017, 2019, and 2021 Editions (References [9], [11], and [12], respectively) and defined as S' in the 2023 Edition [3].

S_c = characteristic strength (defined as $S_{c095\%}$ in the 2017, 2019, and 2021 Editions (References [9], [11], and [12], respectively) and defined as $S'_{c0.05}$ in the 2023 Edition [3].

S_c-S_0 = adjusted characteristic stress (also referred to as scale parameter).

m = Weibull modulus (shape parameter), (defined as $m_{095\%}$ in the 2017, 2019, and 2021 Editions (References [9], [11], and [12][12], respectively) and defined as $m'_{0.05}$ in the 2023 Edition [3].

The steps for calculating the POF are specified in HHA-3217. These steps are summarized in Figure 18.

Two blocks are highlighted in red in Figure 18 because they are critical to the POF methodology:

- Condition 1 for the process zone volume, V_m.
- Condition 2 for the stress range parameter, Δ.

Section 4 of Reference [16] provides detailed background information related to the development of the full assessment methodology (including the POF methodology) and discusses that the full assessment methodology was originally verified using experimental test case results on specimens constructed and fabricated from medium-grain-size graphite [[ ]]P,E. Subsequently, it was found that this definition of the process zone volume (V_m) based only on medium-grain graphite grades results in unrealistically small values for fine-grain graphite grades and the full assessment methodology was overly conservative for these grades. In an attempt to make the full assessment methodology more appropriate for fine-grain graphite grades, the definitions of the process zone volume (V_m) and the stress range parameter (Δ) were changed in ASME BPVC III-5, 2021 Edition [12], as summarized in Table 13. The process zone volume (V_m) was defined based on the fracture toughness (i.e., K_{IC}). However, the change introduced in the 2021 Edition resulted in smaller process zone volumes for medium-grain-size graphite and, as such, made the full assessment methodology less appropriate for those grades.[3]

[[

]][P,E]

Furthermore, the calculation of POF values will be performed with the Weibull shape parameter scaling step. See further discussion of the Weibull shape parameter scaling step in Reference [17]. Reference [17] indicates that the simplified assessment is more conservative than the full assessment provided the Weibull distribution shape parameter is appropriately scaled in the full assessment.

[3] A change in ASME BPVC III-5 that intends to address this issue by introducing a single and consistent approach for all graphite grades is currently in progress. X-energy may elect to implement that updated methodology once it is implemented in ASME BPVC III-5 itself.

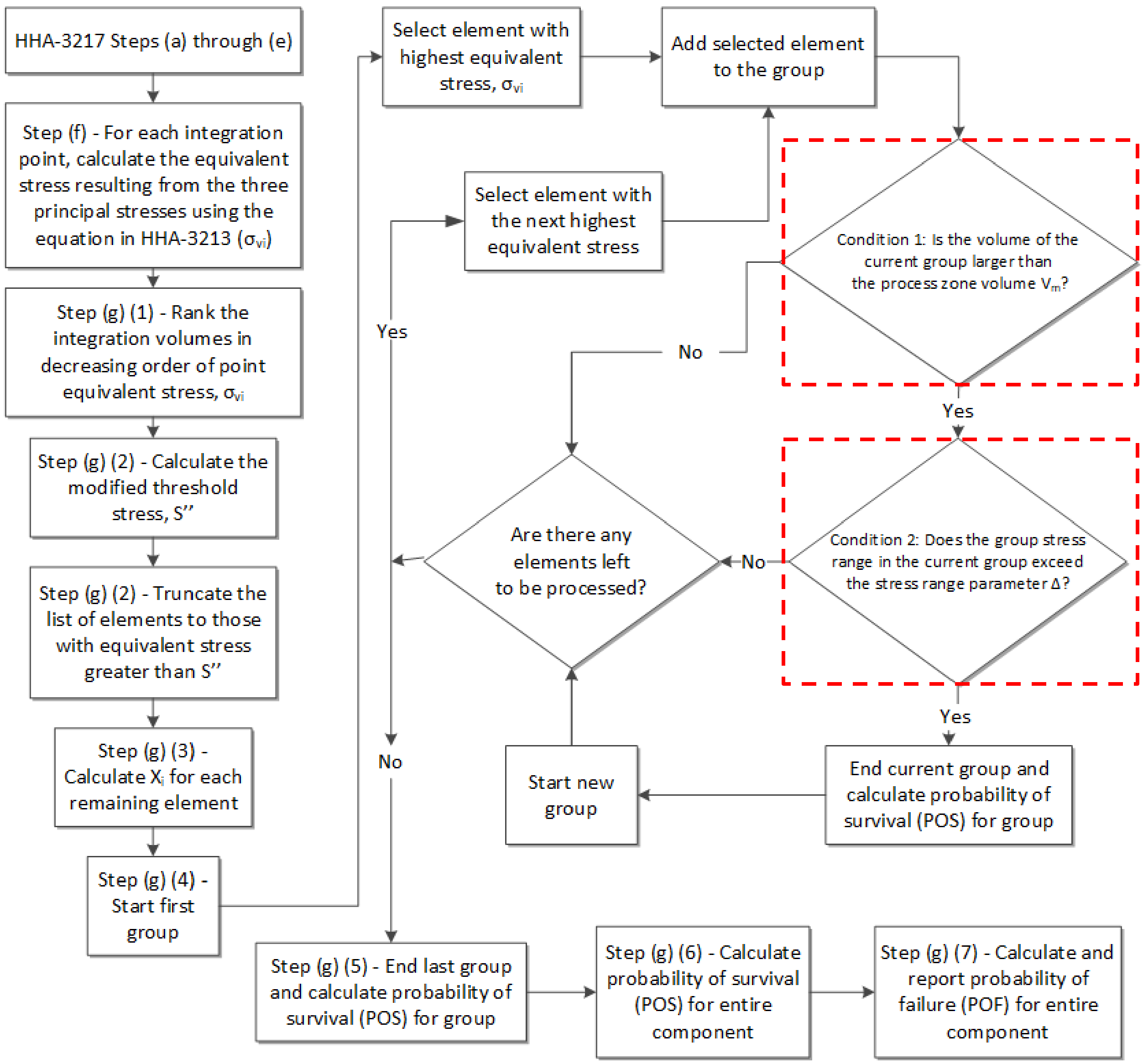

Figure 18: ***Flow Chart for HHA-3217 Calculation of Probability of Failure***

Table 13: *Summary of Process Zone Volume (V_m) and Stress Range Parameter (Δ)*

Edition of the ASME BPVC III-5	Process Zone Volume, V_m	Stress Range Parameter, Δ
2017 and 2019	$V_m = (Grain\ size \times 10)^3$	7%
2021 and 2023	$V_m = \left(\frac{\pi}{2} \times \left(\frac{K_{IC}}{\sigma_m}\right)^2\right)^3$	20%

Note: K_{IC} is the critical or reference stress intensity factor for mode I and σ_m is the mean tensile strength.

6.2.2 Material Properties

Appropriate material property inputs are needed for the structural design methodology. HHA-3227 indicates that material property requirements for design, and for material property deterioration during service, are given in HHA-2200. These material property requirements include as-manufactured material properties, irradiated material properties and oxidized material properties as detailed in Mandatory Appendix HHA-II. These appropriate material property for design are documented in the Material Data Sheet (MDS). Discussion of the material property data requirements from ASME BPVC III-5 (e.g., HHA-2000) and the graphite material qualification methodology for the Xe-100 is provided in Section 4 of this LTR.

6.3 Differences, Changes, Exceptions, and Limitations for Structural Design Methodology

The following subsections discuss:

- The differences between the structural design methodology developed for the Xe-100 GCA and ASME BPVC III-5, 2023 Edition design methodology – Section 6.3.1.
- Changes in design methodology made between ASME BPVC III-5, 2017 Edition and ASME BPVC III-5, 2023 Edition – Section 6.3.2 0.
- Exceptions and limitations from NRC's endorsement of ASME BPVC III-5, 2017 Edition – Section 6.3.3.

6.3.1 Differences between the Xe-100 GCA Structural Design Methodology and ASME BPVC III-5, 2023 Edition HHA Design Methodology

As discussed in introduction in Section 6.1, the design methodology defined in HHA-3000 of ASME BPVC III-5, 2023 Edition, is leveraged as guidance for the design methodology of the GCA and its GCCs for the Xe-100 design. The ASME BPVC III-5 design methodology is augmented and modified as summarized in Table 14.

Table *14*: *Differences Between the Xe-100 GCA Design Methodology and HHA-3000 of ASME BPVC III-5 (2023 Edition)*

Provision from HHA-3000 of ASME BPVC III-5 (2023 Edition)	Summary of the Approach for the Xe-100 GCA (see Section # for Further Detail)	Section # in this Document
HHA-3217, Calculation of Probability of Failure	[[ ]]P The calculation of POF values will be performed with the Weibull shape parameter scaling step. See further discussion of the Weibull shape parameter scaling step in Reference [17].[5]	6.2.1
HHA-3142.2, Wigner Energy	HHA-3142.2 requires that, for graphite irradiated to damage dose greater than 0.25 dpa at a temperature <200°C, the effect of stored (Wigner) energy buildup shall be accounted for when evaluating reactor thermal transients. [[ ]]P,E However, as discussed in Section 3.1.2 of Reference [14], Wigner energy accumulates even when graphite is irradiated at higher temperatures. The release of Wigner energy in graphite irradiated to high dose in HTGR conditions remains an area of uncertainty. X-energy's irradiation program at ORNL will test the release of Wigner energy in graphite irradiated to high dose at around 200°C (see Section 4.5.7) and the significance of Wigner energy for safety in the Xe-100 will be evaluated.	6.6.1

6.3.2 Changes in Design Methodology between ASME BPVC III-5, 2017 Edition and ASME BPVC III-5, 2023 Edition

The purpose of this subsection is to discuss the key changes made in the design methodology between the ASME BPVC III-5, 2017 Edition and the ASME BPVC III-5, 2023 Edition [3] based on the effort documented in Reference [15].

The key changes are summarized as follows:

- Change in the definition of the process zone volume and stress range parameter, both used in the full assessment methodology: HHA-3130 and HHA-3217.

[4] A change in ASME BPVC III-5 that intends to address this issue by introducing a single and consistent approach for all graphite grades is currently in progress. X-energy may elect to implement that updated methodology once it is implemented in ASME BPVC III-5 itself.

[5] A change in ASME BPVC III-5 that intends to address this issue is currently in progress.

