INGSM-11

Eleventh International
Nuclear Graphite
Specialists' Meeting

The Eastbourne Centre
UK

12th – 15th September 2010

Technical Programme

The organising committee reserves the right to amend this programme at short notice if circumstances demand. The schedule printed here was believed to be correct at the time of going to press (August 28th 2010)

PLEASE TAKE NOTE THAT THE ORAL SESSIONS ON ***WEDNESDAY*** *START AT* ***08H40****!!*

Sunday 12th September 2010

14h00 onwards Registration

All oral sessions take place in the main Conference Hall (Level 1)

16h00 Welcome and Opening of the Conference

Launches and Programmes

Chairman: Dr. G.B. Neighbour, The University of Hull, UK

16h10 Launch of the IAEA International Knowledge Base on Nuclear Graphite
A.J. Wickham, G. Haag and L. Ralph

16h50 The Database for Irradiation-Induced Creep in Graphite
G. Haag

17h10 Introducing the EPSRC FUN-GRAF Consortium: Fundamentals of Current and Future Uses of Nuclear Graphite
M.I. Heggie and B.J. Marsden

17h30 The USA's NGNP Graphite R&D Programme
W. Windes, T. Burchell, W. Swank, J. Strizak and M. Carroll

Characterisation of Manufactured Graphite

Chairman: *Mr. Neil Salstrom, Toyo Tanso USA Inc*

17h50 NBG-17: A New Graphite Grade for HTRs and VHTRs
P.Béghein, G.Berlioux, F.Hiltmann, M.Melin and B.du Mesnildot

18h10 Characterisation of Baseline Properties in Nuclear-Grade Graphites
M.C. Carroll

18h30 Microstructural Characterisation of Next Generation Nuclear Graphites
J. Kane, C. Karthik, R. Ubic and D. P. Butt

19h00 **Welcome Reception** (*and **POSTERS***) **Seminar Rooms 1, 2, 3, 4 (Level 1)**

Posters will be on display throughout the meeting

A second 'manned' poster session is scheduled during Tuesday morning's extended coffee break

19h30 **Dinner** (The Eastbourne Centre Restaurant) **(Mezzanine Level)**

Monday 13th September 2010

Fundamental Theory and Models

Chairman: *Dr. L.L. Snead, Oak Ridge National Laboratory, TN, USA*

09h00 First-Principles Calculations on the Thermal Properties of Irradiated Graphite
G. Haffenden, M.I. Heggie and G. Sheehan

09h20 A New Methodology for Modelling the Failure Characteristics of Nuclear Graphites
G.D. Kipling and G.B. Neighbour

09h40 Progress in Developing Finite Element Models replicating Flexural Graphite Testing
R.L. Bratton and S. Duffy

10h00 Microstructural Characterisation of Irradiated Nuclear Graphite using Finite-Element Analysis
A.T. Bakenne, B.J. Marsden, A.N. Jones and G. N. Hall

10h20 Estimation of Dimensional Change Curves for AGR Moderator Graphite using a Bayesian Emulator and Inspection Data
K. McNally, G.N. Hall, G.B.Heys, N.Warren and B.J.Marsden

10h40 *Coffee/Tea* **(In front of Conference Hall and Seminar Rooms 1, 2, 3 – Level 1)**

Graphite Irradiation Behaviour

Chairman: *Dr. S-H Chi, Korea Atomic Energy Research Institute*

11h10 The Effect of Annealing on Thermal Properties, Crystalline Properties and Kinetic Parameters of Wigner Energy Release of BEPO Irradiated Nuclear Graphite
M. Lasithiotakis, B.J. Marsden and T. J. Marrow

11h30 A Review of Stored Energy Release of Irradiated Graphite
N.C. Gallego, T.D. Burchell, and M. Srinivasan

11h50 The NGNP Graphite Creep Experiment Irradiation in the ATR
S.B. Grover

12h10 Design and Results of the HFIR Creep Rabbit Capsule
L.L. Snead, T.S. Byun and Y. Katoh

12h45 **Lunch** (Restaurant)

13h45 Group Photograph - Esplanade

Standards and their Application

Chairman: *Mr. Leo Lanzel, Mersen USA*

14h00 Development of a New ASTM Guide for Measurements on Small Graphite Specimens
M.P. Metcalfe

14h20 Expansion of Irradiation Data by Interpolation and Extrapolation for Design of Graphite Components in High Temperature Gas-cooled Reactors
E.Kunimoto, T.Shibata, M.Eto, S.Shiozawa, K.Sawa, T.Maruyama and T.Oku

14h40 Strength Evaluation of Ultra-fine Grained Graphite by Miniature Equi-biaxial Flexural Test
Y. Katoh, G. Vasudevamurthy and S. Kondo

15h00 Size Effects in the Mechanical Properties of NBG-18 Graphite
T.S. Byun, G. Vasudevamurthy, W.D. Lewis, L.L. Snead and T.D. Burchell

15h30 *Coffee/Tea*

Graphite Waste Management

Chairman: *Dr. W. Windes, Idaho National Laboratory, USA*

16h00 Characterizing the Origin, Distribution and Chemical Form of ^{14}C in Irradiated Graphite
M.L. Dunzik-Gougar, S. McCrory, J. Cleaver and J. Fachinger

16h20 Validation of Nuclear-Physics Calculations against Measured Radionuclide Fingerprints of Oldbury and Wylfa Magnox Graphite Samples
R. Mills and Z. Riaz

16h40 On the Formation and Behaviour of ^{14}C in Irradiated Graphite
A.J. Wickham

17h00 Characterisation and Leaching of British Experimental Pile Zero (BEPO) Nuclear Graphite
L. McDermott, A. N. Jones, B. J. Marsden and T. J. Marrow

17h20 Potential Options for the Management of Radioactive Graphite Wastes
E.J. Harvey, E.M. Michie and S.M. Wickham

17h40 Carbowaste - Work Package 5 - Graphite Recycle
J. Goodwin

18h00 Progress Update on Segregation and Retrieval of Irradiated Graphite
M. Grave (Doosan Babcock) and A. Banford (National Nuclear Laboratory, UK)

19h00 **Dinner** (Restaurant)

Tuesday 14th September 2010

Characterisation and Properties

Chairman: Mr. L. Batty, GrafTech International, USA

09h00 Pre-Peak Load Displacement Characteristics of PGA Reactor Core Graphite
S. Nakhodchi, P. Heard, A. Hodgkins, R. Moskovic, M.R. Wootton ,D.J. Smith , and P.E.J. Flewitt

09h20 The Effects of Equi-biaxial Stress on Crack Nucleation and Propagation in Nuclear Graphite
M. Mostafavi and T.J. Marrow

09h40 Pre-Irradiation Characterisation of AGC-2 Specimens
W. D. Swank, J. R. Lord, D. Rohrbaugh and W. E. Windes

10h00 The Simulation of Neutron Damage in Nuclear Graphite using Ion Irradiation
M. Lasithiotakis, B. J. Marsden and T.J. Marrow

10h20 *Coffee/Tea* and ***POSTERS*** – ***Seminar Rooms 1,2,3***
Posters will be on display throughout the meeting

Graphite Oxidation

Chairman: Dr. J.A. Vreeling, NRG Petten, The Netherlands

11h00 Determining Pore Structures in Nuclear Graphite
M. Figueroa and W. Windes

11h20 The Modelling of Graphite Oxidation Behaviour under Reactor Normal Operating Conditions
S.Y. Yu, X.L. Yu, L. Brissonneau and C. Bourdeloie

11h40 Probabilistic Predictions of Graphite Weight Loss
I. Robertson, M.R. Bradford and P.C. Matthews

12h00 Pore Size Change in Depth and Oxidation Reaction of Selected Nuclear Graphites in 873-1273K Air
S.H. Chi, G.C. Kim, J.H. Jang, H.J. Kim and E.S. Kim

12h30 **Lunch** (Restaurant)

13h15 Departure for Dungeness 'B' NPP (*nominated delegates only*)
13h45 Departure of remaining guests for Leeds Castle

18h30 Drinks Reception (*Fairfax Hall Terrace, Leeds Castle*)
19h00 Conference Banquet (Leeds Castle: entertainment – Jeanette Cordery, harp)

Wednesday 15th September 2010

Reactor Experiments, Inspection and Monitoring

Chairman: *Prof. Yu SuYuan, INET, Tsinghua University, Beijing, China*

08h40 Eddy Current Technique Development: Initial Survey Results from AGR Reactor Cores
M. Brown, A Greenaway and J.Williams, EdF Energy (British Energy), UK

09h00 Update on the Blackstone AGR Graphite Irradiation Programme
O. Wouters, B.P. Jonker, P. Matthews and M. Joyce

09h20 Irradiation of HTR Graphites in HFR Petten
A. Vreeling

09h40 The CHANSELA Software: a Tool to Aid the Selection of Channels for Inspection at Reactor Outages
P. R. Maul, P. C. Robinson and C.E. Watson

10h00 An Integrated Approach to Inspection Data and Material Properties
M.R. Bradford, L. Chidwick, A. Price and M. Joyce

10h20 Evidence Supporting the Use of MCBEND Dosimetry Predictions within AGR Core Condition Assessments
D.A. Thornton

10h40 *Coffee/Tea*

Stress and Component Analysis (i)

Chairman: *Mr. Marc Melin, SGL Carbon, Germany*

11h10 Anisotropy of IG110 and NBG18 under Compressive Stresses
S.Y. Yu, X.W. Zhou and H.T. Wang

11h30 Models of Coefficient of Thermal Expansion for Gilsocarbon Graphites Irradiated in Inert and Oxidising Environments
E.D. Eason, G.N. Hall, B.J. Marsden and G.B. Heys

