ACHIEVING THE RIGHT BALANCE BETWEEN CONSERVATISM, COMPLEXITY AND CONFIDENCE TO SECURE A SAFE AND EXTENDED AGR LIFETIME

15th – 18th October 2018
at
THE CASTLE GREEN HOTEL
KENDAL

PROGRAMME AND ABSTRACTS

organised under the auspices of
The British Carbon Group

and sponsored by
Wood Nuclear Ltd
EdF Energy Generation Ltd
Atkins Global
Frazer Nash Consultancy
National Nuclear Laboratory

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PREFACE

The United Kingdom has fourteen gas-cooled, graphite-moderated Advanced Gas-Cooled nuclear reactors that have approached the end of their design lives but are capable of additional safe operation subject to an improved understanding and continuing evaluation of their graphite core behaviour. All graphite moderated reactors suffer ageing and degradation to the graphite during service. The degradation poses a threat to the functionality of the graphite core with particular regard to refuelling and control-rod movement and, therefore, to the safe operation of the reactor. This is especially true against a background of an increased rate of keyway-root cracking now observed in certain reactors.

As the graphite components of these reactor cores cannot be replaced, it is important to ensure that effective strategies are in place to secure safe and reliable operation ultimately beyond their planned design life. The importance of extending the safe operating life of a reactor is increasing due to the political and socio-economic demands to reduce greenhouses gases and to diversify energy supply, along with the widely-publicised delays in providing replacement generation capacity as older reactors have been shut down along with fossil-fuelled plant. Thus, the importance of understanding reactor core graphite behaviour and demonstrating confidence in predictions become ever more significant as life extension continues.

The objective of the present conference is again to display and to discuss the extensive research and analysis performed by a number of expert organisations in support of the EdF-Energy Advanced Gas-Cooled Reactors, and especially to consider where uncertainties lie in predicting core graphite performance: thus, on this sixth occasion, we consider the need to achieve confidence in prediction whilst accommodating sufficient conservatism in data predictions to guarantee a safe outcome. A key feature of this conference is that time is again allocated within the programme for debate on three important and related issues: through these, and through the main programme of presentations, the conference seeks both to obtain independent peer review of its technical developments and to identify where there remain deficiencies in modelling and prediction.

This conference is the most recent in a series, with the first five being held in Cardiff, Nottingham, Aston, again in Nottingham and then Southampton in 2005, 2008, 2011, 2014 and 2016 respectively. The proceedings of these earlier conferences can be found in the following books.

1. Neighbour, G. B. (Editor) (2007). Management of Ageing in Graphite Reactor Cores (308 pages). RSC Publishing (Royal Society of Chemistry, Cambridge) (ISBN: 978-0-85404-345-3).
2. Neighbour, G. B. (Editor) (2010). Securing the Safe Performance of Graphite Reactor Cores (270 pages). RSC Publishing (Royal Society of Chemistry, Cambridge) (ISBN - 978-1-84755-913-5).
3. Neighbour, G.B. (Editor) (2013). Modelling and Measuring Reactor Core Graphite Properties and Performance (211 pages). RSC Publishing (Royal Society of Chemistry, Cambridge) (ISBN 978-1-84973-390-8)
4. Flewitt P.E.J. and Wickham A.J. (Editors). Engineering Challenges Associated with the Life of Graphite Reactor Cores (477 pages). EMAS Publishing (Warrington) (ISBN: 978-0-9576730-5-2)
5. Flewitt P.E.J. and Wickham A.J. (Editors). Science and Engineering in Collaboration to Support Safe Operation of the Graphite Reactor Cores (417 pages). EMAS Publishing (Warrington) (ISBN: 978-0-9576730-4-5)

PROGRAMME AND ABSTRACTS

ACHIEVING THE RIGHT BALANCE BETWEEN CONSERVATISM, COMPLEXITY AND CONFIDENCE TO SECURE A SAFE AND EXTENDED AGR LIFETIME

A Conference at The Castle Green Hotel, Kendal

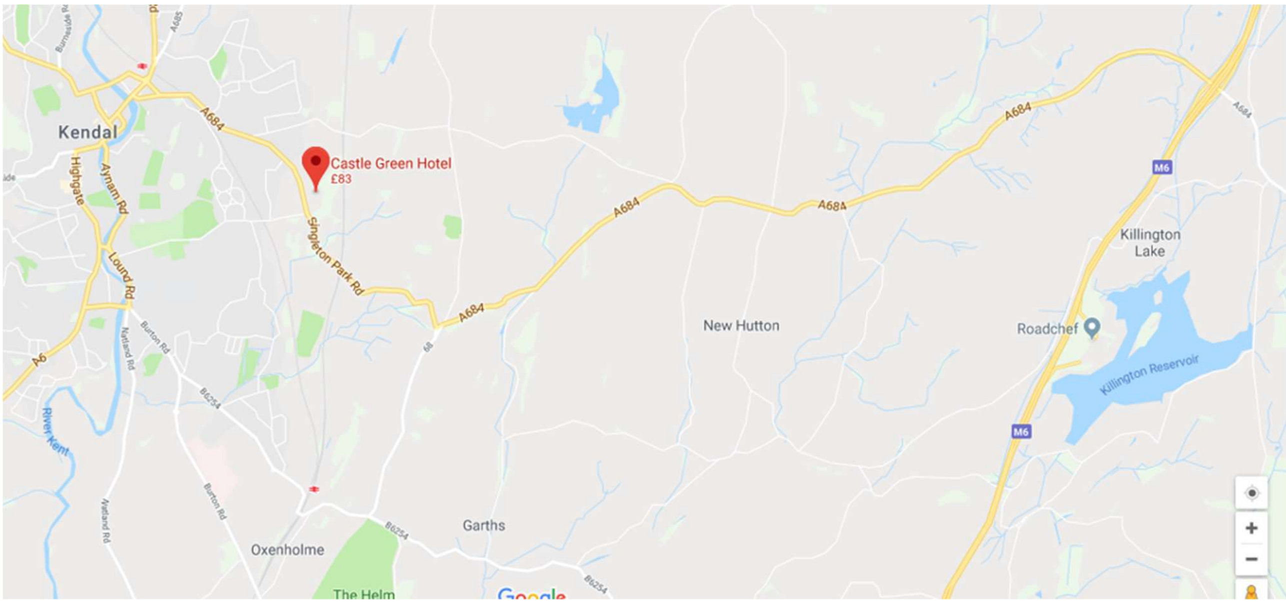
15th – 18th October, 2018

under the auspices of The British Carbon Group

REGISTRATION

Registration will begin at 1400 hrs on Monday October 15th 2018 in the Function Suite Foyer at the hotel. Access to rooms cannot be guaranteed before this time, but a secure baggage store is available.

Car parking is available at the hotel (at own risk). Drivers are recommended to leave the M6 at Junction 37 and then take the A684 towards Kendal. Upon sighting the town, the hotel driveway will appear on the right-hand side. The venue is 5 minutes by taxi from Oxenholme railway station, on the west coast main line from Euston. An hourly service to Edinburgh via Birmingham and Crewe calls here.



From London Heathrow airport, travel to Euston via London Underground – Piccadilly line to Warren Street, then Northern line. From Gatwick airport, take a Southern or Gatwick Express train to Victoria, then London Underground Circle Line (either direction) to Euston Square.

A secure baggage storage area will be available on the final day of the conference since rooms must be vacated at 10am on Thursday October 18th.

Free WiFi is available to registered delegates in all areas: obtain the code from main reception upon arrival. The hotel offers an indoor swimming pool and fitness facilities. All rooms have tea and coffee making facilities. Please note that any issues arising in connection with your hotel accommodation must be taken up directly with the hotel management: the conference organisers accept no responsibility in this regard.

All further enquiries regarding the conference arrangements should be directed to the conference office in the Function Suite Foyer. Conference Manager, Tony Wickham, confer@globalnet.co.uk, (07785 567 577); or Sam Church (07969 840 681).

PRESENTATIONS & AUDIO/VIDEO FACILITIES

Presentations will be 15 minutes plus 5 minutes for discussion, except for the Plenary Lecture. All presentations will take place in the Function Suite (North).

Presenters are encouraged to bring their presentations on a USB memory stick and not to use their own computers. Whilst the latest PowerPoint software and computer facilities will be available, presenters should check their presentation prior to their session. To facilitate smooth running of the programme, presenters are also encouraged to introduce themselves to the session chairpersons prior to their session.

A copy of all presentations must be left on the computer so that they may be collected and issued to delegates before the end of the conference.

CATERING ARRANGEMENTS & CONFERENCE DINNER

Breakfast will be served in the Hotel Restaurant. Coffee breaks and lunch will be available in the conference room, and dinner will be served in the Function Suite (South). Breakout sessions on Wednesday will also be located in the Function Suite. Delegates should retain their conference badges at all times and may be requested to show their badges to obtain service.

Any alcoholic beverage ordered ‘at table’ in the restaurant is at your own cost, and accounts for items charged to your room must be settled with the hotel upon checkout.

A Gala Conference Banquet will be held on the Wednesday evening in the Function Suite South. Dress code for this event is business casual.

Those with special dietary needs are asked to make themselves known to the waiting staff in the restaurant, whereupon their requirements will be met.

