April 14, 2009

MEMORANDUM TO: Timothy R. Lupold, Chief
Corrosion and Metallurgy Branch
Division of Engineering
Office of Nuclear Regulatory Research

FROM: Makuteswara Srinivasan, Senior Materials Engineer **/RA D. Dunn for/**
Corrosion and Metallurgy Branch
Division of Engineering
Office of Nuclear Regulatory Research

SUBJECT: SUMMARY OF THE ORNL/NRC PUBLIC WORKSHOP ON NUCLEAR GRAPHITE RESEARCH, MARCH 16 – 18, 2009, ROCKVILLE, MARYLAND.

During March 16 through 18, 2009, the Oak Ridge National Laboratory (ORNL) conducted a category 3 public workshop to assess the current status of worldwide nuclear graphite research with a panel of international experts in the subject area. This workshop was sponsored by the NRC under a Office of Nuclear Regulatory Research (RES) Contract, JCN: N6640. The workshop was held at the Legacy Hotel and Meeting Center, 1775 Rockville Pike, Rockville, MD 20852. Enclosure 1 lists the workshop attendees.

A public meeting notice was issued on March 3, 2009, and was posted on the NRC's external (public) web page (ADAMS Accession No. ML090620211). The notice included the workshop agenda (ML090620241), which was also available as a handout at the meeting. The purpose of this workshop was 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 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.

A summary of the workshop presentations and the deliberations of the expert panel is provided in Enclosure 2. As a part of the milestone deliverable of the RES contract, ORNL will submit a draft report of the workshop, including the panel recommendations for NRC's future nuclear graphite research during late May 2009. The NRC staff intends to review this draft report during June 2009. The final report is expected to be published and made available to the public during July 2009.

Enclosures: As stated

CONTACT: Makuteswara Srinivasan, RES/DE
301-251-7630 Makuteswara.Srinivasan@nrc.gov

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ADAMS ACCESSION NO.: ML090850118

OFFICE	RES/DE/CMB	SUNSI Review
NAME	MSrinivasan (D. Dunn for)	M. Srinivasan (D. Dunn for)
DATE	04/14/09	04/14/09

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ORNL/NRC WORKSHOP ON NUCLEAR GRAPHITE RESEARCH
Legacy Hotel, Rockville, Maryland.
March 16 – 18, 2009

Attendees Sign-up for Monday, March 16, 2009

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1	Dr. Makuteswara Srinivasan	US NRC/RES	301-251-7630, makuteswara.srinivasan@nrc.gov
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5	Scott Penfield	Technology Insights	615-735-2666 penfield@ti-sd.com
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8	Timothy D. Burchell	ORNL	865 576 8595 burchelltd@ornl.gov
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Enclosure 1

ORNL/NRC WORKSHOP ON NUCLEAR GRAPHITE RESEARCH
Legacy Hotel, Rockville, Maryland.
March 16 – 18, 2009

Attendees Sign-up for Monday, March 16, 2009

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ORNL/NRC WORKSHOP ON NUCLEAR GRAPHITE RESEARCH
Legacy Hotel, Rockville, Maryland.
March 16 – 18, 2009

Attendees Sign-up for Wednesday, March 18, 2009

	Name	Organization	Telephone/email
1	Makuteswara Srinivasan	USNRC/RES	301-251-7630, makuteswara.srinivasan@nrc.gov
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19			
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Page 1 of 2

ORNL/NRC WORKSHOP ON NUCLEAR GRAPHITE RESEARCH
Legacy Hotel, Rockville, Maryland.
March 16 – 18, 2009

Attendees Sign-up for Tuesday, March 17, 2009

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A Summary of the
ORNL/NRC Workshop on Nuclear Graphite Research, March 2009

Overall Summary:

During March 16 through 18, 2009, the Oak Ridge National Laboratory (ORNL) conducted a category 3 public workshop 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 a Office of Nuclear Regulatory Research (RES) Contract, JCN: N6640. The venue for workshop was the Legacy Hotel and Meeting Center, 1775 Rockville Pike, Rockville, MD 20852. Enclosure 1 lists the workshop attendees.

We issued a public meeting notice on March 3, 2009, and posted the notice on the NRC's external (public) web page (ADAMS Accession No. ML090620211). The notice included the workshop agenda (ML090620241), which was also available as a handout at the meeting. The purpose of this workshop was 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 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, South Africa, and regulatory staff from U.K.'s Nuclear Installations Inspectorate and from National Nuclear Regulator (NNR) from South Africa. The panel members were:

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, NGNT 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 (RSA).
7. Mr. Schalk Doms, Senior Regulatory Officer, PBMR Programme, National Nuclear Regulator (NNR) – RSA.
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 – United Kingdom.
10. Mr. Graham Heys, HM Principal Inspector, (Nuclear Installations), HM Nuclear Installations Inspectorate, Health and Safety Executive – United Kingdom.

