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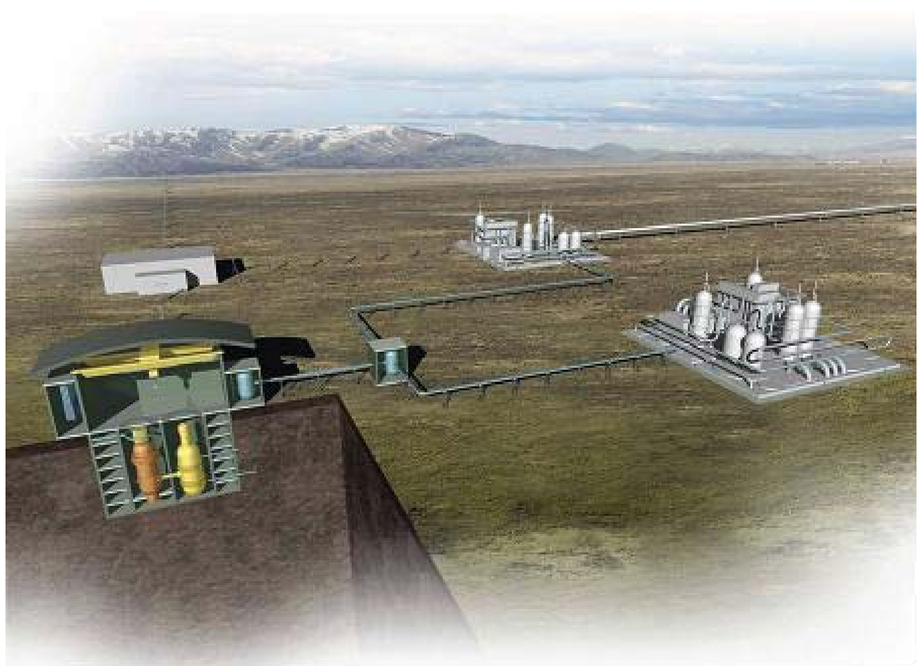
Graphite Technology Development Plan

W. Windes
T. Burchell
R. Bratton

September 2007



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Idaho National Laboratory
Next Generation Nuclear Plant Project
Idaho Falls, Idaho 83415

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Next Generation Nuclear Plant Project

Graphite Technology Development Plan

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Authored by:

W. Windes, NGNP Materials R&D Lead

9/25/07
Date

Reviewed by:

for M. Metcalfe

M. Metcalfe, Nexia Solutions Commercial

9/25/07
Date

Approved by:

D. Petti, NGNP R&D Manager

9/25/07
Date

SUMMARY

The Next Generation Nuclear Plant (NGNP) will be a helium-cooled High Temperature Gas Reactor (HTGR) with a large graphite core. Graphite physically contains the fuel and comprises the majority of the core volume. Graphite has been used effectively as a structural and moderator material in both research and commercial high-temperature gas-cooled reactors. This development has resulted in graphite being established as a viable structural material for HTGRs. 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 upon which the thermomechanical design of the structural graphite in NGNP is based. This *Technology Development Plan* establishes the research and development (R&D) activities and associated rationale necessary to qualify nuclear grade graphite for use within the NGNP reactor.

Background information from past graphite reactor experience, other relevant graphite grades, and the state of graphite technology developed for past gas reactors is presented to provide a perspective on what has been achieved previously in this area of research. The technology required to qualify the graphite for use in NGNP is being developed based on the historical graphite fabrication and performance database, the anticipated NGNP graphite design service conditions, and gaps in the fabrication and performance database.

The resultant data needs are outlined and justified from the perspective of reactor design, reactor performance, or the reactor safety case. The approach allows direct comparison between data needs and the resulting technology development activities. Because there are many variables (i.e., reactor designs, multiple graphite types, a range of operating temperatures and fluence, etc.) that can significantly affect the development of graphite technology for the NGNP, a "baseline" reactor design was chosen to simplify the identification of needed data. The prismatic HTGR design with an outlet temperature of 950°C was chosen as the baseline technology. In this case, the expected doses and operation service lifetimes are expected to be fairly moderate. Technology development needs to satisfy requirements beyond this baseline (i.e., much higher doses expected at graphite reflector surfaces facing the fuel pebbles in the pebble-bed core) are presented separately to provide a more complete understanding of the important differences in the technical requirements for prismatic and pebble-bed HTGRs.

The irradiation program proposed for the prismatic HTGR design consists of eight irradiations that span the proposed temperature-dose envelope for a prismatic NGNP. These irradiations will contain specimens of sufficient size, number, and type to support statistical assessments necessary to capture the inherent variability in graphite; to support traditional American Society for Testing and Materials (ASTM) requirements for sample analysis; and to fully characterize the physical, thermal, and mechanical properties of the irradiated graphite. The current status of the first irradiation capsule, Advanced Graphite

Capsule (AGC)-1, is also presented to provide the reader with a sense of the size and complexity of the irradiation capsules that will be used in the program.

The plan discusses in detail the specific material characterization techniques that will be used to characterize the graphite microstructure and establish the key material properties for both non-irradiated and irradiated specimens that will be used to support American Society of Mechanical Engineers (ASME) codification of graphite. Factors that can significantly affect the R&D program, such as graphite acquisition, test standard development, sample preparation (e.g., grain sizes, sample sizes, etc.), are discussed within each characterization section. In addition, the role of the modeling activities from the engineering-scale to the micro- or meso-scale to the nanoscale is discussed in the context of this qualification program, and the inter-relationships between the experimental and modeling activities are presented to establish a complete picture of the technology development required for NGNP graphite qualification.

Beyond the near-term NGNP graphite qualification program presented here, a more complete evaluation of the processing route and raw material constituent's influence on graphite behavior is required for full commercialization of the HTGR graphite technology in the long term. In addition, appropriate graphite recycling and disposal options must be considered to reduce the volume and costs of anticipated waste disposal. Recycle is considered as a long-term strategy and would only be pursued by vendors when large numbers of HTGRs are developed and a "nuclear graphite economy" is established. The magnitude of the R&D program necessary to establish a standard nuclear grade graphite, whether from a new coke source and/or from recycled material for use within any HTGR design, cannot be firmly estimated today given the current limited knowledge of the linkage between graphite fabrication, material properties, and in-reactor performance. It is anticipated that the work proposed to qualify graphite for the initial NGNP cores will provide the strong technical basis needed to establish a long-term graphite development and qualification program that meets these more ambitious commercialization goals.

Finally, the costs of the baseline NGNP graphite technology development program are presented. The costs are enumerated for experimental, modeling, and pebble-bed design activities. The additional long-term considerations of recycling and coke source qualification are not included in the final cost estimate, but each are discussed since they may have an impact on the other technology development areas.

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ACRONYMS

AGC	Advanced Graphite Capsule
AGR	Advanded Gas Reactor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
AVR	Albeitsgemeinschaft Versuchsreaktor (Germany)
CT	x-ray tomography
CTE	coefficient of thermal expansion
DSC	Differential Scanning Calorimetry
FEM	finite element models
FPY	full-power year
FSVR	Fort St. Vrain Reactor
GT-MHR	Gas Turbine-Modular Helium Reactor
HFIR	High-Flux Isotope Reactor
HTGR	High Temperature Gas Reactor
HTR	High Temperature Reactor (China)
HTTR	High-Temperature Engineering Test Reactor (Japan)
HTV	high temperature vessel
INL	Idaho National Laboratory
ISI	in-service inspection
LLW	low-level waste
MTR	Materials Test Reactor
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PBR	Pebble-bed Reactor
PBMR	Pebble-Bed Modular Reactor
PIE	post-irradiation examination
PIRT	Phenomena Identification and Ranking Table
PMR	Prismatic Modular Reactor
QA	Quality Assurance
R&D	research and development
RBMK	Reactor Bolshoi Moschnosti Kanalynyi (Russia)
SEM	Scanning Electron Microscope
SGL	SGL Group, The Carbon Company
TEM	Transmission Electron Microscope
THTR	Thorium Hochtemperatur Reaktor (Germany)
UT	ultrasonic testing

Graphite Technology Development Plan

1. Introduction

Graphite has been used effectively as a structural and moderator material in both research and commercial high-temperature, gas-cooled nuclear reactors (i.e., Magnox, Advanced Gas Reactor [AGR], Albeitsgemeinschaft Versuchsreaktor [AVR], Reactor Bolshoi Moschnosti Kanalynyi [RBMK], Thorium Hochtemperatur Reaktor [THTR], Fort St. Vrain Reactor [FSVR], etc.). This development has resulted in graphite being established as a viable structural material for High Temperature Gas Reactors (HTGRs). 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.

Beyond structural integrity, the reactor lifetime for specific graphite types cannot be established based on the current state of the art; establishing lifetime is complex because of the influence of fabrication and radiation damage on microstructural changes and associated changes in material properties. Lifetime predictions of graphite components with the service demands and reactor operating mode anticipated for NGNP is a practical but much more complex problem than simply determining whether a graphite type is more stable or less stable in an irradiated environment. Graphite properties, such as strain to failure, dimensional change rate, and irradiation dependence of thermal expansion coefficient, can constrain the reactor design by limiting lifetimes for critical components. For example, irradiation-induced dimensional changes to graphite can be severe enough to require limiting the temperature and flux gradients within graphite components or possibly requiring the need for added design features to physically hold components in position over time.

A complicating factor to establishing a qualified fabrication and performance dataset is the inherent variability in the graphite product. Variability within-billet, intra-billets, and lot-to-lot of the graphite must be accounted for in a statistical manner because of its influence on material properties. This variability must also be characterized to enable credible designs and to support the ongoing development of the probabilistic American Society of Mechanical Engineers (ASME) graphite design methodology. The previous Fort St. Vrain design used deterministic performance models for H-451, which was unacceptably conservative given the understanding of graphite at that time. With our current knowledge, probabilistic performance models can be developed to characterize the new graphite grades for NGNP.

Furthermore, to provide a consistent nuclear grade graphite material for eventual standardization and commercialization of HTGRs, an American Society for Testing and Materials (ASTM) standard specification for isotropic and near-isotropic nuclear graphites (D 7219-05) is being developed along with a standard specification for nuclear graphite suitable for components subjected to low neutron irradiation dose. Additionally, ASME codes and guides for materials selection and qualification, design, fabrication, testing, installation, examination, inspection, and certification will be needed and thus are under development by the international graphite community. Development of these standards will be necessary to approve future grades of nuclear graphite for new HTGRs.

Therefore, the overall objectives to qualify the current NGNP graphite for initial operation are:

1. Establish statistical non-irradiated thermo-mechanical and thermo-physical properties
    a. Characterize lot-to-lot and billet-to-billet variations (for probabilistic baseline data needs)

2. Establish irradiated thermo-mechanical and thermo-physical properties
3. Develop understanding of life-limiting phenomena at high dose and temperature (e.g., irradiation induced creep)
4. Develop appropriate constitutive relations
5. Establish reliable, predictive thermo-mechanical finite element models (FEM)
6. Establish relevant ASTM standards and ASME design rules.

Beyond the initial NGNP graphite objectives, the graphite research and development (R&D) program needs to evaluate processing route and raw material constituents influences on graphite as well as recycling and disposal issues. The current world market share for nuclear graphite is extremely small. While graphite manufacturers are willing to produce nuclear grade graphite, the petroleum industry, which produces the raw starting material – specialty coke, is much less interested. The material specifications for specialty coke are much more exacting than what is needed for electrode production, the majority market share for graphite. Since this material's market share is so small, the coke suppliers have very little financial interest in changing their production process to enable manufacture of these small batches of specialty coke necessary for nuclear graphite production.

As a consequence, there may not be enough specialty coke material needed for sustained production of nuclear graphite for HTGR applications. In the longer term, a full evaluation of the processing route and raw material constituents influence on graphite behavior is required for full commercialization of the HTGR graphite technology. The magnitude of the program necessary to establish a standard nuclear grade graphite for use within any HTGR design is difficult to estimate given the current limited knowledge of the linkage between fabrication, material properties, and performance in reactor.

Finally, the lower power density of current HTGRs and the large inner and outer graphite reflector volumes will generate large quantities of LLW that would have to be disposed of in the absence of recycling. This is complicated by the presence of carbon-14 due to activation of residual nitrogen in the graphite microstructure. Appropriate graphite recycling and disposal options must be considered to reduce the volume and costs of anticipated waste disposal. Two options are currently envisioned: (1) reuse of blocks after heat treatment to anneal out radiation damage or (2) form new blocks using reconstituted graphite material by crushing and jet milling irradiated blocks to fine powder. Such graphite fabrication methods have been employed before (e.g., BAN graphite). Recycle is considered as a long-term strategy and would only be pursued by vendors when large numbers of HTGRs are developed and a "nuclear graphite economy" is established. R&D would be needed to demonstrate that the recycled graphite demonstrated acceptable in-reactor performance.