Monday 15th |October 2018

14.00: Registration Desk Opens (Function Suite Foyer)

18h30: **The British Carbon Group AGM (all are invited)**

19.00: **Chair: Ms Athanasia Tzelepi** (NNL)

"Summary of Hunterston B Inspections in 2018 and Consequences for Future Operation"
Dr James Reed, Chief Graphite Engineer, EdF Energy Generation Ltd

19.45 Pre-dinner drink

20.00 Dinner

Tuesday 16th October 2018

07h30: Breakfast (Hotel Restaurant)

08h30: Conference Service Desk Opens, Function Suite Foyer

09h00: **SESSION: Graphite Properties**
Chairman: Dr Tjark van Staveren (NRG)

09h00: **Materials Test Reactor Programme: 'BLACKSTONE' Phase 3**
Van Staveren T., Knol S., Chidwick L., Gray R., Joyce M., Brown M. and Matthews P.

Project Blackstone is EDF Energy's materials test reactor program that provides data on material property changes due to fast neutron irradiation and radiolytic oxidation in advance of the operating Advanced Gas-cooled Reactors (AGRs). The project makes use of the High Flux Reactor in Petten to achieve accelerated materials aging under conditions similar to those in an AGR. The primary objective is to produce and characterise Gilsocarbongraphite, for each of the station graphite material grades, at the expected AGR end of generation conditions.

Within Blackstone Phase 3, two irradiations are performed sequentially with graphite specimens loaded into Capsules 05 and 06 respectively. Capsule 05 has been designed, built, commissioned and loaded in a high flux position of the HFR. At the time of writing, Capsule 05 is currently undergoing irradiation in the HFR.

The first part of this presentation will highlight the aims and objectives of the Phase 3 experiment to support AGR life extension, the specimen selections for design and the expected experiment outcome. The second part of the presentation will describe the experiment assembly, the thermomechanical design aspects and present the measured and controlled irradiation performance.

09h20: **Assessment of Existing PIE Techniques for Measuring End of Reactor Life Graphite Properties**
Wilkinson S., Tzelepi N., Shaw T., Brown M. and Crelling M.

All Advanced Gas-cooled Reactor (AGR) graphite sampling post irradiation examination (PIE) campaigns are measured at the NNL Windscale Laboratory. The measured property data from these inspection campaigns is used in assessments to support the continuing safe operation of the AGR fleet. As the stations move closer to the end of life core states, it is anticipated that there will be trepanned graphite samples with challenging geometries and surface finish. It is therefore important to consider whether the current PIE techniques can continue to produce reliable and reproducible data for highly degraded graphite samples. The major impact for all techniques will be sample weight loss due to oxidation. However, dynamic Young's modulus, thermal conductivity and coefficient of thermal expansion techniques will also be noticeably affected by the changes in sample properties due to irradiation. The presentation examines the risks associated with such measurements and suggests possible mitigations when measuring samples at high weight loss, and dose, remotely in a shielded facility. This utilizes and builds on previous experience and findings from Magnox graphite PIE and Materials Test Reactor experiments. It also investigates the implications of any changes to measurements techniques and what is practical.

09h40: **Comparison of Gas Permeability Measurement Methods for AGR Graphite Simulants**

Dinsdale-Potter J., Charlton D., Crelling M., Copeland G., Shaw T. and Tzelepi A.

Gas permeability is an important property for Advanced Gas-cooled Reactor (AGR) graphite weight loss predictions and forecasts, characterising how the (carbon dioxide) coolant permeates through the graphite bricks, which make up the reactor core. This presentation shows the results of a comparative study of two different methods for measuring gas permeability. These methods are the active transient or 'vacuum leak' method, which is an established method utilized in the historical AGR oxidation studies and in the Magnox inspection and sampling campaigns, and a novel static equilibrium method currently under development at NNL.

The active transient method uses only one gas and utilizes the principle of equalizing pressure across two sealed chambers, connected by the sample, whereas the static equilibrium method utilizes the principle of equilibrium of fluxes under pressure driven flow for two gases in counter flow to determine the mass transport rate. The static equilibrium method operates at a much lower differential pressure than the cross-brick pressure drops that are typically found in the reactor core. This offers two advantages: firstly the low-range differential pressure transducer ensures better precision at zero pressure differential, required to measure gas diffusivity. Secondly this allows the trepanned sample to be sealed with removable polyolefin, unlike the active transient method which requires the samples to be encased in resin and hence prevents further measurements.

Three Gilsocarbon samples and two PC45 samples were used to study geometry effects in the entire range of anticipated permeability values. This proof of concept project is intended to form the basis for using the static equilibrium method to measure permeability as part of the routine graphite inspection and sampling campaigns.

10h00: **Dimensional Change of Gen IV Nuclear Graphite using Ion-Beam Irradiation**

He Z., Smith A., Theodosiou A., Jones A. and Marsden B.

The development of Generation IV reactors requires development and understanding of new nuclear graphite grades and as such their dimensional change under fast neutron irradiation is critical to their application. This has previously been obtained by using materials test reactors (MTR), however, MTR irradiation is difficult due to access, expensive and time. Ion beam irradiation shares the comparable damage mechanisms with fast neutron irradiation and is frequently used as a surrogate for fast neutron irradiation without resulting in radioactivity. Limited by ion projection range in material, it is difficult to measure the dimensional change induced by the ion beam irradiation, however we present a new method is designed to estimate the dimensional change behavior of graphite based on measuring the bending of thin graphite samples caused by ion beam irradiation. By calculating the stress and strain gradient in the curved sample with a finite element model, the dimensional change of graphite under irradiation can be can be estimated. The parameters of ion beam irradiation will be optimized by varying the ion beam species and energy according to the existing date from fast neutron irradiation. Nano-CT and TEM will also be applied to observe the microstructure response to the irradiation and compare the difference between ion beam irradiation and neutron irradiation.

10h20: **Understanding the Mechanisms of Retention of Fission Products in Virgin and Irradiated Gen IV Nuclear Graphites**
Theodosiou A., Jones A., Sharrad C. and Marsden B.

Graphite has been used for neutron moderation and structural support since the beginning of the nuclear reactor era, there is a renewed interest in graphite motivated by its use in the next generation of High Temperature and Molten Salt Reactors. The retention of the activated fission products (FP) is paramount during normal operating and accident conditions, and a mechanistic understanding of the physio-chemical and transport processes is vital for predicting the release rates and designing appropriate barriers to ensure inherent safety and lifetime performance.

A new research partnership funded jointly between the DOE (USA) and EPSRC (UK) aims to study the mechanisms of retention and transport of fission products in virgin and irradiated nuclear graphite. At Manchester we will study the microstructural and spectroscopic properties of virgin and irradiated graphite. In particular, this project aims to obtain an insight into FP behaviour using TEM tomography in order to understand the migration, retention and release of particular FP's in graphite. Complimentary analysis such as X-ray photon spectroscopy (XPS), polarized optical microscopy and diffractometry will also be used to understand FP interactions.

The highlighted graphite grades are POCO (ZXF-5Q/AXF-5Q), IG-110, NBG-18, PCEA along with older graphites such as PGA and Gilsocarbon. The project will focus on several fission products, known to be mobile and potentially problematic in high temperature environments; these are Iodine (I), Caesium (Cs), Krypton (Kr), Strontium (Sr), Ruthenium (Ru), Silver (Ag) and Europium (Eu). The various fission products will be implanted into the graphite specimens and the samples will be analysed in order to understand the chemical and physical interaction between the graphite and the fission products, along with the diffusivity behavior of each fission product through the host lattice.

10h40: *Coffee Break*

11h00: **SESSION: Graphite Failure**
Chairman: Dr Mark Joyce (Frazer-Nash Consultants)

11h10: **Size Effects on Nominal Net Section Fracture Strength in Unirradiated Gilsocarbon Containing Notches**
Ye R., Novovic M. and Bowen P.

Nuclear graphite Gilsocarbon is used as the neutron moderator, reflector and also a structural material in the UK's advanced gas cooled reactors (AGRs). The graphite bricks are locked in position through a key-keyway locking system. However, the keyways in graphite bricks also act as stress raisers, which increase the local effective stress and may promote keyway root cracking (KWRC).

For nuclear graphite material, the fracture strength of plain samples has been found to first increase with increasing sample size and then decrease with further increases of sample size. It is of great interest to find out whether such a size effect is also present for samples containing notches. Small scale laboratory samples containing simulated keyway notches have been used to study the nominal net section fracture strength of unirradiated Gilsocarbon. Understanding the structural size effect of notched Gilsocarbon is important for correct predictions of net section fracture strength in reactor bricks. In order to minimise all other factors on nominal net section fracture strength, geometrically similar unirradiated Gilsocarbon samples containing simulated keyway notches were used in this current study and the results will be presented.

11h30: **Fracture Properties and Size Effects of Irradiated Graphite**
Jordan M., Dinsdale-Potter J., Wilkinson S., Preston H. and Tzelepi A.

Keyway root cracking is a major life-limiting factor for the advanced gas-cooled reactors (AGRs). The main material properties required for fracture prediction are strength, stiffness, notch sensitivity and the work of fracture (WOF). Methodologies for measuring the flexural strength, stiffness and WOF of ex-reactor irradiated graphite have been developed. Due to the nature and source of the material the sample sizes are restricted, raising questions of representability of the core properties. Data on notch sensitivity of irradiated graphite is limited, and further questions arise over the suitability of testing using small notched beams.

