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Project No. 99902071

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Submission of X Energy, LLC (X-energy) Xe-100 Licensing Topical Report: Graphite Qualification Methodology

The purpose of this letter is to submit the subject licensing topical report (LTR) to the U.S Nuclear Regulatory Commission (NRC) on behalf of X Energy, LLC ("X-energy"). This submission provides both proprietary and non-proprietary versions of the report. It is provided for NRC review and approval as indicated in the report and is expected to be referenced in future Xe-100 licensing applications. The specific review schedule will continue to be developed with X-energy's NRC project manager.

This report contains commercially sensitive, proprietary information and, as such, we are requesting that this information be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, request for withholding," paragraph (a)(4). Additionally, certain information in this report was determined to contain Export Controlled Information (ECI). This information must be protected from disclosure pursuant to 10 CFR 810. Enclosure 1 is the Proprietary, Non-Public version of the report which contains nonredacted sensitive, proprietary information that is appropriately marked. Enclosure 2 provides an affidavit with the basis for this request. Enclosure 3 provides a redacted copy of the report that contains non-proprietary, publicly available content.

This letter contains no commitments. If you have any questions or require additional information, please contact Paul Loza at ploza@x-energy.com or Ingrid Nordby at inordby@x-energy.com.

Sincerely,

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Enclosures:

1) Xe-100 Licensing Topical Report, "Graphite Qualification Methodology" (Proprietary)

2) Affidavit Supporting Request for Withholding from Public Disclosure (10 CFR 2.390)

3) Xe-100 Licensing Topical Report, "Graphite Qualification Methodology" (Non-Proprietary)

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Enclosure 1

X Energy, LLC
Xe-100 Licensing Topical Report,
“Graphite Qualification Methodology”
(Proprietary)

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Enclosure 2

Affidavit Supporting Request for Withholding from Public Disclosure

(10 CFR 2.390)

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Affidavit Supporting Request for Withholding from Public Disclosure (10 CFR 2.390)

I, Travis A. Chapman, Manager, U.S. Licensing, Xe-100 Program, of X Energy, LLC (X-energy) do hereby affirm and state:

1. I am authorized to execute this affidavit on behalf of X-energy. I am further authorized to review information submitted to or discussed with the Nuclear Regulatory Commission (NRC) and apply for the withholding of information from disclosure. The purpose of this affidavit is to provide the information required by 10 CFR 2.390(b) in support of X-energy's request for proprietary treatment of certain commercial information submitted in Enclosure 1 to X-energy's letter XE-NRC-2022-027 from myself to the NRC which provides the X-energy Topical Report, "Graphite Qualification Methodology" for the X-energy Xe-100 Nuclear Reactor.

2. I have knowledge of the criteria used by X-energy in designating information as sensitive, proprietary, confidential, and export-controlled.

3. Pursuant to the provision of paragraph (b)(4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

a. The information sought to be withheld from public disclosure in Enclosure 1 is owned by X-energy. This information was prepared with the explicit understanding that the information itself would be treated as proprietary and confidential and has been held in confidence by X-energy.

b. The information sought to be protected in Enclosure 1 is not available to the public.

c. The information contained in Enclosure 1 is of the type that is customarily held in confidence by X-energy, and there is a rational basis for doing so. The information X-energy is requesting to be withheld from public disclosure includes technical information related to the design, analysis and operations associated with our Xe-100 high-temperature, gas-cooled, pebble bed advanced reactor design that directly impact our business development and commercialization efforts. X-energy limits access to this proprietary and confidential information in order to maintain confidentiality.

d. Enclosure 1 contains information about the planned activities of X-energy related to the development of the Xe-100 design bases, TRISO-X fuel design bases, forecast design development timeframes, and relate to the commercialization strategy for our Xe-100 advanced reactor. Public disclosure of the information contained in Enclosure 1 would create substantial harm to X-energy because it would reveal valuable technical information regarding X-energy's design development, competitive expectations, assumptions, current position and strategy. Its use by a competitor could substantially improve the competitor's position in the design, manufacture, licensing, construction and operation of a similar competing product.

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e. Additionally, Enclosure 1 is assessed to contain certain information that is considered Export Controlled Information (ECI) under the provisions of 10 CFR 810. I have personal knowledge of the criteria used by X-energy to evaluate documents for ECI and affirm that this information should be withheld from public disclosure.

f. The Proprietary Information contained in Enclosure 1 is transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390; it is to be received in confidence by the NRC. The information is properly marked.

I declare under the penalty of perjury that the foregoing is true and correct. Executed on October 31, 2022.

Sincerely,

DocuSigned by:

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Travis Chapman
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Enclosure 3

X Energy, LLC
Xe-100 Licensing Topical Report,
"Graphite Qualification Methodology"
(Non-Proprietary)

Xe-100 Licensing Topical Report

Graphite Qualification Methodology

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Department of Energy Acknowledgement and Disclaimer

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SYNOPSIS

This report describes the method for qualification of graphite core structures used by X-energy for the Xe-100 reactor. 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 structure, 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 graphite core structure 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 importance of the safety functions to be performed. The safety functions of the core graphite structure are ensured by the structural integrity and material properties of the graphite throughout the life of the components. The qualification of the graphite material used for the construction of the graphite core structure is part of the requirements to demonstrate the structural integrity and material properties, and results in a quality product consistent with the required safety functions. The qualification methodology described herein for the Xe-100 graphite core structures follows the approach described in DG-1380 (RG 1.87 R2) and ASME Code, Section III, Division 5 [12]. Departures from ASME Code requirements, if any, will be identified and justified as part of the Construction Permit and Operating License Applications.

CONFIGURATION CONTROL

Document Change History

Rev.	Date	Preparer	Changes
1	28-Oct-2022	P Loza	Sr Licensing Engineer

Document Approval

Action	Designation	Name	Signature	Date
Preparer	Sr Licensing Engineer	P Loza		
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Approver	Manager, Xe-100 Design	W Kriel		

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ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
Ø	Diameter
AG	Against Grain
AGC	Advanced Graphite Creep
AGR	Advanced Gas-cooled Reactor (UK type)
ANLWR	Advanced Non Light Water Reactor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CB	Core Barrel
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations (USA)
CTE	Coefficient of Thermal Expansion
DOE	Department of Energy (USA)
dpa	displacements per atom
FEA	Finite Element Analysis
FGG	Fine Grain Graphite
GCC	Graphite Core Component
GDC	General Design Criterion
HDG	High Dose Graphite
HFIR	High Flux Isotope Reactor
HOPG	Highly-Oriented Pyrolytic Graphite
K	Plain Strain Fracture
LWR	Light Water Reactor
MGG	Medium Grain Graphite
MDS	Material Data Sheet
MTR	Material Test Reactor
MWth	Megawatt thermal
NGNP	Next Generation Nuclear Plant
ORNL	Oak Ridge National Laboratory
Pa	Pascals

Abbreviation or Acronym	Definition
PDC	Principal Design Criterion
PIRT	Phenomena Identification and Ranking Table
RCCS	Reactor Cavity Cooling System
RPV	Reactor pressure vessel
RSF	Required Safety Function
SSC	Structures, Systems, and Components
TRISO	Tristructural isotropic (coated particle fuel)
US	United States
USA	United States of America
WG	With Grain
X-energy	X Energy, LLC

1. INTRODUCTION

X energy, LLC (X-energy) is developing the Xe-100 high-temperature gas-cooled reactor design as an advanced nuclear power source for multiple applications. The Xe-100 is a Generation IV Advanced Reactor based on pebble bed HTGR technology, which utilizes U.S.-developed Uranium Oxycarbide TRISO-coated fuel embedded in spherical fuel elements to form fuel pebbles. The ‘pebbles’ are roughly the size of billiard balls, and the TRISO coating creates an airtight seal around the uranium kernel which helps retain fission products and gases that are produced during operations. The pebble graphite is the primary fuel moderator. The fuel pebbles are contained in a graphite core structure, which serves as an additional moderator and supports the pebble bed of fuel. 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 reactor generates 200 MWth and produces high-quality, super-heated steam at 565°C and 16.3 MPa, which is suitable for many different energy applications. This reactor design uses a safety-related core graphite structure, which supports the RSFs of control reactivity and control heat removal. This graphite requires a qualification program to ensure the performance of these safety-related functions. This report provides the details of the methodology for the safety-related graphite core structure material qualification for the Xe-100.

This methodology is largely consistent with the Section III, Division 5 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), “Rules for Construction of Nuclear Facility Components, High-Temperature Reactors” (abbreviated as “Division 5 code” or “code”) [12] requirements. The code year used will be identified in the licensing submittal. Departures from ASME Code requirements, if any, will be identified and justified as part of the Construction Permit and Operating License Applications.

The 2017 code version does not specifically list any qualified structural graphite material for Xe-100 reactor use, rather, it explains how to qualify a grade of graphite, including the requirements to address oxidation and irradiation effects.

This report describes the qualification of unirradiated and irradiated structural graphite, as well as the qualification of the graphite oxidation behavior. Seismic qualification is outside the scope of this report.

Note that inspection and aging management considerations for the Xe-100 will be described as part of the licensing application and their approval are not part of this report.

X-energy requests NRC review and approval of the Xe-100 safety-related structural graphite material qualification methodology for use by 10 CFR 50 or 10 CFR 52 licensing applicants.

1.1. DESIGN OF THE XE-100

The design of the Xe-100 is described in the following sections to facilitate NRC review and approval of this report. Essential Xe-100 design features that support the safety review of this report are not expected to change in the remaining design completion.