- Minor nomenclature changes in the simplified and full assessment methodologies: HHA-3237, HHA-II-3100, HHA-II-3200, HHA-II-3300.
- Requirement for radiolytic oxidation evaluation was removed: HHA-2230 and HHA-3140. In any case, all effects of oxidation (both thermal and radiolytic) are considered and evaluated, as discussed further in Section 6.6.2.

6.3.3 Exceptions and Limitations from NRC's Endorsement of HHA-3000

The ASME BPVC III-5, 2017 Edition [9] was endorsed by the NRC in RG 1.87, Revision 2 [4]. The endorsement from the NRC includes several exceptions and limitations. The exceptions that directly impact the design methodology from ASME BPVC III-5 (i.e., HHA-3000), and the approach to address each NRC exception and limitation associated with HHA-3000 for the Xe-100 GCA are summarized in Table 15. The table also provide the relevant comments from the NRC in NUREG-2245 [10], which documents the NRC staff's review of the 2017 Edition.

Table 15: *Exceptions Taken by the NRC on ASME BPVC III-5, HHA-3000 (2017)*

Provision from ASME BPVC III-5, HHA-3000	Comment from the NRC in RG 1.87, Revision 2 [4]	Comment from the NRC in NUREG-2245 [10]	Approach to Address the NRC's Exception	Section # in this Document
HHA-3141, Oxidation	The NRC staff is not endorsing the provisions of HHA-3141(c) that set the weight loss limit as 30 percent for geometry reduction in the oxidation analysis. Designers should determine the amount of weight loss above which the region should be regarded as completely removed from the structure and justify that the limit is adequate for the design-specific oxidation analysis.	The staff finds HHA-3141(c) to be acceptable because it is appropriate to place an upper limit on the amount of weight loss allowed in the oxidation analysis. During the review, the staff commented to the ASME Working Group for Graphite Core Components on the amount of weight loss allowed by HHA-3141(c). Specifically, HHA-3141(c) could be interpreted to allow up to a 30-percent reduction in the cross-sectional geometry of a component with oxidation and doing so could be detrimental to maintaining the structural integrity of the GCCs. The Working Group is still developing its formal response to the comment but has informed the staff that it agrees and is making plans to address it. The staff determined that HHA-3141(c) is acceptable despite the open comment because the 30-percent limit in HHA-3141(c) does not relieve the Designer of the responsibility to ensure the integrity and functionality of the individual GCCs, and of the GCA, in its design, taking due account of irradiation and oxidation. However, the staff also notes that the limit may not be generically applicable to all high-temperature reactor designs and therefore is not endorsing the provisions of HHA-3141(c) that set the weight loss limit as 30 percent for geometry reduction in the oxidation analysis. As such, designers should determine the amount of weight loss above which the region should be regarded as completely removed from the structure and justify that the limit is adequate for the design specific oxidation analysis.	An analysis plan has been developed for the Xe-100 GCA for determining the extent of graphite mass loss due to oxidation during normal operation (chronic) and after postulated accident scenarios (acute). A weight loss limit (defined such that any region that experiences weight loss above this amount needs to be considered as completely removed from the structure) may be eventually defined depending on the results of the analysis.	6.6.2
HHA-3142.4, Graphite Cohesive Life Limit	The NRC staff is not endorsing the provisions of HHA-3142.4 that set the graphite cohesive life limit fluence to the fluence at which the material experiences a +10 percent linear dimensional change in the with-grain direction. Designers should determine the graphite cohesive life fluence limit beyond which the material is considered to provide no contribution to the structural performance of the Graphite Core Component (GCC) and justify that the limit is adequate for the GCC design.	The staff finds HHA-3142.4 to be acceptable because the staff agrees with the approach of setting a fluence limit beyond which the material is considered to provide no contribution to the structural performance of the GCC. However, the staff also notes that a +10-percent linear dimensional change limit may not be generically applicable to all high-temperature reactor designs. Therefore, the staff is not endorsing the provisions of HHA-3142.4 that set the graphite cohesive life limit fluence to the fluence at which the material experiences a +10 percent linear dimensional change in the with-grain direction. Designers should determine the graphite cohesive life fluence limit beyond which the material is considered to provide no contribution to the structural performance of the GCC and justify that the limit is adequate for the GCC design.	The fluence distributions at 100% power for 30-year and 60-year operation, respectively, for inner side reflectors and the balance of the GCCs in the GCA are provided in Reference [18][18]. Based on preliminary temperature and irradiation dose inputs, Reference [18][18] shows that only portions of the inner reflector of the side reflector group [[ ]]P,E Appendix C of Reference [18] plots the changes in graphite strength (tension, compression, and flexural) as a function of irradiation dose and concludes that strength of irradiated graphite is higher than that of unirradiated graphite until at least crossover. The change in graphite stiffness (i.e., elastic modulus) due to irradiation dose is modelled directly in the material model IGNIS. In summary, the approach described above fully addresses the structural performance of the GCCs at these limiting fluence conditions.	--

[6]Turnaround is defined as the point at which volumetric dimensional change strain due to irradiation reverses.
[7]Nullity is defined as the point after turnaround at which the volumetric dimensional change strain due to irradiation is equal to zero. It is also referred to as crossover.

Provision from ASME BPVC III-5, HHA-3000	Comment from the NRC in RG 1.87, Revision 2 [4]	Comment from the NRC in NUREG-2245 [10]	Approach to Address the NRC's Exception	Section # in this Document
			In addition to the structural performance evaluation, it will be shown there are sufficient gaps remaining at end of life such that any GCC deformation will not result in constraint and thus hinder expansion or shrinkage, or that they will otherwise affect the function and stability of the GCA. This is consistent with the requirements of HHA-3212.	
HHA-3143, Abrasion and Erosion	The NRC staff is not endorsing the provisions of HHA-3143 (b) that set the mean gas flow velocity limit of 100 meters per second (330 feet per second) for evaluating the effects of erosion on the GCC design. Designers should determine the mean gas flow velocity limit above which an evaluation of erosion is necessary and justify that the limit is adequate for the GCC design.	The staff finds HHA-3143 to be acceptable because the staff agrees that it is appropriate and reasonable to evaluate abrasion and erosion of GCCs if the requisite conditions exist in a given design. However, the staff also notes that the mean gas flow velocity limit may not be generically applicable to all high-temperature, gas-cooled reactor designs. Therefore, the staff is not endorsing the provisions of HHA-3143 that set the mean gas flow velocity limit of 100 meters per second (330 feet per second) for evaluating the effects of erosion on the GCC design. Designers should determine the mean gas flow velocity limit above which an evaluation of erosion is necessary and justify that the limit is adequate for the GCC design.	Erosion will be addressed in a revision to this report as parallel efforts are underway to investigate this degradation mechanism.	6.6.4
HHA-3144, Fatigue	None	HHA-3144 is shown as "in the course of preparation." Therefore, the staff did not perform a review and is not endorsing HHA-3144, and the rest of the 2017 Edition of ASME Code III-5 remains valid without this provision.	In the absence of requirements from the ASME BPVC III-5, a methodology for the evaluation of fatigue life of GCCs was developed and is documented in Reference [19][19].	6.6.5
HHA-3330, Design of the Graphite Core Assembly	The NRC staff is not endorsing the provisions of HHA-3330(g) because provisions for inservice inspection are outside of the scope of ASME Code, Section III, Division 5.	The staff finds HHA-3330, paragraphs (a) through (f), and paragraph (i) to be acceptable because the provisions are sufficient to provide a minimum acceptable standard for the design of GCAs. However, the staff is not endorsing the provisions of HHA-3330(g) because requirements for inservice inspection are outside of the scope of ASME Code III-5, Subsection HH, Subpart A. The scope of the rules in Subsection HH, Subpart A, is clearly defined in HAB-1100 and HHA-1100, both of which do not include provisions for inservice inspection.	The NRC did not endorse the provisions of HHA-3330(g) only because in-service inspection was considered outside of the scope of its review. Similarly, in-service inspection is outside of the scope of this report.	--

]]P

6.4 Stress Analysis Methodology

Stress analyses are used to evaluate the GCCs during all loading conditions and taking into account the degradation mechanisms discussed in Section 6.6. All loads or effects on the GCC that cause stresses or deformations are evaluated. The stress analysis is typically completed by making use of a three-dimensional finite element model.

Per HHA-3215.3, the stress analysis of an irradiated GCC[8] shall take into account the effects of irradiation damage on the properties of graphite. To address this requirement, a material model named Irradiated Graphite Numerical Iterative Solver (IGNIS) is being developed to support the Xe-100 GCA/GCC design. See further discussion in Section 6.5.

These stress analyses are typically organized in the following analysis categories, which are discussed further in Section 2.3 of Reference [14]:

- Static structural analyses, which evaluate the GCCs during normal operating and cold shutdown conditions.
- Transient structural analyses, which evaluate the GCCs during short-term operational events that change the environmental conditions (temperature, pressure) to which the GCCs are exposed (such as AOOs). These analyses use the results from the static structural analyses as initial conditions.
- Dynamic structural analyses, which evaluate the GCCs during loadings such as in response to seismic events (Operating Basis Earthquake (OBE) and Safe-Shutdown Earthquake (SSE), also called the Design Basis Earthquake).

6.4.1 Uncertainty Analysis

HHA-3215 states that "Uncertainty of the calculated results shall be assessed as part of the analysis." To address this requirement and support demonstration of design margin, an uncertainty strategy has been developed for the stress analysis of the GCCs in the Xe-100 GCA, which is presented in Section 2.3.4 of Reference [14][14]. The uncertainty strategy seeks to quantify the uncertainty in the design-by-analysis route, and outlines a methodology for identifying, quantifying, and, where appropriate, minimizing uncertainties within the route.

An overview of the key sources of uncertainty within each modeling component of the design-by-analysis route is provided in Figure 19.

Three approaches may be used to assess the effect of uncertainty in the design-by-analysis route:

- **Approach I – ASME BPVC III-5 Most Probable Value Assessment**. This approach propagates the most probable values (i.e., best estimates) of uncertainty sources through the ASME BPVC III-5

[8] For stress analysis purposes, an irradiated GCC is defined as a component for which the damage dose is greater than 0.25 dpa at any point in the component per HHA-3142.1. See further discussion in Section 6.6.1.

assessments (i.e., simplified and full assessments). The output from this analysis is a single best estimate prediction of the quantities of interest, which are compared to the ASME BPVC assessments allowable limits.