Irradiated graphite from AGR installed sets (Gilsocarbon) and Magnox (PGA) large cores were tested destructively to investigate the effects of size and corner notch geometries, and demonstrate the interdependencies of the four properties for fracture prediction. Having successfully machined high weightloss graphite to the required fracture geometries, measurements of dynamic Young's modulus (DYM) indicated a consistent trend with density and both properties correlated linearly with WOF. Testing on a range of beams and 'wing' samples further demonstrated WOF size insensitivity. Conversely, for plain beams (6 x 6 x 19 mm compared to 8 x 8 x 38 mm) a size dependence of the fracture strength was found. While there was a significant scatter in the data, the crack initiation strengths of notched Gilsocarbon beams also appear to show size dependence, but not the corresponding PGA beams.

11h50: **Localized Deformation in Virgin Nuclear Graphite, observed by Digital Volume Correlation of Limited-Angle Synchrotron Tomographs**
Zillhardt T., Liu D. and Marrow J.

The development of keyway root cracking of graphite is a potential limiting factor for the lifetime of the UK AGRs. To understand how this may be affected by microstructural differences, it is important to study how damage occurs and is accommodated at the microstructural level, prior to fracture. Gilsocarbon graphite is a heterogeneous material that contains many defects at different length scales. These defects, which are distributed in both the matrix and the filler particles, play an important role in the damage development that contributes to strain accommodation. Advanced characterization techniques can now provide a clearer view of this and allows discrimination of damage development at a microstructural level.

An experiment at the UK Diamond Light Source (beamline I12) provided high-resolution limited angle tomography of a virgin graphite disc (20 mm diameter) tested under diametral compression, using a conventional mechanical test frame of high load capacity. The data, at a pixel resolution of 3.2 µm, have been reconstructed with advanced iterative algorithms to provide higher quality tomographs than previously obtained. These could then be processed via digital volume correlation to map the internal displacements with high precision and good spatial resolution. It is clearly observed that the deformation of the microstructure is heterogeneous, with most of the applied strain is taken up by the matrix rather than the filler particles.

12h10: **Failure from Notches in Irradiated and Virgin Graphite**
Treifi M., Jordan M., Tzelepi N. and Mummery P.

In order to predict the onset of cracking at keyway roots of irradiated graphite bricks, large scale mechanical tests have been performed on whole or sections of virgin graphite bricks. This allows the effect of the notch geometry on the failure load to be determined, which has been termed the notch strengthening factor. This is then conflated with the change in material strength due to irradiation and radiolytic oxidation to produce a value of the crack initiation stress at the corner. For this analysis to be robust, it must be assumed that the fracture behaviour of virgin graphite is the same as irradiated graphite. As the mechanical behaviour of graphite is affected by irradiation, this assumption should be tested; this study is such an attempt.

The strength of small (6x6x18 mm) plain-sided and notched beams of virgin and irradiated graphite was measured in 3- or 4-point bending at the National Nuclear Laboratory. The irradiated material was sourced from installed sets at Hinkley Point B. Finite element analysis of the mechanical tests was performed to determine the notch strengthening factors (Figure 1).

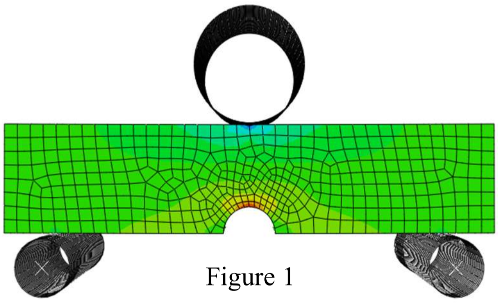


Figure 1

The notch strengthening factors of irradiated and virgin graphites were different. The virgin graphite notch strengthening factor was larger than that of the irradiated graphite, with the virgin notch strengthening factor about 1.9, and the irradiated notch strengthening factor about 1.4. This is probably due to the limited plastic deformation that is accessible to virgin graphite that allows energy from the deformation process to be absorbed by non-crack growth mechanisms. This casts doubt on the validity of the assumption.

12h30: *Lunch*

13h30: **SESSION: Physical Modelling of Cracking**
Chairman: Dr Peter Robinson (Quintessa)

13h30: **Improved Understanding of Crack Propagation in AGR Graphite through the Development of Novel Modelling Techniques**
Booler, O.

The understanding of fracture and crack propagation in nuclear graphite components is an interesting yet difficult topic. The question “when and where graphite bricks will break?” can be answered with established stress analysis and finite element models representing the irradiation damage mechanisms of graphite bricks through the operating life of the reactor. Subsequently, the question “What rate will the graphite bricks crack?” can be answered by considering the inherent variability and uncertainty of the graphite material properties and behaviour. However, the question “What shapes of cracks can be expected and what is the stability of said cracks?” poses a significant challenge. Demonstrating an ability to predict the fracture behavior of graphite components will increase confidence in the understanding of the evolution of the ageing core. Efforts have been ongoing in this area for some time to develop robust numerical modelling tools to tackle this issue. Recent developments and future prospects look promising. This paper outlines the beginnings, recent developments and future expectations in the modelling techniques employed to improve the understanding of graphite fracture and thereby increasing the confidence in predictions.

13h50: **Cracking of AGR Graphite Components: Advances in Computational Modelling**
Pearce C., Kaczmarkzyc L., Martinuzzi P. Baylis S. and Steer A.

Understanding the behaviour of AGR graphite cores with multiple cracked bricks is paramount to the assessment of structural integrity, safe operation and life extension. In this paper the latest developments in the finite element modelling and simulation of crack propagation are presented.

Configurational mechanics (CM) provides the theoretical basis for our work on crack propagation. This approach has a strong physical motivation, exploiting the 1st and 2nd laws of thermodynamics to establish crack front equilibrium and the crack path direction. We have also developed the numerical techniques to implement this theory within a finite element analysis framework (MoFEM). We are able to simulate propagating cracks in 3D solids that are discretely and continuously resolved by adapting the FE mesh in a smooth manner (exploiting the crack front equilibrium condition), thereby avoiding the need for element splitting or enrichment. This paper presents an extension of this work to account for crack propagation driven by the internal stress states that develop in the graphite bricks.

The performance of the formulation and its numerical implementation in MoFEM will be demonstrated with a number of benchmark problems. It will be shown to be computationally efficient, robust and physically accurate.

14h10: **Cracking of AGR Graphite Components: An Integrated Modelling Approach**
Martinuzzi P., Gangnant A., Baylis S., Steer A., Kaczmarkzyc L. and Pearce C.

Understanding the behaviour of AGR graphite cores with multiple cracked bricks is paramount to the assessment of structural integrity, safe operation and life extension. In this paper, the capability to perform crack propagation studies in irradiated graphite is demonstrated. A series of tools, integrated in the in-house EDF platform SalomeMeca, is used.

The behaviour of graphite with radiolytic oxidation and fast neutron irradiation has been studied extensively by EDF Energy and its partners. In particular, Wood developed an ageing model that was implemented as an ABAQUS UMAT routine. This routine can also be utilised in the finite element software code_aster. By chaining this ageing analysis with the crack propagation tools of MoFEM, the effect of cracks on the local behaviour of the reactor can be studied for any particular moment in time of the reactor's life.

The process is rendered as efficient and user-friendly as possible in order to reduce the complexity of the model. It opens the path to large scale parametric analyses, enabling multiple scenarios (such as the number and location of cracks) to be considered, to improve the prediction in the deformation of the core by finding best-fit material parameters, and reduce the degree of conservatism of current industrial models.

14h30: **Influence of Contact on Crack Propagation in Nuclear Graphite**
Athanasiadis I., Ullah Z., Kaczmarkzyc L. and Pearce C.

Numerical investigations have been performed to better understand crack propagation in nuclear graphite by matching laboratory test results. However, influence of frictionless contact between interstitial keys and nuclear graphite brick on fracture is not well understood yet. The present study focuses on investigation of the effect of key-brick frictionless contact on quasi-static crack propagation.

A novel unilateral smooth mortar contact active set strategy is proposed. This has been tailored for hierarchical basis functions to support integration with the existing crack propagation approach in MoFEM. The surfaces of two potential interacting bodies are denoted as master and slave surfaces. To solve the constrained contact problem, Lagrange multipliers are defined on slave surfaces. Under the assumption of small tangential displacements, triangles located on master and slave surfaces are identified only at the beginning of an analysis. Once potential contact of triangles is detected, each master and slave triangle pair is used to create special prism elements. These special elements are used to perform mapping of displacement shape functions from master to slave surfaces and numerical integration to evaluate mortar contact matrices. At each step of an analysis, prisms can either be in an active or passive state (representing contact or gap).

The proposed model is demonstrated with the analysis of a nuclear graphite brick slice with one initial crack located at a keyway corner. Loading is applied through contact with the four interstitial keys - two keys are fully fixed, and a force is applied on the other two. Two initial crack positions are investigated, and results are compared to results from tied meshes.

14h50: **AGR Brick Model Calibration and Prediction Using a Response Surface**
Pogson M., Bond A. and Robinson P.

Keyway root cracking of reactor bricks is an important consideration for the continued operation of AGRs. Quintessa has developed a COMSOL Multiphysics® model of individual bricks to provide a route to predict keyway cracking rates for EDF Energy. Monte Carlo simulation is required to account for variability in brick behaviour, resulting in long run times. A simple response surface model has been developed to calculate bore shapes and cracking times which accurately fits the model outputs and is suitable for use in optimisation methods.

Parameter distributions for brick properties have been defined to represent realistic ranges based on AGR measurements, without global calibration. We will present a method which uses the response surface models to optimise the fit of bore shapes and primary keyway root cracking rates against observations, and hence calibrate the parameter distributions to provide more accurate model predictions.