Dr. Nidia Gallego, Research Scientist, Carbon Materials Technology Group, ORNL, is the principal investigator of the RES contract and coordinated the workshop arrangements.
Dr. Makuteswara Srinivasan, Senior Materials Engineer, NRC, acted as the overall facilitator of this workshop. A photograph of the panel members is shown in the Attachment.

During the first day of the workshop, the panel members presented information on 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 the Nuclear Regulatory and previously conducted by National Laboratories;
(2) the status of worldwide research on nuclear graphite and its applicability to the design consideration of the Next Generation Nuclear Plant (NGNP);
(3) the international regulatory practices in licensing and regulating graphite moderated gas cooled reactors; and,
(4) the results of the NRC-DOE Phenomenon Identification and Ranking Table (PIRT) exercise on nuclear graphite, which was conducted in 2007.

The second day of the workshop began with a presentation by a NRC staff on some of the challenges in assessing the structural integrity of graphite components, which was followed by a presentation by an 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, which will be submitted to the NRC Project Manager by end of May 2009.

We have made all presentations made at this workshop publicly available on the NRC's public web page through ADAMS.

Brief Summary of the Panel presentations:

Dr. B. Sheron, RES Director opened the workshop welcoming domestic and foreign experts and other attendees, and provided an overview of RES mission, and involvement in past NRC-licensed graphite reactors (ML090790850). Dr. M. Srinivasan of RES provided a short history of some of the past nuclear graphite research by the NRC (ML090640480). During the 1970s, the NRC conducted research at Franklin Institute Research Laboratories, Philadelphia, PA, on design rules for nuclear graphite core components. During 1970s, under a RES contract, Brookhaven National Laboratory evaluated possible combustion hazards associated with a high temperature gas cooled reactor. They studied the reactions between graphite and steam or air which produce the combustible gases H_2 and/or CO. They determined that, when mixed with air in the prestressed concrete reactor vessel (PCRV), flammable mixtures may be formed. During 1980s, following the fire in the Chernobyl reactor, under RES sponsorship, Brookhaven National Laboratory evaluated the potential for graphite fire in NRC-licensed and regulated U.S. research reactors and Ft. St. Vrain reactor. They concluded that no credible potential for a graphite burning accident existed in the analyzed reactors. During 2002 – 2003, the NRC initiated research at ORNL to form technical committees in the American Society of Mechanical Engineers (ASME) and the American Society for Testing Materials (ASTM) to enable the development of design, inspection, and operation codes and standards for HTGR graphite core components. In 2007, the NRC conducted a graphite phenomenon identification and ranking table (PIRT) exercise, in cooperation with DOE, for guidance on prioritization of NRC's graphite research. Presently (2009), the NRC has sponsored research at ORNL to conduct a graphite workshop with a panel of international experts, and identify potential research areas that the NRC may conduct in the future, which augments DOE and NGNP applicant's research and fills the technical gaps to license a HTGR NGNP.

Dr. N. Gallego (ORNL) provided an overview of the objectives of the workshop, highlighting the background documents, which have been previously provided to the expert panel for study (ML090850106). These included: (1) NUREG/CR-6944, "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRT), Volume 5: Graphite PIRTs" (ML081140463); (2) INL/EXT-07-13165, "Graphite Technology Development Plan"[1]; (3) ORNL/TM-2007/153, "NGNP Graphite Selection and Acquisition Strategy"[2]; (4) ORNL-GEN4/LTR-06-019, "Experimental Plan and final Design Report for HFIR High Temperature Graphite Irradiation Capsules HTV-1 and -2"[3]; and (5) ORNL/TM-2008/129, "Generation IV Reactors Integrated Materials Technology Program Plan: Focus on Very High Temperature Reactor Materials"[4]. Ahead of the workshop, Dr. Gallego also had suggested the following technical themes to the panel for discussion:

– 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

In his presentation (ML090850110) Dr. T. Burchell (ORNL) discussed the graphite phenomena identification and ranking table (PIRT) review, which was conducted in 2007. Dr. Burchell

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[1] This document is available to the public via http://www.inl.gov/technicalpublications/Documents/3867694.pdf.
[2] This document is available to the public via http://www.osti.gov/bridge/servlets/purl/921767-EDMpxk/921767.pdf
[3] This document is available to the public by contacting the ORNL library by email, library@ornl.gov,
[4] This document is available to the public via http://www.ornl.gov/info/reports/2008/3445605786584.pdf.

provided the panel with background information and answered questions related to graphite PIRT, which is the basis for comparing DOE-proposed graphite research to identify technical gaps. He provided detailed information on the five (5) graphite behavior phenomena, which were identified by the PIRT panel to be of high importance with low knowledge. These were:

1. Irradiation-induced creep (irradiation-induced dimensional change under stress);
2. Irradiation-induced change in the coefficient of thermal expansion (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; and,
5. Blockage of coolant channel in reactivity control block due to graphite failure and/or graphite spalling.