11h50 The Biaxial Strength of NBG-18 Graphite
T.D. Burchell, R. Battiste and J. Strizak

12h10 Nuclear Graphite Reliability Analysis Investigation and Software Development Space-Act Agreement
N.N. Nemeth, J.P. Gyekenyesi, E.H. Baker and R.L. Bratton

12h30 Comparison of the Probabilistic and Deterministic Methods for the Stress Evaluation of Nuclear Graphite
H.T. Wang, Z.S. Zhang and L.B. Sun

13h00 **Lunch** (Restaurant)

Stress and Component Analysis (ii)

Chairman: *Dr. T.D. Burchell, Oak Ridge National Laboratory, USA*

14h00 Progress in Providing Improved Input Data to Graphite Brick Behaviour Modelling
P. Rogers

14h20 Carbon Composite R&D for High Temperature Gas-cooled Reactors
Y. Katoh, L.L. Snead, K. Ozawa, C. Contescu, T.D. Burchell, S. Gonczy and W.E. Windes

14h40 Models of Bending Strength for Gilsocarbon Graphites Irradiated in Inert and Oxidising Environments
E.D. Eason, G.N. Hall, B.J. Marsden and G.B. Heys

15h00 A 3D Methodology for the Generation of AGR Graphite Dosimetry Data
D.A. Allen, D.O. Morgan, A.T. Robinson, D.A. Thornton and G.S. Whiley

15h20 Comparison of FEAT Predicted and Measured AGR Fuel Sleeve Residual Stresses
L. Chidwick and M. Joyce

15h40 Announcement of INGSM-12, 2011

Close of conference

15h50 *Coffee/Tea and departure*

Poster Papers

The following posters were notified to us at the time of going to press. Additional posters have been invited.

P1 An Experimental Study of the Fracture Toughness of Nuclear Graphite IG-11
Li Shi, H.T. Wang, H.T. Wang, L.B. Sun, Y.Q. Hu, X.F. Yao and C. Xiong

P2 Irradiated Graphite Waste: Analysis and Modelling of Radionuclide Release with a View to Long-Term Disposal
G. Black, A.N. Jones and B.J. Marsden

P3 Understanding the Effect of Nuclear Graphite Porosity Size and Shape on Young's Modulus using Finite-Element Analysis
A.T. Bakenne, B. J. Marsden, A.N. Jones and G.N. Hall

P4 Thermal Creep in Polycrystalline Graphite
L. Luyken, M. Figueroa, W. Windes, T.J. Marrow and B.J. Marsden

P5 High Temperature Young's Modulus and Bending Strength of Nuclear Grade Graphites
Eung-Seon Kim, Se-Hwan Chi and Sung-Deok Kim

P6 Encapsulation of Irradiated Graphite Waste
B. Hagos, A.N. Jones, T.J. Marrow and B.J. Marsden

P7 EdF Energy Graphite Core Work Programme
A. Whittle, J. Reed, D. Knowles and S. Brasier

P8 *In-situ* Study of Primary Regime Proton Irradiation-Induced Creep in Graphite
A.A. Campbell, G.S. Was and Y. Katoh

P9 Acute Oxidation of Nuclear Graphite: Porosity Development and Loss of Strength
C. Contescu, J. Strizak and T. Burchell

P10 Investigating the Feasibility of Recycling Irradiated Graphite: A First Demonstration
P. Pappano and T. Burchell

We also welcome exhibitions from:

Netzsch Instruments (throughout the conference) and
KorteQ Ltd (Sunday – Monday only)

Abstracts

Sunday

Sunday 16h10 Launch of the IAEA International Knowledge Base on Nuclear Graphite

A.J. Wickham[1], G. Haag[2] and L. Ralph[3]

[1]Nuclear Technology Consultancy, Builth Wells, LD2 3XA, UK
[2]Nuclear Technology Consultancy, Am Malefinkbach 3, D-52441 Linnich, Germany
[3]KorteQ Ltd, UK

The established IAEA International Database on Irradiated Nuclear Graphite Properties has now been incorporated into a wider Knowledge Base intended both for the archiving of documents and for the creation of specific reference articles, intended to create a permanent record of the accumulated knowledge on all aspects of the production and irradiation of nuclear graphites. This web-based system will be demonstrated and this will be followed by a discussion on the appropriate 'population' of the Knowledge Base in order to maximise the benefit. All graphite specialists are invited to contribute!

Sunday 16h50 The Database for Irradiation-Induced Creep in Graphite

G. Haag, Nuclear technology Consultancy, Am Malefinkbach 3, D-52441 Linnich, Germany

The Database for Irradiation-Induced Creep in Graphite is a continuation of the IAEA International Database on Irradiated Nuclear Graphite Properties. However, creep is not a classical material property to be determined by a set of physical measurements derived from a clear definition formula. Therefore the IAEA Database may need modification. Physical understanding of what creep really is or what it depends upon is not yet finally developed, but it is in the focus of several working groups in different countries. To give them a common data basis it is intended (as a first step) to collect the available data and helpful additional information from past creep experiments. In this framework, re-evaluating the existing information has revealed some basic difficulties which will be discussed. This is also important with respect to assessing the quality of the data and to assist the safety authorities in the assessment of safety aspects of graphite moderated reactors.

Sunday 17h10 Introducing the EPSRC FUN-GRAF Consortium: Fundamentals of Current and Future Uses of Nuclear Graphite

M.I. Heggie[1] and B.J. Marsden[2]

1 The University of Sussex, UK
2 The University of Manchester, UK

From September 2010 a new collaborative research project sponsored by the EPSRC will commence, this research investigates the Fundamentals of Current and Future Uses of Nuclear Graphite and is part of a novel and exciting major collaborative research initiative between the Universities of Leeds, Manchester, Nottingham, Salford and Sussex. Further details for this research program will be announced.

Sunday 17h30 The USA's NGNP Graphite R&D Programme

W. Windes, T. Burchell, W. Swank, J. Strizak and M. Carroll
Idaho National Laboratory and Oak Ridge National Laboratory, USA

The NGNP Graphite R&D program is currently generating quantitative data (i.e. Quality Assured data) necessary for predicting the behavior and operating performance of the new nuclear graphite grades available for use within the core of new VHTR designs. A brief outline of the major activities and the progress within each area will be presented; as-fabricated material property testing, irradiation program, and mechanisms development. The as-fabricated testing activities will establish a large, statistically accurate material property database to ascertain the material property limits for these new graphite types. An extensive material irradiation program has been initiated, the Advanced Graphite Creep (AGC) experiment, and will determine the in-service behavior of the graphite for both popular reactor designs (pebble bed and prismatic). Key material properties such as thermal diffusivity, CTE, elastic modulus, mechanical strength, and irradiation induced dimensional change rate and irradiation creep for a wide variety of nuclear grade graphite types will be measured over a dose range of 1 to 7 dpa. Finally, a number of key scientific research initiatives will be outlined that are currently being pursued to better understand the behavior of graphite within an irradiation environment. These research projects tackle complex issues such determining the actual irradiation creep mechanisms in graphite, thermal creep mechanisms within a graphite crystallite, or determining the evolution of pore microstructure during irradiation.

Sunday 17h50 NBG-17: A New Graphite Grade for HTRs and VHTRs

P.Béghein, G.Berlioux, F.Hiltmann, M.Melin and B.du Mesnildot
SGL Carbon Ltd

A nuclear graphite grade, NBG-18, for moderator and reflector application in gas-cooled nuclear reactors is established. Prismatic-type reactors however require a material with lower maximum grain size due to the rather small spacing between cooling channels in the fuel block. With NBG-17, a suitable material is developed using same proven raw materials and process route, including vibro-moulding forming process. Production process and material characteristics are described, demonstrating the general suitability for nuclear application, with irradiation tests in progress. Other materials to accompany the industrial application – powder matrix for fuel spheres, nuclear carbon blocks, and carbon fibre-reinforced composites (CFRC) – are briefly described.

Sunday 18h10 Characterisation of Baseline Properties in Nuclear-Grade Graphites

M.C. Carroll, Idaho National Laboratory, USA

The economic and technical complexity of advanced irradiated graphite characterization programs limits the size and number of samples that can contribute to resultant databases. The limited sample populations have necessitated a joint program between Idaho National Laboratory and Oak Ridge National Laboratory that will provide extensive baseline graphite properties, and will be used to ascertain more quantifiable material property changes following irradiation in the Advanced Test Reactor via the Advanced Graphite Creep (AGC) experimental program. Along with the as-fabricated material properties, the inherent variability of bulk characteristics within a single billet of nuclear-grade graphite will be presented based on approximately 750 mechanical tests at specific positions and orientations. Additional physical and thermal analyses have also been carried out in order to provide a thorough understanding of property characteristics based on location within an as-produced graphite block. Observations pertaining to the underlying causes behind these variations will be presented, along with an examination of how these results will compliment the AGC program. Discussion will also address the future of the program in examining variability comparisons between different batches of the same graphite as well as across the available spectrum of new graphite grades.

Sunday 18h30 Microstructural Characterisation of Next Generation Nuclear Graphites

J. Kane, C. Karthik, R. Ubic and D. P. Butt
Boise State University, Idaho, USA

Irradiation-induced damage and creep in graphite are considered two of the life-limiting factors of nuclear graphite and hence next generation high temperature gas cooled reactors. To gain a fundamental, materials science based, understanding of the mechanisms involved in irradiation damage and creep, the microstructural changes in graphite upon irradiation must be studied. This research reports the preliminary characterization of virgin nuclear graphites NBG-18, PCEA, IG-110, and PGX via transmission electron microscopy. Observed microstructural features such as quinoline insoluble (QI) particles, microcracks, turbostratic regions, and amorphous regions have been compared. We have also observed in-situ the atomic level processes involved in the swelling of the graphite under irradiation using high resolution electron microscopy (HREM) in conjunction with electron energy loss spectroscopy (EELS). It was observed that under e-beam irradiation, the structural ordering parallel to the basal plane gradually deteriorated because of the appearance of dislocation loops, bending, and fragmentation of the basal planes in addition to a dilation in the lattice parameter along the c-axis causing the swelling of the lattice and the closing of the micro cracks.