The method provides a straightforward means of comparing input data with the required calibration, building transparency and confidence in the modelling process. It also provides a means to estimate confidence bounds for comparing model predictions with measurement data using limited samples and to estimate the probability of keyway root cracking for a range of bore shapes and burnup times.

15h10: **A Data-Led Numerical Investigation of Keyway Root Cracking At-Shutdown**
Bond A. and Robinson P.

Quintessa has developed a diverse modelling capability for predicting brick shapes, stresses and hence keyway root cracking times using COMSOL Multiphysics®. These models show that stresses associated with keyway root cracking will increase significantly when the thermal strain associated with 'at-power' conditions has been reduced. The magnitudes of the stress changes are sufficiently large that primary keyway root cracks are expected to occur at shutdown for the Hinkley Point and Hunterston reactors. However, even using the simplified statistical models to define parameters, these models are very complex and thus require additional support to build confidence in key conclusions drawn from modelling studies.

In this paper we will show how a simplified 'data-led' modelling investigation is of value in demonstrating that primary keyway root cracking occurs at shutdown for these reactors. The analysis presented will look only at the stress/strain change associated with the thermal transient, with parameterisation only using information derived from direct reactor observation wherever possible. Using highly conservative estimates of key parameter uncertainty and variability, we will show that it is extremely likely that keyway root cracking will occur at shutdown. Detailed comparisons with the full models show good agreement which builds confidence in the understanding of primary keyway root cracking which will be an important part of demonstrating the continued safe operation of the reactors as lifetimes are extended.

15h30: *Coffee*

16h00: **Development in the Assessment of AGR Graphite Brick Damage Using the CBNA Model**

Teng H., Slater I., Jones C., Baylis S. and Wright J.

In some of the UK's Advanced Gas-cooled Reactors (AGR), some cracking of the graphite core components has been observed, as the components continue to age with irradiation. It is essential to understand the cracking behaviour, particularly keyway root cracking, to help underwrite the safety cases for operation of the cores to their ultimate lifetimes, and enable EDF Energy to take informed lifetime investment decisions. An integrity analysis tool named as the Cracked Brick Neighbourhood Array (CBNA) has been developed to assist this understanding.

The CBNA model is a 3-D finite element array model which is configured with several different types of graphite components including fuel brick, interstitial brick, filler brick, loose bearing key, spacer key and filler key. The contact interaction between the components was fully included using a 'general contact' concept; the evolution of the irradiation and oxidation of the graphite components were simulated with a user material subroutine. The CBNA model can accommodate different combinations of crack location, crack starting time, and the different fuel brick geometries; and is capable of modelling both at-power and shutdown processes. The CBNA model has been successfully used to investigate the interactions for a number of cracked brick configurations in the UK Hunterston B (HNB) graphite core.

Over the past years a number of improvements have been carried out on the CBNA model. These improvements include:

- Optimizing the mesh refinement
- More realistic crack insertion
- Use of latest material property and field variables

These improvements were aimed at making the computational analysis more efficient, reducing the conservatism, and making more realistic predictions. This paper presents details of the improvement of the CBNA model and application to the Hunterston B (HNB) graphite core. Some indicative results are provided.

16h20: **Expectations for Crack Opening Post-Onset of Keyway-Root Cracking**

Davies B., Baylis M. and Bradford M.

Irradiation induced changes to graphite dimensions and the coefficient of thermal expansion lead to the generation of tensile stresses at the keyway roots of fuel bricks late in operational life. Eventually these internal stresses, combined with external loads that arise from interaction with adjacent bricks, are sufficient to generate a full height axial crack. The irradiation induced material property changes continue to develop after cracking and this causes cracks to progressively widen during further operation.

Whole core models such as AGRIGID and GCORE are used to assess the graphite core tolerability to component damage, of which an important aspect is the component geometry which is affected by cracking and crack opening. As such, an ability to predict component geometry is required for the graphite core safety case.

Brick cracking and the behaviour of bricks post-cracking has been modelled using ABAQUS for both an isolated single fuel brick and for a fuel brick within an array (CBNA) and the crack widths are an output of this. In addition to the modelling we have observations of fuel bricks which have been inspected two or three times since cracking and the rate of crack opening at the bore can be tracked.

This paper presents a method that uses modelling and inspection data to forecast crack widths that can be used to inform whole core damage tolerance assessments within the graphite safety case.

16h40: **SESSION: Core Component Degradation Experiments**
Chairman: Mr Laurence Poulter (ONR)

16h40: **The Robustness of AGR Graphite Components (Part 1)**
McLachlan N., Baylis S., Gilliver B., Pountney S., Hutchinson P., Birkett P., Walker P. and Salih H.

In the AGR graphite cores, and indeed other graphite moderated reactors, there are some forms of loading which have not been widely assessed, either in terms of their magnitude, or their potential effect on component integrity. This is largely because such loadings were considered to be well below any that might challenge component integrity.

These forms of loading include compression along their edges and corners at shallow angles, localised loading along edges, pressure and temperature cycling in ranges and numbers comparable to shutdown and at power in a typical operating reactor, rapid pressure transients, temperature cycling to ranges beyond normal operation, internal stresses due to manufacturing processes and impacting. Where these have been investigated elsewhere, they have been on small samples.

This is the first of two papers, together covering the context, overview of work programme, structural analyses and results of experiments addressing these forms of loading, and specifically on blocks of nuclear-grade graphite close to the full size of reactor components.

17h00: **The Robustness of AGR Graphite Components (Part 2)**
McLachlan N., Baylis S., Gilliver B., Pountney S., Hutchinson P., Birkett P., Walker P. and Salih H.

In the AGR graphite cores, and indeed other graphite moderated reactors, there are some forms of loading which have not been widely assessed, either in terms of their magnitude, or their potential effect on component integrity. This is largely because such loadings were considered to be well below any that might challenge component integrity.

These forms of loading include compression along their edges and corners at shallow angles, localised loading along edges, pressure and temperature cycling in ranges and numbers comparable to shutdown and at power in a typical operating reactor, rapid pressure transients, temperature cycling to ranges beyond normal operation, internal stresses due to manufacturing processes and impacting. Where these have been investigated elsewhere, they have been on small samples.

This is the second of two papers, together covering the context, overview of work programme, structural analyses and results of experiments addressing these forms of loading, and specifically on blocks of nuclear-grade graphite close to the full size of reactor components.

17h20: **Shaking-Table Testing to Explore Progressive Failure of Graphite Bricks in an AGR Core**

Crewe A., Dihoru L., Dietz M., Horseman T., Turton L. and Taylor C.

Advanced Gas Cooled Nuclear Reactor (AGR) cores comprise of many graphite components whose geometry and mechanical properties change under prolonged exposure to neutron irradiation. The changes in the mechanical properties of the graphite have the potential to result in cracking of the graphite bricks later in the operational life of the core. This could result in disruption to the core geometry with possible negative implications on fuel cooling and/or control rod insertion. This component ageing issue needs addressing in both the computational and the physical models employed in the seismic resilience assessments. This paper looks at the development of methods to measure the dynamic loads in the model AGR bricks during physical seismic testing at Bristol University. The paper then explores the issue of the potential for progressive failure of multiple graphite bricks during a seismic event as dynamic loads get redistributed around the core. A variety of model AGR core bricks have been designed measure forces in the array as well as some bricks that will 'crack' when a predefined force is applied to them. For practical reasons it was not appropriate to create such a brick using modelled materials, and it was also desirable that the bricks could be repeatedly reset and allowed to crack again in subsequent tests. These "Crack on Demand" CoD bricks are therefore manufactured from Acetal in two halves.

At the start of a seismic test the brick halves are held together using high strength electromagnets which connect the brick halves via a load cell. During the tests the axial and shear loads between the two brick halves are monitored by the electronics within the brick. When the applied loads exceed the 'breaking force' of the brick, the electromagnets are automatically switched off and the two halves of the brick are allowed to move freely. This paper describes the design and calibration of the various bricks incorporating load measurements and outlines the results from initial testing of an AGR model including these bricks. An initial assessment of the potential for progressive failure of graphite bricks in an AGR Core during seismic excitation is also discussed.

17h40: **Use of Image Recognition to Investigate Crack-Pattern Effects on Seismic Performance of an AGR Core**

Turton L., Crewe A., Dihoru L., Dietz M. and Horseman T.

This paper describes how motion-based multiple object tracking can be used to quickly visualise and interpret large multidimensional experimental data sets. These data sets were produced by a series of experimental tests investigating the response of the UK's Advanced Gas Cooled Reactors (AGRs) to seismic inputs. At the University of Bristol, a quarter-scale single layer array (SLA) with the same layout as a layer of a real AGR was produced. This array was used to investigate the effect of cracking on the seismic resilience of AGRs. Experimental testing involved the placement of a set of cracked bricks to explore the effect of crack number and configuration on seismic performance. The bricks were tracked using a high-speed video system, and multiple object tracking was used to quantify the movements of each brick. Data challenges included processing the large video files efficiently and tracking many hundreds of points moving dynamically and simultaneously. The influence of cracking on response is discussed. Multiple object tracking of dynamic video has many applications toward interpretation of video in other disciplines such as surveillance.

18h00: ***In Situ* Crack Growth in Virgin and Irradiated Graphite**

Krishna R., Wade J., Taylor A. and Mummery M.