2. Background

The basic feasibility of graphite planned for the NGNP has previously been demonstrated in former high-temperature, gas-cooled reactor plants (e.g., DRAGON, Peach Bottom, AVR, THTR and FSVR). These reactor designs represent two design categories: the Pebble-bed Reactor (PBR) and the Prismatic Modular Reactor (PMR). Current commercial examples of potential NGNP candidates are the Gas Turbine-Modular Helium Reactor (GT-MHR) from General Atomics, the High Temperature Reactor concept (ANTARES) from AREVA, and the Pebble-bed Modular Reactor (PBMR) from the PBMR Pty, LTD consortium. Furthermore, the Japanese High-Temperature Engineering Test Reactor (HTTR) and Chinese High-Temperature Reactor (HTR) are demonstrating the feasibility of the reactor components and materials needed for NGNP (HTTR reached a maximum coolant outlet temperature of 950°C in April 2004). This experience has in large part formed our understanding of graphite response within a HTGR nuclear environment.

2.1 Radiation Effects on Graphite

Radiation damage to a solid, crystalline microstructure occurs from either ballistic (atomic or sub-atomic kinetic collisions) or radiological (conversion of radiation-induced electronic excitations to kinetic energy) events. These events can result in significant atomic lattice disruptions, the magnitude of which is significantly dependent upon the bonding energy of the individual atoms.[1] Generally, ballistic events have higher damage efficiencies per event and thus provide a limiting case for materials exposed to such an environment (i.e., a high neutron flux in the HTGR core).

Ballistic neutron damage of graphite and graphitic materials has been studied extensively for decades, and the mechanisms are well understood.[2] Neutron irradiation causes the ballistic displacement of carbon atoms from their equilibrium lattice positions into interstitial positions throughout the microstructure (see Figure 1). Single vacancies and vacancy loops/clusters are left within the basal planes of the crystalline structure causing the basal planes to collapse/shrink (plane destruction) as further damage accumulates and vacancy clusters grow.

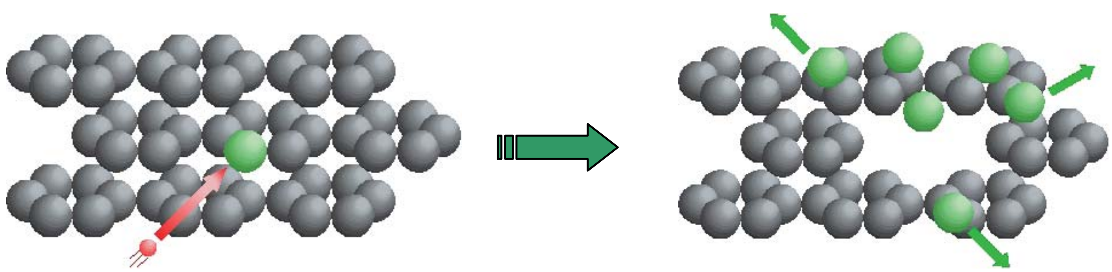

Figure 1. Illustration of (a) subatomic particle (neutron) striking a carbon atom in one of the graphitic basal planes and (b) the resulting ballistic damage to basal planes.

Due to the anisotropic crystal structure of graphite, the interstitial atoms preferentially diffuse and accumulate in the lower energy areas between the basal planes (van der Waals bonds between the covalently bonded basal plane atoms).[3] These small mobile groups of interstitial atoms aggregate into larger clusters, physically forcing the basal layer planes apart. The atoms within the clusters eventually rearrange themselves into new basal planes (see Figure 2), resulting in the expansion of the graphite crystal in the c-axis direction. The corresponding contraction in the a-axis direction (parallel to the basal planes) occurs from vacancy collapse and plane destruction as discussed previously.

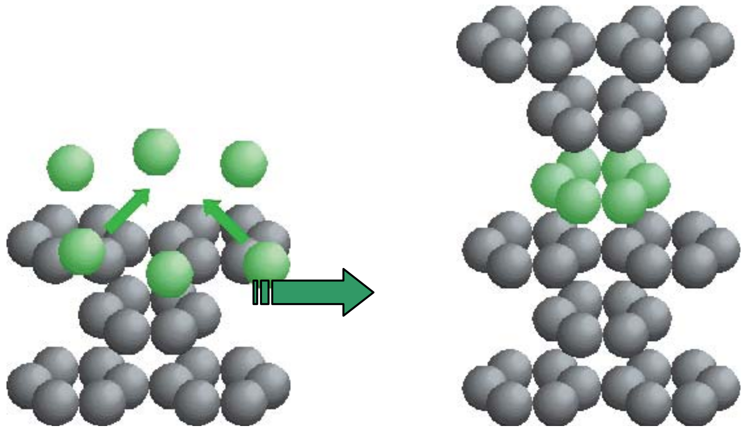

Figure 2. Illustration of (a) interstitial atoms diffusing to lower energy positions between basal planes in the graphite crystal structure and (b) the cluster rearranging itself into a new basal plane.

The mechanical or material effects resulting from these basic radiation damage mechanisms are controlled by a number of factors, including the operating temperature, the degree of crystallinity within the microstructure, the variation of crystallite orientation, and the microdamage within the formed graphitic microstructure during fabrication processes.[4] All these parameters significantly affect the thermo-mechanical response of the graphite, but temperature plays the key role in determining the effects on graphitic structures.

At lower irradiation temperatures (RT - 300°C), graphite structures with high crystallinity and low amount of fabrication defects show significant dimensional swelling in the c-axis direction (depending upon the dose and grades of graphite). Dimensional shrinkage in the a-axis direction occurs, but the rate is considerably smaller, indicating vacancy line/loop collapse or plane destruction. At these (and lower temperatures), significant levels of stored energy (in the form of damage) accumulate within the microstructure and can be released as heat in the graphite crystals.[5] This issue is effectively eliminated at higher operating temperatures where increased point defect mobility promotes significant recombination and the formation of more stable defect clusters.[6]

The physical and microstructural characteristics of the graphite significantly alter the graphite response at higher irradiation temperatures (> 400°C). Processing defects and crystallite misalignment imposed upon the microstructure during fabrication cool down (i.e., cracks parallel to the c-axis planes) physically accommodate the c-axis swelling, and the cracks close as the material swells perpendicular to the c-axis. This crystallite misalignment and damage within the microstructure provides a ready volume of space that can initially accommodate the crystallite swelling during irradiation. Since the c-axis swelling is mitigated, the macroscopic response is one of overall shrinkage due to a-axis shrinkage throughout the graphite volume.

With further irradiation, enough cracks eventually close and the c-axis swelling is no longer accommodated. The macroscopic material response is rapid and irreversible dimensional growth (see Figure 3). When this reversal, or “turnaround”, of the c-axis dimensional change occurs, it is a function of the intrinsic misalignment of the crystallite orientation as well as the level and orientation of the microdamage present within fabricated graphite structures.[7,8] Increasing the irradiation temperature causes faster crack closure due to thermal expansion of the crystallites. Faster turnaround rates are the result (see Figure 4).

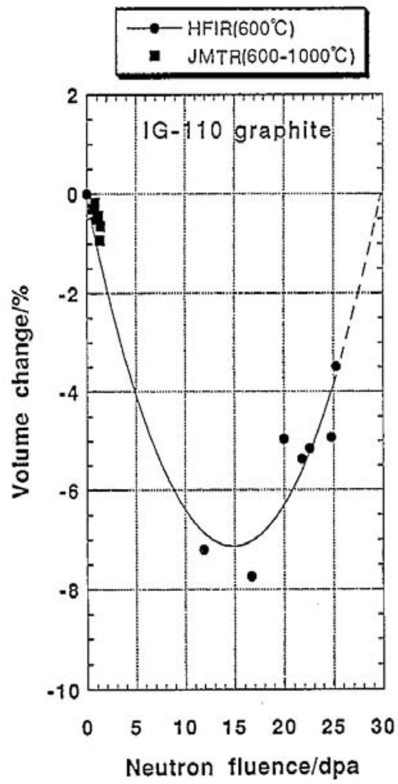

Figure 3. Volumetric changes in an isotropic graphite illustrating turnaround behavior (taken from *S. Ishiyama et al, J. Nuc. Mat., 230 (1996) pp. 1-7*)

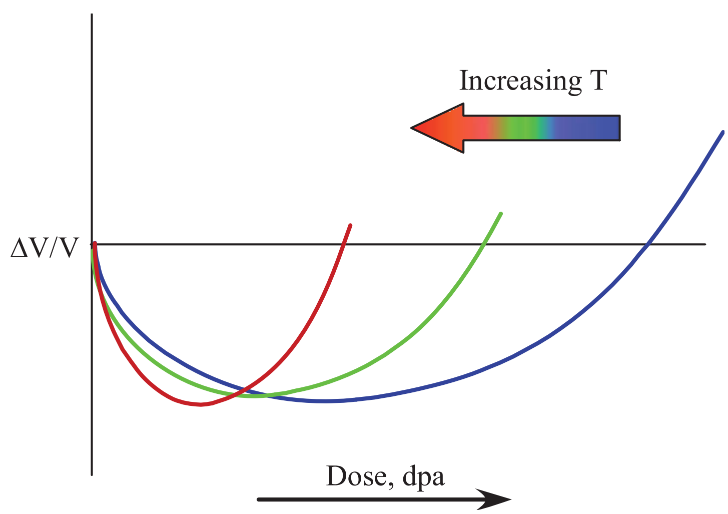

Figure 4. Schematic diagram illustrating the effects of irradiation temperature on turnaround rates.

It is readily seen that the magnitude and rate of dimensional change and the point of "turnaround" are directly related to the degree of crystallinity within the microstructure, the variation of crystallite orientation, process conditions, and the resident micro-damage within the individual graphite types. In addition, the rate of dimensional change is also significantly affected by the irradiation temperature. Typically, the useful lifetime for a graphite type is defined as the time/dose it takes for the material to contract and then swell back to zero dimensional changes.

During operation, graphite components within the reactor core undergo neutron irradiation-induced dimensional change[9,10] (see Section 2.2 below). Local differences in neutron dose and temperature cause differential strains and resultant stresses to develop in the graphite. These stresses are relaxed by neutron irradiation induced creep strains. Thermal creep of graphite is not expected at the temperatures experienced in the reactor core (< 1100°C). The irradiation induced creep strains in graphite can be very large, exceeding several percent; premature failure of the graphite would occur if they were not accommodated by irradiation induced creep. This phenomenon has been shown to be particularly important for Magnox and RBMK plants in which creep is necessary to explain the absence of cracked core components.

2.2 Nuclear Grade Graphite

Nuclear grade graphite is a specially developed composite material manufactured from a filler coke and pitch binder. Nuclear graphites are usually manufactured from isotropic cokes (petroleum or coal-tar derived) and are formed in a manner to make them near-isotropic or isotropic materials. Figure 5 shows the major processing steps in the manufacturing of nuclear graphite. After baking (i.e., carbonization), the artifact is typically impregnated with a petroleum pitch and re-baked to densify the part. Impregnation and re-bake may occur several times to attain the required density. Graphitization typically occurs at temperatures >2500°C with the entire process taking six to nine months.

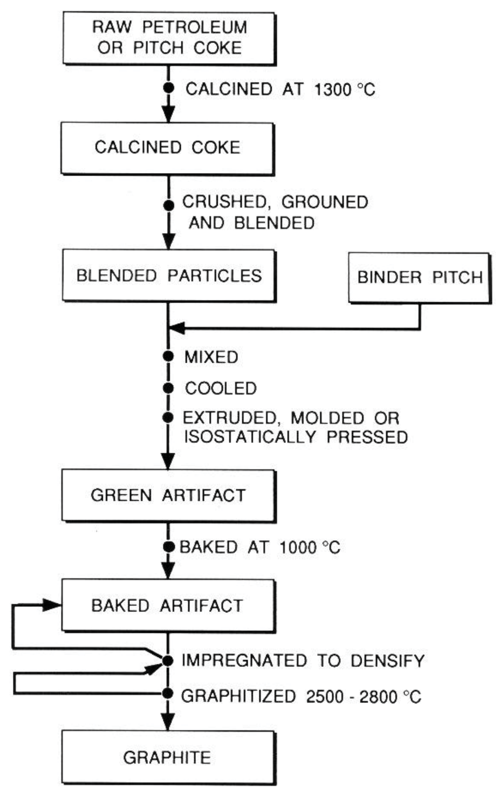

Figure 5. Typical process steps in the manufacturing of nuclear graphite.

Nuclear grade graphite has been specially developed to meet reactor design requirements. Attributes required for modern nuclear grade graphite are:

- Acceptable dimensional change (isotropy)
  - Near isotropic graphite
- High purity
  - Low elemental contamination, especially boron
- Fabricability
  - Ability to machine into large graphite components
- Characterized irradiated material performance
  - Must possess irradiation design database
  - Each graphite type has a unique response to irradiation
  - Graphite of similar grade will not have exact behavior.

While these are minimum attributes necessary to achieve acceptable component lifetimes for use within an irradiation environment, they may not be sufficient to demonstrate adequate structural integrity for all design configurations. It is known that individual "nuclear grade" graphites will have distinctly different responses to the irradiated environments based on the extent of anisotropy, grain size, microstructural defects, microstructure orientation, purity, and fabrication method. As an example and for reasons not fully understood, orthotropic Magnox reactor graphite components show no evidence for cracking, whereas isotropic AGR graphite components show extensive cracking. Thus, the response of each graphite type must be verified for use as a structural component within the NGNP.