Monday

Monday 09h00 First Principles Calculations on the Thermal Properties of Irradiated Graphite

G. Haffenden, M.I. Heggie and G. Sheehan
The University of Sussex, UK

First principles calculations have been performed on perfect and defected graphite to establish, on the atomic level, the causes of post irradiation changes in thermal properties such as heat capacity and coefficient of thermal expansion. The heat capacity of irradiated graphite at very low temperatures shows deviation from that of unirradiated graphite and we believe point defects caused by the irradiation process are the cause of this change. At temperatures above 3000K a steep increase in heat capacity is observed and we believe this is caused by the reversible formation of Stone-Wales defects. The coefficient of thermal expansion also shows a large change in the expected behaviour upon irradiation, this too can be explained by the presence of certain point defects. These defects pin the layers in place and restrict the vibrational freedom of the structure, thus altering the thermal properties.

Monday 09h20 A New Methodology for Modelling the Failure Characteristics of Nuclear Graphites

G.D. Kipling and G.B. Neighbour
The University of Hull, UK

Carbon materials commonly exhibit complex failure mechanisms. Nuclear graphite demonstrates non-linear failure characteristics and an unpredictable fracture path, thereby making it difficult to accurately and consistently model the material computationally. These complexities are due, in part, to the heterogeneity of the materials microstructure. A new approach for modelling the failure of nuclear graphite involves the application of finite element analysis (FEA) to a simulated microstructure model. The methodology for generation of a probabilistic microstructure based on observed flaw distributions is outlined as well as an appropriate FEA approach to produce meaningful failure statistics.

Monday 09h40 Progress in Developing Finite Element Models replicating Flexural Graphite Testing

R.L. Bratton and S. Duffy
Idaho National Laboratory, USA

A four point bend finite element model will be presented where a new material model is employed taking into account the non-linear moduli for tension and compression in graphite. The flexure strength as determined by the ASTM standard will be compared to the flexure strength determined by the finite element model with non-linear moduli. The flexural strength is used in current ASME Graphite Core Component design rules. Column buckling is also affected by the difference in flexural strength.

Monday 10h00 Microstructural Characterisation of Irradiated Nuclear Graphite using Finite-Element Analysis

A.T. Bakenne, B.J. Marsden, A.N. Jones and G. N. Hall
The University of Manchester, UK

Generation IV nuclear reactors require graphite components that will operate efficiently at very high temperature (800 – 1000 0C) and over a longer core life than previous generation reactors. Examples of two possible graphite grades for Generation-IV technology include NBG from SGL Carbon group and PCEA produced by GrafTech. During operation, graphite will experience significant property changes such as dimensional change and Young's modulus, as a result of fast neutron irradiation. Mechanical stresses that build up around the reactor core during degradation may lead to component failure resulting in a shortened lifespan of the reactor. Therefore validation of the through life performance of graphite moderator is critically dependent on the knowledge of graphite properties under the influence of fast neutron. Numerous graphites have been developed with anisotropic or near isotropic material properties. NBG- 10, is one example of graphite grade which uses a pitch coke extruded graphite near isotropic with a grain size of 1.6mm (max). PCEA nuclear graphite is petroleum coke extruded graphite near isotropic with a grain size of about 0.8mm (max). The directional dependence is a result of the geometry of the filler particles used and the manufacturing process. This project aim is to develop confidence in understanding the influence of irradiation induced microstructural changes on the bulk properties. In particular Young's modulus within graphite will be considered. Graphite structures were characterised using high resolution X-ray Tomography, which provides faithful reprehensive data for the input for Finite Element modelling. Digital image correlation (DaVis) will be used to measure local deformation of the tomography data in order to validate modelling predictions. Both graphites have also been characterised using microstructural analysis techniques such as Optical Microscopy, Scanning Electron Microscopy and X- ray diffraction.

Monday 10h20 Estimation of Dimensional Change Curves for AGR Moderator Graphite using a Bayesian Emulator and Inspection Data

K. McNally[1], G.N. Hall[2], G.B.Heys[3], N.Warren[1] and B.J.Marsden[2]

[1] Health and Safety Laboratory, Buxton, UK
[2]Health and Safety Executive, Nuclear Installations Inspectorate, Bootle, UK
[3]The University of Manchester, UK

Historically predictions of graphite moderator brick integrity in the UK's Advanced Gas-cooled Reactors (AGR) have been made using finite element methods and materials data from Materials Testing Reactor (MTR) programmes. Brick internal stresses develop due to restrained dimensional change. Predictions rely on a detailed knowledge of graphite irradiation behaviour; dimensional change and irradiation creep are of high importance. However, the availability of such data, under relevant AGR operating conditions, is limited. During reactor outages, fuel channel brick and bore dimensions are routinely measured. Changes from as-manufactured dimensions are a consequence of: irradiation-induced dimensional change; development of brick internal stress; alignment of fuel within the channel. Whilst previous investigations were reasonably successful in predicting brick shape from MTR data, prediction of brick internal stress was less convincing. The overall aim of this work is to devise methodology to relate observed bore shape changes to internal brick stresses caused by irradiation-induced dimensional and properties changes modified by irradiation creep. The first stage in this work was to reverse-engineer the underlying dimensional change behaviour from inspection data using a Bayesian approach. This paper presents the concepts behind this and preliminary results.

Monday 11h10 The Effect of Annealing on Thermal Properties, Crystalline Properties and Kinetic Parameters of Wigner Energy Release of BEPO Irradiated Nuclear Graphite

M. Lasithiotakis, B.J. Marsden and T. J. Marrow
The University of Manchester, UK

In nuclear fission reactors the physical properties of graphite are altered by fast neutron irradiation. An apparent restoration of some of the damage, particularly that created at low irradiation temperatures, is achieved with thermal annealing above the irradiation temperature. During this annealing process, the accumulated energy is released in the form of heat, known as Wigner energy. The potential for heat generation on annealing of low-temperature-irradiated graphites must be accounted for during the choice of a nuclear reactor decommissioning procedure and disposal method. In this work, Differential Scanning Calorimetry was utilized to anneal BEPO (British Experimental Pile Zero) irradiated nuclear graphite specimens, whilst also measuring the rate of Wigner energy release. The BEPO graphite was irradiated at temperature below 100°C. X-Ray diffraction and Raman Spectroscopy were used to evaluate the extent of irradiation damage in the graphite crystal structure and the effects of annealing. Both isothermal and linear heating rate experiments were performed. The kinetics of the response of these measurements to thermal annealing have been assessed, using an independent parallel reactions model in combination with Weibull kinetic analysis. An annealing kinetics model has been constructed and is presented in this paper.

Monday 11h30 — A Review of Stored Energy Release of Irradiated Graphite

N.C. Gallego[1], T.D. Burchell[1], and M. Srinivasan[2]

[1]Oak Ridge National Laboratory, TN, USA
[2] Nuclear Regulatory Commission, USA

The NRC-sponsored Nuclear Graphite Workshop, conducted by Oak Ridge National Laboratory (ORNL) in 2009, identified that technical information is currently insufficient for understanding the potential release of stored energy of high temperature irradiated graphite when subsequently heated to higher than its previously irradiated temperature. Such phenomena could occur, for example, in a loss of coolant accident in an HTGR. The general prevailing belief is that the potential for such stored energy in high temperature irradiation is negligible. However, experimental data are sparse to support this opinion. The NRC has initiated research at ORNL to: (a) conduct a comprehensive literature review and evaluate the current knowledge on the effects of energy stored on previously irradiated nuclear graphite and to assess the adequacy of this information to understand the expected operational conditions of NGNP HTGR; and, (b) examine the irradiation-induced stored energy on previously irradiated graphite, such as the grade H-451, which was used in Ft. St. Vrain and Peach Bottom reactors and other grades, by conducting designed experiments. At this meeting, we will present a summary of the findings from the literature review and discuss a draft experimental design to study stored energy release.

Monday 11h50 — The NGNP Graphite Creep Experiment Irradiation in the ATR

S.B. Grover, Idaho National Laboratory, USA

The United States Department of Energy’s Next Generation Nuclear Plant (NGNP) Program will be irradiating six gas reactor graphite creep experiments in the Advanced Test Reactor (ATR) located at the Idaho National Laboratory (INL). The graphite experiments will be irradiated over the next six to eight years to support development of a graphite irradiation performance data base, including irradiation creep at different temperatures and loading conditions on the new nuclear grade graphites now. The experiments will consist of a single capsule containing six stacks of graphite specimens, with half of each stack under a compressive load, and no load on the other half of the stack. A seventh stack will not have a compressive load. The specimens will be irradiated in an inert sweep gas atmosphere with on-line temperature and compressive load monitoring and control, and capability to sample the sweep gas effluent for oxidation products or off-gassing of the specimens. The first experiment started its irradiation in September 2009, and is anticipated to complete irradiation in late calendar 2010 or early calendar 2011. This presentation will discuss the experiment including the test train design, temperature and compressive control systems, and the irradiation experience to date.