In addition, Dr. Burchell discussed nine (9) phenomena, which were ranked of high importance with medium knowledge, and two (2) phenomena, which were ranked of medium importance with low knowledge. Research addressing all these PIRT-identified graphite behavior phenomena will potentially enable informed decisions related to NGNP design certification review and NRC staff regulatory technical review of topical reports submitted by potential applicants, ahead of design certification application.

Dr. W. Windes, INL, provided an overview of the current status of the NGNP concept design (ML090850115). Within the reactor core, in terms of service severity, the graphite fuel block is expected to experience a temperature of approximately 1200 °C during normal operations and approximately 1400 °C during off-normal operations. Furthermore, the expected cumulative dose, for this component is approximately 0.8 dpa per year. The least severe exposure would be experienced by outer graphite reflector blocks with an approximate temperature of 800 °C during normal operations and approximately 1100 °C during off-normal operations. These components are expected to accumulate dose levels of approximately 1 dpa per year. He mentioned that a decision on the specific type of graphite has not yet been made. However, the INL has procured graphite from several vendors, and are characterizing their baseline properties using ASTM standards, as applicable. Dr. Windes also provided the program plan and schedule for Advanced Test Reactor (ATR) graphite creep experiments (AGC) under irradiation to various dose ranges at temperatures from 600 °C to 1,200 °C. Irradiation at lower temperatures and post irradiation examination (PIE) will precede the 1,200 °C irradiation tests, which are currently scheduled to begin during 2015. All the irradiation are planned to be completed by 2019 and the PIE completed by 2020. At INL, efforts are also underway for modeling the whole core graphite behavior.

Professor B. Marsden of the University of Manchester presented an overview (ML090850121) of the nuclear graphite research being conducted at the United Kingdom universities, and the European nuclear graphite research initiatives. The European nuclear graphite research initiative consists of thirty three (33) partners or organizations representing Belgium, Czech Republic, France, Germany, Italy, The Netherlands, Slovak Republic, Spain, Switzerland, and the U.K. The University of Manchester (UK), NRG (The Netherlands), AMEC (UK), FZJ (Germany), Framatome ANP (France), CEA (France), SGL Carbon (Germany), and UCAR

(France) have partnered in the RAPHAEL graphite irradiation program, with the irradiation conducted at Petten, Netherlands. The NRI Rez (Czech Republic), Paul Scherer Institut (Switzerland) participate as observers. Basic research by Professor Marsden and his colleagues supports the British Energy (the operator of the AGRs) and Magnox reactors and the U.K.'s NII in establishing a fundamental understanding of the behavior of graphite in these reactors. He presented information on some example nuclear graphite research in the U.K., which included:

(1) the prediction of brick cracking rates using statistical analysis;
(2) the development of semi-empirical models for use in graphite component structural integrity assessments including irradiation creep;
(3) the statistical analysis of reactor installed and trepanned data of component dimensional change, weight loss and property changes for use in structural integrity assessments;
(4) whole core modeling;
(5) refueling trace monitoring;
(6) seismic modeling; and,
(7) component life prediction (numerical finite element modeling).

Professor Marsden also discussed the 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.

Mr. M. Mitchell (PBMR (Pty) Ltd, RSA) presented an overview (ML090850127) of his company's graphite research supporting the design activities as a vendor. Their research targets addressing technical issues to meet:

(a) South African regulatory requirements;
(b) graphite life-cycle model; and,
(c) graphite reactor safety case model.

The PBMR has recently completed an effort within the NGNP Project to reconcile the results of the October 2007 PIRTS to U.S. NGNP requirements. The PBMR is committed to the establishment of international standards and actively participates in ASTM and ASME codes and standards committees. The PBMR also integrates its graphite research with international programs, such as the Gen IV and the VHTR research.