It is well known that fast neutron irradiation introduces defects into the structure of graphite that, in turn, significantly affect the mechanical behaviour. This can be confirmed by performing mechanical tests, with a measure of the work-of-fracture given by the stress/strain response. In this study, the effect of irradiation on the crack growth behaviour is studied *in situ* in 3D for the first time.

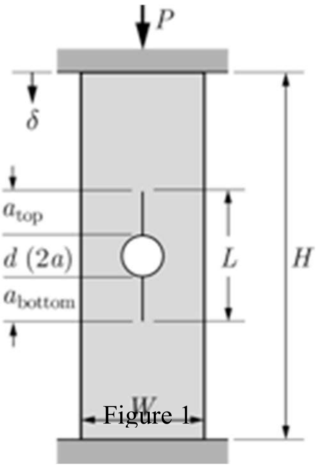


Figure 1

A novel specimen geometry (Figure 1), has been developed to allow tomographic imaging of toxic and irradiated specimens while under load and at elevated temperatures. The stress state in the specimen is such that a crack is always moving into a compressive field and a stable configuration is attained, enabling 3D imaging. In addition, by measuring the crack length as a function of applied load, the fracture toughness of the material can be calculated.

Virgin gilsocarbon and irradiated specimens from installed sets at Hinkley Point B were machined to the appropriate geometry. The imaging experiments were performed at beamline I13 at the Diamond Light Source. The load was incremented until a crack was initiated from the central hole when tomographic imaging was performed. The load was incremented again, and the process repeated until the crack was fully through the specimen.

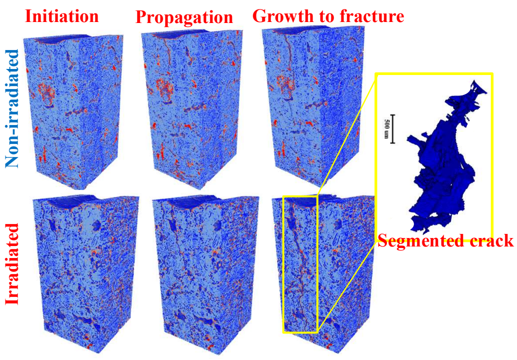


Figure 2

The crack growth mechanism in both materials was similar, with the crack passing through the same features (Figure 2). However, there was significantly less microcracking ahead of the crack tip in the irradiated material. This was mirrored in the crack growing significantly further for a similar displacement increment in the irradiated material than the virgin, with a concomitant lower measured toughness. This has consequences for the interpretation of fracture experiments using virgin material for predicting the behaviour of irradiated material.

19h00: *Dinner*

Wednesday 17th October 2018

07h30: Breakfast (Hotel Restaurant)

08h30: Conference Service Desk Opens, Function Suite Foyer

09h00: **SESSION: Whole-Core Assessments: Damage Tolerance**
Chairman: Prof. Peter Flewitt (University of Bristol)

09h00 **Core Damage Tolerance Assessement: State of the Art**
McLachlan, N.

Tolerance of AGR graphite cores to damage in the form of cracked graphite components, arising during normal operation, faults or seismically, measured against the nuclear safety requirements of shutdown/holddown, cooling and refueling reliability, is an integral part of the safety cases for the graphite cores.

Development of the Damage Tolerance Assessments (DTA) methodology and data has continued over last few years, and is anticipated to continue into the future, reducing conservative simplifications in the current approaches and converting these into assessed safety margins, which can then be formally claimed as tolerance to more damaged core states (e.g. in terms of cracked brick morphologies, deformations and numbers).

This paper gives an overview of recent DTA developments, and also a preview of anticipated future developments, including core external loading, brick load capacities, cracked brick key/keyway interactions respecting crack morphology, treatment of multiply cracked bricks and fragments and assessment of dynamic control rod behaviour.

09h20: **Development and Application of User Elements in ABAQUS/Explicit to Improve the Fidelity of Whole Core Modelling for Seismic Damage Tolerance Assessments**
Sawyer J., Dickenson R., Wedd B., Lessmann M. and Steer A.

As the graphite cores of the UK's Advanced Gas-Cooled Reactors age, radiation induced weight-loss and shrinkage act to weaken the bricks and cause them to crack. As this happens there is a possibility that the graphite core behavior during a bottom line seismic event could alter to challenge the fundamental nuclear safety requirements for the AGRs. Seismic assessments of the AGR cores use a finite element modelling approach called GCORE.

For the later-in-life core conditions the GCORE models predict multiple components of the graphite keying system will be overloaded during the seismic event. Previously the GCORE finite element method has approximated the consequences of this overloading using an incremental approach. This modelling methodology has been updated to take advantage of the user element capability within ABAQUS/Explicit (VUELs).

This paper presents a summary of the expected failure mechanisms and progression based on the experimental evidence available to date and how this has been idealized in the GCORE modelling methodology alongside complementary enhancements required to ensure the model remains stable and physically representative.

09h40: **GCORE Validation for Large Arrays of Radially Keyed Bricks**
Sheehan H., Riley H., Gittoes J., Cruz Garcia L., Beckett J., Collett P., Bale T., Norman R. and Steer A.

To ensure safe and reliable operation of the UK's Advanced Gas-cooled Reactors (AGRs), seismic qualification of the graphite cores is required to demonstrate safe shut down during an infrequent seismic event. Seismic assessments of the AGR cores use a finite element modelling approach called GCORE. The GCORE methodology has undergone validation based on dynamic response tests with small arrays of intact keyed bricks and static loading tests on large arrays containing simulated cracked bricks. To extend the validation of the GCORE methodology to the dynamic response of large arrays with cracked bricks, a shaking table experiment with a multi-layer keyed array (MLA) test rig with quarter-sized components based on late in life core geometry is been undertaken at the University of Bristol.

This paper extends the validation of the GCORE methodology for large arrays of radially keyed bricks with high levels of simulated damage and investigates the potential causes of differences between the computational predictions and experimental measurements.

10h00: **Statistical Analysis of Channel Functionality**
Crawford R. and Clayton C.

Analyses using the whole-core modelling framework AGRIGID and the channel functionality assessment method LEWIS have provided evidence to a number of safety cases. The work presented here looks at the statistical distribution of results from these methods with particular inputs relevant to Hinkley Point B and Hunterston B (HPB/HNB) power stations.

For one percentage of cracks, 64 load cases have been assessed and for 6 other percentages of cracks, 10 load cases have been assessed. The variability between these load cases in the distortion or functionality of the most distorted channel is used to calculate a probability that control rods and fuel will be challenged (according to the conservative calculation of LEWIS).

Seven different cracked core analyses were performed, each containing a different number of cracked bricks: 179 (10%), 358 (20%), 538 (30%), 717 (40%), 896 (50%), 1254 (70%), and 1792 (100%). Two statistical distributions have been used: the Gaussian distribution and the Generalised Extreme Value (GEV) distribution. In both cases, these were fitted to the set of LEWIS results from the most distorted channel in each load case.

The statistical results are discussed, including methods to reduce uncertainty and the implications of having an "outlier" result.

10h20: **SALCOR Coupled Simulation of the Distortion of an Ageing AGR Graphite Core and Fuel-Stringer/Control-Rod Functionality Assessment**
Salih H. and McLachlan N.

An ageing and distorted AGR graphite core might inhibit the free movement of the fuel stringers and control rods (CR). A de-coupled core distortion and stringer/CR functionality assessment is performed using AGRIGID and LEWIS. This approach has the benefit of assessing the functionality of all the stringers/CRs in a timely manner, as a result of the adopted simplified assumptions. SALCOR solid modelling methodology allows a coupled simulation of core distortion and stringer/CR functionality assessment to be carried out, without the simplifications of the de-coupled approach.

In the SALCOR coupled approach, both the core components and the fuel stringer/CR are represented using solid modelling, unlike AGRIGID and LEWIS. The keyed and filler bricks of the HPB/HNB CR interstitial channels are modelled explicitly, unlike AGRIGID.

SALCOR analysis of a 5x5 twelve-layer high array solid model of the middle region of the HPB/HNB core with either a fuel stringer or a CR solid model will be presented. Singly and doubly cracked fuel bricks in the fuel channels are included randomly at prescribed burn-up times of core life. The SALCOR analysis, using ABAQUS/Standard, with UMAT and field variables for the ageing effects of dimensional changes and crack fuel brick openings is used to assess the functionality of the fuel stringer/CR.

10h40: *Coffee*

11h10: **SALCOR Prediction of Control Rod and Fuel Channel Shapes of AGR Distorted and Cracked Graphite Cores**
Salih H. and McLachlan N.

The control rod (CR) channels cannot easily be inspected during reactor outages, as the channels are occupied by the CRs required for shutdown of the reactor. Fuel channels are inspected and monitored, and channel shapes are predicted using AGRIGID. The distortion of the CR channels can potentially be inferred from computational and inspection knowledge of the surrounding fuel channels. However, a solid modelling computational method predicting representative CR and fuel channel distorted shapes of an ageing and cracked AGR core is desirable and has been developed.

In SALCOR (Solid Analysis of Loaded CORe), the keyed and filler bricks of the HPB/HNB graphite core CR interstitial channels are explicitly included; cracked fuel bricks are also explicitly included. Hence, a SALCOR analysis of a full height array of core components will produce representative CR and fuel channel shapes. The influence of different configurations of cracked fuel bricks, on the distorted shape of the CR channel, can also be investigated.

