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Milestone Report on the "Workshop on Nuclear Graphite Research"

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Materials Science and Technology Division

MILESTONE REPORT ON THE "WORKSHOP ON NUCLEAR GRAPHITE RESEARCH"

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for the
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ABBREVIATIONS

AEA	Atomic Energy Authority
AGC	Advanced Graphite Capsule
AGR	Advanced Gas Reactor
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATR	Advanced Test Reactor
C/C	Carbon/Carbon
CFR	Code of Federal Regulation
CMT	Carbon Materials Technology
CRP	Cooperative Research Program
CSC	Core Structural Component
CT	Computer Tomography
CTE	Coefficient of Thermal Expansion
DSC	Differential Scanning Calorimeter
FEM	Finite-Element Modeling
GA	General Atomics
GCR	Gas-Cooled Reactor
GDM	Graphite Degradation Model
GRSAC	Graphite Reactor Severe Accident Code
GTMHR	Gas-Turbine Modular Helium Reactor
HFIR	High Flux Isotope Reactor
HFR	High Flux Reactor
HM	Her Majesty's
HSE	Health and Safety Executive
HTGR	High-Temperature Gas-Cooled Reactor
HTR	High-Temperature Reactor
HTTR	High-Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
INERI	International-Nuclear Energy Research Initiative
INL	Idaho National Laboratory
ISI	In-Service Inspection
JAEA	Japan Atomic Energy Agency
KAERI	Korean Atomic Energy Research Institute
KAIST	Korea Advanced Institute of Science and Technology
LD	Licensing Document
LWR	Light Water Reactor
MTR	Materials Test Reactor
NDE	Nondestructive Evaluation
NECSA	The South African Nuclear Energy Corporation
NERI	Nuclear Energy Research Initiative
NEUP	Nuclear Energy University Program
NGNP	Next Generation Nuclear Plant
NGRG	Nuclear Graphite Research Group
NII	Nuclear Installations Inspectorate
NIL	Nuclear Installation License
NNR	National Nuclear Regulator
NPL	National Physical Laboratory

NRC	Nuclear Regulatory Commission
NRG	Nuclear Research and Consultancy Group
ORNL	Oak Ridge National Laboratory
PB	Pebble Bed
PBMR	Pebble Bed Modular Reactor
PIRT	Phenomenon Identification and Ranking Table
PMR	Prismatic-core Modular Reactor
PR	Prismatic Reactor
PWR	Pressurized Water Reactor
RES	Office of Nuclear Regulatory Research
RSA	Republic of South Africa
R&D	Research and Development
SANS	Small-Angle Neutron Scattering
SEM	Scanning Electron Microscopy
TEM	Transmission Electron Microscopy
UK	United Kingdom
UKAEA	The United Kingdom Atomic Energy Authority
US DOE	United States Department of Energy
VHTR	Very High Temperature Reactor
WAGR	Windscale Advanced Gas-Cooled Reactor
XRD	X Ray Diffraction

SUMMARY

Here we report on information gathered during a public workshop held during March of 2009 in Rockville, MD, on nuclear graphite research, related to high-temperature gas-cooled reactors (HTGR). Oak Ridge National Laboratory organized this workshop and convened an international panel of recognized nuclear graphite specialists and regulators from the U.S.A., the U.K., Japan, and South Africa. Prior to the workshop, the panel members were provided with a series of documents describing the DOE graphite research plan and the NRC–DOE Graphite Phenomena Identification and Ranking Table (PIRT) report, along with other supporting documents for their review.

During the workshop, the expert panel assessed the status of worldwide research on nuclear graphite and held technical discussions to identify the technical gaps between the planned DOE research and the outcome of the Graphite PIRT conducted earlier. The panel deliberated on the behavior of graphite under neutron irradiation and identified technical gaps which are common to either pebble-bed design or prismatic core design. Research organizations which are currently involved in graphite irradiation and characterization activities and analytical modeling of irradiation behavior were identified.

The consensus of the panel was that NRC staff should develop a broad knowledge base in nuclear graphite technology and actively participate in the development of irradiation data, behavior modeling and interpretation, and codes and standards development, such that informed regulatory decisions can be made. The panel recommended several areas of research to address technical gaps needed for safety assessment and licensing review of HTGR designs. The panel did not attempt to prioritize these research areas or define the scope of the required research. In the future, the NRC staff should consider initiating independent research to generate technical bases in the recommended areas and, as appropriate, might enter into cooperative agreements to create and share technical information and knowledge with the organizations identified herein.

1. Introduction

During March 16–18, 2009, the Oak Ridge National Laboratory (ORNL) conducted a Category 3 public workshop[1] to assess the current status of worldwide nuclear graphite research with a panel of international experts in the subject area. The NRC sponsored this workshop under an Office of Nuclear Regulatory Research (RES) Contract, JCN: N6640. The venue for the workshop was the Legacy Hotel and Meeting Center, 1775 Rockville Pike, Rockville, MD 20852.

The objectives of the workshop were to convene an international nuclear graphite expert panel and hold technical discussions to (1) identify areas of research the NRC may initiate to provide technical safety information and data for aiding licensing decisions on the high-temperature gas-cooled reactor (HTGR) or very high temperature reactor (VHTR) for the Next Generation Nuclear Plant (NGNP) at Idaho; (2) identify graphite confirmatory research which NRC may conduct; and (3) propose paths for conducting the recommended research.

The expert panel consisted of recognized nuclear graphite specialists from universities and national laboratories from the U.S.A., the U.K., Japan, and South Africa and regulatory staff from U.K.'s Nuclear Installations Inspectorate and from National Nuclear Regulator (NNR) from South Africa. A representative from China was initially invited to be a member of the panel but was unable to attend. The panel members were as follows:

1. Dr. Timothy Burchell, Group Leader, Carbon Materials Technology Group, ORNL, U.S.A.
2. Dr. William Windes, Leader, Graphite Group, Idaho National Laboratory (INL), U.S.A.
3. Dr. Robert Bratton, NGNP Graphite Group, INL, U.S.A.
4. Mr. Scott Penfield, Technology Insights, San Diego, CA, U.S.A.
5. Dr. Robert Wichner, ORNL Consultant, U.S.A.
6. Mr. Mark Mitchell, Leader Materials Group, Pebble Bed Modular Reactor (Pty) Ltd., Republic of South Africa (R.S.A.)
7. Mr. Schalk Doms, Senior Regulatory Officer, PBMR Programme, National Nuclear Regulator (NNR), R.S.A.
8. Dr. Motokuni Eto, Technical Consultant, Toyo Tanso Co. Ltd., Japan
9. Professor Barry Marsden, School of Mechanical, Aerospace and Civil Engineering, The University of Manchester, U.K.
10. Mr. Graham Heys, HM Principal Inspector (Nuclear Installations), HM Nuclear Installations Inspectorate, Health and Safety Executive, U.K.

Dr. Nidia Gallego, Research Scientist, Carbon Materials Technology Group, ORNL, the principal investigator of the RES contract, coordinated the workshop arrangements. Dr. Makuteswara Srinivasan, Senior Materials Engineer, NRC, acted as the overall facilitator of this workshop.

During the first day of the workshop, the panel members presented technical information on past and present nuclear graphite research, which included the following general topics:

(1) a short history of relevant nuclear graphite research sponsored by the NRC's Office of Nuclear Regulatory Research and previously conducted by national laboratories;
(2) the status of worldwide research on nuclear graphite and its applicability to the design review of the NGNP;

[1] http://www.nrc.gov/reading-rm/doc-collections/commission/policy/67fr36920.html

(3) the international regulatory practices in licensing and regulating graphite-moderated gas-cooled reactors; and
(4) the results of the NRC–DOE PIRT exercise on nuclear graphite, which was conducted in 2007.

The second day of the workshop began with a presentation by NRC staff on some of the challenges in assessing the structural integrity of graphite components, which was followed by a presentation by ORNL staff on technical issue gaps between the PIRT-identified data and information needs and the research proposed in DOE's NGNP research plan on graphite. The NRC staff then provided the panel with background information and ground rules for discussions by the panel on previously identified technical areas.

During the morning of the third day of the workshop, the expert panel continued with the discussion of the topics. The workshop coordinator, Dr. Gallego, also provided the expert panel with final assignments to prepare a draft report.

A detailed summary of the workshop, including summaries of presentations, questions and answers, panel deliberations, biographies of panel members, attendee list, etc., is provided in Attachment 1, and individual panel member technical presentations are provided in Attachment 2.

In this letter-type technical report, we provide a background for this research in Section 2, followed by the recommendations from this workshop in Section 3 and 4, and a list of research organizations and universities performing nuclear graphite R&D in Sections 5 and 6, respectively.

2. Background

The next generation nuclear plant (NGNP) will be a modular HTGR of either the prismatic-core modular reactor (PMR) or pebble bed reactor (PBR) designs. Various energy utilization systems are being considered, including direct- and indirect-gas turbine (Brayton) cycles for electric power production, steam (Rankine) cycles for electricity and/or process steam, and indirect-cycle systems for direct heat, such as hydrogen production.

The two (PMR and PBR) reactor designs utilize nuclear-grade graphites as construction materials for the moderator and core structures. The reactor operating temperature ranges for the two concepts are broadly similar, but the peak cumulative neutron dose for the graphite core component in a PBR is substantially greater than that in a PMR. A significant challenge for new construction HTGRs in the United States is that the previous graphite grade qualified for nuclear service in the United States, H-451, is no longer available. The precursors (raw materials) from which H-451 graphite was manufactured no longer exist. The present understanding of graphite behavior is not sufficient to enable the available H-451 database to extrapolate to the expected reactor operation conditions and currently available nuclear graphite grades.

In qualifying new grade(s) of graphite, there exists a need for a more reliable and commonly acceptable fundamental understanding of irradiated graphite behavior to develop new theories and models having a sound, in-depth, scientific basis. Such effort will provide increased confidence for design and licensing and could reduce the extent of costly experimental irradiation verification, which would be needed when additional new graphite grades are developed for the HTGR.

Because of the inherent variability in the important properties of graphite, a good understanding of the variability of the physical, chemical, mechanical, and thermal properties for a given graphite grade (within billet, between billets, and between production lots) is needed to establish properties degradation behavioral models of phenomena during reactor life. The effects of reactor environment (temperature, neutron irradiation, and chemical attack) on the physical properties must be elucidated. Finally, for each grade of graphite, the irradiation-induced dimensional changes (which drive the generation of graphite component stresses) and irradiation creep behavior (which relieves graphite component stresses) must be determined over a representative temperature and fluence range.

During early 2007, the NRC conducted a PIRT exercise, with the support of international experts, to identify those phenomena that could potentially lead to accidents which could release radionuclides outside the containment of the NGNP. The objectives of the graphite PIRT were to identify significant phenomena related to nuclear graphite performance which could affect reactor safety from the degradation of moderator and structural graphite components. The evaluation considered both routine (normal operation) and postulated accident conditions for the NGNP.

The graphite PIRT panel identified several phenomena, of which five were ranked to be of high importance–low knowledge (I-H, K-L). Nine phenomena were ranked to be of high importance and medium knowledge (I-H, K-M). Two phenomena were ranked as medium importance and low knowledge (I-M, K-L), and a further 14 were ranked as medium importance and medium knowledge (I-M, K-M). The last 12 phenomena were ranked as low importance and high knowledge rank (or similar combinations suggesting they have low priority) (I-L, K-H).

During 2009 NRC sponsored ORNL to review the PIRT findings and compare them against the current DOE research plan and identify technology gaps. The results of this review[2] (see attachment 3) and several other documents [3, 4, 5, 6, 7] were reviewed by the workshop panel prior to the workshop.

During the workshop, the panel analyzed the results of the graphite PIRT and compared the currently postulated DOE research to identify gaps in needed research, which could be addressed in the future by research sponsored and conducted by NRC. After technical presentations by various panel members and deliberations around previously identified technical areas, the workshop panel revisited the ORNL findings [2] and identified and recommended several areas for NRC's future graphite research.

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[2] N. C. Gallego and T. D. Burchell, *Comparison of NRC Graphite PIRT and DOE Planned Research Activities for Graphite*, ORNL/NRC/LTR-09-01 (2009).
[3] T. D. Burchell, *Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRT), Volume 5: Graphite PIRTs*, NUREG/CR-6944 Vol. 5 (2007).
[4] W. Windes, R. Bratton, and T. Burchell, *Graphite Technology Development Plan,* INL/EXT-07-13165 (2007).
[5] T. Burchell, R. Bratton, and W. Windes, *NGNP Graphite Selection and Acquisition Strategy*, ORNL/TM-2007/153 (2007).
[6] T. D. Burchell, D. Heatherly, J. McDuffe, D. Sparks, and T. Thoms, *Experimental Plan and Final Design Report for HFIR High Temperature Graphite Irradiation Capsules HTV-1 and -2,* ORNL-GEN4/LTR-06-019 (2006).
[7] W. R. Corwin, *Generation IV Reactors Integrated Materials Technology Program Plan: Focus on Very High Temperature Reactor Materials*, ORNL/TM-2008/129 (2008).

3. Panel Recommendations

The panel had several general recommendations and also identified specific areas of graphite research addressing phenomena affecting graphite behavior in a reactor environment (see Section 4). The consensus of the panel was that NRC staff should develop a broad knowledge base in nuclear graphite technology and actively participate in the development of irradiation data, behavior modeling and interpretation, and codes and standards development, so as to be able to make informed regulatory decisions. Therefore the NRC staff should consider the following:

- maintaining and developing additional expertise in graphite technology and irradiation behavior through engagement with the scientific community;

- actively participating in design, construction, and inspection codes and standard development;

- developing independent confirmatory behavioral models and structural analysis codes; and

- supporting research programs to address the identified technical gaps.

4. Suggested Areas for Research Identified by Panel Members

The panel identified 10 specific areas where NRC-sponsored research would enable the generation of specific information which would provide the technical bases for evaluating HTGR designs for certification and licensing. The panel did not attempt to prioritize these research areas or define the scope of the required research. The panel also did not provide an opinion on the urgency of these research areas. The panel was not unanimous in the depth and breadth of needed research in any of these areas to address regulatory concerns, and recommended that any research effort be predicated by a review of existing literature to scope the require R&D effort. These specific technical research areas are addressed below.

4.1 Oxidation models

A comprehensive modeling capability of graphite oxidation is needed, which should include both oxidation kinetics and diffusion behavior of species within the graphite structure. The goal is to predict oxidation behavior of large graphite component blocks during normal operation conditions and different accident scenarios. This effort requires two tasks. The first is to collect detailed experimental information on oxidation rates measured at laboratory scale (with small size specimens) for representative graphite materials. These measurements must be conducted under conditions where perturbations caused by oxidant diffusion within the pores of graphite and mass transfer at the gas-solid interface are negligible. The second task involves the measurement of effective diffusivity of the oxidant in the porosity of graphite for pristine graphite materials (not oxidized) and for graphite oxidized to various levels. These measurements should clarify the dependence of effective diffusivity of the oxidant on graphite structure and the effect of oxidation-induced gradual structural changes (opening of porosity). With these two sets of information available, modeling should first be able to reproduce oxidation rates and structural variations observed on tests of small size specimens; in a second stage, modeling should have the capability to predict the oxidation behavior of large graphite blocks.

4.1.1 Oxidation under normal operation conditions

For an assessment of the long-term behavior of graphite components under *normal operating conditions,* especially for the core support structure, specific oxidation rate data are needed for the selected graphites for oxidation by H_2O (moisture) and O_2. The oxidation rate constants must be measured under carefully selected conditions, such that the perturbing effects of diffusion and mass transfer are negligible.

There is a great need for information on oxidant penetration (either H_2O or O_2) under normal operating conditions in order to understand the diffusion behavior of the governing species in graphite. This is not a traditional area of study, and consequently there is a major uncertainty in the data, interpretation, and its application to assessing the VHTR graphite performance. Penetration depth depends on both the effective diffusivity of the oxidant and the intrinsic chemical reaction rate.[8,9,10] Thus, it is highly dependent on temperature. Theory indicates that oxidant penetration and consequently the oxidation rate is significantly reduced by the high pressure typical of normal operation. It would be a great advantage to take full benefit of this effect for safety evaluations. A few measurements of this effect have been made in the U.S.A. (at General Atomics), in the U.K. (Dragon project), and reportedly in Germany. During early 1980s, the NRC sponsored research at Brookhaven National Laboratory to study the effect of load during oxidation.[11] Both oxidation penetration depth and long-term oxidation affect lifetime characteristics. In these circumstances, perhaps the best approach would be a review of the theoretical aspects of pressure effect on oxidation and a comparison with reported data.

4.1.2 Oxidation under accident conditions

In order to develop oxidation modeling capability under *accident conditions*, the oxidation rate constants need to be determined as well as oxidant penetration information, in particular for two categories of cases of major accident scenarios, namely air ingress and steam ingress (for the case of steam generator concepts). Such measurements should be done in the range of projected temperature/time/pressure conditions.[12] Laboratory-scale rate constant experiments are needed as well as larger scale validation tests of larger portions of the core exposed to air at high temperature. Finally, oxidation rate and oxidation penetration measurements need to be combined with model tests to determine structural strength loss as a result of long-term oxidation under normal operating conditions. Some theoretical and experimental work concerning graphite corrosion during accidents in various scenarios is available from U.S. and German sources, for both pebble bed and prismatic fueled reactors. The applicability of such work for the required accident case analyses needs to be determined.[13]

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[8] R. P. Wichner, *Effect of steam corrosion on core post strength loss: Part I. Low ingress rates*, ORNL TM-5534 (1976).
[9] R. P. Wichner, T. D. Burchell, and C. I . Contescu, *Note on graphite oxidation by oxygen and moisture*, ORNL/TM-2008/230 (2008).
[10] R. P. Wichner, T. D. Burchell, and C. I. Contescu, "Penetration depth and transient oxidation of graphite by oxygen and water," *J. Nucl. Mater*. (in print, 2009) doi:10.1016/j.nucmat.2009.06.032.
[11] M. Eto and F. B. Growcock , *Effect of prestress and stress on the strength and oxidation rate of nuclear graphite*, NUREG/CR-2316 (1981).
[12] R. P. Wichner, *Effect of steam corrosion on HTGR core post strength loss: Part II. Steam generator tube rupture event*, ORNL/TM-5580 (1977).
[13] R. Moormann, "Phenomenology of graphite burning in massive air ingress accidents" Proc. HTR2006, 3rd Intl. Meeting High Temp. Reactor Technology, Johannesburg, South Africa (2006).

4.1.3 Current effort and future needs

The subject of graphite oxidation kinetics is being actively researched in the U.S.A., primarily under the auspices of the DOE for the HTGR version of the NGNP[14] and also for the development of new American Society for Testing and Materials (ASTM) standards.[15] A new ASTM standard for characterization of air oxidation of graphite specimens with standard size and shape has been verified in inter-laboratory tests and has been approved by ASTM International.[16] Designed mostly as a means for qualification and selection of graphite materials, the new standard method is not suitable for an analysis of material-specific factors which determine intrinsic oxidation rates and the effective diffusivity of the oxidant. It is believed that, with some modification, the new standard may be adapted for reliable determination of intrinsic kinetic constants (not perturbed by diffusion) for graphite and carbon-based materials.

Experimental data on the oxidation of each candidate graphite for NGNP core components are indispensable, although the general tendency of oxidation behavior of nuclear graphite has been relatively well-known [17,18,19,20,21,22] A systematic survey of both analytical and experimental work on the oxidation of HTGR graphite, which has been carried out in the past, would help expedite the plan for the oxidation experiment on the candidate graphites.

The panel observed that NRC may conduct a literature search initially and develop a white paper on areas that are open and need additional research. This literature search would review oxidation that occurs during normal operation as well as under accident conditions. The NRC should have their own modeling capability for oxidation which includes coolant impurities and develop code which provides an active capability to evaluate oxidation.

Furthermore, because the structure of graphite is considerably affected by irradiation, from a regulatory perspective, none of the previous studies have addressed the oxidation of irradiated graphite, which is of importance in potential air-ingress and steam-ingress scenarios.

To develop the appropriate oxidation modeling capabilities, the following steps are suggested:

- Systematic measurement of intrinsic kinetic rate constants (not perturbed by diffusion) of selected graphite materials during oxidation with O_2 and with H_2O. A standard method should be selected and used, which would ensure that the results are not affected by diffusion of

[14] C. H. Oh, *Development of Safety Codes and Experimental Validation for a VHTGR*, INL/EXT-06001362 (2006).
[15] C. I. Contescu, S. Azad, D. Miller, M. J. Lance, F. S. Baker, and T. D. Burchell., "Practical aspects for characterizing air oxidation of graphite", *J, Nucl. Mater*. **381**, 15–24 (2008).
[16] ASTM D-7542-09, "Standard test method for air oxidation of manufactured carbon and graphite in the kinetic regime."
[17] M. Eto, T. Kurosawa, H. Imai, S. Nomura, and T. Oku, "Estimation of graphite materials corrosion with water vapor in coolant of the VHTR and oxidation effect on the materials properties," in *Graphite component structural design*, IAEA IWGGCR-11 (1986), http://www.iaea.org/inisnkm/nkm/aws/htgr/abstracts/abst_iwggcr11.html
[18] S. Nomura, T. Kurosawa, M. Eto, and T. Oku, "A graphite corrosion rate equation under high concentration water vapor in helium and an estimation of the VHTR support post corrosion and strength," in *Graphite component structural design*, IAEA IWGGCR-11 (1986), http://www.iaea.org/inisnkm/nkm/aws/htgr/abstracts/abst_iwggcr11.html
[19] M. Okada and T. Sogabe, "Behavior of gas desorption and gas permeability of carbon materials," in *The status of graphite development for gas cooled reactors*, IAEA TECDOC-690 (1991), http://www.iaea.org/inisnkm/nkm/aws/htgr/abstracts/abst_24041368.html
[20] S. Nomura et al., "Relation between gasification rates and gas desorption behavior with metallic impurities of carbon and graphite materials for the HTTR," in *The status of graphite development for gas cooled reactors*, IAEA TECDOC-690 (1991), http://www.iaea.org/inisnkm/nkm/aws/htgr/abstracts/abst_24041368.html
[21] M. Eto and F. B. Growcock, *Effect of oxidizing environment on the strength and oxidation kinetics of HTGR graphites*, NUREG/CR-2480 BNL-NUREG-51493 (1981).
[22] M. Eto and F. B. Growcock, *The effect of pretreatment on the initial reaction rates of PGX and H451 graphites with H_2O and O_2*, NUREG/CR-2315 BNL-NUREG-51447 (1981).

oxidant in the graphite body. Initial steps are being made by ORNL and PBMR (Pty) Ltd in a collaborative project (work in progress).

- Measurements of effective diffusivity of oxidants of interest (O_2 and H_2O) in pristine graphite and graphite with various levels of oxidation. The experimental range should cover the temperature conditions specific for normal operation, potential transients, and for accident conditions.
- Experimental work for understanding the effect of oxidation on development of porosity in graphite accompanied by changes in existing pore configuration would be helpful in rationalizing the observed penetration of oxidant, and the effect on mechanical properties.
- Research, or at a minimum a theoretical and literature review, on the effect of irradiation on oxidation behavior by H_2O and O_2.
- Development of a code is suggested, with the capability to use the results of measurements on small size specimens, for prediction of behavior of large structural blocks under a broad set of conditions, significant for normal operation, transients, and various accident scenarios.
- There is a need to verify current predictions of CO production rates during air or water ingress accident conditions and possible accumulation to flammable concentrations. There have been several studies of CO production due to oxidation by O_2. However, because of its importance, and the possible uncertainties in the current model, the issue should be re-evaluated and extended to oxidation by H_2O.
- Formation of dust by preferential oxidation of the binder component of graphite under accident conditions is insufficiently studied.

4.2 Graphite codes and standards

Currently, various testing standards are used to obtain data on small irradiated graphite samples (which are used as a necessity). These standards need to be formalized, including standardized and preferred methods of recording and presenting statistical data. However, several potentially important aspects need to be considered in estimating the behavior of the large graphite blocks in the reactor from the small-specimen material test reactor data. First, the methods to extrapolate irradiation data obtained from small (non-conforming to ASTM standard sizes) specimens should be validated and verified independently by using the same sizes under non-irradiated conditions. Any deviations could then be scaled with model parameters, which could perhaps be adapted to irradiation property determination. Second, these methods should be verified and validated using ASTM round-robin test protocol to establish the envelope for accuracy and sufficiency in the specimen test population. Third, the degree of correspondence between the data obtained using fast flux neutrons for irradiation in the material test reactor and the thermal neutrons in the power reactor should be established. Fourth, the degree of correspondence between the irradiation data obtained on the same graphite specimens between different material test reactors should be established to utilize the overall composite data, if possible, for both historical behavioral model and predictive model.

In the U.K., reviews of the Magnox and advanced gas reactor (AGR) testing techniques were recently carried out by the National Physical Laboratory (NPL).[23,24] The techniques developed for using data to predict properties under reactor operating conditions such as Young's modulus, thermal conductivity, and the coefficient of thermal expansion[25] need to be further developed, validated, and included in the appropriate standards. Attempts have been made to understand

[23] R. Morrel, *A review of UK testing methodologies for AGR Trepanned graphite core samples*, NPL Report MATC (D) 134 (2003).
[24] R. Morrel., *A review of UK testing methodologies for AGR Trepanned graphite core samples*, NPL Report MATC (D) 180 (2003).
[25] D. K. L. Tsang, B. J. Marsden, S. L. Fok, and G. Hall, "Graphite thermal expansion relationship for different temperature ranges," *Carbon* **43**(14), 2902–2906 (2005).

failures in graphite in terms of the Weibull modulus,[26] but more data and analysis is required in this area, particularly for irradiated graphite samples.

The Atomic Energy Society of Japan, in response to a request from the Japan Atomic Energy Agency (JAEA), organized a special committee to work on the development of Codes and Standards for Graphite Core Components, and a draft document was issued in April 2008. Based on the experiences and knowledge acquired during the design and construction of the high-temperature engineering test reactor (HTTR), a revised Codes and Standards for Graphite Core Components for the Future HTGR was drafted in March 2009.

The development of industry standards to allow current and future HTGR stakeholders to capitalize on the significant international experience on gas-cooled reactors should be a general priority. Key areas of interest are as follows:

- Material specifications (ASTM)
- Material test standards (ASTM)
- Guidelines for completion of material characterization (including irradiation testing)
- Design and construction standards (ASME)
- Standards providing guidance for the use of the materials in the reactor systems such as safety classification and system design requirements, possibly including a standard for damage tolerance assessment

The availability of broad-based consensus standards, which include regulatory involvement, is key to the acceptance of the graphite component fabrication technology and the bases for design of components.

The panel observed that consistent and continued active participation of the NRC staff in working group and subgroup technical committees of the codes and standards development organizations is very important and necessary for a variety of reasons, which include providing general guidance and direction in structuring the efforts to address regulatory expectations; providing general technical expertise to define consensus performance and failure criteria for graphite core components, flaw acceptance criteria, and disposition methods development; and addressing the specific nuclear-related quality assurance requirements for component inspection prior to installation and during reactor operation. Additionally, collaboration and coordination should be extensively pursued among the various countries towards the development of "global standards."

4.3 Tribological behavior of graphite

Graphite is not inherently lubricious (slippery) yet exhibits well known lubricious (slippery) behavior in the presence of air and common vapors.[27] Graphite is thus widely used as a dry lubricant to reduce friction between contacting surfaces in such environments. Graphite tribology, however, is fundamentally modified in the absence of air, most notably so in vacuum and in the dry helium environment of HTGRs. The abrasion of graphite blocks on one another, of the fuel pebbles against themselves, and of the fuel pebbles against the graphite moderator blocks can produce graphite dust. An assessment of dust formation, the possibility of pebbles sticking to each other, and resulting potential blocking of channels should be derived from detailed studies

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[26] B. C. Mitchell, J. Smart, S. L. Fok, and B. J. Marsden, "The mechanical testing of nuclear graphite," *J. Nucl. Mater*. **322** (2–3), 126–137 (2003).
[27] B. K. Yen, B. E. Schwickert, and M. F. Toney, "Origin of low-friction behavior in graphite investigated by surface x-ray diffraction," *Appl. Phys. Lett.* **84** (23), 4702-4704 (2004).

on the tribological behavior of graphite, as a function of environment, pressure, temperature, and dose.

The DOE plan includes the study of graphite tribological properties only at ambient conditions, and there is no mention of understanding the effect of the helium environment, high pressures and temperatures, and/or irradiation dose. In addition, the issues of dust formation and sticking of pebbles and the consequences of the events are not addressed in the research plan. Limited literature exists[28, 29] on this subject, mostly from the past German program. For example, it was observed that about 3 kg of dust was produced in AVR per operating year.[30] Dust generation should be evaluated, especially from the perspective of the graphite dust acting as a transport medium for fission products during accidents.

A critical review of existing literature should be conducted in order to evaluate the adequacy of existing data. Additionally, a research effort needs to be initiated to gather information about the chemistry, quantity, size, and shape distribution of graphite dust in order to enable accurate modeling of fission product transport. In addition, graphite dust, when agglomerated, could to lead to blockages of coolant flow or free movement of control rods. It is also important to understand the adsorption and adhesion behavior of graphite dust on various materials surfaces present in HTGR (e.g., various metals and alloys, ceramic insulation and core graphite, and core supports). The presence of graphite dust at elevated temperatures on metallic surfaces could potentially accelerate carburization at the surface and near-surface regions.

Although there was significant discussion about who should take the lead in this area, the panel generally agreed that the graphite programs should contribute to this area by assuming responsibility for modeling dust production rates and dust characterization. It was recommended that other aspects of dust vectoring of fission products be left in the fission product group area. However, assessment of the association of fission products with carbonaceous dust may, in the final analysis, be best performed as a part of the dust production evaluation.

The surface condition (non-oxidized or oxidized due to the long term exposure to helium impurities) will also impact the tribological behavior and must be understood. A predictive capability for oxidation weight loss is needed. This requires knowledge of both oxidation kinetics and the diffusion characteristics of reactive species in the graphite (see Section 4.1).

The future graphite research program should plan to address the requirements of fission transport models through graphite dust. The experimental variables for graphite should include input from fission product model development in order that the output of graphite research may be properly used in the development and application of fission product model and code development activities.

4.4 Oxidative reactivity of graphite dust/powder

The potential effects of the rapid oxidation of graphite dust during an air ingress event (and the transport of associated fission products) should be assessed by a measurement of the oxidation

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[28] R. Moormann, W. Schenk, and K. Verfondern, "Source term estimation for small-sized HTRs: Status and further needs, extracted from German safety analyses," *Nucl. Technol*. **135** (3), 183–193 (2001).

[29] R. Moormann, *A safety re-evaluation of the AVR pebble bed reactor operation and its consequences for future HTR concepts*, Berichte des Forschungszentrums Jülich JUEL-4275 (2008).

[30] J. J. Cogliati and A. M. Ougouag, "Pebble bed reactor dust production model," Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology HTR2008, Washington, DC U.S.A. (2008).

rates of dust under projected accident conditions, combined with a modeling of estimated accident conditions.

A significant amount of work has been carried out in Europe in this area, particularly related to the decommissioning of the U.K. and French Magnox reactors, the Windscale Advanced Gas-Cooled Reactor (WAGR), and the Windscale Piles.[31,32] This work has involved a detailed survey and analyses of a wide range of experiments as well as analytical studies, which were previously conducted in Italy, France, and the U.K. Safety cases were made to use thermal cutting of steel support structures in WAGR which were approved, and the core was successfully dismantled without any incident involving graphite dust. Intrusive surveys, trepanning of graphite core samples, TV inspections, etc., have been carried out on the Windscale Pile cores, which contain a significant amount of carbonaceous dust containing stored energy, without any incident involving graphite dust.

In general, it was agreed by the panel that adequate documentation on dust generation exists with respect to decommissioning issues. Thus no further research is needed. However, the NRC should have the research reviewed and summarized in a white paper. The NRC may also conduct specific research related to dust generation (Section 4.3) and its oxidation pertaining to fission product transport phenomena.

4.5 Improved mechanistic modeling and predictive capability for irradiation-induced dimensional changes and creep

During reactor service, the graphite blocks in the reactor are subject to direct stress by imposed mechanical load, thermal stress due to temperature-dependent coefficients of thermal expansion (CTEs), and stresses that occur from differential irradiation-induced dimensional changes. These stresses may be relaxed by irradiation creep. Consequently, calculating the stress state at any point in a reactor component requires precise knowledge of the spatial variations of neutron dose and temperature, dimensional change, the creep rate, and physical properties (as a function of dose and temperature) of graphite, typically, thermal conductivity, CTE, elastic modulus, and strength. All of these properties have associated uncertainties due to anisotropy and the inherent variability of graphite properties. Consensus behavior model and computational codes have to be developed for each of these behavioral properties for NGNP graphites, which will provide the input data to the overall structural stress analysis code for the core graphite component.

Professor Marsden recited from personal experience the difficulties of working on graphite reactor safety and life extensions with an incomplete database and mechanistic understanding of graphite behavior in the reactor. This lack of basic knowledge had necessitated the implementation of models based on the limited information available and engineering judgment where such models were not available. These models have been changed several times when it was realized that the graphite cores of interest were not behaving as expected.[33] This has finally

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[31] A. J. Wickham and D. Bradbury (principal authors), *Graphite dust deflagration: A review of international data with particular reference to the decommissioning of graphite moderated reactors*, Electric Power Research Institute, Final Report ID #1014797 (2007).
[32] B. J. Marsden and A. J. Wickham (Editors), *Characterization, treatment and conditioning of radioactive graphite from decommissioning of nuclear power plant*, IAEA TECDOC 1521 (2006).
[33] J. Reed, "An overview of graphite core assessment methodology. Securing the safe performance of graphite reactor cores," presented at Securing the Safe Performance of Graphite Reactor Cores, Nottingham, U.K., November 24–26, 2008 (papers to be published; book by the Royal Society of Chemistry (RSC) Publishing).

resulted in materials test reactor (MTR) experiments that have been carried out late in reactor life.[34]

The key to prediction of the generation of stresses in components is the prediction of the rates of dimensional change and irradiation creep.[35] As changes to the properties of irradiated graphite supplied to the nuclear industry will vary due to changes in raw materials and manufacturing routes, it will not be cost-effective or desirable to obtain a complete set of data every time a new or different graphite is considered for reactor use. In addition, operational experience in the U.K. and elsewhere has shown that reactors operate often outside the original design intent, in some cases beyond the scope of the original data, be it graphite or metals. For this reason, it is essential that a well-founded mechanistic understanding, based on microstructural observations in irradiated graphite, is available to give confidence to models used to interpolate and extrapolate available data. In the U.K. the use of x-ray tomography, optical, TEM, SEM, Raman, XRD[36,37,38] and other techniques along with digital image correlation and multi-scale modeling [39,40,41,42] has started giving an insight into property-microstructure relationships in non-irradiated graphite and now in irradiated nuclear graphite; however, this is only the beginning and further work would be needed.

There is considerable variability in the properties of both non-irradiated and irradiated graphite. This variability in material behavior and uncertainty in irradiation data needs to be rigorously accounted for using modern statistical methods, such that the uncertainty in the prediction of component stress is adequately enveloped. This effort is especially important to ascertain that a sufficient factor of safety is provided in the design, considering the potential degradation of the design factor of safety during reactor operation. In the U.K., modern statistical methods using pattern recognition techniques and curve fitting techniques have been used to fit derived models for Gilsocarbon graphite.[43] These methods provide not only empirical fits to the data but also an insight into important mechanisms that may have not been appreciated previously, thus providing guidance for future research areas.

In Japan, in parallel to the activities on drafting the Standards for Graphite Core Components, the JAEA has reviewed and analyzed the existing irradiation data to investigate the possibility of their extrapolation to the higher fluence.[44] To verify the equations derived for evaluating various properties, more irradiation data, at higher fluences and higher temperatures in particular, are required. To fulfill this requirement, appropriate irradiation programs are being proposed,

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[34] M. Bradford, L. Pearson, and J. Reed, “Materials test reactor project. Securing the safe performance of graphite reactor cores,” presented at Securing the Safe Performance of Graphite Reactor Cores, Nottingham, U.K., November 24–26, 2008 (papers to be published; book by the Royal Society of Chemistry (RSC) Publishing).
[35] B. J. Marsden, B. Rand, D. K. L. Tsang and G. N. Hall, “Revisit of UK graphite irradiation data and law,” Proc. International Carbon Conference, Aberdeen, Scotland, 2006.
[36] A. N. Jones, G. N. Hall, M. Joyce, A. Hodgkins, K. Wen, T. J. Marrow, and B. J. Marsden, "Microstructural characterization of nuclear grade graphite," *J. Nucl. Mater*. **381**(1–2), 152–157 (2008).
[37] K. Y. Wen, T. J. Marrow, and B. J. Marsden, "The microstructure of nuclear graphite binders," *Carbon* **46**(1), 62–71 (2008).
[38] K. Wen, J. Marrow, and B. Marsden, "Microcracks in nuclear graphite and highly oriented pyrolytic graphite (HOPG)," *J. Nucl. Mate.* **381**(1–2), 199–203 (2008).
[39] C. Berre, P. M. Mummery, B. J. Marsden, T. Mori, and P. J. Withers, "Application of a micromechanics model to the overall properties of heterogeneous graphite," *J. Nucl. Mater.* **381**(1–2), 124–128 (2008).
[40] C. Berre, S. L. Fok, B. J. Marsden, P. M .Mummery, T. J. Marrow, and G. B. Neighbour, "Microstructural modeling of nuclear graphite using multi-phase models," *J. Nucl. Mater.* **380**(1–3), 46–58 (2008).
[41] L. Babout, B. J. Marsden, P. M. Mummery, and T. J. Marrow, "Three-dimensional characterization and thermal property modelling of thermally oxidized nuclear graphite," *Acta Materialia* **56**(16), 4242–4254 (2008).
[42] G. Hall, B. J. Marsden, and S. L. Fok, "The microstructural modeling of nuclear grade graphite," *J. Nucl. Mater.* **353**(1–2), 12–18 (2006).
[43] E. D. Eason, G. N. Hall, B. J. Marsden, and G. B. Heys, "Development of a Young’s modulus model for Gilsocarbon graphites irradiated in inert environments," *J. Nucl. Mater.* **381**(1–2), 145–151 (2008).
[44] E. Kunimoto et al., “Expansion of Irradiation Data by Interpolation and Extrapolation for Design of Graphite Components in HTGR,” JAEA-Research 2009-008 (2009) (in Japanese; Translation into English is now in progress).

preferably in the framework of international collaboration. In this regard the activities carried out in the Generation IV VHTR Materials Program and the recently initiated International Atomic Energy Agency (IAEA) Irradiation Creep Cooperative Research Program (CRP) are of value.

Current DOE R&D efforts are directed at acquiring design data for the creep rates of candidate NGNP graphites and are being conducted collaboratively by INL and ORNL. In partnership with universities (DOE–Nuclear Energy University Program funding), INL and ORNL are examining fundamental deformation and multi-scale damage mechanisms in graphite, with major attention given to the mechanisms of irradiation-induced creep. Limited work on mechanisms is being carried out by INL and ORNL. An improved understanding of the fundamental irradiation damage and creep mechanisms will underpin the development of predictive models for irradiation-induced dimensional changes and irradiation creep. These analytical predictive models, supported by phenomenological data and relationships, provide required input to core behavioral models used to determine core structural degradation and end of life of core components.

The panel recommended that the DOE's current effort for this model development be augmented and accelerated, and that the NRC develop its own models independently and then validate and refine the models. The NRC may have a university or an independent laboratory develop new or evaluate existing models and refine these models, independent of the licensee. Additionally, NRC should consider participation in the IAEA coordinated research program on irradiation creep mechanisms in graphite.

4.6 An accepted fracture criteria for irradiated nuclear graphite

Predicting when and where the fracture of an irradiated NGNP graphite component may occur is at present very difficult because of the lack of reliable data on properties that are influenced by variations in the reactor environment, and the lack of a validated predictive model for NGNP graphite properties. Having established an estimate of the stress state of a graphite component, a suitable fracture criterion must be adopted to ascertain the probability of failure, or the predicted failure stress must be compared to the design stress to ensure design factors of safety margins have not been exceeded.

In order to predict component life, as well as graphite component stresses, we must be able to compare these predictions against suitable validated failure criteria, which account for not only the uncertainty in operational stress but also the uncertainty in the failure criteria. For this reason any failure criterion needs to be not only capable of serving as a comparison tool for different grades of graphite but also able to be used along with finite-element-based stress analysis codes to predict component life. Some limited work has been done in this area in developing a two-criteria graphite damage model for graphite, which is both stress and fracture based, but this was only for two-dimensional geometries.[45] Furthermore, knowledge of failure stress (in compression, tension, and bending) and an understanding of the fracture behavior of graphite are required. At present the form that failure criteria for nuclear graphite components should take is not clear; therefore, in conjunction with the acquisition of strength and fracture data, failure models will need to be developed and validated. Again, modern probabilistic statistical methods will be required to address uncertainty.

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[45] Z. Zou, S. L. Fok, B. J. Marsden, and S. O. Oyadiji, "Numerical simulation of strength test on graphite moderator bricks using a continuum damage mechanics model," *Engineering Fracture Mechanics* **73**(3), 318–330 (2006).

The current DOE plan includes examining potential fracture criteria (Weibull theory, microstructure fracture model, etc). Room temperature experimental data, in ambient atmosphere, are being obtained from strength measurements (tensile, compressive, flexure) from ASTM-specification specimens machined out of candidate NGNP graphite billets, including biaxially loaded specimens, in order to determine the fracture envelope and statistics of fracture. Potential fracture criteria will be tested against experimental strength data. However, the panel felt that more work is needed to develop individual graphite component and whole core models, which integrate all the inputs and enable prediction of local stresses within graphite components.

The panel recommended that the NRC staff participate in efforts to establish graphite fracture criteria, which, in part, would define what constitutes a failure. The definition of "failure" should take into account the function of the component (e.g., core support structure vs. non-load-bearing reflector block) as well as the properties of the material and the conditions of operation. The NRC staff should actively participate in the activities of the ASME code committee involved in the development of stress limits.

4.7 Accepted improved methods for graphite-core-component volumetric inspection (production and in-service) and online monitoring

Reliable methods are needed for the detection of flaws in key graphite components (e.g., core support structure components). Consideration should be given to proof-testing methodologies as well as nondestructive techniques. Current DOE research being conducted at INL is directed towards nondestructive evaluation (ultrasonic) and flaw detection in large graphite blocks. At ORNL, the relationship between fracture and the flaw population (disparate and background flaws) is being examined. Both optical microscopy, coupled with automated image analysis, and 3D X-ray computer tomography (CT) are being applied to characterize the flaw population. In Japan, the JAEA is also developing inspection methods based on 3D X-ray CT imaging.

The DOE laboratories are currently involved in the development of design codes through ASME (Section III) and will be involved in the development of in-service inspection (ISI) requirements for HTR graphite cores (ASME Section XI).

The panel recommended that the NRC be involved in code development activities (ASME Section III and XI). Moreover, the NRC should quantify the current limitations of existing nondestructive evaluation (NDE) methods and identify those areas that will need additional design factor of safety margin/alternate monitoring capability (potentially surveillances) to accommodate the potential limitations in the NDE area.

4.8 Independent capability to verify graphite-core stress analysis methods

Sophisticated codes have been developed to predict stresses in graphite components.[46,47] However, a major difficulty is the ability to validate the prediction of stresses and failure in irradiated graphite components. This is because most graphite moderated components remain in the graphite core for the life of the reactor. Consequently, a need exists to develop general guidelines for specific finite-element behavioral codes that map the core stress as a function of reactor operation.

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[46] D. K. L. Tsang and B. J. Marsden, "The development of a stress analysis code for nuclear graphite components in gas-cooled reactors," *J. Nucl. Mater.* **350**(3), 208–220 (2006).
[47] D. K. L. Tsang and B. J. Marsden, "Constitutive material model for the prediction of stresses in irradiated anisotropic graphite components," *J. Nucl. Mater.* **381**(1–2), 129–136 (2008).

In the U.K., a limited number of experiments were conducted that involved slitting irradiated graphite fuel sleeves, but these were thin-walled components and are not typical of the large blocks to be used in prismatic and pebble high-temperature reactor (HTR) designs. In addition, the analysis was not particularly rigorous and was carried out a number of years ago. There is now the opportunity in some countries to apply similar techniques to components in decommissioned reactors, either in the core or during dismantling. However, in the past it has proved very difficult to find the desire or funding to do such experiments, as decommissioning companies tend to be driven by tight cost and deadlines. Such an exercise would only be worthwhile if there were sufficient data from graphite stress analysis.

In the U.K., measurements of component deformations, such as channel bore dimensional changes, are regularly carried out. Currently (2009) in the U.K. there is a significant new initiative, funded through the Health and Safety Executive (HSE), to determine what information can be derived from this channel bore information.

Methods will also need to be developed to assess whole core behavior, taking into account the interactions between graphite components as they change dimension due to thermal and irradiation-induced strains and the implications of these deformations in the event of component failure. Work in this area is being carried out for the AGRs;[48] however, such analysis is reactor specific and, therefore, new models will be required for HTRs.

The panel recommended that the NRC develop its own independent verification and confirmatory research capability for stress analysis. The finite-element modeling (FEM) codes should be available to predict core stresses.

4.9 High-temperature stored energy release

Published data show that graphite which has been irradiated at low temperatures (e.g., below 150^{o}C) exhibits a second stored energy release peak when heated to high temperatures (T>1200^{o}C). Currently, data are not available for the high-temperature release of stored energy from graphite irradiated at HTR-relevant temperatures.

A significant amount of work in this area has been carried out in the U.K., and much of the data is available in the open literature. Experimental data show that the accumulation of total stored energy as a function of fast neutron dose saturates and reduces in magnitude with increased irradiation temperature. In addition, after the 200^{o}C peak, the rate of release of stored energy decreases, falling below the specific heat, at irradiation temperatures above 100^{o}C as a function of neutron dose.[49] However, there are literature data which show that in addition to the "200^{o}C" rate of release peak, there is another peak in the rate of release curve at around 1200^{o}C for graphites irradiated at lower temperatures.[50] The existence of this additional energy was predicted as it had been observed that all the stored energy had not been released in samples heated in the differential scanning calorimeter (DSC) to 600^{o}C. However in the U.K., heat released at this temperature would be considered insignificant compared to the heat generated by thermally oxidizing graphite at 1200^{o}C, that is, if sufficient oxygen could be made available to the graphite

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[48] D. K. L. Tsang, B. J. Marsden, and G. B. Heys, "Two dimensional analysis of AGR core using a superelement technique," presented at Securing the Safe Performance of Graphite Reactor Cores, Nottingham, U.K., November 24–26, 2008 (Papers to be published; book by the Royal Society of Chemistry (RSC) Publishing).
[49] B. J. Marsden (Editor), *Irradiation damage in graphite due to fast neutrons in fission and fusion systems*, IAEA_TECDOC-1154 (2000).
[50] J. Rappeneau, J. L. Taupin, and J. Grehier, "Energy released at high temperature by irradiated graphite," *Carbon* **4**(1), 115–124 (1966).

internal porosity, which is unlikely at such a high temperature. The heat of combustion of graphite is 3.26×10^4 J/g, and the highest level of stored energy[49] ever measured is 2.7×10^3 J/g.

The panel recommended that the NRC conduct a literature review and develop a white paper corroborating the above. It further recommended that the NRC support the experimental verification by conducting DSC experiments up to 1600°C on graphite samples irradiated at temperatures, which are appropriate to the NGNP designs, in order to confirm this understanding.

4.10 Graphite decommissioning

Considerable research is required to address the handling and disposal issues of discharged graphite. The reuse of the PMR block should be considered. In addition, some of the reactor vendors are considering the option of recycling graphite for nuclear use. However, significant research effort is required in this area before graphite could be recycled.

The DOE is sponsoring a research program (Deep Burn) which is directed at the transmutation of light water reactor (LWR) fuel in a HTGR. As part of this program, ORNL is investigating options for graphite reuse, recycle, and disposal. ORNL is also exploring collaboration opportunities with the European CARBOWASTE program.

In Europe there is a € 12 million European Commission FP7 framework program (CARBOWASTE)[51,52,53] aimed at investigating the options available for the disposition of the irradiated graphite waste. Twenty-eight (28) partners from 11 countries are participating in this program. Various options are being considered, including recycling as well as volume reduction and deep geological or shallow disposal. Collaboration between European and U.S. initiatives in this area would be most beneficial.

In Japan, a long-term R&D program focused on decommissioning technology and a probable recycling process is to be prepared based on a comprehensive literature survey of previous and current activities. The literature survey may take 1–2 years with extensive support provided by individuals worldwide.

The panel recommended that the NRC monitor worldwide activities in this area through staff participation in IAEA, DOE Deep Burn, and European CARBOWASTE meetings.

5. Organizations Performing Nuclear Graphite R&D

Below is a non-comprehensive list of organizations conducting research on nuclear graphite, with whom NRC, as appropriate, might enter into cooperative agreement to generate and share technical information and knowledge.

5.1 Oak Ridge National Laboratory (ORNL)

ORNL is performing graphite R&D in support of the DOE NGNP project (in collaboration with INL) and the Republic of South Africa's Pebble Bed Modular Reactor (PBMR) project.

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[51] W. von Lensa, G. Cardinal, D. Bradbury, H. Eccles, J. Fachinger, B. Grambow, M. J. Grave, B. J .Marsden, and G. Pina, "Treatment and disposal of irradiated graphite and other carbonaceous waste," Proc. 4th International Topical Meeting on High-Temperature Reactor Technology, HTR2008, Washington, D.C., USA., September 2008.

[52] http://www.carbowaste.eu/

[53] http://cordis.europa.eu/fetch?CALLER=FP7_EURATOM_PROJ_CROSSTOPICS_EN

For the NGNP Program, ongoing research activities include the following:

- Irradiation damage studies
    - Graphite irradiation experiments, High Flux Isotope Reactor (HFIR)
    - Irradiation effects modeling
- Irradiation creep studies
    - Irradiation creep experiments (HFIR)
    - Creep modeling
- Irradiation damage mechanism (jointly with NC State University)
- Characterization of nuclear graphite physical and mechanical properties
- Determination of nuclear graphite biaxial strength
- Nuclear graphite oxidation studies (air oxidation)
- Fracture mechanism, fracture toughness, and fracture modeling of nuclear graphite
- Structural characterization (pore structure, texture)
- Effect of oxidation on structure

The results of this research are reported to the NGNP Program in monthly progress reports, data packages, and topical research reports (ORNL technical memorandum). The NGNP research is expected to continue to 2015–2020, depending upon budget and schedule. The reports from the DOE NGNP project are available to the NRC.

For the PBMR project (RSA), ORNL is performing the Materials Test Reactor Program for grade NBG-18 graphite (anticipated duration ~10 years). This program, being conducted in the HFIR, will provide data on production-grade NBG-18 graphite for evaluating the effects of neutron irradiation on key physical and mechanical properties, including strength (tensile, bend, compressive), elastic constants, thermal conductivity, and coefficient of thermal expansion. The ORNL is also conducting oxidation kinetic studies for graphite-grade NBG-18 and matrix carbon, in air and moist helium environments. Experimental data will be reported to PBMR and may be available to the DOE/NRC via the Generation IV International Forum.

Additionally, the ORNL is conducting research on the reuse and recycle of irradiated graphite under the DOE Deep Burn project. Annealing studies and studies addressing the recycle of irradiated graphite are currently being performed at ORNL. Moreover, in a joint study with GrafTech, the amount of recycled graphite in new graphite stock is being reexamined in a pilot trial. This project will continue through FY 2011. Monthly technical reports are prepared for the DOE as well as topical reports.

ORNL has the capability to conduct MTR experiments and perform pre- and post-irradiation examination.

5.2 Idaho National Laboratory (INL)

INL is performing graphite R&D in support of the DOE NGNP project (in collaboration with ORNL). Ongoing research activities include the following:

- Irradiation damage and irradiation creep studies in the advanced test reactor (ATR)
- Characterization of graphite physical and mechanical properties (billet characterization)
- Air oxidation of graphite
- Core behavior model development
- NDE methods development

Research output is reported in NGNP monthly technical reports, data packages, and topical reports and is available to the NRC via the DOE NGNP project.

INL has the capability to conduct MTR experiments and perform pre- and post-irradiation examination.