Dr. M. Eto (Toyo Tanso, Japan), in collaboration with Dr. T. Shibata of the JAEA, presented an overview (ML090850141) of the Japanese graphite research supporting the high temperature test reactor (HTTR). The lessons learned from the HTTR operations are to be used for the future construction of a VHTR. The maximum fluence level for the VHTR will be about 4 times that for the HTTR at approximately 6×10^{25} n/m^2. The graphite grade IG-110 is considered to be mature and will be the primary candidate for the VHTR. However, an improved graphite grade, IG-430, is being developed and studied, which exhibits approximately 20 -40 % greater strength than that for the IG-110 grade. At the JAEA, micro-hardness indentation testing is being developed to determine residual stresses in irradiated graphite. As an indicator of the extent of oxidation, ultrasonic velocity measurements are being made, with excellent Arrhenius (reaction kinetics) behavior with weight loss due to oxidation. The 3-dimensional x-ray computed tomography (CT) investigations are being conducted with IG-100 and PGX graphites to evaluate the effects of irradiation on graphite microstructure. In the HTTR, specimens have also been installed to enable inservice inspection analysis of dimensional changes, strength, Young's modulus, irradiation-induced changes in CTE, and surface oxidation rate. Monitoring by TV camera is also being practiced. The JAEA and Toyo Tanso are also collaborating with the Peten and the INL ATR irradiation programs, IAEA CRP, and Gen IV International Forum (GIF) in conducting graphite research.

Mr. G. Heys of the U.K. NII provided an overview of the U.K. nuclear graphite research from a regulatory perspective (ML090850151). Presently, about 20% of U.K.'s electrical power is nuclear consisting of 14 advanced gas cooled reactors (AGR), 4 Magnox reactors, and 1 PWR. Reactor. Other than the PWR, the reactors are graphite-moderated. The basis for NII regulation and licensing system resides in the Health and Safety Work Act (1974) and the Nuclear Installations Act (1965), as amended. The NII has guidance documents, which provide safety assessment principles (SAP) to assess graphite safety cases. There is also a separate technical assessment guide "Graphite Reactor Cores" for safety evaluation of AGR and Magnox graphite cores.

For the graphite-moderated reactors in the U.K., the safety functions of the graphite core include:

(a) Shutdown and reactivity control post shutdown;
(b) Fuel cooling; and,
(c) Fuel integrity and unimpaired refueling.

Thus far, NII has encountered the following challenges to the graphite core safety functions:

(1) Observations of unpredicted graphite moderator brick cracking in AGRs, pre stress reversal;
(2) Predictions of cracking in AGRs, post stress reversal; and,
(3) High radiolytic weight loss.

Limiting technical issues have been determined to be highly complex involving onerous graphite duty to greater than 20 dpa exposure, data scatter and high uncertainty, limited data validation, empirical models with lack of fundamental behavioral understanding, complex stress calculations involving many inputs, interactions and iterative loops. These issues have resulted in limited predictability or non-predictability of the initial and current state of graphite core bricks. The NII has initiated several fronts to address these issues:

(a) Develop revised assessment guidance;
(b) Secure maintenance and improvement of graphite core safety cases;
(c) Develop credible and cost effective graphite research program; and,
(d) Establish graphite technical advisory committee of experts from several science and engineering disciplines to provide independent advice.

Besides, the NII also has several nuclear research arrangements with the licensee and independent research conducted at various Universities and commercial research support organizations. Such research is targeted to provide data and information to address the previously mentioned technical challenges for graphite.

Mr. S. Doms (NNR,RSA) provided information on the regulatory practices of the RSA regulator, NNR, particularly with respect to the PBMR (ML090850181). The licensing of a nuclear facility in RSA is governed by the the NNR Act (Act No 47 of 1999). 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. A multi - staged licensing process has been adopted by the NNR, which includes the following major licensing stages:

(1) Acceptance of concept safety case;
(2) Site preparation, construction and manufacturing phase;
(3) Fuel on site, fuel loading, testing and commissioning;
(4) Plant operation; and,
(5) Decommissioning.

Graphite and other ceramic core structure requirements are contained in a licensing document, LD-1097, and include qualification requirements, defining safety functions, qualification of material and tests, and the qualification of structures and assembly. Current NNR research includes the development of an independent 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.

Dr. M. Eto (Toyo Tanso, Japan), in collaboration with Dr. T. Shibata of the JAEA presented an overview (ML090850190) of the Japanese regulatory perspective for high temperature gas cooled reactor. Basically, the assessments are based on the generation of statistically significant database to support design of the graphite core components, the use of a deterministic structural analysis design code that allows no cracking or “unexpected” oxidation, the use of inspection standards and QC/QA management during manufacturing and construction of graphite components, and inservice inspection. During the HTTR operation, regional temperature distribution and the amount of regional fission product are monitored. During outage, inspection by TV camera is performed and surveillance test coupons are tested. Proof tests, which consist of bottom structure seismic test, core components seismic test, support post bucking test, dowel/socket fracture test, and key/keyway fracture test are also performed. A special committee at the Atomic Energy Society of Japan (AESJ) has been established to formulate technical criteria for VHTR graphite components, based on HTTR standards. Their report and other report on irradiation data and analysis are expected to be published during 2009.