1.1.1. Design Background

Graphite has been successfully used as a moderator in nuclear reactors since the first Chicago Pile as shown in Figure 1. Graphite is used in the Xe-100 design in both the pebble fuel and the reflector.

The defining characteristic of the pebble bed reactor is its spherical fuel elements or pebbles. Each fuel sphere consists of approximately 19,000 UCO TRISO-coated particles that are embedded in matrix graphite to form a 60mm diameter pebble. The UCO kernels make use of 15.5 wt% high-assay, low-enriched uranium for the equilibrium pebble, and a slightly reduced enrichment for start-up pebbles. Fuel pebbles are circulated from the top of the pebble bed core to the bottom, extracted, measured for physical integrity and burn-up, and either recirculated for another pass or discharged as used fuel. Online refueling allows for significantly reduced excess reactivity in the reactor at any given time, limiting the magnitude and contributing to the slow response of the reactor to reactivity transients.

The reactor operates in the thermal neutron spectrum and uses graphite as the moderator. The Xe-100 fueled zone is a cylindrical volume filled with approximately 220,000 fuel pebbles that make up the pebble bed core.

The pebble bed core is formed by and contained within a structure comprised of keyed and dowelled graphite blocks that both define the geometry of the core and provide for neutron reflection and moderation. This core graphite structure also establishes the internal flow paths that route the helium heat transport medium from the reactor inlet through the pebble bed core to the reactor outlet. Some helium is allowed to bypass the pebble bed core to cool the core graphite structure blocks.

The core graphite structures are enclosed within a metallic core barrel (CB) to provide both lateral and vertical support. These and other components internal to the reactor are contained and supported within a steel Reactor Pressure Vessel (RPV).

Additional information on the Xe-100 design is given in the “Xe-100 Technical Report Technology Description” document, but this is not relied on to support the NRC findings to approve the subject report. See select design data for the Xe-100 in Table 1.

1.1.2. Key Features

The Xe-100 safety design approach ultimately relies on the inherent properties of selected materials, the laws of physics governing heat transport and reactivity, and the incorporation of passive SSCs.

Key design features of the Xe-100 include high-temperature fuel and a chemically inert, neutronically transparent heat removal medium. This combination of the TRISO particle fuel and stable helium results in a robust reactor design with intrinsic passive safety.

1.1.2.1. Heat Transport Systems

The Xe-100 has a single loop for heat removal via the steam generator.

The inherent safety of the Xe-100 is enhanced using single-phase helium (He) as an inert heat transport fluid, the large thermal inertia of the reactor system materials, and the associated slow response of the reactor to upset conditions.

The reactor inlet (260°C) and outlet (750°C) temperatures were selected to provide significant design margins for the fuel, core metallic structures, and the RPV. The maximum fuel temperatures during normal operation are below 1000°C. The reactor's internal flow paths are arranged such that its hottest parts, i.e. the reactor core, are at the lowest pressure. There is thus an inflow from the colder metallic regions to the hotter graphite regions. The inlet temperature of 260°C ensures that the RPV and CB can be designed to meet existing ASME construction codes.

The design also includes passive decay heat removal via the RCCS supported by:

- Low power density (to ensure that decay heat can be passively removed while maintaining acceptable fuel temperatures),
- Optimal core geometry aspect (Height/Diameter) ratio (to match the decay heat generation with the capability of the passive heat removal pathway),
- Strong negative temperature coefficient of reactivity over the full operational range of the Xe-100 (passive shutdown of the nuclear chain reaction in the event of the core heating up),
- Passive decay heat removal capability to ensure long term plant safety without the need for emergency power.

1.1.2.2. Containment Approach

The Xe-100 design uses a functional containment as defined in RG 1.232 [1] and example design criterion for the functional containment (MHTGR Criterion 16). Note that the NRC has reviewed the functional containment concept and found it "generally acceptable," provided that "appropriate performance requirements and criteria" are developed. See the proposed methodology for establishing functional containment performance criteria for non-LWRs in Commission-approved SECY-18-0096.

Note that design relies primarily on the multiple barriers within the TRISO fuel particles and fuel pebble to ensure that the dose at the site boundary (from postulated events) meets regulatory limits.

1.1.2.3. Graphite Core Structure

The Xe-100 design uses a graphite reflector assembly in addition to graphite-based fuel pellets.

The Core Graphite Structures provide both neutron reflection as well as mechanical support and guidance for the fuel pebbles to move through the core. Cooling gas flow is also directed via the graphite and the flow that bypasses the core is an important design consideration. The principal design methodology for the graphite blocks is following a single-column principle. In this way, no block is supported by more than two blocks in parallel. This prevents tension loading of the graphite as it undergoes shape changes during its lifetime. [[

]]P,E Figure 2: shows a cross-section of the graphite reflector highlighting the sub-components. Additional information on the Xe-100 design is given in the "Xe-100 Technical Report Technology Description" document.

The graphite core structure performance supports the Xe-100 Required Safety Functions (RSFs) [2] of Control Reactivity and Control Heat Removal.

To support the RSF of Control Reactivity, the graphite core structure maintains core geometry and the pebble flow path boundary for the fuel pebbles to move through the core. This ensures the core/fuel geometry remains consistent with the assumptions made in reactor physics analyses. The core graphite structure also provides channels for the insertion of control/shutdown rods.

To support the RSF of Control Heat Removal, the core graphite structure provides a heat transfer path from the core through the core graphite structure. This heat transfer path to the vessel, then the reactor cavity and finally to the Reactor Cavity Cooling System (RCCS) and the environment shall be maintained for all AOOs, DBEs and DBAs regardless of the primary heat transport system pressure and fluid composition. The design manages block geometries and tolerances to support heat conduction and radiation.

The core graphite structure maintains core geometry to support the NSRST PSF for forced circulation. The core graphite structure provides a helium coolant flow path through the core.

The Safety Functions from NGNP Graphite Phenomena Identification and Ranking Tables (PIRTs) [3] listed below also support our analyzed RSFs, NSRST PSFs and subfunctions as shown:

(1) ability to maintain passive heat transfer; RSF – Control heat Removal

(2) maintain ability to control reactivity; RSF – Control reactivity

(3) thermal protection of adjacent components; RSF – Control heat Removal

(4) shielding of adjacent components; sub-RSF – Maintain Core Geometry

(5) maintain coolant flow path; PSF – Control Heat Removal

(6) prevent excessive mechanical load on the fuel; and sub-RSF – Maintain Core Geometry

(7) minimize activity in the coolant. RSF – Retain Radionuclides

The core graphite structure also shields the Core Barrel and Reactor Pressure Vessel from neutron irradiation and provides thermal insulation from hot helium in the core.

The NGNP PIRT's [3] major safety significant functional requirements for the graphite core structure are also aligned with similar Xe 100 core graphite structure functions:

- Maintain core geometry,
- Provide neutron reflection and some moderation,
- Provide helium coolant flow paths,
- Allow movement of pebbles through the core and prevent pebble blockages for online fueling and defueling,
- Limit neutron streaming and attenuate fast neutron flux from the core to shield the CB and RPV,
- Provide thermal insulation from hot helium for metallic CB structures,
- Provide thermal conduction from the core and hot helium to metallic CB structures, RPV, and Reactor Cavity Cooling System (RCCS) when RCCS is being used to cool the core.
- Provide channels for insertion of control/shutdown rods,
- Limit core bypass/leakage flows (i.e., radially through the side reflector).

The graphite reflector is made of three subassemblies: the side reflector, the top reflector, and the bottom reflector. Each of these subassemblies is further broken down into its components in subsequent sections.

1.1.2.3.1. Materials and Masses

The graphite reflector structures will be made from multiple forms of graphite materials. All materials shall conform to the minimum requirements of Mandatory Appendix HHA-I of Subsection HH of Division 5 of the ASME B&PV Code, Section III [12]. [[

]]^{P,E}

The actual density of graphite is about 80% of the theoretical density and this is an important consideration when calculating the weights of the overall system.

Table 2 provides the different sections of the reflector and their respective volumes, graphite types, and graphite masses assuming a graphite density of 1,850 kg/m^3.

1.1.2.3.2. Graphite Core Structure Design Considerations

The design of the graphite to act as a reflector, moderator, shield, coolant 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 with temperature and fast neutron fluence are complex but well characterized as graphite has been used since the very first reactors. Owing to the mechanical, geometrical, thermal, and neutronic changes of the graphite, strict design principles must be followed. The blocks must follow the "single column principle", cannot be directly bonded, and gas leakage must be minimized. Additionally, the side reflector must be designed with mid-life replacement in mind as it reaches the end of life (between turnaround and crossover)[4].

The radiation damage mechanism of concern for graphite is when fast neutrons interact with carbon atoms in a kinetic fashion causing a "cascade" plume of atoms within the lattice. This cascade event has a duration on the order of picoseconds after which the vast majority of the atoms come to rest in the original graphite planes. However, a small fraction of displaced atoms results in defects in the graphite crystal structure. The nature of the damage to the graphite crystal lattice after a cascade event is a strong function of irradiation temperature, hence all irradiation-effect properties are likewise a function of irradiation temperature.