- **Approach II – ASME BPVC III-5 Uncertainty Assessment**. This approach quantifies uncertainty in ASME BPVC assessments. The main output from this analysis is a distribution of the prediction of the quantities of interest, which are compared to the ASME BPVC assessments allowable limits and used to calculate a probability of exceeding the allowable limit.
- **Approach III – Reliability Analysis**. This approach calculates the POF through a reliability analysis approach. The main output from this analysis is a single POF value that incorporates all the uncertainties in a probabilistic manner.

A high-level schematic overview of the three uncertainty assessment approaches is provided in Figure 20.

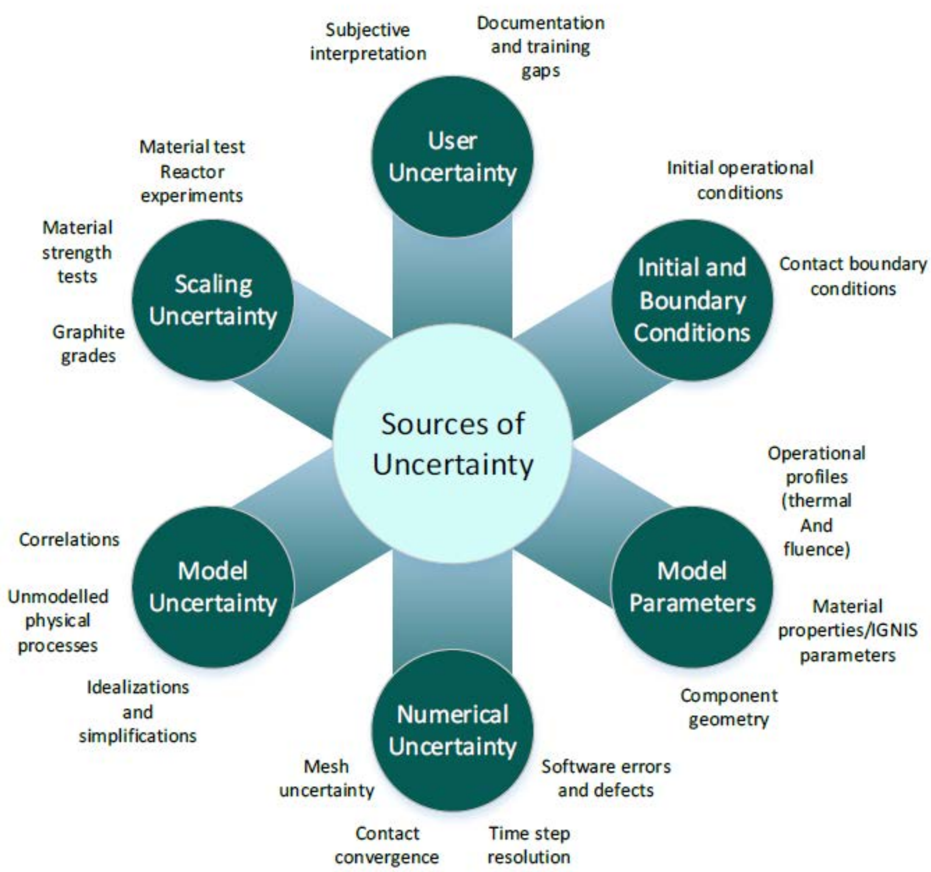

Figure 19: ***Potential Sources of Uncertainty in the Design-by-Analysis Route***

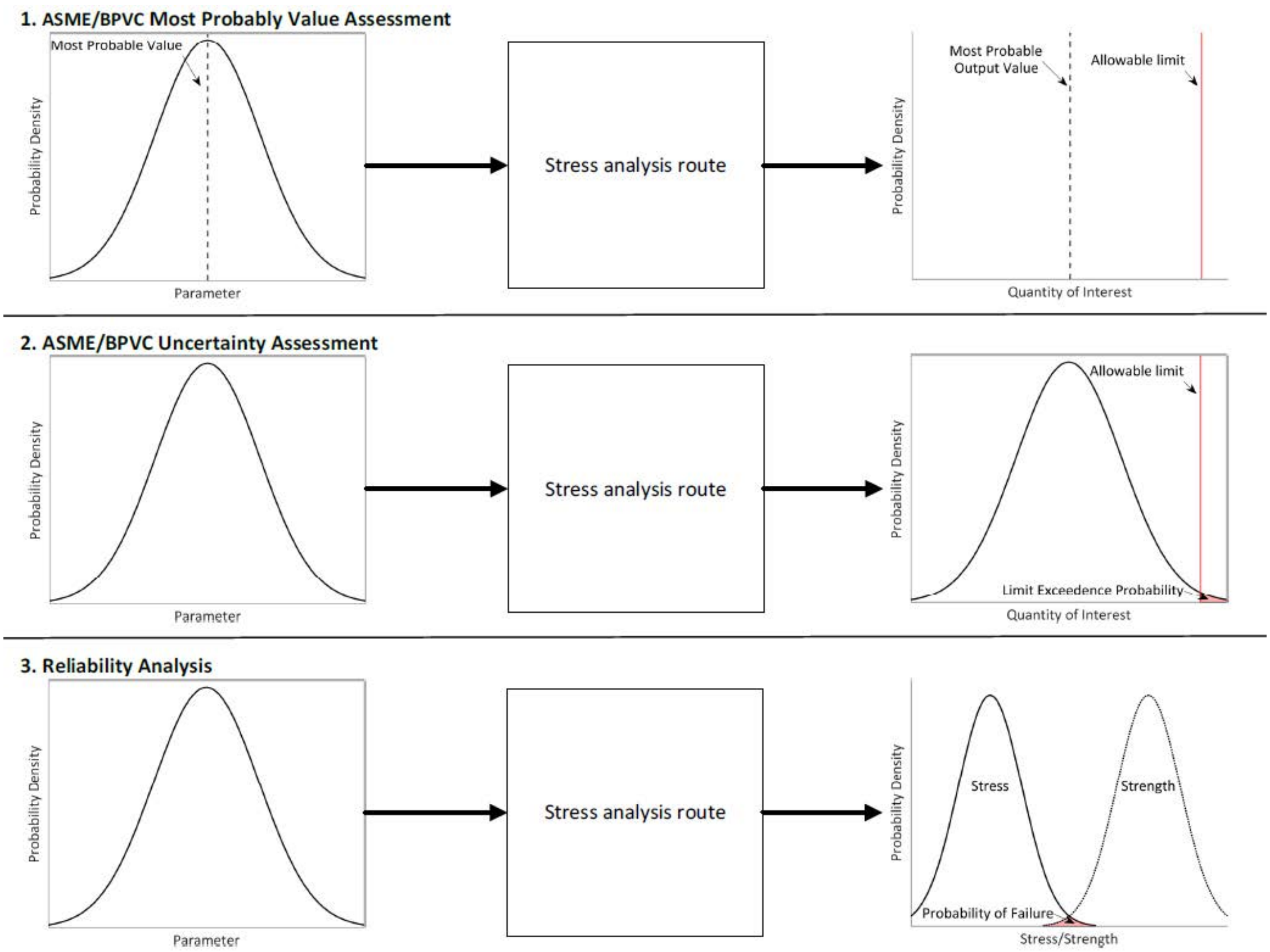

Figure 20: ***A High-Level Schematic Overview of the Three Uncertainty Assessment Approaches***

6.5 Graphite Custom Material Model – IGNIS

As discussed in Section 6.4, the stress analysis of an irradiated GCC shall take into account the effects of irradiation damage on the properties of graphite. To address this requirement, a material model (IGNIS) is being developed to support the Xe100 GCA/GCC design. As noted in Section 4, due to limited grade-specific data available to complete the Material Data Sheet, especially for irradiated data, the data set to be used for design analysis is enhanced by addition of data from other grades, leveraging common underlying behavior of graphite and tuning a constitutive material model to the grades selected for construction. The augmented data set and implementation approach in the material model (IGNIS) to be used design analysis is presented herein. This material model is presented in detail in Reference [21].

Material properties of graphite are complex functions of temperature and neutron irradiation dose which vary spatially in the reactor and throughout its operating lifespan. Irradiation dose from fast neutron fluence over exposure time causes dislocation in or damage to the crystal lattice and general structure of the graphite. This results in irradiation induced dimensional change and affects the material properties of

GCCs. Graphite also displays other unique material behaviors, such as irradiation induced creep. Furthermore, any excessive deformation of GCCs can lead to distortions of control rod channels and affect coolant flow paths throughout the core. Therefore, accurate modeling of GCC materials is essential in the structural design process.

Due to these complexities, the custom graphite material model IGNIS is being developed. IGNIS works in conjunction with the Ansys FEA program to model the properties and behavior of graphite material under neutron irradiation effects and other loading conditions expected within the Xe-100 reactor. Reference [21][21] describes the background, material model theory, implementation, and qualification plan for development of IGNIS. Specifically, the following sections within Reference [21] summarize the key aspects of the IGNIS graphite material model:

- Theory and Implementation (Section 2),
- Verification and Validation (Section 3),
- Qualification and Configuration Management (Section 4), and
- Future Development (Section 5).

Section 2 of Reference [21][21] describes the physical phenomena selected from NUREG-6944 Volume 5 [22] that are most important to the structural modeling of irradiated graphite. This NUREG includes the graphite Phenomena Identification and Ranking Tables (PIRTs) for next generation HTGR, such as the Xe-100 reactor. Most of those irradiated graphite physical phenomena have been captured in the current version of IGNIS and a few are to be considered in the next phase of IGNIS development. Section 2 also summarizes the theory behind modeling irradiated graphite, governing equations, and the mathematical material model used (i.e., a modified Maxwell-Kelvin viscoelastic model shown in Figure 21) and how it is implemented.

IGNIS interacts with the Ansys FEA program to solve the governing equations based on applied force (pressure, temperature, deadweight, and fast neutron fluence), displacement, and stiffness matrix iteratively for a typical nonlinear static structural model. The IGNIS-Ansys interaction structure is shown in Figure 22.