The first day ended with a presentation by Dr. S. Rubin of the NRC on the regulatory research perspectives related to the NGNP V/HTGR licensing (ML090690834). He briefly discussed the 2005 Energy Policy Act and NRC’s role in licensing and regulating the NGNP. He discussed how the graphite behavior and the understanding of graphite modeling and predictive tools are important in contributing to the HTGR accident analysis safety evaluation model for several technical arenas. The NRC’s NGNP R&D infrastructure needs assessment has identified key technical, safety, safety research, and policy issues. To summarize, the NRC NGNP V/HTGR R&D will continue to focus on NGNP V/HTGR COL technical review needs and support the application review schedule. It will utilize cooperative research agreements to generate data and predictive tools where possible. The R&D will also be consistent with the NRC’s “Role of Research” and with completed PIRT results.

The second day of the workshop began with a presentation by Dr. M. Srinivasan of the NRC on some of the challenges involved in the safety evaluation of the NGNP HTGR graphite components (ML090640475). He explained the many aspects of various models and data that provide estimates of the probability of (functional) failure of graphite components, which is used as an input and contribution to the overall core performance risk measures for a HTGR. The various models are:

(a) the graphite material degradation model, based on limited material test reactor data and operational experience;
(b) the extrapolation and translation of such model to graphite components via a scale-up model;
(c) the graphite component structural integrity model, based on finite element stress analysis and fracture analysis (behavior) model; and,
(d) the graphite component inspection model.

He discussed the limitations of each of these models and their interactions, data and model uncertainties, lack of verification and validation of the data and the models, lack of operational experience, variations in reactor operations, and decisions regarding the definition of graphite component failure on the overall estimate of the range of failure probabilities. He cautioned that: (1) when initial risk measure is very low there may be a tendency to ignore potential model weaknesses (incompleteness); and, (2) the robustness of results is dependent on the quality, quantity, and confidence in the information supplied. A major element that influences the robustness of the estimate of risk measures is the adequacy of inspections and the confirmation of model predictions from inspection data. He outlined the need and the challenges involved in developing consensus ASME graphite codes and standards, which would be applicable for regulating NGNP. These are:

(1) a component failure criteria, graded on safety significance;
(2) a component performance criteria;
(3) component inspection criteria;
(4) core surveillance requirements, core restraint monitoring, core support structure monitoring, and testing protocols;
(5) acceptance/replacement criteria for flawed graphite component In service; and,
(6) graphite component degradation management program and procedure to assess Its efficacy.

Dr. T. Burchell (ORNL) provided a template for the objectives of the panel deliberations, which will follow and the expected outcome as a set of recommendations to NRC graphite research areas which NRC could conduct in the future. These research areas would be those which are either not already covered by the DOE graphite research plan or deemed to be insufficient may provide adequate information to provide technical basis for future NRC guidance documents related to design certification review of the NGNP. Dr. Burchell also presented (ML090850214) preliminary results from a NRC contract research in which the ONRL has recently completed a comparison of the DOE-NRC graphite PIRT results with the DOE graphite research plan and identified weaknesses in the DOE research plan, and technical gaps which would need to be addressed by future research. These were classified into three categories:

(1) normal plant operations;
(2) anticipated operational occurrences, and,
(3) design basis accidents.

For each of the PIRT phenomena that were ranked of high and medium importance. Additionally, the ORNL also recommended the following research areas:

(1) Capability to model graphite oxidation;
(2) Accelerated development of ASME codes and standards for graphite core components;
(3) Tribological behavior of graphite in (NGNP) helium environment;
(4) Study of oxidation of graphite dust;
(5) Enhanced analytical modeling and predictive capability for irradiation induced dimensional change and creep;
(6) Development of a consensus fracture criteria for graphite;
(7) Development of consensus inservice inspection methods for graphite core component;
(8) Development of a comprehensive graphite degradation prediction model;
(9) Development of graphite core stress analysis procedure;
(10) Study of irradiation-induced stored energy release at higher annealing (or accident) temperatures; and,
(11) Preliminary efforts to gather knowledge on graphite decommissioning.