1.1.2.3.2.1. Dimensional Change

For temperatures of relevance to HTGRs (>250°C), fast neutron irradiation leads to an effectively continuous and non-saturating process of A-axis contraction (parallel to the basal planes) and C-axis expansion (perpendicular to the basal planes) in graphite crystals. See Figure 3. This behavior can be seen on a macroscopic scale in the irradiation of Highly-Oriented Pyrolytic Graphite (HOPG), with the dimensional change depending on the irradiation temperature and graphitization temperature [5, 6]. Unlike HOPG, nuclear graphite is a polygranular material containing many graphite crystallites with different orientations and various scales of porosity. The bulk dimensional change of graphite under irradiation depends on the net bias in the orientation of constituent crystallites as well as the pore structure of the material. With careful control of raw materials and manufacturing processes, graphite can be manufactured to have isotropic or near-isotropic properties (ASTM D7219-19), leading to near-isotropic irradiation-induced dimensional change. In these materials, graphite initially shrinks, with the rate of shrinkage gradually decreasing up to a point referred to as dimensional change "turnaround".

Beyond turnaround, the material expands with irradiation, passing through the point of "nil swelling" where it has returned to its original volume. See Figure 4 and Figure 5.

In the initial shrinkage process, C-axis expansion is accommodated by porosity. With continued irradiation, this porosity becomes used up. At the same time, stress accumulates within the material, generating new porosity. The combined processes of crystal dimensional change, pore closure, and pore generation lead to the observed bulk dimensional change. Graphite dimensional change behavior depends on a range of factors including the graphite grade (defined by the raw materials, manufacturing processes, and graphitization temperature), the irradiation temperature, applied stress, and oxidation, in addition to fast neutron fluence.

A perfect, unirradiated graphite crystal is composed of graphene planes with strong in-plane covalent bonding and relatively weak interplane van der Walls forces. The introduction of irradiation-induced defects can impact the graphite engineering properties.

1.1.2.3.2.2. Thermal Conductivity

Conduction of heat in graphite occurs primarily through lattice vibrations (phonons). Crystal lattice imperfections lead to rapid degradation in thermal conductivity, with this occurring more quickly at lower temperatures. A further decrease in thermal conductivity is seen at high fast neutron doses (around the point of dimensional change turnaround) due to the generation of new porosity spaces.

Regarding erosion, DG-1380 [7] states "The NRC 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."

The NGNP High Temperature Materials White Paper [8] noted "Erosion and corrosion can reduce the wall thickness of components, and oxidation can affect the surface of materials by various means including degradation of the grain boundaries, thereby reducing the strength of the affected volume of surface material."

ASME code, Section III, Subsection NG-3121 "Construction Rules For Core Support Structures" indicates that "Material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provision made for these effects during the design or specified life of the structure by a suitable increase in or addition to the thickness of the base metal (graphite) over that determined by the design formulas." Therefore, the thickness selected for areas of the Xe-100 design subject to erosion and corrosion of graphite will include an allowance for erosion and corrosion that has not yet been established.

1.1.2.3.2.3. Mechanical Properties

Neutron irradiation has a complex effect on mechanical properties of interest to HTGR design. Elastic modulus, strength, and fracture toughness increase during the initial graphite shrinkage phase of irradiation. However, following turnaround these properties will begin to reduce and eventually substantially degrade to the point where the graphite is no longer structurally sound. This is largely due to the generation of new porosity at high fluence. In design, the return to nil-swelling is often treated as a limit for the useful life of a structural graphite component, although certain components exposed to steep gradients in fast neutron flux, high rates of oxidation, and/or particularly high applied loads may degrade more rapidly.

The coefficient of thermal expansion follows a similar process, with an initial low-dose increase followed by a decrease towards a plateau at higher fluence. While graphite is highly resistant to thermal shock, gradients in the coefficient of thermal expansion can lead to changes in stress when the reactor temperature is changed, particularly at startup and shutdown. The coefficient of thermal expansion is known to vary with irradiation temperature, fast neutron fluence, instantaneous temperature, and irradiation creep.

1.1.2.3.2.4. Oxidation

Graphite components are sensitive to oxidation-related changes in material properties. While the normal operating environment of the graphite blocks is in inert helium gas, certain accident conditions (e.g., steam generator tube ruptures) may permit an oxidizing environment to be established. Analyses of these transients shall consider environmental effects on the material properties of each graphite grade.

1.1.2.3.2.5. Reducing Life-Cycle Graphite Burden

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

Additionally, dimensional changes owing to thermal expansion warrant additional design principles. Some of the key design criteria are the use of single columns, stacking, joining, and sealing to prevent helium leakage, and designing for replacement.

1.1.2.3.2.5.1. Single Column Principle

All block stacks must conform to the “single column principle.” This means that blocks stacked in the vertical direction can only be supported by a block directly beneath. There must be no areas where the potential for bridging can occur as this will place the blocks under a tensile stress condition which could result in block cracking and ultimately failure. In areas where overhangs occur, the analysis must be conducted to ensure proper counter-levering exists without applying an excessive amount of tensile stress.

1.1.2.3.2.5.2. Stacking and Joining

Considering the changes due to temperature and neutron irradiation, it is not possible to bond the graphite blocks 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 graphite stack to keep the alignment of the blocks. The blocks are keyed in such a way as to prevent excessive movement. Rounded and chamfered edges prevent stress concentrators that could lead to chipping.

1.1.2.3.2.5.3. Sealing

One of the main graphite functions is to provide coolant flow channels, acting as the riser for the coolant. Additionally, where the coolant is in the downflowing region, flow bypass of the pebbles must be minimized. As it is not possible to bond the graphite, perfectly tight sealing is virtually impossible. However, using overlapping sleeves, dog-bone keys, rounded edges and keyways, block offsetting, and taking advantage of thermal expansion, sealing is achieved.

1.1.2.3.2.5.4. Graphite replacement

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1.2. REGULATORY INFORMATION

The planned licensing of the Xe-100 plant is under Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." Under the provisions of 10 CFR 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The General Design Criteria listed in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

Appendix A, General Design Criterion (GDC) 1, "Quality Standards and Records" requires for water-cooled reactors that structures, systems, and components (SSCs) be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Further, GDC 1 requires that where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the RSF. Appendix A to Part 50 also states that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the principal design criteria for such other units.

Regulatory Guide (RG) 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," [1] lists the NRC's proposed guidance on how the GDC in 10 CFR Part 50, Appendix A, may be adapted for non-LWR designs. Advanced Reactor Design Criterion 1 in RG 1.232 states that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

Xe-100 Principal Design Criteria Licensing Topical Report, XE00-Z-ZZ-ZZ-004799, Revision 1 [7] describes the development of the PDC for the X-energy Xe-100 pebble-bed, high-temperature gas-cooled reactor (HTGR). The PDC are developed using guidance from Regulatory Guide (RG) 1.232, NEI 21-07, "Technology Inclusive Guidance for Non-Light Water Reactor Safety Analysis Report: Content for Applicants Using the NEI 18-04 Methodology," Revision 1 [10], and Xe-100-specific RSFs and design requirements.

Xe-100 PDC 1 requires "Safety-significant structures, systems, and components shall be designed, fabricated, erected, and tested to quality standards commensurate with the safety significance of the functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the safety-significant function. A quality assurance program shall be established and implemented in order to provide reasonable assurance that these structures, systems, and components will satisfactorily perform their safety-significant functions. Appropriate records of the design, fabrication, erection and testing of safety-significant structures, systems, and components shall be maintained by or under the control of the nuclear power unit licensee for an appropriate period of time."

Xe-100 PDC 10 requires "The reactor system and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

Xe-100 PDC 16 requires "A reactor functional containment, consisting of multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to control the release of radioactivity to the environment and to ensure that the functional containment design conditions that are required to perform safety - significant functions are not exceeded for as long as licensing basis event conditions require."

Xe-100 PDC 34 requires "A passive system to remove residual heat shall be provided. During postulated accidents, the passive system required safety function shall provide effective cooling. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure the system required safety function can be accomplished in such a manner that acceptable fuel, core components, reactor pressure vessel, and reactor building temperature limits are not exceeded."

Xe-100 PDC 70 requires "The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.

The ASME Code establishes standards relating to the pressure integrity of boilers, pressure vessels, transport tanks, and nuclear components. ASME Code, Section III, "Rules for Construction of Nuclear Facility Components," establishes rules for the materials, design, construction, testing, and quality assurance of mechanical systems and components and their support of high-temperature reactors. ASME Code, Section III, Division 1, establishes rules for components where material strength and deformation are time-independent and the maximum allowable temperature is 425°C (800°F). The NRC incorporates by reference portions of the ASME Code, Section III, Division 1, in 10 CFR 50.55a, "Codes and standards."

As discussed in NUREG-2245, Technical Review of the 2017 Edition of ASME Code, Section III, Division 5, "High Temperature Reactors," Draft Report for Comment [11], advanced non-light water reactor (ANLWR) designers have expressed interest in operating in thermal ranges that vary widely between 425 and 1,000 degrees C (800 and 1,832 degrees F), but as of October 2022, there is no NRC-endorsed code of construction for nuclear reactors operating above 425 degrees C (800 degrees F). Also as discussed in NUREG-2245, the NRC staff recognizes that the absence of an NRC-endorsed code of construction for nuclear reactors operating above 425 °C (800° F)) is a significant obstacle for ANLWR designers as the review of an elevated temperature code of construction during a licensing review of a new nuclear power plant would result in substantial costs and a longer review schedule for the requested action. ASME Code, Section III, Division 5, "High Temperature Reactors," [12] extends the rules in ASME Code, Section III, Division 1, to provide consensus standards for the construction of metallic nuclear plant components that would operate within the creep regime (time-dependent), which would include temperatures above 425°C (800° F). In addition, ASME Code, Section III, Division 5 [12], provides new rules for the construction of certain nuclear plant components using graphite and composite materials.