The nuclear graphite previously used in the United States for HTR applications (i.e., H-451) is no longer available. New types have been developed and are currently being considered as candidates for the NGNP, but a qualified properties database on these new candidate grades of graphite must be developed to support the design of graphite core components within the specific reactor service conditions of the NGNP. Non-irradiated and irradiated data are required for the physical, mechanical (including radiation induced creep), and oxidation properties of the new graphite. To meet these requirements, a radiation effects database must be developed for the currently available graphite materials. Much more detailed information on graphite fabrication, properties, and the acquisition of bulk material is discussed within the *NGNP Graphite Selection and Acquisition Strategy* report, ORNL/TM-2007/153.

Component lifetime calculations using new graphite types will be determined from both the initial non-irradiated, "as-received" material properties and the property changes that will occur due to radiation damage or environmental degradation to the graphite during operation. The non-irradiated mechanical and material property values will be used as baseline data for initial reactor startup and operation. The "as-received" property values of the graphite components will be used to calculate the initial core thermal properties (e.g., conductivity, specific heat, etc.) and physical response (e.g., applied stresses, dimensional tolerances, etc.).

The evolution of these property changes is dependent upon a number of factors, including temperature, fluence/dose, graphite microstructure/orientation, chemical purity, and applied stresses during operation. Obviously, those components located physically closer to the fueled region of the core will experience higher temperatures and doses than components on the edge of the reactor, and a faster rate of change is expected. The extent of property changes include physical changes to the component (i.e., dimensional changes); changes in the thermomechanical properties, especially irradiation-induced creep; and changes to thermophysical properties, such as thermal conductivity, coefficient of thermal expansion, etc. All of these will affect the prediction of graphite lifetime.

3. Requirements and Service Conditions

Reactor design considerations, design service operating conditions, and reactor safety requirements are key considerations in determining the change in material properties of nuclear graphite. Physical parameters, such as component size, geometry, and machining, may require specific billet sizes and grain size. Operating conditions will specify expected fluence/dose, temperatures, and initial imposed loads upon the graphite components. Finally, safety considerations may require additional material property measurements, such as oxidation rate, properties after oxidation, wear/friction for dust formation, and post-irradiation thermomechanical and thermophysical properties.

3.1 Physical Parameters of Core

3.1.1 Fuel Blocks and Pebbles

For the NGNP graphite program, a compromise between superior material properties and material cost is an important consideration in selecting a nuclear grade graphite. The Japanese IG-110 graphite with its very small grain size and isotropic microstructure shows excellent nuclear response (high stability) and is considered one of the best commercially available nuclear graphites on the market. However, it is prohibitively expensive and the fabrication technique is exacting. As a consequence, the Japanese only use IG-110 in limited applications within the harshest nuclear environments (i.e., inner core components). These issues have lead the Japanese to evaluate different graphites other than IG-110 for future HTGR applications. Similar logic is being applied to the graphite selection for the NGNP program where in the service conditions and applications for each component are evaluated and a suitable nuclear graphite is selected for optimal performance within those particular parameters.

Fuel blocks for the prismatic reactor design have a number of fuel and coolant channel holes drilled axially down the length of the block. The pitch between these holes can be quite small leaving the graphite webbing no thicker than 2 to 3 mm. If the graphite grain size is large, the webbing may only have one to two grains between channels. One to two grains of a material will not represent the true properties of a material providing uncertainty in irradiation stability and strength of the channel webbing. Thus, it is necessary to select a small-grained graphite that can provide 10 or more grains (at least) between the channels. Pebble fuel has no such machining constraints and can easily accommodate large grain sized graphite. A different graphite type can be used if a PBR design is selected for NGNP.

3.1.2 Reflector Blocks

A similar rationale is used when determining the parameters required for graphite blocks used in the inner and outer reflector regions. The continuous refueling design of the PBR allows it to operate without having to shut down for periodic re-fueling. However, the stationary reflector blocks do sustain radiation damage and must be replaced periodically. For economic reasons, a PBR is operated continuously for as long as possible, forcing the reflector blocks to withstand much longer times and higher doses than prismatic reflector blocks are expected to withstand.

Most pebble-bed designs would like the reactor to operate for at least 20 to 25 years before having to de-fuel the entire core to replace the reflector blocks. At expected NGNP fluence levels, this can equate to doses as high as 25 dpa, which is well past turnaround even for the most stable nuclear graphites. Such a high dose level will require careful analysis of the irradiation response of the graphite selected for reflector block use. For purely economic reasons, replacing the reflector blocks within a prismatic core as infrequently as possible would be desirable as well. The more stable the response of the graphite blocks is to irradiation, the lower the costs for replacement material and less down time for the reactor. These economic considerations are important factors for determining the appropriate graphite for the NGNP reactor.

3.1.3 Peripheral Graphite Components

Graphite components outside of the central core region will receive considerably lower doses and operate at much lower temperatures. These include permanent reflector blocks, core support columns/structures, and other graphite components surrounding the core. As a result, the concern is not necessarily irradiation stability but environmental attack or abnormally large stress states on the graphite components. This can dramatically alter the fracture strength, compressive strength, changes to thermal conductivity, and/or emissivity resulting in structural integrity concerns and loss of efficient reactor heat flow.

Oxidation rates, both acute and long-term chronic degradation, are of specific concern during operation. Oxidation of graphitic components can lead to a loss of strength and may affect the thermomechanical and thermophysical properties as well. Further, generation of dust and small particulates from wear and friction can provide a means for spreading contamination when the coolant is released from the primary system. Finally, seismic and large applied loads may exceed the strength of compromised graphite (such as the support columns) causing failure in critical core components. Material properties for all graphite components must be characterized to determine the response of the reactor core and support structures under normal and off-normal conditions.

3.2 Normal and Off-Normal Operating Conditions

Normal operating conditions have been calculated from reactor models based on both PBR and PMR designs. Normal steady-state temperatures and fluxes (dose) based on a 950°C coolant outlet for both reactors are summarized in Table 1. The range of expected dose is rather large and needs to be refined further as designs mature since higher dose rates will lead to a need for greater testing and longer irradiation programs.

Table 1. Reactor operating conditions

Parameter	Prismatic	Pebble-bed
Temperature (Normal operations)		
Inner reflector blocks	1050°C	600-1040°C
Fuel centerline	1200-1250°C	<1100°C
Peak fast fluence (> 0.1 MeV)		
Inner reflector	$1.7 – 12.2 \times 10^{20}$ n/cm^2	$1.6 – 12.2 \times 10^{20}$ n/cm^2
Dose (0.78×10^{21} n/cm^2 = 1 dpa)	0.19 – 0.85 dpa/FPY	0.18 – 0.85 dpa/FPY

If the designs evolve toward the lower range of the estimated dose rates, some of the data requirements (and associated testing needs) driven by higher levels of irradiation damage may not be necessary. For example, if the inner reflector walls only receive a dose of 0.2 dpa, then it will take more than 20 years before the inner walls will obtain a dose approaching turnaround at 1000°C, 5 dpa. However, if the reflectors received a dose of 0.85 dpa/full-power year (FPY), the walls will reach turnaround levels in approximately five to six years. Higher dose data experiments are, therefore, necessary for 0.85 dpa/FPY but not for 0.2 dpa/FPY.

Since turnaround is a function of temperature as well as dose, this also applies to lower operating temperatures within the reactor.

3.3 Anticipated Licensing Data Needs

3.3.1 Research Topics Identified from NRC PIRT

The Nuclear Regulatory Commission (NRC) Phenomena Identification and Ranking Table (PIRT) process was applied to the issue of nuclear grade graphite for the moderator and structural components of an NGNP. An international group of graphite experts used this process to identify and rank by importance any phenomena that may adversely affect the performance of a nuclear reactor during both normal and off-normal operation. Material property changes as well as material response during accident conditions are considered during this process.

A specified PIRT process has been developed to identify and rank phenomena affecting the performance of a nuclear reactor. The first part of the process is identification of phenomena that may significantly affect the performance of the reactor. Second, the phenomena are ranked as high, medium, or low importance and whether there is a high, medium, or low amount of data available to characterize the phenomena. Obviously, those phenomena that have a high impact on the performance with a low knowledge base will score significantly higher in the assessment. The initial ranked phenomena anticipated for graphite components within the NGNP core have been established in the NRC PIRT report, NUREG/CR-6944, Vol. 1-6.

Table 2 summarizes those phenomena within the PIRT report that are deemed pertinent for the anticipated core design and operation requirements of the NGNP graphite R&D program. Both normal and off-normal operation (postulated accident conditions) were considered for either a prismatic reactor design or a pebble-bed reactor design.

Table 2. Research areas containing the identified PIRT performance phenomena.

Structural integrity of graphite	Retention of long-term structural stability and mechanical strength under specified loads. Specified by ASME requirements.
Thermal response of graphite – normal operation	Changes in thermal properties at peak dose and temperatures.
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.
Changes to by-pass flow	Potential coolant flow issues due to shrinkage and swelling of graphite components.
Chemical and mechanical core stability	Oxidation and subsequent structural stability of oxidized graphite. For both acute (accident) and chronic (normal operation) conditions.

Most of the identified performance phenomena from NUREG/CR-6944, Vol. 1-6, can be summarized within these main research areas. All of these areas of research are currently being integrated into the NGNP graphite R&D program to characterize reactor design, licensing, and operational performance of graphite.

3.3.2 Full Operation or Partial Operating License

The NGNP program may elect to apply for a partial (or demonstration) license to the NRC for this first reactor. The assumption is that a fully qualified graphite will not be required for use within a demonstration reactor and the program will not need to perform some of the higher dose experiments to support a full operating license before startup can occur. In addition, the demonstration plant may not operate at full design power, thus producing less fluence and lower temperatures than expected for full power operation.

As a result, the regulator may be satisfied with some or only part of the data needed for full qualification effectively giving the program more time to gather the required data for full licensing of the graphite. Experiments necessitating longer times and higher irradiation dose can be delayed until the reactor will actually be operated at the higher temperatures and fluences expected at full design power.

3.3.3 Full Data Set or Extensive Core Inspection Program

Rather than fully characterizing the graphite before building the reactor, the NGNP program may elect to have an extensive core in-service inspection (ISI) program. As stated previously, one can be relatively certain that any of the current nuclear graphites (isotropic, pure “nuclear” grade graphite) will be stable for a short period of time within an HTGR core. However, an extensive core inspection program will be required to assure the NRC (and other regulatory groups) that the graphite is behaving as predicted since there will be insufficient verification data. In addition, since an ISI program can only monitor a fraction of the core, there will need to be additional verification data in the form of a characterization program (non- and irradiated material) conducted in parallel while the reactor is operating. The characterization program can be limited in scope since a large portion of the verification resides with the core inspection.

The PBMR reactor in South Africa is pursuing just such a “defense in depth” core inspection approach. Using the NBG-18 nuclear grade graphite from SGL Group, Inc., the PBMR project has chosen to build the reactor core without a complete database of material properties. The PBMR project will supplement some non-irradiation material properties for NBG-18 with extensive in core inspections during the first four to five years of operation. Meanwhile, an irradiation program specifically characterizing NBG-18 irradiation response will be conducted in parallel to provide a more comprehensive database for use in developing the long-range goal of a predictive model for graphite behavioral response.

3.4 Extent of Design Codes and Methodology Required (ASME Controlled)

To provide a consistent nuclear grade graphite material for eventual standardization and commercialization of HTGRs, an ASTM standard specification for isotropic and near-isotropic nuclear graphites (D 7219-05) is required. Additionally, ASME codes and guides for materials selection and qualification, design, fabrication, testing, installation, examination, inspection, and certification are needed. These standards will be necessary to approve future grades of nuclear graphite for new HTGRs and are under development by the international graphite community.

Establishing a nuclear graphite code case by the ASME will require specific and detailed data. The current direction from ASME is to specify a prescriptive method for gathering the requisite graphite characterization data. All gathered data, constitutive relationships, and the associated predictive models resulting from these relationships must be demonstrated to be adequate before the appropriate committees prior to approval of an ASME code case. Two approaches can be used to demonstrate the validity of the code case to the ASME committees – deterministic and probabilistic.

A probabilistic approach will be more accurate to describe failure within a ceramic material such as graphite (i.e., flaw size and distribution through out the microstructure) than the deterministic approach. Furthermore, the probabilistic approach can (a) accommodate the inherent variability in as-fabricated and irradiation-induced changes to the graphite microstructure and properties and (b) reduce unnecessary conservatisms inherent in deterministic models that result in the need for excessive design margins.

4. Material Property Needs

Based on the parameters discussed previously, the technical areas for R&D are outlined below. Material property values within three primary areas (physical, thermal, and mechanical) will be required before any graphite can be used in the NGNP. Specific material properties within each area are identified and the reasoning for obtaining this data is defined for each property.

4.1 Physical

Generally, physical material properties are concerned with characterizing the microstructure and the effects of microstructure on the macroscopic response of the material (i.e., dimensional changes).

Microstructural Characterization. Determination of grain size, morphology/anisotropy, and pore size/distribution within the graphite microstructure is critical to determining the macroscopic physical, thermal, and mechanical properties. These parameters must be determined before an accurate analysis of the graphite performance can be made.

In addition, a key technological deficiency is the inability to determine microstructural features within a graphite material, specifically with non-destructive techniques. This is important for determining not only the evolution in test specimen microstructures as a function of irradiation, but also for determining defects within the large graphite billets. Inspecting billets without damage to ensure proper microstructural development is one of the largest problems facing any Quality Assurance (QA) program for purchasing of nuclear grade graphite. The implementation of non-destructive techniques will be necessary for accurate quality assurance.