An array 5x5 twelve-layer high of the middle region of the HPB/HNB core is analysed using SALCOR solid modelling methodology. SALCOR analyses, using ABAQUS/Standard with UMAT and field variables for ageing effects, have been carried out to predict the distorted shapes of the CR channel and the surrounding cracked fuel channels. The presentation will outline the study carried out and the conclusions reached.

11h30: **SALCOR – Simulation of Local AGR Core Region with Fragmented Singly Cracked Bricks and their Effect on Core Geometry**
Bazell S. and McLachlan N.

As the AGR graphite core ages, the fuel bricks are postulated to crack. As a consequence, the geometry of the core will alter from the original design; therefore, there is a need to determine the distorted geometry in order to demonstrate the safe functionality of the core. Singly-axially cracked bricks (SCB) are one possible way that a fuel brick can crack, resulting in an axial crack extending radially through the keyway root of the keyway and extending the full height of the fuel brick. The SCB can in turn cause secondary cracking, resulting in fragments, which could potentially enter the crack opening and affect the geometry of the core and the movement of the fuel stringer.

Using SALCOR, (a whole core modelling and analysis methodology using 3D solid finite elements to model the graphite components of the AGR core), a set of FEA models representative of regions of Hinkley Point B (HPB) and Hunterston B (HNB) has been built and analysed to investigate different types of fragment. In particular, the analysis addresses the question of what happens to the fragment and neighbouring components.

This paper is a continuation of previous SALCOR work which investigated the consequence of SCBs and DCBs [1]

[1] Bazell, S. and McLachlan, N, (2016). "The Application of 3D Solid Finite Modelling and the Effect of Cracked Bricks in support of Hinkley Point B and Hunterston B". Proceedings of the 5th EDF Energy Nuclear Graphite Conference 'Science and Engineering in Collaboration to Support Safe Operation of the Graphite Reactor Cores'. May 9th-12th, 2016. Grand Harbour Hotel, Southampton, UK.

11h50: **Progress with Core Damage Tolerance Modelling using SOLFEC and PARMEC**
McLachlan N., Koziara T., Martinuzzi P., Brasier S. and Cannell B.

Tolerance of AGR graphite cores to damage in the form of cracked graphite components, arising during normal operation, faults or seismically, measured against the nuclear safety requirements of shutdown/hold-down, cooling and refueling reliability, has traditionally been performed using "stick and spring" models, where components are represented by bodies with only single nodes at each contact and single non-linear springs in each interaction direction, necessitating conservative simplifications. Despite their "simplicity", these models give just the required through-put using commercial solvers and hardware.

Quantifying these conservative simplifications to demonstrating safety margin requires consideration of more: complex cracked component geometries, representative modes of cracking, and precise representation of interactions, beyond the level of resolution and tractability of current "stick and spring" models. Consequently, EDF Energy and partners have been investigating use of alternative solvers, SOLFEC and PARMEC, capable of representing many 10,000s of components as either solid-bodies or "stick and spring", with state-of-the-art efficiency benefitting from parallel processing over ~100s of CPUs. It is also possible to combine these to produce hybrid solid-body / "stick and spring" models, with increased resolution only where required, and perform run-time damage of components.

This is the first of several papers, covering the context. Further papers cover the theory and progress with verification and validation of these models for AGR core damage tolerance applications.

12h10: **SOLFEC and PARMEC Software and the Hybrid Modelling Approach**
Koziara T., Brasier S., Cannell B., Martinuzzi P. and McLachlan N.

SOLFEC implements an instance of the Non–Smooth Contact Dynamics Method using MPI and C. It includes finite element and rigid modeling capabilities, velocity based Signorini–Coulomb contact/impact law, and a parallel time stepping combined with dynamic load balancing. PARMEC is a vectorized and task based Discrete Element Method software, implemented using ISPC and C++. It includes a capability to model systems of rigid bodies and nonlinear springs, mimicking some of the LS-DYNA's functionality, currently in use in the AGR context.

While SOLFEC has been used to model AGR cores as resolved in terms of geometry and approximating deformations and contact conditions of/between components, PARMEC can be used to run simplified models, where AGRs are represented as systems of rigid bodies and nonlinear springs. The hybrid approach combines both approaches and allows mixing of regions of solid and simplified modeling within a single model. This juxtaposes the benefits of refined SOLFEC modeling with the efficiency of PARMEC's models.
The current paper outlines the methodologies of SOLFEC and PARMEC modeling and details the hybrid modeling approach. It includes comparison of performance between the three approaches, NSCD, DEM and hybrid, using a set of simple examples.

12h30: *Lunch*

13h30: **Verification, Validation, Comparison and Visualisation of Whole-Core Models (including SOLFEC and PARMEC)**
Martinuzzi P., Koziara T. and McLachlan N.

Whole core models of the AGR reactor core exist in many different forms and software. In particular, models with 3D components using the non-smooth contact dynamics software SOLFEC, and stick-and-spring "0D" models using various finite element software (PARMEC, ABAQUS, Code_Aster, LS_DYNA). All these models and software have different degrees of maturity and nicely complement each other.

At the EDF Energy R&D UK Centre, efforts are being conducted to improve the degree of confidence of the different models by performing a series of verification and validation of the codes. While the structures of PARMEC and SOLFEC have been thoroughly looked at, batteries of test cases aiming at increasing the coverage of the code and evaluating the maturity, validity and variability of the codes have also been conducted.

In addition, a common post-processing tool, MEDCoupling, is being developed. MEDCoupling enables to map results onto 3D geometries, rendering the visualization of "0D" models or experimental data much easier to interpret. Additional fields can be created to view exactly the information desired, facilitating decision making. By using the same tool, direct comparison is also enabled, which becomes more and more useful with the number of models and experiments constantly increasing.

13h50: **Application of SOLFEC and PARMEC Dynamic Analysis Codes to AGR Seismic Analysis**
Brasier S., Cannell B., McLachlan N., Koziara T. and Martinuzzi P.

Dynamic modelling of non-linear structures on the scale of an AGR core is pushing the boundary of what is possible with conventional commercial analysis codes. This has led EDF Energy to support the development of two novel dynamic codes to try to extend the available seismic analysis capabilities; SOLFEC for solid models and PARMEC for "stick-and-spring" models, with a "hybrid" analysis approach coupling both codes. Atkins have been in the unusual position for analysts of being heavily involved in guiding the development of these codes, whilst also carrying out validation exercises and seeking to use the emerging capabilities to conduct engineering analyses where possible. This paper presents what has been achieved to-date with the PARMEC and SOLFEC codes and outlines the current state of validation for them.

14h10: **BREAKOUT SESSIONS**

Managing CONSERVATISM in Modelling: Moderator: Dr C. Wheatley
Function Suite North

Reducing the COMPLEXITY of Arguments: Moderator: Dr M. Warnes
Function Suite South (1)

Maximising CONFIDENCE in Predictions: Moderator: Dr A. Harker
Function Suite South (2)

…with *Coffee ~ 15h20*

15h50: **SESSION: Graphite Structure**
Chairperson: Prof Abbie Jones (The University of Manchester)

15h50: **Graphite Oxidation as a Tool for Microstructural Investigation**
Badenhorst H.

The oxidation of graphite has been studied for a very long time, but seldom has it been used a tool for understanding the microstructure of graphite. When any catalytic impurities are removed from graphite, oxidative attack is only possible at the edges, or active sites of graphite. This provides valuable insights regarding the underlying crystallography and structure of graphite which are not immediately evident from a topological examination using for example scanning electron microscopy. Using new techniques such as field emission guns and sensitive detectors, modern scanning electron microscopes are capable of operating down to a few hundred volts. When surface features and effects are being considered this is absolutely critical, as the low voltages imply low sample penetration of the electrons, thus the true surface morphology of the material can be resolved. In combination this provides a valuable technique for investigating the microstructures found in graphite. This can be used in combination with conventional bulk techniques such as XRD and Raman to come to more clear conclusions regarding the material under investigation. The approach has been applied to natural graphite but is even more relevant to synthetic and therefore nuclear materials, which have an exceedingly complex microstructure.

16h10: **Analyses of X-Ray Diffraction Data Obtained from Unirradiated and Irradiated Gilsocarbon Graphite**
Darnborough J., Hallam K., Flewitt P. and Bradford M.

Changes in the mechanical and physical properties of synthetic polygranular graphite used as a moderator in gas-cooled civil nuclear reactors in the UK can be driven by interactions at the nano length-scale of the microstructure. As a consequence, it is important to understand the role of neutron irradiation on parameters such as crystallite size and micro-strain. X-ray diffraction data acquired as part of the EDF Energy Blackstone project, from specimens of both unirradiated and irradiated Gilsocarbon graphite and following different temperature exposures and neutron doses, have been subjected to further evaluation. The aim was to extract more information from these data with regards to crystallite size and micro-strain (stress). In addition to conventional Scherrer analyses, the approach centred on application of the Williamson-Hall method to allow separation of coherence length in the direction of the scattering vector (crystallite size) and micro-strain. The analyses considered specifically diffraction peak position, shape and width. Within necessary constraints when the Williamson-Hall method was applied, the measured changes with increasing neutron irradiation dose included:

- An expansion of the c lattice parameter.
- A reduction in the a lattice parameter.
- A reduction in average coherence length (crystallite size).
- An increase in compressive micro-strain.

This work discusses complexities of the Bragg peak profiles due to the structure of synthetic graphite and assumptions used to implement the Williamson-Hall method. The observed peak overlap could be interpreted as a complex interaction between hexagonal graphite and the turbostratic structure, which will be the focus of a future study.