5.3 NRG Petten, The Netherlands

NRG Petten is performing graphite irradiations in the High Flux Reactor (HFR) under the European 6th and 7th framework agreement. The irradiation experiments are aimed at identifying new graphites suitable for use in HTRs and are at reasonably high temperature and dose (beyond turnaround). The experimental data will be made available to the DOE (and NRC) via the Generation IV International forum. NRG Petten is also engaged in a creep experiment for the U.K. Utility British Energy, albeit with a CO_2 atmosphere in the capsule. The data will be proprietary. NRG Petten will be active in the field of graphite research for approximately 10 years.

NRG Petten has the capability to conduct MTR experiments and perform pre- and post-irradiation examination.

5.4. Japan Atomic Energy Agency (JAEA)

The JAEA (formerly Japan Atomic Energy Research Institute, JAERI) is conducting R&D in support of the HTTR. Current research is limited by the unavailability of JAEA's materials test reactors. However, some activities involving inspection of HTTR's graphite core have been conducted. Recent work has been directed at NDE of nuclear graphites and characterization of grade IG-430 (Toyo Tanso). The decision of the Chinese to use Toyo Tanso grade IG110 for the demonstration PBMR has given JAEA added impetus to initiate new nuclear graphite R&D.

JAEA has the capability to conduct MTR experiments and perform pre- and post-irradiation examination.

5.5 Korean Atomic Energy Research Institute (KAERI)

KAERI is conducting limited R&D activities in support of their national HTR project. Graphite research has been in the areas of ion-bombardment damage of graphite (displacement damage), to simulate neutron damage, and in graphite air oxidation. KAERI is researching the kinetics of air oxidation[54] and the effects of oxidation on pore structure (in collaboration with ORNL).[55]

KAERI has the capability to conduct MTR experiments and perform pre- and post-irradiation examination.

5.6 National Nuclear Laboratory (NNL, UK)

The NNL conducts sponsored graphite research in support of the U.K.'s MAGNOX operating reactors.

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[54] S-H. Chi and G-C. Kim, "Comparison of the oxidation rate and degree of graphitization of selected IG and NBG nuclear graphite grades," *J. Nucl. Mater.* **381**(1–2), 9–14 (2008).

[55] S-H. Chi, C. I. Contescu, and T. D. Burchell, "Density change of an oxidized nuclear graphite by acoustic microscopy and image processing," *J. Eng. for Gas Turbines and Power – Trans. ASME* **131** (5) 052904 (4 pages) (2009).

5.7 Electrical Power Research Institute (EPRI)

The EPRI conducts industry-sponsored graphite research.

5.8 Institute for Metals Research (IMR), Chinese Academy of Science (CAS), China

The IMR is engaged mainly in research and development of high performance metallic materials, new types of inorganic nonmetallic materials and advanced composite materials covering their structures, properties, performances, corrosion and protection, as well as the relationship among them. They are expanding their work on nuclear graphite.

5.9 GrafTech International

GrafTech International is a nuclear graphite manufacturer and has the capability to conduct characterization of non-irradiated graphite.

5.10 Toyo Tanso Company Ltd.

Toyo Tanso Company Ltd. is a nuclear graphite manufacturer and has the capability to conduct characterization of non-irradiated graphite.

5.11 SGL Carbon

SGL Carbon is a nuclear graphite manufacturer and has the capability to conduct characterization of non-irradiated graphite.

5.12 British Energy Generation Ltd (BEGL)

BEGL has a substantially funded graphite research program aimed at continued safe operation and life extension of the U.K. AGRs. This program includes an AGR graphite MTR program at NRG Petten which is currently under way and a proposed graphite irradiation creep experiment. They also have invested in large-scale AGR core rigs used to model core behavior and are investigating the possibility of building a large seismic rig. BEGL have spent a substantial amount of funding on the development of various computer codes to model various aspects of AGR core behavior. However, although there are many synergies between AGR and HTR technology, the AGR operates with a carbon dioxide coolant; thus, the AGR graphite is subject to a considerable amount of radiolytic weight loss, up to ~40% peak, leading to significant modification of the graphite material properties above caused by fast neutron damage.

6. Universities Performing Nuclear Graphite R&D

Below is a non-comprehensive list of universities conducting research on nuclear graphite, with whom NRC, as appropriate, might enter into cooperative agreements to share technical information and knowledge.

6.1 U.S. Universities

Recently funded government initiatives such as the Nuclear Energy Research Initiative (NERI) and The Nuclear Energy University Program (NEUP) have encouraged and facilitated university research on nuclear graphite. In the U.S.A. two universities have been successful in competing for NEUP funding.

- North Carolina State University (NCSU)

In a collaborative effort with ORNL, NCSU will be performing research directed at understanding creep mechanisms in graphite with experiments, multiscale simulations, and modeling. Research will focus on dislocation mechanisms in graphite and the interaction of dislocations with irradiation-induced crystal damage. Creep mechanisms indentified in metals (dislocation climb) will be examined as candidates along with the Kelly-Foreman pinning-unpinning graphite creep model. Characterization of crystal defect/dislocations will be conducted using TEM, XRD, and small-angle neutron scattering (SANS). Moreover position annihilation spectrometry will also be applied in an attempt to understand structural changes. Some experiments will be performed in the PULSTAR nuclear reactor at NCSU, as well as the HFIR at ORNL. Simulations and model development are an integral part of the NCSU project. This project has a 3-year duration beginning in mid-2009. Quarterly and annual reports to DOE are required and should be available to the NRC.

- Boise State University

This project will be conducted collaboratively with INL and is directed at graphite structure–property relationships, crystal deformation mechanisms, and dislocation flow and pinning mechanisms. Manchester University will also be involved with this project.

- Clemson University

Clemson University is developing carbon fiber suitable for use in nuclear reactors (collaborations with ORNL).

- Colorado School of Mines (CSM)

CSM has recently announced that it will be setting up a nuclear graphite research and education program. However, this program is currently inactive.

- University of Maryland

The University of Maryland has initiated a research program to understand low-dose neutron damage effects on graphite.

- Massachusetts Institute of Technology (MIT)

MIT has a carbon program looking a various aspects of a variety of carbon materials; including irradiation effects on graphite. Additionally, MIT has a gas-cooled reactor R&D program.

6.2 Non-U.S. Universities

- The University of Manchester, U.K.

The University of Manchester (UK) is conducting research on nuclear graphite properties and behavior, funded by the U.K. Regulator and the U.K. Nuclear Power Reactor Operators. Their work will be funded during the life of the currently operating reactors.

Major research areas include the following:
  - Radiation damage mechanism
  - Characterization technique development
  - Materials behavioral modeling
  - Oxidation and effects of oxidation on structure and properties
  - Stress analysis methodologies

Research data are reported to the work sponsor and are presumably proprietary. However, much of the data is published in the open literature and presented annually at the International Nuclear Graphite Specialists Meeting. The U.K. is not currently active in the Generation IV International Forum.

- The University of Hull

Current research program includes the analysis of U.K. gas-cooled nuclear reactor core designs, particularly materials performance and the functionality of graphite core components to support life extension using various modeling and analytical techniques.

- The University of Sussex

Current research programs are directed at ab initio simulations of the effects of radiation on graphite structure and properties.

- South African Universities

PBMR (Pty) Ltd is sponsoring research into nuclear graphite and carbon materials at several universities in South Africa.

  - University of Pretoria: PBMR sponsors a "carbon chair," currently held by Professor Brian Rand. Research activities currently include graphite air oxidation (grade NBG-18) and development of domestic feedstocks for the production of nuclear-grade graphites.

  - University of Cape Town: Active in the field of fracture mechanics testing of graphite.

  - North-West University: Actively researching fatigue and multiaxial fracture behavior of NBG-18 graphite. Developed novel multiaxial specimen design; reported in open literature and in the Ph.D. Thesis of Johan Roberts.

- Korea Advanced Institute of Science and Technology (KAIST):

KAIST is working collaboratively with INL on air-ingress scenario modeling, including air oxidation of the graphite core via an I-NERI project.

- <u>Institute of Nuclear Energy Technology (INET) at Tsinghua University:</u>

INET is working on areas related to the Chinese Pebble Bed Reactor Project. Tsinghua University is the reactor design authority and has a keen interes in the irradiation behavior of graphite. INET participates in ASME and IAEA nuclear graphite activities.

Attachment 1

Summary of the ONRL/NRC Workshop on Nuclear Graphite Research

1. Workshop Format

The format of the workshop was as follows:

- Day 1: Background on International HTGR Nuclear Graphite Research and Regulatory Perspective:
  - Presentations from the U.S.A., U.K., RSA, Japan panel members (*China unable to attend due to visa issue)

- Day 2: Identification of Technology Gaps and Future Nuclear Graphite Research Activities: Group discussion sessions on the following proposed themes:
  - Graphite qualification
    - INL plans, and vendor plans
    - Comments on plans
      - Adequacy of properties and database
    - Quality assurance requirements
  - Requirements for core behavioral models
    - Irradiation properties
    - Models for fundamental understanding for structural integrity analysis
    - Handling of data and model uncertainties
  - Oxidation of graphite by coolant impurities
  - Status of codes and standard development / future challenges
    - Design and construction code (Section III)
      - Stress analysis
      - Adequacy of margins
    - In-service inspection (Section XI)
  - Tribology and oxidation leading to graphite dust
  - Air ingress and water ingress (accident)
    - Safe shutdown and safe cool down
  - Defining end of core-component life (criteria and safety margins)
  - Decommissioning and disposal
  - Other themes as suggested by panel members

- Day 3 (1/2 day): Panel Documentation of Recommendations for NRC's Nuclear Graphite Research

2. Biography of Panel Members

- *Dr. Timothy D. Burchell*: Leader of the Carbon Materials Technology (CMT) Group within the Materials Science and Technology Division of the Oak Ridge National Laboratory (ORNL), which is engaged in the development and characterization of carbon (graphite) materials for the U.S. Department of Energy. Prior to assuming the position of CMT Group Leader, Dr. Burchell was manager of the Modular High Temperature Gas-Cooled Reactor Graphite Program and was responsible for the development of a multi-year research program to acquire reactor graphite property design data. Prior to joining ORNL Dr. Burchell was a research officer at Berkeley Nuclear Laboratories, Berkeley, Gloucestershire, U.K. where he worked on monitoring the condition of graphite moderators in gas-cooled power producing nuclear reactors. Dr. Burchell's work experience also includes six years in the aerospace industry. He is the author of numerous papers on the subject of graphite fracture behavior and modeling, the effects of neutron damage on carbon materials structure and properties, and adsorbent carbon composites. He has also authored several book chapters on aspects of carbon materials and is the editor of the book entitled "Carbon Materials for Advanced Technologies".

- *Professor Barry J. Marsden:* Professorial Fellow of Science in Nuclear Graphite Technology in the School of Mechanical, Aerospace and Civil Engineering at the University of Manchester, UK. Professor Marsden joined the UKAEA in 1983 to work on Fast Reactor technology. Since 1989 Professor Marsden has headed nuclear graphite technology research in the UKAEA, then AEA Technology before moving to the University of Manchester in 2001 to set up the Nuclear Graphite Research Group (NGRG). His work has involved research into the safety, life extension and decommissioning of graphite-moderated reactors. At the University of Manchester under the direction of Professor Marsden, the NGRG has pioneered new research into behavior of graphite components in Magnox, AGR and HTR systems. Professor Marsden is also actively involved in multiple international research programs and committees related to the area graphite-moderate reactors

- *Graham Heys*: HM Principal Inspector (Nuclear Installations) of the UK HM Nuclear Installations Inspectorate (NII). NII is that part of The Health & Safety Executive responsible for licensing of nuclear installations and regulation of nuclear safety in the United Kingdom. Following 13 years experience as a metallurgist and materials scientist in the UK nuclear and defence industries Graham joined the NII in 1991. Since 1991, Graham has undertaken a number of roles regulating nuclear power reactors and chemical plant. In 2001, he was promoted to NII lead on graphite core integrity assessment for all AGR and Magnox reactors and graphite nuclear safety research. During this period he secured reasonably practicable safety improvements at Magnox and AGR power stations, secured HSE funding to establish the Nuclear Graphite Research Group at The University of Manchester under the direction of Professor Marsden, and set-up the NII Graphite Technical Advisory Committee (GTAC) which advises NII on nuclear graphite technology.

- *Mark N. Mitchell:* Chief Design Engineer, CSC, Pebble Bed Modular Reactor (Pty) Ltd – Republic of South Africa (RSA). Mr. Mitchell currently leads the Pebble Bed Modular Reactor Core Structure Ceramics (graphite structures in the reactor) Design Team. He is a mechanical engineer, specializing in design and structures. He initially worked in the defense and aerospace industries, where he specialized in design and manufacturing with composite materials. In 1999, he joined the Pebble Bed Modular Reactor Company.

- *Dr. Motokuni Eto*: Technical consultant to Toyo Tanso with Japan Atomic Energy Agency. He is a member of the Graphite and Carbon Materials Performance Evaluation Group which was established in JAEA in 2007 in accordance with a collaboration agreement between JAEA and Toyo Tanso. He also worked for the Secretariat of the Nuclear Safety Commission of Japan as a technical counselor. He started working on mechanical properties of HTGR graphites in early 70's to provide various data essential for the design and construction of the HTTR. From 1978 to 1980, as a visiting engineer, he conducted research on the oxidation of HTGR graphites in the HTGR Safety Division of Brookhaven National Laboratory. After the HTTR attained criticality in 1997, he has been conducting research on the development of high performance materials including C/C composites.

- *Dr. Robert L. Bratton:* Dr. Bratton obtained his undergraduate's and master's degree in nuclear engineering and his doctorate's in applied mechanics. He has been working at the Idaho National Laboratory for seventeen years. He worked on the NPR MHTGR program and is now working on Next Generation Nuclear Plant program researching graphite mechanical properties to support conceptual design, ASME code development and pre-licensing activities.

- *Dr. Robert P. Wichner:* Retired nuclear/chemical engineer with greater than 25 years experience in both gas-cooled reactor (GCR) and light-water reactors (LWR), addressing normal operation and safety issues, principally dealing with chemical and mass transfer aspects. As a group leader in ORNL's Chemical Technology Division he directed experimental work on diffusion of fission products in graphite and graphite oxidation. His most recent work has dealt with graphite oxidation under normal operating conditions by oxygen and water vapor. He is the principal contributor of the mass transfer and chemistry models in the GRSAC safety code. Since retirement in 2000, he has worked on explosive dispersal of radiation and a variety of source terms projects.

- *Mr. Schalk Doms:* Senior Regulatory Officer for the National Nuclear Regulator of South Africa. Mr. Doms is a also registered as a Professional Engineering Technician with the Engineering Council of South Africa. He is currently involved with the licensing of the new Pebble Bed Modular Reactor under development for Eskom. Mr. Doms is a qualified Metallurgist with a National Higher Diploma in Physical Metallurgy from Technikon Pretoria and a Post Graduate Diploma in Science from Wits University. He has been in the Metallurgical field for more than 32 years, of which 8 years as Manager of a Metallurgical Test Laboratory. He has gained experience in areas such as Nuclear Materials, Powder Metallurgy, Impregnated Diamond Products, Tungsten Carbide Machine Tools, Heat Treatment of Metals, Metallography, Foundry Practices, Metal Surface Finishing, Vacuum Technology, Failure Analysis, Corrosion Prevention and Control, Welding Metallurgy and Quality Assurance and Control.

- *Scott R. Penfield, Jr., PE:* is a Principal of Technology Insights, an engineering consulting firm that specializes in advanced energy systems, with particular emphasis on high-temperature gas-cooled reactors (HTGRs). Mr. Penfield has over 40 years of experience in nuclear energy systems and components, of which more than 30 years are specific to HTGRs. Areas of particular emphasis have included advanced energy conversion systems and components, assessment of high-temperature process energy applications, R&D planning and management, and the selection and use of advanced materials in HTGRs. A current assignment for PBMR is the development of a white paper for submission to NRC on the design and qualification of core structures ceramics, including graphite, ceramic insulation and carbon composites.

- *Dr. William E Windes:* Staff scientist at Idaho National Laboratory (INL). His research interests are in the areas of processing, characterization, and analysis of novel material systems for both nuclear and non-nuclear applications including materials for use in high temperature and irradiation environments. Dr. Windes is the Technical Lead for nuclear graphite and composites for the Next Generation Nuclear Plant (NGNP) program. In this roles, he is responsible for all research activities concerning nuclear graphite and composite development for use within the new high temperature gas-cooled reactor (HTGR) design. Specific duties include thermo-mechanical testing of non-irradiated and irradiated graphite and composites, development of new ASTM test standards for nuclear graphite and composites, and ASME code case development for determining material properties of nuclear graphite and composites for use within the NGNP. Dr. Windes is a member of a number of ASTM/ASME subcommittees critical to the development of national & international testing standards for VHTR material systems.

3. Summary of Presentations

- ***Nuclear Graphite Workshop - Welcome*** by Dr. Brian Sheron, Director of Research (RES), Office of Nuclear Regulatory Research, NRC.

  Summary: Dr. Sheron opened the workshop welcoming all attendees and thanking panel members from around the world for their participation. His presentation included an overview of NRC/RES mission, and a summary of NRC's experience with graphite reactors (i.e., Peach Bottom Unit 1 and Fort St. Vrain). He concluded his presentation re-stating the objectives of the workshop and emphasizing the importance of expert opinions on research conducted by the NRC.

  Q&A: no questions were asked after this presentation.

- ***A Short History of NRC Nuclear Graphite Research*** by Dr. Makuteswara Srinivasan, Senior Materials Engineer, Office of Nuclear Regulatory Research, NRC

  Summary: Dr. Srinivasan initiated his presentation by describing the objectives of the nuclear graphite research programs conducted by the NRC. Then he described the various research areas sponsored by NRC, including:

  a. Research at the Franklin Institute Research Laboratories, Philadelphia, PA, during the mid-1970's, 'Rules for Design of Nuclear Graphite Core Components – Some Considerations and Approaches.'
  b. Research at Brookhaven National Laboratories during the mid-1980's, 'A Safety Assessment of the Use of Graphite in Nuclear Reactors Licensed by the U.S. NRC.'
  c. Research at Brookhaven National Laboratories during the mid-1970's, 'An Appraisal of Possible Combustion Hazards Associated with a High-Temperature Gas-Cooled Reactor.'
  d. NRC-sponsored workshops on 'high-temperature gas-cooled reactor safety and research issues,' – 2001 and 2002.
  e. Pre-application review of technical papers (pebble bed modular reactor) submitted by Exelon – 2001 – 2002.
  f. The NRC assigned a staff to U.K.'s Nuclear Installations Inspectorate, Fall, 2002.
  g. Research at ORNL to initiate the formation of committees in ASME and ASTM to enable development of design, inspection, and operation codes and standards for graphite core components – 2002 – 2003.
  h. Active participation of NRC staff in ASME and ASTM meetings and in the International Nuclear Graphite Specialist Meetings.
  i. The staff participated in developing a NRC-DOE document on licensing strategy (August 2008) on proposed NGNP, as per Energy Policy Act, 2005.
  j. The NRC conducted a graphite phenomenon identification and ranking table (PIRT) exercise, in cooperation with DOE, in 2007 for guidance on prioritization of graphite research.
  k. The NRC has kept specific research options open, pending DOE HTGR design selection.
  l. The NRC has sponsored research (FY 2009) at ORNL to conduct a graphite workshop and identify potential research areas that the NRC may conduct in the future, which augments DOE and NGNP applicant's research.

Q&A: no questions were asked after this presentation.

- ***Workshop Overview and Goals*** by Dr. Nidia Gallego, Research Scientist, Carbon Materials Technology Group Oak Ridge National Laboratory (ORNL)

  Summary: Dr. Gallego presented an overview of the objectives of the workshop, highlighting the background documents provided to the panel members. She explained the format and agenda of the workshop. She also thanked the panel members for accepting the invitation to participate in the workshop and for their help during the planning stage.

  Q&A: no questions were asked after this presentation.

- ***Overview of Graphite PIRT Findings*** by Dr. Timothy Burchell, Group Leader, Carbon Materials Technology Group, ORNL

  - *Summary:* Dr. Burchell presented a summary of the Graphite Phenomena Identification & Ranking Table (PIRT) Review conducted by the NRC in 2007. He provided information on the PIRT panel and summarized the process used to identify the various graphite behavior phenomena and their categorization into Importance High or Low and/or Knowledge High or Low. Then he discussed in detail the five phenomena, which were identified as Importance High, Knowledge Low (I-H, K-L): *1) Irradiation-induced creep (irradiation-induced dimensional change under stress; 2) Irradiation-induced change in CTE, including the effects of creep strain; 3) Irradiation-induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress); 4) Blockage of fuel element coolant channel due to graphite failure and/or graphite spalling; 5) Blockage of coolant channel in reactivity control block due to graphite failure and/or graphite spalling.* This was followed by a summary of the nine phenomena ranked Importance High, Knowledge Medium (I-H, K-M) and the two phenomena ranked Importance Medium, Knowledge Low (I-M, K-L). The PIRT findings provided the basis for comparison with DOE planned graphite research to identify technical gaps.

  Q&A:
  Q1: Has any thing come up that has changed the ranking from the PIRT?
  A1: Perhaps the issue of dust formation; from the German report that describes that dust was found in areas they did not expect.

  Q2: Are there any phenomena that were missed?
  A2: I don't think so; perhaps the ranking could have been changed.

  Q3: Do we (the panel) need to revisit the ranking of the PIRT?
  A3: No, we may question it and suggest updates based on the fact that we have some more data on pre-irradiation characterization of current graphites. The panel also needs to consider the quality and adequacy of the information that has been obtained since PIRT.

- ***Overview of US DOE NGNP Research Plan*** by Dr. William Windes, Leader Graphite Group, Idaho National Laboratory (INL)

  Summary: Dr. Windes explained the NGNP concept, describing the (design) normal, off-normal and dpa conditions expected for the various core components: the graphite fuel block is expected to experience the higher temperature and dose (1200 ^{o}C normal, 1400 ^{o}C off-normal and ~0.8 dpa/yr). He then listed the various assumptions that they are working with: either prismatic core or pebble bed core; no decision of specific graphite type; starting operating temperature of 750 ^{o}C and gradual increase to 900-950 ^{o}C; 400-600 MWt; Helium inert gas coolant; current vendors are Westinghouse (PB), General Atomics (PR), and AREVA(PR). Dr. Windes then summarized the current DOE research plan and the various activities including participation in the development of ASTM standards development for air oxidation, fracture toughness, X-ray diffraction (XRD) techniques, NDE techniques and shear tests. Some of the current standards may not be appropriate for the following properties: thermal diffusivity, 3-point bend, and sonic velocity measurements. He then provided a summary of the baseline characterization activities and irradiation activities including (AGC experiments) for the proposed nuclear graphite (NGB-18) for the NGNP core components. He also described the current work for whole core modeling and some of the initial results.

Q&A: no questions were asked after this presentation.

- ***UK and European Research*** Activities by Professor Barry Marsden, School of Mechanical, Aerospace and Civil Engineering, The University of Manchester – U.K.

  Summary: Professor Marsden's presentation included an overview of the European Nuclear Graphite Research Initiatives, the UK-Specific Nuclear Graphite Research Activities and a summary of the Nuclear Graphite Research activities at the University of Manchester. The main drivers for nuclear graphite research in Europe are the continued safe operation of the UK AGRs and remaining Magnox reactors, including the possibility of life extension. Activities under the European Framework Programmes FP5, FP6, and FP7 include RAPHAEL and CARBOWASTE. RAPHAEL is the ReActor for Process heat, Hydrogen And ELectricity generation program with 33 partners or organizations from 10 countries (Belgium, Czech Republic, France, Germany, Italy, The Netherlands, Slovak Republic, Spain, Switzerland, UK), and it currently has 8 sub-projects. The CARBOWASTE program aims to develop a solution for integrated irradiated graphite waste management and it has a membership of 28 partners, from the UK, France, Germany, Italy, Spain, Lithuania, Belgium Romania, the Netherlands, Sweden and South Africa. Professor Marsden then presented some examples of nuclear graphite research in the UK, including: statistical analysis for the prediction of brick cracking rates; development of semi-empirical models for use in graphite component structural integrity assessments, including irradiation creep; statistical analysis of installed and trepanned reactor samples providing data on component dimensional change; weight loss and property changes for use in structural integrity assessments; whole core modelling; refuelling trace monitoring (i.e. monitoring the forces involved during refuelling in order to detect any changes in the load traces that may indicate an issue related to graphite component deformation); seismic modelling (numerical analysis and proposed rigs); component life prediction (numerical finite element modelling); inspection methods (crack detection using eddy currents); biaxial testing of graphite strength; and atomistic modeling of defects in irradiated graphite.

  Q&A:
  Q1: Please comment more on the HSE(ND) independent nuclear graphite data (slide # 12) weight loss vs. strength.
  A1: That weight loss refers to radiolytic oxidation, which is not a problem in the HTGR.

  Q2: In the Magnox reactor, does weight loss translate into dust?

A2: No, it goes into the gas. A considerable amount of the coolant gas in both the AGR and Magnox reactors is lost to the atmosphere through leakage, although measurement shows that the activity releases due to the gas loss is insignificant

Q3: In the CARBOWASTE program, which organization is responsible for the program?
A3: This EU project is being managed by Forschungszentrum Jülich (FZJ).[52]

Comment: US has started a project on recycling graphite, the Deep Burn program.

- ***PBMR Research Activities*** by Mr. Mark Mitchell, Chief Design Engineer: CSC, Pebble Bed Modular Reactor (Pty) Ltd – Republic of South Africa (RSA)

  Summary: Mr. Mitchell's presentation included a review of the PBMR and South African design/regulatory framework, followed by a summary of the R&D activities at PBMR, and the graphite reactor safety case model and life cycle. He finished his presentation with a summary of some example activities that are being completed at PBMR and how these integrate with international programs. Mr. Mitchell explained that PBMR fulfils the role of a nuclear vendor and, as such, it focuses on the development of the fuel, the reactor system and its components, and minimization of waste. PBMR's research efforts are directed toward near-term needs and are focused on applications, qualification and characterization. PBMR collaborates with NECSA and other institutions such as universities. PBMR contributes to the Carbon Chair at the University of Pretoria, which was established by the South African Department of Science and Technology, and currently has research contracts with several other universities and companies.

  Q&A:
  Q1: A terminology clarification, is 'safety case' the same as 'safety analysis'?
  A1: An important part of 'safety case' is the 'safety analysis'; I would say that 'safety case' for PBMR is the same as 'licensing documentation' in the US.

  Q2: What are your findings regarding determination of variability of properties (Slide # 12)? Are the differences big?
  A2: There are statistical significant differences at all these levels (between charges, within charges and within billets). The differences are big but not huge.

  Q3: What provisions is PBMR taking to account for these differences?
  A3: Close monitoring during the production process. Keep all data available.

  Q4: What does it take to convince PBMR that it is the same grade of graphite if there are any changes in the manufacturing, i.e., change graphite furnace?
  A4: There are no standard that defines which changes are significant. PBMR will address any deviation from the qualified production route on a case-by-case basis. For instance, PBMR qualifies two furnaces routes. Other changes such as coke source are significant. Pitch changes are not very significant, but pitch quantity changes are significant. Significant changes are coupled with method of re-qualification.

  Q5: Increased inspection may be required to offset uncertainty in the lifetime prediction, due to uncertainty in material response to irradiation. What are the provisions planned for increased inspection to address this uncertainty?
  A5: We have not finalized graphite inspection program yet, but it will include a suitable inspection level to cater for this uncertainty.

- ***Chinese Research Activities*** by Professor Suyan Yu, Institute of Nuclear and New Energy Technology, Tsinghua University – China:

  Professor Yu was unable to attend the workshop and therefore this presentation was cancelled.

- ***Japanese Research Activities*** by Dr. Motokuni Eto, Technical Consultant, Toyo Tanso Co. Ltd – Japan

  Summary: Dr. Eto described the goals of the HTTR Project of JAEA: demonstration of hydrogen production using the iodine sulfide (IS) process, and the development of a commercial HTGR system. Within this framework the following research areas are being pursued on graphite and carbon materials.

  a) Development of micro-indentation technique to evaluate the stress generated in HTGR graphite components.
  b) Development of 3D-Xray technique to observe the microstructural change in irradiated graphite, aiming at modeling of irradiation effects.
  c) Development of in-service inspection techniques and surveillance test for the HTTR graphite components.
  d) Further irradiation experiments are planned for the high-fluence data acquisition

  Q&A:
  Q1: Are PGX and ASR-0RB graphites used in high flux areas?
  A1: No. PGX graphite is used as permanent reflector and plenum block. ASR-0RB carbon is used as thermal insulator at the core bottom.

  Q2: On Slide # 19: How do they plan to release the stress of blocks? Would they cut the whole block or sections?
  A2: X-ray techniques can be applied to relate to residual stress. They have not decided yet whether to evaluate the whole block or just some parts.

  Comment on Slide # 20 (on evaluation of fission product (FP) transport): very good method for FP transport

  Q3: What is the status of IG-430? Will irradiation studies be as extensive as for IG-110?
  A3: There are plans to irradiate IG-430 graphite, but the irradiation plans are less extensive at the moment.

- ***U.K. Regulatory Perspective*** by Mr. Graham Heys, HM Principal Inspector (Nuclear Installations), HM Nuclear Installations Inspectorate (NII), Health and Safety Executive (HSE) – United Kingdom

  Summary: Mr. Heys' presentations included an overview of the regulatory regime in the UK, the regulatory challenges and strategy, the regulatory guidance relating to graphite assessment, safety case improvements, and graphite nuclear safety research (research arrangements, graphite research strategy and current graphite research program). Currently around 20% of UK electrical power supply comes from nuclear power generation: 14 Advanced Gas Cooled Reactors, 4 Magnox reactors and 1 PWR; all of these reactors but the PWR, are graphite-moderated. Mr. Heys described briefly the AGR graphite core and brick design; as well as the Magnox graphite core. He then described the regulatory regime in the

U.K., based on the Health and Safety at Work Act (1974) and the Nuclear Installations Act of 1965, as amended. For graphite-moderated reactors, the graphite core safety functions include: shutdown and reactivity control post shutdown; fuel cooling; and, fuel integrity and unimpaired refueling. The challenges to core safety functions have arisen from observations of unpredicted graphite moderator brick cracking in AGRs (pre stress reversal), predictions of cracking in AGRs (post stress reversal), and high radiolytic weight loss. He presented several examples of cracks found on graphite bricks. He then, explained the HSE-coordinated nuclear safety research program. Finally, he summarized the current work undertaken by licensees and the NII, on several technical areas including:

a) Graphite core material properties;
b) Core component integrity;
c) Whole core structural response;
d) Inspection, sampling and monitoring; and,
e) Regulatory access to independent advice

Q&A:
Q1: On the Regulatory Regime (Slide # 13), how is that different from the one in the U.S.?
A1: The U.K. regulatory system is non-prescriptive. The U.S. has an equivalent licensing regime that is prescriptive.

- ***RSA Regulatory Perspective*** by Mr. Schalk Doms, Senior Regulatory Officer, PBMR Programme, National Nuclear Regulator – RSA

Summary: Mr. Doms presentation included a description of the South Africa's regulatory framework, a summary of the graphite requirements, and an overview of the current research work conducted by the NNR. In July 2000, the NNR received a Nuclear Installation License (NIL) application from Eskom for a PBMR Demonstration Power Plant. The NNR has adopted a multi-staged licensing process, which includes the following licensing stages:

a) Acceptance of Concept Safety Case;
b) Site preparation, Construction and Manufacturing Phase;
c) Fuel on Site, Fuel Loading, Testing and Commissioning;
d) Plant Operation; and,
e) Decommissioning

The NNR has also developed a Requirements Document, LD-1097: "Qualification Requirements for the Core Structure Ceramics (CSC) of The Pebble Bed Modular Reactor". This licensing document (LD) stipulates the requirements for the qualification of the CSC materials and structures, and the quality control related to the manufacturing processes of the CSC components. It also covers the requirements and recommendations for surveillance of the CSC from the construction stage up to the decommissioning of the plant.

Current research work by the NNR includes developing of its own model on the irradiation behavior of graphite. The NNR is in the process of developing a position paper on graphite waste taking into account the current international status of graphite waste management. The first stage of the process is to gather all available information and compile an overview report on international requirements, approaches, and positions on graphite waste minimization, management, and disposal.

Q&A:

Q1: On slide 12, what types of things are recommendations vs. required?
A1: The list of materials properties are recommendations.

Q2: The NNR position paper on graphite waste, is it just internal to NNR?
A2: Yes. However, NNR also participates in CARBOWASTE

- ***Chinese Regulatory Perspective*** by Professor Suyan Yu, Institute of Nuclear and New Energy Technology, Tsinghua University – China

  Professor Yu was unable to attend the workshop and therefore this presentation was cancelled.

- ***Japan Regulatory Perspective*** by Dr. Motokuni Eto, Technical Consultant, Toyo Tanso Co. Ltd – Japan

  Summary: Dr. Eto's presentation covered the Japanese regulatory perspective for HTGR. It is considered that the Japanese regulatory process for the design and construction of future HTGRs will make use of the knowledge and experiences accumulated during the design and construction of the HTTR. To expand the rule for the HTTR into a more generalized standard for future HTGR graphite components, a special committee had been set up in the Atomic Energy Society of Japan to prepare 'Standard for HTGR Graphite Components (draft)', which was completed in March 2009. At the same time the method in which the lower irradiation fluence data are extrapolated to higher fluence has been discussed as to various properties and summarized as a research report. It is important to align this effort with the ASME activities.

  Q&A: no questions were asked after this presentation.

- ***NRC Regulatory Research Perspectives Related to NGNP V/HTGR Licensing*** by Dr. Stuart Rubin, Senior Technical Advisor, Dr. Sudhamay Basu, Senior Nuclear Engineer and Dr. Makuteswara Srinivasan, Senior Materials Engineer, U.S. NRC.

  Summary: Dr. Rubin opened his presentation with an overview of the 2005 Energy Policy Act in which Congress required a NRC-DOE coordinated NGNP licensing strategy. He explained the role of the NRC in the licensing of the NGNP and the importance of NRC independently developing its own analytical tools to verify the NGNP design and its safety performance. In the NGNP Licensing Strategy Report to Congress[56] various graphite safety R&D issues were identified. Dr. Rubin also explained the role of the NRC/NGNP regulatory research, including:

  - Develop NRC staff knowledge, expertise, capabilities and review guidance
  - Independently confirm technical basis for requirements and criteria
  - Develop NRC independent analytical capabilities
  - Confirm or interpret technical information involving significant uncertainty
  - Validate/scope-out technical issues to justify request for follow-up resolution by the applicant

  Dr. Rubin concluded with a summary of NRC's NGNP V/HTGR graphite R&D plans, which includes:

[56] http://www.ne.doe.gov/pdfFiles/NGNP_reporttoCongress.pdf

- Support codes and standards development
- Conduct graphite workshop
- Participate in international irradiations
- Develop independent evaluation capability
- Develop capability to predict failure probability
- Conduct selective R&D to support regulatory decisions
- Support NRC HTGR accident evaluation model development

Q&A:
Q1: About Slide #14, do you know what are you getting into with the mechanistic source term calculation? This is an extremely expensive approach and I have the feeling that it is a largely unnecessary.
A1: I do not disagree. DOE is aware of the price tag. Also, besides the cost, time and uncertainty are issues. The question becomes 'how do you define mechanistic?'

- ***Some of the Challenges in NGNP HTGR Graphite Component Safety Evaluation*** by Makuteswara Srinivasan, Senior Materials Engineer, NRC

  Summary: Dr. Srinivasan initiated his presentation by outlining the materials-related challenges for NGNP HTGR safety evaluation and explaining the various integrating predictive models used as input for regulatory decision. The models are:

  a) Graphite degradation model
  b) Graphite (structural) component integrity model
  c) Graphite inspection model
  d) Contribution to risk (normal, AOO, accident)
  e) Risk assessment model
  f) Integration into regulatory decision

  After explaining how each one of these models works, he summarized the challenges in developing consensus codes and standards, including performance acceptance criteria, in-service inspection and surveillance requirements. He emphasized the importance for NRC to develop the staff expertise, technical tools and data to support an effective and efficient independent safety evaluation of the NGNP HTGR graphite components.

  Q&A:
  Comment: On Slide # 6, Influence of graphite behavior on risk assessment. The input MTR data, is at room temperature, but we need a validation mechanism or data at temperature and irradiation conditions

  Q1: Is failure related to the presence of a crack or inability to fulfill a function?
  A1: From a regulatory perspective, the applicant has to define what constitutes degradation, particularly with respect to the factor of safety used in the design during all stages of reactor operation. The degraded condition should be able to be monitored to assure that design assumptions and structural integrity of graphite components during reactor operation is still maintained and other functional requirements used in design assumptions are successfully met.

  Q2: The model shown on Slide #6 looks like a complicated model; what rules does NRC have to deal with such a complex model?

A2: Graphite core components are passive components and subjected to degradation in a complex manner in the gas cooled high temperature reactor. This apparently complicated model to address the potential reduction in operational safety margin due to passive component degradation is really split into models of individual contributing elements to the overall probability of failure. In many cases, the degradation itself is difficult to determine during operation or shut down inspections. However we need to keep in mind that 'absence of evidence' is not necessarily 'evidence of absence'. Thus the uncertainties involved in estimations of the probability of failure need to be considered in a comprehensive manner and the time-dependent change in the assumptions involved in estimating such probabilities need to be better understood, and considered in component assessments. The governing rules are stated in the Appendix A, "General Design Criteria", of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities".

Q3: What is considered to be confirmatory research by the NRC?
A3: Confirmatory analysis of graphite behavior in HTGR is recommended when:

1. Consensus codes and standards for stress analysis are not available;
2. The applicant is or will be submitting a new methodology or approach, consisting of complex, interacting, and iterative computer codes, which is unfamiliar to the staff;
3. The applicant's key parameters have not been determined to the level of adequacy deemed necessary due to insufficient data, inadequate or questionable extrapolation of data, imprecise modeling, and large uncertainties in predicting reactor environment, namely the temperature and stresses in graphite bricks;
4. Relatively narrow design margins (factor of safety) are used for temperature and stress distributions for graphite components for assumed probability of failure or fracture; and,
5. The prediction and estimation of the changes in design margins, as a function of reactor life, can not be made to the degree required by modeling, and when operating experience and inspection data may not be available well into the future to support any assumptions used in the design.

- ***Comparison of Graphite PIRT Results with DOE Research Plan*** by Dr. Timothy Burchell, Group Leader, Carbon Materials Technology Group and Dr. Nidia Gallego, Research Scientist, Carbon Materials Technology Group, ORNL

Summary: Dr. Burchell discussed the NRC-sponsored research performed at ORNL, in which the NRC Graphite PIRT was compared with the DOE's planned research activities and presented a summary of the findings. He reviewed in detail the five phenomena ranked I-H, K-L, the nine phenomena ranked I-H, K-M, the two phenomena ranked I-M, K-L and seven of the phenomena ranked I-M, K-M. He concluded his presentation with a list of 11 research areas recommended by ORNL:

a) Oxidation modeling capability;
b) Accelerated development of ASME code for graphite core components;
c) Graphite tribological behavior in helium;
d) Oxidative reactivity of graphite dust powder compared to graphite blocks;
e) Enhanced analytical modeling and predictive capability for irradiation induced dimensional change and creep;
f) An accepted fracture criteria for nuclear graphite;
g) An accepted in-service inspection method for graphite core components;
h) Overall graphite degradation (prediction) model (GDM)
i) A graphite core stress analysis method;

j) The potential for stored energy release in irradiated graphite exposed to high temperatures during reactor accidents; and,
k) Knowledge needs for graphite decommissioning

Q&A:
Q1: Would it be possible to specify a set of properties to graphite manufacturers instead of worrying about changes on coke source?
A1: The problem is that there is a lack of fundamental understanding of the relationship between graphite structure and irradiated properties

4. Panel Discussion Results

Graphite qualification and characterization
- INL plans, and vendors plan
- Comments on plans
  - Adequacy of properties and database
- Quality assurance requirements

Panel Member	Comments
Burchell	Adoption of an ASME code and ASTM standard is needed in near term Need to identified which reactor concept will be chosen
Wichner	No comment
Marsden	Property database is not sufficient in itself. There is a need to understand what microstructural features determine the property values of unirradiated graphite, and what the implications are when graphite microstructure is changed by irradiation. NRC needs to be assured that qualification is based on sound science and that captures irradiation behavior
Heys	There is a need for improved understanding of structure / property relationships for nuclear graphites, in particular fundamental investigation of the effect of manufacturing process parameters on development of microstructure and material properties. QA is very important (QA vs. QC). Develop scientific knowledge of structure/property relationships in the longer term.
Eto	There is a collaboration established between INL and JAEA to 'share' irradiation data NRC needs to develop arrangements and international collaborations (Codes and irradiation properties)
Doms	Need to combine safety culture with QA
Bratton	Participation of NRC when INL is conducting NQA and Audits NRC needs to be a participant on ASME and ASTM activities Qualification for purchase vs. qualification for design
Windes	NRC must develop their own independent expertise to be able to properly evaluate the data they will receive from DOE NRC must be involved in the NQA process NRC needs to be involved in ASME and ASTM activities (beyond as an observer; needs to be an active participant) to develop codes and standards for licensing NGNP
Penfield	NRC needs to be involved in ASME and ASTM activities Develop criteria for acceptability of flaws (criteria for acceptability of flaws in service should be related to the function of the component) In defining program priorities, we should distinguish between 'musts' for an initial reactor vs. 'needed' or "desired" for a fleet of reactors.

Mitchell	Definition of "qualification" is important. For the purposes of this discussion I propose it means Characterization. Specifically, determining both the virgin material properties and the effect of irradiation on properties. This begs the question, which properties are required. NRC needs to be in agreement with what the necessary requirements for qualification are. It is anticipated that for Graphite Core Assemblies and Components, the requirements will be captured in the ASME code. Therefore, NRC participation in ASME/ASTM activities is key. Graphite qualification is a vendor responsibility, so the NRC focus should be standardization in the industry. This is already an important part of the NRC's activities. In terms of the supplied NGNP plans, they appear adequate. The weakness is that the approach to graphite qualification is not completed to an overall standard so there is no guarantee that it is of adequate scope or directly comparable to other programs results. It would be ideal if international data were all generated to the same minimum standard. This would allow for direct comparison between the results of the various programs.

Summary:

- NRC needs to have/develop the ability to understand/interpret data brought to them by applicants
- Active NRC involvement in ASME code and ASTM standard development is desirable and encouraged to get early input in the comment/approval process

Requirements for core behavioral models

- Irradiation properties
- Models for fundamental understanding for structural integrity analysis
- Handling of data and model uncertainties

Panel Member	Comments
Burchell	There is not an 'agreed' model for creep or an agreement on mechanism. R&D on fracture criteria is needed.
Wichner	A set of component failure criteria should be developed for structural graphite components. The failure criteria, when adopted into the whole core model, would define core life.
Marsden	As well as validation of techniques used for testing materials properties both in the irradiated and unirradiated condition, validated methodologies are required for using that data for predicting changes at reactor operating conditions i.e. as a function of irradiation temperature and fast neutron fluence There is evidence that irradiation creep recovery on loading is greater than is predicted by current models. Therefore, load and unloading irradiation creep data and microstructural studies are required to develop new mechanistic based creep models taking account of the unloading behavior. Failure models and data should be devised that can be used in conjunction with probabilistic stress analysis finite element based codes to estimate the life of graphite components. Failure data and techniques that can only be used for comparisons of different graphite grades are of limited use in core component assessment.
Heys	Irradiated properties: • There is little or no data on effect of test temperature on irradiated properties. Most properties are historically measured at room temperature. This data needs to be complemented by data at irradiation conditions as damage processes may be sensitive to temperature.

	• The effect of biaxial or multi-axial loading on irradiation creep behavior may be important to prediction of component integrity as irradiated graphite components are subject to complex stress states. • Fundamental research on fracture behavior for unirradiated and irradiated graphite is necessary to validate component and whole core models and improve understanding of the effects of ageing processes on component integrity. There is a need to improve understanding of the effect of biaxial loading on fracture (at temperature and after irradiation). • Effects of stress concentration features, such as keyways, on fracture initiation is still not adequately understood for irradiated or indeed unirradiated material. • Effect of component size effects on irradiated material properties is worthy of further investigation. • Measurement of Poisson's ratio of irradiated material continues to present challenges and this needs to be resolved. Modeling of irradiated properties. : • There is a lot of work being undertaken in the UK. It would be beneficial to develop an international consensus on mechanisms of graphite ageing processes to inform materials models and predictive trends. Validation of deterministic and probabilistic methodologies • There is a potential opportunity for validation of structural integrity models by PIE of graphite components from decommissioned reactors or in-reactor validation, such as internal stress, component failure criteria, etc. However, this will be expensive and is only likely to happen through international collaboration. The UK is working towards improved understanding of cracking observed in the AGR stations and there is perhaps an opportunity to validate probabilistic methodologies using operating reactor data.
Eto	Conducting experiments to generate data for validation of models and equations is strongly suggested. JAEA analyzed wide varieties of existing data to derive equations but still needs much more data especially for high fluence high temperature irradiations. NRC to support the analyzing activities as well as the experimental program.
Doms	Validation of models specific for seismic activity
Bratton	V&V models needed NAFEMS models or analytical models
Windes	Data/information gaps being addressed by DOE research program: 1) Understanding fundamental mechanisms: 2) Validation/confirmation data 3) Computer whole core models
Penfield	No comment
Mitchell	For the purpose of this discussion, we identified whole core behavior modeling as comprising multiple points. – (A) Calculation of load inputs (Temperatures, fluence etc.) – (B) Response of the material to irradiation – (C) Integration of material response into a part stress analysis model (Stress Analysis) – (D) Comparing the stresses in the parts to the limits (Stress Evaluation) – (E) Integrated assessment, of the parts assembled to form the Graphite Core

	Assembly. Some specific comments on the various steps are:- (B) Response of the material to irradiation. This needs to be supported by modeling of the materials. This allows us to build upon the empirical knowledge that will be gained from the vendor's experimental programs with a basic understanding of materials properties and how it affects the performance of the material in reactor. (C) Stress analysis: NRC needs to consider the following aspects of the prediction of stresses within the graphite core components: • Requirements for conservatism, what is adequate conservatism for a stress prediction? This should be considered in conjunction with the reactor vendors. (D) NRC needs to complete enough independent research to establish a position on what the suitable acceptance criteria for the stresses in a graphite core component should be. (E) Whole core modeling: NRC needs to establish positions on the minimum requirements for integrated core assessment. In addition, as it is not possible to eliminate the possibility of cracking of the GCC, it must be demonstrated that cracking of components does not result in an unacceptable state in the assembly. Thus, the techniques and approach to damage tolerance assessment need to be defined and accepted by the NRC. There is room for research and development in this area.
Summary: • Needs data to validate models. Opportunity to do PIE on discharged components (this is an extremely expensive task. Requires a major commitment. May need international collaboration to do this). Component failure criteria should be developed and applied in core models.	

Oxidation of graphite by coolant impurities	
Panel Member	**Comments**
Burchell	Long term degradation by He impurities on strength for structural components. Independent models (GERSAC) to be developed
Wichner	NRC should have an activity on lifetime degradation of the core support area • Verification of oxidant environment (ingress, amount, distribution of impurity) • Assessment of effect of variation of temperature environment on substructure • Normal operating condition oxidation rate • Oxidation penetration prediction • Verification of strength loss with degree of oxidation Dust production and characterization from graphite; disposition of dust throughout the system needs to be determined.
Marsden	No comments
Heys	No comment
Eto	NRC needs to gather historical data from previous oxidation studies; initiate an activity for a review of existing international data. In parallel to the activities to grasp the general trend of oxidation behavior of HTGR graphites, there should be confirming oxidation experiments on candidate graphites from the aspect of safety.

Doms	No comment
Bratton	Basic oxidation/adsorption along in a tube needed
Windes	A current research area/gap being addressed by DOE-NGNP program (helium impurities, not air ingress). Potential need for a long term research program to identify potential chronic, low-level degradation issues in graphite. This is being pursued by NGNP program as a long-term research objective (i.e. university work).
Penfield	No comment
Mitchell	PBMR has a small program on Normal Operation oxidation. This is focused on estimating the extent of oxidation as well as the effect of this on the operation of the graphite core. This is of operational significance and should not affect the safety of a HTR. It is probably better to focus regulatory research on oxidation during the LBEs, rather than on the normal operation case.
Summary: • There is a need for a full and comprehensive literature review. The issue of creation (if any) of dust needs to be addressed. • NRC needs independent modeling work for oxidation (normal operating and accident conditions).	

Status of codes and standard development / future challenges
– Design and construction code (Section III)
    – Stress analysis
    – Adequacy of margins
– In-service inspection (Section XI)
    – Online monitoring	

Panel Member	**Comments**
Burchell	Accelerate codes & standards activities. Encourage NRC participation in this activities ISI considered a gap. NRC should be able to make informed decisions on applicants' stress analysis. Need a defined failure mechanism.
Wichner	Component failure criteria need for structural graphite components.
Marsden	AGR and Magnox reactors measure channel bore dimensional changes and, in some Magnox reactors, overall core shrinkage is recorded. This data is used to ensure that the core is fit for purpose and the deformations are compared to predictions made using MTR graphite data. Similar techniques should be considered in HTR design. Stress analysis codes based on various graphite constitutive models have been developed in the UK. Predictions from these codes are compared with failure criteria to predict component life. It is important when developing these methods that they are applied using an appropriate probability assessment methodology.
Heys	On-line monitoring of AGR and Magnox reactors has been optimized over recent years in response to ageing processes and findings from core inspections. Online monitoring includes: freedom of movement of fuel and control rods and rod drop trending; channel outlet gas temperature monitoring or channel power; 3D models relate ISI and other observations to give a holistic view of core ageing phenomena. ISI at AGR and Magnox reactors includes: channel bore measurements to monitor brick shape development and whole core shape changes (including channel tilt and bow); remote visual inspection using TV camera examination during outages; and trepanning of graphite samples enable tracking of graphite property changes and

	radiolytic oxidation. Work is in-hand to deploy an eddy current device at periodic shutdowns to enable volumetric inspection. There is a need to embed ISI and on-line monitoring into the design. For example, it would be desirable to: establish where, when and how to trepan graphite samples; consider the need for inclusion of surveillance samples, their location in the core and the types of samples; consider if gas flow measurements would be beneficial; consider the required coverage by thermocouples; techniques and methods of ISI. If incorporated at the design stage this will enable optimization so that the benefit from these activities may be maximized to provide improved tracking and understanding of ageing processes. Codes and standards should address whole core modeling. NGNP needs to develop techniques for ISI, on-line monitoring, surveillance and sampling at the design stage to permit development and implementation of optimized methods and techniques.
Eto	In the ISI to be done in the HTTR, camera to inspect core support posts will be used, which would give a lot of valuable information as to the surface characteristics of graphite posts.
Doms	No comment
Bratton	ASME Code, Section 11 activities. NGNP Engineering trade study may be relevant. Lifetime of components depends upon design, prismatic block not highly irradiated.
Windes	ISI technique development is being pursued by DOE-NGNP program. New ISI techniques need to be developed and investigated. Current ISI and monitoring technologies used for other reactors need to be assessed for use in the NGNP.
Penfield	Codes must not be design specific
Mitchell	ISI not the only need. Online monitoring may be more effective (He leak detection for example). ISI not well treated in RES plans. Maybe a timing issue. The ASME work on Section XI, Div 2 should be expanded to include graphite. The timing of this needs to be determined. This will allow for a Reliability and Integrity Management program to be established for the graphite, where all sources of information regarding the performance on the reactor components are combined to provide adequate confidence in the ongoing functional reliability of the structure. This should also consider the role of graphite in the safety case of the HTRs. As the NRC is instrumental in accepting the safety case, it is critical that the NRC take a proactive interest in this field. NRC research should support the development of such standards and the NRC's ability to endorse them. The issues with the development of the construction code, specifically the graphite assessment and evaluation methods, are included in the discussion on "core behavioral models" above.
Summary: • No specific NRC R&D in the ISI area; however, DOE plans will cover needed R&D. NRC staff needs to participate in codes and standard development (especially Sect 11) • Could include coupons, trepanned samples, etc. • Failure criteria are needed for structural components.	