Brief Summary of Panel Deliberations:

The panel then began their deliberations on the topics which have been suggested prior to the workshop.

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

The INL staff described their plans on qualifying nuclear grade graphites at vendor sites and vendors' plan to follow NQA-1 requirements. However, in their opinion, the procedures are still being in the formative stages because of the newness of the NQA-1 requirements to the vendor, who have been following ISO standards in their manufacturing and quality assurance processes. The INL staff mentioned that active involvement by the NRC staff in providing guidance on nuclear Part 50, Appendix B requirements for graphite would be useful.

The panel observed that the selection of manufactured graphite properties for qualification should have nexus with expected graphite performance and implications to irradiated properties. This is not straightforward and the regulatory staff should be actively involved in the early definitions and requirements for graphite qualification. The qualification should be based on sound science and should account for the irradiation behavior.

The traceability should involve individual core component and its location in the reactor. The panel was of the opinion that the NRC should develop, via long-term research, scientific knowledge of the microstructural features of graphite which govern the irradiation behavior. The NRC should develop and foster international cooperative research in this area.

As a regulator, the NRC may need to consider combining safety culture with the QA requirements. The NRC staff should participate when the INL staff is conducting NQA assessments with their graphite vendor.

– Requirements for core behavioral models
  - Irradiation properties
  - Models for fundamental understanding for structural integrity analysis
  - Handling of data and model uncertainties

The panel spent a considerable amount of time discussing the issue of core behavioral model. The major knowledge in this area is from the U.K. graphite reactor experience. The U.K. reactor experience has indicated considerable caution required in establishing and understanding the graphite core behavior, especially at high doses and at later stages of reactor life. Deviations from expected behavior have been encountered, due to a variety of reasons including: (a) assumptions and limitations in extrapolating MTR irradiation data to reactor component for conditions outside of the data range; (b) lack of a fundamental understanding of the irradiation behavior of graphite to predict the graphite component in the reactor; and (c) extrapolating and understanding the data and model uncertainties to whole core performance.

Traceability of the material test reactor tested graphite and its correspondence with the graphite being used for the HTGR is needed to extrapolate the test data to reactor graphite performance. The panel observed that most of the irradiated graphite properties are usually determined at room temperature, either in air or in an inert environment. The data should be generated at temperatures of NGNP operation and environment, since the properties may be governed by thermally activated phenomena or processes; and, the damage processes are temperature-sensitive. The use of data which were obtained at room temperature and in an environment other than the HTGR environment could be misleading.

Appropriate methods to extrapolate the small specimen data to large component are not generally available. There exists a gap in the verification and validation of the methods used for predictions. The extrapolation of data into regions of unknown importance could potentially be quite erroneous.

Irradiation can change the notch sensitivity and needs further research. The panel expected that one of the pre-licensing issues would be the validation of the probabilistic graphite design method that may be used since many components lead to many interactions.

Irradiated creep is an issue of considerable importance, and this has not been addressed adequately in prior research. The panel was of the opinion that NRC should foster research that advances the fundamental understanding of basic mechanisms via long term research.

The validation of the behavior models could be augmented by the PIE results on discharged components from decommissioned reactors. However, this is quite an expensive research and requires major commitment of resources in personnel and equipment. A good avenue will be international collaboration and sharing.

– Oxidation of graphite by coolant impurities

The panel was of the opinion that more research needs to be conducted to obtain oxidation data on NGNP graphite in the HTGR environment containing potential impurities in the coolant. Oxidation data for irradiated graphite is currently not available for even older graphites. In order to understand the effects of irradiation-induced microstructural changes on the kinetics of oxidation, selective oxidation experiments are warranted on irradiated NGNP graphite. The panel was of the opinion that, more likely, oxidation by radiolysis, would not be a consideration for the HTGR environment.

The NRC should review the current knowledge regarding oxidation effects relevant to HTGR, and publish a white paper.

– Status of codes and standard development / future challenges
- Design and construction code (Section III)

The panel was of the opinion that the current progress for the development of the ASME codes and standards, though good, needs to be accelerated to meet the NGNP schedule requirements. However, challenges remain since the important irradiation data for the NGNP graphites, which are needed for the code will not be available until 2014 – 2015. The panel recommended that the NRC staff participate in the ASME code meetings more aggressively and actively, providing more guidance and review of early draft documents. Benchmarking of the code with cases, verification and validation of the codes were considered to be significant and were identified as potential shortcomings, if they were not conducted independently by the regulator.