16h30: **Studies of Graphite Irradiation Behaviour by Thermal Annealing**

Tzelepi N., Dinsdale-Potter J., Wilkinson S., Shaw T., Davies B. and Davies M.

Irradiation creep is very important in reactor core operation as it is believed to relieve the stresses caused by irradiation induced dimensional change in reactor components. The effects of irradiation creep are currently being investigated in a number of MTR experiments. The common characteristic of creep irradiation programmes is that they are technically difficult, time consuming and expensive. This work uses a different approach to investigate irradiation creep and it is based on the MTR studies and subsequent thermal annealing undertaken by Brocklehurst and Brown (1969).

This paper presents the results of the second phase of the annealing programme that was funded by the Innovate-UK project "Influence of Creep and Geometry on Strength of Irradiated Graphite Components". The programme included thirteen irradiated samples for Hinkley Point B and Hunterston B and three unirradiated controls annealed at temperature steps of 1400 °C, 1800 °C and 2140 °C. Before and after each annealing step, measurements of sample lengths, diameters and coefficient of thermal expansion in all orientations were performed. In addition, dynamic Young's modulus measurements and laser Raman spectroscopy were carried out at the start and at the end of the programme. The paper concludes with the comparison with the historical data and discusses potential reasons for some inconsistencies.

16h50: **Characterisation of the Crystalline Structure of Neutron-Irradiated Graphite**

Alnairi M., Mironov B., Windes W., Scott A., Westwood A. and Brydon R.

In this work, microstructural parameters, such as lattice dimension and the disorder within the lattice of four different neutron-irradiated graphite grades have been investigated using X-ray diffraction (XRD) and Raman spectroscopy techniques, which produced consistent results (see Figs 1a, b). The graphite samples (Generation-IV candidates) were irradiated at the Advanced Test Reactor at the Idaho National Laboratory (grades PCEA and PCIB both based on petroleum coke) and subjected to neutron irradiation doses ranging from 1.5 - 6.8 dpa with the irradiation temperature varied between 350°C - 670°C. Compared to virgin specimens of the same grade, XRD diffractograms of the two tested graphites illustrated that crystallite size decreased by roughly 40% (for both L_a and L_c) for low dose, low temperature samples, while for high dose, high temperature samples it reduced by roughly 50% (for both L_a and L_c) [1]. As correlating evidence, quantitative analysis of the G peak obtained from Raman spectra provided evidence for the fragmentation of crystallites following the same changes in irradiation conditions [1].

continued overleaf

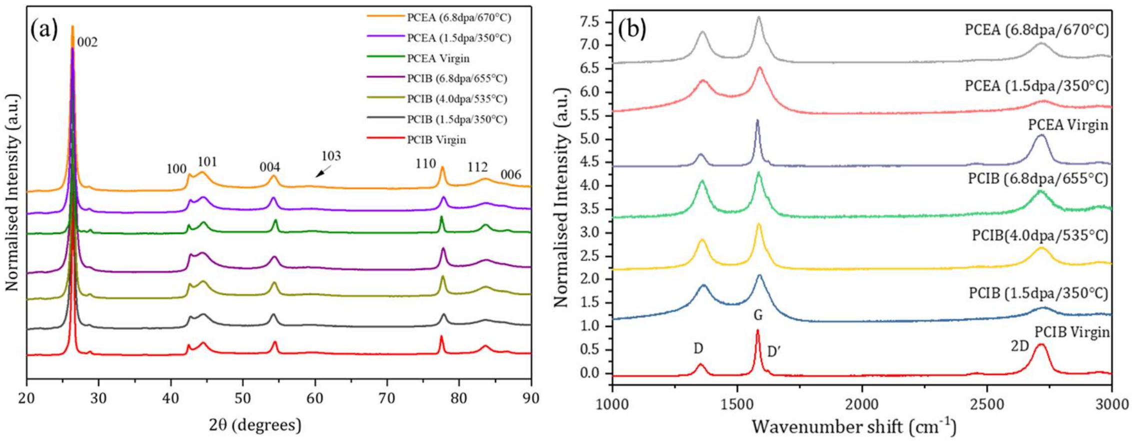


Figure 1: Schematic diagram showing (a) the measured XRD patterns of virgin and irradiated PCIB and PCEA graphites, (b) Raman spectra in 1000-3000 cm^{-1} wavenumber range of raw of the aforementioned nuclear graphite grades (virgin and irradiated).

[1] Freeman, H.M., Mironov, B.E., Windes, W., Alnairi, M.M., Scott, A.J., Westwood, A.V.K. and Brydson, R.M.D., 2017. Micro to nanostructural observations in neutron irradiated nuclear graphites PCEA and PCIB. Journal of Nuclear Materials, 491, pp.221-231.

[2] This work was financially supported by the Umm AL-QURA University Al-Laith branch, Saudi Arabia.

[3] Irradiated samples were provided by the Idaho National Nuclear Laboratory, USA.

17h10: **Nano- and Microstructural Characterisation of Irradiated AGR Graphite**

Tzelepi N., Jordan M., Ward M., Lo I-H. and Krishna R.

Physical, chemical and mechanical properties of AGR graphite are thoroughly examined through EDF Energy’s extensive core monitoring programmes. Although optical microscopy is now part of the monitoring programmes and has provided some useful insights on the microstructure of irradiated Gilsocarbon from different AGR stations, other microstructural characterisation is not carried out routinely yet.

This programme is aimed at understanding the relation between property and microstructural changes of irradiated Gilsocarbon as a function of fluence and weight loss. A number of trepanned samples have been selected that cover a dose range of 4.6 to 18.7 dpa; that is to say, irradiated samples have been selected to represent the microstructure before, at and after dimensional change turnaround. In addition, samples with low and high weight loss at the same dose have been selected to investigate the effect of weight loss on the evolution of the microstructure with dose. The weight loss of the selected samples ranges from 2.9 to 29.3 %.

This study has used X-ray micro-tomography, helium pycnometry, transmission electron microscopy, Raman spectroscopy and scanning electron microscopy to characterize the nano- and micro-structure of the samples. In addition, a size-effect study on helium pycnometry has been carried out. The aim of the size effect study is to inform investigations on samples from Materials Test Reactor experiments that normally use smaller samples than the ones used in the AGR core monitoring campaigns. This paper presents the results of the helium pycnometry study and relates the findings of the microstructural characterisation to those of the NNL polycrystalline model.

17h30: **A Porosity Analysis of Ex-Reactor Gilsocarbon based on a Semi-Automatic Segmentation of X-Ray Tomography**
Jordan M., Taylor J. and Tzelepi N.

The pore structure of graphite subjected to fast neutron irradiation and radiolytic-oxidation changes dramatically as weight loss progresses. Initially the material contains both open and closed pore networks, which regulate gas flow through the bulk. As radiolytic-oxidation progresses many physical properties are known to follow a negative trend as the material is oxidised, including strength, stiffness, electrical conductivity. Historically, porosity has been measured by: indirect measurement techniques, such as pycnometry; two-dimensional imaging, such as impregnated optical microscopy or SEM; or small volume destructive techniques, such as FIB-tomography or TEM. Advances in the recent years mean that non-destructive X-ray computed tomography (XCT) has become a viable method for bulk measurements of 3-D pore networks, capable of distinguishing open and closed pore networks. However, no standard for porosity determination by XCT exists, and so in current practice the measurement of every tomograph is dependent on the analyst's judgement or reliance on an ideal model which is typically unsuitable for graphite.

As part of an Innovate UK project to examine size effects in fracture test measurements, X-ray tomographs were captured of a series of ex-reactor Gilsocarbon samples prior to fracture testing. A method for semi-automatic porosity segmentation was developed, which despite the simplicity of the underlying segmentation rule extracted plausible pore networks for material with a range of weight loss. The method will be described, and a quantitative analysis of the segmented pores and the relation of the porosity and fractures properties will be presented.

17h50: **Modelling of the Combined Void-Size Distributions of Gilsocarbon obtained from Pycnometry, Gas Adsorption and Mercury Porosimetry**
Jones K., Laudone G. and Matthews G.

The pore-level characterisation of nuclear graphite is critical for predicting the reactor safety and for assessing the viability of different grades of graphite material. The majority of studies often focus on the impact of one specific length scale (macroscopic/mesoscopic/ microscopic) but few studies have attempted to provide void size information that spans multiple length-scales.

Gilsocarbon graphite is particularly difficult to characterise at a pore level because the accessible pore matrix is thought to comprises voidage which spans over several orders of magnitude. It is evident that one technique along would be insufficient for measuring the entire pore range, therefore a combination of techniques has been used in best effort to provide a complete picture.

Firstly the samples were analysed by helium pycnometry to obtain values of the accessible and inaccessible pore volumes for each sample. This was followed by a series of gas adsorption experiments before the final investigation using mercury porosimetry.

The experimental information was then brought together via a series of modelling procedures. Gas adsorption and mercury porosimetry data are converted to void size distributions. The bespoke software package PoreXpert, designed at the University of Plymouth, was used to inverse model the experimentally measured percolation characteristics and total accessible porosity, obtained from the combination of techniques, to generate simulated void network structures. The simulation software can also perform calculations on the chosen representative structure in order to evaluate properties such as the network flow capacity of gases, with the eventual aim of improving the understand of gas flow through nuclear graphite. This method was initially validated using the homogeneous and isotropic IG-graphites.