Monday 12h10 Design and Results of the HFIR Creep Rabbit Capsule

L.L. Snead, T.S. Byun and Y. Katoh
Oak Ridge National Laboratory, TN, USA

The design and irradiation of a bellows-loaded creep capsule has been carried out for the relatively small and versatile "rabbit" position of the High Flux Isotope Reactor at ORNL. This design loads a 25 mm long, 6 mm diameter cylinder of H-451 nuclear graphite at 10 Mpa and a temperature of approximately 800°C. Each cylindrical sample included a SiC temperature monitor to determine irradiation temperature. Two prototype capsules were irradiated at 2 and 6 dpa, respectively. Following irradiation the load train stiffness (bellows and holder) were tested to ensure integrity of the bellows during irradiation. The graphite samples were then removed from the load train and post-irradiation-examination was carried out including effects on dimensional change, elastic modulus, coefficient of thermal expansion, and microstructure. This paper will report on both the integrity of the prototype design and results from the crept specimens.

Monday 14h00 Development of a New ASTM Guide for Measurements on Small Graphite Specimens

M.P. Metcalfe, National Nuclear Laboratory, UK

There is currently a suite of ASTM standards that can be applied to graphite covering a range of physical, mechanical and thermal property measurements. These standards have been developed with the objective of optimising the method of measurement in the absence of any constraints on test sample production. A new guide is being developed to provide advice on how the application of selected standards to small samples under non-compliant conditions can be tested for suitability. This presentation describes progress on this initiative.

Monday 14h20 Expansion of Irradiation Data by Interpolation and Extrapolation for Design of Graphite Components in High Temperature Gas-cooled Reactors

E.Kunimoto[1], T.Shibata[2], M.Eto[1], S.Shiozawa2[2], K.Sawa[2], T.Maruyama[2] and T.Oku[3]

[1] *Toyo Tanso Co. Ltd, Japan*
[2] *Japan Atomic Energy Agency*
[3] *University of the Air, Japan*

Graphite materials are used for core components of HTGRs. In Japan, the authors prepared a technical document entitled “Draft of Standard for Graphite Core Components in High Temperature Gas-cooled Reactor”. The draft standard contains graphical expressions in the wide range of fluence for various properties of irradiated IG-110 Graphite. The graphs were obtained based on interpolation and extrapolation of the existing irradiation data. The necessary database can be established by expansion of existing irradiation data with appropriate interpolation and extrapolation methods. This presentation shows the reasonable interpolation and extrapolation method for strength and other properties of IG-110 graphite which is used for the HTTR and a major candidate for the VHTR. The interpolation and extrapolation method was developed so as to be applicable to the irradiation data on other graphites.

Monday 14h40 Strength Evaluation of Ultra-fine Grained Graphite by Miniature Equi-biaxial Flexural Test

Y. Katoh, G. Vasudevamurthy and S. Kondo,
Oak Ridge National Laboratory, TN, USA

Equibiaxial flexure is an alternative flexural test configuration to conventional four-point bending of brittle materials. This test method is particularly suitable and attractive for irradiation effect study on fine-grained, isotropic nuclear graphite due to 1) the applicability of very small specimens, 2) freedom from the edge effect, and 3) lack of anisotropy effect on biaxial strength. In the present work, an equibiaxial flexural test procedure for using miniaturized specimens was first developed. Influences of the test configurations, loading conditions and the Young's modulus to fracture stress ratio on the stress uniformity were studied by a finite element analysis and validated by an experiment. Based on that and the recommended standard practice in ASTM C1499, a statistically significant population of miniature specimens suitable for some of the planned irradiation programs were tested. The material used was Tokai Carbon grade G347S isotropic, ultra-fine grained nuclear graphite. Uniaxial flexural tests were also conducted in a four-point-1/3 configuration using the full-size (ASTM C651) and the 1/2 size (ASTM C1161) specimens machined from the same graphite block. The results obtained from different test configurations and different specimen dimensions are compared and discussed in terms of the size scaling.

Monday 15h00 Size Effects in the Mechanical Properties of NBG-18 Graphite

T.S. Byun, G. Vasudevamurthy, W.D. Lewis, L.L. Snead and T.D. Burchell
Oak Ridge National Laboratory, TN, USA

The fracture strength and static elastic constants of NBG-18 graphite have been characterized using sub-size cylindrical specimens and statistical treatments. Tensile fracture tests have been performed for eight sets of cylindrical specimens with different diameters (4, 6, 8, and 12 mm,), lengths (18 – 46 mm), and grain orientations. Elastic constants were obtained for selected specimens by reading two-dimensional microstrains from specimen surfaces. To have specimen heads for loading in tensile machine, aluminium head pieces were glued to both specimen ends. Only a limited size effect was found in the strength data: the mean values for the six sets of with-grain (WG) specimens ranged from 21.7 to 22.9 MPa. The fracture strengths measured from the two sets of against-grain (AG) specimens were slightly lower. The standard deviations for the specimens with WG-orientation were in the range of 2 – 3 MPa, which resulted in the Weibull moduli ranging from 8.3 to 12.4. Meanwhile, no size effect was observed in the elastic constants regardless of the large variation in specimen size: the average Young's modulus was 6.85 GPa and average Poisson ratio was 0.165. A general conclusion of this research was that the mechanical property data from a sub-size cylindrical sample with glued end-pieces can be representative of the ASTM compliant sample for the NBG-18 graphite.

Monday 16h00 Characterizing the Origin, Distribution and Chemical Form of ^{14}C in Irradiated Graphite

M.L. Dunzik-Gougar[1], S. McCrory[2], J. Cleaver[2] and J. Fachinger[3]

[1] Idaho State University, USA
[2] Idaho National Laboratory, USA
[3]FNAG, Germany

During the operation of graphite-moderated reactors, neutron interactions with graphite impurities produce activation products, including ^{14}C, which is formed through three distinct reactions with ^{13}C, ^{14}N, and ^{17}O. Study of irradiated graphite from many reactors indicates ^{14}C is concentrated in the outer 5 mm of the graphite structure. ^{14}C is of concern with respect to long-term disposal of the graphite due to its half-life and high mobility in the environment. The aim of research to be presented is to identify the chemical form of ^{14}C in irradiated graphite and develop a practical method by which ^{14}C can be removed. Pre- and post-irradiation characterization of nuclear grade graphite is being conducted to determine bonding characteristics, functional groups, location and concentration of ^{14}C and its precursors. A number of characterization methods are under investigation for analyzing the graphite surfaces, where 14C is concentrated. Specifically, ^{14}C bonds with oxygen and nitrogen are expected and must be distinguished from the bulk carbon bonding. Preliminary results from X-ray Photoelectron Spectroscopy (XPS), InfraRed and SIMS analyses will be presented and discussed. Also presented will be developments in experiments for thermal treatment of irradiated graphite to remove ^{14}C.

Monday 16h20 Validation of Nuclear-Physics Calculations against Measured Radionuclide Fingerprints of Oldbury and Wylfa Magnox Graphite Samples

R. Mills and Z. Riaz, National Nuclear Laboratory, UK

This study reports comparisons between calculations and measurements of the activation of graphite samples installed within interstitial channels of the Oldbury and Wylfa reactors during their construction. These samples have been irradiated during the operating life of the reactors until their removal in 2006 and 2008. The calculations were carried out with MCNPX and the UK FISPIN codes. An investigation into the nuclear data used in these calculations high-lighted recent improvements in the neutron cross-sections used to calculate ^{14}C activities that on the effect of graphite impurities on ^{14}C production is described.

Monday 16h40 On the Formation and Behaviour of ^{14}C in Irradiated Graphite

A.J. Wickham, Nuclear Technology Consultancy, Builth Wells, UK
(in association with Bradtec Decon Technologies Ltd, UK and EPRI, USA)

^{14}C, a long-lived β-emitter, concerns radioactive waste-disposal authorities because of the potential take-up into the biosphere and the environment. In graphite, it is formed by two principal routes, from ^{13}C and from ^{14}N, with some contribution from ^{17}O in reactor systems cooled with air of CO_2. There are strongly differing opinions on the importance of the ^{14}N route, and these reflect largely upon the true nitrogen content of the graphite at the time of irradiation. Equally, the literature contains differing opinions on the relevance of recoil energies at the point of formation, and this leads to important questions about the nature and distribution of the ^{14}C atoms, and their subsequent behaviour under the continuing fast-neutron damage processes to which the graphite is subjected. A major study of this topic, commissioned by EPRI, has provided answers to some of these points and indicates the areas where further information is required. A number of questions remain unresolved and will be posed to this specialist audience. The impact of the study with respect to the management of radioactive graphite wastes will be discussed.

Monday 17h00

Characterisation and Leaching of British Experimental Pile Zero (BEPO) Nuclear Graphite

L. McDermott, A. N. Jones, B. J. Marsden and T. J. Marrow
The University of Manchester, UK

The British experimental pile Zero (BEPO) reactor was one of the first experimental reactors in the UK. BEPO operated for twenty years at Harwell between 1948 and 1968, was graphite moderated, air cooled, fuelled with natural uranium metal in aluminium cans. Nuclear moderator graphite is extremely pure, with impurities in the region of parts per million (ppm) as any impurities may lead to the production of undesired radioactive isotopes. The radioactive isotopes present within the BEPO graphite matrix are currently known to be ^{3}H, ^{14}C, ^{60}Co, ^{152}Eu and ^{154}Eu. These isotopes are commonly found within all irradiated graphites and are of particular concern to the NDA due to their long potential half lives. The activities and characterisation of these isotopes within UK nuclear graphites is therefore of significant consequence to decommissioning, waste treatment and future reprocessing options. Characterisation studies of irradiated BEPO graphite have been carried out within the Nuclear Graphite Research group; this research focuses on developing methods to accurately determine the activity of the radioisotopes, particularly ^{14}C and ^{3}H. The aims are to understand the nature and quantity of impurities in the initial microstructure after neutron irradiation and to investigate novel treatment methods in order to remove the isotopic contamination. Characterisation techniques including Synchrotron X-ray Tomography, Raman spectroscopy and polarised optical microscopy have been utilised. This presentation will focus on the most recent treatment results achieved from aggressive leaching of BEPO graphite in order to quantify the release of ^{14}C and ^{3}H from the graphite.