Mass and Dimensional Measurements. Dimensional change is one of the key parameters defining a nuclear grade graphite. Determination of volumetric and density changes as a function of temperature and dose will be necessary to understand critical performance measures, such as turnaround, irradiation creep, and internal stresses imposed upon graphite components.

4.2 Thermal

Thermal material properties are critical for protecting the fuel particles during off-normal events as well as for predicting thermally induced stress states within solid graphite components (i.e., reflector blocks). Degradation in thermal properties – conductivity, specific heat, and coefficient of thermal expansion (CTE) – will significantly impact the ability of the graphite to both absorb energy and transfer the heat load out of the core region during an off-normal event. Without adequate removal of the heat, fuel particle centerline temperatures will exceed the design limits resulting in unacceptable numbers of particle failures and radiation release levels. In addition, thermally induced stresses can be exacerbated between and within graphite blocks with significantly altered thermal properties. Elevated stress levels can exceed the structural strength of the graphite blocks, resulting in cracking, spallation, and structural stability.

Thermal Expansion. The CTE for graphite components is critical for determining the dimensional changes that occur as a result of temperature increases. Localized external stresses can be imposed upon mechanically interlocked graphite core components as the individual pieces suffer differential expansion. Internal stresses can occur within larger graphite components if there is a temperature gradient causing differential expansion within the piece (i.e., one side has a higher temperature than the other). Finally, the thermal expansion is greatly dependent upon the graphite microstructure, such as orientation/anisotropy, pore size and distribution, and crystallinity.

Irradiation damage can alter graphite CTE values significantly, and changes must be quantified to determine the extent of change to the CTE. A reduction or increase in CTE can significantly affect the stresses (internally and externally) imposed upon the graphite components within a reactor core and will directly affect component lifetime. Determining the changes to the CTE as a function of irradiation dose and temperature will be a key parameter for reliable calculation of stress states within graphite components, volumetric changes, and irradiation creep rates.

Thermal Conductivity. The ability to conduct heat through the graphite core is critical to the passive removal of decay heat. Reduction of the thermal conductivity within graphite can significantly affect the passive heat removal rate and thus the peak temperature that the core and, subsequently, the fuel particles will experience during off-normal events. Determining changes to the conductivity as a function of irradiation dose and temperature is important for the safety analysis for the HTGR.

Oxidation. The oxidation rate of graphite during an off-normal, air-ingress event is required to determine the effect of oxidation on the specific graphite properties as well as the entire core performance. There are two primary concerns: failure of individual graphite blocks (due to strength and thermal conductivity reduction as a result of pore formation and growth) and general core geometry configuration issues (the entire core fails due to acute oxidation and catastrophic graphite failure). Kinetic models resulting from experimental data will be required to predict weight loss in specific areas of the core. It is expected that the damage will be limited and that core geometry remains intact; however, some data will be required to confirm this assessment.

Additionally, based on regulatory requirements, thermal and mechanical testing of previously oxidized material will need to be performed to determine the chronic effects oxidation may have on graphite material properties. Mechanical and thermal properties will be investigated from both acute and chronic oxidized material. The affects due to chemical and physical (pores) differences for each graphite type will be required.

Emissivity. Emissivity values for graphite must be high enough to allow heat energy to pass across the gap between the core and pressure vessel walls. Graphite emissivity is primarily a function of surface conditions for the graphite components (i.e., roughness, porosity, etc.). The as-received graphite is assumed to have high enough emissivity values to meet the heat conduction values. Confirmation that emissivity values do not degrade extensively due to oxidation and/or irradiation is needed.

Specific Heat. There are concerns that energy stored within the graphite microstructure as a consequence of irradiation damage can be released if graphite is raised to a high temperature (the Wigner energy release phenomenon). If there is an off-normal event where the graphite is undergoing air oxidation, this additional stored energy, along with the heat generated from graphite oxidation, may exceed the specific heat value and produce a run away reaction. A few studies suggest that very high energy peaks may occur at the high operating temperatures of the NGNP.

It is generally understood that irradiation damage energy (accumulation of Frenkel defects) is only available when it is “frozen” in the microstructure at low irradiation temperatures (< 300°C). When graphite is irradiated at temperatures higher than 300°C the increased point defect mobility affectively anneals out this accumulated damage within the graphite microstructure as fast as defect pairs can occur. Thus, at higher irradiation temperature it is assumed that Wigner internal energy release will be minimal, but some testing to determine the stored energy levels within graphite will be required.

Electrical Resistivity/Conductivity. Electrical conductivity values for graphite are not specifically required. Electrical conductivity is used as a rapid, simple means to determine grain orientation, structure, and crystallinity of the graphite. In conjunction with optical microscopy, it can be used to determine the microstructural texture of the graphite components without a great deal of sample preparation work.

4.3 Mechanical

The graphite single crystal is highly anisotropic due to strong covalent bonds between the carbon atoms in the basal in the plane and weak van der Waals bonds between the basal planes. This anisotropy is transferred to the filler coke particles and also to the crystalline regions in the binder phase. Thus, the mechanical and physical properties of graphite vary within a billet due to texture introduced during forming and thermal processing. Moreover, there is statistical variability in the properties between billets within the same lots, between lots, and between batches due to variations on raw materials, formulations, and processing conditions. Accurate characterization of the mechanical properties is fundamental to determining the induced and applied stresses to the graphite components. Determining the resulting stresses in (and on) the components from exposure to a reactor environment is necessary to calculate the ability of the graphite to withstand the imposed loads and continued service conditions during operation. Therefore, it is necessary to develop a statistical data base of the properties for a given graphite grade.

Irradiation Creep. Strain relief of induced stresses (i.e., irradiation creep) within irradiated graphite microstructures allows the graphite to withstand irradiation damage. However, graphite will continue to suffer from irradiation creep even after initial internal stresses are relieved (i.e., primary irradiation creep), resulting in continued dimensional changes. The resulting macroscopic behavior is similar to the changes in CTE discussed above. Thermal creep does not occur at expected NGNP operating temperatures.

Because irradiation creep can alter the underlying microstructure, it can affect the material properties in nuclear graphite during long-term exposure. The graphite performance and changes to the material microstructure and properties during long-term exposure must be characterized and understood to validate the design and establish accurate lifetimes for new graphite types.

Finally, since irradiation creep specimens are physically large, it is relatively easy to irradiate a large number of specimens simultaneously inside an irradiation creep experiment. Therefore, while investigating irradiation creep rates, all other irradiated material property values can also be determined utilizing both the creep samples and piggy-back irradiation samples within an irradiation capsule.

Elastic Constants and Stress-Strain Curve. The mechanical properties of graphite are necessary to determine the structural integrity of graphitic components. These properties are vital to determining the viability of the structural strength and integrity of the reactor core. The as-received and irradiated values are needed for whole core models, which will be used for the graphite design code. This is discussed in later Sections of this plan. Specific material properties required for the whole core modeling are listed in Table 3:

Table 3. Mechanical properties

Static and dynamic elastic modulus
Shear modulus
Poisson's ratio
Creep Poisson's ratio
Strength values : (σ_{flex}, $\sigma_{tensile}$, $\sigma_{compression}$)
Strain to failure
Fracture toughness : (K_{Ic}, G_{Ic}, σ_f)
Multi-axial failure criteria*

* Multi-axial stresses are anticipated to be induced in the front reflector blocks, which experience the largest dose and temperature levels. Gradients in dose and temperature within the block lead to tensile stresses on the front of the block gradually changing to compressive stress states on the back of the block, which induce high multi-axial stresses resulting in potential failure within the block.

Tribology (wear/friction) . Tribology is primarily a concern with pebble-bed designs. The concern is that wear on the pebbles during movement can generate dust, which will act as a means for transporting fission products during loss of coolant. To determine the amount of dust to be generated, the tribological properties of the graphite must be determined.

5. Technology Development Plan

The scientific and engineering techniques described within this section encompass all the anticipated tests required to validate and qualify nuclear grade graphite for use within the NGNP. The plan presented here represents the information needed for full operational license of a prismatic NGNP reactor design. The test matrixes could be limited to reduce the scope of the testing in support of a limited licensing strategy (i.e., demonstration plant license) if necessary to meet NGNP deployment schedules. For a pebble-bed HGTR design, additional testing will be required to support the longer design life (i.e., high dose levels) of the front facing reflector blocks. In addition, the slightly lower inlet temperature (from prismatic design levels) may require changes to the test matrix parameters to ensure the tests bounds the operating envelope. The high dose irradiation experiment are included here for completeness but will be separated from the prismatic cost baseline. Ultimately, data from all tests will be required for commercialization of the HTGR technology in order to use a new graphite type within an HTGR.

Constitutive relationships and model development using the data acquired from this R&D program will be required for codification of the graphite. The appropriate role of model development and the extent of development are discussed as it pertains to perceived ASME and regulatory requirements.

5.1 Experimental Data

Since many graphite components will be exposed to the full neutron flux generated in the NGNP core, any changes to pertinent material properties must be determined to understand the long-term behavior of the graphite in reactor. As a consequence, an extensive non-irradiated and irradiated material characterization program is planned and currently in progress.

The non-irradiated characterization program focuses on developing a statistically valid material database for each of the graphite types selected for irradiation testing. This will establish baseline values for material properties that can be used to determine the quantitative changes during irradiation. A large irradiated specimen population will then be exposed to the expected NGNP reactor environment (temperature and dose ranges). As stated previously, this experiment is expected to yield pertinent information for all irradiated material property values necessary to qualify a new nuclear grade graphite. However, doses and temperatures experienced within the prismatic reactor design are expected to be different than the pebble-bed design. Generally, the prismatic reactor design operates at slightly higher temperatures while the pebble-bed design is expected to operate for long periods of time (see Section 3.1.2).

Specific descriptions of test sample preparation, non-irradiated material characterization, irradiation experiment descriptions, and material characterization comprising the experimental data needs are outlined below.

5.1.1 Test sample preparation

Before any material characterization testing (non-irradiated or irradiated) can be carried out, an optimal method of machining the graphite samples from the bulk material must be developed to ensure representative samples can be obtained. The NGNP program has developed an extensive sample cutting and sectioning plan to guarantee not only statistically valid sample numbers but also spacial validity so that microstructural changes within the bulk material (i.e., billet) affecting material property changes are well characterized.[11] Particular attention has been given to the traceability of each specimen to its spatial location and orientation within a billet.

These graphite billet cutting plans were developed to promote a more complete or finer resolution material property "mapping" of material property changes within the billets. This was achieved by maximizing the number of test specimens that could be obtained from each billet. However, to provide statistically significant results from the various test methods, a minimum of four samples are needed from each location/orientation within the same billet (per ASTM methodologies). Since this is physically impossible, it is assumed that the billets have some level of symmetry in material properties throughout the entire structure, which allows samples from different sections of the billet to effectively be "similar" with respect to material properties (see Figure 6). Using samples from similar locations within each billet section will yield enough samples to provide for statistical validity within a single billet.

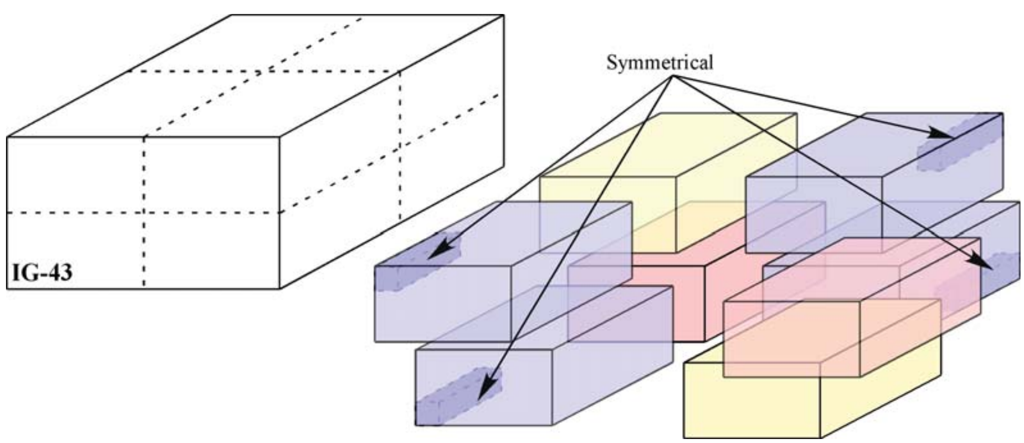

Figure 6. Example of symmetrical quadrant sections within a graphite billet

To facilitate machining, the billets are cut into successively smaller sections designated as "slabs", and each slab is sized to accommodate the proper grain orientation within the test specimens. All test specimen blanks machined for a given slab will have the same grain orientation (ag or wg) as the slab. The slabs will be further sectioned into sub-slabs to allow the rectangular test specimen blanks to be machined to the correct size. Finally, a tracking methodology is used that will account for every specimen machined from a graphite billet. A unique identification number will be assigned to each test specimen providing the exact location and orientation of the sample within the graphite billet. This identification system is based on the cutting methodology to provide an easy and concise method for identifying the different samples. This methodology (and the corresponding assumptions) will be used for producing all test specimens for material characterization.