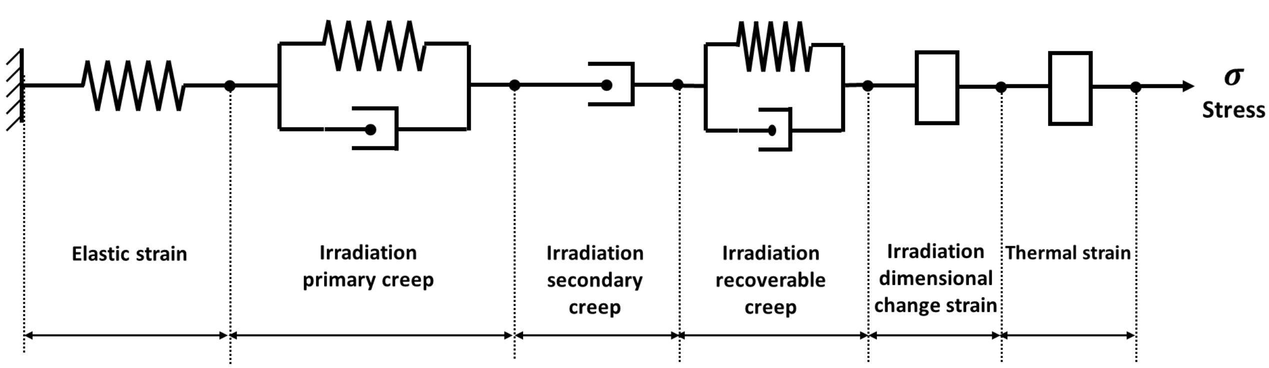

Figure 21: ***Modified Maxwell-Kelvin Material Model***

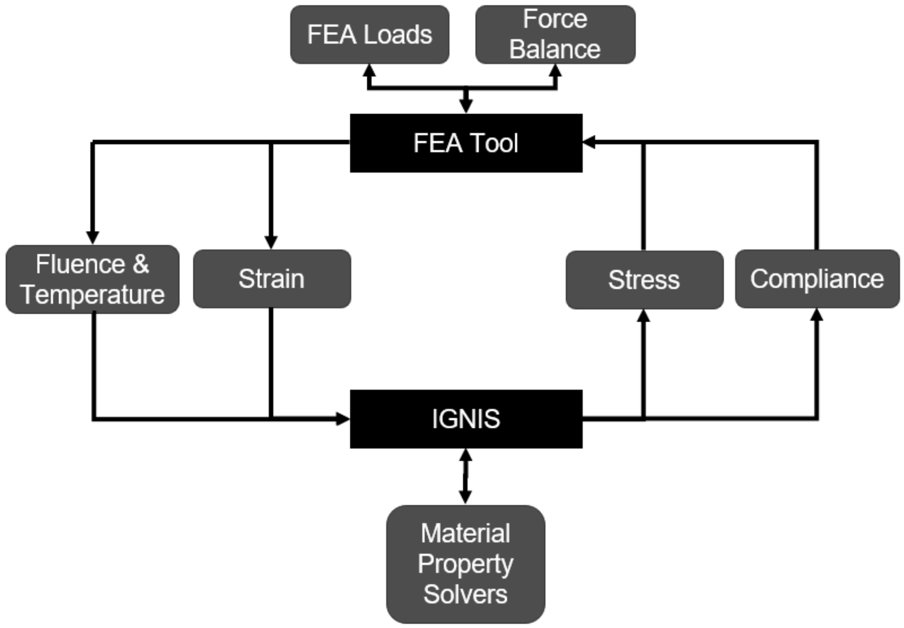

Figure 22: ***IGNIS-Ansys Interaction Structure***

Section 3 of Reference [21] describes the V&V activities that are performed to ensure proper functioning of IGNIS. The verification activities ensure that IGNIS solves the governing equations correctly. In the validation activities the results of IGNIS are compared to experimental data.

Section 4 of Reference [21] describes the plan that guided the qualification efforts for bringing IGNIS into compliance with ASME NQA-1 [23] and other standards as needed.

In summary, the Xe-100 GCA design analysis will implement a graphite material model that captures the elastic, thermal expansion, irradiation dimensional change, and creep strain behavior, including appropriate V&V activities, as a necessary enabling step to perform realistic stress analysis that models the important phenomena. IGNIS v2.0 is currently the latest version to date.

6.6 Graphite Degradation Mechanisms

The methodology selected for the design of the Xe-100 GCA and GCCs addresses the effects of the following degradation mechanisms, which are discussed further in the following subsections:

- Irradiation – Section 6.6.1
- Oxidation – Section 6.6.2
- Abrasion – Section 6.6.3
- Erosion – Section 6.6.4
- Fatigue – Section 6.6.5

6.6.1 Irradiation Effects

Internal Stresses Due to Irradiation

Neutron irradiation affects the crystalline structure of graphite and results in graphite material property changes, which induce internal stresses in irradiated GCCs.

HHA-3142.1 specifies the threshold limits to be used in determining how the effects of cumulative fast neutron irradiation fluence should be considered for a given GCC. HHA-3142.3 requires that the internal stresses in irradiated GCCs (i.e., that exceed the dose limits described in HHA-3142.1(c)) shall be calculated.

Most GCCs in the Xe-100 are expected to experience damage doses greater than 0.25 dpa. Therefore, these GCCs are considered to be irradiated per HHA-3142.1 (c). For these irradiated GCCs, the ASME BPVC requires the use of a viscoelastic model to characterize the graphite material behavior under neutron irradiation. To address this requirement, a custom material model named IGNIS**,** which predicts the behavior of graphite under the conditions that exist within the Xe-100, is being developed. IGNIS is discussed further in Section 6.5.

Stored Wigner Energy

Wigner energy (WE) is a type of stored energy that accumulates in graphite material when irradiated at low temperatures. The buildup of Wigner energy can be relieved by heating the material, a process known as annealing. A sudden release of Wigner energy can be a serious concern for the safe operation of nuclear reactors as it can cause a rapid and significant increase in temperature.

HHA-3142.2 requires that, for graphite irradiated to damage dose greater than 0.25 dpa at a temperature <200°C, the effect of stored (Wigner) energy buildup shall be accounted for when evaluating reactor thermal transients.

[[

]][P,E] However, as discussed in Section 3.1.2 of Reference [14], Wigner energy accumulates even when graphite is irradiated at higher temperatures. The release of Wigner energy in graphite irradiated to high dose in HTGR conditions remains an area of uncertainty. X-energy's irradiation program at ORNL will test the release of Wigner energy in graphite irradiated to high dose at around 200°C (see Section 4.5.7) and the significance of Wigner energy for safety in the Xe-100 will be evaluated.

6.6.2 Oxidation

Oxidation as a degradation mechanism for GCCs is introduced as follows in HHA-3141:

> *GCCs may be oxidized by moisture, oxygen, or carbon dioxide in the coolant. The oxidizing gas mixtures diffuse into the porous structure of the graphite. The local weight loss in the GCC varies depending on the conditions at which the oxidation occurs and the distance from the surface*

exposed to the gas flow. Subsurface weight loss (or density) profiles provide a means to estimate oxidation gradients versus distance from the exposed surface. The local weight loss decays exponentially from the surface down to the maximum penetration depth, which depends on temperature. At equal overall weight loss, oxidation at lower temperatures penetrates deeper under the surface than at higher temperatures.

In addition to material property changes, weight loss due to oxidation might result in loss of strength and reduction in geometry, thus it potentially impacts the structural integrity of the GCCs.

HHA-3141 requires that oxidation analysis shall be carried out in detail to estimate the weight loss profiles of graphite structures, since reaction rates depend on the temperature, reactants, and graphite grade. Additionally, as documented in RG 1.87 [4] and in Section 6.3 of this document, the NRC staff did not endorse the provisions of HHA-3141 (c) that sets a weight loss limit of 30%. RG 1.87 suggests that designers should determine the amount of weight loss above which the region should be regarded as completely removed from the structure and justify that the limit is adequate for the design-specific oxidation analysis.

As part of detailed design, as mentioned in Section 4.6, detailed analysis will be conducted to determine the extent of graphite mass loss due to oxidation during normal operation (chronic) and after postulated accident scenarios (acute). The assessment will consider the following items to characterize the extent of oxidation in the GCCs:

1. Primary helium environmental conditions,
2. Impurities over the life of the reactor,
3. Temperature of the GCCs,
4. Irradiation of GCCs, and
5. Reaction rates derived from industry research and testing.

The objective of the analysis is to both ensure the applicable requirements from the ASME BPVC III-5 are met, and that the RSFs of the GCA are not impacted by both chronic oxidation and acute oxidation. The oxidation analysis plan for the Xe-100 GCA is presented in Section 3.2 of Reference [14].

Fundamentally, as the Xe-100 is an HTGR which uses highly inert helium as the coolant, it is expected that chronic oxidation of the graphite in normal operation due to the low concentrations of gaseous impurities will be very low (i.e., lower than 1%, which is the threshold over which the graphite material is considered oxidized per HHA 3141 (a)), as will be demonstrated through analysis. The introduction of impurities during operation will be controlled to ensure a low inventory is present, and a Helium Purification System (HPS) is also available to assist with extracting impurities further. Oxidation rates are exponentially higher at elevated temperatures. Regions of the pebbles in the pebble bed are at elevated temperatures and have a high surface area and graphite matrix volume available for preferential interaction with the gaseous impurities. Most of the GCCs operate in low temperatures at which oxidation rates are close to negligible. As for acute oxidation, it is expected to occur during an event such as a steam generator tube rupture (SGTR). Due to the large oxidant ingress during such postulated event, calculating a mass loss due to oxidation in all GCCs less than 1% is not expected. As such, the graphite material for these GCCs is likely to be considered oxidized. This intent is to demonstrate that the acute oxidation of GCCs do not impede the GCA from meeting its RSFs.

6.6.3 Abrasion

HHA-3143 (a) requires that abrasion shall be evaluated if there is relative motion between GCCs, GCCs and interfacing components, or GCCs and the fuel of a pebble bed reactor. A methodology to address the abrasion concern has been developed for the Xe-100 GCA which is presented in Section 3.4 of Reference [14].

Based on the preliminary results discussed in Section 3.4 of Reference [14], the impact of abrasion on the structural integrity of these GCCs as well as on the GCA's capability to perform the RSFs is likely negligible.

6.6.4 Erosion

HHA-3143 (b) requires erosion shall be evaluated in areas where the mean gas flow velocity in the cross section of the channel exceeds 330 ft/sec (100 m/s). However, as documented in RG 1.87 [4] and in Section 6.3 of this document, the NRC staff did not endorse the provisions of HHA-3143 (b) that set the mean gas flow velocity limit of 100 meters per second (330 feet per second) for evaluating the effects of erosion in the design of GCCs. RG 1.87 suggests that designers determine the mean gas flow velocity limit above which an evaluation of erosion is necessary and justify that the limit is adequate for the GCC design.

Erosion will be addressed in a subsequent revision to this report as parallel efforts are underway to investigate this degradation mechanism. Based on conversations with industry experts, it is expected it will be shown that erosion is not a significant degradation mechanism to the performance basis in the flow regimes to which the Xe-100 GCCs are subjected.