Draft Regulatory Guide DG-1380, Proposed Revision 2 to Regulatory Guide 1.87, Acceptability of ASME Code, Section III, Division 5, "High Temperature Reactors," [7] describes an approach that is acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) 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. It endorses, with exceptions and limitations, the ASME BPVC Section III, "Rules for Construction of Nuclear Facility Components," Division 5 [12], "High Temperature Reactors." The qualification methodology described herein for the Xe-100 reflector graphite material follows the approach described in DG-1380 and ASME Code, Section III, Division 5 [12]. Departures from ASME Code requirements, if any, will be identified and justified as part of the Construction Permit and Operating License Applications.

2. NUCLEAR GRAPHITE AND GRAPHITE SELECTION

Nuclear-grade graphite materials are used in the Xe-100 HTGR following a long tradition of use. This section discusses graphite grading and the proposed material qualification plan.

2.1. BACKGROUND ON NUCLEAR GRAPHITE

Nuclear graphite is any grade of graphite, usually synthetic graphite, manufactured for use as a moderator or reflector within a nuclear reactor. Graphite has been an important material for the construction of nuclear reactors for 80 years due to its performance as a moderator, having a high density of light atomic nuclei with low neutron absorption cross-sections. It also has the ability to withstand a range of harsh environmental conditions.

Graphite was used in 1942 to build Chicago Pile 1, the first nuclear reactor to achieve a sustained chain reaction. Graphite was chosen over heavy water due to the large manufacturing capacity that existed in the US, primarily to supply electrodes. Large quantities of high-purity graphite could be procured from several manufacturers, while heavy water required the costly and time-consuming creation and operation of new production facilities. Although neutron damage was identified as a potential problem early on, with experience it was found that graphite structures could be designed that could perform well even when subjected to relatively high doses of fast neutron irradiation [13]. Heavy water reactor research continued in parallel, notably in Canada.

The first generation of large, high-power reactors was designed to produce plutonium using natural uranium fuel, prioritizing neutron economy and speed of construction over thermal efficiency or longevity. These reactors used coarse grain needle coke graphite with strongly orthotropic properties. Similar material was found to be acceptable for use in early power reactors: typical grades included PGA (UK), GR-280 (USSR), and H-327 (US). As nuclear power technology developed, the focus shifted to reactors that could operate for decades at higher temperatures and relatively high power densities. This led to the development of near-isotropic medium-grain graphites including Gilsocarbon (UK), H-451 (US), and ATR-2E (Germany). More recently, iso-molded fine-grain graphites have been developed, achieving near-isotropic behavior and consistent mechanical properties. [[

]]P,E Currently available materials include iso-molded and medium-grain grades, but the major historic medium- and coarse-grain nuclear graphite grades are no longer manufactured.

Nuclear grade graphite is manufactured from 'filler' coke mixed with 'binder' pitch. Coke and pitch may both be the by-products of petroleum refining and coal coking processes or may be produced specifically for graphite.

Isotropic graphite is desired for uniform behavior. Isotropic coke is used for manufacture. It is heated, extruded into billets, and then baked at 1,000 °C for several days. To reduce porosity and increase density, the billets are impregnated once or more times with pitch at high temperature and pressure before a final graphitization stage at high temperature. Individual billets are machined into the final required shapes.

Modern graphite manufacture commences with a high molecular weight hydrocarbon, often coal tar pitch or a residue of crude oil distillation, which is first converted to coke by heating in the absence of air. This is a long and complex process, usually taking several weeks to perform. The result of the process is that the carbon atoms order themselves in extensive hexagonal clumps to create a good coke.

The coke is then calcined, crushed, and sieved to get a specific distribution of particle sizes. Next, these particles are bound together using hot pitch, and the mixture is extruded or molded to form rough blocks of the shape eventually desired.

The binder is then coked by baking, and an extensive pore network is created. The baked article is therefore reimpregnated and rebaked until an adequate density is reached. Finally, the material is converted to graphite at up to 3000°C by passing a current through a conducting coke bed surrounding the blocks. At 1600°C the localized regions of order begin to link up and graphitic properties begin to appear. Beyond 2400°C individual crystals grow to some extent, but large scale re orientation does not occur. Most impurities are volatile and so disappear at graphitization temperatures. Further purification may be achieved during or after manufacture of the graphite by using halogen gases.

As the newly made graphite cools the individual crystals contract anisotropically creating large stresses in the overall matrix. As a result, a network of fine cracks (known as Mrozoski cracks) is created. Together with the remaining volatilization pores, the graphite is networked with pores, occupying about 20% of the overall volume, and as the pores are interconnected, they provide access to the external atmosphere. The porous structure plays an important role in the behavior of graphite under irradiation.

See the graphite manufacturing steps in Figure 6. See graphite properties and performance data in Table 3.

2.2. GRAPHITE SELECTION FOR THE XE-100

X-energy reviewed the global market of nuclear graphite to select the type most suitable for the needs of the Xe-100. [[

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For most of the graphite reflector in the Xe-100, X-energy plans to use medium grain material. Medium grain graphite offers good behavior under irradiation, with higher fracture toughness [14] and better resistance to oxidation [15] than typical fine grain grades. Although unirradiated tensile strength is typically lower, acceptable performance can still be achieved. The majority of historic graphite moderated reactors have used medium or coarse grain graphites, including all of those to have irradiated graphite to beyond dimensional change turnaround.

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2.2.1. Expert Review Phenomena Identification and Ranking

Nuclear graphite users have learned about graphite material performance in the 80 years of use. The NRC published "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRT)" [3], which identifies and ranks the phenomena for the use of nuclear-grade graphite for the moderator and structural components of a next generation nuclear plant (NGNP), considering both routine (normal operation) and postulated accident conditions for the NGNP. This report published in 2008 states:

> *"Much has been learned about the behavior of graphite in reactor environments in the 60-plus years since the first graphite reactors went into service. The extensive list of references in the Bibliography is plainly testament to this fact. Our current knowledge base is well developed. Although data are lacking for the specific grades being considered for Generation IV (Gen IV) concepts, such as the NGNP, it is fully expected that the behavior of these graphites will conform to the recognized trends for near isotropic nuclear graphite. Thus, much of the data needed is confirmatory in nature. Theories that can explain graphite behavior have been postulated and, in many cases, shown to represent experimental data well. However, these theories need to be tested against data for the new graphites and extended to higher neutron doses and temperatures pertinent to the new Gen IV reactor concepts. It is anticipated that current and planned future graphite irradiation experiments will provide the data needed to validate many of the currently accepted models, as well as providing the needed data for design confirmation."*

The NRC held a public workshop during March of 2009 in Rockville, MD, on nuclear graphite research, related to HTGRs. Oak Ridge National Laboratory organized this workshop and convened an international panel of recognized nuclear graphite specialists and regulators from the U.S.A., the U.K., Japan, and South Africa. Prior to the workshop, the panel members were provided with a series of documents describing the DOE graphite research plan and the NRC DOE Graphite Phenomena Identification and Ranking Table

(PIRT) report, along with other supporting documents for their review. The panel recommended several areas of research to address technical gaps needed for safety assessment and licensing review of HTGR designs. [20]

Idaho National Laboratory (INL) then issued a "Graphite Technology Development Plan" [21], based on NUREG/CR-6944, providing research areas for graphite development. This report indicates that "While the general characteristics necessary for producing nuclear grade graphite are understood, historical "nuclear" grades no longer exist. New grades must be fabricated, characterized, and irradiated to demonstrate that current grades of graphite exhibit acceptable non-irradiated and irradiated properties so that the thermomechanical design of the structural graphite in the Next Generation Nuclear Plant (NGNP) can be validated." The INL report anticipated the graphite development research areas to address areas of low knowledge and high importance. Those research areas are shown in Table 6 with their relationship to the Xe-100 structural graphite material qualification plan.

The NRC issued the "HGTR NRC Research Plan," ML110310182 [22] on March 3rd, 2011. This plan includes a plan for graphite performance. The plan references the PIRT [3] and states:

> *"The PIRT identified seven major safety significant functional requirements for GCC. There are: (1) ability to maintain passive heat transfer; (2) maintain ability to control reactivity; (3) thermal protection of adjacent components; (4) shielding of adjacent components; (5) maintain coolant flow path; (6) prevent excessive mechanical load on the fuel; and (7) minimize activity in the coolant. A number of phenomena were identified which could potentially affect these requirements. The most significant phenomena included: (a) irradiation effect on graphite properties; (b) consistency of graphite quality and performance over the service life; and (c) potential generation of graphite dust, which could impact source term."*

Based on the graphite PIRT [3] and the data gap analysis from the 2009 public workshop, the NRC listed the following phenomena having the highest importance with low knowledge.

These data needs involve:

1. Irradiation-induced creep (irradiation-induced dimensional change under stress)
2. Irradiation-induced change in CTE, including the effects of creep strain
3. Irradiation-induced change in thermal conductivity
4. Irradiation-Induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress).

The NRC stated that all these phenomena are being addressed by the data generated for NGNP graphites by either DOE or other worldwide research (or both), along with the applicant's data. This will be used for NRC staff evaluation of the HTGR designs. The X-energy PIRT ("Xe-100 Phenomena Identification and Ranking Tables") [23] builds off of the NGNP PIRT [3].