Tribology and Oxidation leading to graphite dust, F-P transport	
Panel Member	**Comments**
Burchell	There is a gap in information related to wear, dust formation, and transport. Impact of operating environment on dust formation is needed
Wichner	Dust is the principal vector for release during accidents of many types of the fission products. Therefore, dust production during normal operation and accidents is an important part of GCR safety analysis. It was generally agreed by the panel that the graphite area should assist the fission product transport effort by providing information on carbonaceous dust production. It is also necessary to characterize the dust in order to predict transport and deposition. It is noted that that there are several dust production mechanisms (including tribology) and also non-carbonaceous dusts. The degree of association of fission products with dust may be most effectively dealt with as an integral part of the dust formation effort.
Marsden	No comments
Heys	Helium purity levels and potential effect on graphite need to be considered.
Eto	There are some data on desorption and permeability of gas in graphite, but almost no data are available regarding graphite dust.
Doms	No comment
Bratton	There are journal papers on the subject that should be summarized for the NRC
Windes	This issue seems to be driven by the F-P transport group. While graphite dust may provide a ready vector for fission product release the real issue is the amount of F-P that are available to be transported out of reactor; i.e. the source term. As discussed, the amount of dust is not the issue (small amounts can still carry out large activity) so the main issue is how much FP is available to be carried out. Currently, there is no tribology R&D in the NGNP program.
Penfield	F-P should have the lead for this issue. We can help with dust formation (tribology and oxidation) and request feedback from F-P community.
Mitchell	Single most important GAP in information that NRC can pursue is the FP on graphite dust and its contribution to the FP source term. How much FP does dust carry? Quantification of the amount of dust and the inventory of FPs is needed. NRC R&D focused in this area will be important to support the NRC licensing of any HTR reactor concept.
Summary: • Graphite community needs to address the issue of carbonaceous dust formation (tribology plus other mechanisms). Although the FP community will take the lead in assessment of dust vectoring of fission products, dust formation studies may be the most effective means for dealing with FP association with dust. FP community needs to explore/address the issue of dust activity and transport.	

Air ingress and water ingress (accident)
Safe shutdown and cool down

Panel Member	**Comments**
Burchell	Kinetics data is not sufficient, and NRC must also understand the diffusion and mass flow behavior.
Wichner	The air ingress accident could become a focus of attention, equivalent to the LOCA for the LWRs. Therefore, all aspects need to be fully evaluated. Public perception could bring up, in an unschooled way, the possibility of graphite burning (as in conflagration, or flame) during an air ingress event. A forceful response should be developed in anticipation of this public issue, The possibility of core burning in the pebble bed concept should especially be positively dealt with. CO production and possible burning is the approximate equivalent of H_2 production and burning in the LWR LOCA. Experience indicates that this issue could be a focus of close public (and perhaps ACRS) attention. I would recommend the following: •Review CO production correlations and data. There have been studies of CO production, but the resulting correlations have improbable features from a chemical point of view. Therefore, the field should be reviewed, as a minimum. Possibly additional experiments may be called for focusing on accident conditions. •Modeling exercises should indicate whether or not CO accumulation to flammable or explosive concentrations can occur during an air ingress accident. There should be assurance that the oxidation rate and oxidation penetration correlations apply to the conditions of the range of air ingress accidents. The extent of dust production due to air ingress should be clearly defined. Air ingress could introduce dust production mechanisms not present in normal operation. The behavior of the dust during air ingress should be predicted, i.e., in what way does it interact with events, (There already has been an inquiry on the possibility of dust explosions, possibly motivated by a rough equivalency with possible H_2 explosions in the LWR LOCA.) Similar considerations arise for H_2O ingress events. The rate of H_2O oxidation and oxidation penetration needs to be measured under ingress conditions. Information on dust production during moisture ingress should be obtained. Flammable gas production during moisture ingress should be assessed, but is less significant than for the comparable air ingress. There may be no studies on CO production from moisture comparable to air. In addition, there is the question of H_2 production by reaction of CO with H_2O in the water shift reaction.
Marsden	There is information in the UK on the Windscale fire and Magnox air ingress assessments that may be useful for validation of codes developed for HTR applications.
Heys	No comment
Eto	JAEA performed simulation experiments on air ingress as part of licensing HTTR.

Doms	No comment
Bratton	Panel members interested on this topic should consult ASME for various documents they have published in this area.
Windes	NGNP conducting kinetics experiments on oxidation during accident scenarios.
Penfield	No comment
Mitchell	In addition to Dr Wichner's comment it must be added that experiments have been completed in this field. Specific examples are the NACOC experiment sequences in Germany at FZ Jülich. Some of he results are published publicly while other experimental series are proprietary. These should be used to increase the understanding of the integration of the phenomena as they occur and interact in the transient and provide validation of the codes developed and deployed in the assessment of these LBEs. A possible target for NRC R&D in the area is the modeling of the events to determine the driving phenomena and support the staff in reviewing applications which address these events.
Summary: • NGNP design evolving (steam generator) therefore steam ingress should be considered. NRC staff needs to be aware of activities on oxidation. NRC need to do independent analysis • NRC needs independent oxidation modeling capability	

Defining end of core-component life (criteria and safety margins)	
Panel Member	**Comments**
Burchell	Licenses application – period of operation. The replacement schedule for graphite components must be considered; strength vs. stress; failure criteria.
Wichner	Component failure criteria need to be developed for the key graphite structural components.
Marsden	Component failure may not signify the end of core life. Assessment should be based on fitness for purpose, with the new HTR designs there is the possibility to design tolerance to failure into the graphite structure.
Heys	It would be desirable for NRC to consider the effect of uncertainty and complexity of predictive models and data inputs in determining the adequacy of safety margins to secure defense-in-depth.
Eto	JAEA is now developing a micro-indentation technique to estimate stresses generated within a graphite component. This may be utilized to evaluate the end of core component life.
Doms	No comment
Bratton	End of life depends on vendor design which is not known at this time.
Windes	NGNP has extensive program to indentify property degradation over component life. Data may not currently be available but is in the process of being acquired.
Penfield	Program to support plant life, data for planned operating period.
Mitchell	Margins defend against failure. How to define margins. Need to establish logical framework for NRC to endorse. Risk based assessments This is an extension of the discussion on "core behavioral models" above.

Summary: • NRC staff must be aware of complexity and provide input to question of how to define adequacy of data and margins

Decommissioning and disposal	
Panel Member	**Comments**
Burchell	DOE deep burn is addressing issues related to determining whether graphite blocks can be reused. Current NGNP program irradiations do not cover reuse of graphite, higher dose experiments are needed
Wichner	Attention is called to the C14 data measured in the Peach Bottom HTGR Surveillance Program. In addition, tritium distribution measurements are reported. Refs: "Distribution and Transport of Tritium in the Peach Bottom HTGR", R.P. Wichner and F.F. Dyer, ORNL-5497, August 1979.
Marsden	There is a new 12m€ European Commission FP7 framework program (CARBOWASTE) to address the issues of graphite disposal including recycling as well, volume reduction and deep geological or shallow disposal
Heys	No comment
Eto	There is a need for a long term disposal (or reuse) policy for HTGR graphite. To prepare a long term R & D program, comprehensive survey on the past and present activities is necessary.
Doms	No comment
Bratton	Vendors are being solicited by NGNP Deep Burn project on this subject. Should be revisited later
Windes	NGNP has a program on the area of decommissioning. DOE Deep burn program addressing graphite reuse and recycle. Particular activities/research will become clearer when a specific reactor design is selected. Decommission and disposal issues are being addressed as much as possible, however we anticipate a huge gap.
Penfield	Interested in the longer term graphite irradiation data so that life of graphite reflector blocks could be extended.
Mitchell	In South Africa: Disposal of the waste is anticipated to occur after approximately 60 years, or when a suitable repository becomes available. It is anticipated that the waste will be classified as Low/Intermediate level long-lived (LILW-LL), due to the fact that the primary activity is in the form of 14C (half-life 5700 years) with a specific activity from long-lived β emittance > 4000 Bq/g, and minimal heat generation is expected within the material. This position or those like it need to be considered by the NRC for compatibility with the waste requirements in the US. In addition, South Africa participates in the European Carbowaste program where among other issues we are studying possible options for reducing the volume of waste from the graphite.
Summary: • Collaboration and exchange of information with the European Commission FP7 Carbowaste program would be beneficial.	

5. Other themes

1. International collaborations (G. Heys)

- Existing collaboration forums (IAEA, GIF, INGSM) do not provide demonstrable opportunity for discussion of regulatory issues.

- Meet with INGSM:
- Session in meeting

2. INGSM should publish proceedings (Marsden)

3. OECD/ NEA activity on regulatory issues including HTGR (Eto). MDEP, Multinational Design Evaluation Program, http://www.nea.fr/mdep/welcome.html

The MDEP program incorporates a broad range of activities including:

- An enhanced multilateral cooperation within existing regulatory framework
- A multinational convergence of codes, standards and safety goals
- Implementation of MDEP products to facilitate licensing of new reactors, including those being developed by the Generation IV International Forum

Attachment 2

Technical Presentations

- “Nuclear Graphite Workshop: Welcome” – Dr. Brian Sheron, Director of Research, Office of Nuclear Regulatory Research, NRC.
- “A Short History of NRC Nuclear Graphite Research” – Dr. Makuteswara Srinivasan, Senior Materials Engineer, Office of Nuclear Regulatory Research, NRC.
- “Workshop Overview and Goals” – Dr. Nidia Gallego, Research Scientist, Carbon Materials Technology Group Oak Ridge National Laboratory (ORNL).
- “Overview of NRC Graphite PIRT Findings” – Dr. Timothy Burchell, Group Leader, Carbon Materials Technology Group, ORNL.
- “NGNP Graphite Program” – Dr. William Windes, Leader Graphite Group, Idaho National Laboratory (INL).
- “UK and European Research Activities” – Professor Barry Marsden, School of Mechanical, Aerospace and Civil Engineering, The University of Manchester – UK.
- “PBMR Research Activities” – Mr. Mark Mitchell, Leader Materials Group, Pebble Bed Modular Reactor (PBMR) (Pty) Ltd – Republic of South Africa (RSA).
- “Japanese Research Activities” – Dr. Motokuni Eto, Technical Consultant, Toyo Tanso Co. Ltd – Japan.
- “A U.K. Regulatory Perspective” – Mr. Graham Heys, HM Principal Inspector (Nuclear Installations), HM Nuclear Installations Inspectorate, Health and Safety Executive – U.K.
- “South African Perspective on Nuclear Graphite Qualification and Manufacturing” – Mr. Schalk Doms, Senior Regulatory Officer, PBMR. Programme, National Nuclear Regulator – RSA.
- “Japanese Regulatory Perspective” – Dr. Motokuni Eto, Technical Consultant, Toyo Tanso Co. Ltd – Japan.
- “NRC Regulatory Research Perspectives Related to NGNP V/HTGR Licensing” – Dr. Stuart Rubin, Senior Technical Advisor, NRC.
- “Some of the Challenges in NGNP HTGR Graphite Component Safety Evaluation” – Makuteswara Srinivasan, Senior Materials Engineer, NRC .
- “Comparison of Graphite PIRT Results with DOE Research Plan” – Dr. Timothy Burchell, Group Leader, Carbon Materials Technology Group, Carbon Materials Technology Group, ORNL.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Nuclear Graphite Workshop

Dr. Brian Sheron, Director
Office of Nuclear Regulatory Research

Welcome
March 16, 2009

U.S.NRC

Nuclear Graphite Workshop

- Welcome
- The NRC/RES Mission
- Graphite Reactor Experience
- Workshop Objectives

2

U.S.NRC

The NRC Mission

- License and regulate the Nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of **public health** and **safety**, promote the common defense and security, and protect the environment.

3

U.S.NRC

RES Mission

- Further the regulatory mission of the NRC by providing technical advice, technical tools and information for identifying and **resolving safety issues**, making regulatory decisions, and promulgating regulations and guidance. RES conducts **independent experiments** and analyses, develops technical bases for supporting realistic safety decisions by the agency, and prepares the agency for the future by **evaluating safety issues** involving current and new designs and technologies. RES develops its program with consideration of Commission direction and input from the program offices and other stakeholders.

4

Graphite Reactor Experience

- Peach Bottom Unit 1
- Fort St. Vrain
  – Stuck Control Rod
  – Moisture intrusion problems
  – Cracks in steam generator tubes

5

Workshop Objectives

- Learn about graphite research activities being conducted and planned around the world
- Consider this in addition to DOE plans for NGNP
- Compare these to the NRC PIRT
  – Identify areas where the NRC may conduct research

6

U.S.NRC

Protecting People and the Environment

- Thank you for participating

7

A Short History of NRC Nuclear Graphite Research

ORNL/NRC Workshop on Graphite Research
Rockville, MD, U.S.A.

Dr. Makuteswara Srinivasan
Senior Materials Engineer
Office of Nuclear Regulatory Research
March 16, 2009

1

Graphite R&D Objectives

- Develop scientific information to establish independent technical bases for development of staff regulatory positions and guidance documents to enable safety decisions on graphite and composite materials used in HTGRs; address uncertainty in behavior of graphite under HTGR environments.
- Use research results to confirm materials specifications, codes, and standards and to provide information and data for NRC HTGR EM (graphite dust) and for evaluating HTGR PRAs.

2

Background - 1

- **The NRC sponsored research at the Franklin Institute Research Laboratories, Philadelphia, PA during mid-1970s.**
  - **V. Svalbonas, T.C. Stilwell and Z. Zudans, "Rules for Design of Nuclear Graphite Core Components – Some Considerations and Approaches", Nucl. Eng. & Design, 46, 313-333 (1978).**

    Identified Research Requirements:

    1. **Oxidation-strength models for graphite;**
    2. **Seismic testing and analysis correlations;**
    3. **Fatigue behavior, and description by fracture mechanics procedures; and**
    4. **Further correlative experiments involving stress categories and associated limits and failure criteria.**

3

Background - 1

- **The NRC sponsored research at the Brookhaven National Laboratories during mid-1980s.**
  - **D.G. Schweitzer, D.H. Gurinsky, E. Kaplan, and C, Sastre, "A Safety Assessment of the Use of Graphite in Nuclear Reactors Licensed by the U.S. NRC", NUREG/CR-4981 (ML061990375), September 1987.**
    - **Evaluated the potential for graphite fire in U.S. research reactors and Fort. St. Vrain reactor.**
      - **Considered stored energy contributions.**
      - **Established necessary conditions of geometry, temperature, oxygen supply, reaction product removal, and a favorable heat balance for self-sustained rapid graphite oxidation.**
      - **Concluded that no credible potential for a graphite burning accident in reactors analyzed.**

4

Background - 1

- **The NRC sponsored research at the Brookhaven National Laboratories during mid-1970s.**
  - **H.B. Palmer, M. Sibulkin, R.A. Strehlow, and C.H. Yang, "An Appraisal of Possible Combustion Hazards Associated With A High-Temperature Gas-Cooled Reactor", BNL-NUREG-50764, March 1978.**
    - **Studied combustion hazards resulting from a primary coolant circuit depressurization followed by water or air ingress into the prestressed concrete reactor vessel (PCRV).**
    - **Studied reactions between graphite and steam or air which produce the combustible gases H_2 and/or CO; when mixed with air in the PCRV, flammable mixtures may be formed.**
    - **Possible circumstances leading to these hazards and the physical characteristics related to them were delineated and studied.**

5

Background - 2

- **The NRC held a high-temperature gas-cooled reactor safety and research issues workshop, Rockville, MD, October 10-12, 2001 and during October 22-23 (2002).**
- **During 2001 – 2002, the NRC conducted a pre-application review of technical papers (pebble bed modular reactor) submitted by Exelon.**
- **The NRC assigned a staff to U.K.'s Nuclear Installations Inspectorate, Fall, 2002.**
- **The NRC initiated graphite research (ORNL, FY 2002 – 03) to form committees in ASME and ASTM to enable development of design, inspection, and operation codes and standards for graphite core components.**

6

Background – 2

- The NRC staff has been actively participating in ASME and ASTM meetings and has been participating in International Nuclear Graphite Specialist Meetings.
- The staff participated in developing a NRC-DOE document on licensing strategy (August 2008) on proposed NGNP, as per Energy Policy Act, 2005.
  - The applicant or the industry will generate all required experimental data pertaining to graphite to support the NGNP design development effort.
  - The NRC will have full access to the data to support its development effort for confirmatory tools and for an independent assessment of the safety performance of materials.

7

Background - 3

- The NRC conducted a graphite phenomenon identification and ranking table (PIRT) exercise, in cooperation with DOE, in 2007 for guidance on prioritization of graphite research.
- The NRC has kept specific research options open, pending DOE HTGR design selection.
- The NRC has sponsored research (FY 2009) at ORNL to conduct a graphite workshop and identify potential research areas that the NRC may conduct in the future, which augments DOE and NGNP applicant's research.

8

ABBREVIATIONS

ASME	American Society for Mechanical Engineers
ASTM	American Society for Testing Materials
DOE	U.S. Department of Energy
EM	Accident Analysis Evaluation Model
HTGR	High Temperature Gas Cooled Reactor
NGNP	Next Generation Nuclear Plant
ORNL	Oak Ridge National Laboratory
PIRT	Phenomenon Identification and Ranking Table
PRA	Probabilistic Risk Assessment

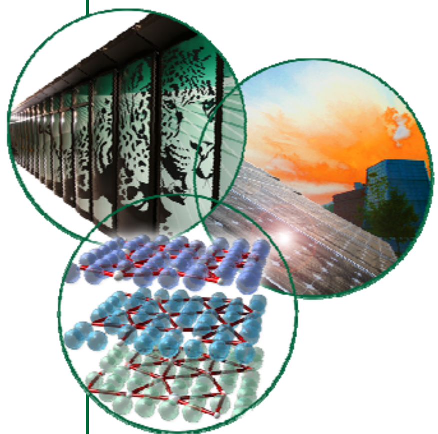
9

Workshop Overview and Goals

Nidia Gallego

Carbon Materials Group

Oak Ridge National Laboratory



Presented at the NRC Workshop on Graphite Research

Rockville, MD

March 16-18, 2009

U.S. DEPARTMENT OF ENERGY

OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

Workshop on Nuclear Graphite Research

Organized by ORNL – Sponsored by NRC

- **Objectives:**
  - To provided NRC with an independent assessment of worldwide progress of graphite research, specifically related to NGNP HTGR, and regulatory challenges and expectations.
  - To identify additional research activities related to graphite, beyond those planned by DOE and other organizations, needed to evaluate safety margins, failure points, and quantify uncertainties.

Workshop on Nuclear Graphite Research

Organized by ORNL – Sponsored by NRC

- Impact of this Workshop:
  - The outcome of this workshop and the assessment are expected to identify those technical areas for future NRC research, which could augment DOE and other international research to enable the acquisition of data and knowledge with high quality and sufficiency.

Documents Provided for Reference

- NUREG/CR-6944, “Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRT), Volume 5: Graphite PIRTs”.
- INL/EXT-07-13165, “Graphite Technology Development Plan”.
- ORNL/TM-2007/153, “NGNP Graphite Selection and Acquisition Strategy”.
- ORNL-GEN4/LTR-06-019, “Experimental Plan and final Design Report for HFIR High Temperature Graphite Irradiation Capsules HTV-1 and -2”.
- ORNL/TM-2008/129, “Generation IV Reactors Integrated Materials Technology Program Plan: Focus on Very High Temperature Reactor Materials”.

Workshop Format

- Background on International HTGR Nuclear Graphite Research and Regulatory Perspective:
  - Presentations from USA, UK, RSA, Japan (*China unable to attend due to visa issue)
- Identification of Technology Gaps and Future Nuclear Graphite Research Activities:
  - Group discussion sessions on proposed themes
- Panel Documentation of Recommendation for NRC's Nuclear Graphite Research

A G E N D A

Day 1 *Topic: Background on International HTGR Nuclear Graphite Research and Regulatory Perspective*

Welcome by Dr. Brian Sheron, Director of Research, Office of Nuclear Regulatory Research, NRC.

A Short History of NRC Nuclear Graphite Research – Dr. Makuteswara Srinivasan, Senior Materials Engineer, Office of Nuclear Regulatory Research, NRC

Workshop Overview and Goals – Dr. Nidia Gallego, Research Scientist, Carbon Materials Technology Group Oak Ridge National Laboratory (ORNL)

Overview of Graphite PIRT Findings – Dr. Timothy Burchell, Group Leader, Carbon Materials Technology Group, ORNL

Overview of US DOE NGNP Research Plan – Dr. William Windes, Leader Graphite Group, Idaho National Laboratory (INL)

Break

UK and European Research Activities – Professor Barry Marsden, School of Mechanical, Aerospace and Civil Engineering, The University of Manchester – UK

PBMR Research Activities – Mr. Mark Mitchell, Leader Materials Group, Pebble Bed Modular Reactor (Pty) Ltd – Republic of South Africa (RSA)

Lunch Break

Chinese Research Activities – Professor Suyan Yu, Institute of Nuclear and New Energy Technology, Tsinghua University – China

Japanese Research Activities – Dr. Motokuni Eto, Technical Consultant, Toyo Tanso Co. Ltd – Japan

U.K. Regulatory Perspective – Mr. Graham Heys, HM Principal Inspector (Nuclear Installations), HM Nuclear Installations Inspectorate, Health and Safety Executive – United Kingdom

Break

RSA Regulatory Perspective – Mr. Schalk Doms, Senior Regulatory Officer, PBMR Programme, National Nuclear Regulator – RSA

Chinese Regulatory Perspective – Professor Suyan Yu, Institute of Nuclear and New Energy Technology, Tsinghua University – China

Japan Regulatory Perspective – Dr. Motokuni Eto, Technical Consultant, Toyo Tanso Co. Ltd – Japan

NRC Regulatory Research Perspectives Related to NGNP V/HTGR Licensing – Dr. Stuart Rubin, Senior Technical Advisor, Dr. Sudhamay Basu, Senior Nuclear Engineer and Dr. Makuteswara Srinivasan, Senior Materials Engineer, U.S. NRC

ADJUORN

Day 2 Topic: Identification of Technology Gaps and Future Nuclear Graphite Research Activities

Some of the Challenges in NGNP HTGR Graphite Component Safety Evaluation – Makuteswara Srinivasan, Senior Materials Engineer, NRC

Comparison of Graphite PIRT Results with DOE Research Plan – Dr. Timothy Burchell, Group Leader, Carbon Materials Technology Group and Dr. Nidia Gallego, Research Scientist, Carbon Materials Technology Group, ORNL

Panel Discussion to Identify Gaps

Purpose: NGNP H(VH)TGR Research Needs Identified For:

- Design Certification Review and Approval (near term)
- Operation Review (near term)
- Inspection (ISI) (mid term)
- Operating License Extension Beyond The existing NGNP Graphite Irradiation Database (long term)
- Decommissioning Graphite Core Components (long term)

Day 2 Topic: Identification of Technology Gaps and Future Nuclear Graphite Research Activities

Suggested Panel Discussion Themes:

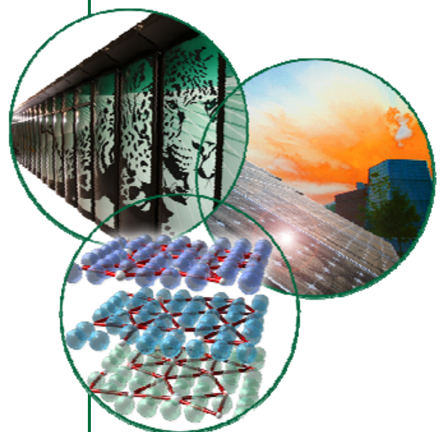
- Graphite qualification
  - INL plans, and vendors plan
  - Comments on plans
    - Adequacy of properties and database
  - Quality assurance requirements
- Requirements for core behavioral models
  - Irradiation properties
  - Models for fundamental understanding for structural integrity analysis
  - Handling of data and model uncertainties
- Oxidation of graphite by coolant impurities
- Status of codes and standard development / future challenges
  - Design and construction code (Section III)
    - Stress analysis
    - Adequacy of margins
  - In-service inspection (Section XI)
- Tribology and oxidation leading to graphite dust
- Air ingress and water ingress (accident)
  - Safe shutdown and safe cool down
- Defining end of core-component life (criteria and safety margins)
- Decommissioning and disposal
- Other themes as suggested by panel members

Day 3 *Topic: Expert Panel Documentation of Recommendation for NRC's Nuclear Graphite Research*

- Continuation of discussion
- Summary and wrap up
- Document any changes necessary to Graphite PIRT
- Review proposed NRC research topics

Overview of NRC Graphite PIRT Findings

Tim Burchell & Nidia Gallego
Carbon Materials Group
Oak Ridge National Laboratory



Presented at the NRC Workshop on Graphite Research
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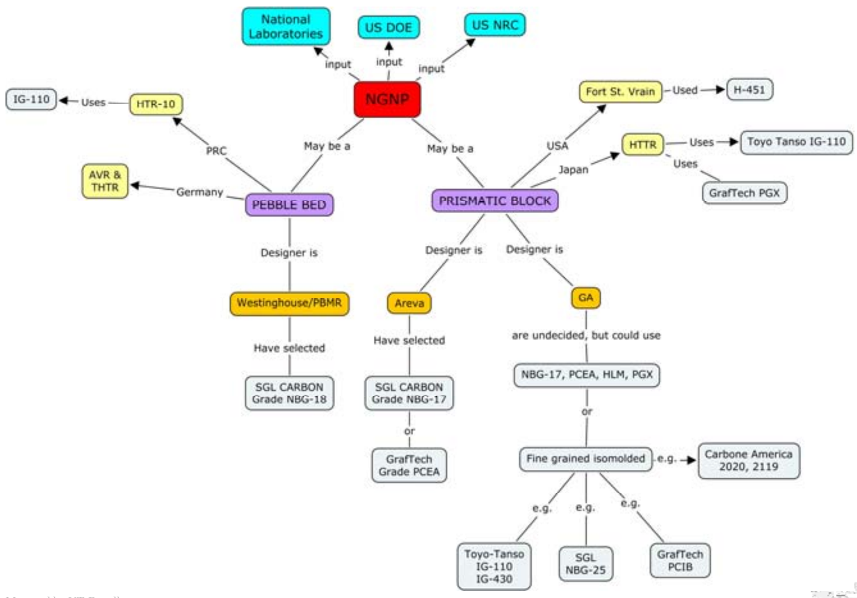
Introduction

- The NRC conducted a Graphite Phenomena Identification & Ranking Table (PIRT) Review in 2007
- The PIRT Panel members were:
  - Dr. Timothy Burchell (Chairman)
  - Dr. Robert Bratton (INL)
  - Professor Barry Marsden (University of Manchester, UK)
    - Mr. Scott Penfield (Technology Insights, Observer)
    - Dr. Will Windes (INL, Observer)
    - Mr. Mark Mitchell (PBMR, Observer)
  - Makuteswara Srinivasan, NRC Facilitator
- A full PIRT Report was subsequently Issued

DOE NGNP: Players & Options

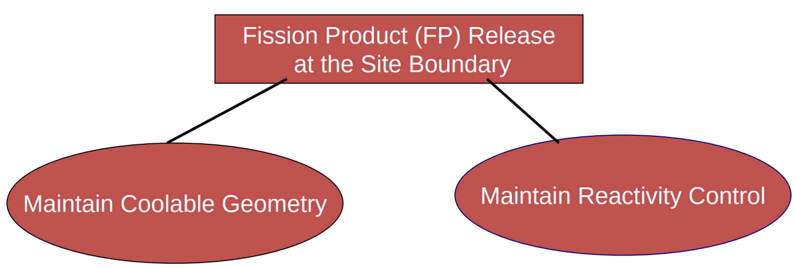


Rationale for Developing Evaluation Criteria (EC) for Graphite

Graphite functions as neutron moderator and as the core structural material.

Maintaining core geometry is essential for:

(a) maintaining core cooling (coolant flow and thermal conduction paths) and
(b) maintaining reactivity control (fuel and control rod channels).



These were determined to be the two EC or Figures of Merit

Note: Fuel elements, including matrix, have been covered by earlier Fuels PIRT (NUREG/CR-6844)

Evaluation Criterion: Reactivity Control

Evaluation Criterion : Maintain Coolable Geometry

Design
- PBR
- PCR

Properties
- He Coolant
- Irradiated Strength
- Critical Defect
- Graphite Specification (impurities & properties)
- Variability in properties (Textural and Statistical)
- Tribology

Phenomena
- Channel Blockage
  - Graphite dust and debris
  - Non-graphite debris
- Channel Distortion
  - Channel Bowing
  - Brick Shear
- Column Collapse (PCR)
- Collapse of upper insulation and deposition onto channel (PCR)
- Collapse of top reflector (PBR)
- Collapse of Bottom Plenum Structure (PBR)
- Chemical attack (Air/Moisture Ingress)

PBR: Pebble Bed Reactor
PCR: Prismatic Core Reactor

Needed Input: Codes & Standards

Spatial and temporal distribution of Fluence and Temperature for graphite core

Contributing Factors To Phenomena Occurrence

(Inadequate) Codes & Methodologies

Insufficient Data & Understanding of Dimensional & Properties Changes Due to Environment

Insufficient data to validate codes – analysis of uncertainty

Lack of reliable irradiated graphite behavioral model based on fundamental microstructural investigations

(Inadequate) Failure Criteria

Component NDE, On-Line Monitoring (OLM), and In-Service Inspection (ISI)

Annealing (Incl. Stored Energy Release)/Thermal Conductivity

Detailed List of Contributing Factors To Phenomena Occurrence

Effect of air oxidation on properties after air ingress

External (applied) loads

Creep strain (Irradiation-induced stress-modified dimensional change)

Internal stress (strain) temp, Fluence, CTE, E, Dim. Change, f(gradient in T, Fluence)

Chemical attack (Impure helium, graphite purity, for example)

Variability in properties (Textural and Statistical)

Consistency in graphite quality over the lifetime of the reactor fleet (for replacement, for example)

Temperature-induced change in Specific Heat

Change in Thermal Properties due to annealing, including stored energy

Graphite dust generation (tribological behavior in Helium, f(T,Pressure,Fluence)

Detailed List of Contributing Factors To Phenomena Occurrence

Graphite Specification

Oxidation of graphite dust

Emissivity, f (surface roughness)

Cyclic Fatigue

Thermal Shock

Subcritical Crack Growth

Component NDE

Online Monitoring

In-service Inspection

Irradiation Induced dimensional change

Irradiation induced strength change

Irradiation-induced thermal conductivity change

Irradiation-induced Young's modulus change

Irradiation-induced change in CTE

Irradiation-induced change in Shear Modulus

Irradiation-induced change in stress-strain curve

Irradiation-induced change in Fracture Behavior

Figures of Merit

FOM ID	Figure of Merit	No. of Phenomena
3-1	Ability to maintain passive heat transfer	22
3-2	Maintain ability to control reactivity	25
3-3	Thermal protection of adjacent components	22
3-4	Shielding of adjacent components	11
3-5	Maintain coolant flow path	23
3-6	Prevent excessive mechanical load on the fuel	14
3-7	Minimize activity in the coolant	19

Summary Of The Phenomena Importance and Knowledge Rankings

PIRT Rank	No. of Phenomena
I-H, K-L	5
I-H, K-M	9
I-M, K-L	2
I-M, K-M	14
I-L, K-H	0
I-L, K-M	2
I-L, K-L	1
I-H, K-H	8
I-M, K-H	1

The Following PIRT Rankings are Discussed

- I-H, K-L (5 Phenomena)
- I-H, K-M (9 Phenomena)
- I-M, K-L (2 Phenomena)

Phenomena ranked I-H, K-L

- *I.D. No. 7: Irradiation-induced creep (irradiation-induced dimensional change under stress)*

- Stress due to differential thermal strain and differential irradiation-induced dimensional changes would very quickly cause fracture in the graphite components if it were not for the relief of stress due to irradiation-induced creep. The phenomena and mechanism of irradiation-induced creep in graphite is therefore of high importance. Currently there are no creep data for the graphite grades being proposed for use in the NGNP. However, creep at low dose follows a linear law that can be explained through a dislocation pinning/unpinning model due to Kelly and Foreman [39]. Marked deviation from this law has been observed at intermediate neutron doses. The applicability of the law has been extended by taking into account changes in the pore structure that manifest themselves as changes in the CTE with creep strain [15]. However, the current creep law breaks down at high-temperature, moderate-dose and moderate-temperature high-dose combinations. A new model for creep is needed that can account for the observed deviations from linearity or the creep strain rate with neutron dose. Existing and new models must be shown to be applicable to the currently proposed graphite grades. Knowledge rank was therefore considered as low.

Phenomena ranked I-H, K-L (continued)

- *I.D. No. 10: Irradiation-induced change in CTE, including the effects of creep strain*

- Differential thermal strains occur in graphite components due to temperature gradients and local variation in the CTE. Variations in the CTE are a function of the irradiation conditions (temperature and dose) and the irradiation induced creep strain [20, 33, 15, 10]. Thus the importance ranking is high for this phenomenon. Irradiation-induced changes in CTE are understood to be related to changes in the oriented porosity in the graphite structure. The changes are observed to be different when graphite is placed under stress during irradiation. The direction and magnitude of the stress (and creep strain) affect the extent of the CTE change. Only limited data are available for the effect of creep strain on CTE in graphite, and none of this data is for the grades proposed for the NGNP. Thus, the knowledge rank is low.

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Phenomena ranked I-H, K-L (continued)

- *I.D. No. 11: Irradiation-induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress)*

- The properties of the graphite are known to change with neutron irradiation, the extent of which is a function of the neutron dose, irradiation temperature, and irradiation-induced creep strain. Differential changes in moduli, strength, and toughness must be accounted for in design. The importance of this phenomenon is thus ranked high. Although data exist for the effect of neutron dose and temperature on the mechanical properties of graphite, there are few data on the effects of creep strain on the mechanical properties. Moreover, none of the available data is for the grades currently being considered for the NGNP. Knowledge ranking is therefore low.

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Phenomena ranked I-H, K-L (continued)

- *I.D. No. 25(b): Blockage of fuel element coolant channel due to graphite failure and/or graphite spalling*
- *.D. No. 27(b): Blockage of coolant channel in reactivity control block due to graphite failure and/or graphite spalling.*

- Significant uncertainty exists as to the stress state of any graphite component in the core. Moreover, the strength of the components changes with dose, temperature, and creep strain. The combination of these factors makes the probability of local failure, graphite spalling, and possible blockage of a coolant channel in a reactivity control block difficult to determine. Consequently the panel rated this phenomenon's importance as high. Although the changes in properties of graphite have been studied for many years there are still data gaps that make whole core modeling very difficult (e.g., effect of creep strain on properties). Moreover, data on the grades selected for NGNP are not available. Therefore, the panel rated the knowledge base for this phenomenon as low.

Phenomena ranked I-H, K-M

- *I.D. No. 1: Statistical variation of non-irradiated properties*

- The graphite single crystal is highly anisotropic due to the nature of its bonding (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. Therefore, it is necessary to develop a statistical data base of the properties for a given graphite grade. Variations in the chemical properties (chemical purity level) will have implications for chemical attack, degradation, decommissioning). Probabilistic design approaches are best suited to capturing the variability of graphite. The panel rated this phenomenon as high importance. Although other nuclear graphites have been characterized and full databases developed, allowing an understanding to de developed of the textural variations, only limited data exist on the graphites proposed for the NGNP. Therefore, the panel rated this phenomenon's knowledge level as medium.

Phenomena ranked I-H, K-M (continued)

- *I.D. No. 2: Consistency in graphite quality over the lifetime of the reactor fleet (for replacement, for example)*

- Graphite is manufactured from cokes and pitches derived from naturally occurring organic sources such as oil and coal (in the form of coal tar pitch). These sources are subject to geological variations and depletion, requiring the substitution of alternate sources. Therefore, consistency of graphite quality and properties over the lifetime of a reactor, or the reactor fleet (for replacement, for example), is of importance. The panel ranked the importance of this phenomenon as high. Our understanding of this phenomenon is sufficient that we are able to develop generic specifications (ASTM DO2.F, D 7219-05) that should assure quality and repeatability. However, this has not been proven. The panel assessed the knowledge base for this phenomenon as medium.

Phenomena ranked I-H, K-M (continued)

- *I.D. No. 6: Irradiation-induced dimensional change*

- Neutron irradiation causes dimensional changes in graphites. Theses changes are the result of anisotropic crystal growth rates (a-axis shrinkage and c-axis growth), the interaction of crystal dimensional change with porosity, and the generation of new porosity. The amount of irradiation-induced dimensional change is a function of the neutron dose and irradiation temperature. Consequently, gradients in temperature or neutron dose will introduce differential dimensional changes (strains). Irradiation induced dimensional changes are the largest source of internal stress. Because of the significance of dimensional changes in generating core stresses, the panel gave this phenomenon as high importance. Irradiation-induced dimensional changes have been researched for many years, and several dimensional change models have been proposed. However, there is a paucity of data for the dimensional changes of the graphites proposed for the NGNP. Therefore, the knowledge rank was considered as medium.

Phenomena ranked I-H, K-M (continued)

- *I.D. No. 8: Irradiation-induced thermal conductivity change*

- Displacement damage caused by neutron irradiation introduces additional phonon scattering sites to the graphite crystal lattice and consequently reduces the thermal conductivity. The nature of the irradiation-induced damage is sensitive to the temperature of irradiation. Consequently, the extent of degradation is temperature dependant. In addition, phonon-phonon (Umklapp) scattering increases as the measurement temperature increases, and thus the thermal conductivity falls as the temperature increases. At very high irradiation dose, thermal conductivity reduces further, at an increased rate, attributed to porosity generation due to large crystal dimensional change. The thermal conductivity is also subject to some recovery (annealing) on heating above the irradiation temperature (such as during an accident thermal transient). The exact thermal conductivity under all core conditions is therefore subject to some uncertainty. A thermal conductivity lower than required by design basis for LBE heat removal due to (a) inadequate database to support design over component lifetime, or (b) statistical and textural variations in characteristics of graphites from lot to lot have the potential to allow fuel design temperatures to be exceeded during LBEs. The importance of this phenomenon was therefore considered high. Irradiation-induced thermal conductivity changes have been researched for many years and several conductivity change models have been proposed. However, there is a paucity of data for the conductivity changes of the graphites proposed for the NGNP. Therefore, the knowledge rank was considered as medium.

Phenomena ranked I-H, K-M (continued)

- *I.D. No. 9: Irradiation-induced changes in elastic constants, including the effects of creep strain*

- Neutron irradiation induces changes in the elastic constants of graphite. Initial increases in the moduli are attributed to an increase in dislocation pinning points in the basal plane, which reduce the crystal shear compliance, C44. Subsequent changes in the elastic modulus are attributed to pore-structure changes (initial pore closures followed by pore generation). Although the understanding of irradiation modulus changes is plausible behavior, there are no direct microstructural observations or sufficiently well developed models of these mechanisms. Therefore, the knowledge rank was considered as medium.

Phenomena ranked I-H, K-M (continued)

- *I.D. No. 17: Tribology of graphite in (impure) helium environment*

- Graphite is a naturally lubricious material. However, its behavior is modified by the helium environment of the NGNP. The abrasion of graphite blocks on one another or of the fuel pebbles on the graphite moderator blocks may produce graphite dust. Studies are needed to assess the effect of the helium environment on the friction and wear behavior of graphite. The possibility that fuel balls can "stick" together and cause a fuel flow blockage must be explored, although German pebble bed experience was positive in this regard (i.e., no blockages). The consequences of dust generation (possible fission product transport mechanism) and possible fuel ball interactions resulted in the panel ranking the importance of this phenomenon as high. Some literature exists on this subject mostly from the past German program. Consequently, the panel ranked the knowledge level as medium.

Phenomena ranked I-H, K-M (continued)

- *I.D. No. 21: Degradation of thermal conductivity (see also No. 8)*

- The degradation of thermal conductivity in graphite components has implications for fuel temperature limits during loss-of-forced cooling accidents.

Phenomena ranked I-H, K-M (continued)

- *I.D. No. 28(b): Blockage of Reactivity Control Channel due to graphite failure, spalling*

- Significant uncertainty exists as to the stress state of any graphite component in the core. Moreover, the strength of the components changes with dose, temperature, and creep strain. The combination of these factors makes the probability of local failure, graphite spalling, and possible blockage of a coolant channel in a reactivity control block difficult to determine. Consequently the panel rated this phenomenon's importance as high. Although the changes in properties of graphite have been studied for many years there are still data gaps that make whole core modeling very difficult (e.g., effect of creep strain on properties). Moreover, data on the grades selected for NGNP are not available. Therefore, the panel rated the knowledge base for this phenomenon as low.

Phenomena ranked I-H, K-M (continued)

- *I.D. No. 36: Graphite temperatures*

- All graphite component life and transient calculations (structural integrity) require time-dependent and spatial predictions of graphite temperatures. Graphite temperatures for normal operation and transients are usually supplied to graphite specialists by thermal-hydraulics specialists. Although, in some cases, gas temperatures and heat transfer coefficients are supplied, and the graphite specialists calculate the graphite component temperatures from these.

Phenomena ranked I-M, K-L

- *I.D. No. 15: Graphite dust generation*
- Abrasion between adjacent block, or fuel pebbles and reflector blocks, will cause the formation of dust. This may become a vector for fission products or could possibly impede coolant flow (see below).

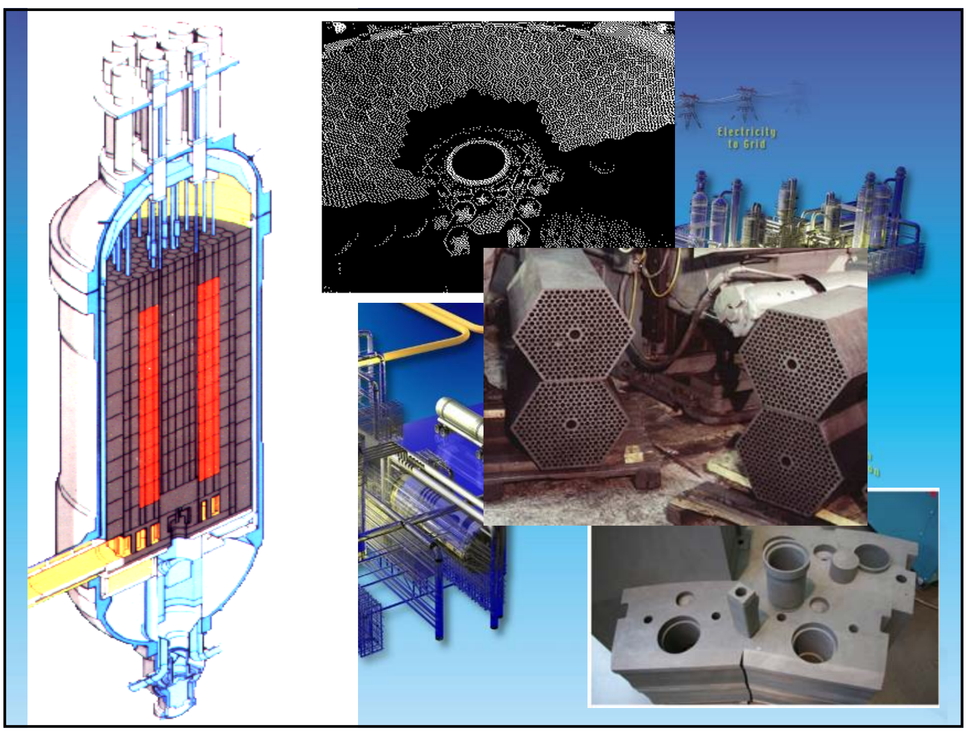
- *I.D. No. 26(b): Blockage of reflector block coolant channel—due to graphite failure, spalling*
- Blockage of coolant channels by graphite debris could cause local hot spots in the core.

NGNP Graphite Program

NRC Graphite Workshop

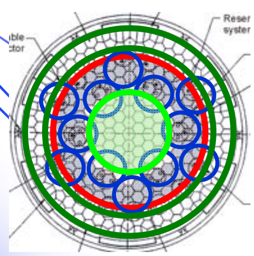
Will Windes, NGNP graphite technical lead

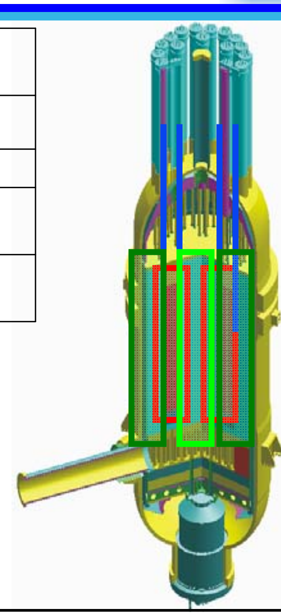
Rockville, MD, 16-18 March, 2009



NGNP core components

Component	Normal Operation	Off-Normal	dpa
Graphite fuel block	~1200 ºC	~1400 ºC	~ 0.8/yr
Control rods	~1000 ºC	~1200 ºC	~ 0.5/yr
Reflector blocks – inner	~900 ºC	~1200 ºC	~ 0.5/yr
Reflector blocks - outer	~800 ºC	~1100 ºC	~ 0.2/yr

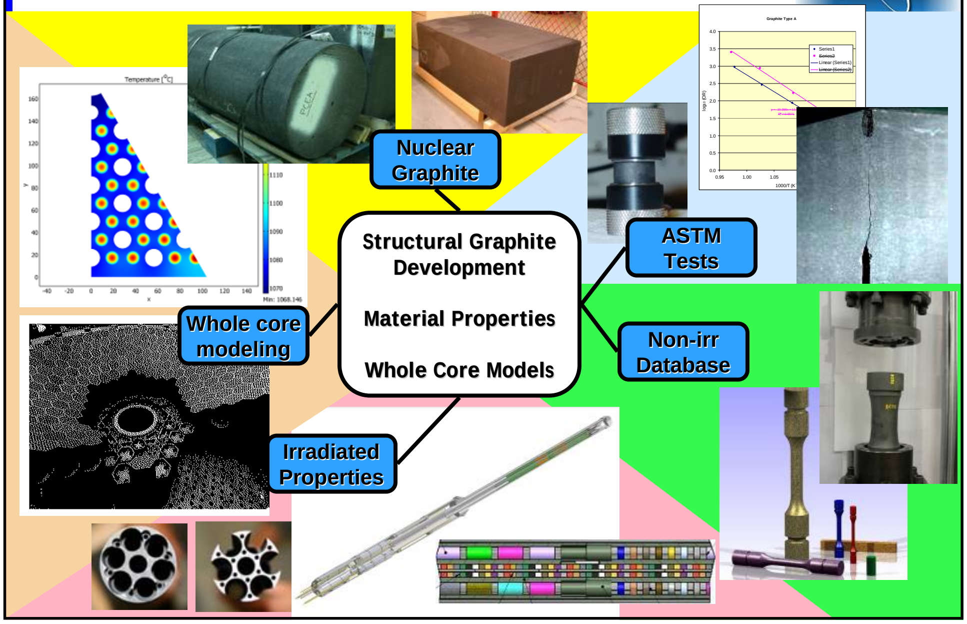




NGNP assumptions

- Either a prismatic core or pebble bed core
  - Graphite – (compressive applications)
  - Ceramic composites – (tensile applications)
- No decision of specific graphite type
- 750 C (to start with) - 900-950 C (eventually)
- 400-600 MW_t
- He inert gas coolant (limited impurities)
- Funding Partnership with private sector
- Vendors = Westinghouse (PB), GA, & AREVA

Research Map

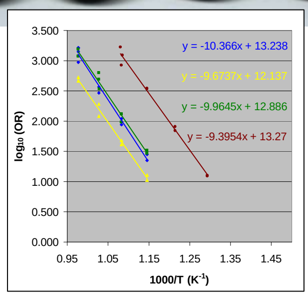


ASTM Standards

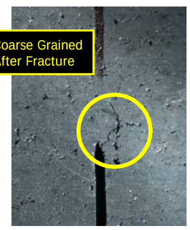
- All licensing data generated using ASTM standards
  - Except when none currently exist
- New standards being developed
  - **Air oxidation** = Nearly complete
  - **Fracture toughness** = Working data from RR
  - **XRD techniques** = Evaluating
  - **NDE techniques** = Evaluating/screening studies
  - **Shear tests** = Evaluating
- Evaluation of existing standards
  - DYM, small sample, sonic measurements, thermal

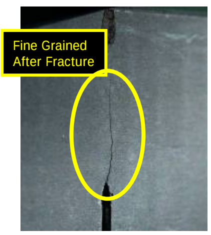
ASTM: Development

Graphite Oxidation

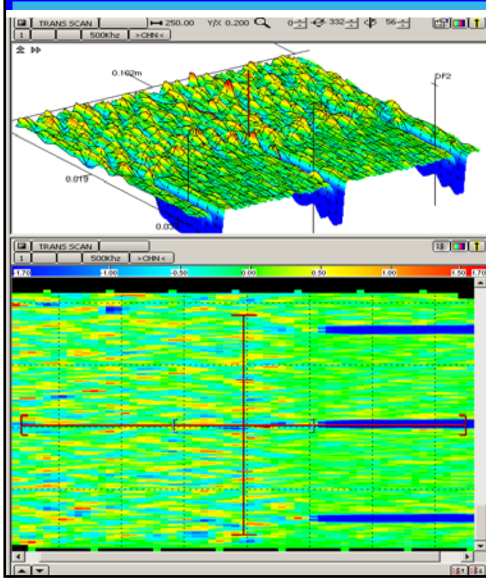


Fracture toughness





ASTM: Development



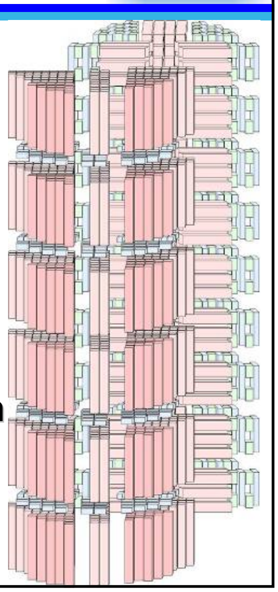
- NDE Techniques for large graphite components
  - Flaw type determination
  - Penetration depth
  - Spacial resolution
- Needed to identify disparate flaws
  - Different than “normal” distribution flaws
  - Similar to what metals have for validation

ASTM: Evaluations

- Current standards may not be appropriate
  - Thermal diffusivity – “short” samples needed (4-6 mm)
    - But grain size = 2-3mm reducing averaging effect
    - Sample size : Grain size ??
  - 3-point bend – “long” and “wide” samples needed
    - Irradiation samples are difficult
    - Where does the dimensional ratio fall apart ??
  - Sonic measurements – “large” samples needed for accurate time of flight
    - Again, small samples are difficult
    - Signal data interpretation (i.e. start & stop)
    - Coupling issues, specifically for irradiation samples

Baseline characterization

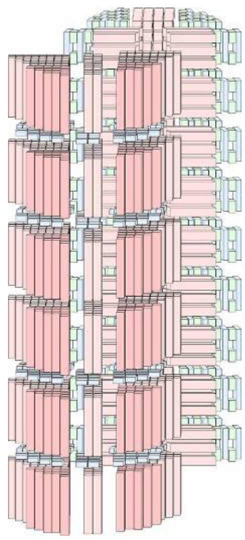
- **Baseline material properties**
- **Variations in:**
  - **Intra-billet**
  - **Inter-billet**
  - **Lot-to-lot**

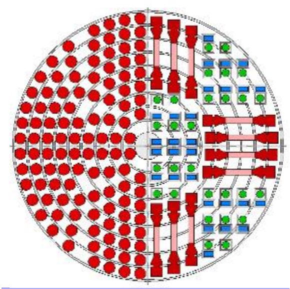


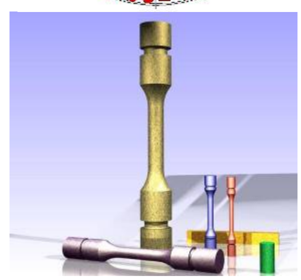
- **Irradiation results superimposed on top of baseline values**
  - Irr. results are **not** true property values

Baseline characterization

- **Statistically representative sampling matrix**
- **Finest spacial resolution – min. sample size**
- **Physical, thermal, and mechanical property measurements**