6.6.5 Fatigue

GCCs are subjected to cyclic loading caused by varying temperatures and thermal gradients during reactor start-up, load-following operations, reactor trips, cold pressurized shutdowns, cold de-pressurized shutdowns, and many more LBEs during the lifecycle of the Xe-100. Cyclic loading would also be experienced during off-normal events, in which cases it is desirable to confirm fatigue does not challenge component integrity. Fatigue under these conditions may cause propagation of nucleated micro-cracks and form inherent flaws in graphite. Thus, there is a need for a method to assess fatigue mechanisms as part of qualification of GCCs.

HHA-3144 is shown as "in the course of preparation." Therefore, the NRC staff did not review HHA-3144.

In the absence of requirements from the ASME BPVC III-5, a methodology for the evaluation of fatigue life of GCCs was developed in Reference [19] and is presented in Section 3.5 of Reference [14]. The methodology leverages the recommendations from a NUMARK technical letter to the NRC [20].

An overview of the fatigue methodology is shown in Figure 23 of Reference [19] and is summarized as follows:

- The methodology uses publicly available fatigue data for [[ ]]^{P,E} from multiple stress ratios R (Step a) in Figure 23.
- A statistical model from Price, discussed in Reference [19], is used to characterize the relationship between number of cycles to failure and homologous stress (defined as the ratio of the maximum applied stress over the mean strength). Specifically, low-cycle fatigue data (less than 2,500 cycles) is used to calibrate the Price model. Fatigue data beyond 2,500 cycles then are used to validate the Price model. (Steps b), c), and d) in Figure 23). Lower tolerance limits are defined for several levels of POF targets: 10^{-4}, 10^{2}, and 5×10^{-2}. This effort is repeated for each stress ratio R.
- For each given POF target, a Modified Goodman Diagram (Step e) in Figure 23 is developed. The X and Y axes represent the minimum and maximum stresses experienced during the cyclic stress event, respectively. These stresses are ratioed by the mean strength.[9] Several thresholds of number of cycles to failure are drawn on the Modified Goodman Diagram. For example, the green highlighted region in Figure 23 is defined as the "safe zone" for which any combination of minimum and maximum stresses within that region can withstand 10^5 cycles or more before failure.
- Finally, the damage accumulation from different cyclic stress event combinations is calculated using the Miner's rule (Step f) in Figure 23.

Fatigue assessment is expected to be performed for applicable loading scenarios where cyclic loading behavior might exist, such as in mode state changes, accident events, or in response to an earthquake.

[9] The mean strength can be either the mean tensile strength or the mean compressive strength depending on the loading condition (i.e., tensile for pull-pull and pull-push, and compressive for push-push).

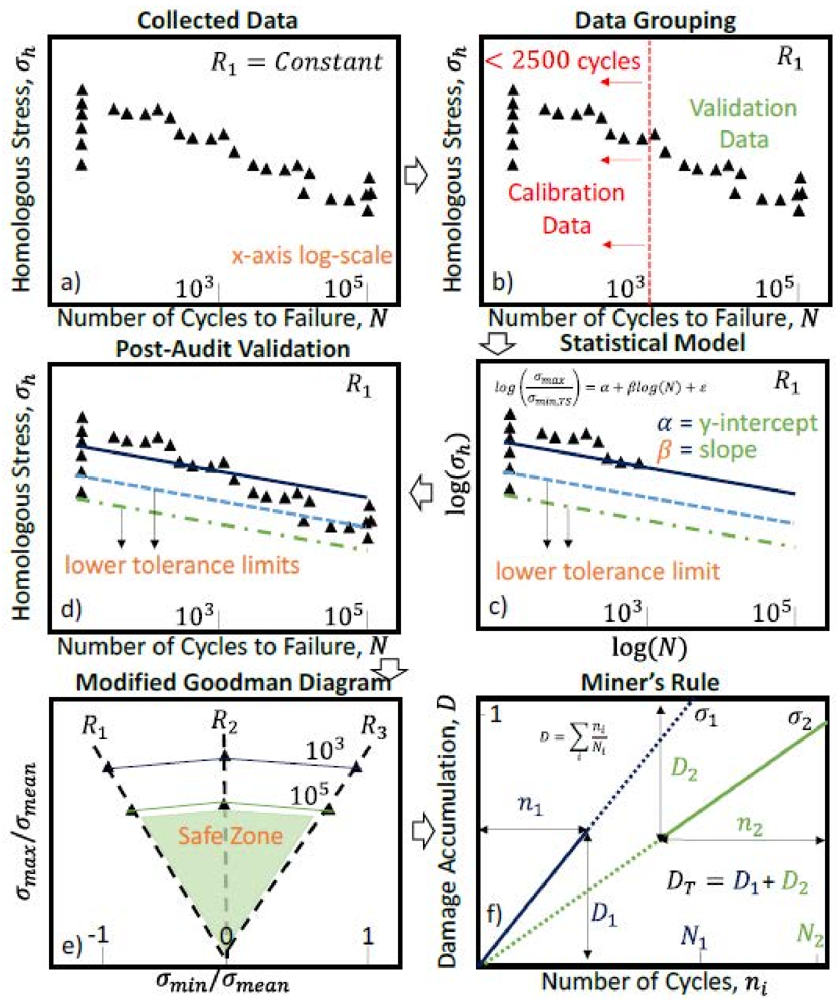

Figure 23: ***Overview of the Fatigue Methodology***

6.7 Acceptance Criteria

The definition of acceptance criteria for the GCA and its GCCs starts from the following fundamental concepts:

- The GCA as a whole supports several safety functions for reactivity control, fuel cooling, and fuel movement. The GCA determines the geometry of the pebble bed and control rod channels ("Maintain Geometry" and "Control Reactivity") and is relied upon to conduct heat away from the pebble bed in certain accident conditions, maintaining acceptable fuel temperatures ("Control Heat Removal").
- The GCA consists of a large number of loosely stacked GCCs (i.e., components, keys, and sleeves) with gaps between them, rather than a singular monolithic structure (for which degradation of the structural integrity of any part is more likely to impact the entire structure).

- Graphite is a brittle material which does not experience substantial plasticity.
- At the component level (i.e., GCCs), some level of deformation, displacement, and damage (including cracking) is expected and must be anticipated in all conditions, including normal operation.
- At the assembly level (i.e., GCA), this degradation may include movement of GCCs relative to each other and the other reactor components at places where there are already gaps. It may also include significant, permanent relative motion of GCCs and/or fragments of fully cracked GCCs. The results may include changes to coolant flow, changes to fuel movement and distortion of control rod channel shapes with different end states depending on the event severity.
- Reliability of individual GCCs (i.e., absence of cracks) does not inherently support safety functions of the GCA. Degradation of the structural integrity (by crack initiation and crack propagation) of individual GCCs might or might not affect the RSFs of the GCA, depending on the design, the percentage of individual GCCs affected, and the severity and location(s) of degradation.

In combination, these concepts mean that the conventional treatment for metallic components, which are generally designed to deterministically preclude any type of structural integrity degradation, is not applicable to GCCs and GCA.

The structural design methodology defined in the ASME BPVC III-5 is based on probabilistic methods and relies on structural reliability targets (i.e., “Probability of Failure” targets) as an effort to minimize the number of GCCs with cracks, and thus as an attempt to eventually minimize the probability that the GCA will be unable to perform its RSFs.

However, as discussed above, evaluation of the structural integrity degradation (by crack initiation) of individual GCCs may not be sufficient to ensure that the GCA successfully supports all applicable RSFs (specifically RSF 1.1.2.1c, which is directly related to “control reactivity” and RSF 1.2.1.1c, which is directly related to “control heat removal”). Furthermore, evaluation of GCCs’ deformation, displacement, and material property changes (e.g., change in thermal conductivity) also play a role in this determination. Therefore, additional derived acceptance criteria will be developed at the component level (i.e., GCC level) to demonstrate that the GCA successfully supports the applicable RSFs during all LBEs. The methodology that will be used to define these additional acceptance criteria is expected to fit within the framework of the methodology presented in Reference [14], which leverages the risk-informed, performance-based process outlined in NEI 18-04 [1].

In summary, the design methodology presented herein intends to leverage both (1) the structural design methodology acceptance criteria defined in HHA-3000 of the ASME BPVC III-5 and (2) the additional acceptance criteria to be defined subsequently leveraging the risk-informed, performance-based process outlined in NEI 18-04 [1].

6.7.1 Damage Tolerance Assessment

While the design objective is to minimize the probability for cracking, the ASME BPVC III-5 acknowledges that this may not be achievable in practice. Specifically, HHA-3100 states that:

> *The design approach selected is semiprobabilistic, based on the variability in the strength data of the graphite grade. Due to the nature of the material, it is not possible to ensure absolute reliability, expressed as an absence of cracks, of Graphite Core Components. This is reflected in the setting of Probability of Failure (POF) targets. Also note that due to the complex nature of the loadings of graphite components in a reactor combined with the possibility of disparate failures of material due to undetectable manufacturing defects, the Probability of Failure values used as design targets may not be precisely accurate predictions of the rate of cracking of components in service. The Designer is required to evaluate the effects of cracking of individual Graphite Core Components in the course of the design of the Graphite Core Assembly and ensure that the assembly is damage tolerant.*

In addition to the design by analysis approach from the ASME BPVC III-5, a Damage Tolerance Assessment (DTA) will be leveraged to demonstrate that the design is damage tolerant provides additional assurance that the design is safe and ensures its reliable operation. A DTA may provide multiple, partially independent, arguments in support of the safety of the GCA. A DTA ensures a robust design that remains functional even if there is unexpected behavior of GCCs or unexpected loading of the GCA.

7. NEI 18-04 Implementation and Special Treatments Identification

This section presents the methodology to:

- Develop appropriate reliability and capability performance-based targets for the GCA,
- Develop appropriate special treatment requirements, as would be included in the Design Specification, and
- Demonstrate that the performance-based targets intended in the design are met as would be included in a Design Report.

This section is based largely on information contained in the "NEI 18-04 Implementation Methodology" document [13][13].

7.1 GCA Safety Basis

The approach used to ensure the GCA performs its RSFs is risk-informed and performance-based by use of the NEI 18-04 [1] approach, rather than purely compliance-based. By following NEI 18-04 principles, the GCA design will develop and use a graded approach to qualification, use of design data, and analytical approach/assumptions commensurate with contribution to safety.