Modelling of the graphite performance is outside the scope of this LTR and will be covered in future LTRs.

2.3. QUALIFICATION PLAN OVERVIEW

As noted above, the Xe-100 graphite reflector will undergo a neutron fluence which will modify its physical characteristics and its behavior. The Xe-100 structural graphite qualification criteria include the verification of the unirradiated mechanical and thermal properties of the graphite to be used (Section 3), and verification that the graphite will maintain its structural integrity within a neutron flux (Section 4).

3. QUALIFICATION OF UNIRRADIATED GRAPHITE MECHANICAL AND THERMAL PROPERTIES

The ASME Division 5 [12] code defines a method to qualify the use of high-temperature graphite in a nuclear power reactor. This method is applied here on the Xe-100 structural graphite. The code provides details on the qualification of commercially available (unirradiated) graphite, as well as on irradiated material properties. The process described provides a method to show the selected grade meets a component's design criteria.

This section explains how X-energy uses the code process to qualify the properties of unirradiated structural graphite. Mechanical and thermal property information from this qualification process is used to assess the structural graphite integrity using the probabilistic approach described in the Division 5 code [12], Article HHA-II-3000, and its sub-articles.

Sections 3.1, 3.2, and 3.3 cover the use of the Division 5 code [12] and corresponding ASTM Standards requirements on the measurement of graphite properties, the sampling requirements, and the determination of graphite purity. The measurements of mechanical and thermal properties also capture the effect of pore size on graphite material properties, so no separate pore size measurements are included in this qualification plan.

The code provides standards and guidance on the Designer and Supplier Certification:

- G certificate (HAB-3300):

  Designer of Graphite Core Components must have a G-Certificate (HAB-3320). The Designer is responsible for designing the components and assemblies, preparing the Design Specification, Design Report, Design Drawings, and Construction Specification

- GC certificate (HAB 3400):

  Holder is responsible for construction and assembly of components, prepares the Construction Report, tests components (HAB-3420)

- G Quality System certificate (HAB-3800):

  Holder is a Graphite Material Organization (GMO), and the certification is provided by ASME or a GC-certificate holder, based on the GMO's quality system program. The holder is responsible for material manufacture, graphite core component machining and installation (HAB-3830)

All machining (HHA-4220), examination (HHA-4230, HHA-5220, HHA-5400, HHA-5500), testing (HHA-4240), packaging (HHA-4250) and installation (HHA-5300) must be done by a responsible party (G-certificate, GC-certificate or G Quality System-Certificate holder). X-energy has begun working with the ASME Inspection Groups and our vendors to obtain G and GC certificates. As of the writing of this report, the next group to apply for either certificate will be the first to ever apply. The applicable ASME code and ASTM standards for structural graphite material qualification are listed in Table 7.

3.1. MECHANICAL AND THERMAL PROPERTIES

The point of mechanical and thermal properties testing is to develop the representative data of each graphite property and to evaluate the intra- and extra-billet changes of graphite from the 3 graphitization charges (bakings). This data is also used to calculate the probability of failure.

The plan for structural graphite meets HHA-II-2000, as the parameters used to meet the structural graphite RSF are included in the MDS. Tests are conducted as described in the Division 5 code [12] to collect the information, showing the number of specimens, the orientation, and the temperatures used in each test. The number of tests per dataset is determined by the sample cutting plan.

Table 8 lists the standards followed in the data generation. The resulting mechanical and thermal properties are shown to be compliant with the code and ASTM test standards listed in Table 7.

The Division 5 Code [12] states strength test requirements for both room and high temperatures. Analysis of mechanical properties (i.e., strength) will follow the guidance in the Division 5 code, article HHA-II-3100, Material Reliability Curve Parameters [12]. As graphite strength increases with temperature, the use of room temperature strength data is conservative, plus ASTM standards do not yet cover high-temperature strength testing.

The INL Plan [21] test details include:

- 3 room temperature strength measurement types
- Room temperature bulk density
- Room temperature Young's and shear modulus
- CTE and thermal conductivity tests use the same specimen for all temperature measurements, covering the desired range and using the required 200 deg C intervals
- Specimens per dataset are set by the sample cutting plan

Based on the Division 5 code, article HHA-III-4100, As-Manufactured Graphite [12], an appropriate sampling plan is used to calculate the number of billets, specimens, and sampling locations to use in the INL testing.

There are three alternative approaches to design:

- Simplified assessment (HHA-3220): meet reliability targets based on stress limits from the material reliability curve only, using peak stress limits.
- Full assessment (HHA-3230): meet reliability targets based on stress limits from the material reliability curve and the distribution of stresses in the component with combined stress limits.
- Design by test (HHA-3240): meet reliability targets based on experimental proof of component performance within margins derived from the material reliability curve.

Components must be classified based on importance to safety and environmental conditions (HHA-3111); as well as by the service level of the component's expected loads- these classifications are used to determine stress limits used in design analyses.

- SRC-1 – Important to safety and may be subject to environmental degradation
- SRC-2 – Not important to safety and are subject to environmental degradation
- SRC-3 – Not important to safety and not subject to environmental degradation

3.2. PROPERTY VARIATION

Property variation per billet and by the manufacturer is assessed as described below, as graphite varies more than metallic components. Division 5 code, HHA-III-5000, Use of Historical Data [24], allows the use of applicable data versus fresh testing.

3.2.1. Local Variation

Local variations (Intra-billet) in graphite properties are introduced by the method of manufacturing and are more related to the specific material features versus quality control. The billet cutting pattern for component development is determined based on these properties.

The manufacturer provides local variation data. This data would be considered historic data which is evaluated to meet Code Article HHA-III-5000. This information will be compared with INL test results. Strength characteristically varies across the billet with the middle being weaker than the ends.

3.2.2. Lot-to-Lot Variation

Variation over months/years of production time can be evaluated by comparing each single billet's inspection results to the manufacturer's historical record. Lot-to-lot variation data would be considered historic data which would be evaluated to meet Code Article HHA-III-5000.

3.3. PURITY

X-energy plans to satisfy the graphite purity requirement as specified in the ASTM D7219-08 standard and will evaluate the expected impurities from manufacturers specifications against experimental data to confirm that they do not impact the safety-related function of the material. Impurities considered include ash content and boron concentration. Other impurities are only considered if they affect the ability of the graphite to perform its safety-related function, but may also be considered with respect to their relation to final graphite disposal cost. ASTM D7219-08 requires that high-purity nuclear graphite for a gas-cooled reactor should have ash content equal to or less than 300 ppm, and an equivalent boron content (EBC) equal to or less than 2 ppm.

4. QUALIFICATION OF GRAPHITE UNDER IRRADIATION

4.1. IRRADIATION ENVIRONMENT

The Xe-100 typical operating temperatures and fast neutron fluxes are shown in Figure 9. [[

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4.2. ASME IRRADIATION QUALIFICATION REQUIREMENTS FOR STRUCTURAL GRAPHITE

"Irradiated graphite" is graphite exposed to fluences greater than 0.25 displacement per atom (dpa) per Division 5 code, Article HHA-3142.1 [12]. HHA-2200, Material Properties for Design, and HA-III-3000, Properties to be Determined [12], indicate that the following properties should be measured on irradiated graphite.

4.2.1. Basic Properties

- Irradiation-induced dimensional change
- Irradiation-induced change in the coefficient of thermal expansion, and temperature relation
- Irradiation-induced change in strength
- Irradiation-induced change in elastic modulus
- Irradiation-induced change in thermal conductivity, and temperature relation

4.2.2. Irradiation Creep Properties

- Irradiation-induced creep coefficient
- The effect of creep strain on the coefficient of thermal expansion and elastic modulus.

HHA-III-3000, Irradiated Graphite, and HHA-II-4000, Detailed Requirements for Derivation of the Material Data Sheet - Irradiated Material Properties call out additional irradiated measurements of structural graphite:

- Envelope the temperature range of material in operation in the reactor for all service levels
- Use temperature increments of at most 200°C
- A single grain orientation may be used for measurements in near-isotropic graphite grades except for dimensional changes

4.3. IRRADIATION QUALIFICATION PROGRAMS FOR THE XE-100 MATERIAL

Irradiated Material Properties are addressed in Code sections HHA-2220 and HHA-III-3300. The Qualification Envelope states that a maximum of 200°C temperature increments are allowed for damage dose measurements and for any properties that are temperature dependent.

The properties of interest are:

- Dimensional Change
- Strength
- Elastic Modulus
- Creep Coefficient
- Coefficient of Thermal Expansion
- Thermal Conductivity

The material being tested for qualification use must be representative (HHA-III-4200); Historical Data must be shown to be applicable if used (HHA-III 5000); Reporting Requirements for Material Data Sheets is addressed (HHA-II-4000), and Material Specification and use of ASTM standards are covered (HHA-III-1000 and HHA-I 1110).

4.3.1. Basic Properties

The Irradiation Fluence Limits are addressed (HHA-3142.1):

- < 0.001 dpa the effects of neutron irradiation are negligible and may be ignored
- >0.001 dpa the effect of neutron irradiation on thermal conductivity shall be taken into account
- 0.25 dpa all effects of neutron irradiation (described in HHA-2200) shall be considered, and a viscoelastic analysis applied

Stored (Wigner) Energy (HHA-3142.2) is addressed for irradiated graphite to the significant fluence limit (> 0.25 dpa) and at a temperature < 200°C. If irradiation > 0.25 dpa and temperature < 200°C, then, the effect of the stored (Wigner) energy buildup shall be accounted for during thermal transient evaluation. NOTE: The Xe-100 does not feature graphite irradiated at <200°C.