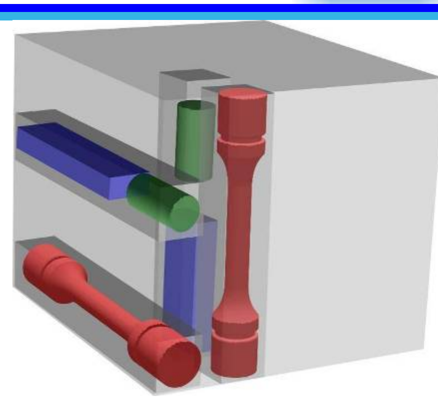
Baseline characterization

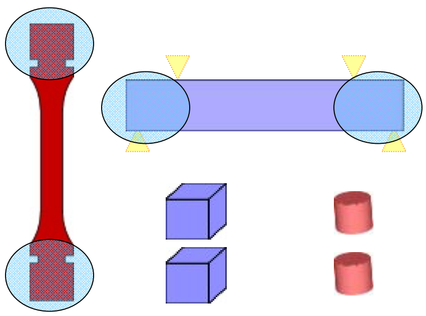
- Baseline material properties to be measured
  - Detailed pre-irradiation dimensional characterization (for creep experiments)
  - Mass, (bulk density)
  - Chemical analysis
  - Fundamental frequency (Young's modulus)
  - Sonic velocities (Shear and Young's modulus,
  - Poisson's ratio
  - Electrical Resistivity
  - Coefficient of thermal expansion (25-800 ^{o}C)
  - Thermal conductivity (25-1000 ^{o}C)
  - Mechanical Strength
    - Compressive, tensile, flextural (bending)
  - *Fracture toughness, shear strength, oxidation ?*

- **PCEA**
- **NBG-18**
- NBG-17
- IG-110
- IG-430
- Others

Baseline characterization

- Full-size mechanical samples
  - No grain size verses sample size issues
- NDE before destructive

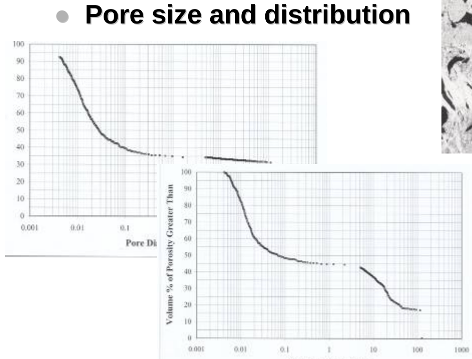


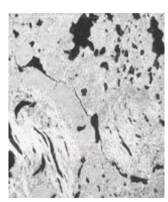


- Thermal analysis after mechanical
  - Small samples OK
  - “Same” location in billet

Other “baseline” properties

- **Multi-axial mechanical testing**
  - **Stresses are complex – need complex tests**
  - **Data used to model whole core behavior**
- **Microstructure analysis**
  - **Pore size and distribution**

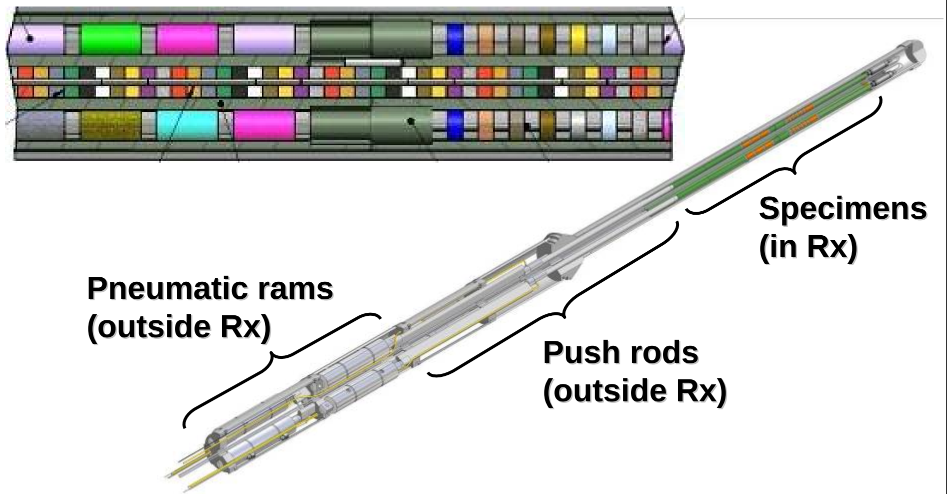




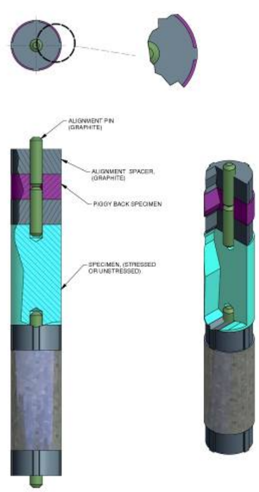
- **Microstructural modeling**
  - **Needed to figure out what’s really going on in microstructure during irradiation? (basal plane pinning. Or is it?)**

Irradiation activities

ATR Graphite Creep (AGC) experiments

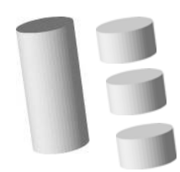


AGC Samples



AGC sample loading scheme

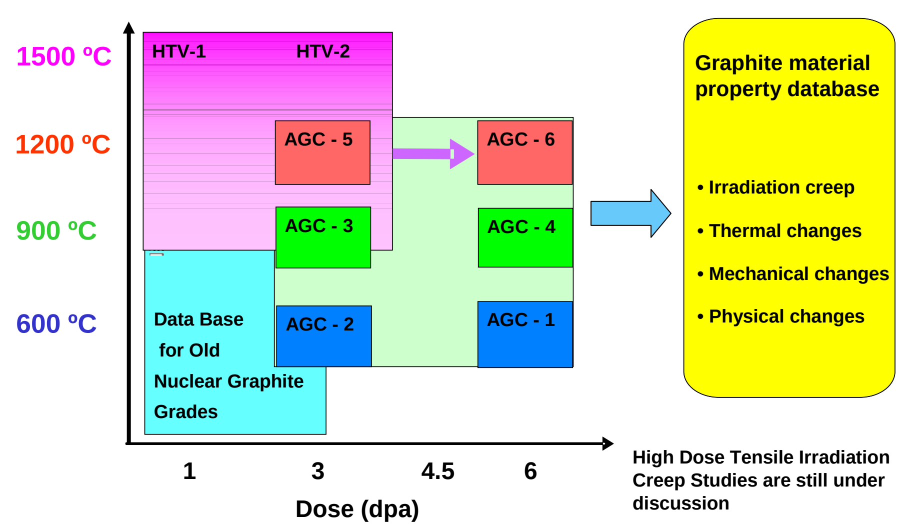
- Creep samples
  - Ø12 mm x 25 mm (1/2”x1”)
- Piggyback samples
  - Ø12 mm x 6 mm (1/2”x1/4”)
- Six major (creep) grades
  - H-451, IG-110, PCEA, NBG-18, NBG-17, and IG-430
- Ten piggyback grades
  - NBG-25, PCIB, PPEA, NBG-10, BAN, HLM, PGX, S2020, HOPG, and A3 matrix



AGC Experiments

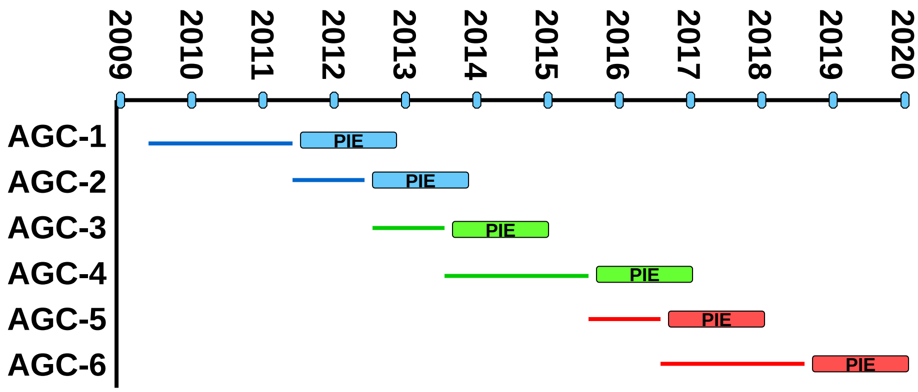
- Total of six irradiation capsules operating at 600, 900, and 1200 C.
- Fluence range 0.5-7 dpa (covers both PB and prismatic)
- 90 creep specimen pairs (stressed and unstressed) and over 300 piggyback specimens
- Compressive loads applied to creep grades
  - 2, 2.5 and 3 Ksi compressive
  - No applied load to piggyback specimens
- Material properties ***after*** irradiation creep
- Provides change to baseline graphite properties

AGC Experiments

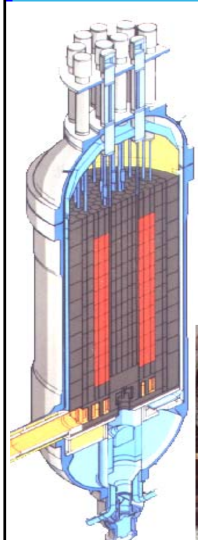


High Dose Tensile Irradiation Creep Studies are still under discussion

AGC Schedule

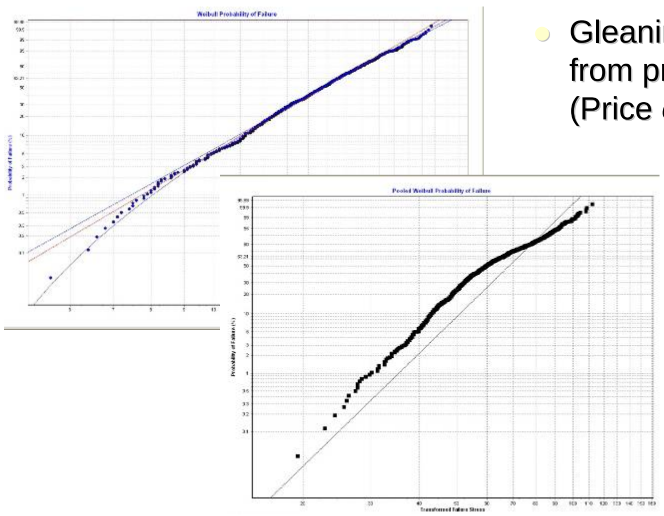


Whole core modeling



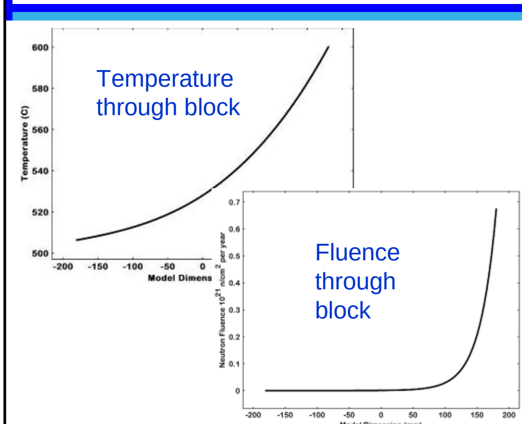
- Ground work needed before modeling begins
  - Investigate previous work and relationships
  - Gather past data and results
- Start to build relationship between reactor conditions and graphite behavior
- Employ probablistic rather than deterministic approaches

Modeling: Ground work

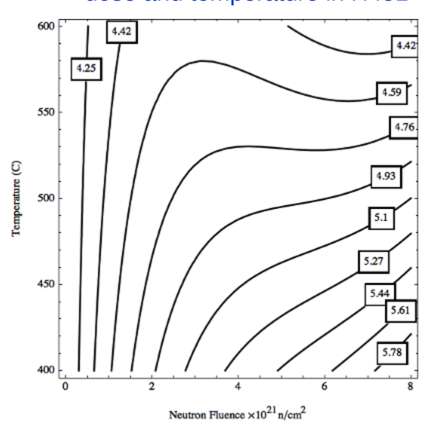


- Gleaning useful data from previous work (Price *et al.*)
- Now need to start to build relationship between current Rx conditions and new graphite types

Modeling: Assumptions and initial data

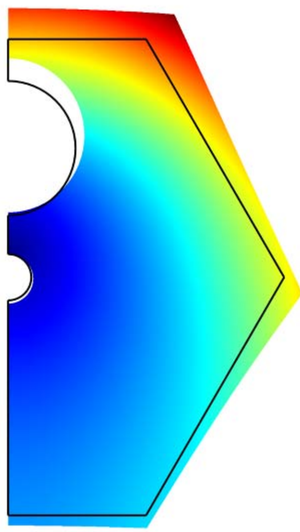


Thermal expansion due to dose and temperature in H451



- Current and past information combined to determine graphite behavior

Modeling: Initial models



Initial models based upon NGNP conditions

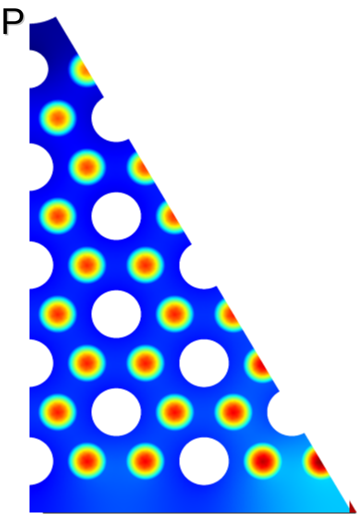
- Temperature, flux, load, dose, dimensions, etc.

- Initial behavior modeled
  - Thermal displacements

Modeling: Initial models

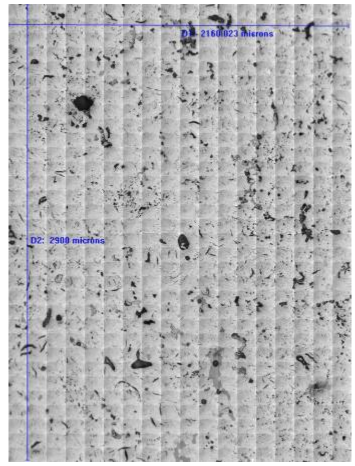
- Initial models based upon NGNP conditions
  - Temperature, flux, load, dose, dimensions, etc.

- Initial behavior modeled
  - Temperature gradients
  - Fuel & graphite temperature



Basic Research

- Minimal direct role from NGNP
- Long range research objectives
  - NEUP and university projects
  - IAEA
- Areas of interest
  - Irradiation creep
  - Strength/fracture
  - Microstructure evolution
  - Ceramic composites



- Considered critical to understanding the data generated from the rest of the program

ASME support

- ASME simply provides an accepted process for approving any data generated for the materials
- Code case = How to use a material in a specific application
- Looks at the whole picture
  - Baseline data, irradiation data, component modeling results, whole core modeling, and compares to the expected operating conditions

- **NRC is interested in having a code case for graphite**
  - Makes it easier to approve of graphite types
  - Makes it easier to use graphite over time with common understanding of behavior
  - But a code case is not mandatory, just harder

In summary

- All licensing data is taken by ASTM Standards
- Baseline material properties are the **real** values for graphite
  - Statistically valid, no weak pockets/areas
- AGC (irradiation) results are the changes to the material properties
  - Measured values are not the true material properties
- Whole core model is the final repository for the data
  - Data results and graphite behavior are interpreted using the model
- Research activities (NEUP/University) needed to understand results
- ASME simply provides an accepted process for approving any data generated for the materials

Thank You

- Questions ?

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WORKSHOP ON GRAPHITE RESEARCH

UK and European Research Activities

Professor Barry J Marsden

The University of Manchester

School of Mechanical, Aerospace and Civil Engineering

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Overview

- European Nuclear Graphite Research Initiatives
- UK Specific Nuclear Graphite Research
- Nuclear Graphite Research at Manchester
- Conclusions

Nuclear Graphite Research in Europe

- The main drivers for nuclear graphite research in Europe are:
  - The continued safe operation of the UK AGRs and remaining Magnox reactors, including the possibility of life extension
  - The European Framework Programmes FP5, FP6, FP7
    - RAPHAEL
    - CARBOWASTE
  - The decommissioning and waste disposal of irradiated graphite waste from graphite moderated reactors (including operational waste) in the UK, France, Lithuania, Italy, Spain and Germany
- Depending on the definition of Europe there are operating RBMK reactors in Western Russia and Lithuania as well as shutdown RBMK reactors in the Ukraine
  - These activities also have issues of continued safe operation and final decommissioning and waste disposal

RAPHAEL-IP

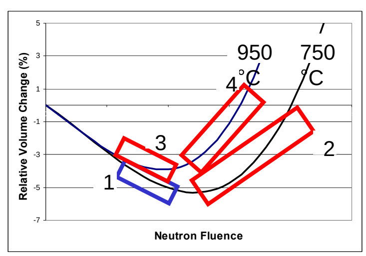
- RAPHAEL-IP
  - ReActor for Process heat, Hydrogen And ELectricity generation
- European Commission's 6^{th} Euratom Framework Programme
  - support research co-operation and integration of research efforts in the nuclear fission and fusion area
  - April 2005 – April 2009
- 33 Partners or organisations
  - 10 countries
    - Belgium, Czech Republic, France, Germany, Italy, The Netherlands, Slovak Republic, Spain, Switzerland, UK

RAPHAEL-IP

- 8 Sub-projects
  - coupled reactor physics and core thermo-fluid dynamics (SP CP)
  - fuel technology (SP FT)
  - back-end fuel technology (SP BF)
  - materials development (SP ML)
  - component development (SP CT)
  - safety (SP ST)
  - system integration (SP SI)
  - project management (SP PM)
- Materials development (SP ML)
  - 4 work packages
    - vessel materials (WP-ML1)
    - high temperature materials (WP-ML2)
    - graphite materials (WP-ML3)
    - codes and standards for graphites and composites (WP-ML4)

RAPHAEL-IP

- Graphite materials (WP-ML3)
  - 4 tasks
    - continuation of 750ºC irradiation tests (Task 1)
    - graphite irradiation test at higher temperatures (Task 2)
    - corrosion and heat-up tests (Task 3)
    - graphite modelling (Task 4)
  - 8 partners
    - University of Manchester (UK), NRG (The Netherlands), AMEC (UK), FZJ (Germany), Framatome ANP (France), CEA (France), SGL Carbon (Germany), UCAR (France)
  - 2 observers
    - NRI Rez (Czech Republic), Paul Scherer Institut (Switzerland)



INNOGRAPH-1A: (FP5)	750°C, low/medium dose
INNOGRAPH-1B:	750°C, high dose
INNOGRAPH-2A:	950°C, low/medium dose
INNOGRAPH-2B:	950°C, high dose

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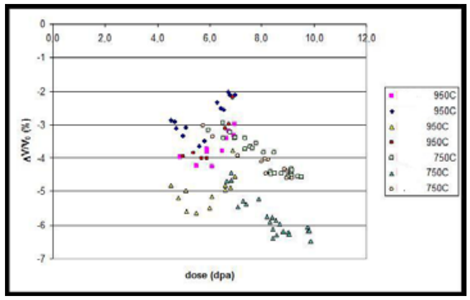
RAPHAEL-IP

- Codes and standards (WP-ML4)
  - 7 partners
    - NNC (UK), JRC (The Netherlands), Framatome ANP (France), Framatome ANP (Germany), EA (Spain), SGL Carbon (Germany), UCAR (France)



750°C and 950°C Turnaround

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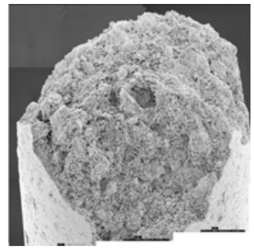
Carbowaste EU FP7

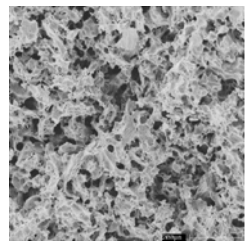
- Aims to develop a solution for integrated irradiated graphite waste management through investigation of:
  - The nature of the contamination (how it is contained within the graphite? What are its release mechanisms?)
  - The mobility and immobility of the various species within the graphite
  - Seek opportunities for safe and fit for purpose disposal solutions
  - Assist predictability of long-term disposal performance
  - Develop opportunities for more economic solutions for legacy waste and future use of the material.
  - Investigate recycling possibilities
- Membership: 28 partners, from the UK, France, Germany, Italy, Spain, Lithuania, Belgium Romania, the Netherlands, Sweden and South Africa
- Funding Total Funding 12,081,363€ (6,000,000€ EU)
  - Manchester **Funding 623,062 € (470,833 € EU)**

Some Examples of Nuclear Graphite Research in the UK

Licensees AGR and Magnox reactors

- Statistical analysis – prediction of brick cracking rates
- Graphite irradiation Data
  - Development of semi-empirical models for use in graphite component structural integrity assessments including irradiation creep
- Statistical analysis of reactor installed and trepanned data of component dimensional change, weight loss and property changes for use in structural integrity assessments
- Whole core modelling
- Refuelling trace monitoring
- Seismic modelling (numerical analysis and proposed rigs)
- Component life prediction (numerical finite element modelling)



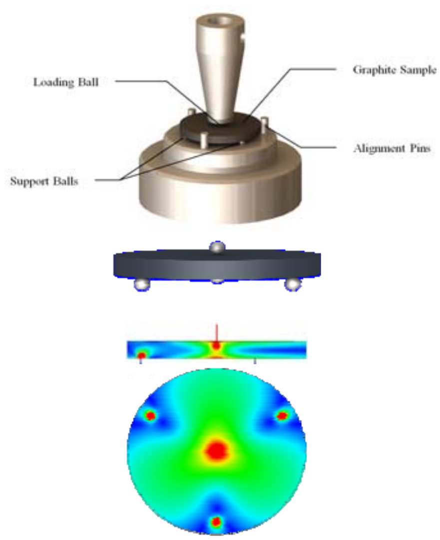


PGA Graphite ~40% weight loss Magnox North/SERCO

Some examples of Nuclear Graphite Research in the UK

Licensees AGR and Magnox reactors

- Inspection methods (crack detection using eddy currents)
- AGR Gilsocarbon graphite property MTR programme
- Possible AGR Gilsocarbon graphite irradiation creep MTR programme
- Microstructural – property investigation in highly oxidised (~40% weight loss) Magnox PGA graphite
- Improved methods of property measurement, particular Young's modulus and Poisson's ratio.
- Biaxial testing of graphite strength
- Atomistic modelling of defects in irradiated graphite

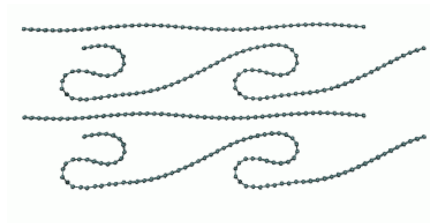


Biaxial testing due to Gareth Neighbour University of Hull

Research Contractors to the UK Licensees

- AMEC Nuclear
- Atkins
- British Energy Generation Ltd (EDF)
- Doosan Babcock
- Frazer-Nash Consultancy Ltd
- Jacobs Engineering UK Ltd
- Magnox North
- Magnox South
- National Nuclear Laboratory (NNL)
- NRG Petten
- Nuclear Decommissioning Authority (NDA)
- Nuclear Technology Consultancy
- Quintessa Limited
- Serco
- University of Birmingham
- University of Bristol
- University of Glasgow
- University of Hull
- University of Manchester
- University of Strathclyde
- University of Sussex

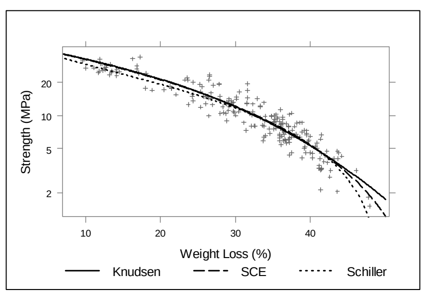
One quarter size rig of AGR core AMEC



Malcolm Heggie (University of Sussex) Atomistic Modelling

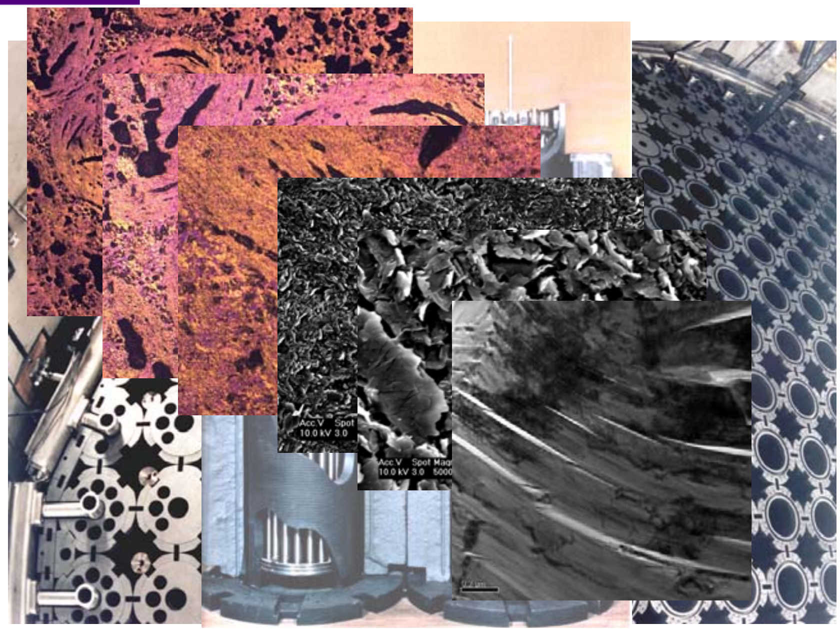
HSE(ND) Independent Nuclear Research

- The Graphite Technical Assessment Group (GTA)
  - Hosted by the University of Manchester
- Statistical assessment of irradiated graphite properties
  - The University of Manchester and Models and Computing Services (Ernie Eason)
- Fracture in graphite
  - The University of Birmingham
- AGR Brick cracking network
  - The University of Manchester
  - The University of Birmingham
  - The Health and Safety Laboratory (HSL)
- Development of Whole Core Models
  - The University of Manchester

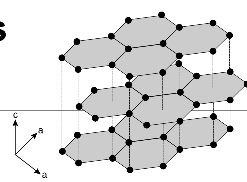


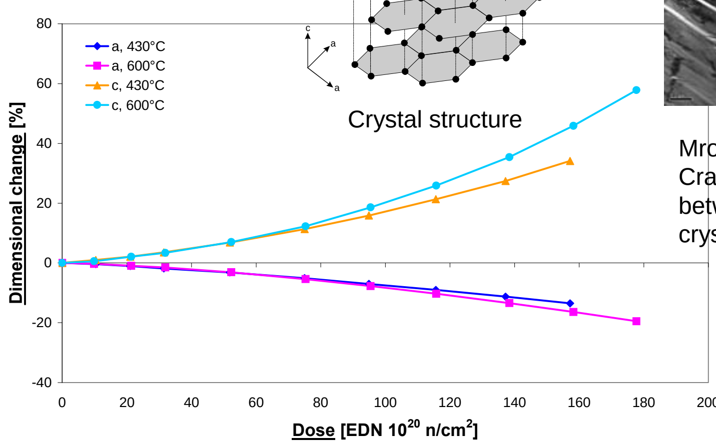
High weight loss strength data showing evidence "percolation limit"

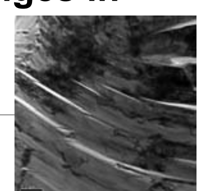
Due to HSL/HSE



Irradiation Induced Dimensional Changes in Graphite Crystals

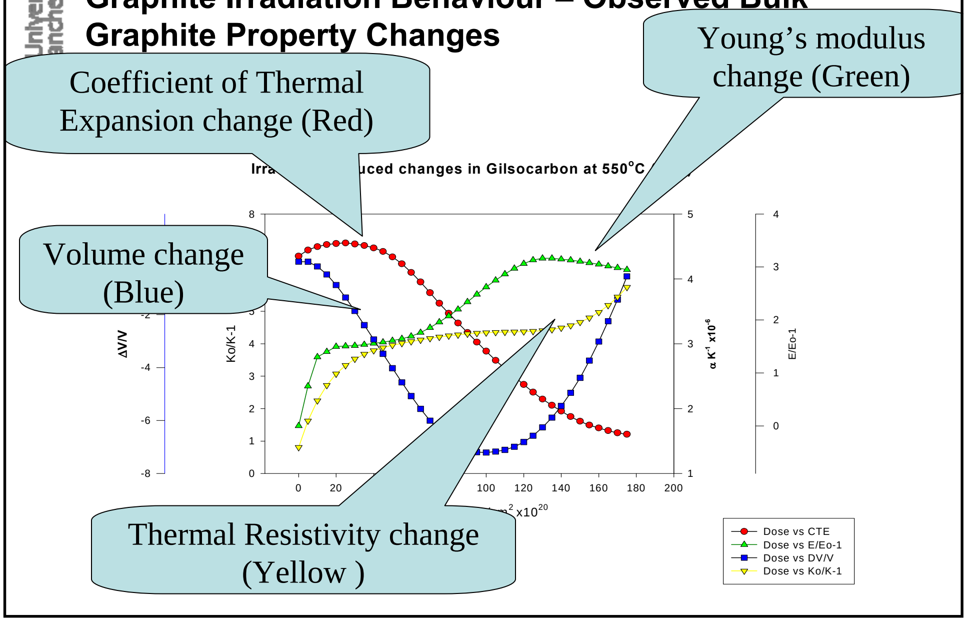




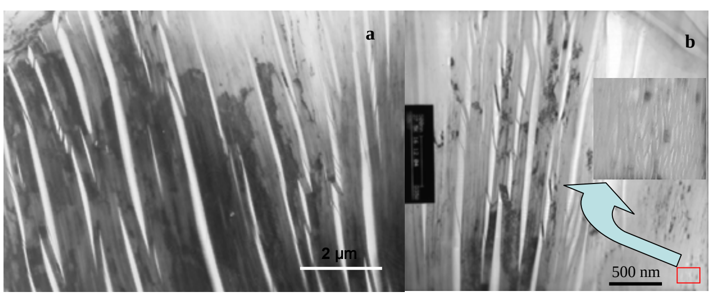


Mrowozski Cracks between crystallites

Graphite Irradiation Behaviour – Observed Bulk Graphite Property Changes

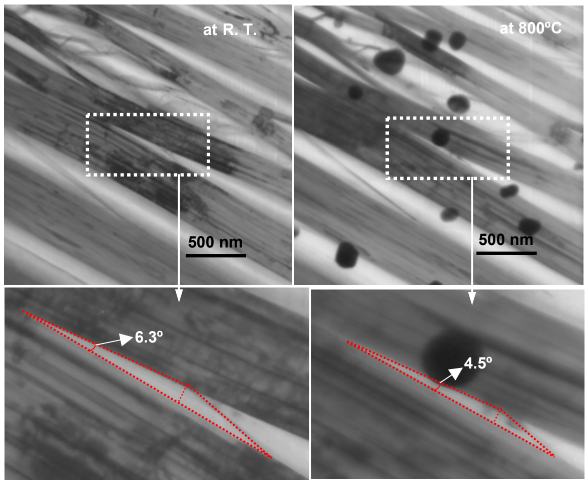


Study of the influence of micro-cracks on the behaviour of nuclear graphite



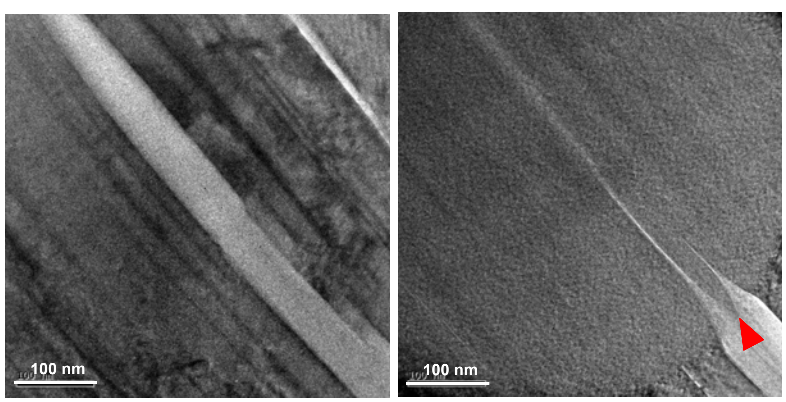
- TEM images of Mrozowski cracks in Pile Grade A Graphite
- Mrozowski cracks have an important role in determining both the unirradiated and irradiated the properties nuclear graphite
- Microcracks were formed by cleavage along basal plane during graphitisation
- The length of the cracks varies from tens of nanometres to more than 10 mm
- The width of the cracks varies from less than 50 nm to ~200 nm.
- The very fine crystallites and cracks in the inset of the figure on the left may be in the binder.

Results: in situ heating



Upon heating, a gradual closure of cracks is observed because of the thermal expansion of the graphite crystallites surrounding the cracks.

Results: in situ electron irradiation



Closure of a crack in Gilsocarbon after In-situ electron irradiation.

The feature with bright contrast does not disappear completely.

Note a small part of crack (indicated by arrow), which was covered by the electron beam has not closed completely..

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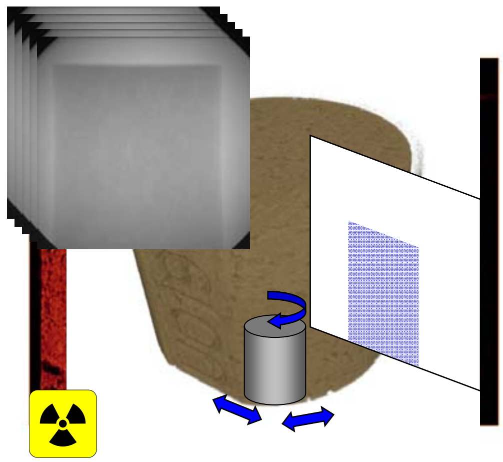
Characterisation and modelling of microstructure heterogeneity in nuclear graphite

- Synchrotron tomography
  - Non-destructive
  - 3D microstructure
  - Technique
    - sample
    - X-ray source
    - detector
    - reconstruction



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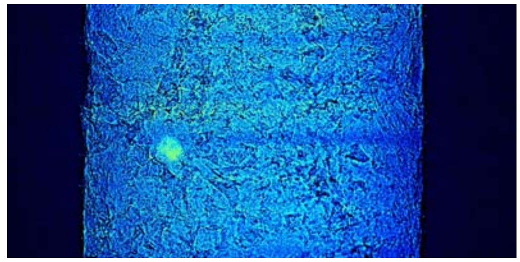
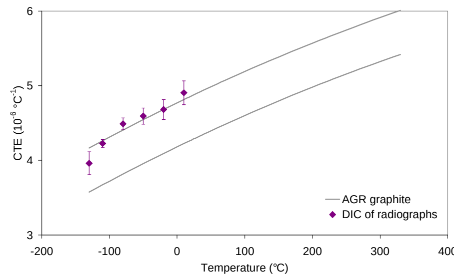
Synchrotron tomography

- Swiss Light Source (SLS)
  - Paul Scherrer Institut, Villigen
- TOMCAT beamline
  - TOmographic Microscopy and Coherent rAdiology experimenTs
- Thermal experiments
  - Gilsocarbon samples
    - ø1mm × 2mm region of interest
  - CT scanned at different temperatures
    - 0°C, -30°C, -60°C, -90°C, -120°C, -140°C

Synchrotron tomography
Bulk CTE

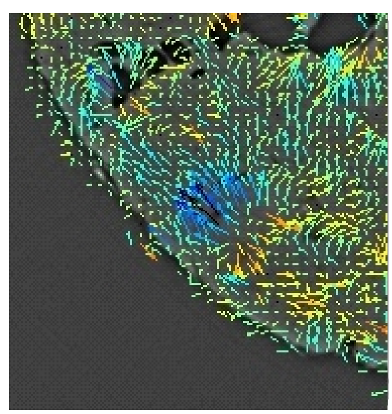
- Bulk CTE changes
  - radiographs
  - digital image correlation (DIC)
    - DaVis
  - compared well with literature data





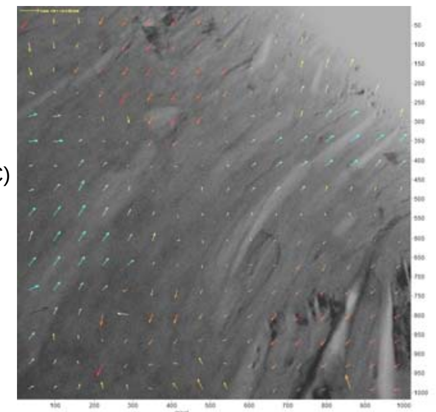
Synchrotron Tomography Localised CTE

- Local CTE
  - reconstructions
  - regions of well-defined microstructure
  - digital image correlation (DIC)
  - heterogeneity in thermal strains



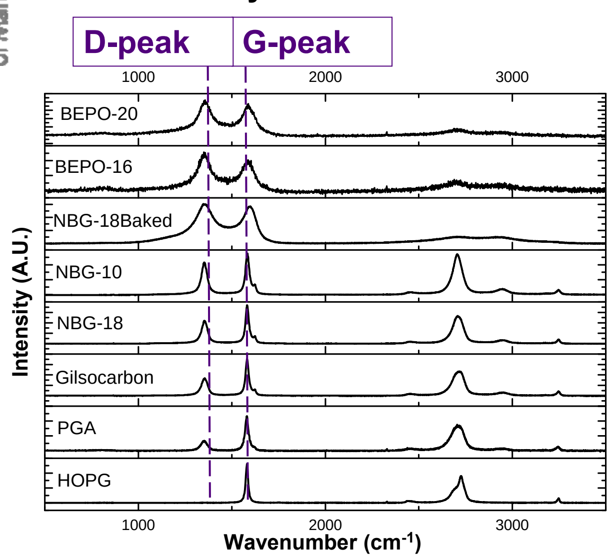
Transmission electron microscopy - Localised CTE

- Microstructural thermal strain
  - sample heated to 600°C
  - closure of Mrozowski cracks
  - directionality
    - Digital Image Correlation (DIC)
  - local CTE (non-cracked)
    - 11.7×10^{-6} K^{-1}
  - local CTE (cracked)
    - *c*-axis 34×10^{-6} K^{-1}
    - *a*-axis -0.5×10^{-6} K^{-1}



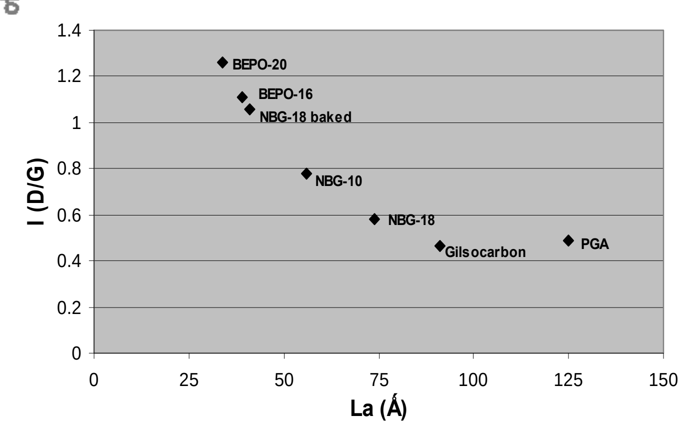
400-500°C

Raman Spectroscopy – structural characterisation, phase purity and crystallite size.



- Raman scattering measurements were performed using a Renishaw inVia Raman microscope using an Argon 514.5nm laser.
- G-peak noted at 1580 cm^{-1} indicative of in plane bond stretching of C sp^2 atoms.
- D peak noted at 1350 cm^{-1} is the breathing node of sp^2 and disordered carbon.

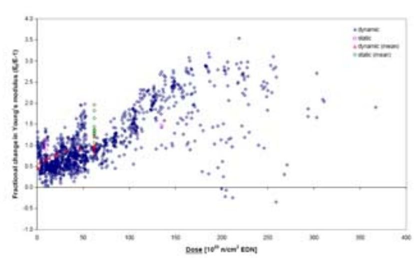
Raman crystallite size



- I(D)/(G) is an indication of the structural disorder.
- Structural disorder increases a crystallite size decreases.

Models of Young's Modulus for Gilsocarbon Graphite grades Irradiated in an Inert Environment

- Ernest D. Eason, Modeling & Computing Services LLC Boulder, Colorado USA
- Graham Hall, Barry Marsden, Nuclear Graphite Research Group The University of Manchester
- Funded by HSE(ND)
- Properties being investigated are: dimensional change, CTE, Young's modulus, strength and thermal conductivity
- Later analysis will also include irradiation in an oxidising atmosphere
- Only Young's modulus discuss here
- Database
  - UKAEA MTR data

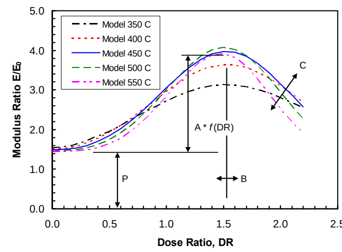


Young's modulus inert MTR data

Inert Model for Young's Modulus

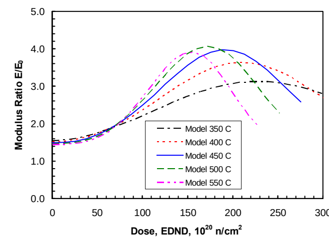
The model below was fitted to the inert MTR data

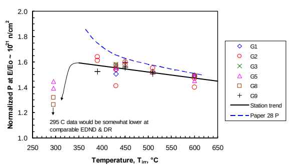
$$\frac{E}{E_0} = P(T_{irr}) + A(T_{irr}) \cdot \left(\frac{DR}{B}\right)^{C-1} \exp\left[-\left(\frac{DR}{B}\right)^{C}\right]$$

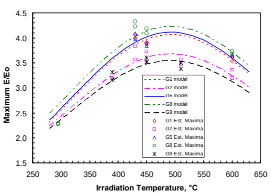


Dose ratio DR, is dose normalised to dimensional change "turnaround"

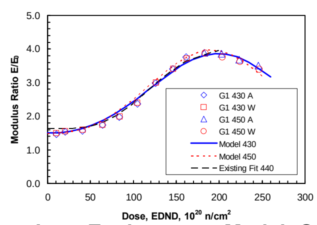
Functions *P* and *T* were found to be function of graphite grade as well as temperature, see below



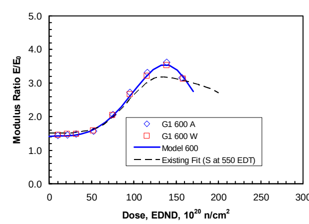




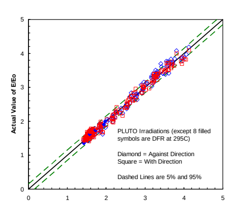
Model fits to various graphite grades



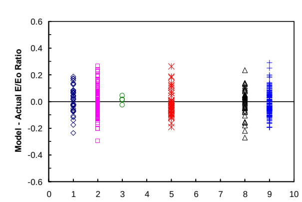
Inert Environment Model, Graphite Group 5 (HYA), 430 - 450°C Data



Inert Environment Model, Graphite Group 5 (HYA), 600°C Data



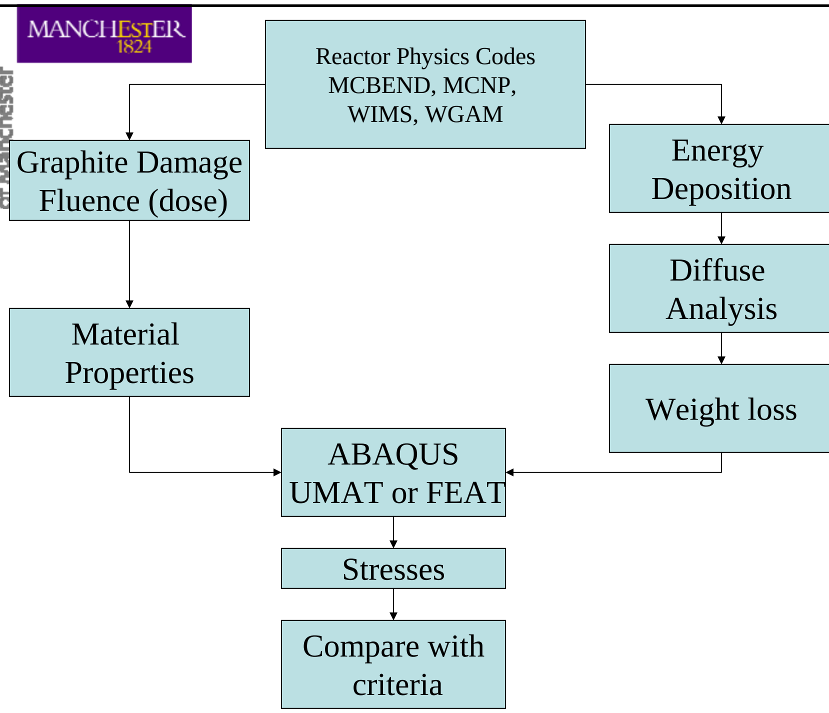
Actual *vs.* Model Values of E/E0 for Station Moderator Data Used for Calibration



Residual Plot Showing no Residual Trend with Graphite Group

Graphite Component Stress Analysis

- Nuclear Graphite – is an artificial, porous polycrystalline material
- In reactor graphite properties and dimensions change due to fast neutron irradiation and radiolytic oxidation
- Also in reactor graphite exhibits irradiation creep
- Property changes are a function of:
  - Fast neutron fluence
  - Temperature
  - Radiolytic weight loss
- Irradiation creep is a function of
  - Fast neutron fluence
  - Radiolytic weight loss
  - Stress
- There are other complications related to creep, CTE which are ignored here



Total Strain Increment

Elastic Strain

Total Strain

Primary creep strain

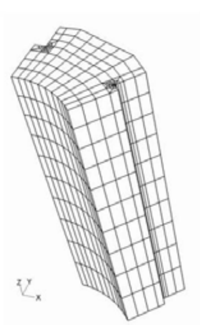
$$\{\varepsilon_E\} = \{\varepsilon_{TOT}\} - \{\varepsilon_{pc}\} - \{\varepsilon_{sc}\} - \{\varepsilon_{\gamma}\} - \{\varepsilon_{th}\}$$

Secondary creep Strain

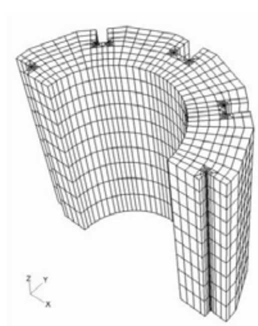
Thermal Strain

Dimensional Change

Example HPB/HNB 3-D Finite Element Meshes

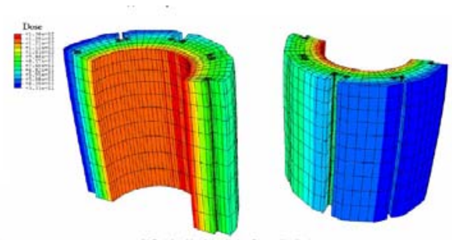


a) Central



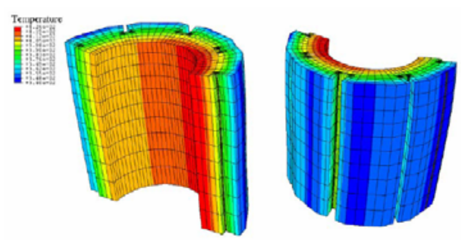
b) Peripheral

Variation in Fluence (Dose), temperature and weight loss



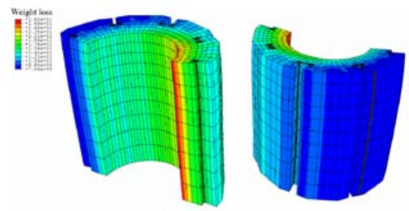
(b) Peripheral brick irradiation fluence distribution

Fluence



(b) Peripheral brick temperature distribution

Temperature

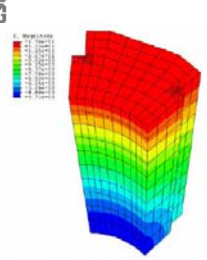


(b) Peripheral brick weight loss distribution

Figure 8: Three-dimensional weight loss distribution at 30.6 fpy

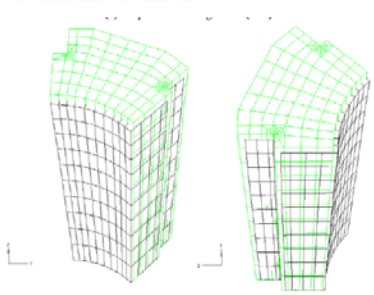
Weight loss

Displacements - Central



(a) Displacement magnitude (mm)

QuickTime™ and a
decompressor
are needed to see this picture.



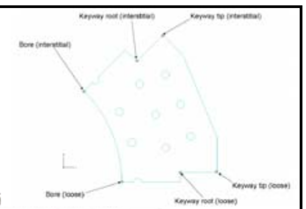
(b) Deformed and undeformed shape

Figure 28: Three-dimensional central brick position displacement after the shutdown step

Hoop stresses – Central Brick – 2D

(b) History profiles for node numbers form the minimum hoop stress curve.

Central Brick
Bore Stresses – 3D



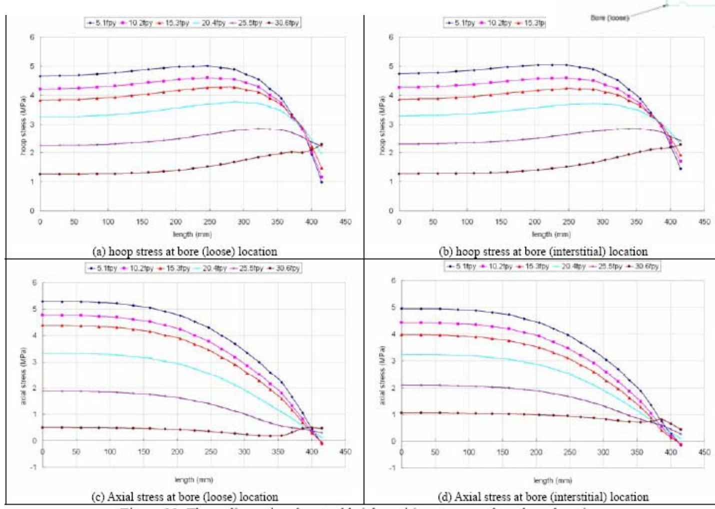
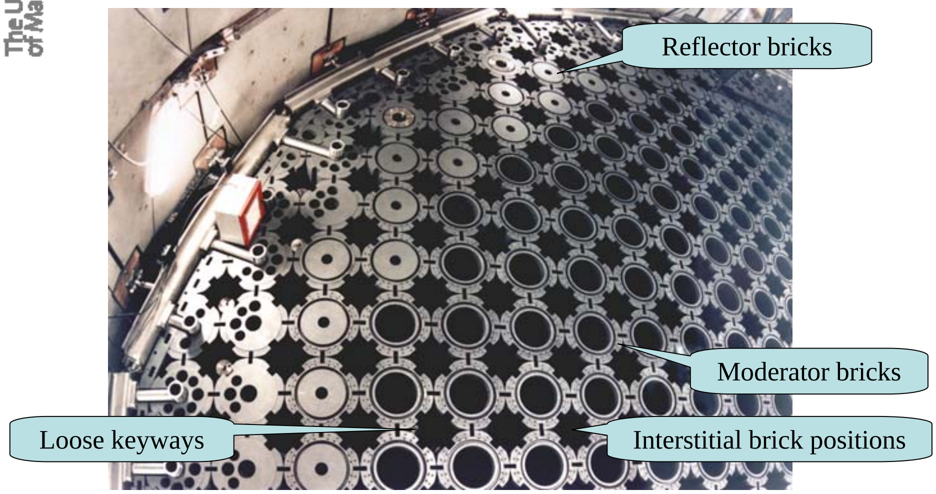


Figure 30: Three-dimensional central brick position stresses along bore locations

AGR core Whole Core Modelling



Construction of AGR Core

AGR Graphite Brick Superelement

- In order to model arrays of AGR bricks supper elements have been developed
- The stiffness matrix for a brick is reduced to only include the nodes at the boundaries thus including the number of degrees of freedom in the finite element model reducing the size and time of assessments
- A special irradiated graphite superelement has been developed for use with the ABAQUS finite element code

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Superelement analysis of moderator brick with a crack

- Superelement for 1/8 brick combined with ABAQUS elements calling the UMAT

QuickTime™ and a
decompressor
are needed to see this picture.

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Superelement

- 10 x 10 fuel bricks model

QuickTime™ and a
decompressor
are needed to see this picture.

QuickTime™ and a
decompressor
are needed to see this picture.

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Characterisation of irradiated graphite waste EC FP7 Carbowaste

- The UK graphite moderated reactors will have produced somewhere in the region of 90,000 tonnes of irradiated nuclear graphite after operation ceases.
- In order to make informed decisions of how best to dispose of this large volume of waste it is necessary to understand the character of the irradiated graphite waste and the effectiveness of the various proposed decontamination and immobilisation treatments.
- This work is funded by Carbowaste and NDA
- The objective is to use microstructural and radiochemical techniques in order to quantify the isotopic location and distribution within the graphite.
- This data will be used to compare with theoretically calculated isotopic inventory from trace elemental analysis.