Monday 17h20

Potential Options for the Management of Radioactive Graphite Wastes

E.J. Harvey, E.M. Michie and S.M. Wickham
Galson Sciences Ltd, UK

Geological disposal has generally been regarded as the most practicable approach to manage radioactive graphite wastes and this is the current baseline strategy for the management of UK bulk reactor graphite. However, alternative strategies are increasingly being sought that might enable a reduction in the volume of waste requiring geological disposal. Several recent and ongoing initiatives have explored different options for managing the UK's radioactive graphite wastes. Other countries, notably France, are also developing alternative strategies for irradiated graphite management. In light of these recent developments, Galson Sciences Ltd has carried out a review, on behalf of the Environment Agency, of potential options for the management of UK radioactive graphite wastes. The review will be used to help guide the work of the Site License Companies and the Nuclear Decommissioning Authority in developing graphite management strategies, and will help to ensure that this work delivers solutions that are likely to be acceptable to the Environment Agency. The outputs will also be used to inform the Environment Agency as it prepares to review any forthcoming proposals for the management of radioactive graphite wastes. This presentation will discuss the approach followed to consider graphite management options, and the findings of the review.

Monday 17h40 Carbowaste - Work Package 5 - Graphite Recycle

J. Goodwin, Hyder Consulting Ltd., UK and D. Bradbury, Bradtec Decon Technologies Ltd, UK

Work Package 5 of the EC funded CARBOWASTE programme is focused on the development of recycle pathways for irradiated graphite derived from the decommissioning activities worldwide. Graphite can potentially be recycled by processing the irradiated graphite without decontamination. This would require fully active (and radiation shielded) recycle facilities and this is being considered elsewhere. Decontamination of the graphite would allow easier recycle, and this can be achieved in part through gasification which allows effective decontamination for radionuclides other than ^{14}C through their retention in solid form for disposal as radioactive waste. This may be preceded by an initial 'roasting' phase the result of which would be ^{14}C rich synthesis gas, which with further concentration by isotope separation may yield a suitably pure alternative source of ^{14}C for use e.g. as a medical radioisotope. Re-deposition of the synthesis gas into solid form through the use of the Sabatier and Bosch reactions provides a suitable intermediate (e.g. Carbon Black) for subsequent incorporation into new products which, due to ^{14}C content, would be restricted for use in the nuclear industry. Products identified which have the potential to be manufactured in part (at least) from recycled graphite include graphite electrodes for use in radioactive waste vitrification, and activated carbon for use as a gas absorbent in filters to remove radionuclides. Silicon Carbide produced in part from recycled graphite has potential use as a confinement/packing material for wastes within a repository site. Recycled graphite can also be used for various novel purposes such as an adsorbent for radionuclides from waste streams.

Monday 18h00 Progress Update on Segregation and Retrieval of Irradiated Graphite

M. Grave (Doosan Babcock) and A. Banford
(National Nuclear Laboratory, UK)

This paper is a brief 5 minute update of the status of the CARBOWASTE Retrieval and Segregation of irradiated graphite. A more detailed paper is in preparation for presentation at UKDWM10 in November 2010. The presentation will outline the findings of the Retrieval and Segregation report and indicate the modelling activity that is in progress.

Tuesday

Tuesday 09h00

Pre-Peak Load Displacement Characteristics of PGA Reactor Core Graphite

S. Nakhodchi[1], P. Heard[1], A. Hodgkins[2], R. Moskovic[3], M.R. Wootton[3] ,D.J. Smith[1] , and P.E.J. Flewitt[1]

[1]*The University of Bristol, UK*
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[3]*Magnox North ltd, UK*

Pile grade A (PGA) graphite used in UK gas cooled reactors is a multiphase, polygranular, aggregate material with a complex deformation and cracking behaviour. Indeed, it can be classified as a quasi-brittle material. Experiments have been performed on plain and notched beam geometry and compression disc unirradiated PGA graphite specimens to determine load-displacement characteristics. For the beam specimens the three-dimensional digital image correlation technique has been combined with resistance strain gauge measurements to evaluate, both the localised and the global displacements during testing. The digital image correlation technique used two cameras focusing from two directions at the object surface so that in plane and out of plane displacements can be determined. The non-linear load-displacement characteristics prior to peak load are correlated with the localised displacements which can extend up to ~3mm from the tensile surface of the specimen. At peak-load it is from such a process zone that a macro-crack propagates rapidly along an irregular path controlled by the direction of the applied load and the microstructure of the graphite. Pre-peak load cracking in disc geometry specimens of unirradiated and irradiated graphite has been observed using optical and focused ion beam microscopy. Micro-cracking has been shown to precede macro-crack formation and this mechanism is consistent with the observed pre-peak load non-linearity in the load-displacement curve. The results are discussed with respect to the quasi-brittle displacement characteristics of such porous aggregate materials.

Tuesday 09h20

The Effects of Equi-biaxial Stress on Crack Nucleation and Propagation in Nuclear Graphite

M. Mostafavi and T.J. Marrow, The University of Manchester, UK

Stress state plays an important role in the fracture behaviour of quasi-brittle materials such as nuclear graphite, and it is known that biaxial stress can reduce strength. Equi-biaxial stress is representative of the state of stress in some regions of the graphite core during service. It is therefore important to investigate its fracture behaviour under such loading. The objective of the present study was to determine the criteria for stable and unstable propagation of crack nuclei in nuclear graphite under uniaxial and equi-biaxial loading, by observation of the development of damage using Digital image correlation (DIC). DIC has recently received much attention as a powerful tool to detect and analyse the failure initiation and propagation in a wide range of materials. Useful information can be gained from such analyses including the evolution of the geometry of the fracture initiator and formation of the fracture process zone. In this paper, the damage process in Gilsocarbon has been characterised by direct observation of the fracture of ring-on-ring and four-point bend tests. A cohesive zone finite element model was used to simulate successfully the observed behaviour. Such a model may be used to predict the behaviour of nuclear graphite under other stress states.

Tuesday 09h40 Pre-Irradiation Characterisation of AGC-2 Specimens

W. D. Swank, J. R. Lord, D. Rohrbaugh and W. E. Windes
Idaho National Laboratory, USA

Graphite has been effectively used as a structural and moderator material in both research and commercial HTGRs (high temperature gas-cooled reactors). While the processes necessary for producing nuclear grade graphite are understood, historical nuclear grades no longer exist. New grades must therefore be fabricated, characterized, and irradiated to demonstrate that current grades of graphite exhibit acceptable irradiated and non-irradiated properties upon which the thermo-mechanical design of the structural graphite in the Next Generation Nuclear Power (NGNP) plant is based. There are six Advanced Graphite Creep (AGC) experiments planned to study radiation effects on 16 different types of graphite. Described here are the pre-irradiation measurements of over 500 specimens designated for the AGC-2 experiment expected to begin irradiation during the winter of 2011. Measurements of density, thermal diffusivity, coefficient of thermal expansion, resistivity and elastic modulus are presented for isostatically moulded, extruded, and vibrationally moulded graphite. Comparisons of properties and isentropic behaviour are made for the different manufacturing processes.

Tuesday 10h00 The Simulation of Neutron Damage in Nuclear Graphite using Ion Irradiation

M. Lasithiotakis, B. J. Marsden and T.J. Marrow
The University of Manchester, UK

In a nuclear fission reactor the physical properties of graphite are changed by fast neutron irradiation. A convenient method of simulating irradiation damage in small samples is by ion bombardment. However it is important that the nature of the damage due to ions, which are available over a limited energy range, is equivalent to that from neutrons, which have a more extended spectrum of energies. The cascades caused by ion irradiation may also differ from neutron-induced cascades. The properties and microstructure of various ion irradiated nuclear graphite grades have been investigated therefore, to gain confidence that ion-irradiation damage is sufficiently representative of neutron irradiation damage. The surface characterization techniques of Scanning Electron Microscopy, Atomic Force Microscopy and Raman Spectroscopy were used to evaluate the extent of irradiation damage and to compare the effects of ion bombardment with the reported effects of neutron irradiation. The kinetics of the response to thermal annealing of the properties of ion-irradiated samples have been compared with the kinetics of energy release with annealing of neutron irradiated graphite. A qualitative agreement between the two phenomena is found.

Tuesday 11h00 Determining Pore Structures in Nuclear Graphite

M. Figueroa and W. Windes, Idaho National Laboratory, USA

The pore structure in a nuclear graphite microstructure is key to understanding the irradiation behaviour of graphitic components, specifically volumetric shrinkage ratios, irradiation creep, and component strength. However, accurately determining the pore structure is problematic even though most nuclear graphite has significant open porosity. Recent work investigating quantitative pore measurement techniques for new nuclear grade graphite types is presented and discussed. Traditional gas adsorption or liquid intrusion techniques as well as X-ray computer tomography and 2D optical microscopy were all applied to graphite types of current interest to the USA's NGNP program. A Monte Carlo Fortran code has been developed to ascertain a quantitative 3D pore size distribution based upon data generated from the X-Ray CT and multiple 2D images to provide direct comparison of the disparate techniques.

Tuesday 11h20 The Modelling of Graphite Oxidation Behaviour under Reactor Normal Operating Conditions

S.Y. Yu, X.L. Yu, L. Brissonneau and C. Bourdeloie
INET, Tsinghua University, Beijing, China

The oxidation of graphite in normal operating conditions is a very important factor when evaluating the service time of the graphite structural material in a High Temperature Gas-cooled Reactor. This presentation deals with the modelling of graphite oxidation by steam in the helium channel of a fuel block. The FEM software COMSOL is used: the turbulent flow of the coolant is simulated by using the k-å model and the chemical reaction is expressed by the Langmuir-Hinshelwood equation. Calculations were carried out for steam pressures around 1 Pa and for different temperature distributions. The influence of burn-off and the diffusion in graphite porosities were both considered in the oxidation. Results show that oxidation mainly occurred on the graphite surface at the bottom of the core because of the higher temperature. The thickness of graphite with a burn-off higher than 8% was about 1 mm at the core base. Less than 15% of steam was consumed in the coolant channel of the fuel assemblies. Calculations also showed that the mean gasification rate in one channel for the second service time was larger than the first service time.