Graphite Core Components are to be designed such that (HHA-3212):

- All mechanical loads that occur shall be transferred to the adjacent load-carrying or supporting structures within allowable limits.
- Displacement or deformation of adjacent Graphite Core Component in opposing directions do not cause constraint and thus hinder expansion or shrinkage due to temperature or irradiation.
- Changes in the shape of a Graphite Core Component due to irradiation do not adversely affect the stability or functionality of the core assembly.
- The compensation of the differential strains inside the Graphite Core Assembly and in the surrounding structures does not lead to stresses exceeding the HHA-3211 limits in the Graphite Core Components.
- The movement of blocks and the accumulation of gaps inside the Graphite Core Assembly are within allowable limits.
- Changes in the shape of the Graphite Core Component due to radiation and temperature effects are within allowable limits and do not affect the function and stability of the core assembly.
- Design channels for the gas flow through Graphite Core Components are such that the shielding effect of the graphite internals is within allowable limits.

- Grooves, keyways, dowel holes, and other recesses in the blocks are to be blended. The minimum fillet radius shall be five times the maximum grain size as documented in the Mandatory Appendix HHA-II Material Data Sheet.
- The external edges of Graphite Core Components shall be chamfered.

4.3.2. Irradiation Creep Properties

AGC/HDG testing evaluates creep properties for the 3 graphite materials. The measurements will be used to calibrate the creep models. AGC/HDG testing will address the following:

- Test material: The 3 graphite materials in the WG direction
- Stressed and un-stressed specimens will be irradiated at the same time and conditions in the irradiation capsule
- Two target test irradiation temperatures will be used with the maximum 200°C gap allowed by ASME.
- Pre-turnaround & post-turnaround Irradiation fluences will be specified:
- A compressive applied stress will be used
- Post-irradiation examination of creep strain & the effect of creep strain on CTE and Young's modulus

The AGC/HDG data measure creep at 600°C and 800°C and includes [[ ]]^{P,E} data. There is some evidence for enhanced creep rates at lower and higher temperatures (i.e., top and bottom reflectors). Creep data for the [[ ]]^{P,E}.

There is expected to be little variation in the measured secondary creep rate with temperature in the Xe-100 side reflector expected temperature range, however this is an area of some uncertainty that will be addressed through appropriate sensitivity studies.

In his 1981 review of creep data and models, Price [25] considered two correlations:

- The 'UKAEA' line, by Kelly and Brocklehurst [26], which finds no temperature dependence of the secondary coefficient rate below 650°C and an increase above this temperature
- The 'ORNL' line, proposed by Kennedy based on the hypothesis that secondary creep rates follow crystallite strain rates, with a minimum around 400°C and higher rates above and below this temperature

While Price concluded that the UKAEA line was most appropriate for design purposes, uncertainty and variability in the data were such that either interpretation could be supported.

A plot of secondary creep data is shown in Figure 8. Subsequent analysis by Davies and Bradford [32], included in Figure 8, supports the 'ORNL' line while also finding minimal variation in creep rate in the range from around 300-650°C. On this basis, we consider that the lower temperature data from AGC/HDG is likely to be applicable to graphite in the Xe-100 in the 300-600°C temperature range, however there is theoretical support for the conservative use of a lower secondary creep coefficient in the 400-500°C temperature range (a low assumed creep rate is conservative for graphite component design as it leads to less relief of internal stress).

5. DEMONSTRATION OF ENVIRONMENTAL COMPATIBILITY

5.1. OXIDATION OF GRAPHITE

X-energy will determine the potential loss of graphite strength per Article HHA-3141 via 2 paths: weight loss and strength reduction. See the graphite strength versus weight loss depicted in Figure 10 [[ ]]^{P,E}.

The evaluation methodology is to determine the rate of oxidation per ASTM D7542, then obtain gas effusivity, and combine this data within the example INL model. Next, the weight loss value will be determined experimentally. Then, X-energy will determine strength reduction based on the relationship between strength and weight loss already established in the Division 5 code, Article HHA-3141-1 [12].

6. CONCLUSIONS AND LIMITATIONS

6.1. CONCLUSIONS

X-energy plans to use graphite grades [[ ]]P,E in the Xe-100 safety-related structural graphite component design. This report presents the qualification plan for the use of those materials. The qualification plan for unirradiated and irradiated graphite conforms with the Division 5 code [12]. Departures from ASME Code requirements, if any, will be identified and justified as part of the Construction Permit and Operating License Applications.

There is an overlap in the planned timescales for the irradiation program and graphite analysis and licensing. X-energy plans to perform these analyses using available grade-specific data and to use historic data for the proxy grade(s) where data are lacking. X-energy will subsequently update the analysis to confirm acceptability once full data is available. This scenario was foreseen by the NRC as documented in HTGR NRC Research Plan [22]. The NRC noted that per the DOE schedule, some of the data for NGNP graphites may not become available by when a design certification application is expected to be submitted to the NRC. Therefore, NRC staff might have to use other data for a similar graphite class, along with the applicant's technical basis for designing the HTGR GCC, to support staff technical safety evaluations. In the absence of properties data as a function of complete cumulative damage dose of the reactor life, the staff could decide to give a conditional operational license for limited initial period of operation ensuring adequate public safety, pending complete (namely, high irradiation dose) data from irradiation and confirmation of existing data by actual HTGR initial operation.

The ASME III.5 code sets an arbitrary limit on feature size relative to material grain size. HHA-3212(h) indicates that "grooves, keyways, dowel holes, and other recesses in the blocks are to be blended. The minimum fillet radius shall be five times the maximum grain size". X-Energy plans to demonstrate the acceptability of component features through analysis.

X-energy plans to use historic data as part of the analysis, as there is extensive data generated over decades. The data is of generally high quality but was performed to different standards in the US, Europe, and Japan.

X-energy plans to use grade-specific 600°C creep data from AGC for calculations in the 250-750°C range in combination with historic creep data from wider temperature and dose ranges to cover the entire Xe-100 graphite reflector conditions.

X-energy is building a Helium test facility for the testing of various components in the high-temperature helium environment. Additionally, national lab and university system partners with helium test facilities may be used.

Aside from irradiation, other planned experiments are:

- Leak Flow Validation
- Manufacturing and Machinability Studies
- Friction Factor and Dust Creation (As needed)
- Impact Testing
- Wear testing with rod channel liners (as needed)

X-energy will have developed the ability to perform additional tests as they become apparent based on ongoing results.

The simulation and modeling effort is, to some extent, required by the Division 5 code [12], and in other cases allows for a more efficient design. Some of the analyses planned or ongoing are:

- Structural
  - Cold-static
  - Hot-static
  - Seismic
  - Impact
- Oxidation
  - Chronic oxidation
  - Acute Oxidation
- Dust Creation and Migration
- CFD
- Leak-by flow
- Radiation Exposure
- Thermal expansion
- PoF

The Xe-100 Simulation and Modeling Effort expected path is as follows.

In Step 1, perform neutronics analysis. This provides core power profiles, ex-core heating results, and fluence.

In Step 2, generate the detailed computational flow dynamics (CFD). Using neutronics data from Step 1, this provides detailed component temperatures from the low leak flow core structure model.

In Step 3, develop the initial finite element analysis (FEA). Using temperature data from Step 2, perform FEA using static and thermal loads to determine core structure movement. This calculation provides expanded geometry for the next CFD model.

In Step 4, calculate the detailed CFD leak flow analysis. Using thermal expansion data from Step 3 provides detailed component temperatures for the low leak flow core structure model. Also, determine reflector temperatures with medium and high leak flows.

In Step 5, perform the independent reflector block FEA analysis. Using leak flow resistance data from Step 4, perform detailed graphite block analysis for all blocks using temperatures from the CFD analysis and the fluence from the neutronics.

7. CONCLUSIONS AND RECOMMENDATIONS

X-energy is requesting NRC review and approval of the graphite qualification methodology described in this document for use by licensing applicants under 10 CFR 50 or 10 CFR 52.