– Stress analysis

The panel considered independent capability to perform the stress analysis of graphite core component may not require special research. However, the NRC staff should be knowledgeable on the various components that contribute to and constitute a rigorous finite element stress analysis for graphite undergoing irradiation at high temperatures and high fluence. Particularly, the staff should enhance their knowledge and a fundamental understanding to the changes in CTE, strength, dimensions, thermal conductivity, Young's modulus, Poisson's ratio, and irradiation creep as a function of dose and temperature because the phenomenological relationships for these properties form input into the reiterative stress analysis models. The staff should be aware of the assumptions involved in the extrapolation of the small specimen data to large components, and the limitations involved in the extrapolation of the data into operational ranges beyond the experimental data. The panel identified stress analysis as not an immediate issue, but one which should be gradually developed over some time.

– Adequacy of margins
- In-service inspection (Section XI)

The panel arrived at the conclusion that more work is needed in this area, especially to define the criteria and the terms. Such efforts are being carried out currently by potential HTGR vendors in collaboration with the ASME code subcommittee involved in developing codes and standards for graphite core design. Again, the panel urged that the NRC staff should commit more time and effort to attend and actively participate in the ASME code committee meetings.

– Tribology and oxidation leading to graphite dust

The panel discussed this issue at length. Some of the panel considered this as a non-issue for the prismatic reactor; some of the members considered this to be a non-issue even for the pebble bed design. But, the panel agreed that, because graphite dust acts as the carrier of the FP, this is an area that requires further research. The panel also found that the DOE research plan is lacking in graphite tribology research to provide a fundamental knowledge understanding of the origin of dust generation, the dust characteristics, and its interaction with the materials it encounters to either adsorb, adhere and to lift-off. Such information is needed to provide information for decision making on the importance of this phenomenon. The panel was of the opinion that the NRC should initiate research in this area, including a preliminary literature survey of the prior work.

– Air ingress and water ingress (accident)
  • Safe shutdown and safe cool down

The panel discussed this subject area in good detail and was of the opinion that much work has already been done in the past, both by the NRC and others. It would, however, be worthwhile to conduct a literature review and publish a white paper on this issue. The panel also observed that there exist insufficient data on the kinetics of oxidation and a general lack of understanding of the diffusion of the controlling species and the mass flow behavior. The panel recognized this issue as a "high-profile" event and that the NRC needs to stay on the top of this topic, and be able to perform independent analysis.

– Defining end of core-component life (criteria and safety margins)

The subject of what constitutes the end of life of the graphite core component life was discussed considerably by the panel members. It was considered that the functionality of the graphite core was considered to be of more significance than the existence of a few cracks in the graphite component. The "tolerability" of the graphite component to the existence of cracks and minor dimensional changes needs to be examined and understood. The "tolerability" needs to be coupled with the "functionality." Three main functions of the core were identified as the maintenance of:

(a) adequate cooling of the graphite core;
(b) reactivity control and the ability to shut down; and,
(c) maintenance of the core geometry.

Therefore, in addition to understanding the effects of cracks in the graphite core, dimensional changes and irradiation creep are also important to component life and establishing safety margins for graphite performance and the overall safe reactor operation.

The panel was of the opinion that there exists a need to establish a science-based framework to define margins, which will support risk-based assessments. It is also important to properly address the uncertainty and the complexity of the issues to determine the adequacy o the margins.

No particular research was identified by the panel to address this issue. However, the panel suggested that the NRC staff must be aware of the complexity of the issue and provide input to the question of how to define the adequacy of the data and margins.

– Decommissioning and disposal

The panel identified decommissioning and disposal of irradiated graphite core as a long term issue and recommended that NRC monitor worldwide research in this area, rather than initiating any new work in this area. The current NGNP irradiation program does not cover the re-use of graphite, for which higher dose experiments would be needed. It was also mentioned that there should be some interest in the longer term graphite irradiation data so that the life of the graphite reflector blocks could be extended.

– Other themes as suggested by panel members

The panel did not suggest any other theme for further work by the NRC.

Technical gaps in the DOE Research Plan versus PIRT Results:

Summarizing the workshop, Dr. Burchell re-visited the ORNL review of the DOE graphite research plan and the PIRT results, and identified several areas for NRC's future nuclear graphite research for panel discussion. These research areas and the opinion of the expert panel on each of these areas are provided below:

1. Oxidation modeling capability: 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 modeling capability for oxidation which includes coolant impurities and develop an active capability to evaluate oxidation and build code.

2. Graphite codes and standards: The panel observed that the NRC staff attendance has not been consistent and continuous at the ASME, ASTM, and ANS meetings and recommended the active participation of the NRC staff so that early input and guidance may be provided on regulatory positions.