CARBOWASTE

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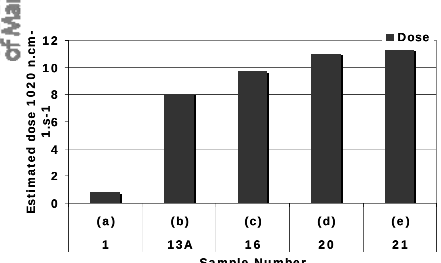
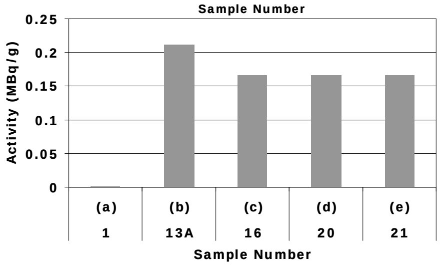
BEPO Graphite Activity





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Isotopic Inventory

- Isotopic Inventory is a result of:
  - Moderator - graphite plus the impurities within.
  - Components of the fuel.
  - Coolant
  - Core Structural Materials - Steel Impurities

Element	Magnox	AGR
Li	0.05	0.05
Be	0.02	0.02
B	0.1	0.5
N	10	10
Na	1.0	4.0
Mg	0.1	0.4
Al	1.0	4.0
Si	35	35
S	50	60
Cl	2.0	4.0
Ca	35	25
Ti	3	0.7
V	12	0.4
Cr	0.35	0.4
Mn	0.04	0.25
Fe	10	28
Co	0.02	0.70
Ni	1.0	6.0
Zn	0.13	1.0
Sr	0.4	0.4
Mo	0.1	2.5
Ag	0.001	0.001
Cd	0.04	0.07
In	0.05	0.06
Sn	0.05	1.0
Ba	1.5	0.5
Sm	0.04	0.05
Eu	0.004	0.005
Gd	0.005	0.01
Dy	0.008	0.006
W	0.12	0.15
Pb	0.12	0.8
Bi	0.08	0.05

Figure 3 Graphite impurity Levels (ppm)*

^{3}H	1.2×10^{14}	^{93}Mo	8.5×10^{8}
^{10}Be	7.1×10^{10}	^{93m}Nb	5.5×10^{8}
^{14}C	8.5×10^{13}	^{94}Nb	1.0×10^{5}
^{36}Cl	9.5×10^{11}	^{99}Tc	1.7×10^{8}
^{41}Ca	7.3×10^{11}	^{108m}Ag	2.3×10^{10}
^{54}Mn	2.7×10^{8}	^{113m}Cd	1.0×10^{10}
^{55}Fe	1.5×10^{13}	^{121m}Sn	4.5×10^{10}
^{59}Ni	9.3×10^{10}	^{133}Ba	5.6×10^{11}
^{60}Co	2.7×10^{13}	^{152}Eu	2.2×10^{11}
^{63}Ni	1.3×10^{13}	^{154}Eu	5.2×10^{12}
^{65}Zn	2.1×10^{8}	^{155}Eu	1.6×10^{12}

Figure 4 Activation Inventory (Bq) of reference Magnox Reactor after 40 years reactor operation followed by 10 years decay.

**White et al. Assessment of Management modes for graphite from reactor decommissioning. CEC 1984.*

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Autoradiography

- Autoradiography is the visual distribution pattern of radiation.
- Autoradiography determine β and γ radiation, not α due to it being stopped by the photographic film which is the recording medium used for this technique.
- Autoradiography differs from the pulse-counting techniques in several ways. Each phosphorous crystal in the photographic emulsion is an independent detector.

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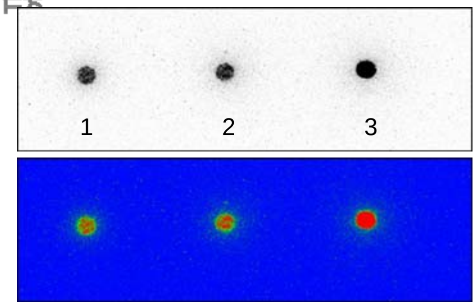
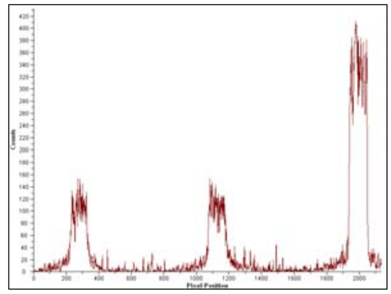
Autoradiography: Low energy detection





- Low energy tritium detection autoradiography (detects all radioactivity above 0.018MeV) results – 3 hour exposure time.
- Grey scales shows different radioactivity intensities.
- The coloured image shows the background in blue, medium intensity in green and high intensity areas in red.

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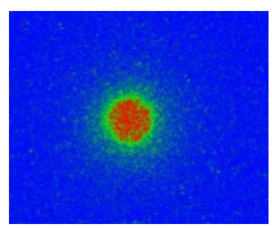
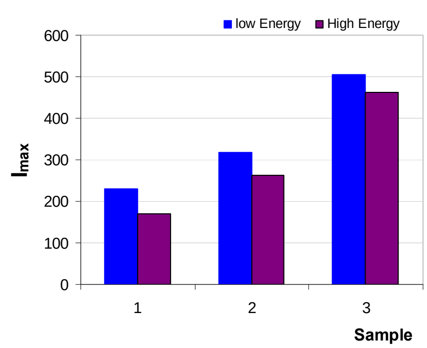
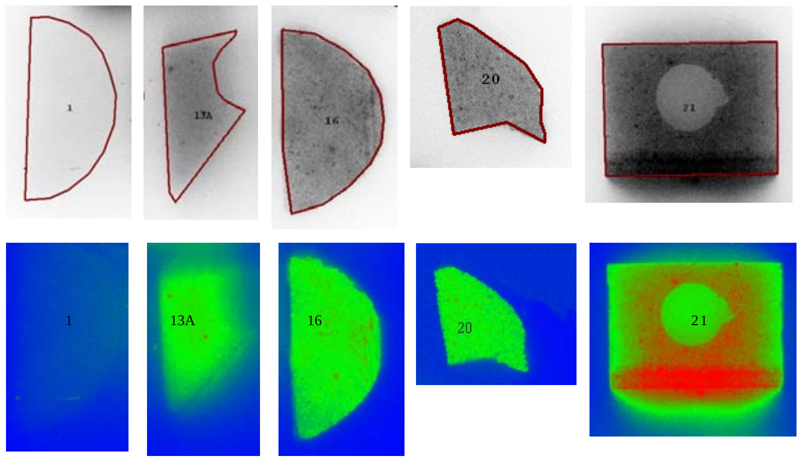
Autoradiography: Comparison



- These preliminary tests show the isotopic inventory is not evenly distributed throughout the graphite matrix.
- Further analysis of intensity shows the isotopic difference between low and high energy films, contributed from tritium present within the graphite.

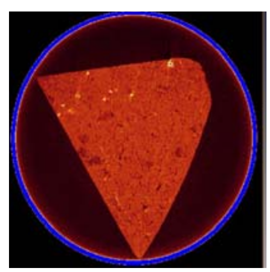


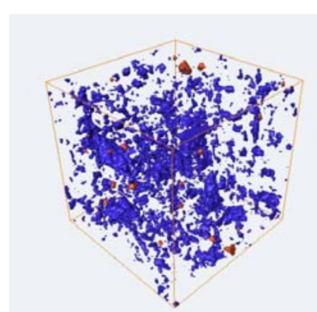
Autoradiography: All BEPO Samples



X-ray Tomography irradiated BEPO 16

- 2D images show slices within the graphite matrix. Several high attenuation spots are visible within the BEPO graphite matrix.
- 3D images show the porosity distribution.
- The blue is the internal porosity and the red is the high attenuation spots.





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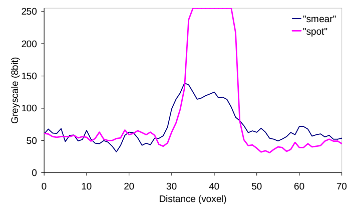
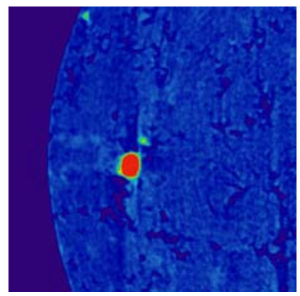
EDX of high attenuation “spot”

- Bright areas in CT reconstruction
  - found in many samples





“spot” (J12)

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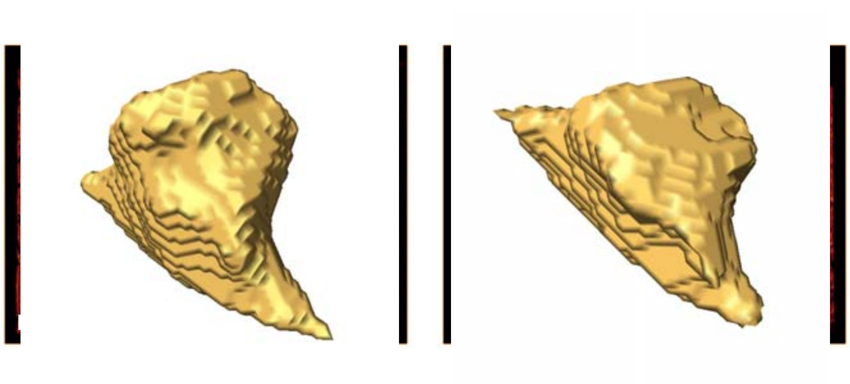
EDX of high attenuation “spot”



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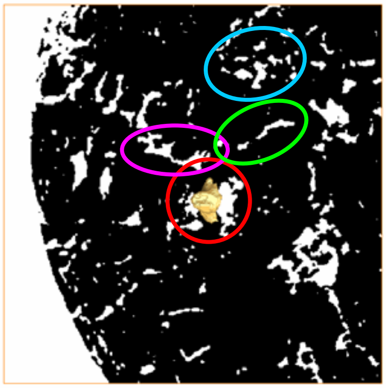
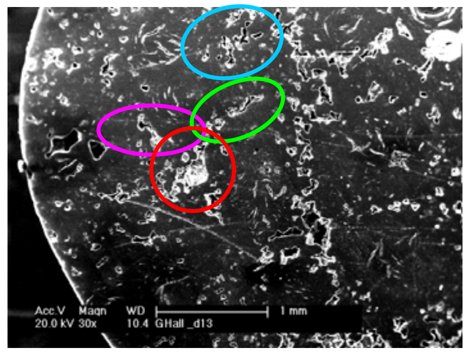
EDX of high attenuation “spot”





tomography

SEM

(SEM and EDX by Dr W. Weaver and Dr A Jones)

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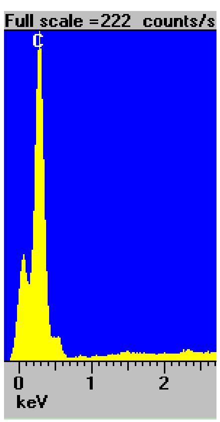
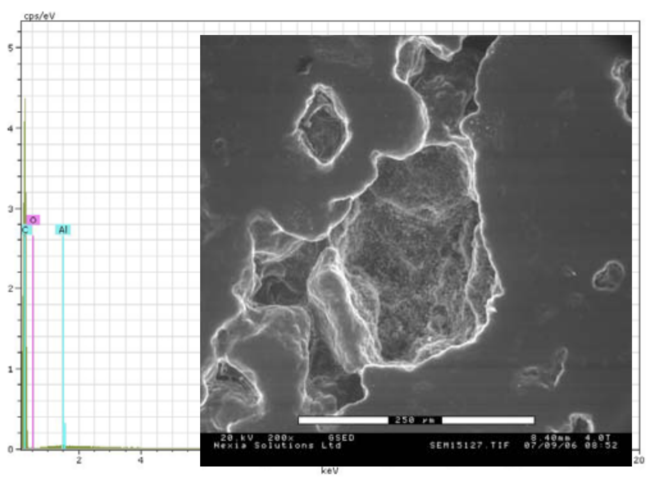
EDX of high attenuation “spot”



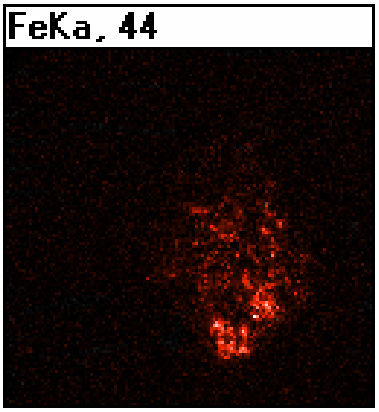


spectrograph of high attenuation area

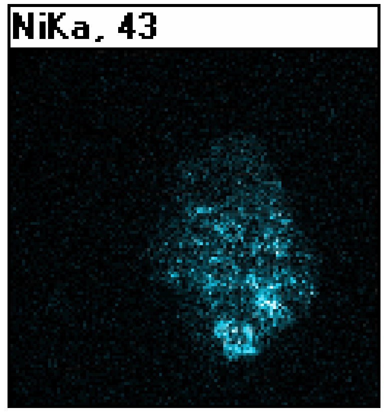
spectrograph of high attenuation “spot”

SEM of high attenuation “spot”

EDX of high attenuation "spot"

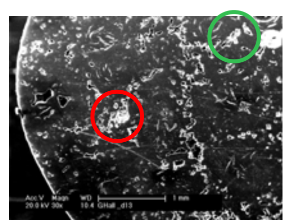


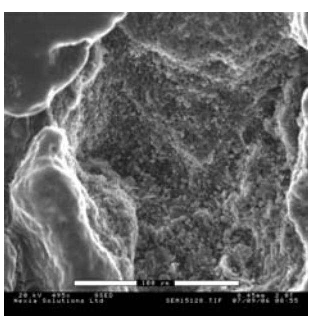
secondary electron image
iron



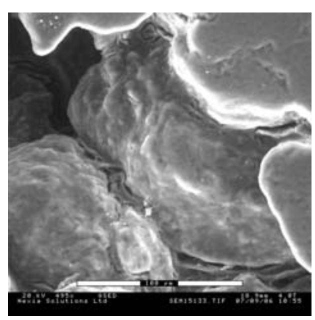
aluminium
nickel

EDX of high attenuation "spot"





high attenuation pore



non-high attenuation pore

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Conclusions

- A significant amount of research is aimed at the continued safe operation of the UK AGRs and remaining Magnox reactors, including the possibility of life extension
- The European Framework Programmes FP5, FP6, FP7 continue with RAPHAEL and CARBOWASTE
- A growing area of European research effort is related to the decommissioning and waste disposal of irradiated graphite waste from graphite moderated reactors (including operational waste) in the UK, France, Lithuania, Italy, Spain and Germany

Acknowledgement

Research discussed in this presentation was funded by the HSE(ND), EPSRC, British Energy, Magnox, European Commission and the NDA

PBMR Research Activities

Mark Mitchell, Scott Penfield, Shahed Fazluddin, Mary Fechter

Presented to: WORKSHOP ON NUCLEAR GRAPHITE RESEARCH, Organized by ORNL and Sponsored NRC. Legacy Hotel and Meeting Center

Rockville, MD – March 16-18, 2009

Overview

- Introduction
  - PBMR and South African framework
  - R&D at PBMR
  - Graphite Reactor Safety Case Model and Life Cycle.
- Activities and Priorities
  - Prioritisation of work:
    - Work supporting build programme
    - Mid Term Research Objectives
  - Presentation of Example Activities
- Integration with International programmes.

PBMR in South Africa

- The following roles are defined in the South African Nuclear Energy Policy*:
- South African Nuclear Energy Corporation (NECSA):
  - The main functions of NECSA are to undertake and promote research and development in the field of nuclear energy and radiation sciences and technology;
  - to process source material, special nuclear material and restricted material and
  - to co-operate with persons in matters falling within these functions.
- The South African utility Eskom
  - Eskom is the owner and operator of the Koeberg Nuclear Power Station. Construction of Koeberg's two reactors commenced in 1976 under a turn-key contract and they have operated safely in the more than 20 years since their commissioning in 1984 and 1985 respectively.
  - Koeberg supplies 1800 MWe to the national grid when both reactors are operating at full power contributing approximately 6% of South Africa's electricity.
  - South Africa’s expertise with respect to the management, operation and maintenance of nuclear power plants resides in Eskom.
- Pebble Bed Modular Reactor (Pty) Ltd. (PBMR)
  - The PBMR Company is developing a fuel manufacturing plant and a first of fleet demonstration plants designed to meet Generation IV requirements, applicable to both electricity generation and process heat applications.

*DEPARTMENT OF MINERALS AND ENERGY. NUCLEAR ENERGY POLICY FOR THE REPUBLIC OF SOUTH AFRICA. JUNE 2008

Implications to R&D at PBMR

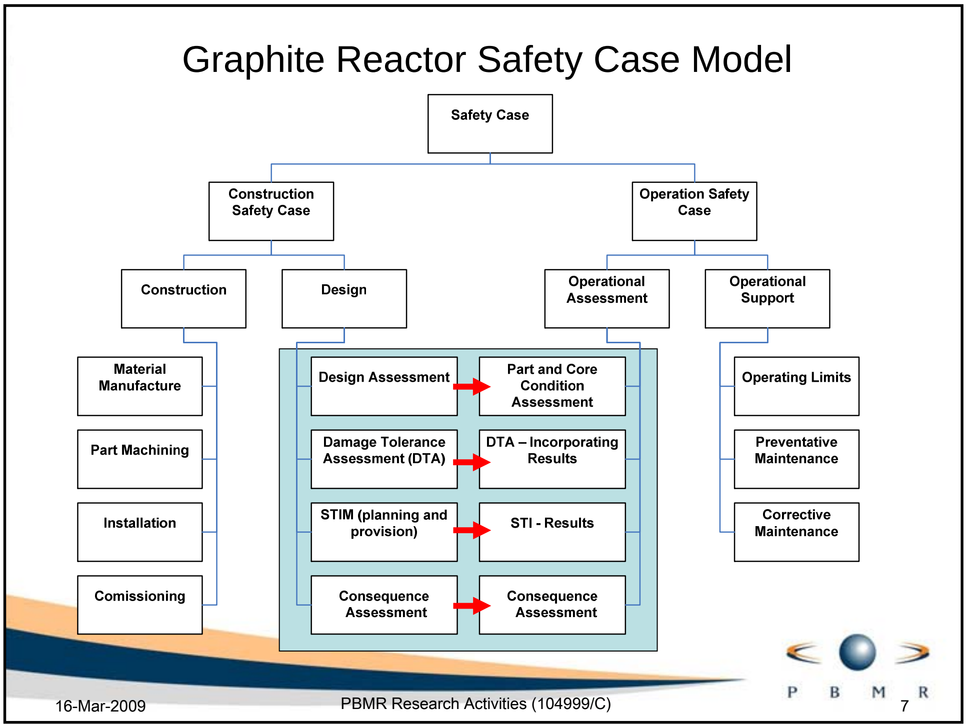
- PBMR fulfils the role of Nuclear vendor.
- PBMR Focuses on development of the Fuel, Reactor System and it’s components, and minimisation of waste.
  - Research efforts directed at near term needs
    - Focus on applications, qualification and characterisation.
  - Expands to consider technology growth path for the PBMR products
  - Collaborate with NECSA and other institutions such as universities where possible on longer term R&D objectives.
- In terms of graphite, the majority of the work in South Africa takes place at PBMR, focussed on supporting our product.
  - PBMR contributes to the Carbon Chair at the University of Pretoria which was established by the DST.
  - PBMR has research contracts with several other universities and companies in this area.

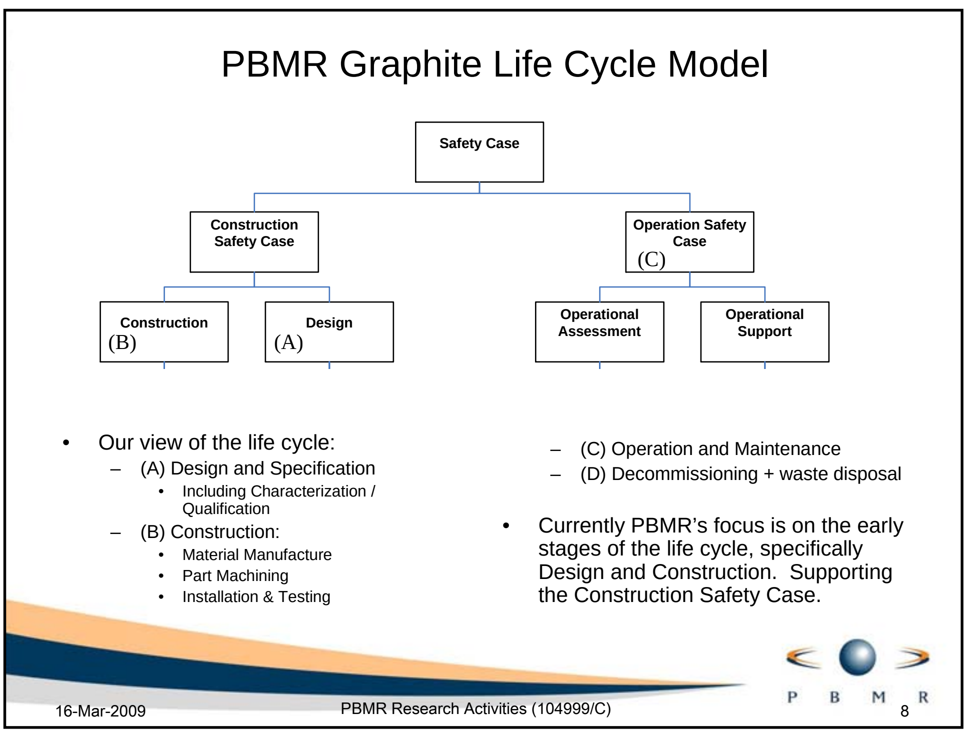
Classification of Areas of Interest

- In order to understand the areas that are receiving attention for graphite, the following information must be considered:
  - South African Regulatory Requirements
  - The Graphite Life-cycle Model: How do we see the graphite life cycle.
  - Graphite Reactor Safety Case Model: The model that is implied by the planned PBMR safety case.
- These two models overlap as the Safety Case Model shows the safety case for the various stages of the life cycle. Both models respond to the South African Regulatory Requirements.
- PBMR has recently completed an effort within the NGNP Project to reconcile the results of the October 2007 PIRTS to U.S. NGNP requirements

South African Regulatory Requirements

- The NNR is not prescriptive, however some specific guidelines and regulations are provided. The key requirements documents are described below.
  - RD-0016 "Requirements for Licensing Submissions Involving Computer Software and Evaluation Models for Safety Calculations"
  - RD-0018 "Basic Licensing Requirements for the Pebble Bed Modular Reactor"
  - RD-0019: Requirements for the Core Design of the Pebble Bed Modular Reactor
  - **LD-1097 "Qualification of the Core Structure Ceramics of the PBMR"**
  - RD-0034 "Quality and Safety Management Requirements for the PBMR"


Graphite Reactor Safety Case Model
Safety Case
Construction Safety Case
Operation Safety Case
Construction
Design
Operational Assessment
Operational Support
Material Manufacture
Part Machining
Installation
Comissioning
Design Assessment
Part and Core Condition Assessment
Damage Tolerance Assessment (DTA)
DTA – Incorporating Results
STIM (planning and provision)
STI - Results
Consequence Assessment
Consequence Assessment
Operating Limits
Preventative Maintenance
Corrective Maintenance
16-Mar-2009
PBMR Research Activities (104999/C)
P B M R
7


PBMR Graphite Life Cycle Model
Safety Case
Construction Safety Case
Operation Safety Case
(C)
Construction
(B)
Design
(A)
Operational Assessment
Operational Support
• Our view of the life cycle:
– (A) Design and Specification
• Including Characterization / Qualification
– (B) Construction:
• Material Manufacture
• Part Machining
• Installation & Testing
– (C) Operation and Maintenance
– (D) Decommissioning + waste disposal
• Currently PBMR's focus is on the early stages of the life cycle, specifically Design and Construction. Supporting the Construction Safety Case.
16-Mar-2009
PBMR Research Activities (104999/C)
P B M R
8

Activities

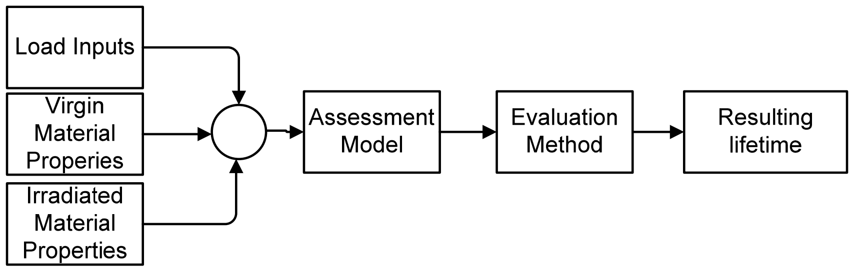
- Work is prioritised:
  - First priority:
    - Design Assessment and Specification
    - Manufacture and Quality Control
    - Establishment of Surveillance, Test Inspection and Maintenance (STIM)
  - Next Priority:
    - Mid-term research goals, focussed on incremental improvements for following reactors.

- Notes:
  - PBMR is committed to the establishment of international standards. Participation in ASTM and ASME.
  - This approach is pragmatic and areas that are of great longer-term interest such as security of supply of graphite raw materials are at present assigned lower priority in the near-term, this is however included in our Technology Development Roadmap.

Design Assessment

- Design Assessment Comprises the following major activities:
  - Integrated Assessments, determine the loads that parts are subjected to due to:
    - Core physics analysis
    - Coolant Flow
    - Temperatures
    - External loads
    - Interaction between parts in the assembly
  - And, Part Assessment, determining the suitability of the parts that comprise the CSC to resist the applied loads. This includes:
    - Non-irradiated part assessment, and
    - Lifetime assessment to determine the life of irradiated parts
  - Additional specialist assessments are completed such as Oxidation (Thermal and Radiolytic) to comply with LD-1097 requirements.
  - Additional Safety Analysis, specifically relating to Dust and Release Calculations.

PBMR Graphite life assessment process



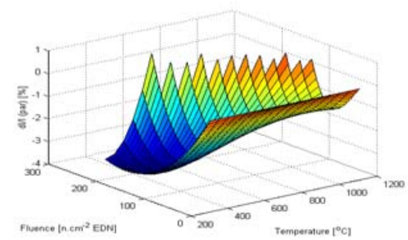
- Described as follows:
  - Load inputs: Flux and temperature fields, external loads etc.
  - Virgin Material Properties: Determine representative material properties for the selected material (Graphite NBG-18)
  - Irradiated Material Properties: Determine representative irradiated material properties
  - Stress analysis: Complete stress analysis to predict the through life stresses in both
    - Non-irradiated parts, and
    - Irradiated parts
  - Stress evaluation, to compare the stress analyses to acceptable limits.

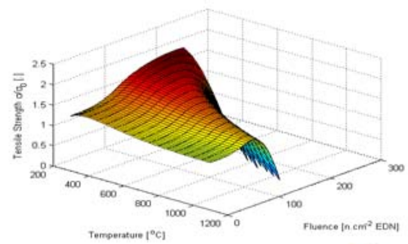
Key Elements Supporting Design Assessment

- Material Selection / Quality Requirements
- Material Production Qualification
- Virgin material characterisation:
  - Material production qualification and property measurement on trial material.
  - Gather data on all properties required in LD-1097 and additional properties used for design.
  - Determination of variability of properties (between charges, within charges and within billets)
  - Provision of design values to be used for design, coupled with determination of specification values to ensure consistent production.
  - Multiple Properties (Strength, Thermal conductivity, Emissivity, …)

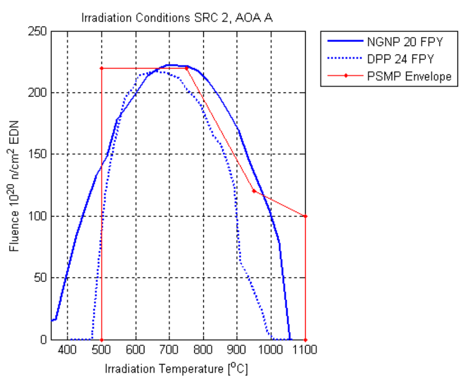
Activities Supporting Design Assessment (2) – Irradiated Properties

- Model based on ATR-2E and VQMB data
  - Dimensional change
  - Volume change
  - CTE
  - Thermal conductivity
  - Elastic Modulus
  - Strength
  - Irradiation creep
- The statistical validity of the empirical fits is ensured through the implementation of polynomials of low order with statistically significant coefficients.
- To be Verified by PSMP.





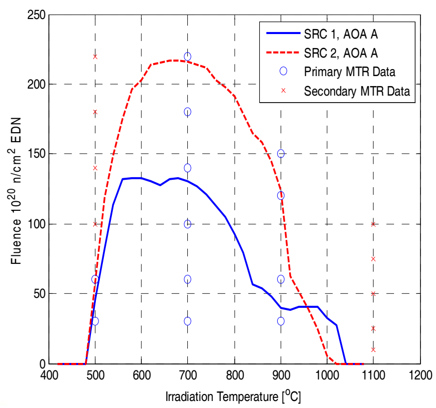
Activities Supporting Design Assessment - PSMP

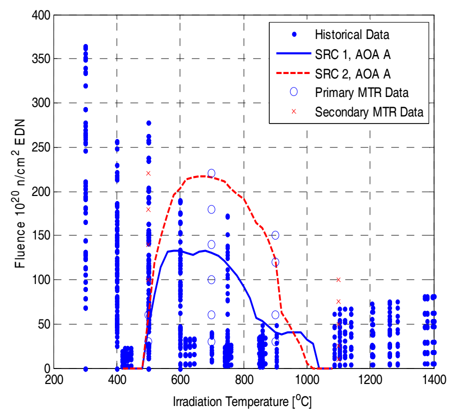


Based on preconceptual design results (Oct-2007)

- PSMP = PBMR Specific Material Test Reactor Programme.
- This programme is designed to characterise the behaviour of NBG-18 under DPP operating conditions to support operation of the CSC. The programme includes the following experiments:-
  - High and low fluence testing of NBG-18, augmented by an extended irradiation strength testing programme.
  - Testing of other materials of interest: SiO_2 based rigid ceramic insulation; Composite materials for control rod applications.

Activities Supporting Design Assessment – PSMP (2)





PSMP Planned Material Property Tests

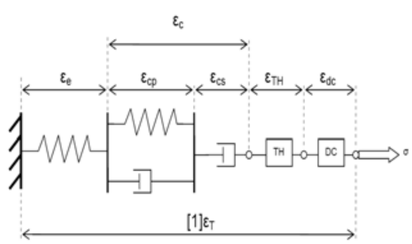
	ID	Property	Direction	Proposed Test Method	Minimum Number of Test Specimens / Remark
PHYSICAL	1.	Mass	Not Applicable	No Standard	Perform measurements on all irradiation test specimens for each irradiation condition
	2.	Dimensions	WG, AG		Measure lateral dimensions on all irradiated specimens for each condition, i.e. AG, WG
	3.	Volume	Not Applicable		
	4.	Density	Not Applicable	ASTM C1039	
	5.	Open Porosity	Not Applicable		
	6.	Pore Size Distribution	Not Applicable	No Standard	Measure 3 select specimens per direction per condition Preference to be given to helium intrusion
	7.	Electrical Resistivity	WG, AG	ASTM C611	6 specimens per direction per condition
	9.	CTE (20°C - 1600°C)	WG, AG		Measure 3 select (20-200°C) specimens per direction per condition
	11.	Thermal Conductivity (RT - 1600°C)	WG, AG		Measure 3 select (RT) specimens per direction per condition
	13.	Microstructural Examination	Not Applicable	No Standard	Examine 3 select specimens per condition
	14.	Activity (Gamma Spectrum)	Not Applicable	No Standard	Measure 2 select specimens per condition after a minimum of 2 decay periods
MECHANICAL	15.	Dynamic Elastic Modulus	WG, AG	ASTM C1198 /	6 specimens per direction per condition
	16.	Shear Modulus	WG, AG	ASTM C1259	
	17.	Poisson's Ratio	WG, AG		
	18.	Tensile Strength	WG, AG	ASTM C749	6 specimens per direction per condition
	19.	Strain to Failure	WG, AG	ASTM C749	Derive from stress-strain measurement for each specimen

Irradiation Creep

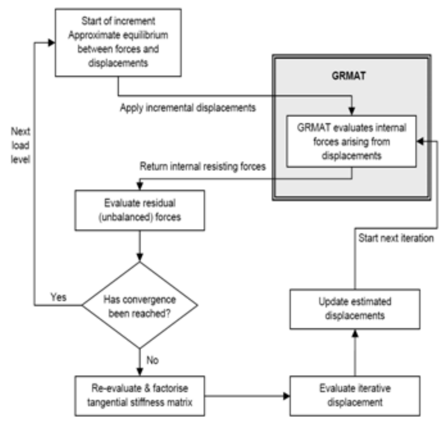
- For Design Assessment, PBMR uses models based on historical data.
  - Material independent
  - Relate Creep to other properties
- Question on creep behavior after turnaround.
- PBMR plans via cooperative efforts to acquire data from our GenIV partners to validate the models in this area.
- This phenomenon affects our certainty on the end of life. Uncertainty leads to the introduction of large margins and increased inspection.

Activities Supporting Design Assessment – Stress Analysis

- GRMAT implementation described in figures.



$$\varepsilon_{TOT} = \varepsilon_e + \varepsilon_{cp} + \varepsilon_{cs} + \varepsilon_{TH} + \varepsilon_{dc}$$

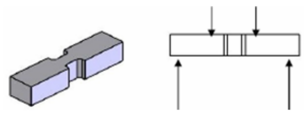


- The accounts for elastic, thermal, dimensional changes and creep strains.

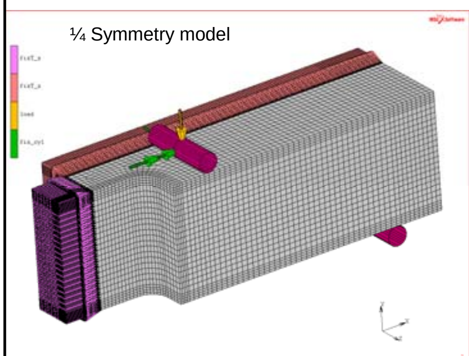
Activities Support Design – Fracture model validation

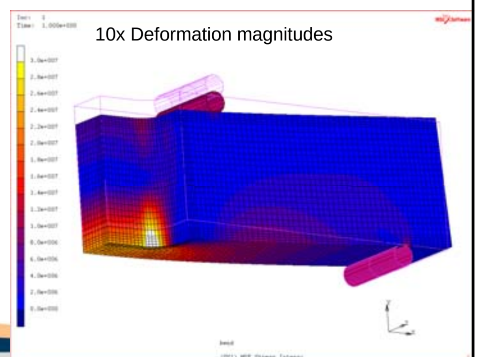
•Example – Failure prediction validation
AN-18 Dog-bone (Test direction 1)

• 10 Samples of 4 Different fillet radii tested
- 1mm
- 5mm
- 10mm
- 20mm



Fillet radius	Median test failure load (N)	Predicted median failure load (N)	Predicted / Test
1 mm	9663	7910	81%
5 mm	10249	8764	86%
10 mm	10991	9142	83%
20 mm	11264	9766	86%





16-Mar-2009 PBMR Research Activities (104999/C) 19

Mid Term Research Activity Focus

- Improved plant performance through management of margins
  - Better understanding of the margins and uncertainties in the design will lead to the ability to improve the efficiency of the design, enhance performance and reduce operating cost.
- Improved plant performance through better quality control on materials and parts
  - The more effective the quality control applied in the manufacture of the CSC materials and components, the better the overall plant performance. This will primarily be due to reduction in outages for repair or replacement and improved confidence in the hardware.
- Improved plant performance through the ability to effectively monitor the performance and condition of the plant in operation
  - The ability to effectively determine the condition of the CSC in service is important to successful plant operation. Any advances made in this field will be beneficial to PBMR.

16-Mar-2009 PBMR Research Activities (104999/C) 20

Integration with International Programmes

- Gen IV International forum, VHTR, Materials and Project Management Group
- 1.1 WP 1 GRAPHITE
  - Task 1. Data, Design Methodology and Construction
    - Task 1.1 Graphite Selection and Qualification Strategy
    - Task 1.2 Graphite Physical and Mechanical Properties (virgin material)
    - Task 1.3 Graphite Fracture Behaviour
    - Task 1.4 Graphite Oxidation Behaviour
    - Task 1.5 Graphite Component Testing
    - Task 1.6 Graphite Irradiation Effects
    - Task 1.7 Graphite Irradiation Induced Creep
    - Task 1.8 Graphite Codes & Standards Development
    - Task 1.9 Graphite Behaviour Model Development
    - Task 1.10 Links to Existing Graphite Irradiation Behaviour Databases
    - Task 1.11 Materials data base
  - Task 2. Data, Design Methodology and Construction
    - Task 2.1 Graphite Qualification Strategy
    - Task 2.1 Methodologies for ISI of graphite cores
  - Task 3. Decommissioning and Disposal
    - Task 3.1 Literature Survey
    - Task 3.2 Defining of project
    - Task 3.3 Decontamination
    - Task 3.4 Recycling
    - Task 3.5 Disposal
- Possible collaboration with other parties.

Conclusion

- PBMR follows a comprehensive approach to developing core graphite technologies for use in the reactor fleet.
  - The examples provided are indicative of both the breadth and depth of the scope.
- This approach is prioritized and supported by mid term R&D objectives.

WORKSHOP ON NUCLEAR GRAPHITE RESEARCH

Organized by ORNL and Sponsored by NRC

Legacy Hotel and Meeting Center, Rockville, MD, March16-18, 2009

JAPANESE RESEARCH ACTIVITIES

Motokuni Eto, Technical Consultant, Toyo Tanso Co., Ltd.
Taiju SHIBATA, High Temperature Fuel & Material Group, JAEA

2009年8月17日

東洋炭素株式会社

Goal of HTTR Project

2005 → 2020-30

Reactor Technology (HTTR)
- Attainment of reactor-outlet coolant temperature of 950℃ (April, 2004)
- Safety demonstration test
- Long-term operation at 950℃

Hydrogen Production Technology
IS Process
- Completion of 1 week continuous hydrogen production (Jun, 2004)
- Improvement of system efficiency
- Pilot test (under planning)

System Integration
- Safety evaluation
- Isolation valve tests

HTGR Plant Design and Gas Turbine Technology
- Design of cogeneration HTGR system (GTHTR300C)
- Tests of compressor, magnetic bearing etc.

Hydrogen Production with HTTR-IS System ($1000 m^3/h$)

Commercial HTGR System

Hydrogen production for commercial use in 2020s

2

Future Plan – R&D items -

JAEA | TOYO TANSO Inspiration for Innovation

Medium-range plan in JAEA → To be proposed

		2005	2010	2015	2020
		VHTR(GTHTR300C) Conceptual design	Detailed design	Evaluation	
Reactor	HTTR test	Performance tests Safety tests	Reactor-IS simulation		
	Fuel	Irradiation test on burnup, Manufacture of ZrC,	High-burnup SiC Irradiation of ZrC		
	Material	Graphite test C/C component test	Irradiation tests		
Heat Utilization	Components	Compressor (Gas turbine) IHX			
	Hydrogen production	Data base system	Pilot plant test	HTTR-IS	

3

Safety Demonstration Test

JAEA | TOYO TANSO Inspiration for Innovation

FY	2002	2003	2004	2005	2006		2009
CRW	★	★	Delayed	★	(Phase 1 in progress)		
TGC1, 2	★	★		★			
TGC3				(Phase 2 under preparation)		★	
VCSS						★	
OT				(Phase 2 planned)		★	

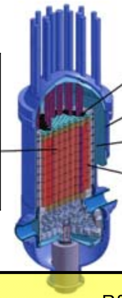
CRW Reactivity insertion test (Control rod withdrawal test)
TGC1, 2 Gas circulators trip test
by running down one and two out of three gas circulators
TGC3 Loss of forced cooling test
by running down all gas circulators
VCSS Vessel cooling system stop test
OT Off-normal load condition test of heat utilization system, etc.

4

R&D Items for VHTR

JAEA | TOYO TANSO Inspiration for Innovation

	HTTR	VHTR
Power density	2.5 MW/m^3	5.8 MW/m^3
Burn-up	22 GWD/t	150~200 GWD/t
Fluence	~ 1.5×10^{25}n/m^2	~ 6×10^{25}n/m^2



Component	HTTR	VHTR	R&D
Fuel, Reflector blocks	IG-110	IG-110 (primary) IG-430 (advanced)	Global code/ standard Irradiation data Lifetime extension

IG-430 has 20-40% higher strength than IG-110

IG-110 is mature nuclear graphite and primary candidate for VHTR.

Property Data and Proof Test on IG-110 for HTTR Development

JAEA | TOYO TANSO Inspiration for Innovation

- ✓Property data
  - Mechanical and thermal properties
  - including irradiation and oxidation effects
- ✓Proof tests on IG-110 graphite components
  - Bottom structure seismic test
  - Core components seismic test
  - Support post bucking test
  - Dowell/ socket fracture test
  - Key/ keyway fracture test

Strength Data for IG-110

TOYO TANSO Inspiration for Innovation

The first loaded IG-110 graphite in the HTTR

	Average (MPa)	Standard deviation (MPa)	Number of specimens	Su value (MPa)
Tensile strength	29.6	1.49	640	26.1
Compressive strength	82.6	2.36	320	76.9

Su values for tensile and compressive strength were decided Survival probability of 99% at confidence level of 95%

The first loaded IG-110 has excellent performance.

It is possible to increase the Su values for proven IG-110 graphite. It gives lifetime extension of components.

Typical Properties of IG-110 and IG-430 Graphites

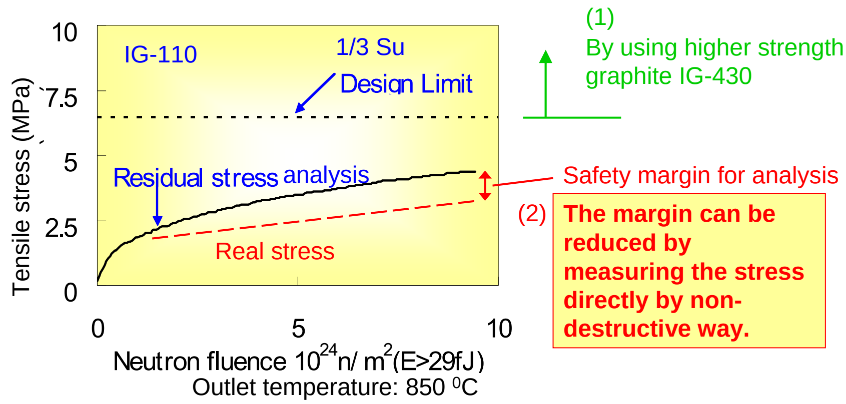
TOYO TANSO Inspiration for Innovation

Grade	IG-110	IG-430
Raw material	Petroleum pitch	Coal-tar pitch
Bulk density (Mg/m^3)	1.78	1.82
Tensile strength (MPa)	25.3	**37.2**
Compressive strength (MPa)	76.8	**90.2**
Young's modulus (GPa)	8.3	10.8
Poisson's ratio	0.14	
Su value for tensile (MPa)	19.4	
Su value for compressive (MPa)	61.4	

Both graphties are fine-grain graphite

IG-430 has 20-40% higher strength than IG-110

For Lifetime Extension of Components



Two possible solutions for lifetime extension

(1) use advanced graphite

(2) evaluate stress by measurement

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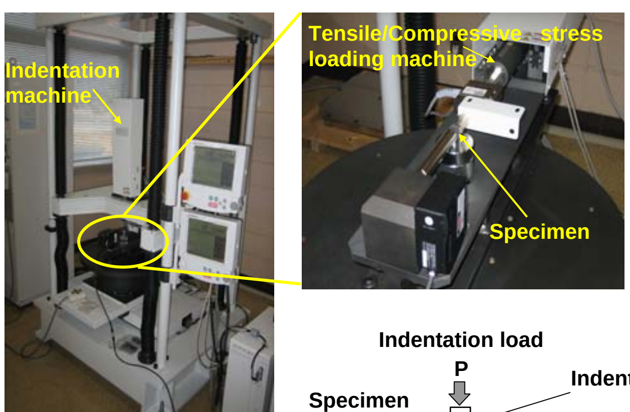
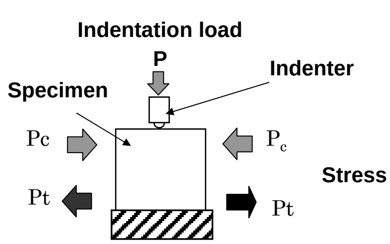
Apparatus for Micro-indentation Test





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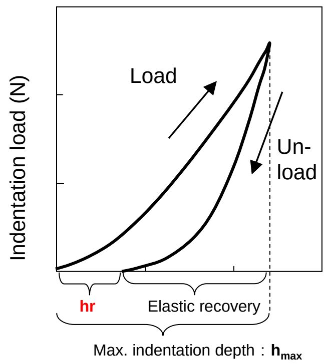
Load-depth curves for micro–indentation test

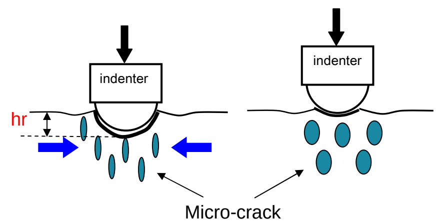
JAEA | TOYO TANSO Inspiration for Innovation

Plastic deformation model for indentation load at compressive stress



hr : depth by plastic deformation

Compressive load decreases micro-crack accommodation at un-loading process



Compressive

No-stressed
Tensile

Plastic deformation accommodates load

Micro-cracks accommodate load

⇩

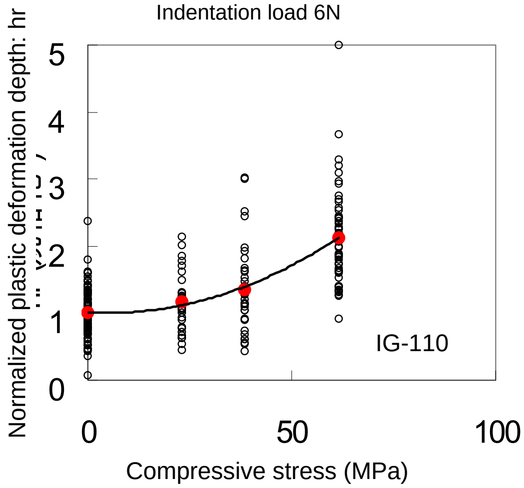
Compressive stress will be evaluated by plastic deformation

Plastic Deformation Depth after Un-loading

JAEA | TOYO TANSO Inspiration for Innovation

Compressive stress was given by testing machine on IG-110 sample

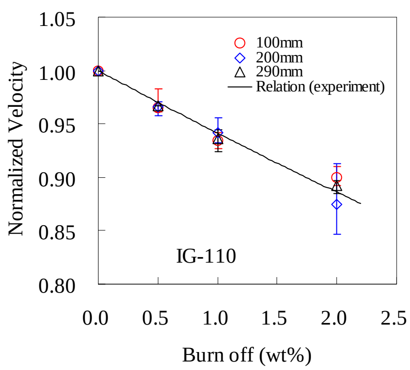
Spherical indenter (R=0.5mm) ,
Indentation load 6N

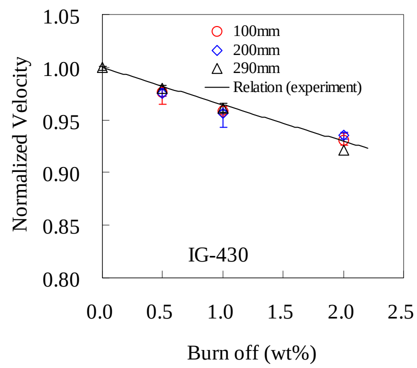


- plastic deformation depth hr was increased by compressive stress as expected
- it is possible to evaluate compressive stress condition by micro-indentation characteristics

- variation of data
  - -reduce by improving experimental condition
  - -evaluated by statistic approach

Ultrasonic Wave Velocity as a Function of Oxidation Burnoff





Specimen size 100×200×290(mm)

Oxidation temperature 500℃

$$v/v_0 = \exp(-6.04B) \quad \text{for IG-110}$$

$$v/v_0 = \exp(-3.64B) \quad \text{for IG-430}$$

v/v_0 **: velocity ratio**

B **: burn-off**

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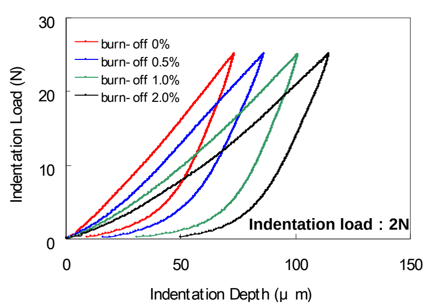
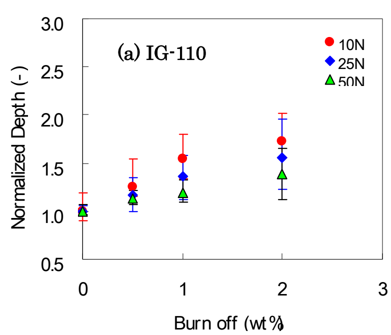
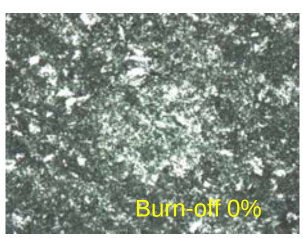
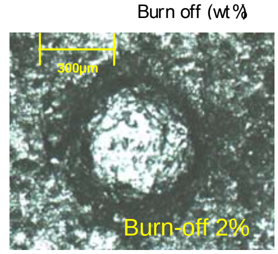
Micro-indentation Test for Oxidized IG-110









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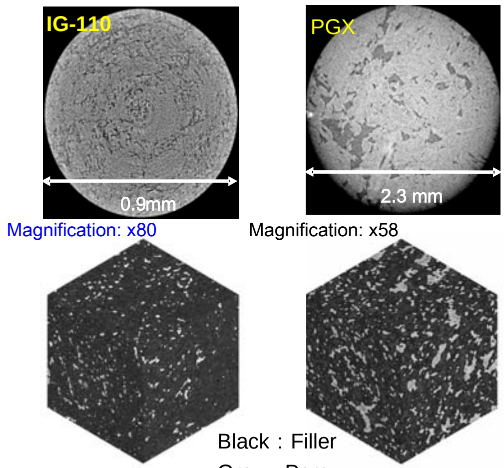
Development of 3D-Xray CT Technique for IG-110 and PGX Graphites



3D-Xray CT images (upper)
3D-image model for filler and pore (lower)

Evaluation for irradiation effects

- Coarse-grained PGX
  Normal magnification of x58
- Fine-grained IG-110 :
  3D-image was successfully obtained by using high magnification of x80

Next study

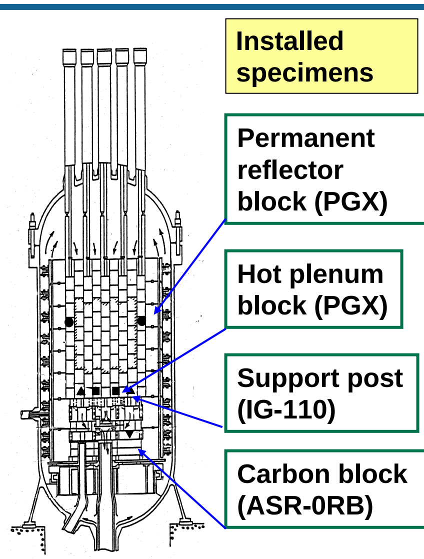
- Modeling of irradiation effects on filler and pore
- Characterization with graphite material property data
- Demonstration by irradiated IG-110 data

In-Service Inspection (ISI) of Graphite

1. TV camera monitoring

2. Surveillance test
   - Dimensional change
   - Bending strength
   - Compressive strength
   - Surface oxidation rate
   - Young's modulus

HTTR graphite blocks can be measured at refueling period.



CTE Measurement in HTTR ISI

Irradiation-induced changes in CTE of IG-110, PGX and ASR-0RB will be measured at ISI of HTTR graphite components

- Specimens for CTE (Coefficient of Thermal Expansion) are installed in the HTTR graphite blocks

- CTE measurements will be carried out by taking out the specimens from blocks after the refueling.