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Table 1: Xe-100 Design Data

Parameter	Value/Description
Reactor Type	HTGR
Core Configuration	Pebble fuel in a right circular cylinder configuration
Physical Dimensions (height, diameter)	9.2 m, 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

Table 2: Reflector Materials and Masses

Component	Material	Volume (m^3)	Mass (kg)
Reflector Top	[[ ]]^{P,E}	[[ ]]^{P,E}	[[ ]]^{P,E}
Reflector Bottom	[[ ]]^{P,E}	[[ ]]^{P,E}	[[ ]]^{P,E}
Funnel and Outlet Plenum	[[ ]]^{P,E}	[[ ]]^{P,E}	[[ ]]^{P,E}
Side Reflector - Outer	[[ ]]^{P,E}	[[ ]]^{P,E}	[[ ]]^{P,E}
Side Reflector - Inner	[[ ]]^{P,E}	[[ ]]^{P,E}	[[ ]]^{P,E}
Rod Sleeve	[[ ]]^{P,E}	[[ ]]^{P,E}	[[ ]]^{P,E}
Riser Sleeve	[[ ]]^{P,E}	[[ ]]^{P,E}	[[ ]]^{P,E}
	Total	[[ ]]^{P,E}	[[ ]]^{P,E}

Table 3: Typical Unirradiated Nuclear Graphite Properties and Performance Data

Properties	Requirement and characteristics	Attributes
Bulk density	A density greater than 1.7g/cm^3 is more effective for neutron moderation/reflection per unit volume. Higher density equals higher strength for similar grades.	Moderation value and structural integrity
Strength and modulus	Reasonable strength is required for structural components. A typical requirement for extruded and molded graphite, is ≥ 15 MPa (tensile strength with the grain (WG)); Isomolded graphite, is ≥ 22 Mpa (tensile strength in WG). Dynamic Elastic modulus (elasticity under dynamic loading), 8-15 GPa (WG) for all the grades. The strength must exceed allowable operating component stresses. Higher strength equals higher modulus but lower fracture toughness, and must be taken into account. Strength and modulus are temperature dependent, increasing with temperature.	Structural value
Coefficient of Thermal Expansion (CTE)	Typical nuclear graphite CTE is 3.5 - 5.5x10^{-6}/°C. Anisotropy, defined as the CTE_{AG}/CTE_{WG} ratio, is typically a value between 1.00 to 1.10 for isotropic graphite and 1.10 to 1.15 for near-isotropic graphite. CTE increases with temperature.	Structural strength
Thermal conductivity	Typical nuclear graphite requires unirradiated thermal conductivity (W/m·K, AG at 25°C): ≥ 100 for extruded and molded graphite, ≥ 90 for isomolded graphite. Unirradiated thermal conductivity decreases with temperature.	Heat transport
Purity	ASTM D7219-08 indicates that expected impurities (ash content and boron) do not impact the safety-related function of the material. Boron should be ≤ 2ppm for high-purity graphite and ≤ 10ppm for low-purity graphite. Ash should be ≤ 300ppm for low neutron irradiation dose application, to minimize impurity acting as an oxidation catalyst.	Neutron efficiency and oxidation rate

Note: largely based on ASTM D7219. Graphite specifications for manufacturing will typically set more restrictive requirements.

Table 4: Proposed Dose/Temperature Matrix for Irradiation Tests

Target Temperature (°C)	Target Dose (dpa) (number of HFIR cycles)					
Low (<250)	1 (1)	2.5 (2)	5 (4)	10 (7)	17 (12)	
300	5 (4)	10 (7)	15 (10)	20 (14)	25 (17)	30 (21)
450	5 (4)	10 (7)	15 (10)	20 (13)	25 (16)	30 (19)
600	4 (3)	8 (5)	12 (8)	16 (10)		
5 low-temperature capsules per material and 48 multi-use capsules per material (106 capsules total)						

Notes:

[[ ]]^{P,E}

Low-temperature (<250°C) capsules are primarily to determine stored energy. Measurements will consist of dimensional change, thermal conductivity, and heat capacity

For the other temperatures, measurements will consist of dimensional change, coefficient of thermal expansion, thermal conductivity, dynamic Young's modulus by ultrasonic velocity, tensile strength by disc compression (ASTM D8289-19), and compressive strength

Table 5: Properties for Un-Irradiated Structural Graphite Grades

Property	[[ ]]P,E	[[ ]]P,E	[[-18]]P,E
Density, g/cm^3	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Dynamic Young's Modulus, GPa	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Elastic Poisson's Ratio ($\nu_{elastic}$)[a]	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Classification, coke type	[[ ]]P,E	[[ ]]P,E	[[-]]P,E
Grain size (max), µm	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Flexural strength, MPa (WG/AG)	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Tensile strength, MPa (WG/AG)	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Compressive strength, MPa	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Coefficient of thermal expansion, $x10^{-6}/°C$	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Thermal conductivity, W·m/K, room temperature	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E
Fracture toughness, K_{IC}, $MPa \cdot m^{1/2}$	[[ ]]P,E	[[ ]]P,E	[[ ]]P,E

Data is from Reference [27] unless noted

a [28]

b [31]

c [14]

d mostly from [29]

e [30]

[[ ]]P,E

Note that unless otherwise indicated, values are as provided by manufacturers and are indicative only.

Table 6: Proposed INL Research Areas and Structural Graphite Material Qualification Plan Elements

Topic	Impact	Relation to Xe-100 Material Qualification Plan
The structural integrity of graphite	Retention of long-term structural stability and mechanical strength under specified loads. Specified by ASME requirements.	Addressed by ASME Section III, Division 5 [12] requirements. See Sections 3.1, 3.2, and 3.3 of this report.
Thermal response of graphite - normal operation	Changes in thermal properties at peak dose and temperatures.	Changes in graphite thermal properties under irradiation are known (see Section 4)
Thermal response of graphite off-normal operation	Verification that changes to thermal material properties is sufficiently small to guarantee the passively safe response of the reactor.	Qualification data generated under the structural graphite material qualification plan will be used in subsequent modeling to predict thermal response.
Changes to by-pass flow	Potential coolant flow issues due to shrinkage and swelling of graphite components.	Qualification data generated under the structural graphite material qualification plan will be used in subsequent modeling of the bypass flow.
Chemical and mechanical core stability	Oxidation and subsequent structural stability of oxidized graphite. For both acute (postulated event) and chronic (normal operation) conditions.	Qualification data generated under the structural graphite material qualification plan will be used in subsequent modeling of the oxidation behavior.

Table 7: Applicable ASME Code and ASTM Standard for Structural Graphite Material Qualification

Property to be Qualified	ASME Section III, Division 5 Code [12], and related ASTM Standard
Mechanical and Thermal Properties	HHA-II-2000, Material Data Sheet, Form MDS-1 HHA-III-3100, As-manufactured graphite HHA-III-4000, Requirement for representative data ASTM C781 Standard Practice for Testing Graphite Materials for Gas-Cooled Nuclear Reactor Components ASTM D7775 Standard Guide for Measurements on Small Graphite Specimens
Property Variation	HHA-III-5000, Use of historical data
Purity	ASTM D7219-19, Standard specification for isotropic and near-isotropic nuclear graphite materials ASTM C1233-15 Standard practice for determining equivalent Boron contents for nuclear materials
Oxidation	ASTM D7542 Standard Test Method for Air Oxidation of Carbon and Graphite in the Kinetic Regime HHA III 3200 Oxidized Graphite

Table 8: ASTM International Standard and Specimen Dimension Requirements

Property	ASTM standard	Specimen dimensional requirement
Bulk density	C559	Ø 10x60 mm, at room temperature
Flexural strength	C651	2.5x5x60mm
Tensile strength	C749	Ø 12.95x120.65 mm, center section Ø 6.35 mm
Compressive strength	C695	Ø 9.5x19 mm
Dynamic YM and Shear Moduli	C747	2x10x50mm
Coefficient of Thermal Conductivity	E228	Ø 10x60mm
Thermal Conductivity	E1461	Ø 10x2mm

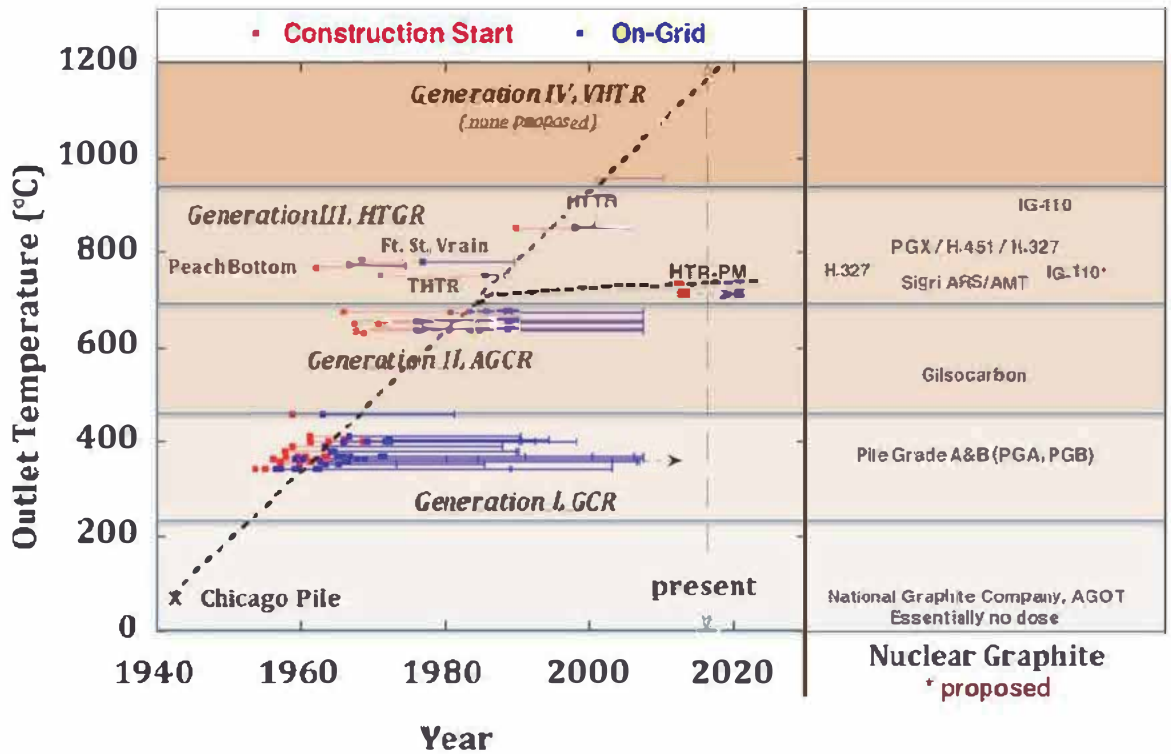

Figure 1: Graphite Use History in Nuclear Reactors

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Figure 2: Graphite Reflector Cross Section