Since a large portion of the testing is with irradiated samples, a minimum specimen size must also be considered for volume restrictions within a materials test reactor (MTR). Each material test will depend on the specific graphite's grain size since ASTM test standards call for specimen sizes to have cross-sectional diameters of at least 5X the grain size across the stressed gauge section of the sample. This sizing requirement helps ensure representative and repeatable testing results. However, most graphite material tests use a minimum of 10X of the maximum grain size for the cross-sectional diameter across the gauge section. Thus, for a graphite with a 1-mm grain size, the minimum diameter in the test gauge area for a typical tensile specimen must be ~10-mm. Fabricating smaller test specimens is not allowed since they will not provide representative material properties across such a few number of grains in the material microstructure. Other graphite types, which have smaller grain sizes, may use smaller gauge sections, again depending on the material's grain size.

5.1.2 Non-irradiated Material Testing

Baseline, "as-received" material properties for each graphite type are needed to establish accurate thermal and mechanical response of the core. Since material properties are expected to vary throughout the rather large billets or blocks of graphite, mapping the magnitude and spacial positions of variability is important to determining an individual component's material properties. To enable credible core designs and to support the ongoing development of a probabilistic graphite design methodology, the maximum variability within graphite components must be well characterized. For example, if the compressive strength is reduced significantly near the edges of the billets, a graphite support column fabricated from a position near the edge may not possess sufficient strength to support the weight of the core blocks above. Thus, determining where the strength begins to be reduced within the larger billet and by how much is important for determining where to fabricate an individual graphite component to meet specific design requirements within the core.

A complicating factor is the variability not only within the individual billets but also from billet-to-billet and finally lot-to-lot. These within-billet, intra-billets, and lot-to-lot variations of the graphite must be accounted for in a statistical manner to determine the maximum range of material property variations expected for components machined from an "average" billet. Such a statistical material property database can only be obtained from extensive non-irradiation characterization of samples taken within billets and compared to sample between different billets and different graphite lots.

Physical, thermal, and mechanical property testing of multiple graphite samples from a large billet sample matrix is necessary for determining the proper statistical ranges of values. The appropriate sample matrix size, sample geometry, and sample dimensions as described above will be important to establishing statistical validity. All material tests to be used to build this material property database are described later in this section. Once the non-irradiated, "as-received" material properties have been determined, the changes due to irradiation will be determined from post-irradiation examination (PIE) and characterization studies on representative graphite types.

5.1.3 Irradiation experiments

A series of irradiation experiments will be required to determine the graphite response under irradiation. After the graphite sample matrix is chosen, the irradiation conditions are determined based upon the expected reactor conditions as described in Section 3. PIE and characterization entails performing the same physical, thermal, and mechanical tests as described for the non-irradiated materials, only this time with irradiated graphite samples.

As discussed previously, thermal creep of graphite is not expected at the temperatures experienced in the reactor core (< 1100°C), so determining the non-irradiation creep rate is not required. However, irradiation induced creep in graphite is expected at these temperatures and will play an important role in the irradiated behavior of the graphite during reactor service. Thus, irradiation creep experiments will form a significant part of all irradiation studies for graphite types in the NGNP.

5.1.3.1 AGC Experiments

The Advanced Graphite Capsule (AGC) experiments are designed to provide irradiation creep rates for moderate doses and higher temperatures of leading graphite types that will be used in the NGNP reactor design. The experiments are designed to provide not only static irradiation material property changes but also to determine irradiation creep parameters for actively stressed (i.e., compressively loaded) specimens during exposure to a neutron flux. The temperature and dose regimes covered by the AGC experiment are illustrated in Figure 7.

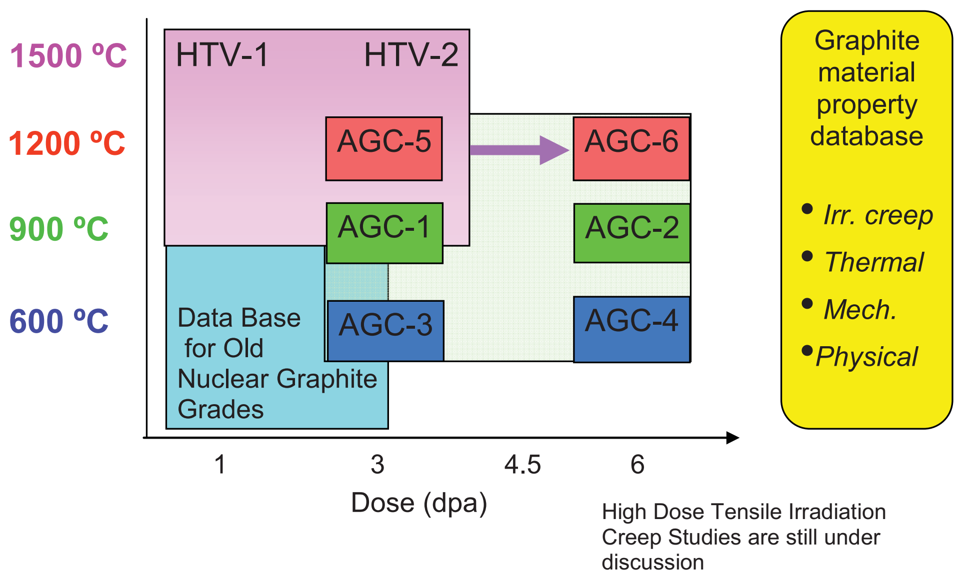

Figure 7. Schematic diagram illustrating dose and temperature ranges for AGC and HTV experiments.

As shown, the dose and temperature is bounding for the prismatic reactor design (dpa ~ 5 – 6 at 1100°C) for both fuel and front facing reflector blocks. This dose limit is intentionally below the expected point of turnaround for the current NGNP graphite types at normal NGNP operating temperatures. Only AGC-6 experiment (6 – 7 dpa at 1200°C) may approach expected turnaround limits for the selected NGNP graphite types.

To determine when (and if) turnaround will occur for the selected NGNP graphite during exposure in the AGC-6 capsule, an additional experiment –high-temperature vessel (HTV) has been postulated. The

HTV 1 and 2 capsules are simple “drop-in” capsules with the exposure parameters illustrated in Figure 7. As shown, these experiments are operated at much higher temperatures (inducing faster turnaround) but at lower doses. As this is a simple dimensional change experiment to determine when turnaround may occur, the graphite is not loaded during irradiation. A detailed description of the HTV 1 and 2 experiment is presented in ORNL-GEN4/LTR-06-019.[12]

Since the prismatic NGNP design estimates that reflector blocks can be replaced well before turnaround should occur at normal operating temperature (<5 – 6 dpa) and fuel blocks are replaced after only two cycles (<4 – 5 dpa), the AGC experiment should fully bound the graphite experience within a prismatic design. The dpa levels achieved in the AGC experiment will not, however, fully bound the pebble-bed NGNP design for high-dose reflector blocks (see below). But, it will certainly provide preliminary data for the first 20 – 25% of the expected dpa levels for these graphite components.

Graphite components located farther from the core region will have correspondingly less dose and operate at much lower temperatures than the fuel region. As a consequence, turnaround and irradiation creep levels for these peripheral graphite components will be at significantly longer times and lower rates and should be fully bounded by the AGC data.

All six capsules comprising the AGC experiment will be irradiated in the South Flux trap of the Advanced Test Reactor (ATR). This will require sequential irradiation campaigns for each capsule. Each capsule will contain approximately 400 specimens – 90 large irradiation creep specimen pairs and over 300 “piggy-back” specimens. The smaller and non-stressed “piggy back” specimens are located in the center channel in the experiment. Other “piggy backs” are used in the lower non-stressed creep specimen channels as offset specimens to account for the slight asymmetry in the ATR flux profile. The larger irradiation creep specimens are sub-divided into six columns of 15 specimens each. Each of the six columns contains stressed and non-stressed specimens. The symmetry of the flux buckling is used to irradiate each stressed and non-stressed specimens at the same fluence level. There are seven non-stressed specimens below core centerline and eight stressed specimens above core centerline. The creep measurement is made on the dimensional difference between the stressed and non-stressed specimens irradiated at the same fluence and temperature. Figure 8 shows the arrangement of the graphite specimens in the experiment.

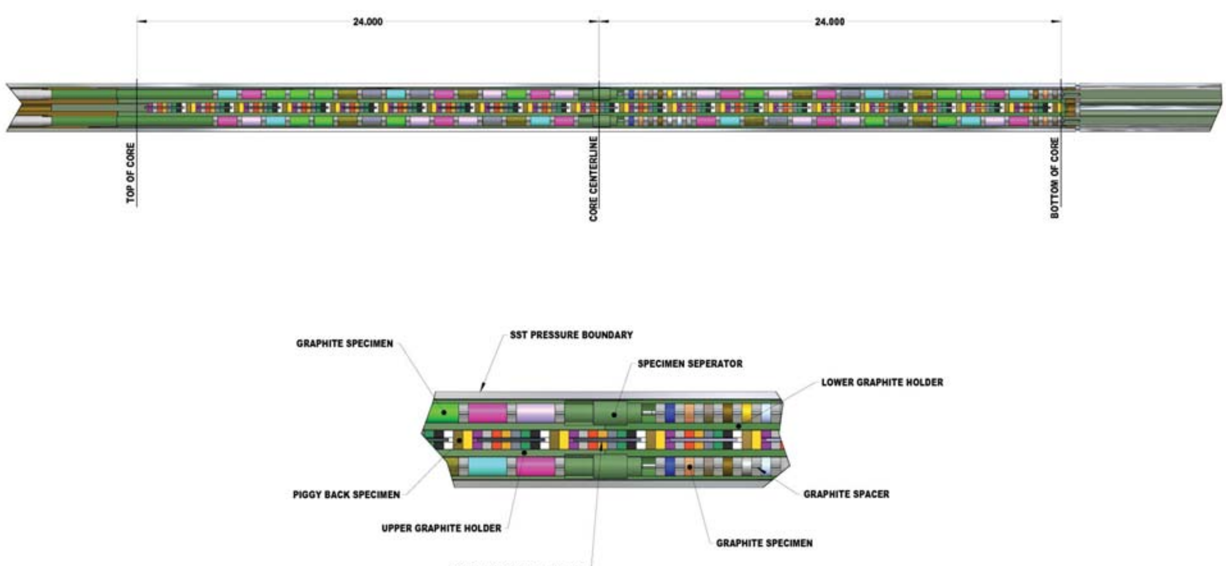

Figure 8. Internal configuration of AGC experiment.

All specimens are maintained at a constant temperature during exposure times of between six and 18 months depending on the equired dose (see Figure 9). PIE characterization is projected to take approximately 14 – 18 months for each capsule even though irradiated graphite samples can be contact handled after a short decay period (~ 6 months).

5.1.3.2 High Dose Irradiation Experiments

The high-dose experiment is designed to provide irradiation exposure for very high doses and moderate temperatures. As noted above, the pebble-bed design expects the facing reflector blocks (inner and outer reflector) to operate at much longer times and thus withstand a maximum of irradiation damage before the core is shutdown, de-fueled, and the blocks replaced. Current expectations are for the reflector blocks to operate approximately 20 – 25 years before replacement. At the higher end of the dose range noted above for a pebble-bed NGNP design, this can correspond to as much as 25 dpa before change-out. While this appears to be a very large dose, the expected temperature ranges are lower than for a prismatic design resulting in longer turnaround times and slower irradiation creep rate.

If a pebble-bed design is selected, the graphite material property changes for the higher expected dose levels will be required. A high dose creep experiment exposing selected graphite to much longer dose levels at moderate temperatures has been tentatively planned in support of this design selection.[1] The temperature and dose regimes covered by this high dose experiment are illustrated in Figure 10.

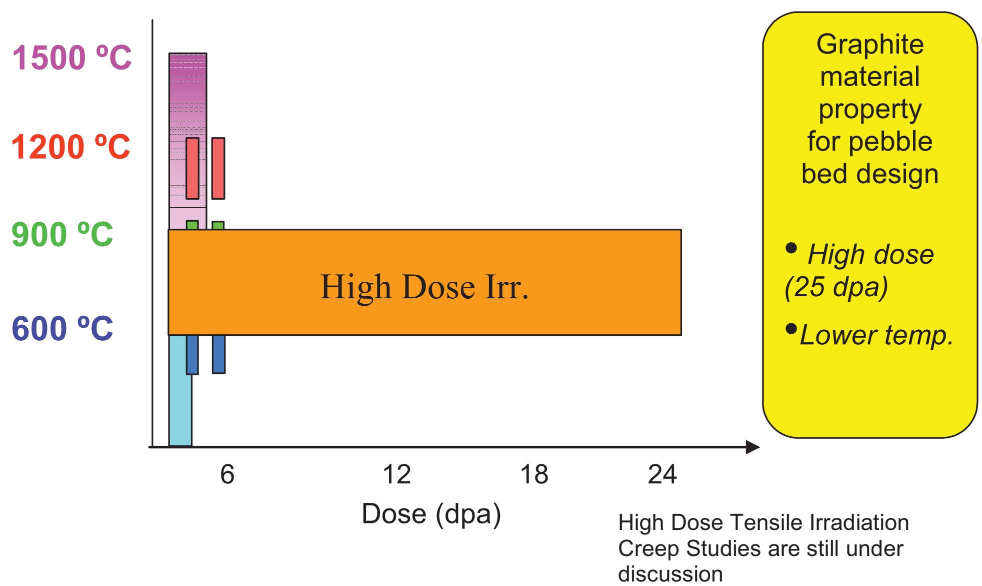

Figure 10. Schematic diagram illustrating dose and temperature ranges for High Dose Irradiation experiments.