Mechanical properties measurement in HTTR ISI

Irradiation-induced changes in mechanical properties will be measured at ISI of HTTR graphite components

- Grade: IG-110, PGX and ASR-0RB

- Dimensional change, bending strength, compressive strength, dynamic Young's modulus and thermal diffusivity

- Specimens are installed in the HTTR graphite blocks

Irradiation Induced Creep in HTTR ISI

Irradiation-induced creep will be measured by the loaded IG-110 block in the HTTR.

- Measurement will be carried out by taking out the specimens from blocks after the refueling.
- Fuel blocks have residual stress which is given by irradiation-induced creep effect on graphite.
- The stress will be released by cutting the block with residual stress. The creep effect will be evaluated by the dimensions of graphite block before and after the cut.

19

Blockage of Fuel Element Coolant Channel or Reactivity Control Rod Block

Evaluation of FP transport by the fuel failure test simulating a blockage.

- Some of the coolant flow pass in the irradiation test blocks with fuels will be blocked by plugs.
- It simulates blockage of fuel element coolant channel due to graphite failure.

20

IRRADIATION EXPERIMENT

Present Status of Irradiation Research Activities:

- **Experimental Data (Graphite Characterization Group: JAEA/Toyo Tanso)**

  Higher fluence, wider temperature range data needed.

  - Irradiation planned in Joyo, JAEA;
  - Collaboration with Petten Programs, INL ATR Irradiation Programs

- **International Collaborations**

  GIF (Generation IV International Forum)

  - VHTR R&D Project, Material Project Plan, Graphite: Irradiation Creep

  IAEA CRP

  - Development of Comprehensive Creep Model and Equations

コンテンツタイトル

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        - » 本文 5

Health and Safety
Executive

Nuclear Graphite Research
A UK Regulatory Perspective

G B Heys

HM Principal Inspector (Nuclear Installations)

HM Nuclear Installations Inspectorate (NII)

Topics

- Background
- Regulatory regime
- Regulatory challenges and strategy
- Regulatory guidance relating to graphite
- Safety case improvements
- Graphite nuclear safety research
  - Research arrangements
  - Graphite research strategy
  - Current graphite research programme

Health and Safety
Executive

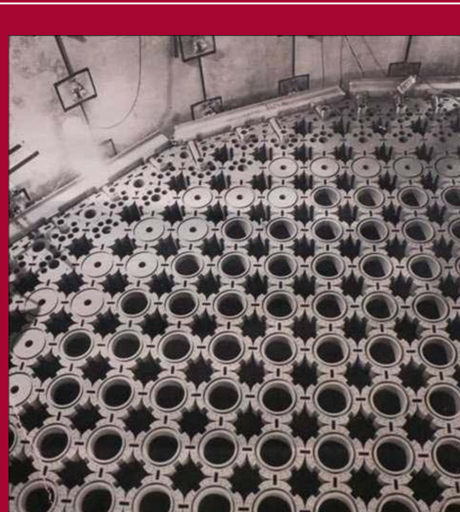
Background

Background

- Around 20% of UK electrical power supply comes from:
  - 14 Advanced Gas Cooled Reactors
  - 4 Magnox reactors
  - 1 PWR
- 18 of these reactors are graphite moderated
- Current planned closure dates for AGRs between 2016 and 2023
- Last of Magnox reactors due to cease generation in 2012

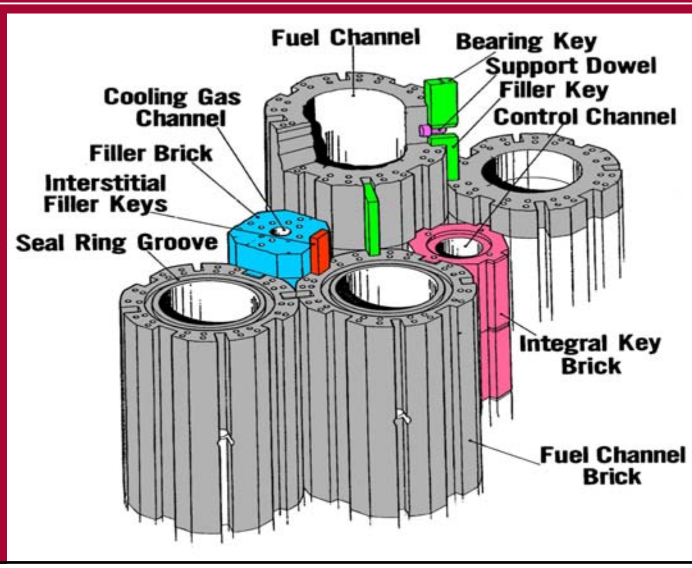

UK Nuclear Sites
Magnox Reactor
Advanced Gas Cooled Reactor
Research Reactor
Pressurised Water Reactor
Chemical Plants & Other Facilities
Submarine Facilities
Dounreay
Coulport & Faslane
Rosyth
Torness
SURR
Hunterston
Chapelcross
Hartlepool
Sellafield
Calder Hall
ICI Billingham
Barrow
Heysham
Springfields
Wylfa
Capenhurst
Trawsfynydd
Derby
Sizewell
Harwell
Berkeley
Amersham
Bradwell
Cardiff
Oldbury
Ascot
Hinkley Point
Aldermaston & Burghfield
Winfrith
Dungeness
Devonport

AGR Core



AGR Core

AGR brick design



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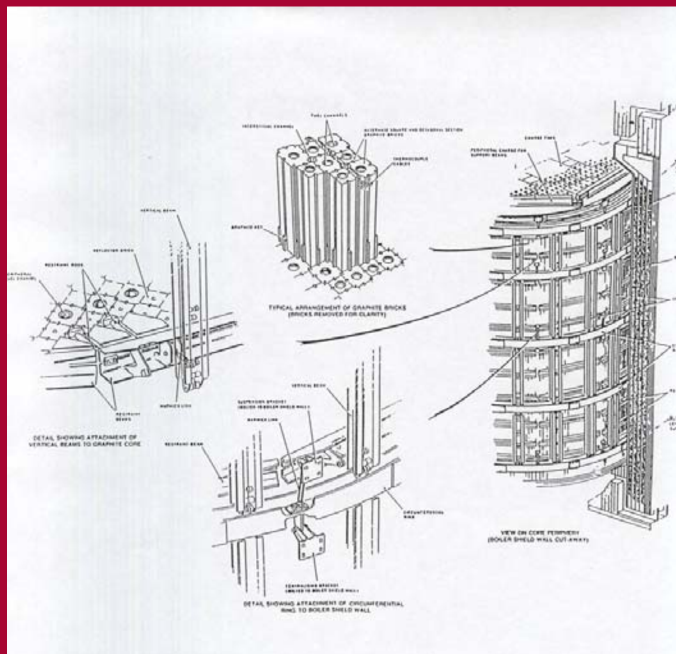
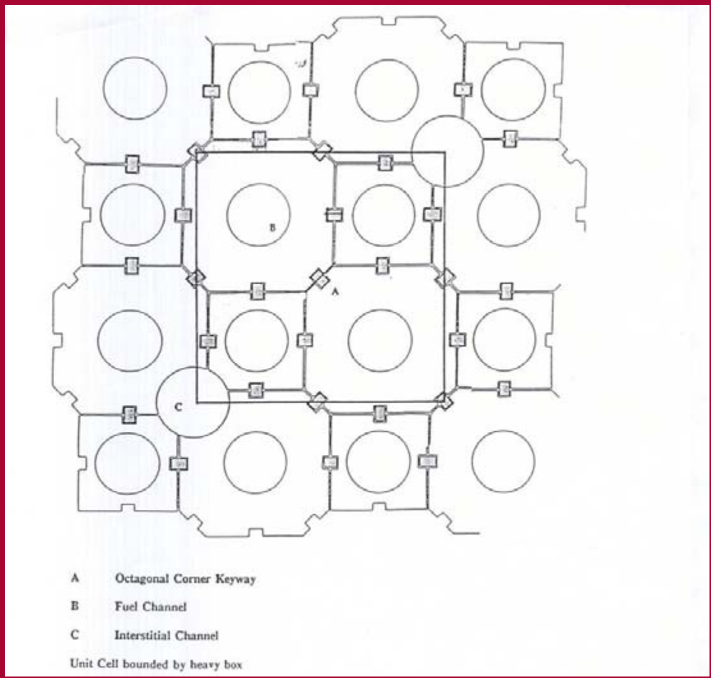


Magnox core

Magnox core

HSE



Magnox core plan view

Health and Safety Executive

HSE

Regulatory Regime

Regulatory Regime

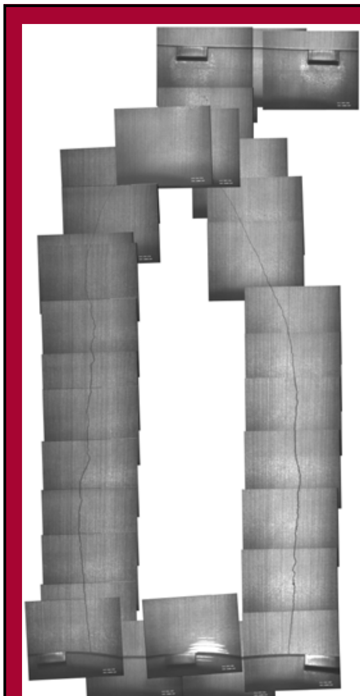
- Health and Safety at Work Act (1974)
  - Health and safety of employees and public
  - Non-prescriptive, goal setting regime
  - Duties on employers and employees
- Nuclear Installations Act 1965 as amended
  - Act established licensing system
  - NII can attach conditions to the site licence in the interest of safety (without recourse to parliament or ministers)
- Conditions attached to nuclear site licence
  - Licence issued by NII to individual sites
  - Compliance with licence conditions is mandatory
  - Safety of an activity should be demonstrated before the activity is undertaken - permissioning
  - Licensee must be in adequate control of safety related activities and an intelligent customer for services provided
  - Licensees set their own design safety standards and criteria

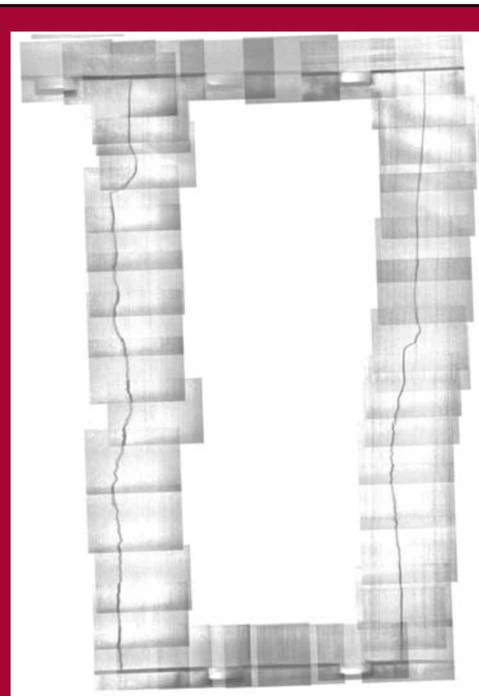
Health and Safety
Executive

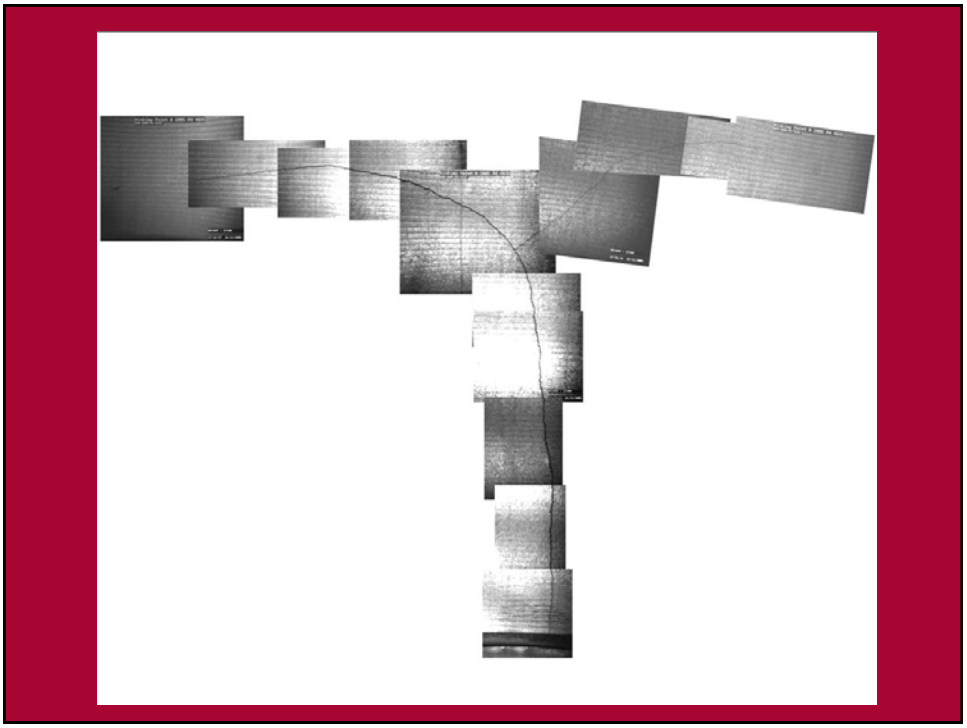
Regulatory Challenges and Strategy

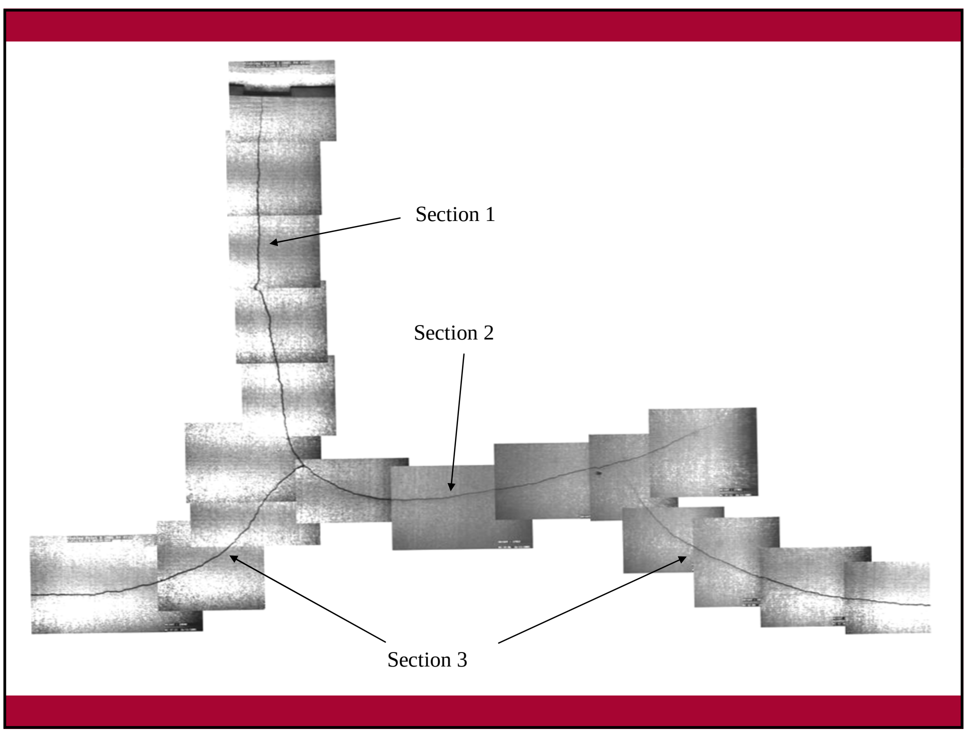
Regulatory Challenges

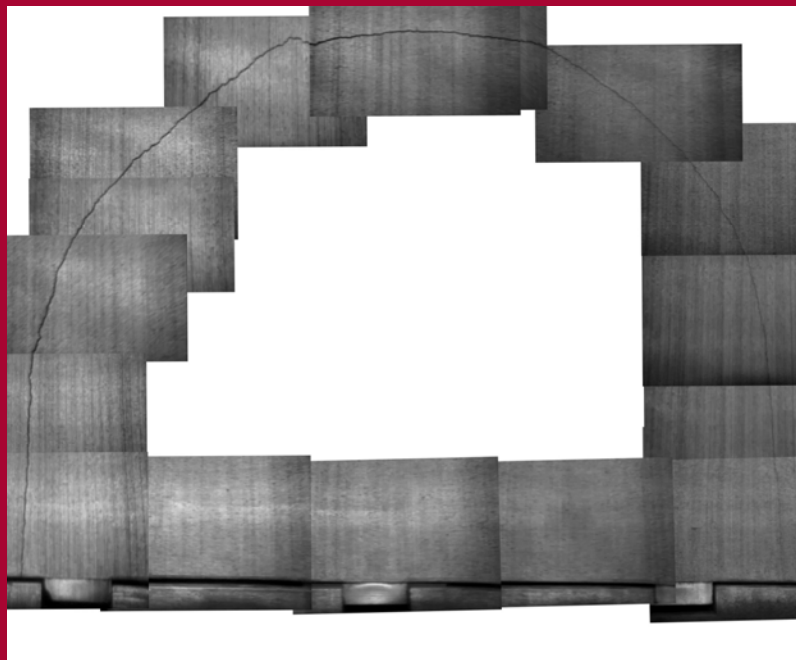
- Graphite core safety functions:
  - Shutdown and reactivity control post shutdown
  - Fuel cooling
  - Fuel integrity and unimpaired refuelling
- Challenges to core safety functions arising from:
  - Observations of unpredicted graphite moderator brick cracking in AGRs, pre stress reversal
  - Predictions of cracking in AGRs, post stress reversal
  - High radiolytic weight loss








Section 1
Section 2
Section 3



2002/04 Strategic Technology Review

HSE

- Onerous material duty, ~20dpa, >40w/o
- Established engineering practice:
  - Licensee bespoke
  - Empirical , lacks in-depth understanding
  - Limited validation
  - Need for bounding data
  - Highly complex
  - Limited user group
- Data scatter
- Significant uncertainty
- High complexity - many inputs, interactions & iterative loops

Handling Uncertainty

HSE

- Reducing Risks, Protecting People (R2P2)
- Precautionary Principle:
  - Rules out lack of scientific certainty as a reason for not taking preventive action
- Types of uncertainty:
  - Knowledge - sensitivity studies
  - Modelling – expert judgement, alternative hypotheses
  - Limited predictability or unpredictably – initial and current state

2004/05 Graphite Strategic Objectives

HSE

1. Develop revised assessment guidance
   - SAPs and TAGs revised
2. Secure maintenance and improvement of graphite core safety cases
   - Significant progress made
3. Develop credible and cost effective graphite research programme
   - Improved research programme in place
4. Provision of independent advice
   - Graphite Technical Advisory Committee (GTAC) established 2004 – contract in place to 2011

Health and Safety
Executive

Regulatory Guidance

Regulatory Guidance

- Safety Assessment Principles for Nuclear Facilities, 2006, SAPs
  - "Safety cases for graphite components and structures are expected to present a multi-legged approach, based upon independent and diverse arguments. The rigour of application and robustness of the supporting data and information should be based upon the safety classification of the graphite components and structures. The multi-legged arguments, when taken together with the various elements of established engineering practice, should provide defence in depth."

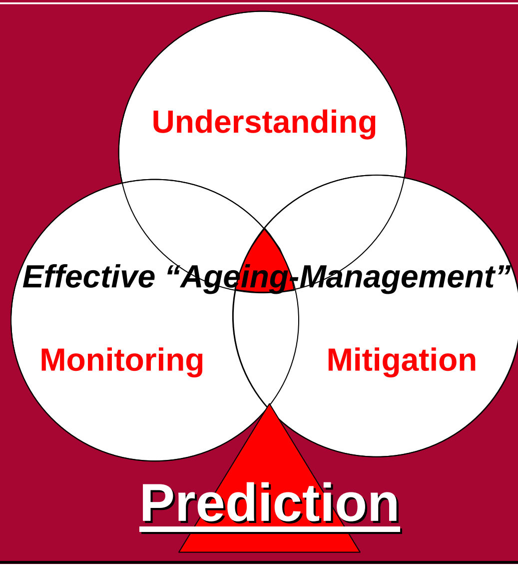
Regulatory Guidance continued

- The safety case should develop multi-legged arguments based upon the following:
  - a) design;
  - b) manufacture, construction and commissioning;
  - c) component and structure condition assessment (CSCA);
  - d) defect tolerance assessment;
  - e) analysis of radiological consequences of defectiveness;
  - f) monitoring; and
  - g) examination, inspection, surveillance, sampling and testing.
- Technical Assessment Guide – “Graphite Reactor Cores”

Health and Safety Executive

Safety Case Improvements

IAEA "Ageing-Management" Programme



Safety Case Improvements

Understanding

- Improved understanding of materials and component behaviour, further work progressing

Monitoring

- More frequent and extensive inspection, surveillance & sampling
- Enhanced on-line monitoring & trending of core behaviour

Mitigation

- ALARP plant modifications & methods of operation implemented, further work progressing
- Safe limiting conditions established, further work progressing

Prediction

- Progress with validation of predictive methods, further work progressing
- Extensive sensitivity studies and probabilistic analyses undertaken and compared with inspection data, further work progressing
- Progress obtaining bounding materials data, further work in-hand

Health and Safety
Executive

Graphite Nuclear Safety Research

Nuclear Research Arrangements

- HSE coordinated nuclear safety research programme. Consists of:
  - Licensee research
  - HSE Levy programme
- Nuclear Safety Studies (NSS)
  - Direct support to safety case assessment
- Process for HSE coordinated research
  - NII writes Nuclear Research Index (NRI) annually
  - Licensee undertakes research in response to NRI and updates this annually

Graphite Nuclear Research Strategy

- Improve fundamental understanding of ageing processes and materials behaviour
- Improve understanding of cracking mechanisms
- Obtain materials data that bounds future operating conditions in relevant environments
- Address data scatter by improved modelling
- Establish limiting component, channel, and core condition or limit of tolerability and associated safety margins
- Validate models
- Improve inspection and monitoring techniques
- Maintenance and development of independent expert advice
- Improve international collaboration on nuclear graphite research

HSE coordinated research Summary of current work undertaken by licensees & NII

- Graphite core material properties
  - MTR experiments investigating:
    - Effect of high radiolytic weight loss
    - Irradiation creep
    - Standardisation of testing procedures
    - Microstructural characterisation
  - Modelling of graphite structure
  - Development of new measurement techniques eg Poisson's ratio
  - Improved dose/damage relationships for irradiated material properties trends

HSE coordinated research
Summary of current work undertaken by licensees & NII

- Core component integrity
  - Fracture behaviour under biaxial loading
  - Fracture modelling
  - Improved spatial and temporal predictive modelling, incorporating:
    - More representative stress analyses
    - Improved probabilistic modelling
    - Prediction and validation of:
      - Brick shape changes
      - Brick cracking rates and morphologies
      - Material properties

HSE coordinated research
Summary of current work undertaken by licensees & NII

- Whole core structural response
  - Development of improved static and dynamic whole core models to assess realistic core driving forces
  - Validation of whole core models
  - Seismic model validation using “shaker” table tests
- Inspection, sampling and monitoring
  - Model based condition monitoring
  - Integrated core inspection in CO_2 - brick dimensions, remote visual, brick internal stress, volumetric inspection
  - Techniques for measurement of core distortion during operation

HSE coordinated research
Summary of current work undertaken by licensees & NII

- Regulator access to independent advice
  - Microstructural studies on irradiated graphite
  - Modelling dose/damage development and understanding of irradiated material properties trends
  - AGR Brick Cracking Network – understanding and prediction of cracking
    - Materials resistance parameters
      - Crack initiation, size effects etc
      - Fracture behaviour under biaxial loading
      - Effect of notches on fracture initiation
    - Driving force parameters
      - Stress analysis sensitivity studies
      - Statistical modelling of brick shape changes
      - Convergence of stresses and brick shape using Bayesian framework
  - Whole core modelling using super-elements
  - ASME graphite code development
  - IAEA international graphite database
  - Graphite Technical Advisory Committee (GTAC)

Summary

- Reviewed regulatory challenges on nuclear graphite technology
- Outlined research and regulatory strategy to address safety issues
- Approach to address current challenges faced by UK on graphite creates opportunity for closer international collaboration to achieve benefits for current and future gas-cooled reactors

South African Regulatory Perspective on Nuclear Graphite Qualification and Manufacturing

March 16 -18, 2009
USNRC Workshop on Safety Related Graphite R&D needs for High Temperature Gas Cooled Reactors
Rockville, Maryland, USA

Mr. S Doms
Senior Regulatory Officer: PBMR Programme

Mr. P Bester
Manager: PBMR Programme

1

OUTLINE

1. Introduction
2. Regulatory Framework
3. Graphite Requirements
4. Current Research Work Conducted by the NNR
5. Conclusions

2

1. Introduction

- In terms of the South African legislation, the NNR Act (Act No 47 of 1999), any person wishing to site, construct, operate, decontaminate or decommission a nuclear installation may apply to the NNR Chief Executive Officer for a nuclear installation licence.
- In July 2000 the NNR received a Nuclear Installation Licence (NIL) application from Eskom for a PBMR Demonstration Power Plant.
- The NNR also developed a number of documents detailing the requirements and recommendations for the licensing of the PBMR.

3

2. Regulatory Framework

(NNR Licensing Documents - 1)

The scope of regulatory assessment for licensing of the PBMR is based on the licensing requirements and safety criteria defined by the NNR in a number of regulatory documents.

#	Title
RD-0018	Basic Licensing requirements for PBMR
RD-0019	Requirements for the Core Design of the PBMR
RD-0024	Requirements on Risk Assessment and Compliance with Principal Safety Criteria for Nuclear Installations
RD-0034	Quality and Safety Management Requirements for the Nuclear Installations
LD-1096	Fuel qualification requirements for PBMR

4

2. Regulatory Framework

(NNR Licensing Documents - 2)

#	Title
LD-1097	Qualification Requirements for the Core Structure Ceramics of the PBMR
RD-0014	Emergency Preparedness and response requirements for nuclear installations
RD-0016	Requirements for licensing submissions involving computer software and evaluation models for safety calculations
RD-0026	Decommissioning of Nuclear Facilities
LG-1045	Guidance for licensing submissions involving computer software and evaluation models for safety calculations

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2. Regulatory Framework

(Dose and Risk Criteria)

Initiating Event Frequency	Limits
Category A	**Plant Personnel:** • annual individual accumulated design dose limit of 20 mSv **Members of the Public (critical group):** • annual individual design dose limit of 250 μSv (per site)
Category B	**Plant Personnel (outside of exclusion areas):** • 50 mSv individual design dose limit for the total accumulated exposure after one single event **Members of the Public (critical group):** • 50 mSv individual design dose limit for the total accumulated exposure after one single event
Category C	Limitation of risk to the values set by the risk criteria: **Plant Personnel:** • 5×10^{-5} y^{-1} peak individual risk due to all nuclear installations, and • 10^{-5} y^{-1} average risk due to all nuclear installations **Members of the Public:** • 5×10^{-6} y^{-1} peak individual risk due to all nuclear installations, and • 10^{-8} y^{-1} average risk per site

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2. Regulatory Framework

(Licensing Approach)

A multi-staged licensing process has been adopted by the NNR, which includes the following major licensing stages:

- Stage 1: Acceptance of Concept Safety Case
- Stage 2: Site preparation, construction and manufacturing phase
- Stage 3: Fuel on Site, Fuel Loading, Testing and Commissioning
- Stage 4: Plant operation
- Stage 5: Decommissioning

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2. Regulatory Framework

(RD-0034: Quality and Safety Management Requirements for Nuclear Installations)

- This RD details the requirements of the NNR for quality and safety management systems for licensees, designers and suppliers of nuclear installations in South Africa.
- All parties and organisations that are in any way involved in activities important to nuclear safety of a nuclear installation must comply with the applicable requirements of this document.
- This RD defines the principles for an Integrated Management System Approach, General Requirements on Organisation and Documentation, Management Responsibility, Resource Management, Process Realisation, Measurement, Analysis and Improvement and Safety Culture.

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2. Regulatory Framework

(Supplier Qualification Process)

- **The regulatory process for the PBMR requires that the applicant must ensure compliance with several requirements on Quality and Safety Management before design, manufacturing, testing and commissioning of safety important components can be initiated .**
- **To this effect the NNR performs joint monitoring activities with the applicant and its designer and is involved as part of its assessment process in the Qualification of PBMR (Pty) Ltd as designer, the Qualification of the applicant (Eskom Client Office) and the Qualification of PBMR suppliers**

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2. Regulatory Framework

(Manufacturing of Long Lead Items)

- **The procurement process requires that interventions are identified by the applicant, designer, independent inspector (if the code or standard requires the involvement of an independent inspector) and the NNR.**
- **The NNR oversight activities are to ensure that the characteristics of the product being produced are consistent with the material and design specifications.**

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2. Regulatory Framework

(Third Party Independent Inspections)

- **No internationally recognized standards for graphite such as ASME III for metallic components with inherent QA measures and independent third party inspection exist for Graphite.**
- **The NNR therefore requires that PBMR implements a framework where independent third party inspection is ensured with the necessary certification processes.**

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3. Graphite Requirements: LD-1097

- **The NNR developed a Requirement Document, LD-1097: "Qualification Requirements for the Core Structure Ceramics of The Pebble Bed Modular Reactor".**
- **This LD stipulates the requirements for the qualification of the CSC materials and structures, and the quality control related to the manufacturing processes of the CSC components.**
- **It also covers the requirements and recommendations for surveillance of the CSC from the construction stage up to the decommissioning of the plant.**
- **The objective of a CSC-QP is to provide confidence in the qualification of the CSC and to ensure that scientifically sound standards and specifications will be applied.**

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3. Graphite Requirements: LD-1097

(Selection of material and definition of required properties)

The LD requires that specific safety functions of the CSC be defined to accommodate the functions and characteristics of the CSC. The specific safety functions of the CSC are related to the FSF, which are Heat removal, Reactivity control and Radioactivity confinement.

This section also addresses the following aspects:

- **Safety Classes and Quality Classes**
- **Past Experience**
- **Specification of Basic Material**
- **Material Data Sheets**

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3. Graphite Requirements: LD-1097

(Manufacturing Processes and Quality Assurance Preconditions for Manufacturing)

The LD states that as a precondition for manufacturing, the manufacturers of the CSC basic materials must be capable of meeting the principal QA requirements.

In addition to the principal QA requirements, the following specific requirements must be considered:

- **Geometrical Control**
- **Surface inspections**
- **Clean Conditions during handling and manufacturing**
- **Marking of the components**
- **Treatment of deficiencies**

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3. Graphite Requirements: LD-1097

(Qualification of material and irradiation testing)

- A Test Programme consisting of physical, mechanical and Irradiation testing must be carried out that addresses the CSC design criteria.
- The Requirements for Basic Material Testing are for Statistics; un-irradiated Material; Material Data Correlation; Material Utilisation; Irradiated Material; In-Situ-Tests and Surveillance During Operation.
- If credit is taken from previous qualification programs, the tests must also demonstrate compliance with the properties and characteristics found in the past.

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3. Graphite Requirements: LD-1097

(Qualification of structures and assembly)

This LD stipulates the following requirements for the Qualification of structures and assembly:

- Requirements for Design
- Positioning and Sealing
- Definition of Loads
- Definition of the Load Cases and the Stress Categories
- Structural Analysis
- Stress analysis of un-irradiated CSC
- Stress Analysis of irradiated CSC
- Fatigue Analysis and Lifetime Assessment

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4. Current Research Work by the NNR

(Development of a Graphite Material Model)

The NNR is developing its own model on the irradiation behaviour of graphite in order for the claims made by the PBMR CSC designers to be independently assessed.

- **Task 1 involves the development of a graphite material model, which simulates the changing graphite material properties due to fast neutron irradiation over the required fluence and temperature ranges.**
- **Task 2 will use the GMM and appropriate finite element models of the individual components of the side and central reflectors to determine the component distortions and the internal stresses that may arise over life.**
- **Task 3 work will involve the generation of a model of the CSC over the height of the active core using solid model representations of individual components.**

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4. Current Research Work by the NNR

(Position Paper on Graphite Waste)

- **Waste treatment, storage and disposal under strict consideration of shielding aspects, protection of workers and environmental impact in terms of the South African context are defined in the "Radioactive Waste Management Policy and Strategy for the Republic of South Africa 2005" published by the Department of Minerals and Energy of South Africa.**
- **The NNR is in the process of developing a position paper on graphite waste taking the current international status of graphite waste management into account. The first stage of the process is to gather all available information and compile an overview report on international requirements, approaches, and positions on minimization, management and disposal of graphite waste.**

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6. Conclusions

- Not only the qualification of CSC material, but also the manufacturing process and associated assurance process has an impact on safety of the Pebble Bed Modular Reactor. The NNR regulatory framework considers all these aspects in detail.
- The NNR is developing independent measures and models to confirm the claims within the safety case.
- The NNR is confident that adequate measures have been implemented towards addressing these issues mentioned and that a rigorous process has been developed that will ensure the safety of the PBMR DPP.

- The End -

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WORKSHOP ON NUCLEAR GRAPHITE RESEARCH

Organized by ORNL and Sponsored by NRC

Legacy Hotel and Meeting Center, Rockville, MD, March16-18, 2009

JAPANESE REGULATORY PERSPECTIVE

Motokuni Eto, Technical Consultant, Toyo Tanso Co.
Taiju SHIBATA, High Temperature Fuel & Material Group, JAEA

2009年8月17日

東洋炭素株式会社

HTTR Graphite Components Design & Manufacturing Procedure

(1) Database

(2) Design

(3) Manufacturing
Construction

(4) In-service

Property data including irradiation effect

Proof test

Structural design code

Inspection standards
QA/QC management

No crack propagation or unexpected oxidation supposed

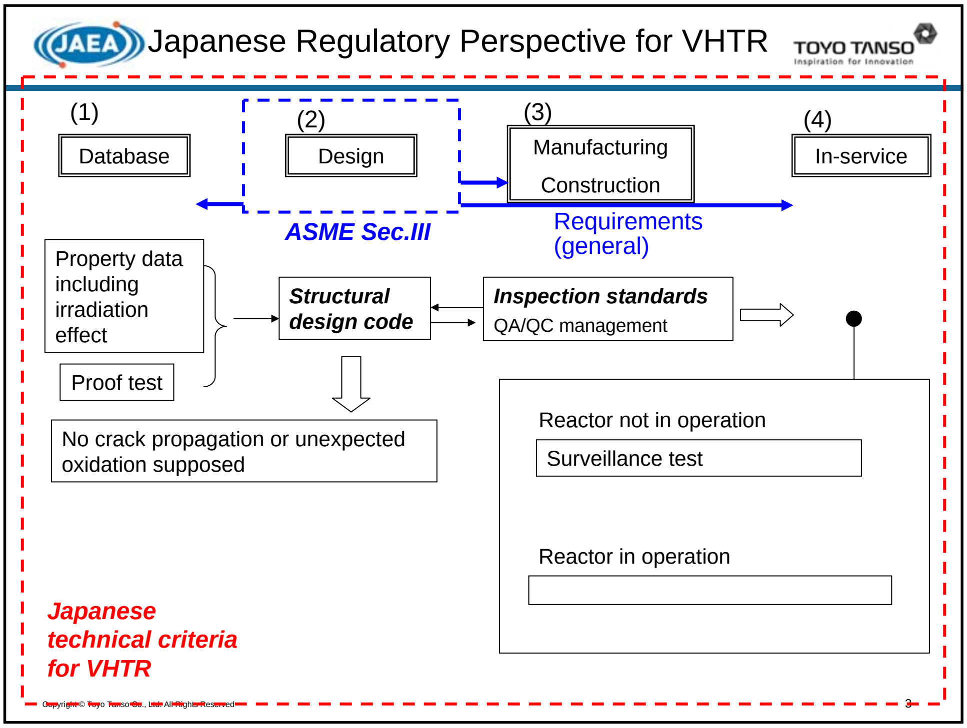
Reactor not in operation

Surveillance test
inspection by TV camera

Reactor in operation

Monitoring
- regional temperature distribution
- regional FP release amount

2


JAEA
Japanese Regulatory Perspective for VHTR
TOYO TANSO
Inspiration for Innovation
(1)
Database
(2)
Design
(3)
Manufacturing
Construction
(4)
In-service
ASME Sec.III
Requirements
(general)
Property data
including
irradiation
effect
Proof test
Structural
design code
Inspection standards
QA/QC management
No crack propagation or unexpected
oxidation supposed
Reactor not in operation
Surveillance test
Reactor in operation
Japanese
technical criteria
for VHTR
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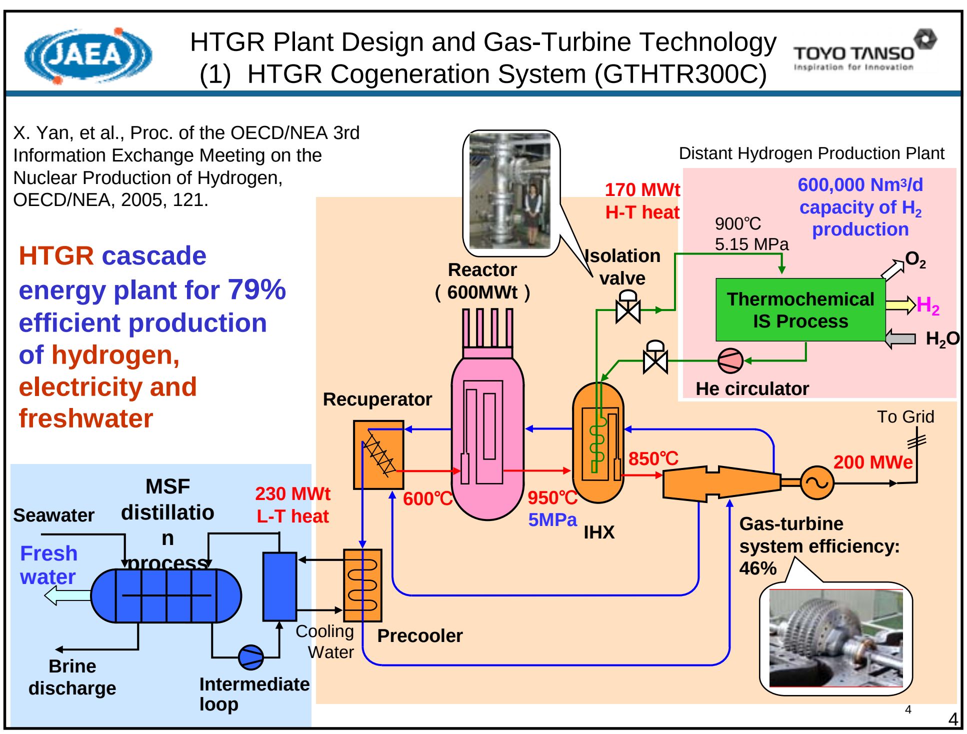
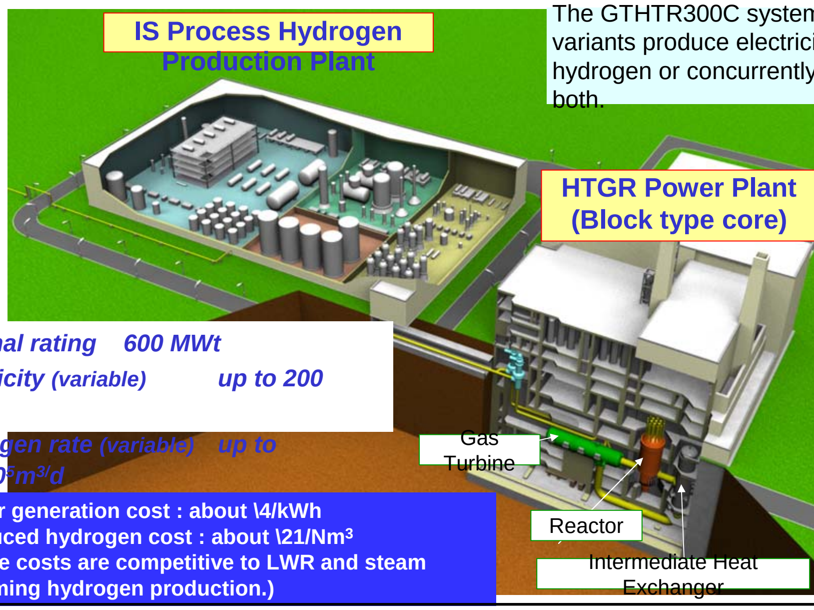

JAEA
HTGR Plant Design and Gas-Turbine Technology
(1) HTGR Cogeneration System (GTHTR300C)
TOYO TANSO
Inspiration for Innovation
X. Yan, et al., Proc. of the OECD/NEA 3rd
Information Exchange Meeting on the
Nuclear Production of Hydrogen,
OECD/NEA, 2005, 121.
HTGR cascade
energy plant for 79%
efficient production
of hydrogen,
electricity and
freshwater
Distant Hydrogen Production Plant
600,000 Nm³/d
capacity of H_2
production
170 MWt
H-T heat
900℃
5.15 MPa
Isolation
valve
Reactor
(600MWt)
Thermochemical
IS Process
O_2
H_2
H_2O
He circulator
Recuperator
To Grid
850℃
200 MWe
600℃
950℃
5MPa
IHX
MSF
distillation
process
Seawater
230 MWt
L-T heat
Fresh
water
Gas-turbine
system efficiency:
46%
Cooling
Water
Precooler
Brine
discharge
Intermediate
loop
4
4

(2) HTGR Plant Design
- Concept of HTGR Cogeneration System (GTHTR300C) -



The GTHTR300C system variants produce electricity, hydrogen or concurrently both.

Thermal rating 600 MWt
Electricity (variable) up to 200 MWe
Hydrogen rate (variable) up to 6.4X10^{5}m^{3}/d

Power generation cost : about \4/kWh
Produced hydrogen cost : about \21/Nm^{3}
(These costs are competitive to LWR and steam reforming hydrogen production.)

5

Future Plan – R&D items -

Medium-range plan in JAEA → To be proposed

		2005	2010	2015	2020
		VHTR(GTHTR300C) Conceptual design	Detailed design	Evaluation	
Reactor	HTTR test	Performance tests Safety tests	Reactor-IS simulation		
	Fuel	Irradiation test on burnup, Manufacture of ZrC,	High-burnup SiC Irradiation of ZrC		
	Material	Graphite test C/C component test	Irradiation tests		
Heat Utilization	Components	Compressor (Gas turbine) IHX			
	Hydrogen production	Data base system	Pilot plant test	HTTR-IS	

6

6

Long High-temp.(50days/950°C) Operation of HTTR

TOYO TANSO
Inspiration for Innovation

Operation and Tests of HTTR

Core Physics | He Chemistry | Fuel & Materials Integrity | Operation & Maintenance Technique | Inherent Safety

Obtain Data for HTGR Development

Core Physics ： Excess reactivity and other core physics parameters change by fuel burn-up

He Chemistry : Impurities behavior at high temperature under radiation environment
Performance of primary He purification system

Fuel and Materials Integrity ： FP release, High temp. component (IHX), In-core components **(Graphite core support and reflectors)**

Operation and Maintenance Technique ： Control of primary coolant (Leak, Purification,・・・)

Structural Design Code for HTGR

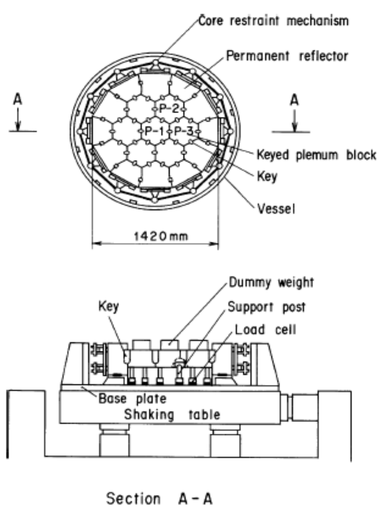
TOYO TANSO
Inspiration for Innovation

- Experiences and knowledge database obtained in the design, construction and operation of HTTR are to be utilized effectively for the larger commercial HTGR.
- For this purpose the detailed evaluation of existing data as well as the new data obtained from the ongoing and future experiments are necessary.

(1) Property Data and Proof Test

- ✓Property data
  - For IG-110, PGX and ASR-0RB
  - Including irradiation and oxidation effects
- ✓Proof tests
  - Bottom structure seismic test
  - Core components seismic test
  - Support post bucking test
  - Dowell/ socket fracture test
  - Key/ keyway fracture test

(1) Property Data and Proof Test

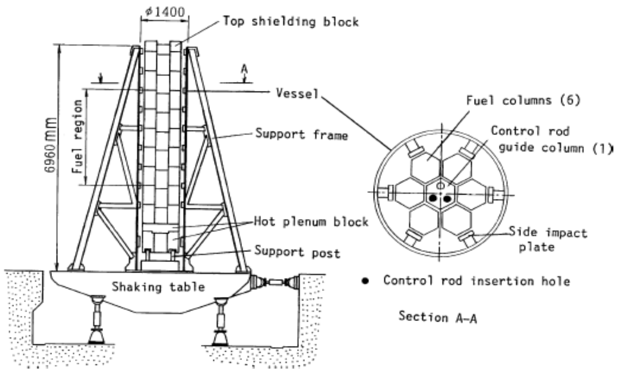


Core bottom structure seismic test apparatus

Integrity of the structure was confirmed.

Validity of technical criteria should be confirmed by proof tests.

(1) Property Data and Proof Test



Core component array seismic test apparatus

Integrity of the structure was confirmed. Seismic analysis code SONATINA-2V was developed. Propose to describe in the ASME code as a specific example.

Japanese seismic standard was revised in 2006. Check for seismic waves following the new standard is underway.

(2) Graphite Structural Design Code

Graphite Structural Design Code

Deterministic approach

1) Classification of graphite components
2) Failure theory

Maximum principal stress + Modified bi-axial stress limit

3) Stress classification

Limit for primary and secondary stress

4) Stress limit
5) Buckling limit
6) Stress analysis

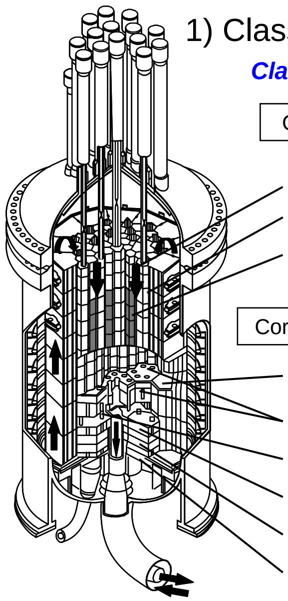
Viscoelastic analysis by proven VIENUS code

7) Specified minimum ultimate strength, Su
8) Oxidation effect
9) Quality control determined by Inspection Standard for Graphite
10) A set of design data

IG-110, PGX and ASR-0RB
including irradiation and oxidation effects

(2) Graphite Structural Design Code

TOYO TANSO Inspiration for Innovation



1) Classification of components

Classified by structural functions (safety viewpoint)

Core graphite components

Replaceability: routine

Irradiation effect: considered

- Replaceable reflector block (IG-110)
- Control rod guide block (IG-110)
- Fuel block (IG-110)

JAEA original design criteria

Core support graphite components

Serious safety function to keep core structure for reactor shutdown

Replaceability: difficult

- Permanent reflector block (PGX)
- Hot plenum block (PGX)
- Support post (IG-110)
- Lower plenum block (PGX)
- Carbon block (ASR-0RB)
- Bottom block (PGX)

Design criteria was established on the basis of concept of former ASME draft

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(2) Graphite Structural Design Code

TOYO TANSO Inspiration for Innovation

4) Stress limit

Operation condition	Primary + secondary stresses: Membrane	Primary + secondary stresses: Menbrane + bending or Point	Peak stress: Peak	Peak stress: Fatigue(*)
I & II	0.25Su	0.33Su	0.9Su	1/3
	0.33Su	0.5Su	0.9Su	1/3
III	0.5Su	0.67Su	0.9Su	2/3
	0.5Su	0.75Su	0.9Su	2/3
IV	0.6Su	0.8Su	1.0Su	3/3
	0.7Su	0.9Su	1.0Su	3/3

(*) Allowable fatigue life usage fraction
Upper line: core support conponents
Lower line: core components

Core support components have more severe limits than *core components* considering safety function.

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(2) Graphite Structural Design Code

The first loaded IG-110 graphite in the HTTR

Table 2.4 Tensile and compressive data of HTTR first loaded IG-110 graphite.

Tensile strength (MPa)	Average	Standard deviation	Number of specimens	S_u value
HTTR first loaded data	29.6	1.49	640	26.1
Design data	25.3	2.43	362	19.4
Compressive strength (MPa)	Average	Standard deviation	Number of specimens	S_u value
HTTR first loaded data	82.6	2.36	320	76.9
Design data	76.9	6.41	373	61.4

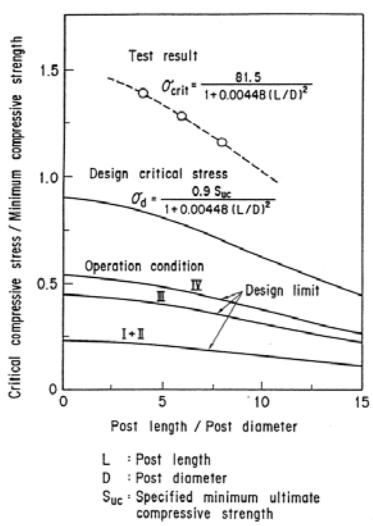
Su values for tensile and compressive strength were decided Survival probability of 99% at confidence level of 95%(JAEA-Technology 2006-048)

The first loaded IG-110 has much higher strength than design data.

It is possible to increase the Su values for proven IG-110 graphite. It gives lifetime extension of components.

(2) Graphite Structural Design Code

5) Buckling limit of core support post



Rankine-Gordon type stress limit criteria was completed by test results

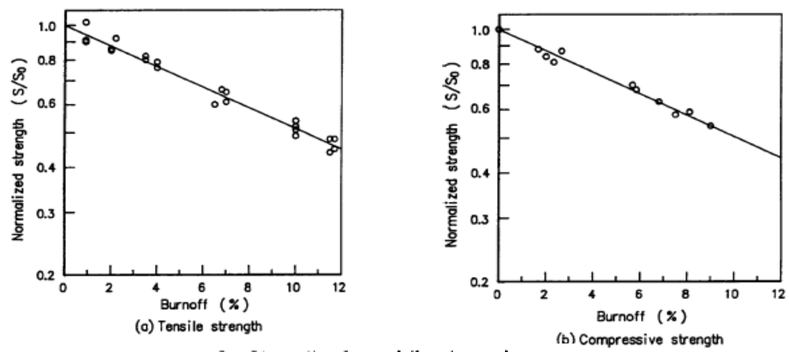
Specific example is not given in ASME code.

Propose to describe in the ASME code.

(2) Graphite Structural Design Code

TOYO TANSO
Inspiration for Innovation

8) Oxidation effect



S : Strength of unoxidized specimen
S_0: Strength of oxidized specimen

The oxidation-induced property change should be considered for safety analysis.

The strength is decreased by oxidation with variation of data.

It is important to decide reasonable criteria.

(2) Graphite Structural Design Code

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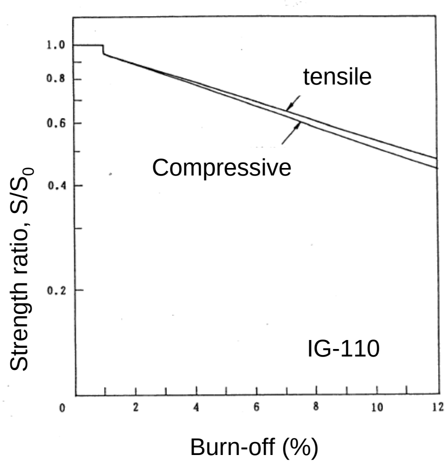
Inspiration for Innovation

8) Oxidation effect



JAEA criteria for oxidation

- ✓ Region oxidized > 80%:
  Geometry reduction
  (regarded as burn away)
- ✓ Strength reduction till 12% is evaluated following the figure

At low oxidation condition, its damage on material properties is negligible for safety analysis.

Propose to describe in the ASME code.