3. Graphite Tribology: Because of the prevalence of conflicting opinion on this subject, as evidenced by the panel discussion, the NRC needs to reach a conclusion in this area. It was recognized that the fission product (FP) transport drives this need for better data and understanding. Graphite dust is the vehicle for the transport once FP escapes from the fuel. The NRC needs to understand the transport mechanism and how the dust acts as a carrier.

4. Oxidative reactivity of graphite dust powder compared to graphite blocks: This topic is believed to have been adequately researched and would only need to have the body of research reviewed and summarized in a white paper/NUREG.

5. Improved mechanistic modeling and predictive capability for Irradiation induced dimensional change and creep: The NRC should participate in IAEA research program. Model development should be augmented and accelerated. The NEUP and IAEA programs will be examining old data and will generate some new data and develop predictive behavior models. The irradiation creep data obtained by INL will be used to validate and refine the model. The panel recommended that the NRC develop its own models independently and then validate and refine the models. The NRC may have an university or an independent laboratory develop new or evaluate existing models and refine these models, independent of the licensee.

6. An accepted fracture criteria for nuclear graphite: The NRC staff should participate in the developments to establish graphite fracture criteria, which defines what constitutes a failure. The NRC staff should actively participate in the activities of the code committee involved in the development of stress limits.

7. The NRC staff should be actively involved early on in developing general guidelines and specific finite element behavior codes to map the core stress and core geometry changes as a function of reactor operation. This is an area that may lead to future confusion as terms, such as, fracture criteria may be used and never fully defined. This is the whole core model. The NRC staff should participate and provide high level guidance on the development of criteria for the whole core model. The specific failure criteria would be developed by the designer. The NRC should develop the capability to understand whole core models that integrate all inputs and predict local stresses of graphite components. Ultimately, the NRC will need to review the licensee applications to ensure the vendor models are appropriate and that the results determined are appropriate. The NRC will need an independent model to verify the results of the vendor/designer analysis. The codes that exist have not been validated to any large degree. The NRC should consider some type of in-house validation capability (i.e. instrument a simplified structure and compare results with a model).

8. An acceptable pre-service and ISI method for graphite core components inspections: The panel opined that existing methods of X-ray radiography and ultrasonic testing, which are used by graphite manufacturers, are not adequate to reliably inspect large nuclear graphite core components. The methods are also hampered by the absence of a definition for a “critical” flaw. The NRC staff should be involved in such criteria development, including requirements for surface and volume examinations, qualification of the techniques and inspector, and the codification of these in ASME Section XI, Appendix VIII, including methods to evaluate flaws. The NRC should review and identify the current limitations and identify those areas that will need additional design margin/alternate monitoring capability (potentially surveillances) to accommodate limitations in the NDE area.

9. Stored energy release: The release of stored energy, during the heat up of graphite either due to accident or deliberately to release such energy could potentially lead to undesired consequences. However, the prevailing knowledge indicates that this is not an issue for

HTGR, which operates at relatively high temperature, and there is no need to conduct research in this area. However, it may be prudent for NRC to conduct a literature search and conduct limited experiments to have a definitive result regarding the effects of the release of stored energy of irradiated graphite on heating to temperatures above that of the HTGR operation.

10. Graphite Decommissioning: The panel was of the opinion that research need not be conducted in this area at present. As specific recycling methods are proposed in the future, the NRC may need to do research. Monitoring of worldwide activities via staff participation in IAEA and other meetings would be adequate.

Overall, the ORNL-NRC graphite workshop, with international experts, provided an opportunity for technical information exchange, assess briefly the status of worldwide nuclear graphite research pertinent to the graphite PIRT, identify areas for further research to improve the knowledge and understanding of complex issues, such as irradiation creep, graphite core behavior (whole core modeling), graphite qualification requirements, inservice inspection, and the need for data on graphites which will be used for the NGNP.

The ORNL contractor will assemble the recommendations from individual panel experts and publish a report to the NRC around June 2009. This report is expected to be made available to the public during July 2009.

Attachment:
Photograph of Panel Members

Nuclear Graphite Expert Panel
ORNL-NRC Nuclear Graphite Workshop
Rockville, MD, March 16 – 18, 2009

Back Row: Mr. Schalk Doms, Dr. Robert Bratton, Dr. William Windes, Mr. Mark Mitchell, and Mr. Scott Penfield.
Front Row: Mr. Graham Heys, Professor Barry Marsden, Dr. Makuteswara Srinivasan, Dr. Nidia Gallego, Dr. Timothy Burchell, and Dr. Motokuni Eto.
Not in the picture: Dr. Robert Wichner

Attachment