[1] Given the similarities between and NGNP pebble-bed and the PBMR in South Africa, were a pebble-bed selected for the NGNP, there may be the opportunity to further optimize the test matrix proposed here by leveraging the planned graphite irradiations for the PBMR.

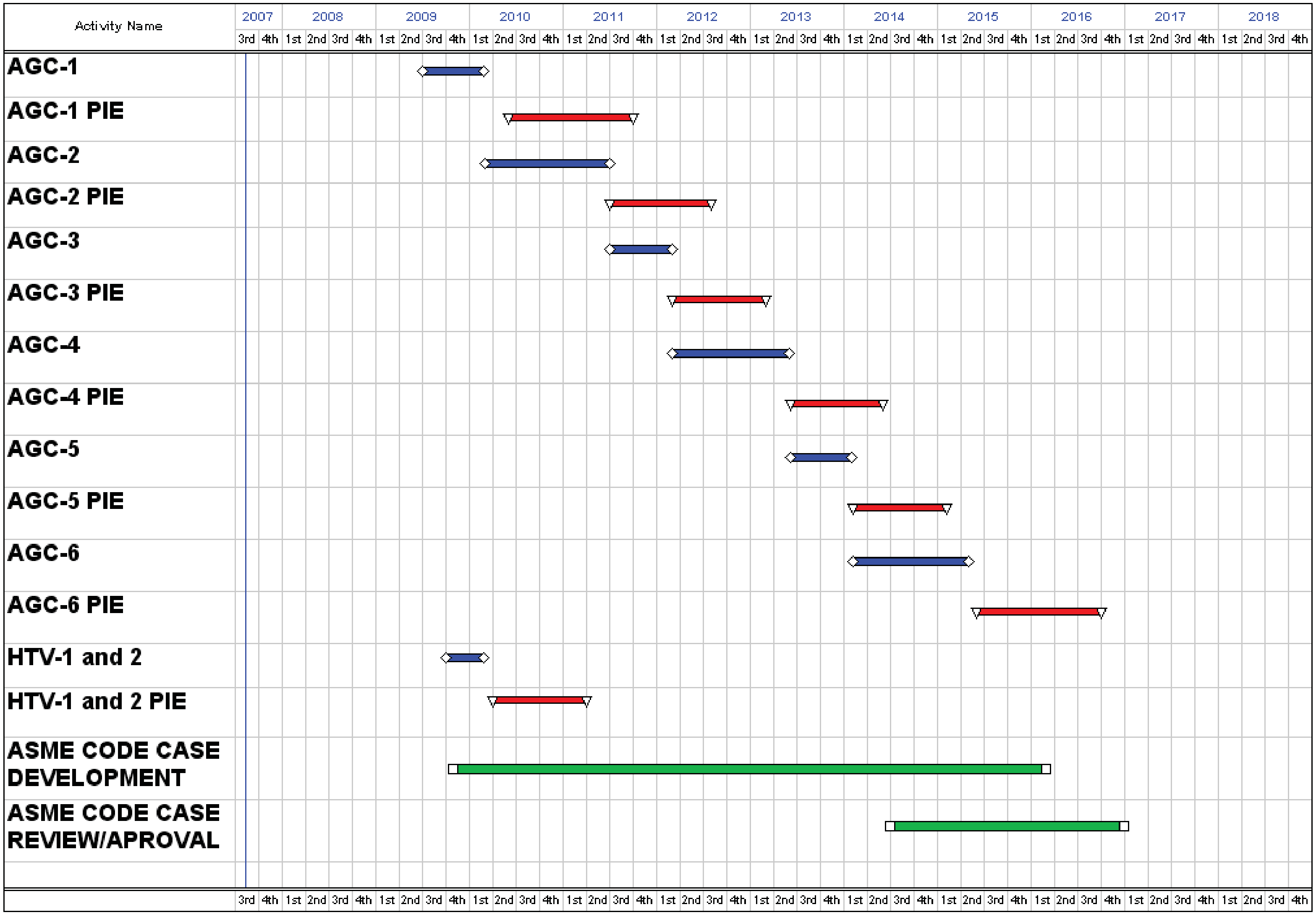

Figure 9. Estimated irradiation and PIE schedule for AGC experiment.

As shown, these samples will be exposed to dose levels considerably higher than expected turnaround dose levels even at the moderate exposure temperatures. In addition, the irradiation creep samples will be tensile loaded during exposure to ensure optimal creep rates (Section 5.1.4). However, since these dose levels are expected after 25 years of service, the high dose experiments are not needed for initial material property ranges specifically required for reactor licensing and startup operations. Results stemming from this experiment can (and most likely will) be delayed for a few years until after reactor startup since data from the AGC experiments will provide sufficient data to support operations for at least 6 – 7 FPY operation of the reactor. The data from the high-dose experiment will be required if any graphite reflector block will be exposed to doses higher than 6 – 7 dpa.

5.1.4 Material Characterization

The following sections describe the material tests anticipated for all non-irradiated and irradiated examination and characterization studies. These material tests will be applied to both irradiated and "as-received" graphite samples to ascertain the changes to the material properties resulting from a neutron radiation field. Where possible, ASTM standard test methods will be employed. If no test standard exists, some additional activity may be required to develop a test standard.

5.1.4.1 Physical testing

Microstructure Characterization. Optical microscopy measurements of grain size, morphology/anisotropy, and pore size/distribution will be used to determine the graphite microstructure. Adjacent optical samples will necessarily be taken as close to test specimens (with similar orientation) as possible from within the graphite billet, and the microstructure will be inferred to the test sample microstructures.

In conjunction with the optical microscopy, non-destructive X-ray tomography (CT) will be investigated for its ability to ascertain the microstructure within individual test samples. X-ray techniques will allow samples to be analyzed before being tested using additional techniques (both non-destructive and destructive). Image analysis techniques will be used to enhance information such as internal pore sizes and pore structure throughout the microstructure.

Changes to the microstructure due to thermal, irradiation, and stress history will be compared to the original microstructure. Microstructural evolution and modifications as a result of exposure to reactor environments can then be established.

Non-destructive methods for large scale analysis (i.e., manufacturing, ISI methods, etc.) will need to be developed. CT methods may be possible but have limited resolutions for these large component sizes. Ultrasonic testing (UT), electrical resistivity/conductivity, impact echo, or other techniques will have to be developed to meet both ISI requirements as well as billet characterization for manufacturing QA.

Mass and Dimensional Measurement. Precision measurements of all irradiation test samples will allow macroscopic dimensional changes and pore formation estimates to be determined. Volumetric and density changes will be calculated and compared to pre-irradiation values for each test sample.

5.1.4.2 Thermal testing

All thermal (and electrical) samples will be button samples having dimensions equal to or less than 12mm diameter x 6mm thickness. These small sample sizes allow for many specimens to be made available for both irradiated and non-irradiated testing. In addition, the small size also allows thermal samples to be machined from the ends of mechanical test specimens, if needed. This ensures spacial uniformity of measurements of relevant thermal, physical, and mechanical material properties within the graphite billet characterization. Additionally, "re-using" the same samples allows for larger sample batches within the irradiation test trains.

Thermal Expansion & Conductivity. Thermal expansion and conductivity values will be obtained from graphite button samples within a laser flash diffusivity analyzer to temperatures of 1600°C (off-normal maximum temperature). Non-irradiated and irradiated button samples will be prepared for testing at all temperature ranges of interest.

Oxidation. There are currently no approved ASTM test methods for measuring the oxidation rate of graphite. The NGNP program is assisting in the development of this test standard. After development, the test will be used to ascertain the oxidation rate of selected graphite types for a variety of operating conditions. Results will be used to develop kinetic models to predict weight loss in specific areas of the core.

Additionally, based on regulatory requirements, thermal and mechanical testing of previously oxidized material will need to be performed to determine the effects oxidation may have on graphite material properties. A core configuration issue as a result of air ingress is establishing that, whatever mitigation techniques are selected for this event in NGNP, an assessment that the damage is limited and that core geometry remains intact is important. In addition, mechanical and thermal properties will be investigated from both acute and chronic oxidized material.

Emissivity. Limited confirmatory measurements of emissivity values for graphite will be measured using standard techniques (i.e., infrared based, etc.). NRC PIRT requirements will demand some comparative studies to determine any changes in emissivity resulting from oxidation and/or irradiation.

Specific Heat. All thermal specimens will be subjected to analysis via Differential Scanning Calorimetry (DSC) to determine the specific heat for individual samples. Changes to the specific heat due to oxidation and/or irradiation will be compared to as-received values. In addition, previously irradiated samples from AGC capsules will be monitored to ascertain the potential reduction in specific heat due to the release of high temperature Wigner energy. These will be limited confirmatory studies to ascertain the potential for Wigner energy storage at the lower NGNP irradiation temperatures.

Electrical Resistivity/Conductivity. Electrical conductivity/resistivity values will be measured through sample button resistivity measurements. Microstructural characteristics will be compared to optical and CT results. These tests will be performed as possible based on the material geometry and size.

5.1.4.3 Mechanical Testing

Mechanical testing is the most extensive and complex part of the graphite test program. Strength, irradiation creep, fracture toughness, and multi-axial testing procedures utilize complex sample geometries, complicated testing techniques, and take a long time to perform.. Therefore, the techniques and plans outlined for these mechanical tests, such as the irradiation creep tests, require careful consideration.

Irradiation Creep. An extensive irradiation creep program is needed to characterize graphite creep response as part of a larger irradiated materials characterization program. A large sample population (both irradiation creep and piggy-back specimens) will need to be exposed to the expected NGNP PMR design in the AGC experiment. If property changes within graphite for higher doses are required, then a second, high-dose irradiation experiment will be implemented.

Generally, at doses below turnaround (0 – 6 dpa for NGNP graphite grades) in the normal operating temperature regime expected for NGNP (~1000 – 1200°C), both compressive and tensile irradiation creep rates are similar. As a consequence, conducting irradiation creep with a compressive load should yield the same response as in a tensile stress. This assumption is true until turnaround occurs. Since turnaround is a function of both temperature and dose (dpa), those graphite types exposed to higher temperatures will experience turnaround at correspondingly lower doses.

After turnaround, graphite loaded in tension enters into a non-linear (tertiary) creep regime where the creep rate is significantly increased (c-axis growth and pore formation). Tensile stresses either promote or at the very least allow unhindered strain relief during irradiation, providing a “worst-case” creep rate for the graphite types exposed to higher doses (see Figure 11). Compressive loads, after turnaround, will tend to retard the creep rate and effectively delay the tertiary creep regime. Therefore, once turnaround has been achieved, graphite samples should be in a tensile stress state to determine the fastest rate of irradiation creep possible within the graphite.

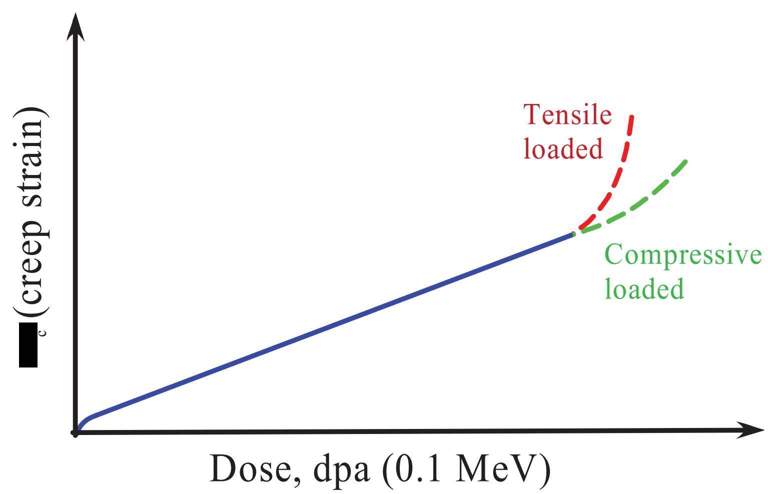

Figure 11. Schematic illustration of tensile and compressive loading on tertiary creep response of graphite.

5.1.4.4 AGC Experiment

The AGC experiment is designed to provide irradiation creep rates for moderate doses and higher temperatures of leading graphite types that will be used in the NGNP reactor design. The experiment is designed to induce irradiation creep within the secondary regime, thus allowing the graphite to be compressively loaded during irradiation, which simplifies the experiments considerably. Static compressive loads of 14.5 – 20 MPa (2-3 Ksi) are applied to the graphite during irradiation. The temperature and dose regimes covered by the AGC experiment are illustrated in Figure 7.

As shown, the dose is intentionally below (or possibly at) the point of turnaround within the graphite at the normal NGNP operating temperatures. Only the AGC-6 experiment (6 – 7 dpa at 1200°C) will approach expected turnaround limits for current NGNP graphite types. Since the AGC experiments are a comparison measurement between stressed and unstressed irradiated specimens, if turnaround were to occur during AGC-6 exposure, the creep rate results would be affected by the compressive loading state of the graphite.

Results from the HTV experiment will provide both turnaround and high temperature irradiation data for all selected graphite types. Turnaround data from these experiments will be used to adjust the exposure, loading, and temperature limits for AGC-6 to extrapolate as much accurate information from it as possible.