(3) Inspection Standard for Graphite

Manufacturing process for components

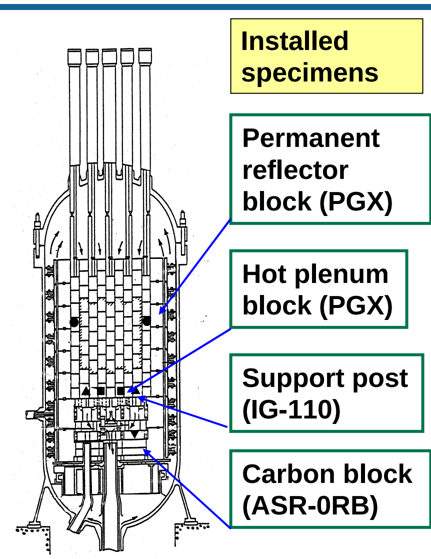
- (1) Material inspection
  - 1) Graphite grade
  - 2) Impurities
  - 3) Mechanical strength
  - 4) Dimensional stability at high temperatures
    (only for carbon material)
- (2) Dimension inspection
- (3) Visual inspection
- (4) Non-destructive test

(Compulsory inspection for as-fabricated graphite)

In-Service Inspection (ISI) of Graphite Components

1. TV camera monitoring

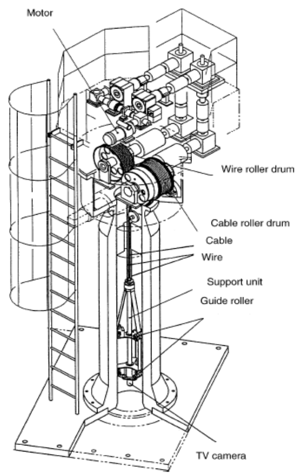
2. Surveillance test
   - Dimensional change
   - Bending strength
   - Compressive strength
   - Surface oxidation rate
   - Young's modulus



HTTR graphite blocks can be measured during refueling period.

(4) In-Service Inspection (ISI) of Graphite Components

(1) Visual inspection by TV camera



Object items: Inclination of support post, Gaps between blocks, etc.

Visual inspection will be carried out at refueling time.

(4) In-Service Inspection (ISI) of Graphite Components

(2) Surveillance test

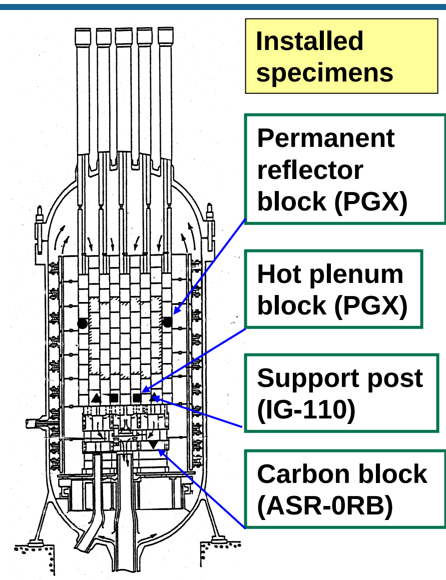
Dimensional change

Bending strength

Compressive strength

Surface oxidation rate

Young's modulus



Preparation of Design Standard for Commercial HTGR Graphite Components

A special committee at AESJ (Atomic Energy Society of Japan) discusses to establish technical criteria for VHTR graphite components.

Schedule: 2008.4 ~ 2009.3

Outline:

- Based on HTTR standards
- Including database, design, materials and in-service requirements
- Extrapolation of irradiation data
  - Dimensional change
  - Young's modulus
  - CTE
  - Thermal conductivity
  - Strength
  - Creep parameters
- Evaluation by fracture mechanics
- Accept probabilistic approach

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Design & In-service Inspection Standards for Commercial HTGR Graphite Components

- Special Committee Report on HTGR Design and In-service Inspection Standards (Draft) is to be published by the end of March 2009.

  The draft standards will be hopefully translated into English in the near future for its global usage for the design of HTGR graphite components.

- Research Report on the Analyzing Methods (Extrapolation) for the Existing Irradiation Data is to be published soon. (in press)

  By analyzing the existing irradiation data on wide varieties of graphites, tentative design curves were obtained for the following properties, particularly of IG-110.

  - Dimensional change, - Young's modulus, CTE, Thermal conductivity, Strength, Irradiation creep parameters

  It is expected that these curves are to be demonstrated by the future irradiation data, especially for irradiation creep.

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Table 2
Comparison between Japan's design criteria and others

Items	Japan	US	Germany
Core support component			
Failure theory	Maximum principal stress + modified Coulomb–Mohr theory	Maximum principal stress theory	Total strain energy theory
Buckling limits	Rankin–Gorden type	Karman type	Not specified
Pure shear stress limits	Considered	Not considered	Not specified
Oxidation effect	Considered	Not specified	Considered
Quality control	Specified	Not completed	Specified
Stress evaluation method	Same for both		Weibull theory
Stress category			Not required
Safety factor	Considered	Considered	Considered
Minimum ultimate strength	Same for both		Considered
Core component	Fundamental concept is the same as the core support component with some exceptions (safety factor, irradiation effects . . . etc.)	Not specified	Not specified

NRC Regulatory Research Perspectives Related to NGNP V/HTGR Licensing

Stuart D. Rubin
Senior Technical Advisor for Advanced Reactors
Office of Nuclear Regulatory Research
March 16, 2009

1

2005 Energy Policy Act: Congress Required an NGNP Licensing Strategy

- The ways in which current NRC LWR licensing requirements will need to be adapted for the types of reactors considered for the project
- The analytical tools that the NRC will need to independently verify the NGNP design and its safety performance
- Other research or development activities that the NRC will need to review an NGNP license application

2

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NGNP Safety R&D Needs Development

- NGNP Licensing Strategy Report to Congress - NRC/DOE
- NRC Advanced Reactor Research Plan - NRC
- NGNP Phenomena Identification & Ranking Tables - NRC/DOE
- HTGR Fuels Phenomena Identification & Ranking Tables - NRC
- HF Phenomena Identification & Ranking Tables - NRC
- NGNP Gap Analysis Report - ORNL

3

U.S.NRC

NGNP Licensing Strategy Report to Congress

- Risk-informed, performance-based approach to establish NGNP design-specific technical licensing requirements
- Analytical tools, models and associated data needed to address VHTR safety-relevant phenomena and perform confirmatory analysis - utilize R&D from DOE, NGNP applicant and cooperative activities to the extent possible
- Regulatory infrastructure development to include: regulatory guides, SRPs, codes and standards, reactor oversight process development and inspection programs

4

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NGNP VHTR Design and Safety Concept

- Higher operating temperatures and accident temperatures
- Graphite used for moderator, core structures and support structures
- Helium coolant is single phase and chemically inert
- High performance TRISO fuel particles in graphite matrix retain fission products
- Metallic pressure vessels thermally insulated by graphite and composites
- Inherent reactor characteristics, passive SSCs mitigate design basis accidents
- Greater emphasis on accident prevention vs. accident mitigation
- Event-specific, mechanistic source term for accident consequence analysis
- Vented low pressure confinement vs. leak-tight containment
- PRA insights used for design, safety and licensing basis decisions
- Highly automated plant controls and advanced digital I&C
- Modular fabrication and construction methods; longer operating cycles
- Deeply embedded, below grade reactor-plant structures
- Reactor protection for multiple BOP functions: electric power, process heat, H2

5

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NRC Advanced Reactor Research Plan Technical Arenas

- Fuel Performance Analysis**
- Nuclear Analysis**
- Thermal-Fluids Analysis**
- Accident Analysis**
- Consequence Analysis*
- Graphite Component Analysis**
- Metallic Component Analysis*
- Structural/Seismic Analysis*
- Risk-Informed Licensing**
- PRA**
- Human Factors**
- Advanced I&C*
- Fuel Cycle/Materials Safety*
- Material Protection
- H_2 Production Facility

* Arena contributes to HTGR Accident Analysis Evaluation Model   * Generic Arena

* Arena depends directly or indirectly on some aspect of graphite R&D

6

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NGNP R&D Infrastructure Needs Assessment

Identified:

- Key NGNP technical, safety, safety research, policy issues
- Gaps in NRC's technical information and analysis capabilities need to support the NGNP licensing review
- Experimental data, models, codes, technical knowledge and technical guidance needed for NGNP regulatory decisions

7

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Perspectives on Research and Development Needed for NGNP Licensing

NGNP vs. LWR Differences →	Involving Differences In →	Resulting in R&D Gaps →	Involving →	Technical Infrastructure Development Goals
Reactor Plant Layout Materials Structures Systems Components Fuel forms Operating modes and states Reactor operating conditions BOP functions/systems Containment design PRA use in licensing basis Source term calculation Expanded Adv digital I & C Concept of operations Manufacturing methods Outage lengths Etc.	Licensing Basis Events (LBE) LBE conditions Safety important phenomena Safety functions Safety-related SSCs Safety figures of merit SSC failure mechanisms Success criteria Maintenance and testing DID approach Risk metrics Role of the operator Man-machine interface Approach to ISI and IST Etc.	PIRTs Experimental/test facilities Experimental data Phenomena modeling Design analysis methods Qualification methods Analytical tools Accident evaluation models Operational data Commission policies Standards (e.g., materials) Aging management Inspection capabilities Technical training Etc.	Analytical tool development Code to data benchmarks Code to code benchmarks Sensitivity studies Uncertainty analyses Analyses and evaluations Policy option development NRC-sponsored research Cooperative research Collaborative research DOE laboratory research NGNP designer research Public involvement Etc.	NGNP technical requirements SSC design criteria Analytical tools Accident evaluation models Fabrication Control Docs Codes and Standards Licensing policy decisions Regulatory requirements Regulatory guidance Inspection procedures Reactor oversight process Technical data bases Technical reference docs Knowledge and Know-How Etc.

8

Illustrative NGNP PIRT Results: Number of Graphite Phenomena Effecting FOMs

Safety Figure of Merit (FOM)	Number of Phenomena
Maintain passive accident heat transfer	22
Maintain reactivity control	25
Thermal protection of metallic components	22
Radiation shielding of metallic components	11
Maintain helium core cooling flow	23
Prevent elevated mechanical loads on fuel	14
Limit fuel fission product release to He coolant	19

9

NGNP Licensing Strategy Report to Congress: Graphite Safety R&D Issues Identified

- Analytical tools and design methods for graphite performance
- Models for graphite reactions in accident analysis tools
- Models for fission product transport through, retention in, graphite
- Effects of neutron fast fluence on graphite properties
- ASME code design requirements and tools for graphite structures
- Corrosion behavior of graphite structures during air ingress
- Construction inspection techniques/procedures for graphite

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Role of NRC NGNP Regulatory Research

- Develop NRC staff knowledge, expertise, capabilities and review guidance
- Independently confirm technical basis for requirements and criteria
- Develop NRC independent analytical capabilities
- Confirm or interpret technical information involving significant uncertainty
- Validate/scope-out technical issues to justify request for follow-up resolution by the applicant

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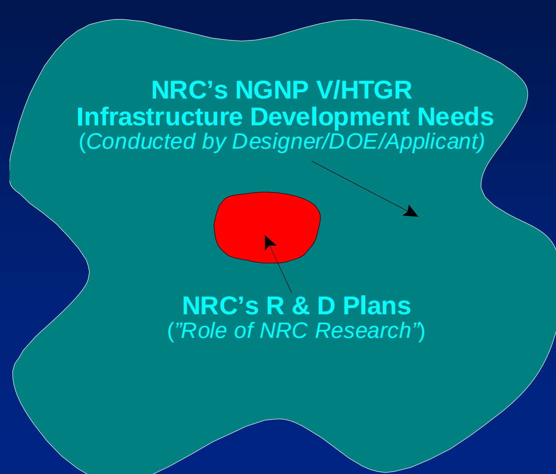
NGNP V/HTGR Infrastructure Development Needs vs. NRC Research and Development Plans



12

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NRC NGNP V/HTGR Graphite R&D Plans

- Support codes and standards development
- Conduct graphite workshop
- Participate in international irradiations
- Develop independent evaluation capability
- Develop capability to predict failure probability
- Conduct selective R&D to support regulatory decisions
- Support NRC HTGR accident evaluation model development

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NGNP Licensing Policy Issues

- Containment functional performance requirements
- Allowable dose consequences for licensing basis event categories
- Use of the PRA (e.g., select LBEs, establish special treatment requirements)
- Acceptable basis for event-specific mechanistic source term calculation, including siting source term
- Necessary DID measures

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NRC NGNP V/HTGR R& D Summary

- Focus on NGNP V/HTGR COL technical review needs
- Consistent with completed PIRTs: NGNP, TRISO Fuels and HFs
- Consistent with the “Role of NRC Research”
- Recognize/utilize DOE NGNP VHTR R&D plans/results
- Include prismatic block & pebble bed reactor designs….for now
- Focus on the NGNP-specific reactor design….after DOE selection
- Utilize cooperative research agreements where possible
- Support the NGNP COL application review schedule

Some of the Challenges in NGNP HTGR Graphite Component Safety Evaluation

ORNL/NRC Workshop on Graphite Research
Rockville, MD, U.S.A.

Dr. Makuteswara Srinivasan
Senior Materials Engineer
Office of Nuclear Regulatory Research
March 17, 2009

The contents of this presentation do not necessarily reflect any position of the NRC.

1

Outline

1. Nuclear Graphite in the NGNP Context
2. Integrating Predictive Models – Input to Regulatory Decision
   a) Graphite Degradation Model
   b) Graphite (Structural) Component Integrity Model
   c) Graphite Inspection Model
   d) Contribution to Risk (Normal, AOO, Accident)
   e) Risk Assessment Model
   f) Integration into Regulatory Decision
3. Challenges in Consensus Codes and Standards
   a) Performance Acceptance Criteria
   b) Inservice Inspection
   c) Surveillance Requirements
4. Summary

2

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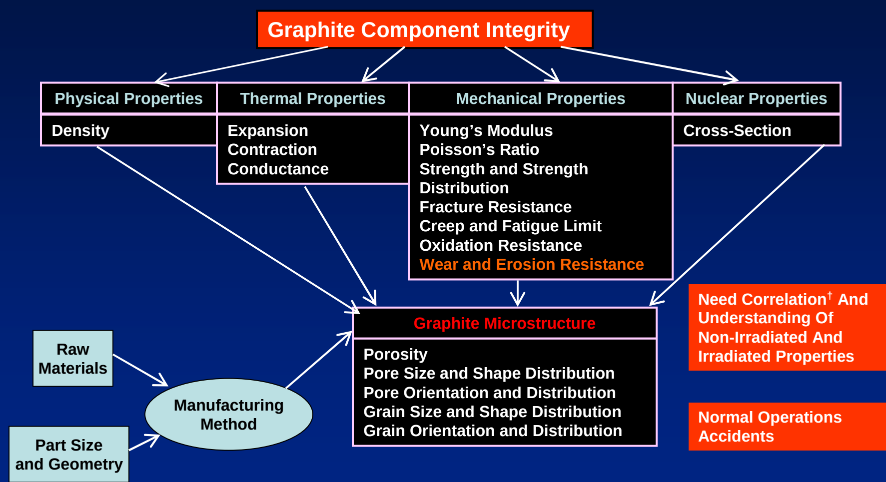
Graphite Component Integrity



† Correlation does not imply cause; PIE and analysis may shed light.

3

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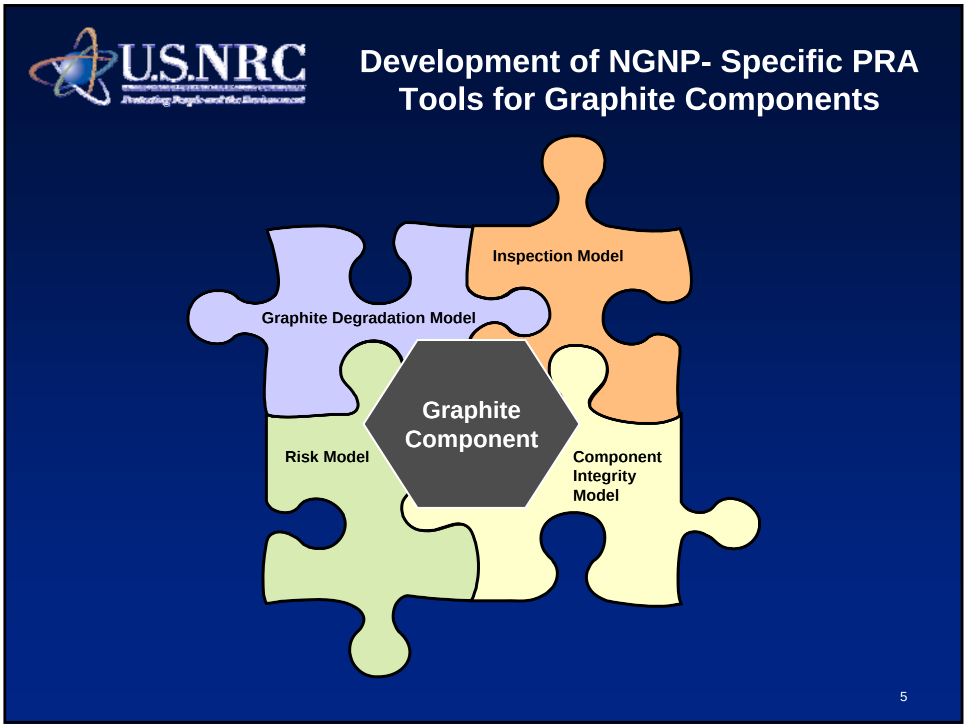
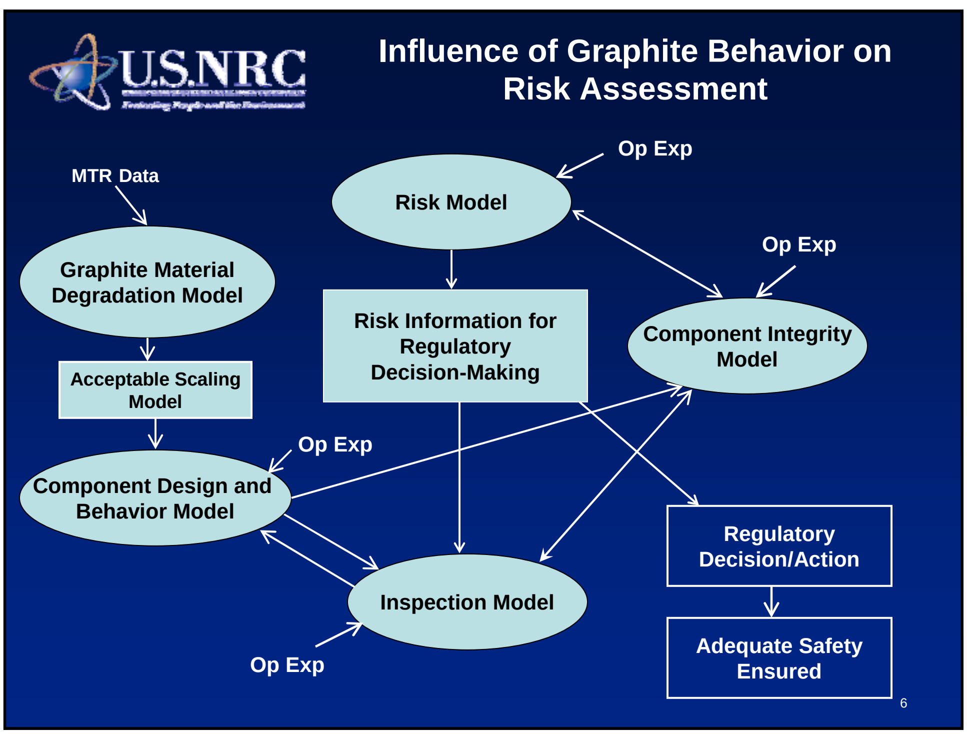
Materials-Related Challenges for NGNP HTGR Safety Evaluation

- Provide acceptable input to risk information and deterministic information in establishing the plant licensing basis
  - Involves safety margin and defense-in-depth requirements to adequately accommodate uncertainties and unknowns for NGNP plant designs which have limited operational experience, but utilize inherent characteristics and passive SSCs to reliably achieve safety functions.
- Establish an acceptable technical basis for the plant safety analysis
  - An acceptable basis from operating experience, experimental data and analysis methods for predicting the performance and behavior of reactor graphite structures and components within the HTGR pressure boundary system operating environment will need to be established.

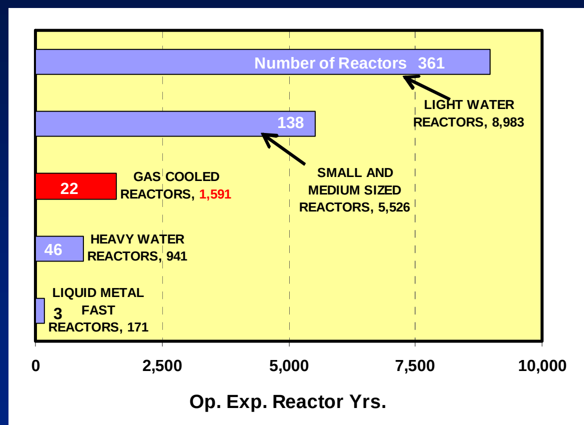
4


U.S.NRC
Development of NGNP- Specific PRA Tools for Graphite Components
Inspection Model
Graphite Degradation Model
Graphite Component
Risk Model
Component Integrity Model
5


U.S.NRC
Influence of Graphite Behavior on Risk Assessment
Op Exp
MTR Data
Risk Model
Op Exp
Graphite Material Degradation Model
Risk Information for Regulatory Decision-Making
Component Integrity Model
Acceptable Scaling Model
Op Exp
Component Design and Behavior Model
Regulatory Decision/Action
Inspection Model
Adequate Safety Ensured
Op Exp
6

World Nuclear Energy Generation

Cumulative Operating Experience of World Nuclear Power Reactors



Ref: "Global Development of Advanced Nuclear Power Plants and Related Activities", IAEA, September 2006.

7

Evaluation of the Risk-Informed PRA for Graphite Components

MTR Data

Op Exp

Material Degradation Model

Specific to Each Degradation

Graphite Aging Effects – CTE, Creep, Thermal Conductivity, Dimensions, Elastic Modulus, and Strength.

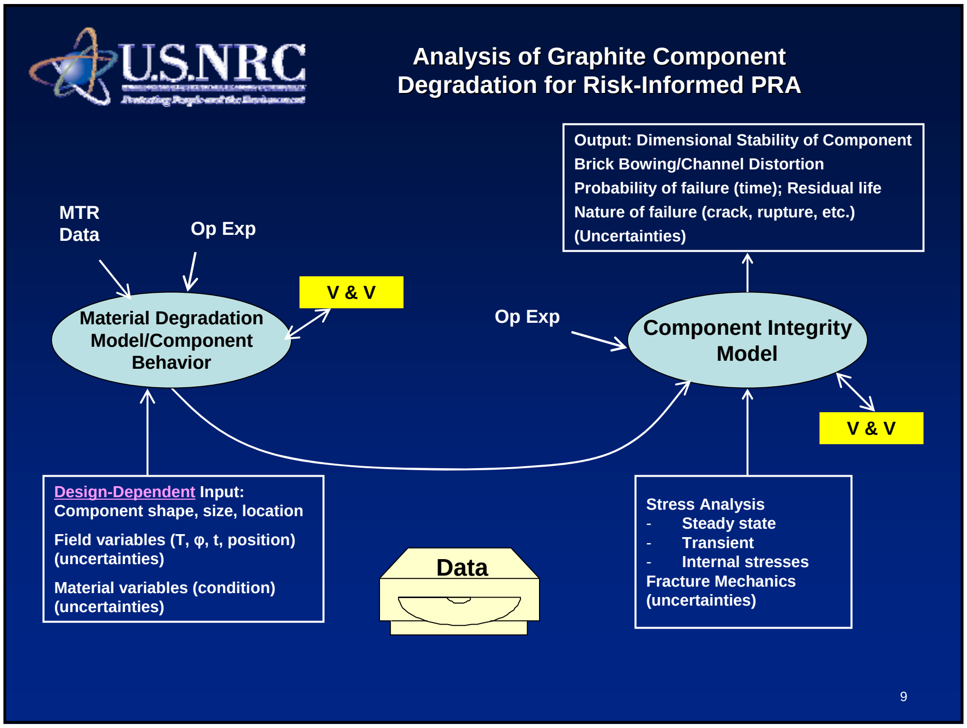
F(time (t), time at temperature t_T), fluence (φ), temperature (T), and atmosphere)

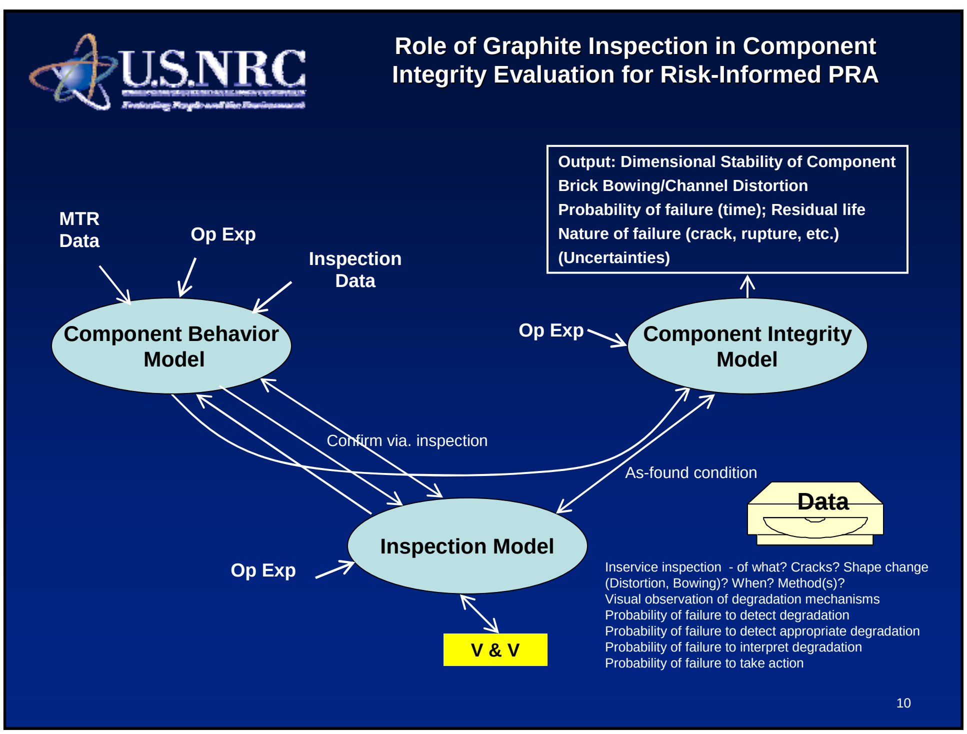
Issues: Generic degradation mechanism – however, extent of degradation may be component and environment (material, stress, temperature, atmosphere) specific

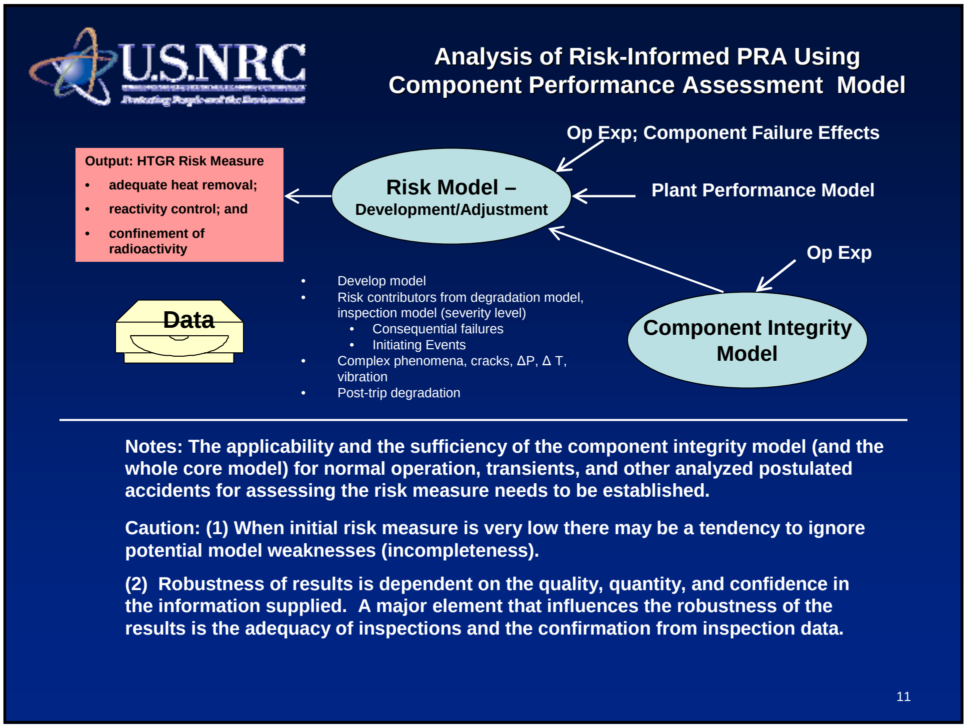
Unknowns (changes in environment)

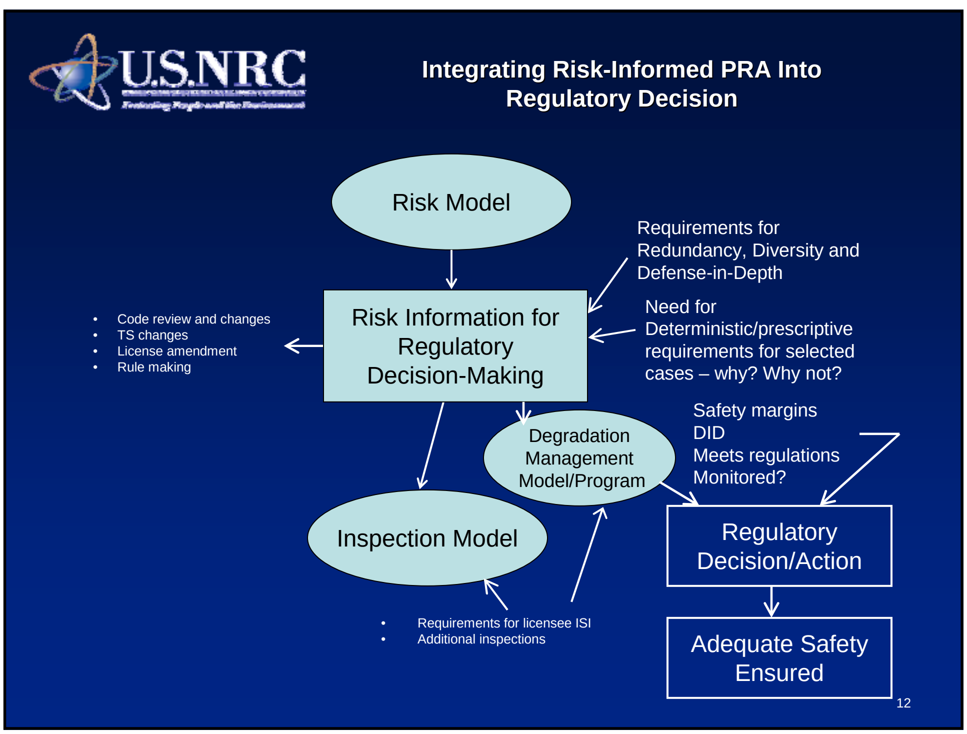
Output: Change in Property as a f(time)
Degradation rate – highly variable (includes modeled and not-modeled variables)
Considers and Quantifies Uncertainties

8

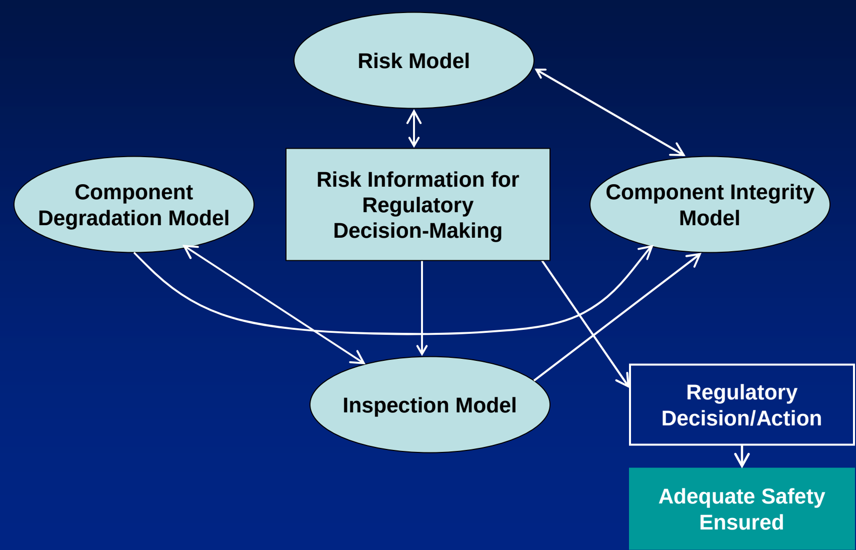

U.S.NRC
Analysis of Graphite Component Degradation for Risk-Informed PRA
Output: Dimensional Stability of Component
Brick Bowing/Channel Distortion
Probability of failure (time); Residual life
Nature of failure (crack, rupture, etc.)
(Uncertainties)
MTR Data
Op Exp
V & V
Material Degradation Model/Component Behavior
Op Exp
Component Integrity Model
V & V
Design-Dependent Input:
Component shape, size, location
Field variables (T, φ, t, position) (uncertainties)
Material variables (condition) (uncertainties)
Data
Stress Analysis
- Steady state
- Transient
- Internal stresses
Fracture Mechanics
(uncertainties)
9


U.S.NRC
Role of Graphite Inspection in Component Integrity Evaluation for Risk-Informed PRA
Output: Dimensional Stability of Component
Brick Bowing/Channel Distortion
Probability of failure (time); Residual life
Nature of failure (crack, rupture, etc.)
(Uncertainties)
MTR Data
Op Exp
Inspection Data
Component Behavior Model
Op Exp
Component Integrity Model
Confirm via. inspection
As-found condition
Data
Inspection Model
Op Exp
V & V
Inservice inspection - of what? Cracks? Shape change (Distortion, Bowing)? When? Method(s)?
Visual observation of degradation mechanisms
Probability of failure to detect degradation
Probability of failure to detect appropriate degradation
Probability of failure to interpret degradation
Probability of failure to take action
10


U.S.NRC
Analysis of Risk-Informed PRA Using
Component Performance Assessment Model
Op Exp; Component Failure Effects
Output: HTGR Risk Measure
• adequate heat removal;
• reactivity control; and
• confinement of radioactivity
Risk Model –
Development/Adjustment
Plant Performance Model
Op Exp
Data
• Develop model
• Risk contributors from degradation model, inspection model (severity level)
• Consequential failures
• Initiating Events
• Complex phenomena, cracks, ΔP, Δ T, vibration
• Post-trip degradation
Component Integrity Model
Notes: The applicability and the sufficiency of the component integrity model (and the whole core model) for normal operation, transients, and other analyzed postulated accidents for assessing the risk measure needs to be established.
Caution: (1) When initial risk measure is very low there may be a tendency to ignore potential model weaknesses (incompleteness).
(2) Robustness of results is dependent on the quality, quantity, and confidence in the information supplied. A major element that influences the robustness of the results is the adequacy of inspections and the confirmation from inspection data.
11


U.S.NRC
Integrating Risk-Informed PRA Into
Regulatory Decision
Risk Model
Requirements for Redundancy, Diversity and Defense-in-Depth
• Code review and changes
• TS changes
• License amendment
• Rule making
Risk Information for Regulatory Decision-Making
Need for Deterministic/prescriptive requirements for selected cases – why? Why not?
Degradation Management Model/Program
Safety margins
DID
Meets regulations
Monitored?
Inspection Model
Regulatory Decision/Action
• Requirements for licensee ISI
• Additional inspections
Adequate Safety Ensured
12

Evaluation of the Risk-Informed PRA for Graphite Components and Regulatory Decision/Action



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Materials-Related Challenges for NRC Staff's Safety Evaluation of NGNP HTGR

- Develop NRC staff expertise, technical tools, and data to support an effective and efficient independent safety evaluation of the NGNP HTGR graphite components.
  - Materials performance analysis codes
  - Structural and component integrity analysis codes
  - Surveillance requirements and inspection codes
  - Tools to evaluate the efficacy of component degradation management programs
- Establish HTGR graphite-specific regulatory positions, guidance documents, or standard review plans for the NRC staff to conduct an effective and efficient design safety review of HTGR graphite components.
- Establish HTGR graphite-specific NRC staff regulatory positions, guidance documents, or standard review plans on staff review of in-service inspection and surveillance plans and techniques.

14

Basic Technical Issues for NGNP HTGR Graphite Component Safety Evaluation

Issues Regarding Graphite Component Safety Requirement:

Early Operation Years: Reactor operation with a cracked graphite component is not allowed.

Mid-Life Operation Years: Reactor operation with cracked graphite component may be allowed, depending upon the safety significance of the observed cracking and assurance of negligible degradation on further reactor operation. Additional ISI may be warranted.

Component Replacement Criteria:

Needs development.

COLE-BAKER A., REED J., Measurement of AGR graphite fuel brick shrinkage and channel distortion, in Management of Ageing Processes in Graphite Reactor Cores, ed.Neighbour, G., Royal Society of Chemistry, London(2007), 201-208.

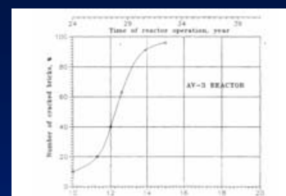

Figure 3. The cracking of reactor bricks during AV-3 operation.

P.A. PLATONOV, O.K. CHUGUNOV, V.N. MANEVSKY, V.I. KARPUKM, "Radiation damage and life-time evaluation of RBMK graphite stack", in Proceedings of a specialists meeting held in Bath, United Kingdom, 24-27 September 1995, INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA-TECDOC-901, pp:79 -89.

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Basic Technical Issues for NGNP HTGR Graphite Component Safety Evaluation

1. **Component Failure Criteria – Graded on Safety Significance**
   - a) **Probability of failure estimates, extrapolated from small population irradiation data.**
   - b) **If cracking of a graphite component is allowed (in some areas).**
     - i. **Maximum number of allowable cracks and the nature of cracking.**
     - ii. **Cracking characterization, and procedures to assess its safety significance and its effect on risk measures.**
   - c) **Cracking will not be allowed in critical areas such as fuel and control rod bricks, coolant channel areas, and core support columns.**

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Basic Technical Issues for NGNP HTGR Graphite Component Safety Evaluation

2. Component Performance Criteria
   a) Maximum allowable permanent deformation (brick bowing and channel distortion) in critical areas
3. Component Inspection Criteria
   a) Design-for-inspection of critical areas
   b) Sizing of flaws and flaw evaluation and procedures to assess their safety significance and their effect on risk measures
   c) Methods to measure and categorize component deformation, and procedures to assess its safety significance and its effect on risk measures
4. Surveillance Requirements Including Coupons, Core Sampling (Trepanning), Core Restraint Monitoring, Core Support Monitoring, and Testing Protocols
5. Acceptance/Replacement Criteria For Flawed Graphite Component In Service
6. Graphite Component Degradation Management Program and Procedure to Assess Its Efficacy

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Summary

1. Independently verified and validated predictive analytical models and codes for NGNP graphite properties are needed for the NRC staff evaluation of the design of NGNP HTGR graphite components.
2. Independently verified and validated predictive codes are needed for the NRC staff evaluation of NGNP HTGR graphite component integrity.
3. Independently verified and validated inspection codes and standards are needed for the NRC staff evaluation of NGNP HTGR graphite component degradation during service.
4. The NRC staff needs technical information on model and data uncertainties, and their overall effect (sensitivity) on failure probability predictions.
5. The NRC staff needs to communicate to risk analysts the importance of properly considering material/component degradation model and data uncertainties in risk evaluation models for risk informed regulatory decisions.

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ABBREVIATIONS

AOO	Anticipated Operational Occurrence
CTE	Coefficient of Thermal Expansion
DID	Defense-In-Depth
HTGR	High Temperature Gas Cooled Reactor
ISI	Inservice Inspection
MTR	Material Test Reactor
NGNP	Next Generation Nuclear Plant
Op E	Operating Experience
ORNL	Oak Ridge National Laboratory
PIE	Post-Irradiation Examination
PRA	Probabilistic Risk Assessment
TS	Technical Specifications
V & V	Verification and Validation

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Comparison of Graphite PIRT Results with DOE Research Plan

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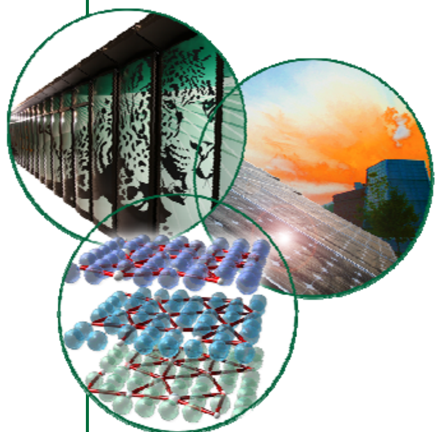


Presented at the NRC Workshop on Graphite Research

Rockville, MD

March 16-18, 2009

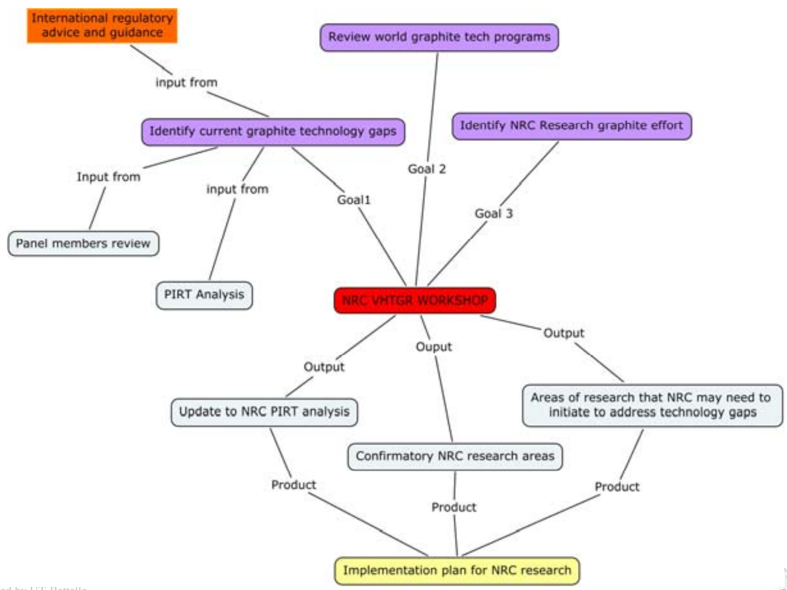
U.S. DEPARTMENT OF ENERGY

OAK RIDGE NATIONAL LABORATORY
MANAGED BY UT-BATTELLE FOR THE DEPARTMENT OF ENERGY

Overview of Presentation

- Background
- NRC Workshop & PIRT/R&D Comparison
- Phenomena Rankings Considered
- U.S. NGNP Graphite R&D Gap Analysis
- Recommended Research Areas

NRC GRAPHITE WORKSHOP LOGIC



Summary Of The Phenomena Importance and Knowledge Rankings

PIRT Rank	No. of Phenomena
I-H, K-L	5
I-H, K-M	9
I-M, K-L	2
I-M, K-M	14
I-L, K-H	0
I-L, K-M	2
I-L, K-L	1
I-H, K-H	8
I-M, K-H	1

Emphasis Was Placed On The Following Categories Of Phenomena

- I-H, K-L (5 Phenomena)
- I-H, K-M (9 Phenomena)
- I-M, K-L (2 Phenomena)
- I-M, K-M (14 Phenomena)

The phenomena were grouped into common areas were it was possible (within rankings)

The Following Documents Were Used for Source Materials Regarding US NGNP Program

- NUREG/CR-6944, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRT), Volume 5: Graphite PIRTs".
- INL/EXT-07-13165, "Graphite Technology Development Plan".
- ORNL/TM-2007/153, "NGNP Graphite Selection and Acquisition Strategy".
- ORNL-GEN4/LTR-06-019, "Experimental Plan and final Design Report for HFIR High Temperature Graphite Irradiation Capsules HTV-1 and -2".
- ORNL/TM-2008/129, "Generation IV Reactors Integrated Materials Technology Program Plan: Focus on Very High Temperature Reactor Materials".

U.S. NGNP GRAPHITE R&D GAP ANALYSIS

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Phenomena Ranked
Importance-High, Knowledge-Low

(I-H, K-L)

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Phenomena:
•Irradiation-induced creep (irradiation-induced dimensional change under stress)
•Irradiation-induced change in CTE, including the effects of creep strain
•Irradiation-induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress)

	GAPS / WEAKNESSES
Normal Operations*	•Experimental data fully bounds conditions for PMR design. •Need additional experimental data if PBR design is selected.
Anticipated operational occurrences	•Data will enable development of predictive models. •Weakness is in model development. Effort for model development needs to be augmented and accelerated
Design basis accidents	•Creep data will have to be combined with data from oxidation studies to account for effects due to accidents involving air and moisture ingress. •Weakness/gap is more related to the lack of knowledge in the area of graphite oxidation.

*Normal operations for a PMR design currently do not include recycling of graphite blocks.

Phenomena:
•Blockage of fuel element coolant channel due to graphite failure and/or graphite spalling
•Blockage of coolant channel in reactivity control block due to graphite failure and/or graphite spalling

	GAPS / WEAKNESSES
Normal Operations*	•Would the design code (ASME code) be available for core design? Need to accelerate code development. •Determining the adequacy of the design code (ASME code)
Anticipated operational occurrences	
Design basis accidents	•Understanding weakening of graphite surfaces due to long term oxidation.

*Normal operations for a PMR design currently do not include recycling of graphite blocks.

Phenomena Ranked Importance-High, Knowledge-Medium

(I-H, K-M)

Phenomena:
•Statistical variation of non-irradiated properties

	GAPS / WEAKNESSES
Normal Operations	•The current plan will provide sufficient experimental data to have a good characterization of the statistical variation of important non-irradiated properties for the selected graphite types. •There is a gap/weakness with respect to the characterization of additional graphites in order to expand the pool of qualified graphites for use in either type of reactor. However, if there are no additional graphites, presently a gap does not exist for this phenomenon.
Anticipated operational occurrences	
Design basis accidents	

Phenomena:
•Consistency in graphite quality over the lifetime of the reactor fleet (for replacement, for example)

	GAPS / WEAKNESSES
Normal Operations	•The major gap is with respect to identifying alternative coke sources to the two currently available (U.S. pet coke and Japanese pitch coke) for manufacturing HTGR graphite, and testing the validity of the ASTM standard specifications to new graphites. •Another gap is with respect to exploring the options for graphite recycling and reuse for nuclear applications.
Anticipated operational occurrences	
Design basis accidents	

Phenomena:
•Irradiation-induced dimensional change
•Irradiation-induced thermal conductivity changes and degradation of thermal conductivity

	GAPS / WEAKNESSES
Normal Operations	•Experimental data bounds conditions for PMR design for up to the expected volume change turnaround behavior with probably not enough margin in dose or temperature. •Need additional experimental data if PBR design is selected •Data will enable the development of predictive models. •Weakness is in model development, i.e., effort for model development needs to be augmented and accelerated. •Weakness is also in the validation of models to include new graphites.
Anticipated operational occurrences	
Design basis accidents	•Dimensional and thermal conductivity changes are large sources of internal stresses. Experimental data from this task will be combined with data from oxidation studies to develop whole core models in order to model behavior under accidents relating to air and moisture ingress. •Weakness/gap is related to the lack of knowledge in the area of graphite oxidation, especially on irradiated graphite.

Phenomena: *•Irradiation-induced changes in elastic constants, including the effects of creep strain*	
	GAPS / WEAKNESSES
Normal Operations	•Experimental data bounds conditions for PMR design for up to the expected volume change turnaround behavior with probably not enough margin in dose or temperature. •Need additional experimental data if PBR design is selected •Data will enable the development of predictive models. •Weakness is in model development, i.e., model development effort needs to be augmented and accelerated.
Anticipated operational occurrences	
Design basis accidents	•These data will be combined with data from oxidation studies to develop/improve predictive models that account for accidents relating to air and moisture ingress. •There is gap/weakness in the understanding and predictive model development for irradiation-induced crystal strain in the graphite block. •Weakness/gap is related to the lack of knowledge in the area of graphite oxidation, particularly on irradiated graphite.

Phenomena: *•Tribology of graphite in (impure) helium environment*	
	GAPS / WEAKNESSES
Normal Operations	•The current DOE research plan considers wear and friction only a significant problem for the PBR design. However, wear and friction have the potential of being significant issues for PMR design as well due to the erosion of relatively soft surface of graphite by the high-velocity helium flow in the coolant channels. •Need to study tribological properties of graphites considered for PMR design. •Need to study the tribological properties of graphite in helium environment, and at high pressures and temperatures. •Need to study the tribological properties over the life of reactor, i.e., as function of temperature, dose, and oxidation weight loss. •Need to study tribological properties for graphites for PBR and PMR design as a function of oxidation and weight loss.
Anticipated operational occurrences	
Design basis accidents	•Graphite dust generation is a serious issue, and therefore significant research is needed focused on characterizing the dust formed and on understanding what happens to the dust generated, as well as to develop models for fission product transport.

Phenomena: *•Blockage of Reactivity Control Channel due to graphite failure, spalling*	
	GAPS / WEAKNESSES
Normal Operations	•Need emphasis on analytical model development capable of accounting for key factors such as: oStatistical variation of inherent properties oMicrostructural differences between grades oAnisotropy due to forming and manufacture, and oGeometric factors, such as stressed volume and specimen/component dimensions oStrength changes due to oxidation, temperature, and neutron irradiation.
Anticipated operational occurrences	
Design basis accidents	•Whole Core model needs to account for changes in properties due to graphite oxidation and weight loss. However, there is a gap regarding oxidation behavior and diffusion of species within the graphite structure, both for non-irradiated and irradiated graphite.

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Phenomena Ranked Importance-Medium, Knowledge-Low

(I-M, K-L)

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Phenomena: *•Graphite dust generation*	
	GAPS / WEAKNESSES
Normal Operations	•Need to study tribological behavior of graphite in a helium environment, both under solid-solid abrasive contact conditions and under high velocity gas-solid contact conditions. •Need to study the effect of irradiation and temperature on tribological behavior. •Need to include graphites considered for PMR design and not only those considered for PBR design. •Need to study the oxidation behavior of graphite dust and whether graphite dust can be a vehicle for the transportation of fission products.
Anticipated operational occurrences	
Design basis accidents	•Need to develop models for fission product transport based on information about chemistry, quantity, size and shape distribution of graphite dust. •Need to understand the effect of surface oxidation of core graphite on the tribological behavior of graphite.

Phenomena Ranked
Importance-Medium, Knowledge-Medium

(I-M, K-M)

- ***Graphite contains inherent flaws:*** Graphite contains a distribution of inherent flaws that controls the strength. The characteristics of this flaw population must be established, and their effects on important mechanical properties understood in order to design NGNP graphite structures. The flaw structure is one of the components of the graphites texture. Characterization of these flaws by nondestructive methods also needs research and development.
- ***Cyclic fatigue (non-irradiated):*** The extent to which a given grade suffers from cyclic fatigue (S-N Curve) strength needs be determined for both non-irradiated and irradiated graphite. However, data from previous studies on non-NGNP graphites indicate that the cyclic fatigue has a small and negligible effect compared to other phenomena. Thus, limited tests could be performed to confirm this assumption.
- ***Annealing of thermal conductivity*:** When graphite is heated above its previous irradiation temperature by ~50°C, annealing of the defect structure (caused by displacement damage) can occur. Thus, there is some recovery of the thermal conductivity because the internal resistance caused by phonon-defect scattering is reduced. Performance of limited testing can clarify this issue.

- ***Channel distortion*:** Channel distortions may occur because of differential strains. These, in turn, are caused by local differences in dimensional change rates due to temperature and dose gradients. However, this is mostly a design issue, and can be addressed from material test reactor data.
- ***Increased bypass coolant flow channels by break, distortion, etc.:*** Channel distortions may occur because of differential strains. These, in turn, are caused by local differences in dimensional change rates due to temperature and dose gradients. Differential strains may eventually cause failure of graphite core components. However, this is mostly a design issue, and can be addressed from material test reactor data.
- ***Effect of chronic chemical attack on properties*:** Oxidation by air of impurities in the helium coolant to chronic levels will reduce graphite’s mechanical integrity and increase the rate of dust formation. Analytical models and predictive methods are needed to estimate the extent of weight loss of graphite, and its effects in reducing graphite strength.
- ***External (applied) loads*:** Such loads must be quantified and properly accounted for in the design process.