Tuesday 11h40 Probabilistic Predictions of Graphite Weight Loss

I. Robertson[1], M.R. Bradford[2] and P.C. Matthews[2]

[1]Frazer-Nash Consultancy Ltd, UK
[2]EdF Energy (British Energy), Gloucester UK

Nuclear Power Station Safety Cases require predictions of various graphite properties in order to support their continued use. In recent years, these predictions have increasingly been based on probabilistic as well as deterministic principles such that both the inherent variability of the system and the uncertainty in data and modelling are considered and accounted for. A methodology to predict graphite weight losses at British Energy's AGR power stations has been formulated which includes the propagation of variability and uncertainty through the solution process. The method attempts to capture and constrain the variability in the graphite physical properties by tuning the predicted results to measured data using various statistical and optimisation techniques. When fully implemented the method will be capable of producing predictions, and associated uncertainty, of graphite weight losses in life limiting control volumes.

Tuesday 12h00 Pore Size Change in Depth and Oxidation Reaction of Selected Nuclear Graphites in 873-1273K Air

S.H. Chi, G.C. Kim, J.H. Jang, H.J. Kim and E.S. Kim
Korea Atomic Energy Research Institute

Temperature dependent oxidation reaction behaviour of four kinds of nuclear grade graphites, *i.e*. IG grades(110, 430) and NBG grades(18, 25), were investigated based on the pore size measurements in depth after 10 percent burn-off in 873-1273K in air. Specimens were cylindrical (1 inch dia. and 1 inch length), and oxidized in a vertical tube furnace at 10 L/min air flow rate. The pore sizes were measured by using an optical microscope with an image processing software at four locations: surface, 710, 1420, and 2140 microns in depth. Results show that, while all grades show a negligible pore size change in depth at 873 and

1273 K, apparent gradients in pore size were observed in depth at 973 K. At 1073 K, while the IG-grades showed a negligible pore size changes in depth, NBG-grades showed an apparent gradient in pore size in depth. Largely, while the IG-grades showed an oxidation behaviour as discussed in the classic oxidation model, the regime 2 reaction range of NBG-grades appeared to extend into a higher temperature. Part of the observed extension in NBG-grades may be attributed to their lower grain boundary area and high density.

Wednesday

Wednesday 08h40 Eddy Current Technique Development: Initial Survey Results from AGR Reactor Cores

M. Brown, A. Greenaway and J. Williams, EdF Energy (British Energy), UK

Advanced Gas Reactor (AGR) graphite cores are inspected during periodic shutdowns using a range of techniques, such as; TV camera, channel bore diameter and tilt measuring, and the trepanning of sample cores. To further enhance the range of inspection information, development of an eddy current capability has been progressed. The technique was initially developed with the objective of detecting subsurface cracks emanating from the outside of the brick (keyway root crack detection). During the development of the technique it also became apparent that the technique had potential to provide an indirect measurement of the graphite brick density, via measurement of electrical conductivity. This paper outlines the development of the proof of principle eddy current tool (PoPECT) and summarises the initial survey results from deployment at Hartlepool and Heysham I reactors.

Wednesday 09h00 Update on the Blackstone AGR Graphite Irradiation Programme

O. Wouters[1], B.P. Jonker[1], P. Matthews[2] and M. Joyce[3]

[1]NRG Petten, The Netherlands
[2]EdF Energy (British Energy) Ltd, Gloucester, UK
[3]Frazer-Nash Consultancy Ltd, UK

The first phase of the Blackstone irradiation programme has nearly been completed. Two capsules filled with graphite samples from British (EDF Energy) Advanced Gas-cooled Reactors have been successfully irradiated in the High Flux Reactor in Petten, The Netherlands. The samples in one of the two capsules were subject to an oxidising CO_2 environment, representative of the AGR. The post irradiation experiments are in an advanced state with the first results being analysed to be used in future AGR safety cases. This presentation will summarise the first phase results including irradiation characteristics (dose, temperature), post irradiation sample properties and potential use of the data. Furthermore an outlook towards the second phase of the programme, aimed at reaching end-of-life graphite conditions, will be given.

Wednesday 09h20 Irradiation of HTR Graphites in HFR Petten

A. Vreeling, NRG Petten, The Netherlands

In 2001 the Nuclear Research & consultancy Group started a graphite irradiation program for the development of High Temperature Reactor technology in a European framework. The irradiation experiments, containing present day available graphite grades, are performed at the High Flux Reactor in Petten. The grades are NBG-10, NBG-17, NBG-18, NBG-20, NBG-25, PCEA, PPEA, PCIB, LPEB, IG-110 and IG-430. In the fifth framework programme (2001-2004) and sixth framework programme (2005-2010) four irradiation experiments have been performed. The post irradiation testing is focused on dimensional changes, dynamic Young's modulus, coefficient of thermal expansion and coefficient of thermal conductivity. The post irradiation testing of two experiments is finished. The post irradiation testing of a selection of samples from the two high-dose experiments is in progress.

Wednesday 09h40 The CHANSELA Software: a Tool to Aid the Selection of Channels for Inspection at Reactor Outages

P. R. Maul, P. C. Robinson and C.E. Watson, Quintessa Ltd, UK

The inspection of graphite cores in Advanced Gas Cooled Reactors provides important information to support safety cases for their continued operation. Decisions on which channels to inspect at outages are complex as different types of channel (defined by their location in the core, inspection history, burnup etc.) provide different information on the core condition. In this paper details of the CHANSELA (Channel Selection Assistant) software are given. CHANSELA allows the user to view historical inspection information from the reactor on a core map; for example channel bow, brick shrinkage and brick cracking. After the user has specified the information that is desired from a particular inspection, and any constraints or preferences that may influence the selection of particular channels, CHANSELA will automatically find the combination of channels that provides the maximal amount of information using an efficient genetic algorithm. Other combinations of channels can also be assessed and compared against one another. The software allows every stage of the process to be documented, providing a clear and transparent audit trail.

Wednesday 10h00 An Integrated Approach to Inspection Data and Material Properties

M.R. Bradford[1], L. Chidwick[2], A. Price[2] and M. Joyce[2]

[1] EdF Energy (British Energy) Lrd, Gloucester, UK
[2] Frazer-Nash Consultancy Ltd, UK

In recent years British Energy have undertaken a thorough review of the material property models used in component condition assessments in support of the Advanced Gas-cooled Reactor (AGR) Safety Case. Whilst the basic model response remains informed by historic Material Test Reactor data, the model's statistical elements are now derived directly from AGR inspection data. Initially, only data from trepanned samples was considered, however more recently, the measured channel bore shapes are being employed to investigate shrinkage and creep processes. A full statistical treatment of model uncertainty and intrinsic material variability is now employed to make fully blind predictions of new data prior to its release. This provides an opportunity to test model performance, before using the data to refine our understanding of the distributions of material property evolution in specific AGRs.

Wednesday 10h20 Evidence Supporting the Use of MCBEND Dosimetry Predictions within AGR Core Condition Assessments

D.A. Thornton, Serco Ltd, UK

This presentation gives a review of the evidence supporting the use of MCBEND to calculate the neutron damage and nuclear energy deposition dose parameters used within AGR core component condition assessments. Specifically, it describes the following activities carried out in support of this objective: • Review and evaluation of the validation evidence supporting the ability of MCBEND to predict neutron damage and nuclear energy deposition; • Identification of deficiencies in the validation database as well as any issues arising; • Proposals for addressing the deficiencies and issues associated with the validation database; • Presentation of arguments supporting the use of neutron damage and total energy deposition as appropriate correlation parameters; • An assessment of uncertainties applicable to neutron damage and nuclear energy deposition data used in AGR core condition assessments which are predicted by MCBEND.

Wednesday 11h10 Anisotropy of IG110 and NBG18 under Compressive Stresses

S.Y. Yu, X.W. Zhou and H.T. Wang
INET, Tsinghua University, Beijing, China

The thermal expansion coefficient (CTE) of nuclear graphite IG-110 and NBG-18 under compressive stresses of 20MPa, 30MPa and 40MPa has been measured by strain gauge method and corresponding anisotropies of CTE under stresses were investigated. With the increasing compressive stresses, the CTE of IG-110 and NBG-18 parallel and perpendicular to the loading directions increased significantly and decreased gradually respectively. The corresponding CTE anisotropies of IG-110 and NBG-18 almost maintain below 1.05 and keep their original near-isotropic properties under compressive stresses which are beneficial for the integrity and safety of the graphite used in the reactor.

Wednesday 11h30 Models of Coefficient of Thermal Expansion for Gilsocarbon Graphites Irradiated in Inert and Oxidising Environments

E.D. Eason[1], G.N. Hall[2], B.J. Marsden[2] and G.B. Heys[3]

[1]Modelling and Computing Services LLC, USA
[2]The University of Manchester, UK
[3] Health and Safety Executive (Nuclear Installations Inspectorate), UK

This paper presents empirical models of fast neutron damage and radiolytic oxidation effects on coefficient of thermal expansion (CTE) for the Gilsocarbon graphites used in Advanced Gas-cooled Reactors (AGRs). This is the next in a series of updated models of the properties of Gilsocarbon graphites being developed by the authors with funding from the UK Health and Safety Executive. Models of dimensional change and Young's modulus developed under the same programme have been presented at past INGSM meetings. An improved model has been developed from the currently-available test reactor data for estimating the change in CTE due to irradiation in inert environments. The new model shows an abrupt increase to an "upper shelf" irradiated CTE value at very low dose, then CTE values decreasing with increasing dose. The effect of radiolytic oxidation is modelled by shifting the inert model in both dose and CTE directions to agree with the CTE measurements on material trepanned from AGR moderator bricks. The shift in the inert model that is needed to match the trepanned data varies significantly by reactor. The new model predicts randomly-selected validation data that were not used in model fitting as well as it fits the calibration data.