5.1.4.5 High Dose Irradiation Creep

As noted previously, the pebble-bed design expects the facing reflector blocks (inner and outer reflector) to operate at much longer times and thus withstand a maximum of irradiation damage before the core is shutdown, de-fueled, and the blocks replaced. Current expectations are for the reflector blocks to operate approximately 20 – 25 years before replacement. At the higher end of the dose range noted above for a, NGNP pebble-bed design, this can correspond to as much as 25 dpa before change out of the reflector blocks. While this appears to be a very large dose, the expected temperature ranges are lower than for a prismatic design, resulting in longer turnaround times and slower irradiation creep rate.

If a pebble-bed design is selected, the creep rate and resultant strain from these higher doses must be determined for accurate lifetime predictions. This will require an extensive design development program to determine an optimal tensile loading configuration that can withstand long-term exposure (i.e., four years within ATR or 2.5 years within the High-Flux Isotope Reactor [HFIR]). In addition, sample size, geometry, and matrix size will need to be considered to determine the most advantageous MTR to use for this experiment.

One benefit is that only one graphite type will be required for these tests since the NGNP pebble-bed design is currently interested in only a single graphite type. Thus, the sample matrix can be significantly reduced allowing multiple MTRs to be considered. However, similar to the AGC experiment, the test temperatures, fluences, and tensile loads must be constant during the test.

Elastic Constants and Strength Testing. Standard strength testing techniques using stress-strain (σ-ε) curve relationships will provide the bulk of the mechanical material properties. Extensive testing programs for both non-irradiated and irradiated graphite samples will be necessary to (1) prove consistency between billets and lots of graphite, (2) provide baseline material property data, and (3) quantitatively demonstrate the material property changes as a result of exposure to a HTGR environment.

ASTM test standards call for specimen sizes to have cross-sectional diameters of at least 5X the grain size across the stressed gauge section of the sample to provide representative and repeatable testing results. Traditional practices tend to use a minimum of 10X of the maximum grain size for the cross-sectional diameter of the gauge section. Thus, for a typical large-grained graphite (i.e., NBG-18) with a maximum grain size of 1.6 mm, the test gauge section will need to be at least 16mm across to provide accurate mechanical property values. Most other graphite types have considerably smaller grain sizes and may use smaller gauge sections.

This imposes a minimum sample size that in many cases may be too large for most MTRs. The ATR facility can and will be used to accommodate the larger-sized specimens, but multiple reactors are anticipated to meet all the irradiation needs. This may force the use of much smaller specimens that are not included in standard ASTM testing methods. A program to develop miniature test specimens for irradiation testing will need to be developed for graphite. This is similar to the on-going miniature sample development for irradiated metal samples. In either case, careful determination of the optimal sample size is required for each mechanical test before irradiation.

Specialized grips are required for the testing of graphite, as specified in the test standards. This is especially true for miniature test specimens that will require specialized fixtures to accommodate the small size. These specialized fixtures must be identified and developed prior to mechanical testing activities.

Finally, some elastic constants will be obtained using non-destructive UT methods. Standard testing procedures will be used to obtain dynamic elastic modulus values (tensile and compression) for specific samples.

Table 4. Mechanical properties

Property	**Test Standard**
Static and dynamic elastic modulus	ASTM C747-05, Standard Test Method of Elasticity and Fundamental Frequencies of Carbon and Graphite Materials by Sonic Resonance
Poisson's ratio	ASTM C747-05, Standard Test Method of Elasticity and Fundamental Frequencies of Carbon and Graphite Materials by Sonic Resonance
Strength values	
σflex	ASTM C1161-02c, Standard Test Method for Flextural strength of Advanced Ceramics at Ambient Temperature
σtensile	ASTM C749-02, Standard Test Method for Tensile Stress-Strain of Carbon and Graphite
σcompression	ASTM C695-05, Standard Test Method for Compressive Strength of Carbon and Graphite
Strain to failure	ASTM C565-93, Standard Test Method for Tension testing of Carbon and Graphite Mechanical Materials
Fracture toughness : (KIc, GIc, σf)	Under development
Multi-axial failure criteria	Under development

Tribology (wear/friction) . Standard "pin-on-wheel" wear testing procedures will be used to determine wear, friction, and dust generation values for selected grades of graphite. Additionally, previously irradiated and oxidized graphite will be subjected to similar tests to determine any changes. These will be limited studies focused on those graphite types of interest to pebble-bed designs (i.e., NBG-18).

5.2 Multi-scale Model Development

Models are required to allow the designer to assess the condition of graphite components and core structure design margins at any point in the lifetime of the reactor. The models are needed to describe interactions between graphite components, specifically, the behavior of the stack of graphite blocks making up the core moderator and reflector. Specific models should be able to calculate external loads imposed upon the components, internal stresses resulting from radiation and temperature induced dimensional changes, movement of components (i.e., dimensional clearances for control rod insertion), and estimates of residual strength both with and without environmental attack (i.e., air-ingress during off-normal event).

Modeling the behavior of a graphite core is complex and will require some fundamental understanding of the graphite physical, thermal, and mechanical behavior as a function of irradiation temperature and neutron fluence. However, the primary objective of these models is to provide the ability to calculate in-service stresses and strains in graphite components and estimate the structural integrity of the core as a whole. Thus, understanding of fundamental mechanistic material behavior during operation will be limited to those aspects required to understand the response of the entire core both during normal operation and during off-normal events (e.g. predict seismic behavior of the core). A physics-based understanding of microstructural damage and its effects on materials structure and properties will provide an initial start to estimating the amount of changes to a graphite component can be estimated but the degree of change is unique to the specific nuclear graphite grade used and these fundamental principles must be supplemented with actual experimental material property data to provide a complete analysis of the core behavior.

For example, the existence of temperature and flux gradients within the core and individual components will generate differential changes in dimensions and, hence, stress. Such stresses will creep out (relax) at the expected temperature and fluence levels experienced during normal operation. In addition, stresses arising due to thermal gradients will also creep out during operation, but will reappear in the opposite sense when the core cools during reactor shutdown. To model the core and component stresses during operation (and cool down), the change in properties of the graphite as a function of temperature and neutron dose must be known. Since the point-to-point flux and temperature data will not be available for all combinations of dose and temperature for the required properties, behavioral models are required to estimate the stress states for all components throughout the core. Thus, the whole core models must use a combination of experimentally derived material properties underpinned by an understanding of the fundamental physics to account for all variations possible within the graphite components of the core. Consequently, a major goal is the development and validation of multi-scale models for the behavior of graphite, core components, and whole graphite cores for use in licensing and continued operational safety assessments.

5.2.1 Whole Graphite Core and Component Behavior Models

Finite element models (FEM) are required to define the core condition at all times during core life. Such models will take core-physics and thermo-hydraulics inputs for point dose and temperature values and apply graphite material behavior models to calculate the changes in properties with neutron dose, temperature, and oxidative weight loss. Core and component-scale models will allow designers to predict core and core block (e.g., reflector or fuel element) dimensional distortion, component stresses, residual strength, and probability of failure during normal or off-normal conditions.

Finite element based codes such as COMSOL™ and ABAQUAS™ offer platforms upon which the desired whole core/component behavioral models may be assembled. It is anticipated that reactor vendors will have their own custom codes to describe and predict the behavior of the core within their particular design for NGNP. However, independent validation of these whole core-scale models may be requested by the NGNP project to ensure the safety envelope of the core during normal and off-normal operating conditions.

Finally, the development and utilization of such codes is an integral part of the design process and is recognized as such by the ASME graphite core components design code, which is currently under preparation by a sub group of ASME B&PV Sect III (nuclear). The sub-group is currently bench-marking core component stress models against a standard set of problems (data sets). Additional validating data for the developed models will come from large multiaxial load specimen testing and, ultimately, from full-scale core components tests.

5.2.2 Macro-scale Materials Behavior Models

Materials behavior models are needed to predict the effects of temperature, neutron dose, and oxidation weight-loss on key physical and mechanical properties. The material behavior results from these models are validated through an extensive program of non-irradiated and irradiation characterization experiments. The properties of interest include:

- CTE and thermal conductivity, specific heat
- Strength (tensile, compressive, flexural)
- Fracture behavior
- Elastic constants (Young's modulus, shear modulus, Poisson's ratio)
- Creep coefficient(s).

The material property models must also take into account the interaction of effects (e.g., neutron damage and weight loss) and the interdependency of effects (e.g., effects of stressed dimensional change [creep]) on the physical properties of graphite. Materials models must be physically based (i.e., based on the materials structural changes) and should incorporate structural damage models and existing physics-based models (e.g., phonon conduction). Existing material property models (in some cases empirically derived models) must be evaluated and new or improved materials behavior models developed. Material property values needed for validation of these models will be obtained from the experimental characterization research, as described above (i.e., AGC experiment, non-irradiated property characterization, and possibly high-dose experiments).

Particular emphasis will be placed on this aspect of the multi-scale modeling as it is most directly applicable to the NGNP R&D program. Individual vendor designs are expected to significantly influence the whole core-scale modeling efforts, and as such, the majority of the development effort for whole core-scale models is expected to resided with the vendors. However, as illustrated above, the material property models necessary for predicting graphite component and core behavior will be essential to developing and validating the whole core-scale models.

5.2.3 Micro/Nano-scale Models

Nano- and micro-scale modeling provides a fundamental understanding of material behavior. *Ab-Initio* models of the atomistic phenomena occurring on irradiation will allow prediction of the displacement damage that can occur and may shed light on the crystal deformation modes. Simulations (e.g., Density Function Theory) of defect structures for relevant combinations of dose and temperature can provide the basis of determining crystal strains. Understanding the physical interactions of the graphite crystallites and the inherent porosity within and around the crystallites is crucial to building microstructural models for the behavior of polycrystalline graphite. Similarly, the deformation processes that occur within the crystallites when graphite is subjected to stress, either externally applied or those that develop within the graphite due to dose and temperature gradients, must be understood and modeled.

Surprisingly, after ~60 years of graphite use in reactors, the microstructural mechanism of irradiation creep and crystal deformation are still being questioned and are not fully elucidated. Recent fundamental studies by Heggie, et al (University of Sussex Group, UK), have suggested displacement damage structures previously considered improbable are indeed energetically favorable, indicating the need for further study. Crystallite damage observations using Transmission Electron Microscope (TEM), coupled with Scanning Electron Microscipe (SEM) and CT studies of irradiated graphite will provide mechanistic data for structural models.

As indicated above, development of nano- and micro-scale models will underpin the macro-scale materials property models, as well as provide valuable input for experimental validation requirements. However, fundamental studies and micro-scale modeling should be supportive of the material property models to enable a basic understanding of the mechanisms driving the material property changes. While important, less direct emphasis will be placed on complete development of these nano- and micro-scale models than on the material property models discussed previously. Some support will be required to fully understand the underlying principles that induce changes to the material properties, but the majority of the work will be left to long-range research funding sources.

6. Costs and Schedule

Experimental testing and data collection are considered to be the largest costs for the graphite R&D program. As indicated in Sections 4 and 5, the list of required material properties are fairly extensive and the irradiation testing program rather long. The activities supporting licensing (i.e., development of whole core models, ASME code case development, and NRC licensing reviews) are assumed to be less time and cost intensive; however, the exact activities are less defined leading to uncertainty for appropriate budgets. As a consequence, the costs are broken into two areas: experimental data collection and licensing support.

6.1 Data Collection Costs

The costs for support activities, such as QA, sample procurement/fabrication, and pre-irradiation tasks, in addition to the actual testing programs, are discussed as well. The experimental work is further divided into non-irradiated and irradiated tasks to better reflect the development plan in Section 5.0. A brief description of the identified activities and the estimated costs are shown in Table 5.

Table 5. Estimated costs for graphite R&D

Activity	Estimated Costs	Comments
	Experimental Testing	
Procurement of graphite lots for sample fabrication Statistical char. Irradiated testing	$250k per lot (10 – 15 billets) 4 lots per graphite $1M per graphite	PBMR was able to use some of the graphite billets from each lot to fabricate components for their first core. This cost saving occurs only for the graphite type selected for the NGNP reactor.
Source qualification	$1 – 2M	1. Will establish the requirements for source qualification (qualify other coke sources for additional graphite production) in future cores. 2. Will be required for design certification over lifetime of reactor if new graphite is used. 3. Actual qualification of new coke/graphite sources will have costs similar to the current NGNP graphite development program but are not included here..
Statistical thermo-mechanical characterization	$3 – 4M per graphite	Non-irradiated material property database. Includes machining costs, all testing (including multi-axial), and data analysis.
AGC irradiation capsule design and review	$5M	It is assumed that the approval costs for future AGC capsules will be significantly reduced once the generic design for all AGC capsules has been approved in FY-08.
Preparation for PIE of irradiated graphite samples	$5M	To meet NGNP schedule requirements, the Idaho National Laboratory (INL) will need to modify existing laboratories to facilitate PIE of graphite irradiation samples in parallel with Oak Ridge National Laboratory (ORNL).