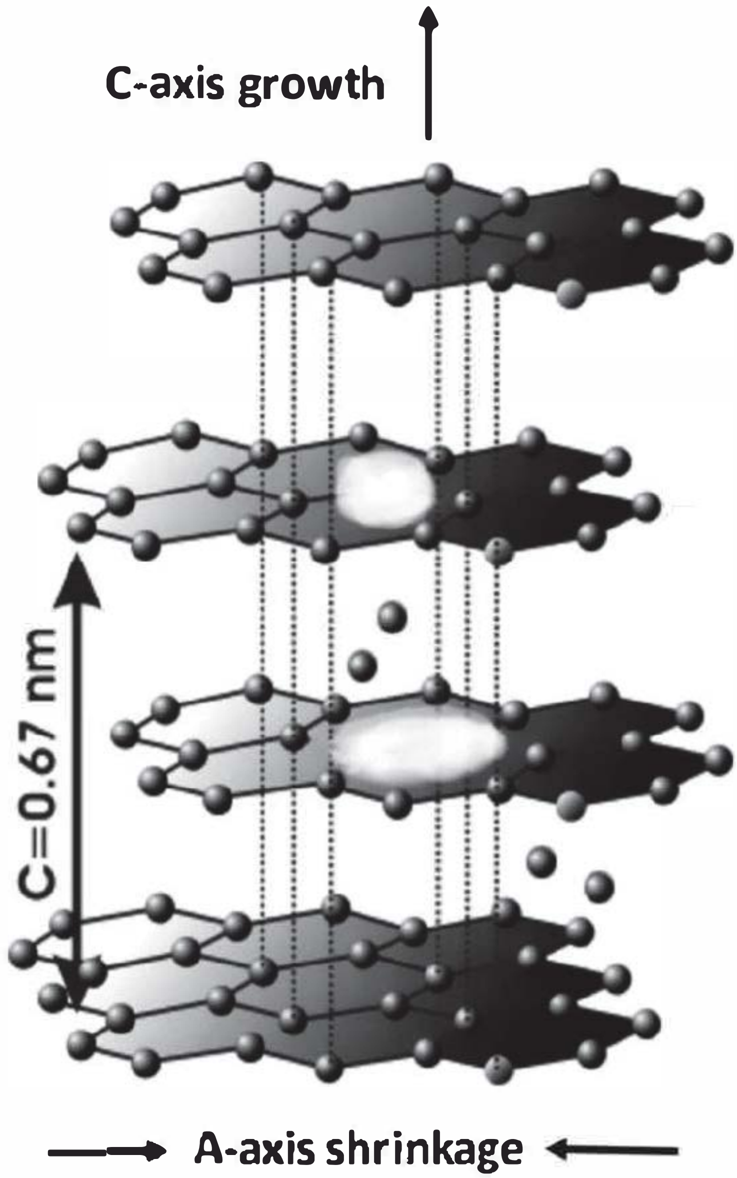

Figure 3: Graphite Dimensional Change with Irradiation

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Figure 4: Illustration of the Relationship of Fluence on Graphite

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Figure 5: Illustration of the Relationship of Fluence on Graphite Turnaround

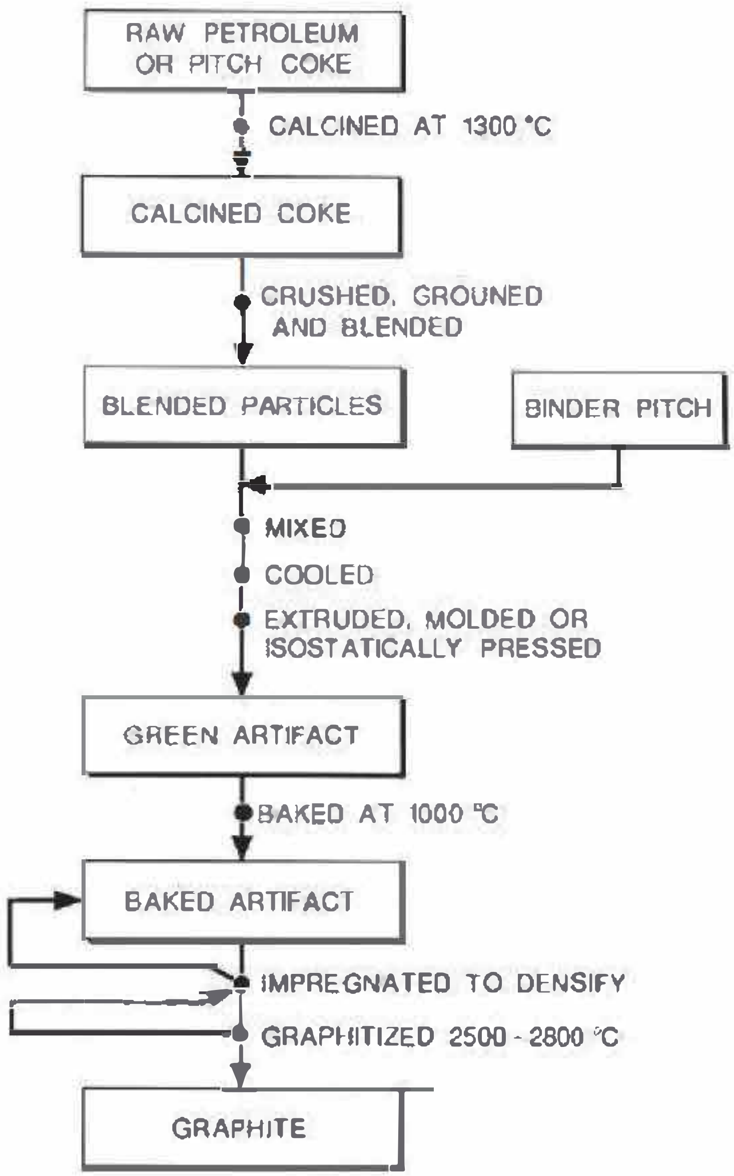

Figure 6: Steps in the Manufacture of Synthetic Graphite

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Figure 7: Summary of Dose/Temperature Test Matrix for Grapite

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Figure 8: Xe-100 Graphite Environmental Conditions – Approximate Dose/Temperature Envelope

Reference [34]
Note: [[ ]]P,E

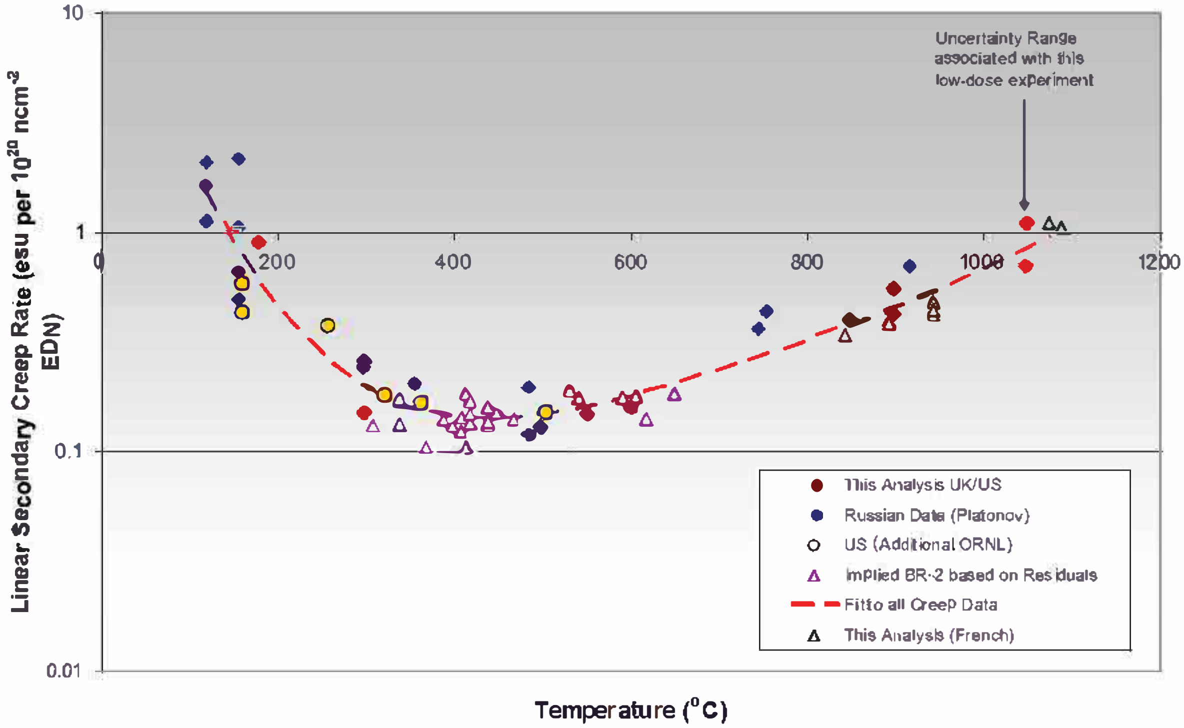

Figure 9: Calculated Secondary Creep Coefficients Plotted Against Irradiated Temperature

Reference [27]

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Figure 10: Graphite Strength versus Weight Loss