The process used to classify the various GCCs that make up the GCA is described in the following paragraphs based on Reference [13]. To help explain the process, a flowchart is presented in Figure 24 that identifies each block with an alphanumeric code. Also, although the flowchart depicts specific functional areas (i.e., Risk-Informed Analysis, Design Analysis, and Graphite Analysis) for executing the assessment process, there will necessarily be overlap among these functional areas.

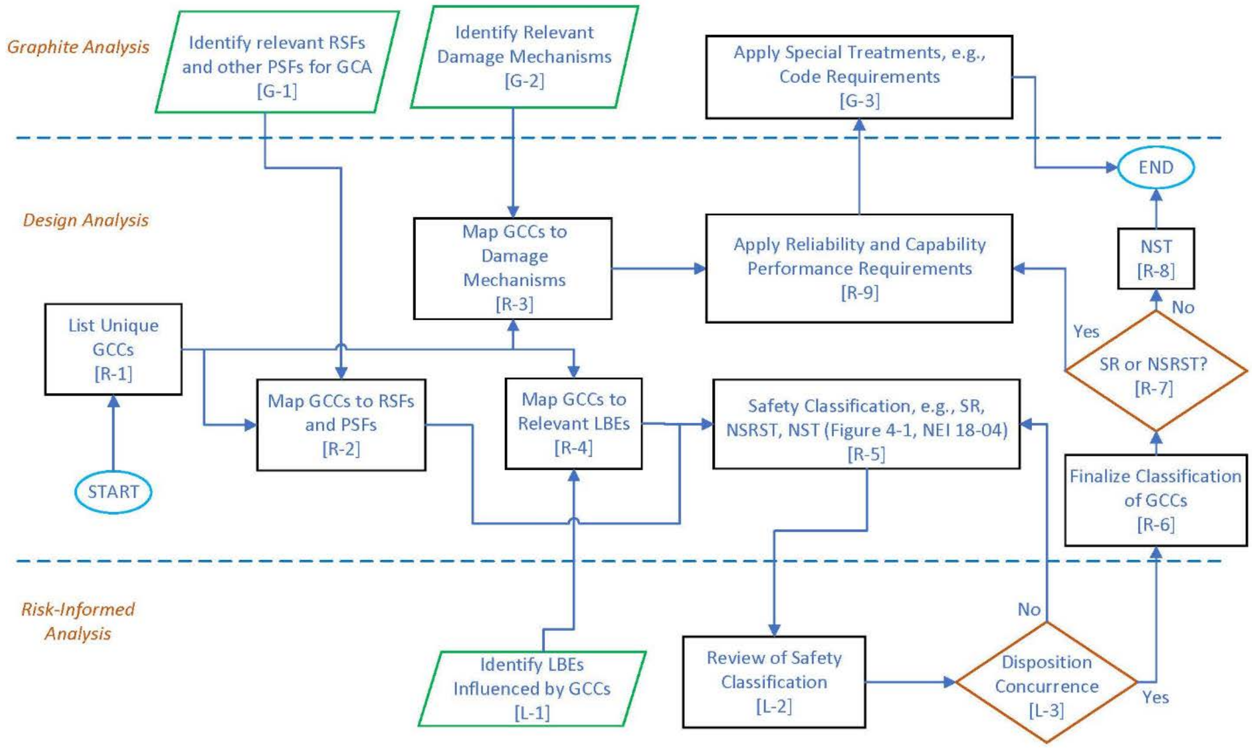

Figure *24*: *GCC Grading Process Flowchart*

The first step in the GCC grading process is to list all GCCs that make up the GCA (*R-1*). The next step (*R-2*) is to map the RSFs and PSFs identified by the Graphite Analysis functional area to each of the GCCs. The Graphite Analysis functional area has identified applicable safety functions [46]based on input from the preliminary safety classification [8], which can be used as an input to map individual GCCs to their respective RSFs and/or PSFs, depending on the nature of the physical process for which they are responsible, e.g., heat transfer, core support, etc. Input from the Graphite Analysis functional area will help the Design Analysis functional area map what GCCs are susceptible to identified damage mechanisms (*R-3*). For example, GCCs closer to the periphery of the reactor core are more susceptible to neutron fluence damage due to the presence of a higher neutron flux in that vicinity, which has an impact on the application of Special Treatments (*G-3*). Input from the Risk-Informed Analysis functional area will identify LBEs in which the GCCs may play an influential role (*R-4*).

Step *R-5* is where the safety classification of GCCs is determined based on upstream information regarding GCC assignment to safety functions (RSFs or PSFs) and associated LBEs, which can be any combination of AOOs, DBEs, or Beyond Design Basis Events (BDBEs). This step is also representative of the safety classification process outlined in NEI 18-04 [1]. The assignment of safety classifications then undergoes an iterative review process (*L-2*) and (*L-3*) to ensure the GCCs are appropriately classified. After review and concurrence of the assigned safety classifications, i.e., either Safety Related (SR), Non-Safety Related with Special Treatments (NSRST), or Non-Safety Related with No Special Treatments (NST), the classification of GCCs is finalized in step *R-6*. Step *R-7* is a decision point in which GCCs classified as either SR or NSRST are further scrutinized by establishing reliability and capability performance targets (step *R-9*) based on information from step *R-3*. Step *R-8* simply assigns the NST category to those GCCs that were not classified as either SR or NSRST.

Step *G-1* is meant to identify what RSFs or PSFs are relevant for the GCA. For example, the GCA is responsible for maintaining structural integrity and the ability to withstand certain physical loads and stresses. However, the GCA can also play an important role regarding progression of LBEs in which it can have an impact on a safety function that is not as important to structural integrity, e.g., heat transfer. It is these identified RSFs and PSFs that are passed to the Design Analysis functional area to help with mapping individual GCCs to their respective safety functions (RSFs and/or PSFs). Step *G-2* is meant to identify what damage mechanisms exist that can physically degrade the GCA due to harsh environmental conditions, e.g., high neutron fluence, the effects of temperature, erosion, etc. Relevant damage mechanisms may be found in Reference [47][47].

Once reliability and capability targets have been established by the Design Analysis functional area, the Graphite Analysis functional area is tasked with assignment of Special Treatments (*G-3*) to ensure the GCCs can meet their reliability and capability targets. Examples of Special Treatment could include the consideration of manufacturing quality specifications, design to a certain ASME Structural Reliability Class per Reference [3] (to achieve a suitably low number of cracks), or other material requirements related to heat transfer properties for transferring heat away from the reactor core.

The contribution from the risk-informed analysis functional area is to identify relevant LBEs (*L-1*) in which the GCA plays an influential role. For example, certain GCCs that make up the GCA are responsible for supporting the fuel pebbles that make up the reactor core and are thus important for those LBEs where core geometry plays a role. As another example, loss of forced coolant flow retards the transfer of heat away from the reactor core, so those GCCs in the vicinity of the pebble bed are significant contributors to conductive heat transfer to maintain fuel temperatures within acceptable limits. Once safety classifications have been assigned by the Design Analysis functional area, the Risk-Informed Analysis functional area is responsible for their concurrence and to ensure the proper classifications have been made within the context of NEI 18-04 [1].

8. Conclusions

8.1 Summary

This LTR described the following methodologies:

- X-energy's methodology for performing design analysis to ensure that the RSFs associated with the GCA are maintained (Section 6). This includes both those functions related to assuring the mechanical/structural integrity of the GCA, the focus of application of ASME BPVC III-5, and other functions (e.g., control reactivity, control heat removal) not governed by ASME BPVC III-5. The methodology conveyed in this LTR informs the development of content necessary for a Design Specification and method of demonstrating the acceptable mechanical/structural integrity and performance of the design in a Design Report. This LTR presented the details of augmentations, exceptions, and limitations taken for this methodology as compared to the methodology taken in the 2017 Edition of the ASME BPVC III-5 [9] as endorsed by the NRC with limitations via RG 1.87 [4].
- X-energy's material qualification methodology for obtaining material property data for as-manufactured, oxidized, and irradiated graphite to arrive at the set of data needed for the Material Data Sheet (MDS) to be used for design (Section 4). This includes, as applicable, the current set of data available for preliminary design analysis, current and future testing discussed in Section 5 that augments the data set currently available for design and for material specifications to be used in component procurement, as well as known or anticipated gaps or exceptions in the data expected or explicitly required by the ASME BPVC III-5.
- X-energy's methodology to develop appropriate reliability and capability performance-based targets by application of the NEI 18-04 process [1], as well as the methodology to develop appropriate special treatment requirements as would be included in the Design Specification, and to demonstrate that the performance-based targets intended in the design are met as would be included in a Design Report (Section 7).

X-energy intends to revise this report in the future with further testing results, responses to NRC comments and questions, and in updates to revised industry guidance.

8.2 Scope Exclusions

The following activities are integrated into the design of the GCA but are not included in the scope of this LTR. They are listed and discussed below to provide context within the overall framework of the design, development, and licensing of the GCA.

- ***Reliability and Integrity Management***

X-energy intends to implement the methodologies presented in this LTR to evaluate and qualify the GCA for the life of the Xe-100 reactor design alongside a Reliability and Integrity Management (RIM) program. Development of the RIM program will implement operational activities that support demonstration that the RSFs, reliability and capability performance-based targets, and special treatment requirements are met throughout the 60-year design life. As one of the first steps in the development of a RIM program for the GCA, degradation mechanisms have been evaluated and presented herein (Section 6.6). The details of the RIM program for the GCA will be provided as part of an Operating License (OL) application.

- ***Testing***

The testing programs applicable to the GCA are presented for information in Section 5 to provide sufficient background on how X-energy intends to leverage them in the context of the methodologies presented in this LTR. Future testing programs will be presented in detail and explicitly mentioned in areas which they support the design or safety assurance of the GCA when those become available. Testing programs will likely be used to provide additional data to support the final GCA design.

- ***Quality Assurance (QA)***

X-energy intends to develop a QA Manual (QAM) compliant with all requirements listed in Subsection HAB of the ASME BPVC III-5, 2023 Edition [3]. This QAM is outside of the scope of this LTR, but it can be inferred that the methodologies presented herein will be aligned with the QAM.

- ***Construction***

Consistent with the requirements from the ASME BPVC III-5, a Construction Specification for the GCA will be prepared. The activities and requirements associated with this Construction Specification are outside of the scope of this LTR. These typically include:

- Machining, examination, testing, and packaging of GCCs;
- Storage of GCCs at site, unpackaging and examination of GCCs, installation of GCCs to form the GCA, and examination requirements during installation and post-installation of the GCA.