Wednesday 11h50 The Biaxial Strength of NBG-18 Graphite

T.D. Burchell, R. Battiste and J. Strizak
Oak Ridge National Laboratory, TN, USA

NBG-18 graphite is one of several grades being considered for the core structures of the Next Generation Nuclear Plant (NGNP) which is currently being developed by the US Department of Energy. The biaxial strength of grade NBG-18 graphite was determined in the first (tension-internal pressure) and fourth stress (compression-internal pressure) quadrants. A total of 46 specimens from two production lots were tested at ten stress ratios. Each biaxial specimen had a companion tensile test specimen that was machined from the core of the biaxial specimen. Here we present details of the specimen geometry, test apparatus, and the test data from both stress quadrants. The tensile data is additionally reported. Moreover, the data in the first stress quadrant has been modelled using a combination of the Burchell fracture model and the Shetty mixed-mode fracture criterion. The predicted first stress quadrant failure envelope is compared with the experimental data.

Wednesday 12h10 Nuclear Graphite Reliability Analysis Investigation and Software Development Space-Act Agreement

N.N. Nemeth[1], J.P. Gyekenyesi[1], E.H. Baker[1] and R.L. Bratton[2]

[1]NASA Glenn Research Center, USA
[2]Idaho National Laboratory, USA

This presentation summarizes the work performed under Space Act Agreement SAA3-824 entitled “Nuclear Graphite Reliability Analysis Investigation and Software Development” between the NASA Glenn Research Center and the Battelle Energy Alliance (i.e. Idaho National Laboratory). There were two main objectives: (1) investigation of stochastic fracture models appropriate for nuclear grade graphite and (2) coupling the NASA developed CARES/Life (Ceramics Analysis and Reliability Evaluation of Structures) code with the COMSOL finite element analysis program. An overview is provided of results and findings from (1) a literature review of stochastic fracture models, (2) results from multiaxial strength prediction of H-451 graphite using the CARES/Life (Ceramics Analysis and Reliability Evaluation of Structures) code, (3) an analysis of statistical distribution of H-451 graphite from historical data – including some updated results, and (4) results of benchmarking CARES/Life analytical results between the ANSYS and COMSOL finite element analysis programmes.

Wednesday 12h30 Comparison of the Probabilistic and Deterministic Methods for the Stress Evaluation of Nuclear Graphite

H.T. Wang, Z.S. Zhang and L.B. Sun
INET, Tsinghua University, Beijing, China

Graphite is a major structural material in the core internal of high temperature gas-cooled reactors. To evaluate the structural integrity of the graphite components a probabilistic method is proposed in KTA 3232. On the other hand, the deterministic method is utilized in ASME. In this presentation, the probabilistic method is compared with the deterministic method, highlighting the relation between the allowable failure probability in KTA and the safety margin in ASME. The translation from the failure probability to the safety margin and the corresponding inverse translation depend strongly on the Weibull Modulus denoted by *m*,

which describes the scattering degree of the tensile strength of nuclear graphite. The numerical examples show that for m = 5~10, which is a typical value for the vibration-moulded graphite, the allowable failure probability required in KTA roughly matches the safety margin in ASME. In addition, the allowable failure probability is independent of m. This feature allows the probabilistic method to be readily used for a wide range of graphite grades.

Wednesday 14h00 Progress in Providing Improved Input Data to Graphite Brick Behaviour Modelling

P. Rogers, Frazer-Nash Consultancy Ltd, UK

The ageing and behaviour of graphite bricks in the UK Advanced Gas-cooled Reactors (AGR), both at the present time and in the future, are modelled using a range of predictive techniques. The capability of these techniques to represent true core behaviour is dependent, in part, upon the input data that they are provided with. Work is currently under way to reduce the uncertainties in several of these input data in order to improve confidence in the predictive techniques and support continued operation of the AGRs. Of particular note is recent work to provide updated graphite brick neutron damage and gamma heating data, and investigations to reduce uncertainties in graphite brick temperature predictions. This presentation provides a high level overview of the work area including current and future strategy, recent activity and justification for the on-going work programme.

Wednesday 14h20 Carbon Composite R&D for High Temperature Gas-cooled Reactors

Y. Katoh, L.L. Snead, K. Ozawa, C. Contescu, T.D. Burchell, S. Gonczy and W.E. Windes

Oak Ridge National Laboratory and Idaho National Laboratory, USA

The use of composite materials, either carbon composites or silicon carbide composites, is considered necessary in some of the reactor unit components for Generation IV high temperature gas-cooled reactors. Based on the analyses of composite materials needs and properties requirements performed by the reactor designers, a technical program toward the qualification of composite materials for use in high temperature helium-cooled reactors is being undertaken in the United States Generation IV Materials R&D program. This paper will provide an overview of the latest development of the technical elements related primarily with the carbon fibre composites research and development. Topics to be covered include the irradiation effects, environmental effects, and codes and standard.

Wednesday 14h40 Models of Bending Strength for Gilsocarbon Graphites Irradiated in Inert and Oxidising Environments

E.D. Eason[1], G.N. Hall[2], B.J. Marsden[2] and G.B. Heys[3]

[1]Modelling and Computing Services LLC, USA

[2]The University of Manchester, UK

[3] Health and Safety Executive (Nuclear Installations Inspectorate), UK

This paper presents an empirical model of fast neutron damage and radiolytic oxidation effects on bending strength for the Gilsocarbon graphites used in Advanced Gas-cooled Reactors (AGRs). This is the next in a series of updated models of the properties of Gilsocarbon graphites being developed by the authors with funding from the UK Health and Safety Executive. Models of dimensional change and Young's modulus developed under the same programme have been presented at past INGSM meetings. The available strength and modulus data on AGR graphites neither support nor conclusively disprove the practice of estimating strength from Young's modulus. However, the larger database now available makes it possible to estimate strength without considering Young's modulus. The model presented here is based on empirical evidence of essentially constant strength as fast neutron dose increases in inert environment. An exponential function is used to model the degree of radiolytic oxidation as measured by weight loss and the variation from one AGR station to another. The combined model of neutron damage and radiolytic oxidation is calibrated to bending strength data on material trepanned from AGR moderator bricks. The new model is found to be superior to alternative models that estimate strength from Young's modulus.

Wednesday 15h00 A 3D Methodology for the Generation of AGR Graphite Dosimetry Data

D.A. Allen, D.O. Morgan, A.T. Robinson, D.A. Thornton and G.S. Whiley[1]
Serco Ltd and [1]EdF Energy (British Energy) Ltd, UK

Historically, graphite dosimetry data for AGR cores have been provided using 2D approaches relying upon supercell Monte Carlo models of one or more channels. Appropriate boundary conditions were applied to represent symmetrical distributions of channels across the core. This approach cannot provide accurate data for channels in strongly asymmetric locations, such as at the edge of the core or adjacent to vacancies or on-lattice control rods. Recently, a new methodology has been developed to exploit the latest capabilities of the Monte Carlo code MCBEND. This method uses detailed 3D models which are capable of providing distributions of dose parameters across the graphite cores. The models developed include accurate representations of graphite in all locations including those with strong asymmetry. They are capable of calculating data in arbitrary geometries but, to improve efficiency, the separability of radial-azimuthal and radial-axial variations in dosimetry data has been demonstrated and exploited. This approach has been qualified by the consistent reproduction of results from explicit 3D calculations. The revised methodology has already been implemented for Dungeness B and for Hinkley Point B / Hunterston B. It is expected that the methodology will be applied to each of the AGR reactor designs.

Wednesday 15h20 Comparison of FEAT Predicted and Measured AGR Fuel Sleeve Residual Stresses

L. Chidwick and M. Joyce, Frazer-Nash Consultancy Ltd

Measured residual stress data exists in the British Energy (BE) Post Irradiation Examination (PIE) database of irradiated fuel sleeves that were discharged at varying levels of irradiation. Of these fuel sleeves, a significant number were VFT anisotropic grade graphite and thus it was recognised that these data could be used to validate the current Magnox material models. The majority of VFT data available were measured residual stresses, however, the measured strains were reported for a small sub-set. Predictions of residual stress and strain were made and material properties (dimensional change, Young's Modulus, etc.) used for these predictions were taken from available literature. A number of assumptions were made when fitting the VFT material data to the B++ model. Both primary and secondary creep were considered, however, recoverable creep was not. Good agreement was found between the measured peak tensile residual strains and the FEAT predicted residual strains. The equivalent predicted residual stresses were found to be lower than the reported measurements, this was due to the unknown weight loss at which the residual stresses were taken. It is possible that the weight loss when the measurements were taken was ignored, as this would have been a conservative assumption. This work was carried out on behalf of Magnox North Ltd.