Recommended Research Areas

- Oxidation modeling capability, which requires both oxidation kinetics and diffusion behavior of species within the graphite structure. Kinetics is being actively researched in the USA. However work on studying the diffusion behavior in graphite is necessary so that the desired modeling capability can be developed.
- Accelerated development of ASME code for graphite core components. Adequacy of ASME codes needs to be evaluated to enable core design. This effort needs to be augmented and accelerated.
- Graphite tribological behavior in helium. The current plan does not address this issue properly. An assessment of dust formation, possibility of pebbles sticking and of blocking of channels should be derived from detailed studies on the tribological behavior of graphite, as a function of environment, pressure, temperature and dose.
- Oxidative reactivity of graphite dust powder compared to graphite blocks. Graphite dust can potentially become a carrier for fission products. Models need to be developed on this area, and the appropriate experimental data to support the development of such models need to be generated.

Recommended Research Areas (cont'd)

- Enhanced analytical modeling and predictive capability for irradiation induced dimensional change and creep. These analytical predictive models, supported by phenomenological data and relationships, provide a key input to core behavioral models used to determine core structural degradation and end of core component life. The current effort for this model development should be augmented and accelerated.
- An accepted fracture criteria for nuclear graphite. More work is needed in the US to develop whole core models that integrate all the inputs and predict local stresses of graphite components. Whole core models are critical to enable an agreement on acceptable fracture criteria for graphite.
- An accepted in-service inspection method for graphite core component
- Overall graphite degradation (prediction) model (GDM).
- A graphite core stress analysis method. Although this depends largely on the design, a need exists to develop general guidelines and specific finite element behavioral codes to map the core stress as a function of reactor operation.

Recommended Research Areas (cont'd)

- The potential for stored energy release in irradiated graphite exposed to high temperatures during reactor accidents (i.e., high temperature energy release).
- Knowledge needs for graphite decommissioning: Considerable research is required to address handling and disposal issues of discharged graphite. PMR block reuse should be considered. In addition some of the reactor vendors are considering the option of recycling graphite for nuclear use, which may become a design requirement. However, significant research effort is required on this are before graphite could be recycled

Attachment 3

ORNL/NRC/LTR-09/01

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ORNL/NRC/LTR-09/01

Comparison of NRC Graphite PIRT and DOE Planned Research Activities for Graphite

Milestone Report

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NRC Project Manager: Dr. Makuteswara Srinivasan

February 2009

UT-BATTELLE

ORNL-27 (4-00)

Comparison of NRC Graphite PIRT and DOE Planned Research Activities for Graphite

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U.S. Nuclear Regulatory Commission
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CONTENTS

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LIST OF FIGURES

Figure **Page**

ACRONYMS

ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CTE	coefficient of thermal expansion
DOE	Department of Energy
FOM	Figures of Merit
GTMHR	Gas Turbine Modular Helium Reactor
HTGR	High-Temperature Gas-Cooled Reactor
INL	Idaho National Laboratory
LBE	Licensing Basis Event
NGNP	Next-Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PBMR	pebble bed modular reactor
PBR	pebble bed reactor
PMR	Prismatic-core Modular Reactor
PIRT	Phenomena Identification and Ranking Table
U.S.	United States

EXECUTIVE SUMMARY

This report compares the NRC graphite PIRT results with the DOE-planned research activities for HTGR (NGNP) graphite in order to identify additional research activities related to graphite needed to evaluate safety margins, failure points, and quantify uncertainties. The results are presented according to their ranking order ascribed by the graphite PIRT. Each of the phenomena is described first, then the DOE planned research activities that addresses the phenomena are summarized, and finally a comparison discussion is presented that outlines gaps and weakness in the research plan with respect to normal operations, anticipated operational occurrences, and design basis accidents.

The report concludes with a list of recommended research activities that the NRC may consider implementing to generate experimental data and information, and to develop analytical models, which would permit the evaluation of safety margins, failure points, and to quantify uncertainties in important graphite properties used to design NGNP graphite components.

1. Introduction

The objective of the current work is to compare the Nuclear Regulatory Commission (NRC) Phenomena Identification and Ranking Table (PIRT) rankings on the identified phenomena in graphite with the DOE-planned HTGR research activities for the phenomena, and to identify additional research activities needed to evaluate safety margins, failure points, and quantify uncertainties.

The research method used to identify the gaps and weaknesses in the DOE research plan in addressing the PIRT results was to conduct: (a) a thorough review of related documents; (b) analyze the strength and weaknesses in addressing PIRT results; and (c) document the results of the analysis.

The following documents were reviewed:

- NUREG/CR-6944, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRT), Volume 5: Graphite PIRTs".
- INL/EXT-07-13165, "Graphite Technology Development Plan".
- ORNL/TM-2007/153, "NGNP Graphite Selection and Acquisition Strategy".
- ORNL-GEN4/LTR-06-019, "Experimental Plan and final Design Report for HFIR High Temperature Graphite Irradiation Capsules HTV-1 and -2".
- ORNL/TM-2008/129, "Generation IV Reactors Integrated Materials Technology Program Plan: Focus on Very High Temperature Reactor Materials".

During the review process, gaps between the NGNP PIRT results and the DOE planned research were identified for each one of the PIRT phenomena. The following items were evaluated in the DOE's research program related to graphite: experimental parameters, such as temperature and fluence ranges for testing properties; number of samples to test and sampling adequacy to provide sufficient statistics; handling of uncertainty in data and modeling; lack of modeling, potential challenges that could arise in interpreting data; interpolating and extrapolating DOE's planned research data to operational conditions; and, the tools to be used for analysis and interpretation.

The results are presented in accordance with the ranking order of the graphite PIRT. Each of the phenomena is described first, and then the DOE planned research activities that addresses the phenomena are summarized, finally a comparison discussion is presented with respect to:

1. ***Normal Operations:*** Normal operating temperature, fluence, impurities in helium coolant and their potential effect on degradation of graphite, graphite strength as a function of temperature in impure helium environment, wear of graphite due to coolant flow erosion, wear of graphite due to fuel pebble-graphite core tribology (pebble bed reactor), and any other.
2. ***Anticipated operational occurrences:*** These would be transients expected during reactor start-up and shut-down (for refueling outage, periodic inspections, etc.). The transients may include field effects such as temperature excursions and other stress excursions, among others.
3. ***Design Basis Accidents:*** This includes loss of forced coolant and decompression accidents, among other. For example, air and moisture ingress, after an analyzed accident.

The report represents only the authors' opinion about the absence of PIRT-identified research needs that are not currently being addressed by DOE's research and the authors' opinion on the completeness and adequacy of DOE's research to provide data and information required to address PIRT-identified information needs for regulatory decisions. This report only provides identified data gaps and does not provide suggestions on how to address the identified gaps.

2. Background

The next generation nuclear plant (NGNP) will be a modular high-temperature gas-cooled reactor (HTGR), either a gas-turbine modular helium reactor (GTMHR) version [a prismatic-core modular reactor (PMR)] or a pebble-bed modular reactor (PBMR) version [a pebble bed reactor (PBR)] design, with either a direct- or indirect-cycle gas turbine (Brayton cycle) system for electric power production, and an indirect-cycle component for hydrogen production.

The two (PMR and PBR) reactor designs utilize nuclear-grade graphites as the material for the moderator and core structures. The reactor operating temperature ranges for the two concepts are broadly similar, but the peak neutron dose for the graphite core component in a PBR is substantially greater than that in a PMR. A significant challenge for HTGRs in the United States is that the previous graphite grade qualified for nuclear service in the United States, H-451, is no longer available for new reactors. The precursors from which H-451 graphite was manufactured no longer exist. The present understanding of graphite behavior is not sufficient to enable the available H-451 database to extrapolate to the expected reactor operation conditions and currently available nuclear graphite grades.

In qualifying new grade(s) of graphite, there exists a need for more reliable fundamental understanding of irradiated graphite behavior to develop new theories and models having a sound, in-depth, scientific basis. Such effort will provide increased confidence for design and licensing and reduce the extent of experimental verification that would be needed when additional new graphite grades are developed for HTGR.

Figure 1 shows a schematic representation of all the factors influencing the NGNP research program on graphite and the reactor-type selection process.

Because of the inherent variability in the important properties of graphite, a good understanding of the variability of the physical, chemical, mechanical and thermal properties for a given graphite grade (within billet, between billets, and between production lots) is needed to establish behavioral models of (degradation) phenomena during reactor life. The effects of reactor environment (temperature, neutron irradiation, and chemical attack) on the physical properties must be elucidated. Finally, for each grade of graphite the irradiation-induced dimensional change (which drives the generation of graphite component stresses) and irradiation creep behavior (which relieves graphite component stresses) must be determined over a representative temperature and fluence range.

During early 2007, the U.S. Nuclear Regulatory Commission (NRC) conducted a PIRT exercise, with the support of international experts, to identify those phenomena that could potentially lead to accidents which could release radionuclides outside the containment of the NGNP. The objectives of the graphite PIRT were to identify significant phenomena related to nuclear graphite performance which could affect reactor safety from the degradation of moderator and structural graphite components. The evaluation considered both routine (normal operation) and postulated accident conditions for the NGNP.

The graphite PIRT panel used the specified PIRT process and procedures in their deliberations. Specifically, the panel first discussed the PIRT process, establishing an understanding of the various steps and requirements to identify the phenomena and rank them in order of their importance and the extent of global understanding. The PIRT exercise:

1. Identified the figures of merit (FOM). Three levels of FOM were defined: The top-level or Level 1 FOM was the requirement to maintain dose levels to the public within the regulatory requirements. The Level 2 FOM consisted of three “System” FOM that could influence the top-level FOM, and were identified as those: *i)* leading to increased activity in the helium coolant; *ii)* leading to challenges to the structural integrity of the primary pressure boundary; and *iii)* adversely affecting the ability to attain and maintain cold shutdown and hold down. The Level 3 “component” FOM were: ability to maintain passive heat transfer; maintain ability to control reactivity; ability to protect adjacent components from excessive heat; ability to shield adjacent components from radiation; ability to maintain coolant flow path; ability to prevent excessive mechanical load on the fuel; and, ability to minimize activity in the coolant.
2. Defined the phenomena that affect FOM;
3. Organized the phenomena at component level;
4. Established the importance of each phenomenon and assigned rank of high (H), medium (M), or low (L) based upon the phenomenon’s influence on the FOM;
5. Established the knowledge base of each phenomenon and rated it as H, M, or L and identified pertinent literature; and,
6. reconciled individual panel rankings, and arrived at a consensus panel ranking.

The graphite PIRT panel identified several phenomena, of which five were ranked to be of high importance–low knowledge (I-H, K-L). Nine phenomena were ranked to be of high importance and medium knowledge (I-H, K-M). Two phenomena were ranked as medium importance and low knowledge (I-M, K-L), and a further 14 were ranked as medium importance and medium knowledge (I-M, K-M). The last 12 phenomena were ranked as low importance and high knowledge rank (or similar combinations suggesting they have low priority) (I-L, K-H). In this report detailed analysis is provided for the highest importance phenomena, grouping them when appropriate. The medium and low importance phenomena are analyzed as a group, because the US DOE Research Plan contains little or no description of research activities to address these phenomena.

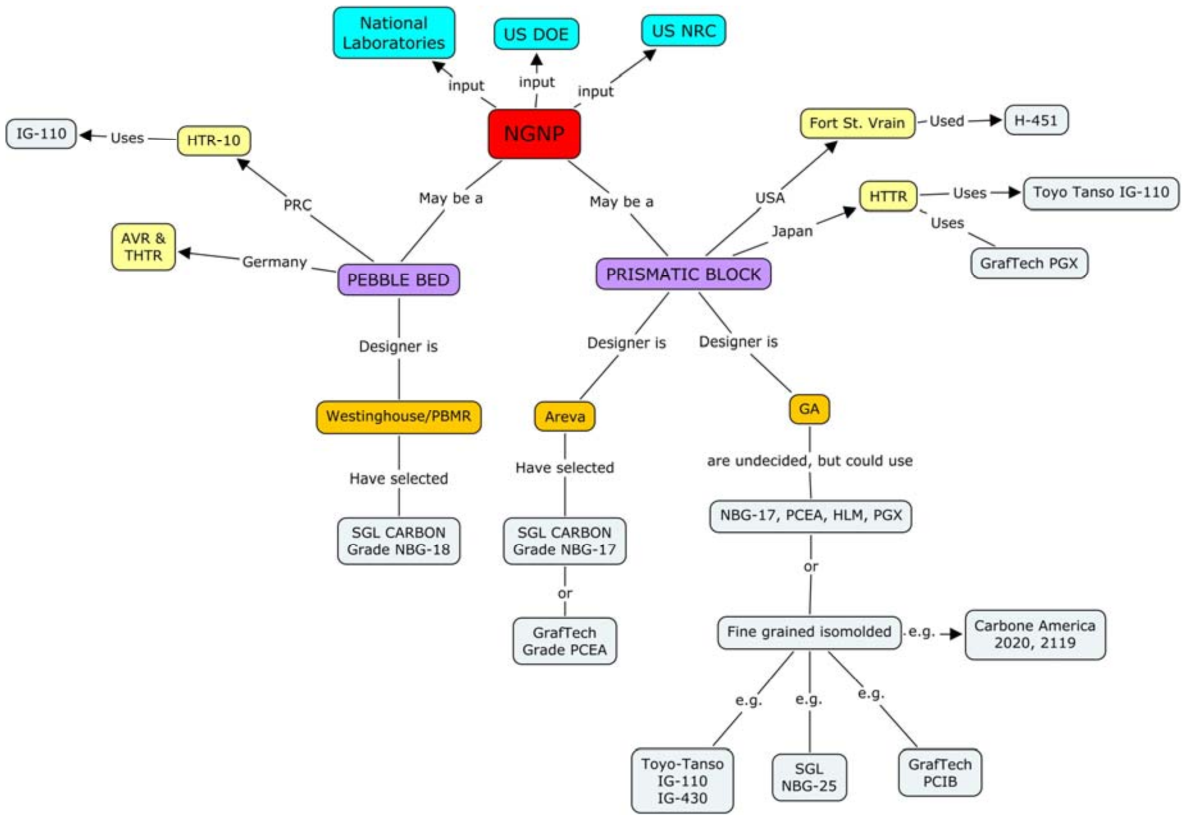

Figure 1. Factors affecting / influencing the NGNP program and the reactor-type selection process

3. Phenomena Ranked Importance-High, Knowledge-Low (I-H, K-L)

Irradiation-induced creep (irradiation-induced dimensional change under stress)

- From PIRT: Stress due to differential thermal strain and differential irradiation-induced dimensional changes would very quickly cause fracture in the graphite components if it were not for the relief of stress due to irradiation-induced creep. The phenomena and mechanism of irradiation-induced creep in graphite is therefore of high importance. Currently there are no creep data for the graphite grades being proposed for use in the NGNP. However, creep at low dose follows a linear law that can be explained through a dislocation pinning/unpinning model due to Kelly and Foreman. Marked deviation from this law has been observed at intermediate neutron doses. The applicability of the law has been extended by taking into account changes in the pore structure that manifest themselves as changes in the CTE with creep strain. However, the current creep law breaks down at high-temperature, moderate-dose and moderate-temperature high-dose combinations. A new model for creep is needed that can account for the observed deviations from linearity or the creep strain rate with neutron dose. Existing and new models must be shown to be applicable to the currently proposed graphite grades. Knowledge rank was therefore considered as low.

Irradiation-induced change in CTE, including the effects of creep strain

- From PIRT: Differential thermal strains occur in graphite components due to temperature gradients and local variation in the CTE. The CTE changes depend on irradiation conditions (temperature and dose) and the irradiation induced creep strain. Thus, the importance ranking is high for this phenomenon. Irradiation-induced changes in CTE are understood to be related to changes in the oriented porosity in the graphite structure. The changes are observed to be different when graphite is placed under stress during irradiation. The direction and magnitude of the stress (and creep strain) affect the magnitude of the CTE change. Only limited data are available for the effect of creep strain on CTE in graphite, and none of this data is for the grades proposed for the NGNP. Thus, the knowledge rank is low.

Irradiation-induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress)

- From PIRT: The properties of the graphite are known to change with neutron irradiation, the extent of which is a function of the neutron dose, irradiation temperature, and irradiation-induced creep strain. Differential changes in elastic moduli, strength, and toughness must be accounted for in design. The importance of this phenomenon is thus ranked high. Although data exist for the effect of neutron dose and temperature on the mechanical properties of graphite, there are few data on the effects of creep strain on the mechanical properties. Moreover, none of the available data is for the grades currently being considered for the NGNP. Knowledge ranking is therefore low.

Summary of DOE work plan: The DOE Graphite Technology Development Plan (INL/EXT-07-13165) reports that the AGC experiments are designed to provide irradiation on creep rate data for moderate doses and higher temperatures of leading graphite types that may be used in the NGNP reactor. The experiments are designed to provide not only static irradiation material property changes (on non-stressed material) but also to determine irradiation creep parameters for actively stressed (i.e., compressively loaded) specimens during exposure to a neutron flux.

The AGC1-AGC6 experiments will cover the temperature range from 500 ^{o}C to 1200 ^{o}C and irradiation doses up to 7 dpa. These conditions bound only the operating conditions for the PMR reactor, and most of them are below the expected point of (volume-change) turnaround for the current NGNP graphite types at normal operating conditions. Only one experiment (AGC-6) may approach expected turnaround limits for selected NGNP graphite types.

Discussion: The current US DOE work plan (AGC1-AGC-6) will generate experimental data to properly evaluate the first three top-ranked phenomena: (a) irradiation-induced creep (irradiation-induced dimensional change under stress, (b) irradiation-induced change in CTE, including the effects of creep strain, and (c) irradiation-induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress) ***only*** for the operational conditions of a prismatic (PMR) NGNP reactor design and for the selected graphites NBG-18 & NBG-17 (SGL) PCEA (GrafTech International), and IG-110 & IG-430 (Toyo Tanso).

The cumulative dose (dpa) levels achieved in the AGC experiments will not bound the conditions for pebble-bed NGNP design reflector blocks, which are expected to experience higher dose. The planned experiments will only provide preliminary data for the first 20 – 25% of the expected dpa levels for the PBR graphite components. A tentative plan for a high-dose creep experiments exposing selected graphite to much longer dose levels at moderate temperatures exists, in case the PBR design is selected. However, funding to carry out this experimental work has not been allocated.

The experimental data from AGC1-AGC6, along with historical data, will be utilized to improve current predictive models and/or develop new models that will allow the reactor designers to predict the conditions of graphite components and core structure design margins at any point in the lifetime of the reactor. The models will provide the ability to calculate in-service stresses and strains in graphite components and estimate the structural integrity of the core as a whole. These models can be used to interpolate point-to-point flux and temperature data bound by the experimental data. However, these models can not, and should not, be used to extrapolate graphite behavior at higher doses, similar to those expected in a PBR reactor design.

Therefore, if in the future the NGNP Project chooses the PBR design, it is critical that the higher dose, moderate temperature experiments be performed and that the predictive models are expanded to cover the operating conditions of a PBR reactor.

Moreover, the accuracy of predictive models must be considered. 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. However, it is important that the NGNP Project have the independent capability to validate these models and ensure the safety envelope of the core during normal and off-normal operating conditions. The US DOE work plan includes Multi-scale Model Development; however, this task needs to be emphasized and accelerated at the different inter-dependent technical areas and then integrated to develop a reliable whole core model.

The above discussion has assumed that if the PMR reactor design is chosen, there will be no reuse or recycling of graphite blocks. However, recycling of graphite blocks is being considered by some of the reactor vendors and may become a design requirement. If this is the case, the experimental plan will need to be expanded to include longer duration irradiation experiments. The option of recycling graphite for nuclear use is attractive from the perspective that reduces the cost and volume of graphite sent to disposal or storage. However, considerable research is required to address handling and disposal issues of discharged graphite.

Phenomena: • *Irradiation-induced creep (irradiation-induced dimensional change under stress)* • *Irradiation-induced change in CTE, including the effects of creep strain* • *Irradiation-induced changes in mechanical properties (strength, toughness), including the effect of creep strain (stress)*	
	GAPS / WEAKNESSES
Normal Operations*	• Experimental data fully bounds conditions for PMR design. • Need additional experimental data if PBR design is selected.
Anticipated operational occurrences	• Data will enable development of predictive models. • Weakness is in model development. Effort for model development needs to be augmented and accelerated
Design basis accidents	• Creep data will have to be combined with data from oxidation studies to account for effects due to accidents involving air and moisture ingress. • Weakness/gap is more related to the lack of knowledge in the area of graphite oxidation.
*Normal operations for a PMR design currently do not include recycling of graphite blocks.	

Blockage of fuel element coolant channel due to graphite failure and/or graphite spalling

- From PIRT: Significant uncertainty exists as to the stress state of any graphite component in the core. Moreover, the strength of the components changes with dose, temperature, and creep strain. The combination of these factors makes the probability of local failure, graphite spalling, and possible blockage of a fuel element coolant channel difficult to determine. Consequently the panel rated this phenomenon's importance as high. Although the changes in properties of graphite have been studied for many years, there are still data gaps that make whole core modeling very difficult (e.g., effect of creep strain on properties). Moreover, data on the grades selected for NGNP are not available. Therefore, the panel rated the knowledge base for this phenomenon as low.

Blockage of coolant channel in reactivity control block due to graphite failure and/or graphite spalling.

- From PIRT: Significant uncertainty exists as to the stress state of any graphite component in the core. Moreover, the strength of the components changes with dose, temperature, and creep strain. The combination of these factors makes the probability of local failure, graphite spalling, and possible blockage of a coolant channel in a reactivity control block difficult to determine. Consequently the panel rated this phenomenon's importance as high. Although the changes in properties of graphite have been studied for many years there are still data gaps that make whole core modeling very difficult (e.g., effect of creep strain on properties). Moreover, data on the grades selected for NGNP are not available. Therefore, the panel rated the knowledge base for this phenomenon as low.

Summary of DOE work plan: Results from physical properties, irradiation effects and creep strain data, will be combined with core model and stress prediction to determine failure criteria. This work is being done under the ASME committee. The NGNP program is funding personnel to be involved and/or lead these tasks in the ASME Committee.

Discussion: Expand knowledge of the effect of creep strain on properties of currently available grades of graphite to facilitate whole-core modeling that will enable the monitoring of the stress state of any graphite component in the core. This will be handled by consensus codes; similarly safety margins could be increase to account for uncertainties in the models

<table>
<tr><td colspan="2">Phenomena:
• Blockage of fuel element coolant channel due to graphite failure and/or graphite spalling
• Blockage of coolant channel in reactivity control block due to graphite failure and/or graphite spalling</td></tr>
<tr><td></td><td>GAPS / WEAKNESSES</td></tr>
<tr><td>Normal Operations*</td><td rowspan="2">• Would the design code (ASME code) be available for core design? Need to accelerate code development.
• Determining the adequacy of the design code (ASME code)</td></tr>
<tr><td>Anticipated operational occurrences</td></tr>
<tr><td>Design basis accidents</td><td>• Understanding weakening of graphite surfaces due to long term oxidation.</td></tr>
<tr><td colspan="2">*Normal operations for a PMR design currently do not include recycling of graphite blocks.</td></tr>
</table>

4. Phenomena Ranked Importance-High, Knowledge-Medium (I-H, K-M)

Statistical variation of non-irradiated properties

- From PIRT: The graphite single crystal is highly anisotropic due to the nature of its bonding (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 graphitized binder region. Thus, the mechanical and physical properties of graphite exhibit anisotropy, and vary within a billet due to texture introduced during forming and thermal processing. Moreover, there is variability in the properties between billets within the same lots, between lots, and between production batches due to variations on raw materials, formulations, and processing conditions. Therefore, it is necessary to develop a statistical data base of the properties for a given graphite grade. Variations in the chemical properties (chemical purity level) will have implications for chemical attack, degradation, and decommissioning). Probabilistic design approaches are best suited to capturing the variability of graphite. The panel rated this phenomenon as of high importance. Although other nuclear graphites have been characterized and full databases developed, thereby allowing an understanding of the textural variations, only limited data exist on the graphites proposed for the NGNP. Therefore, the panel rated this phenomenon's knowledge level as medium.

Summary of DOE work plan: The DOE Graphite Technology Development Plan (INL/EXT-07-13165) (section 5.1.1) has developed an optimal method of machining the graphite samples from the bulk material to ensure that 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 spatial validity so that microstructural changes within the bulk material (i.e., billet) affecting material property changes are well characterized. Particular attention has been given to the traceability of each specimen to its spatial location and orientation within a billet.

The 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 standards). Since this is physically impractical, 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. For example, it is usually understood that a processed graphite billet exhibits a reasonable amount of orthogonal isotropy in its properties (or it is transversely isotropic). Using samples from similar locations within each billet section will yield enough samples to provide for statistical validity within a single billet.

The variability within individual billet, from billet-to billet and from production lot-to-lot will be analyzed in a statistical manner to determine the maximum range of material property variations expected for components machined from a typical billet. Such a statistical material property database can only be obtained from extensive characterization of as-received graphite samples taken within billets, and compared to samples between different billets and different graphite production lots.

Discussion: The current billet cutting plans and the testing matrix from the DOE Graphite Technology Development Plan are very comprehensive, and are designed to properly characterize the variability within graphite components. The data obtained with such statistical validity will provide acceptable technical bases and enable credible core design, and the support of the ongoing development of a graphite probabilistic design methodology.

Therefore, the work, as described in the DOE Graphite Technology Plan, will provided enough experimental data to properly evaluate the statistical variation of non-irradiated graphite properties, for the selected graphite types (NBG-18 and PCEA).

<table>
<tr><td colspan="2">Phenomena:
• Statistical variation of non-irradiated properties</td></tr>
<tr><td></td><td>GAPS / WEAKNESSES</td></tr>
<tr><td>Normal Operations</td><td rowspan="3">• The current plan will provide sufficient experimental data to have a good characterization of the statistical variation of important non-irradiated properties for the selected graphite types.
• There is a gap/weakness with respect to the characterization of additional graphites in order to expand the pool of qualified graphites for use in either type of reactor. However, if there are no additional graphites, presently a gap does not exist for this phenomenon.</td></tr>
<tr><td>Anticipated operational occurrences</td></tr>
<tr><td>Design basis accidents</td></tr>
</table>

Consistency in graphite quality over the lifetime of the reactor fleet (for replacement, for example)

- From PIRT: Graphite is manufactured from cokes and pitches, which are derived from naturally occurring organic sources such as oil (in the form of coal tar pitch) and coal. These sources are subject to geological (natural) variations and depletion, requiring the substitution of alternate sources for coke and pitch manufacture. It is recognized that the important graphite properties are influenced strongly by the nature and type of raw materials used. Therefore, ensuring the consistency of graphite quality and properties over the lifetime of a reactor, or the reactor fleet (for replacement, for example), is of importance, and challenging. The panel ranked the importance of this phenomenon as high. Our understanding of this phenomenon is sufficient because we were able to develop generic specifications (ASTM DO2.F, D 7219-08, and D 7301-08) that should assure the required quality and repeatability. However, these are only recent specifications and the quality and repeatability in manufacturing nuclear graphite as per the specifications have yet to be demonstrated. The panel assessed the knowledge base for this phenomenon as medium.

Summary of DOE work plan: The DOE Graphite Technology Development Plan (INL/EXT-07-13165) has funding for the development of ASTM test standards. Under this plan, the plan participants will be actively involved with the ASTM committee in writing standards, and participating in round robin testing.

Discussion: ASTM standard test methods include material specifications such as the ASTM D02.F, D 7219-08 (Standard Specification for Isotropic and Near-Isotropic Nuclear Graphites) and ASTM D02.F D7301-08 (Standard Specification for Nuclear Graphite Suitable for Components Subjected to Low Neutron Irradiation Dose). Currently, the ASTM subcommittee on nuclear graphite is also reviewing some 35 existing properties determination standards for possible update with new information, and the development of new standards, particularly applicable to HTGR application, such as fracture toughness testing. Standards for nondestructive testing of graphites are also under consideration. Conformance to consensus test standards and the use of graphite conforming to the consensus material specification will provide the required data that will ensure quality and repeatability of graphite properties.

<table>
<tr><td colspan="2">Phenomena:
• Consistency in graphite quality over the lifetime of the reactor fleet (for replacement, for example)</td></tr>
<tr><td></td><td>GAPS / WEAKNESSES</td></tr>
<tr><td>Normal Operations</td><td rowspan="3">• The major gap is with respect to identifying alternative coke sources to the two currently available (U.S. pet coke and Japanese pitch coke) for manufacturing HTGR graphite, and testing the validity of the ASTM standard specifications to new graphites.
• Another gap is with respect to exploring the options for graphite recycling and reuse for nuclear applications.</td></tr>
<tr><td>Anticipated operational occurrences</td></tr>
<tr><td>Design basis accidents</td></tr>
</table>

Irradiation-induced dimensional change

- From PIRT: Neutron irradiation causes dimensional changes in graphites. Theses changes are due to anisotropic crystal dimensional change rates (a-axis shrinkage and c-axis growth), the interaction of crystal dimensional change with porosity, and the generation of new porosity. The amount of irradiation-induced dimensional change is a function of the neutron dose and irradiation temperature (and applied load, i.e, irradiation induced creep as discussed above). Consequently, gradients in temperature or neutron dose will introduce differential dimensional changes (strains). Irradiation-induced dimensional changes are considered to be the largest source of internal stress. Because of the significance of dimensional changes in generating core stresses, the panel gave this phenomenon high importance. Irradiation-induced dimensional changes have been researched for many years, and several dimensional change models have been proposed. However, there is a paucity of data for the dimensional changes of the NGNP candidate graphites. Therefore, the knowledge rank was considered as medium.

Irradiation-induced thermal conductivity changes and degradation of thermal conductivity

- From PIRT: Displacement damage caused by neutron irradiation introduces additional phonon scattering sites to the graphite crystal lattice and consequently reduces the thermal conductivity. The nature of the irradiation-induced damage is sensitive to the temperature of irradiation. Consequently, the extent of degradation is temperature-dependant. In addition, phonon-phonon (Umklapp) scattering increases as the measurement temperature increases, and thus the thermal conductivity decreases as the temperature increases. At relatively large irradiation doses, thermal conductivity reduces further, at an increased rate, and the phenomenon is attributed to porosity generation due to large changes in crystal dimensions. Some amount of thermal conductivity also recovers (anneals on heating above the irradiation temperature (such as during a thermal transient). The exact magnitude of the change in thermal conductivity of the core under reactor operation is therefore subject to some uncertainty. A thermal conductivity lower than that required by design basis for licensing basis event (LBE) heat removal could exist due to: (a) inadequate database to support design over component lifetime: or (b) statistical and textural variations in characteristics of graphites from lot to lot. Such scenario has the potential to allow fuel design temperatures to be exceeded during a LBE. Therefore, the panel considered the importance of this phenomenon as high. Irradiation-induced thermal conductivity changes have been researched for many years and several models have been proposed. However, there is a paucity of data for the conductivity changes of the graphites proposed for the NGNP. Therefore, the knowledge rank was considered as medium.

Summary of DOE work plan: Evaluation of irradiation-induced dimensional changes and thermal property changes are covered in the experimental plan for AGC1-AGC6 (500 ^{o}C to 1200 ^{o}C, and up to 7 dpa) and HTV-1 and HTV-2 (up to 1600 ^{o}C and up to 4 dpa).

Discussion: The current US DOE work plan (AGC1-AGC-6 and HTV-1 and HTV-2) will generate experimental data to properly evaluate these two phenomena only for NBG-18 & NBG-17 (SGL) PCEA (GrafTech International), and IG-110 & IG-430 (Toyo Tanso), and for the operational conditions of a prismatic (PMR) NGNP reactor design. The work plan has a tentative experimental design for higher-dose, moderate temperature conditions, in case that the PBR design is selected. However funding to carry out this experimental work has not been allocated.

<table>
<tr><td colspan="2">Phenomena:
• Irradiation-induced dimensional change
• Irradiation-induced thermal conductivity changes and degradation of thermal conductivity</td></tr>
<tr><td></td><td>GAPS / WEAKNESSES</td></tr>
<tr><td>Normal Operations</td><td rowspan="2">• Experimental data bounds conditions for PMR design for up to the expected volume change turnaround behavior with probably not enough margin in dose or temperature.
• Need additional experimental data if PBR design is selected
• Data will enable the development of predictive models.
• Weakness is in model development, i.e., effort for model development needs to be augmented and accelerated.
• Weakness is also in the validation of models to include new graphites.</td></tr>
<tr><td>Anticipated operational occurrences</td></tr>
<tr><td>Design basis accidents</td><td>• Dimensional and thermal conductivity changes are large sources of internal stresses. Experimental data from this task will be combined with data from oxidation studies to develop whole core models in order to model behavior under accidents relating to air and moisture ingress.
• Weakness/gap is related to the lack of knowledge in the area of graphite oxidation, especially on irradiated graphite.</td></tr>
</table>

Irradiation-induced changes in elastic constants, including the effects of creep strain

- From PIRT: Neutron irradiation induces changes in the elastic constants of graphite. Initial increases in the moduli are attributed to an increase in dislocation pinning points in the basal plane, which reduce the crystal shear compliance, C44. Subsequent changes in the elastic modulus are attributed to pore-structure changes (initial pore closures followed by pore generation). Although there appears to be a significant phenomenological understanding of irradiation-induced modulus changes, there are no direct microstructural observations or sufficiently well developed analytical mechanistic model in support of phenomenological model. Therefore, the knowledge rank was considered as medium.

Summary of DOE work plan: Evaluation of irradiation-induced changes in elastic constants, including the effects of creep strain are covered in the experimental plan for AGC1-AGC6 (500 ^{o}C to 1200 ^{o}C, and up to 7 dpa) for NBG-18 and PCEA graphites.

Discussion: The current US DOE work plan (AGC1-AGC-6) will generate experimental data to properly evaluate this phenomenon for NBG-18 & NBG-17 (SGL) PCEA (GrafTech International), and IG-110 & IG-430 (Toyo Tanso) graphite and for the operational conditions of a prismatic (PMR) NGNP reactor design. The work plan has a tentative experimental design for higher-dose, moderate temperature conditions, in case that the PBR design is selected. However funding to carry out this experimental work has not been allocated.

The experiments will measure changes in modulus as a function of irradiation dose. The resulting data, with the combination of dimensional change (volume change) measurements will provide information necessary to enable an understanding of the pore structure changes. However, there is a gap with respect to a basic understanding of crystal strain in graphite and the development of a predictive model, based in dimensional changes of the crystallites.

Phenomena: • *Irradiation-induced changes in elastic constants, including the effects of creep strain*	
	GAPS / WEAKNESSES
Normal Operations Anticipated operational occurrences	• Experimental data bounds conditions for PMR design for up to the expected volume change turnaround behavior with probably not enough margin in dose or temperature. • Need additional experimental data if PBR design is selected • Data will enable the development of predictive models. • Weakness is in model development, i.e., model development effort needs to be augmented and accelerated.
Design basis accidents	• These data will be combined with data from oxidation studies to develop/improve predictive models that account for accidents relating to air and moisture ingress. • There is gap/weakness in the understanding and predictive model development for irradiation-induced crystal strain in the graphite block. • Weakness/gap is related to the lack of knowledge in the area of graphite oxidation, particularly on irradiated graphite.

Tribology of graphite in (impure) helium environment

- From PIRT: Graphite is inherently lubricious and used to reduce friction between contacting surfaces. However, its behavior is modified by the HTGR helium environment. The abrasion of graphite blocks on one another or of the fuel pebbles themselves, and on the graphite moderator blocks can produce graphite dust. Studies are needed to assess the effect of the helium environment on the friction and wear behavior of graphite. The possibility that fuel pebbles can “stick” together and cause a fuel flow blockage must be explored, although German pebble bed reactor operation did not exhibit such problem.(i.e., no blockages). The consequences of dust generation (possible fission product transport mechanism) and possible fuel pebble interactions resulted in the panel ranking the importance of this phenomenon as high. Some literature exists on this subject mostly from the past German program. Consequently, the panel ranked the knowledge level as medium.

Summary of DOE work plan: The DOE research will perform standard 'pin-on-wheel' wear tests to determine wear, friction, and dust generation data 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 PBR design, i.e., NBG-18.

Discussion: The US DOE plan only intends to study the tribological properties at ambient conditions, and there is no mention of understanding the effect of helium environment, and high pressures and temperatures. In addition, the issues of dust formation and 'sticking' of pebbles and the consequences of the events are not addressed in the research plan.

Dust generation, can be a serious issue, especially from the perspective of the graphite dust acting as a transport medium for fission products. A significant research effort needs to be initiated to gather information about the chemistry, quantity, size and shape distribution of graphite dust in order to enable accurate modeling of fission product transport. In addition, graphite dust, when agglomerated, could to lead to blockages for coolant flow or free movement of control rods. It is also important to understand the adsorption and adhesion behavior of graphite dust on various materials surfaces present in HTGR, like: various metals and alloys, ceramic insulation and core graphite, and core supports.

The surface condition (non-oxidized or oxidized) will impact the tribological behavior and must be understood. A predictive capability for oxidation weight loss is needed. This requires knowledge of both oxidation kinetics and the diffusion characteristics of reactive species in the graphite.

<table>
<tr><td colspan="2">Phenomena:
• Tribology of graphite in (impure) helium environment</td></tr>
<tr><td></td><td>GAPS / WEAKNESSES</td></tr>
<tr><td>Normal Operations</td><td rowspan="2">• The current DOE research plan considers wear and friction only a significant problem for the PBR design. However, wear and friction have the potential of being significant issues for PMR design as well due to the erosion of relatively soft surface of graphite by the high-velocity helium flow in the coolant channels.
• Need to study tribological properties of graphites considered for PMR design.
• Need to study the tribological properties of graphite in helium environment, and at high pressures and temperatures.
• Need to study the tribological properties over the life of reactor, i.e., as function of temperature, dose, and oxidation weight loss.
• Need to study tribological properties for graphites for PBR and PMR design as a function of oxidation and weight loss.</td></tr>
<tr><td>Anticipated operational occurrences</td></tr>
<tr><td>Design basis accidents</td><td>• Graphite dust generation is a serious issue, and therefore significant research is needed focused on characterizing the dust formed and on understanding what happens to the dust generated, as well as to develop models for fission product transport.</td></tr>
</table>

Blockage of Reactivity Control Channel due to graphite failure, spalling

- From PIRT: Significant uncertainty exists on the state of stress of graphite core components. The strength of the components changes with dose, temperature, and creep strain. The interaction of these operation-dependent factors makes the probability of local fracture, graphite spalling, and possible blockage of a reactivity control channel in a reactivity control block difficult to determine. Consequently, the panel rated this phenomenon's importance as high. The panel was also cognizant that the NGNP designs are known to be capable of safe shutdown without control rod entry. Although the changes in properties of graphite have been studied for many years, there are still data gaps that make whole core modeling very difficult (e.g., effect of creep strain on properties). Moreover, data on the grades selected for NGNP are not available. Therefore, the panel rated the knowledge base for this phenomenon as medium.

Summary of DOE work plan: The current plan covers the generation of experimental data required for whole core model that such as statistical variation of properties, anisotropy, microstructural analysis, etc.

Discussion:

Phenomena: • *Blockage of Reactivity Control Channel due to graphite failure, spalling*	
	GAPS / WEAKNESSES
Normal Operations Anticipated operational occurrences	• Need emphasis on analytical model development capable of accounting for key factors such as: o Statistical variation of inherent properties o Microstructural differences between grades o Anisotropy due to forming and manufacture, and o Geometric factors, such as stressed volume and specimen/component dimensions o Strength changes due to oxidation, temperature, and neutron irradiation.
Design basis accidents	• Whole Core model needs to account for changes in properties due to graphite oxidation and weight loss. However, there is a gap regarding oxidation behavior and diffusion of species within the graphite structure, both for non-irradiated and irradiated graphite.

Graphite temperatures

- From PIRT: The graphite component life (structural integrity) and transient calculations require time-dependent and spatial predictions of graphite temperatures. Graphite temperatures for normal operation and transients are usually supplied to graphite specialists by thermal-hydraulics specialists. Although, in some cases, gas temperatures and heat transfer coefficients are supplied, and the graphite specialists calculate the graphite component temperatures from these data input.

Summary of DOE work plan: Some work is planned by the NGNP program at INL, however, it is not done by the NGNP Materials Program.

Discussion: There are well-established codes in this area; however, the adequacy of these codes for the new high-temperature reactor designs needs to be assessed. This assessment resides in the domain of the reactor physicist and thermal-hydraulic analysts.

5. Phenomena Ranked Importance-Medium, Knowledge-Low (I-M, K-L)

Graphite dust generation

- From PIRT: Abrasion between adjacent graphite blocks, or fuel pebbles and reflector blocks, will cause the formation of dust. The dust may become a carrier for fission products or could possibly impede coolant flow, if the dust agglomerates and deposits on the walls of the coolant channels.

Blockage of reflector block coolant channel—due to graphite failure, spalling

- From PIRT: Blockage of coolant channels by graphite debris could cause local hot spots in the core.

Summary of DOE work plan: Current DOE program has plans to evaluate the tribological properties of graphite NBG-18 (for use in PBR design) at ambient conditions. However, there is not a formal plan for studying tribological properties in helium environment or dust formation at elevated temperatures and pressures, and the behavior of the dust under these conditions.

Discussion: Dust generation (as discussed earlier in this report), can be a serious issue, especially from the perspective of dust being a transport media for fission products. A significant research effort needs to be initiated to gather information about the chemistry, quantity, size and shape distribution of graphite dust in order to enable accurate modeling of fission product transport.

In addition, graphite dust could to lead to blockages, and therefore it is important to understand the adsorption/desorption/re-adsorption and adhesion behavior of graphite dust on various materials surface, such as metals and alloys used in HTGR, ceramic insulation and core graphite, and core supports.

<table>
<tr><td colspan="2">Phenomena:
• Graphite dust generation</td></tr>
<tr><td></td><td>GAPS / WEAKNESSES</td></tr>
<tr><td>Normal Operations</td><td rowspan="2">• Need to study tribological behavior of graphite in a helium environment, both under solid-solid abrasive contact conditions and under high velocity gas-solid contact conditions.
• Need to study the effect of irradiation and temperature on tribological behavior.
• Need to include graphites considered for PMR design and not only those considered for PBR design.
• Need to study the oxidation behavior of graphite dust and whether graphite dust can be a vehicle for the transportation of fission products.</td></tr>
<tr><td>Anticipated operational occurrences</td></tr>
<tr><td>Design basis accidents</td><td>• Need to develop models for fission product transport based on information about chemistry, quantity, size and shape distribution of graphite dust.
• Need to understand the effect of surface oxidation of core graphite on the tribological behavior of graphite.</td></tr>
</table>

6. Phenomena Ranked Importance-Medium, Knowledge-Medium (I-M, K-M)

The last set of phenomena was ranked as of medium importance and medium knowledge. The current DOE work plan does not address these issues individually, and therefore it is difficult to make an appropriate assessment of how these phenomena are being addressed. Even though some data exist, these phenomena do need to be researched and addresses properly. More efforts are needed in the following areas:

- ***Graphite contains inherent flaws:*** Graphite contains a distribution of inherent flaws that controls the strength. The characteristics of this flaw population must be established, and their effects on important mechanical properties understood in order to design NGNP graphite structures. The flaw structure is one of the components of the graphites texture. Characterization of these flaws by nondestructive methods also needs research and development.

- ***Cyclic fatigue (non-irradiated):*** The extent to which a given grade suffers from cyclic fatigue (S-N Curve) strength needs be determined for both non-irradiated and irradiated graphite. However, data from previous studies on non-NGNP graphites indicate that the cyclic fatigue has a small and negligible effect compared to other phenomena. Thus, limited tests could be performed to confirm this assumption.

- ***Annealing of thermal conductivity***: When graphite is heated above its previous irradiation temperature by ~50°C, annealing of the defect structure (caused by displacement damage) can occur. Thus, there is some recovery of the thermal conductivity because the internal resistance caused by phonon-defect scattering is reduced. Performance of limited testing can clarify this issue.

- ***Channel distortion***: Channel distortions may occur because of differential strains. These, in turn, are caused by local differences in dimensional change rates due to temperature and dose gradients. However, this is mostly a design issue, and can be addressed from material test reactor data.

- ***Increased bypass coolant flow channels by break, distortion, etc.:*** Channel distortions may occur because of differential strains. These, in turn, are caused by local differences in dimensional change rates due to temperature and dose gradients. Differential strains may eventually cause failure of graphite core components. However, this is mostly a design issue, and can be addressed from material test reactor data.

- ***Effect of chronic chemical attack on properties***: Oxidation by air of impurities in the helium coolant to chronic levels will reduce graphite's mechanical integrity and increase the rate of dust formation. Analytical models and predictive methods are needed to estimate the extent of weight loss of graphite, and its effects in reducing graphite strength.

- ***External (applied) loads***: Such loads must be quantified and properly accounted for in the design process.

7. Summary Remarks

7.1 Graphite brick stress and fracture:

Predicting when fracture of irradiated NGNP graphite component may occur is very difficult because of the lack of reliable data on properties that are influenced by the reactor environment and predictive model for NGNP graphites. During reactor service the graphite blocks in the reactor are subject to direct stress by imposed mechanical load, thermal stress due to temperature-dependent CTE, and stresses that occur from differential irradiation-induced dimensional changes. These stresses may be relaxed by irradiation creep. Consequently, calculating the stress state at any point in a reactor component requires precise knowledge of the spatial variations of neutron dose and temperature, dimensional change, the creep rate, and physical properties (as a function of dose and temperature) of graphite, typically, thermal conductivity, CTE, elastic modulus, and strength. All of these properties have associated uncertainties due to anisotropy and the inherent variability of graphite properties. Consensus behavior model and computational codes have to be developed for each of these behavioral properties of NGNP, which will become the input codes to the overall structural stress analysis code for a core graphite component. Having established an estimate of the stress state of a graphite component, a suitable fracture criterion must be adopted to ascertain the probability of failure. Or, the predicted failure stress must be compared to the design stress to ensure design margins have not been exceeded.

Thus, to determine the fracture probability, but not necessarily the exact time of fracture, the following data must be known:

1. Spatial and temporal variations in fast neutron dose and temperature. Usually this information is provided by thermal fluid analysis codes and core physics codes.
2. Variations of the physical properties within the graphite blocks due to anisotropy induced by manufacture. Consensus code needs to be developed, based on adequate experimental data.
3. Billet to billet and lot by lot statistical properties variations for the graphite grade in question. Consensus code needs to be developed, based on adequate experimental data.

4. Physical properties of the graphite grade in question, including the variations of these properties with temperature, and neutron dose. Consensus code needs to be developed, based on adequate experimental data.
5. Irradiation induced dimensional change rate and creep rate. Consensus code needs to be developed, based on adequate experimental data.
6. Possible strength and property losses due to thermal oxidation of the graphite. Consensus code needs to be developed, based on adequate experimental data.
7. A fracture criterion to allow prediction of the probability of fracture for any given stress level. Consensus method for this prediction and code needs to be developed.

It has to be noted that fracture in a graphite component may not constitute a safety hazard as the function of the component may not be impaired. However, for certain components, such as control rod blocks, local fracture may impede the free in- and out- movement of a control rod. The DOE (US) and international research programs are directed toward providing a complete statistical characterization of graphite properties for the NGNP graphite grades. Similarly, irradiation effects and creep experiments are planned to yield suitable data. Reactor physics codes and thermal hydraulics codes are being developed to provide local spatial and temporal estimates. However more work is needed in the US to develop whole core models that integrate all the inputs and predict local stresses. The international graphite community has yet to agree on an acceptable fracture criteria for graphite, although several have been proposed. Similarly, there is not yet an accepted model for irradiation-induced creep at high doses and temperatures. Consequently, further work on model development is indicated.

Strength losses due to oxidation for the grades of interest have not been elucidated, and models for local weight loss calculations are needed. To this end, knowledge of both oxidation kinetics and the diffusion of oxidizing species through the graphite must be obtained. Some work on oxidation kinetics is being performed by the US program, but the determination of diffusion behavior is not yet being studied.

Finally, efforts are under way to codify the graphite design rules through the American Society of Mechanical Engineers (ASME). This work is vital to assure that conservative margins are established and seismic event and fatigue behavior are considered. The ASME code development should be accelerated such that a consensus code is available for NGNP design.

7.2 Tribology and dust generation:

During operation graphite-graphite contact in the reactor may produce graphite dust. Such wear may occur when adjacent blocks “fret” due to flow or plant induced vibrations. In a pebble bed reactor the motion of the pebbles through the bed against the graphite inner and outer reflector surface may produce graphite dust. Similarly pebble on pebble contact will be a potential source of dust.

There are two consequences of dust formation:

1. Mass loss on graphite surfaces may locally weaken the graphite.
2. The dust transport about the reactor. Any fission products associated with the dust will also be transported and may not be contained in a reactor depressurization event.

Several factors will affect the formation of graphite dust in the reactor. First, the basic tribological behavior of graphite in helium (e.g., friction, wear rates) needs to be evaluated for the graphite grades (and fuel pebble matrix material) in question. Second, the effect of neutron irradiation on tribological behavior need to be determined, and finally, the extent to which graphite/ pebble surface oxidation affects the tribological behavior must be understood, including the potential oxidation of the dust itself.

The loss of brick strength has been addressed (see "Brick Stress and Fracture"). However, the tribological behavior of graphite in a helium environment should be better understood. Currently no work is being performed in the USA on this subject.

The fate of any graphite dust produced has not been examined. Is the dust more chemically reactive than the pebbles or graphite block? If so, it will be preferentially gasified and flushed from the reactor in the coolant gas. The potential transport of fission products through this route need to be understood. A sound understanding of oxidation kinetics and oxidative species diffusion behavior is therefore indicated. In addition a good understanding of the chemistry, quantity, size, and shape distribution of graphite dust is necessary to enable development for fission product.

8. Recommended Research Areas:

After reviewing and comparing the PIRT identified phenomena with the US DOE, the following gaps and weaknesses in the DOE research plan have been identified, which would need additional research:

1. Oxidation modeling capability, which requires both oxidation kinetics and diffusion behavior of species within the graphite structure. Kinetics is being actively researched in the USA. However work on studying the diffusion behavior in graphite is necessary so that the desired modeling capability can be developed.

2. Accelerated development of ASME code for graphite core components. Adequacy of ASME codes needs to be evaluated to enable core design. This effort needs to be augmented and accelerated.

3. Graphite tribological behavior in helium. The current plan does not address this issue properly. An assessment of dust formation, possibility of pebbles sticking and of blocking of channels should be derived from detailed studies on the tribological behavior of graphite, as a function of environment, pressure, temperature and dose.

4. Oxidative reactivity of graphite dust powder compared to graphite blocks. Graphite dust can potentially become a carrier for fission products. Models need to be developed on this area, and the appropriate experimental data to support the development of such models need to be generated.

5. Enhanced analytical modeling and predictive capability for irradiation induced dimensional change and creep. These analytical predictive models, supported by phenomenological data and relationships, provide a key input to core behavioral models used to determine core structural degradation and end of core component life. The current effort for this model development should be augmented and accelerated.

6. An accepted fracture criteria for nuclear graphite. More work is needed in the US to develop whole core models that integrate all the inputs and predict local stresses of

graphite components. Whole core models are critical to enable an agreement on acceptable fracture criteria for graphite.

7. An accepted in-service inspection method for graphite core component

8. Overall graphite degradation (prediction) model (GDM).

9. A graphite core stress analysis method. Although this depends largely on the design, a need exists to develop general guidelines and specific finite element behavioral codes to map the core stress as a function of reactor operation.

10. The potential for stored energy release in irradiated graphite exposed to high temperatures during reactor accidents (i.e., high temperature energy release).

11. Knowledge needs for graphite decommissioning: Considerable research is required to address handling and disposal issues of discharged graphite. PMR block reuse should be considered. In addition some of the reactor vendors are considering the option of recycling graphite for nuclear use, which may become a design requirement. However, significant research effort is required on this are before graphite could be recycled.

9. References

In supporting this work, the following documents were reviewed:

[1] NUREG/CR-6944, Vol. 5; ORNL/TM-2007/147, Vol. 5 “Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRT), Volume 5: Graphite PIRTs”, 2007

[2] INL/EXT-07-13165, “Graphite Technology Development Plan”.

[3] ORNL/TM-2008/129, “Generation IV Reactors Integrated Materials Technology Program Plan: Focus on Very High Temperature Reactor Materials”.

[4] ORNL/TM-2007/153, “NGNP Graphite Selection and Acquisition Strategy”.

[5] ORNL-GEN4/LTR-06-019, “Experimental Plan and final Design Report for HFIR High Temperature Graphite Irradiation Capsules HTV-1 and -2”.