8.3 Evolving Design and Licensing Basis Documentation

The following documents provide the comprehensive set of documentation that forms the design and associated licensing basis for the GCA. This documentation will be generated and updated at various phases during the design process as continuous development occurs until the final GCA design is complete. These documents are discussed below to provide sufficient context to this LTR:

- **Material Data Sheet (MDS)** – The MDS contains the data required for design in accordance with ASME BPVC III-5, Subpart HHA. This data encompasses the material properties in the as-manufactured, irradiated, and oxidized conditions, including creep and dimensional change behavior, and will be used to conduct the analyses that will be documented in the Design Report for the GCA. A subset of this information will also be used to inform the material specifications for a Construction Specification to be used for procurement of the graphite material. The MDS will be prepared for the intended graphite grades in the Xe-100 reactor. Each MDS will be developed in line with the methodologies established herein and will contain the design data which will be used to evaluate the GCA and its GCCs.

- **Design Specification** – The GCA Design Specification is the document that will enforce and control the design process for the GCA and its GCCs. The Design Specification for the GCA will leverage the methodologies presented herein and provide a comprehensive set of design requirements for the GCA and its GCCs including ASME BPVC III-5 requirements (with augmentations, exceptions, and limitations discussed herein), reliability and capability performance-based targets and special treatment requirements, and any additional requirements necessary for maintaining all RSFs applicable to the GCA and its GCCs. The Design Specification for the GCA will be incorporated by reference in a subsequent FSAR transmitted in support of an Operating License Application (OLA).

- **Design Report** – A Design Report for the GCA will be produced based on the methodologies set forth in this LTR. It will be written to summarize the analysis documented and to show compliance with the GCA Design Specification. This report will demonstrate that the GCA performs its RSFs for their design life and that reliability and capability performance-based targets and special treatment requirements in the Design Specification are met to support initial operation. The Design Report will also provide recommendations for operational activities that support demonstration that the RSFs meet performance objectives for the lifetime of the plant as would be demonstrated by an associated RIM program for the GCA. The final Design Report for the GCA will be incorporated by reference in a subsequent FSAR transmitted in support of an OLA.
- **Preliminary Safety Analysis Report (PSAR)** – This LTR will form the basis of information for the GCA-related PSAR content. This document specifies preliminary methodologies which will be leveraged by a PSAR to demonstrate adequate consideration has been given to the GCA.
- **Final Safety Analysis Report (FSAR)** – This LTR will be used as the basis for the final methodologies to be used in the design and analysis of the GCA and its GCCs, which will be finalized in an FSAR. The LTR in tandem with the MDS, Design Specification, and Design Report will form a comprehensive set of design data, methodologies, requirements, and analyses to demonstrate that the GCA performs its RSFs for a 60-year design life.

8.4 Conclusion

X-energy is requesting NRC review and approval of the graphite material qualification and design methodologies as implemented via the NEI 18-04 risk-informed performance-based methodology to support the preliminary design, and subsequent final design, and associated analyses of the GCA.

9. Cross References and References

Document Title Cross References: X-energy documents that may impact the content of this document. References: X-energy or other documents that will not impact the content of this document	**Document No.**	**Rev./ Date of Issuance**	**Cross Reference/ Reference**
[1] Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development	NEI 18-04 (ML19241A472)	2019, Revision 1	Reference
[2] Xe-100 Principal Design Criteria Licensing Topical Report	004799	3	Cross Reference
[3] ASME Boiler and Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors."	N/A	2023	Reference
[4] US NRC Regulatory Guide, "Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors'"	RG 1.87 (ML22101A263)	2	Reference
[5] US NRC Regulatory Guide, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors"	RG 1.232 (ML17325A611)	0	Reference
[6] Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: Content for Applicants Using the NEI 18-04 Methodology	NEI 21-07		Reference
[7] US NRC Regulatory Guide, "Guidance for a Technology-Inclusive, Risk-Informed, Performance-Based Methodology to Inform the Licensing Bases and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors"	RG 1.233 (ML20091L698)	0	Reference
[8] Xe-100 Preliminary Safety Classification of SSCs	001067	4	Cross Reference
[9] ASME Boiler and Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors."	N/A	2017	Reference
[10] US NRC NUREG "Technical Review of the 2017 Edition of ASME Code, Section III, Division 5, "High Temperature Reactors""	NUREG-2245 (ML23030B636)	January 2023	Reference
[11] ASME Boiler and Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors."	N/A	2019	Reference
[12] ASME Boiler and Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors."	N/A	2021	Reference
[13] Xe-100 NEI 18-04 Implementation Methodology for the Graphite Core Assembly (GCA)	009322	1	Cross Reference
[14] Xe-100 Design Methodology for Graphite Core Assembly and Graphite Core Components	009650	2	Cross Reference
[15] ASME BPVC Gap Assessment for the Xe-100 Graphite Core Assembly (2017 Edition to 2023 Edition)	009482	1	Cross Reference
[16] ASME NTB-4, "Background Information for Addressing Adequacy of Optimization of ASME BPVC Section III, Division 5 Rules for Nonmetallic Core Components"	N/A	2021	Reference

Document Title Cross References: X-energy documents that may impact the content of this document. References: X-energy or other documents that will not impact the content of this document	**Document No.**	**Rev./ Date of Issuance**	**Cross Reference/ Reference**
[17] Saitta, M, & Beirnaert, G. "Simplified Method for Adjusting the Shape and Characteristic Strength Parameters of the Weibull Strength Distribution of Graphite Materials" Pressure Vessels and Piping Conference, American Society of Mechanical Engineers, 2023.	PVP2023-105207	2023	Reference
[18] Xe-100 Graphite Core Assembly Preliminary Plots of Fluence and Temperature	003415	2	Cross Reference
[19] Fatigue Methodology for Graphite Core Components	008160	1	Cross Reference
[20] NUMARK Associates, Inc., 'Additional Technical Information in Support of the U.S. Nuclear Regulatory Commission Review of the 2017 Edition of ASME Code, Section III, Division 5, "High Temperature Reactors," Subsection HH, "Class A Nonmetallic Core Support Structures," Subpart A, "Graphite Materials."'	ML21109A123	May 2021	Reference
[21] Graphite Material Model Overview: IGNIS	009815	1	Cross Reference
[22] US NRC NUREG, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs)", Volume 5	NUREG/CR-6944 (ML081140463)	2008	Reference
[23] ASME, Quality Assurance Requirements for Nuclear Facility Applications	NQA-1	2015	Reference
[24] Core Leakage and Bypass Flow Analysis Plan	006676	1	Cross Reference
[25] W. Windes, Presentation: Advanced Graphite Creep (AGC) Experiment, GCR Review Meeting, Idaho National Laboratory, July 2020	N/A	2020	Reference
[26] M. Heijna et al., Comparison of irradiation behaviour of HTR graphite grades, Journal of Nuclear Materials 492 pp 148-156	http://dx.doi.org/10.1016/j.jnucmat.2017.05.012	2017	Reference
[27] A.A. Campbell, Y. Katoh, Summary Report on Effects of Irradiation on Material IG-110 -Prepared for Toyo Tanso Co., Ltd.	ORNL/TM-2018/1040	2018	Reference
[28] Campbell, A.A. et al., "Development of a Graphite Irradiation Qualification Plan for the XE-100 Reactor"	PVP2023-106610	2023	Reference
[29] Graphite Core Assembly (GCA) Crack Propagation Evaluation Methodology	009102	1	Cross Reference
[30] Graphite Core Assembly Oxidation Analysis Plan	008947	1	Cross Reference
[31] Reactor Core Steam Oxidation Chronic Methodology	009163	1	Cross Reference
[32] Reactor Core Air Oxidation Modelling Methodology	008948	2	Cross Reference
[33] Reactor Core Irradiation and Oxidation Integrated Effects Methodology	009165	1	Cross Reference
[34] Xe-100 Resolution of Open Issues in the Methodology of Modelling Oxidation Reactions During SGTR Service Receipt Inspection Report	006696	2	Cross Reference
[35] ORNL Graphite Irradiation Program Work Package Summary	008777	1	Cross Reference
[36] Preliminary Assessment of Wear of Graphite Components and Fuel Pebbles	009099	2	Cross Reference
[37] Transmittal of Presentation Slides on Graphite Seismic, Transient, and Fatigue Methodologies	009162	1	Cross Reference

Document Title Cross References: X-energy documents that <u>may</u> impact the content of this document. References: X-energy or other documents that <u>will not</u> impact the content of this document	**Document No.**	**Rev./ Date of Issuance**	**Cross Reference/ Reference**
[38] GCA Comprehensive Test Plan	009185	1	Cross Reference
[39] GCA Graphite Material Qualification Plan	009186	~~1~~2	Cross Reference
[40] Generic Material Specification for XE-100 Nuclear Graphite Billets	008314	1	Cross Reference
[41] GCA Preliminary Chronic Oxidation Assessment Methodology	008946	1	Cross Reference
[42] Uncertainty Strategy for Stress Analysis of GCCs	009097	1	Cross Reference
[43] GCA Damage Tolerance Assessment Methodology	009049	1	Cross Reference
[44] ASME Code Compliance Envelope Plots for GCA	009801	1	Cross Reference
[45] Graphite Properties Modelling Methodology - Wigner Energy	009098	1	Cross Reference
[46] Xe-100 Required and PRA Safety Functions	001279	6	Cross Reference
[47] GCA Preliminary Failure Modes and Effects Analysis	008744	1	Cross Reference
[48] Preliminary Scoping Effect of Graphite Side Reflector Interface on Thermal Resistance	009103	1	Cross Reference
[49] Effect of Graphite Cracks on Heat Removal Capacities	009104	1	Cross Reference
[50] GCA CAD Model Release Memo	009858	1	Cross Reference
[51] Understanding the Effect of Specimen Size on the Properties of Fine-Grain Isotropic Nuclear Graphite	ORNL/TM-2020/1573	2020	Reference
[52] Xe-100 Mechanistic Source Term Approach Licensing Topical Report	000632	2	Cross Reference
[53] Xe-100 Graphite Core Assembly (GCA) ASME BPVC Compliance Assessment	009152	1	Cross Reference
[54] ASME Interpretation, “Quality assurance requirements for representative and historical data used in design.”	BPV III-5-24-04	10/17/24	Reference
[55] Guidance For a Technology-Inclusive Content-of-Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors	RG 1.253	August 2023	Reference
[56] Preliminary Scoping Inner Side Reflector Reactivity Effect Analysis	008243	1	Cross Reference

Appendix A. ASME BPVC III-5 Compliance Summary

To facilitate NRC's review, this appendix compares the graphite material qualification methodology presented in Section 4 and the design methodology presented in Section 6 against the approach in ASME BPVC III-5 [3]. This summary is also documented in Reference [53].