Posters

P1 An Experimental Study of the Fracture Toughness of Nuclear Graphite IG-11

Li.Shi, H.T. Wang, H.T. Wang, L.B. Sun, Y.Q. Hu, X.F. Yao and C. Xiong
INET, Tsinghua University, Beijing, China

Graphite components used in High Temperature Gas Reactor (HTGR) are subjected to stresses arising from both external and internal sources, such as the seismic load, and differential shrinkage or expansion due to irradiation and thermal loads. At the late stage of the nuclear plant, the stresses of the graphite may lead to the fracture or large deformation, which will affect the size and integration of the reactor core. Fracture toughness of graphite thus has become an important issue in the safety analysis of HTGR. Nuclear graphite is a brittle, polygranular material, which brings difficulties to the research of its fracture toughness. In the present paper, an experimental work on measuring the fracture toughness of graphite IG11 is described. Three-point bending tests were employed. The effects of graphite specimen size, thickness, and the relative crack length a/W are studied in the tests. The deformation and stain at the crack tips are also measured. The results indicate that the fracture toughness of IG11 graphite is in a range of 0.82~1.27 MPa.m1/2, with an averaged value of 1.038 MPa.m1/2. The fracture toughness increases with the size of the graphite specimens.

P2 Irradiated Graphite Waste: Analysis and Modelling of Radionuclide Release with a View to Long-Term Disposal

G. Black, A.N. Jones and B.J. Marsden, The University of Manchester, UK

It is estimated that the UK graphite moderated reactors will produce ~90,000 tonnes of irradiated graphite waste when operation ceases, this is a significant quantity of material for which a treatment/disposal route has not yet been developed. It is known that the graphite waste contains a number of short and long lived isotopes of concern, including ^{3}H, ^{60}Co, ^{36}Cl and ^{14}C, and recent research has shown that the quantity of these isotopes can be significantly reduced using a number of methods, such as chemical leaching and thermal treatment. Although these treatments show promise, a full understanding of the isotope location and release mechanisms involved must be developed before the regulator will accept them as viable disposal options. These concepts will be investigated using both experimental measurements of irradiated graphite samples and computer simulation of the isotopes involved, with the final aim of validating the techniques and developing them as part of safe and economical disposal options for the UK.

P3 Understanding the Effect of Nuclear Graphite Porosity Size and Shape on Young's Modulus using Finite-Element Analysis

A.T. Bakenne, B. J. Marsden, A.N. Jones and G.N. Hall
The University of Manchester, UK

In the nuclear industry, graphite is used as a moderator and reflector material. It is also used for other features of reactor cores, such as fuel sleeves, spacer rings, shield-wall graphite. Examples of two possible graphite grades for Gen-IV technology include NBG-10 from SGL Carbon group and PCEA produced by GrafTech. Numerous graphites have been developed with anisotropic or near isotropic material properties. NBG- 10 is pitch coke extruded graphite near isotropic with a grain size of 1.6mm(max) while PCEA nuclear graphite is petroleum coke extruded graphite near isotropic with a grain size of about 0.8mm (max). The directional dependence is a result of the geometry of the filler particles used and the manufacturing process. The microstructures of un-irradiated (virgin) NBG-10 and PCEA consist of the filler particle, and pores of different sizes resulting from the manufacturing process (Figure 1). Under fast neutron irradiation the microstructure of nuclear graphite is significantly changed. These changes consequently affect the graphite bulk structural properties such as Young's modulus, strength, coefficient of thermal expansion (CTE) and thermal conductivity. Graphite dimensional and properties changes are a function of fast neutron irradiation and temperature, and can lead to component distortion and the generation of internal stress.

P4 Thermal Creep in Polycrystalline Graphite

L. Luyken, M. Figueroa[1], W. Windes[1], T.J. Marrow and B.J. Marsden
The University of Manchester, UK and [1]Idaho National Laboratory, USA

Although there are data and empirical relationships for thermal creep in various grades of graphite, there is no well founded scientific understanding of the mechanism involved in the process. This programme of work is aimed at not only obtaining thermal creep data but determining the underlying deformation mechanism involved and in particular to discover whether the process occurs heterogeneously within the binder and filler at the microstructural level. The experimental programme uses three methods to characterise the effect of thermal creep in polycrystalline graphite: • Size strain analysis of X-ray diffraction results will characterise thermal creep changes to the crystallites. • RAMAN mapping to provide high resolution strain maps of the sample surface have been collected to identify differences in strain between binder and filler regions. • High resolution tomography in conjunction with three-dimensional digital image correlation will detail the strains inside a volume of the sample. The thermal creep experiment presented here was carried out as part of collaboration between the University of Manchester and Idaho National Laboratory.

P5 High Temperature Young's Modulus and Bending Strength of Nuclear Grade Graphites

Eung-Seon Kim, Se-Hwan Chi and Sung-Deok Kim
Korea Atomic Energy Research Institute

For the design and safety analysis of a VHTR, high temperature mechanical properties are required. In this study, Young's moduli and bending strengths of commercially available nuclear graphites were measured and compared. An impulse excitation method was used for Young's modulus mesurement in Ar environment up to a temperature of 1200 deg C. Four point bending strength was measured in Ar environment up to a temperature of 1000 deg C. As a function of temperature up 1000 deg C, the Young's modulus was correlated with the bending strength for each grade. The difference among the grades was discussed based on the microstructures.

P6 Encapsulation of Irradiated Graphite Waste

B. Hagos, A.N. Jones, T.J. Marrow and B.J. Marsden
The University of Manchester, UK

The majority of irradiated graphite waste in the UK reactors originates from moderators and reflector materials, which are exposed to a high irradiation dose during operational lifetime. Irradiation damage induces dimensional change and other physical property changes to graphite. It is necessary to understand fully the microstructural and radioisotopic character of the graphite and proposed decommissioning or immobilisation treatments. Techniques such as encapsulation may limit the diffusion of radionuclides and may fully immobilise the waste rendering it suitable for future geological disposal. This research aims to develop and demonstrate advanced immobilisation of a range of graphite wastes using encapsulation materials, with the objective of reducing this activity of Intermediate Level Waste ILW to that of Low Level Waste (LLW) and possibly even to Ultra Low Level Waste (ULLW). Leach testing will be used in order to quantify the radionuclide release rate under geological repository conditions. To date non destructive high resolution characterisation techniques such as Electron Microscopy, Polarised optical microscopy and X-ray diffraction have been used to understand the microstructural properties of the virgin material and irradiated graphite prior to studying encapsulant materials.

P7 EdF Energy Graphite Core Work Programme

A. Whittle, J. Reed, D. Knowles and S. Brasier, EdF Energy (British Energy), UK

EDF Energy maintains the UK's fleet of ageing AGR reactors. Key to safe operation and life extension is an in-depth understanding of the Graphite Cores in each reactor. The Graphite Core Programme Team (GCPT) is charged with this understanding and the development of numerous safety cases in order to operate to the current planned life extension timescales. Graphite core knowledge is built up from the atomic level, through to the behaviour of samples (from trepanned AGR samples or experimental MTR projects), and then translated to brick and whole core models (software and scale models), and the effect of seismic events.

P8 *In-situ* Study of Primary Regime Proton Irradiation-Induced Creep in Graphite

A.A. Campbell[1], G.S. Was[1] and Y. Katoh[2]

[1]University of Michigan, USA
[2]Oak Ridge National Laboratory, TN, USA

The creep of graphite under high temperature neutron irradiation is critical to predicting the integrity of structural components of the Very High Temperature Reactor. *In-situ* proton irradiation-induced creep experiments were performed at the University of Michigan on graphite samples at 900°C at applied tensile stresses of 6MPa and 20.7MPa, and doses of 0.35 dpa and 0.23 dpa respectively. This poster will discuss the experimental setup, results from the two experiments, and finally compare and discuss the two experiments.

P9 **Acute Oxidation of Nuclear Graphite: Porosity Development and Loss of Strength**

C. Contescu, J. Strizak and T. Burchell, Oak Ridge National Laboratory, TN, USA

Because graphite has limited oxidation resistance at high temperature, it is important to understand the mechanism of porosity development caused by oxidation and the effect of oxidation on mechanical properties. Along with other physical methods for characterization of density gradients in oxidized graphite, optical microscopy coupled with automated image analysis is a powerful and convenient technique. We present recent results on mechanism of porosity development in PCEA grade graphite specimens oxidized in air in different conditions. The effect on mechanical properties is correlated with the penetration depth of oxidised layer. The latter is more a function of oxidation temperature than of the burn-off degree.

P10 **Investigating the Feasibility of Recycling Irradiated Graphite: A First Demonstration**

P. Pappano and T. Burchell, Oak Ridge National Laboratory, TN, USA

A graphite moderated, gas cooled reactor, whether it be a pebble or prismatic design, creates a significant amount of irradiated graphite during its lifetime. This quantity of radioactive structural graphite presents a disposal issue during change out or end of life events. A potential method for mitigating this storage problem is its use as a feedstock for the production of fresh, or recycled, graphite. A nuclear graphite is manufactured from petroleum coke and coal tar pitch. The manufacturing process for recycling irradiated graphite from a reactor would be identical, except that ground irradiated graphite replaces the petroleum coke feedstock. Oak Ridge National Laboratory has performed first of its kind recycling of irradiated graphite. The irradiated samples were ground and mixed with coal tar pitch, formed into cylinders and heat treated. No additional coke particles were used; the filler portion of the graphite was 100% ground irradiated material. The recycled graphites were then tested for numerous properties. It was found that the recycled graphite properties were lower in value then a commercially available nuclear grade graphite. This is because an impregnation step to densify and toughen the recycle piece was not performed as the equipment needed for such an impregnation was not available. This finding was confirmed by GrafTech International who, as part of this program, made pilot scale recycled graphites from unirradiated material. GTI found that a graphite billet made from 100% ground graphite had essentially the same properties as one made from only petroleum coke filler. It is also important to note that the irradiated graphites used as feedstock were irradiated to different temperatures and fluences. The irradiation temperature and fluence did not appear to effect the properties of the recycled graphite. That is, a recycled graphite made from high T_{irr} and fluence starting material had similar properties to a recycled graphite made from low T_{irr} and fluence material.

Additional posters may be exhibited