Activity	Estimated Costs	Comments
AGC experiment – Irradiation	$4M per 3-dpa capsule $8M per 7-dpa capsule $36M total	Nominal review and approval of new capsule design, construction costs, irradiation (neutrons) and monitoring costs
AGC- experiment – PIE	$3.5M per capsule $21M total	All irradiated physical, thermal, and mechanical testing to be performed for each graphite.
HTV 1&2 design and approval	$2M	Includes design, approval, and construction costs for these simpler "drop-in" capsules. Neutron costs will be minimal since these tests are in HFIR.
HTV 1&2 – PIE	$2M	Physical, thermal, and mechanical testing of these un-loaded specimens
Oxidation studies	$2M	Both development of ASTM test standards for oxidation testing of nuclear graphite as well as determining oxidation rates of non-irradiated and irradiated graphite.
Baseline experimental costs (prismatic reactor design)	**$78 – 80M**	1. This is the estimated experimental expense for qualifying a graphite type for use within a prismatic reactor design (the selected baseline design). 2. Additional costs for a pebble-bed design are included at the end of this table.
	Design Validation	
Micro-scale modeling	$2M	1. As in all modeling efforts this activity can be very expensive. Careful selection of specific work focused on license approval will reduce the costs significantly. 2. It is assumed the primary funding source for micro-scale modeling will be NERI type awards. 3. Whole core models in direct support of ASME code case and NRC licensing approval will be funded significantly by reactor venders.
Macro-scale modeling	$6 – 10M	
Whole core modeling	$6M	
NDE Development Pre-irradiation – as-received ISI	$5M	NDE techniques capable of characterizing the as-received graphite components before emplacement within the reactor as well as in-service inspection tools to ensure the integrity of the graphite components within the core are expected to be needed for NRC licensing.
ASTM test standards development	$4M	ASTM committee duties, standard writing, and participation in Round Robin proof testing.
ASME code case support	$3M	ASME committee duties and participation in data collection 10 years participation Two researchers from INL/ORNL

Activity	Estimated Costs	Comments
Project management	$10-12M	Includes funding for project management, quality assurance and records management activities for NQA level 1 program.
Baseline estimated design validation costs (prismatic design)	**$36-42M**	1. As stated above, modeling costs can vary dramatically. These costs are considered the minimum necessary for NRC licensing requirements. Whole core model costs will most likely be shared by vendors in support of licensing their design. 2. NDE costs are relatively unknown at this time since the scope is undefined. However, in-service inspection of graphite components is necessary for UK reactors and anticipated for PBMR and will probably be required for NGNP.
Total Baseline Costs	**$114-122M**	
Beyond Baseline Costs (Pebble-bed Design Additional)		
Procurement of graphite lots for sample fabrication Statistical char. Irradiated testing	$250k per lot 1 – 2 lots per graphite $500K per graphite	1. Costs are reduced since PBMR has currently an extensive non-irradiated materials property database for NBG-18 which is the graphite of choice for a pebble-bed design. 2. May need to order more graphite for additional characterization data to meet USA regulator requirements.
Source qualification	$1M	1. Will establish the **requirements** for source qualification (qualify other coke sources for additional graphite production) in future cores. 2. Will be required for design certification over lifetime of reactor if new graphite is used. 3. Actual qualification of new coke/graphite sources will have costs similar to the current NGNP graphite development program.
Statistical thermo-mechanical characterization	$1.5 – 2M per graphite	1. Costs are reduced since PBMR has currently an extensive non-irradiated materials property database for NBG-18 which is the graphite of choice for a pebble-bed design. 2. Non-irradiated material property database. Includes machining costs, all testing (including multi-axial), and data analysis.
High dose creep capsule – design and approval	$5M	Some cost savings using previous experience with AGC and HTV 1&2 capsule designs.
High dose creep capsule – irradiation	$8 – 12M	This experiment is 4X longer time in ATR than the longest AGC capsules. Potential savings could use HFIR since the flux in HFIR is ~ 3X higher than ATR. However, there is limited volume in HFIR and availability is at 50%.
High dose creep capsule – PIE	$5M	The much higher dose may make these graphite samples more difficult to handle and subsequently test. Additional costs will be associated.

Activity	Estimated Costs	Comments
Beyond baseline additional costs	**~ $21 – 25.5M**	1. As noted above expenses for procurement and qualification of graphite are on a "per graphite basis". This cost estimate will increase for more graphite types being tested. 2. The high dose creep experiments are valid only for long term exposure of graphite (i.e., pebble-bed reflectors). Costs may be reduced depending upon which design is selected.

As seen from the above table, total costs are a function of the number of graphite types to be investigated, the selected reactor design, and the operating parameters (i.e., temperature, dose levels) of the selected reactor. A prismatic design is used for a cost baseline with the assumption that the R&D program will change if a pebble-bed design is selected. A number of variables can adjust the costs for each reactor design. These are discussed in the sub-sections that follow

6.1.1 Prismatic

PCEA graphite has been selected for the prismatic NGNP design, but others may be considered (see *NGNP Graphite Selection and Acquisition Strategy*) adding to the overall costs. A full thermo-mechanical characterization program for all graphite types will be required.

6.1.2 Pebble-bed

NBG-18 graphite has been selected for the pebble-bed NGNP design, but others may be considered (see *NGNP Graphite Selection and Acquisition Strategy*). Graphite irradiations to 25 dpa will be required for front face of reflector blocks, adding to irradiation experiments costs. A partial thermo-mechanical characterization program for NBG-18 graphite will reduce the overall costs since PBMR has already performed significant testing in this area.

6.2 Data Collection Schedules

A preliminary schedule for all graphite work has been recently developed. This master schedule incorporates not only the irradiation schedules for AGC, as discussed previously, but also graphite procurement, non-irradiated data collection, and required ASME and NRC licensing effort timelines. This master schedule is presented in modified form in Figure 12.

As shown, the schedule does not incorporate those tasks needed for support of a pebble-bed design selection (i.e., high-dose experiments, minimal non-irradiated characterization, or adjustments to the existing irradiation program). The schedule is based on the baseline assumption of a prismatic reactor design. Once key design decisions are made by the NGNP project, the schedule and cost baseline will be updated to reflect these decisions and a more detailed resource loaded schedule will be produced.

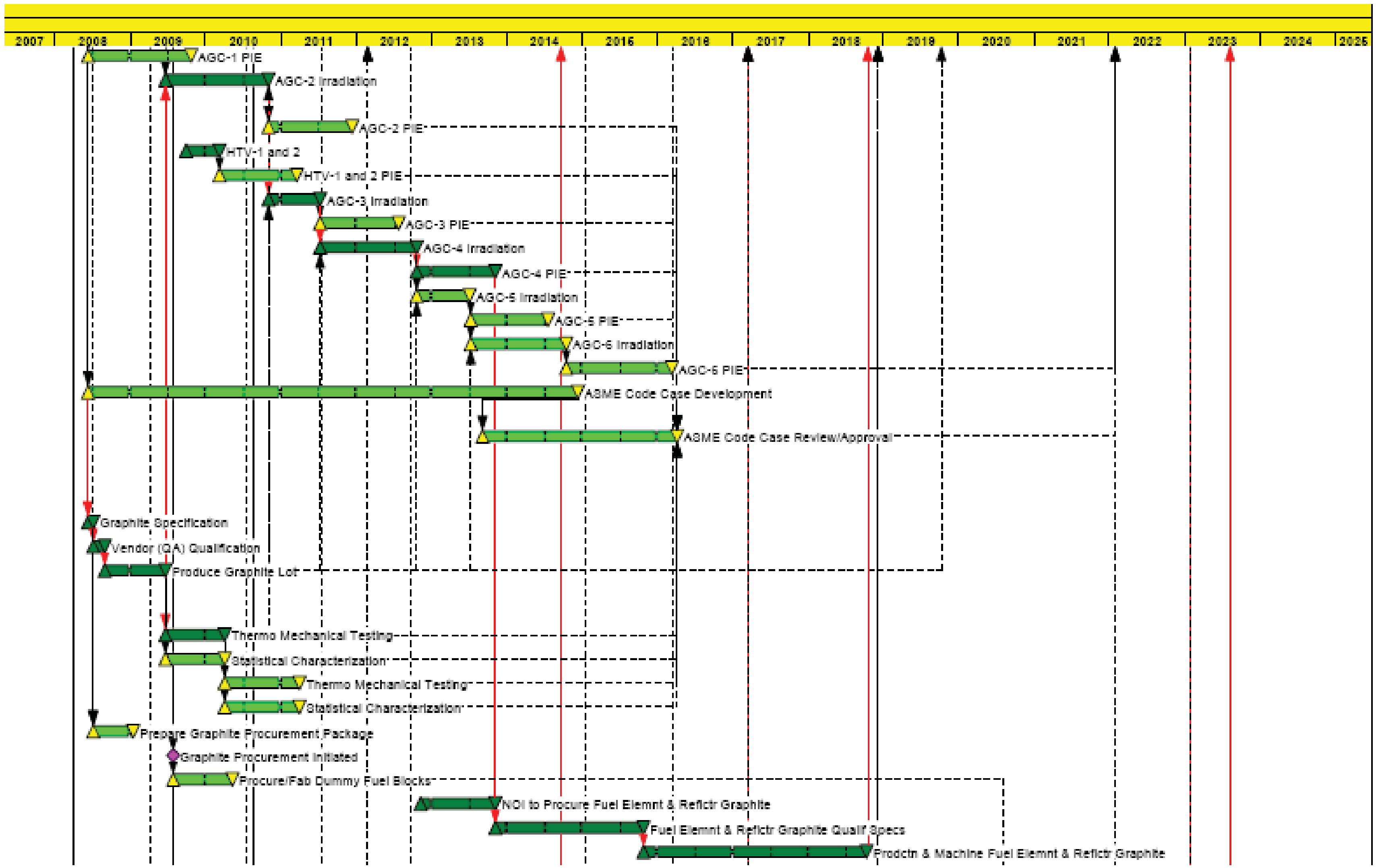

Figure 12. Schematic of master schedule for graphite R&D effort.

7. Longer Term Considerations

What is presented above is a minimal "baseline" estimate for performing the required graphite R&D for the NGNP. Some discussion of each is warranted about longer term issues that would impact the longer R&D program needed for ultimate commercialization of the HTGR technology.

7.1 Graphite Acquisition Plan

Currently, the world market share for nuclear graphite is extremely small. While graphite manufacturers are willing to produce nuclear grade graphite, the petroleum industry, which produces the raw starting material – specialty coke, is much less interested. The material specifications for specialty coke are much more exacting than what is needed for electrode production, the majority market share for graphite. Since this material's market share is so small, the coke suppliers have very little financial interest in changing their production process to enable manufacture of these small batches of specialty coke necessary for nuclear graphite production.

As a consequence, there may not be enough specialty coke material needed for initial or sustained production for nuclear graphite for HTGR applications. Obviously, this can significantly affect the graphite R&D schedule if multiple lots of graphite are required for testing and qualification (see Source Qualification comments in estimated costs, Table 5). This potential shortage of coke sources has been addressed in much more detail within the *NGNP Graphite Selection and Acquisition Strategy* report, ORNL/TM-2007/153.

For full commercialization of the HTGR graphite technology in the long term, a more complete evaluation of the processing route and raw material (e.g. coke source) constituent's influence on graphite behavior is required. The magnitude of the R&D program necessary to establish a standard nuclear grade graphite for a broad range of qualified coke sources for use within any HTGR design cannot be firmly estimated today given the current limited knowledge of the linkage between graphite fabrication, material properties, and in-reactor performance. It is anticipated that the work proposed to qualify graphite for the initial NGNP cores in Section 5 will provide the strong technical basis needed to establish a long- term graphite development and qualification program that meets this more ambitious commercialization goal.

7.2 Graphite Disposition and Recycle Options

Currently, the National Spent Nuclear Fuel Program and the Office of Civilian Radioactive Waste Management will be disposing Fort St. Vrain and Peach Bottom fuel blocks in Yucca Mountain. The C^{14} and Cl^{36} loading from these fuel blocks is insignificant compared to isotope inventories from commercial fuel's long-lived fission products and transuranics. Graphite reflector blocks from Fort St. Vrain were disposed in a lower-level radioactive landfill. Only Europe faces federal controls on C^{14} and Cl^{36} loading in graphite.

Currently, there is no federal guidance on recycling irradiated graphite. Recycling irradiated graphite will depend on a number of factors, including, the number of HTGRs (i.e., volume of graphite generated), the ability to decontaminate irradiated graphite, the performance of recycled graphite, and the total cost of recycling. It is expected that as the volume of irradiated graphite grows due to more HTGRs in operation, the cost-to-benefit ratio of graphite recycle will become more favorable.

Euratom has begun development of a decontaminating processes where the C^{14} is removed from along the grain boundaries of irradiated graphite using a heated oxygen gas. The contaminated gas is captured, and the "clean" blocks are ready for LLW disposal or possible recycling. Complete decontamination of graphite to below LLW thresholds (crushing plus chemical means) is possible but expensive. Once the graphite has been decontaminated, two recycling options are currently envisioned: (a) reuse of blocks after heat treatment to anneal out radiation damage or (b) form new blocks using reconstituted graphite material by crushing and jet milling irradiated blocks to fine powder.

Once a successful technology is developed for decontaminating graphite, the primary issue for recycling is the irradiation performance of the recycled graphite. A new qualification program will be necessary to validate the performance of this recycled graphite source, either for reuse of blocks or reconstituted material.

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