A.1. Graphite Material Qualification Methodology (Section 4)

Table 16 through Table 18 highlight the status of the compliance of the graphite material qualification methodology presented in Section 4 against the applicable requirements from ASME BPVC III-5.

The column of "current compliance" shows if the currently available data (historical data) meets the code, e.g., enveloping reactor normal operation condition and coolant chemistry. The column of "Planned Compliance" shows if the Xe-100 GCA design approach plans to meet the Code either by new test, acquiring additional data or extrapolating available data. Items marked with "No" in both columns identify a Code requirement to which the Xe-100 GCA design will plan to take exception.

Table 16: *ASME Code Compliance Summary for As-Manufactured Graphite*

Qualification Items	Corresponding Code Article	Current Compliance	Planned Compliance
Sampling Plan	HHA-III-4100	N/A	Yes
Bulk Ensity Tensile Strength, Compressive Strength Dynamic Moduli (Room Temperature) Thermal Conductivity Coefficient of Thermal Expansion	HHA-II-2000 HHA-III-3100	To be evaluated	Yes
Flexural Strength	HHA-II-2000	No (Perform 3-points flexural strength test)	No See justification in Section 4.4.4 Test Matrix
Dynamic Moduli (Elevated Temperature)	HHA-II-2000 HHA-III-3100 HHA-III-5000	No (Data envelop reactor condition, but not meeting sampling requirement)	No See justification in Appendix B of Ref. [39]

Qualification Items	**Corresponding Code Article**	**Current Compliance**	**Planned Compliance**
Fracture Toughness (Critical Stress Intensity Factor)	HHA-II-2000 HHA-III-3100 HHA-III-5000	No (not meet sampling requirement)	No See justification in Section 4.4.4 Test Matrix
Property Variation	HAB-4557.2 HHA-III-5000	N/A	Yes (Acquiring quality control data from manufacturer)

Table 17: *ASME Code Compliance Summary for Irradiated Graphite*

Qualification Items	**Corresponding Code Article**	**Current Compliance**	**Planned Compliance**
Design Data General Requirements	HHA-II-4100	No	Partial compliance. See discussion in Section 4.5.8: (a) Justify extrapolation based on sensitivity studies and safety analysis (show uncertainty is tolerable with respect to RSFs) (b) Reporting format will be followed (c), (d) Creep model parameters will be given for a generic model, without full coverage of temperature range for each graphite grade (e) Limited extrapolation beyond the irradiation temperature/dose envelope is expected
Design Data Graphite Grades	HHA-III-1000	Yes: grades conform to ASTM D7219, coke and processing route will be maintained	Yes
General Requirements	HHA-III-2000	Partial compliance: manufacturing process for tested material was not qualified to NQA-1	Yes: X-energy to justify acceptability of QA for test material
Properties to be Determined	HHA-III-3300	No	Partial compliance, see discussion above for HHA-II-4100 and in Section 4.5.8
Representative Data	HHA-III-4200	N/A	Yes: X-energy to justify representativeness of tested material using manufacturer's data
Historical Data	HHA-III-5000	Partial compliance: manufacturing process and testing processes were not NQA-1 qualified	Yes: X-energy to justify acceptability of QA for historic data

Table 18: ***ASME Code Compliance Summary for Oxidized Graphite***

Qualification Items	Corresponding Code Article	Current Compliance	Planned Compliance
Local weight loss Oxidants: steam, air	HHA-3141 HHA-III-5000	To be evaluated	Yes
Oxidation impact to strength, elastic modulus, and thermal conductivity.	HHA-3141 HHA-II-2000 HHA-III-3200 HHA-III-5000	Yes	Yes

A.2. Design Methodology (Section 6)

The compliance of the design methodology presented in Section 6 against the applicable requirements from ASME BPVC III-5 is summarized in Section 6.3.

Appendix B. Safety Basis Example

B.1. Safety Basis Example: Inner Side Reflector

B.1.1. Introduction

This Appendix provides an example of the NEI 18-04 process described in Section 7. It is applied to the Inner Side Reflector (ISR) and intends to illustrate the general approach that may be taken to structuring the safety case arguments for one set of GCCs. It is based on the DTA methodology set out in Reference [43] and intends to be consistent with the NEI 18-04 methodology described in more detail in Reference [13]. The focus here is on the inner side reflector GCCs. The arguments and evidence here are preliminary and subject to change.

The safety case starts from the RSFs assigned to the GCA (Table 1), which are: RSF 1.1.2 (maintain long-term subcriticality), RSF 1.2.1 (maintain core heat removal through passive means), and RSF 1.4.1 (maintain core geometry).

B.1.2. Inner Side Reflector

The ISR is a GCC whose primary function is to reflect neutrons leaking from the reactor core back into the core to improve neutron economy from a criticality standpoint (See Figure *25* [50] for a visual representation of the ISR). However, this objective is not a safety function. During certain accident scenarios, the ISR function is to transfer heat away from the reactor core to the reactor vessel wall, and ultimately to the RCCS. This is associated with the heat removal RSF (1.2.1.1c). The ISR also plays a role in maintaining the geometry of the pebble bed core and of the GCA (RSF 1.4.1.3), and is adjacent to the control rod channels, the geometry of which must be maintained to meet RSF 1.1.2.1b and 1.1.2.1c. These roles are considered below in the context of irradiation damage and cracking that could occur during the reactor lifetime.

Relevant damage mechanisms for this GCC have been identified in Reference [47], which include the following: cracking, dimensional distortion, chemical degradation, thermal conductivity decrease, and erosion. Reliability and capability targets can be derived from safety analysis evaluations that look into how degradation in graphite properties and GCC integrity affects the ability to meet these RSFs.

B.1.3. Control Heat Removal RSF

It will be shown that the ISR assembly can meet RSF 1.2.1.1c even in the presence of irradiation damage (e.g., dimensional change, etc.) and a number of cracks exceeding the expected level. The argument will have a structure similar to the reasoning below:

- In certain accidents, e.g. Pressurized Loss of Forced Cooling or Depressurized Loss of Forced Cooling, it is necessary to transfer heat through the GCA, primarily by conduction. Passive decay heat removal forms a key part of the Xe-100’s inherent safety design.

- Resistance to heat transfer comes from material properties (higher thermal resistance in irradiated graphite), gaps as part of the design (between inner and outer reflector), and new gaps introduced by irradiation dimensional change or cracking. Irradiated properties will be determined using existing irradiated data and further data from the MTR. Preliminary analysis quantifies the effect of gaps between the inner and outer reflector on thermal resistance[48]. Circumferential gaps have the greatest effect on heat transfer out of the core, with radial gaps in the GCA being less important [49]. New gaps due to cracking would at worst have a similar effect to the circumferential gaps, more likely the effect would be smaller (radial gaps).
- Overall, the effect of cracks and gaps can be quantified. Using conservative assumptions, cracking is unlikely to increase the thermal resistance of the side reflector by more than around 1% even if 10% of the ISR GCCs crack. The gap between inner and outer side reflector GCCs adds 5 - 10% to thermal resistance, and assuming 10% of the GCCs crack and each crack is equivalent to an inner/outer reflector gap, the total increase in heat resistance of the inner side reflector due to cracks would be less than 1%. This is small compared to uncertainty from the effect of irradiation damage and judged to be tolerable. Increased thermal resistance due to postulated cracks can be accounted for in safety analysis.

Based on this reasoning, it is expected that the highest level of structural reliability is not required to meet RSF 1.2.1.1c and these GCCs could be assigned a Structural Reliability Class (SRC) as defined in the ASME BPVC of SRC-2. To ensure appropriate transfer of heat through conduction, the irradiated thermal conductivity and quality of as-manufactured material play a larger role.

B.1.4. Maintain Geometry RSF

When considering the importance of ISR component structural integrity for reactivity, it will be shown that RSF 1.4.1.3 can be met even in the presence of significant damage, supporting the assignment to SRC-2. The following argument is given for illustration:

- While the ISR GCCs are supported by the outer side reflector bricks and do not themselves bear the load of other GCCs, the presence of the ISR helps maintain geometry of core, contributing to the reactivity characteristics of the reactor.
- In most extreme example of GCC failure, we could assume complete loss of the ISR – see preliminary analysis in Reference [56] . This would allow the core to expand radially, flattening and moving downwards (away from control rods) into a lower-leakage configuration. Despite this, preliminary analysis of this extreme scenario shows tolerable reactivity effects, and the reactor still shuts down under inherent reactivity feedback due to the nature of TRISO fuel.

It is expected that partial failure of a fraction of ISR GCCs exceeding the threshold for SRC-2 would have a very small effect on core geometry and would therefore allow RSF 1.4.1.3 to be met even with cracking.

B.1.5. Control Reactivity RSF

RSF 1.1.2.1b and 1.1.2.1c require that the GCA allows reactivity control by movable poisons (i.e. control rod insertion). The control rods are inserted through a channel formed by dedicated control rod channel sleeve GCCs. Each control rod channel sleeve may be held in place by the ISR GCC above and below as well as the adjacent ISR GCC. Continuity of the control rod channel is maintained by the sleeves, and while potentially large displacement of the ISR may affect the shape of the control rod channel, isolated cracking of GCCs in the ISR is not expected to directly compromise this RSF. Damage to the ISR may be included in structural models to assess control rod insertion. While further work is required to substantiate these assumptions, it is expected that cracking in the ISR exceeding levels allowed by SRC-2 would be tolerable with respect to control rod insertion.

B.2. Conclusions

When considering heat removal, the effect of cracking is expected to be small, compared to irradiation-induced changes in graphite properties such as thermal conductivity, and tolerable overall. Because of this, the number of cracks is not a primary concern for heat conduction and the GCC can be designed to allow more cracks, e.g., by specifying SRC-2 (ASME Code HHA-3111) [3].

Similarly, it is expected that the effects of cracking on core geometry will not be detrimental to maintaining support of the pebble bed reactor core, and it will be possible to shut down the reactor using control rods even if some GCCs adjacent to the control rod channels have cracked. While further analysis is required to substantiate these, the assignment of SRC-2 will likely be appropriate. This represents the Special Treatment (*G-3*) in Figure 24 to ensure this GCC meets the required level of structural integrity within the risk-informed framework of NEI 18-04.

[[

]]P,E

Figure 25: *Inner Side Reflector*