This work shows the results of the characterisation of a range of virgin and MTR Gilsocarbon samples and how the void volume and void sizes evolved with increasing weight loss and irradiation of the samples.

19h30 for 20h00: *Reception and Gala Dinner. Castle Green Hotel*

Thursday 18th October 2018

REMINDER: ROOMS MUST BE VACATED BY 10am: secure luggage storage at Main Reception or Conference Office (Function Room Foyer)

07.30: Breakfast (Hotel Restaurant)

08.30: Conference Service Desk Opens, Function Suite Foyer

09h00: **SESSION: Inspection, Repair and Monitoring**
Chairman: Prof A.J Wickham (The University of Manchester)

09h00: **Peripheral Reflector Block In-Situ Robotic Reshaping to Accommodate Core Geometry Changes**
Petit de Mange E. and Gurin D.

Following several years of success in completing complex, first-of-a-kind projects of in-situ graphite core moderator block modifications in RBMK reactors for the purpose of ensuring safe operation and life extension – Diakont has recently completed groundbreaking new work; this paper will present an overview.

Core geometry changes on the RBMK fleet over the course of their operating lifetimes have resulted in the radial gaps closing between the outer graphite reflector block stacks and the steel vessel containing the core. This has introduced the potential for mechanical interference, that could lead to reactor vessel damage.

Diakont developed, qualified, and fielded a complex robotic system that entered the reactor cavity below the upper bioshield, traversed out to the core perimeter, lowered the cutting robot into the annular region, removed graphite material from a block stack, measured the effectiveness of the repair and the resultant geometry, and then exited the annular area; all while the reactor remained fueled. This process was repeated several times throughout the course of the outage, working continuously on a critical path schedule.

As a result of this repair program, the necessary geometry of the peripheral brick stacks has been confirmed as restored, allowing for continued safe operation and life extension.

09h20: **3-D Reconstruction of AGR Fuel Channels using RVI Footage**
Law K., West G., Murray P. and Lynch C.

Within the U.K, there is a fleet of 7 Advanced Gas-cooled Reactors (AGR) in operation which are close to or have already exceeded their original estimated design lifetime. As the reactors continue to age, it is critical to gain an understanding of the structural morphology of the AGR fuel channels and to quantify changes in the physical properties of the graphite bricks which occur during reactor operation. To assess such operational changes, periodic Remote Visual Inspection (RVI) is carried out on pre-selected channels using specialist inspection tools which gather video and other sensory data of the channel bore. Current visualisation of the video provides limited 3-D structural information due to the image acquisition process which results in a loss of depth information.

This paper describes a novel image reconstruction framework based on a technique called Structure-from-Motion (SfM) which has been designed to ascertain 3-D structural information directly from 2-D in-core RVI data. This paper also discusses the challenges of applying 3-D reconstruction techniques like SfM to pre-existing inspection data before demonstrating that the proposed framework can be applied to data captured under laboratory conditions and to subsets of real in-core inspection videos to generate singular and circumferential point cloud reconstructions of the channel bore.

09h40: **Automated Analysis of AGR Fuel Channel Inspection Videos**
Devereaux M., Murray P., West G., Buckley-Mellor S., Cocks G., Lynch C. and Fletcher A.

Remote Visual Inspection of the fuel channels which form the reactor cores of the UK's fleet of Advanced Gas-Cooled Reactors (AGRs) occurs during planned periodic outages and provides station operators with a detailed understanding of core condition. A typical single fuel channel inspection generates a large amount of footage which must be analysed before the station is returned to power (provided it is safe to do so). While manual approaches are currently used, inspection videos can be analysed efficiently using techniques from image processing and computer vision. For example, the ASIST (Automated Software Image Stitching Tool) software processes inspection videos to construct a single image known as a chanorama (channel panorama) which allows the full inside surface of a single fuel channel to be viewed in a snapshot. To accurately characterise defects such as cracks in chanoramas, their dimensions need to be measured. This requires channel features of known size to serve as references to calculate scaling factors. These features vary from station to station and include overall brick dimensions, trepanned holes and keyways. In this paper, we propose a series of algorithms to automatically detect and measure the dimensions of each of these known features. In turn, this information can be used to accurately size any cracks detected.

10h00: **Extracting Quantitative Information on Graphite Weight Loss from Eddy Current Data**
Verrier P., Robinson P. and Watson C.

Eddy current data from the graphite core is now routinely gathered at inspections and Quintessa has developed an empirical model to relate the measured resistivity to a weighted average graphite density (and hence weight loss) at each scanning frequency. Weight loss is one of the most important processes affecting safety as the reactors age.

Although eddy current penetration is relatively shallow, the scan covers the whole brick surface and can be gathered for cracked bricks and also repeatedly for the same brick. It can therefore be seen to complement data from measurements on graphite samples obtained by trepanning, but the question remains of how to include quantitative information from eddy current data in weight loss models used to support safety cases.

In this paper, we demonstrate how existing eddy current data can be used to investigate systematic trends within the graphite bricks and determine both the within-brick and brick-to-brick variability in density. This provides insight into the uncertainty in whole-brick density estimates and helps inform how eddy current data should be regarded with respect to trepan data. These uncertainty estimates ultimately help in the derivation of a method to include eddy current data in weight loss models, improving forecasts of weight loss evolution.

10h20: **Fuel Channel Bore Estimation for On-Load Pressurised Fuel Grab Load-Trace Data**
Young A., Aylward W., Murray P., West G., McArthur S. and Rudge A.

Accurate fuel channel bore estimation allows the inference of information about the health of the graphite core of an advanced gas-cooled reactor (AGR). This concept has been extensively explored in previous work focussing on analysing off-load depressurised fuel grab load trace (FGLT) data. By isolating the frictional component of the FGLT and using inspection data as a ground truth, a linear regression model was trained to estimate the fuel channel bore. However, for data gathered during on-load refuelling, which has the added complexity associated with the interaction between the fuel assembly and coolant gas, the same process cannot be used.

This paper describes a new process for removing the aerodynamic effects of the coolant gas in the core from onload pressurised FGLT data. This effect cannot be directly measured, so initially an empirical model is created by comparing the response from both off-load depressurised and onload pressurised refuelling events. This model is then used to estimate the off-load equivalent FGLT response directly from the onload trace. Then, by using a bore estimation model, trained on offload data, it is possible to produce accurate bore estimations for on-load FGLT data. The results from this approach are compared against a theoretical gas flow model to determine the best approach for removal of the aerodynamic effects.

10h40: *Coffee*

11h10: **LoTAS – Load Trace Analysis Software**
Watson C., Robinson P., Suckling P., Chittenden N. and Kite J.

As fuel stringers are raised and lowered within the reactor core, the load experienced by the fuel grab hoist is recorded. Each stringer houses a set of three brushes, at the base of the stringer below the fuel elements, that are in contact with the graphite bricks as the fuel is moved. The variation in load (corresponding to small changes in bore diameter and hence evolving friction and compression force components) can be observed in the load traces.

Work carried out by EDF Energy has shown that it is possible to detect keyway root cracked bricks (which tend to have wider bore diameters than uncracked bricks) by manually analysing the fuel grab load traces (FGLTs). The ability to monitor crack development as the reactors age can provide an important contribution to demonstrating continued safe operation.

The LoTAS (Load Trace Analysis Software) has been developed by Quintessa for EDF Energy to help manage the large database of existing FGLTs. This facilitates EDF Energy in undertaking their analysis, in particular when wishing to compare traces with previous ones for the same channel. In addition, LoTAS includes algorithms for detection of abnormal bore diameters. The aim is to reduce human error and subjectiveness, increasing confidence in the methodology, whilst reducing the effort required to process and analyse traces.

11h30: **Breakout Reports and Further Discussion**
Chairperson: Ms Athanasia Tzelepi (NNL)

Dr C. Wheatley, Dr M. Warnes, Dr A. Harker

12h30: *Lunch*

13h30: **SESSION: Statistical Modelling of Cracking**
Chairman: Mr O. Booler (Wood Nuclear)

13h30: **Successes in Statistical Modelling of AGR Core Evolution**
Rookyard S., Robinson P., Pogson M.

The properties of graphite bricks in nuclear reactors are variable by nature. Statistical models are therefore a crucial tool, for both predicting inspection outcomes and for extrapolating results from the inspected channels to the whole reactor or into the future. Statistical modelling has been used with great success in two particular areas of interest.

Bore-initiated cracking has long been known to occur in AGRs. Quintessa uses statistical modelling on behalf of EDF Energy to predict the number of such cracks that will be seen at each inspection. The models also serve to forecast future numbers of cracks and to assess whether this type of cracking is likely to pose problems for the reactor from a safety perspective. We present a review of bore-initiated cracking predictions and results from past inspections, illustrating the success of the approach in predicting inspection outcomes.

Evolution of the fuel channel wall shape is also seen in AGRs. This is an important indicator of the likelihood of keyway root cracking. Before each inspection Quintessa uses statistical models of shape metrics to predict which bricks are most likely to be keyway root cracked; the accuracy of the predictions is assessed post-inspection. Obtaining a satisfactory statistical representation of keyway root cracking is important for demonstrating continued safe operation.

13h50: **Forecasting Keyway Root Cracking at Hunterston B Reactor 3**
Robinson P. and Bradford M.

In October 2015, the first keyway root cracks (KWRCs) were seen at Hunterston B (HNB) Reactor 3. KWRCs had previously only been seen in some of the limited population of high shrinkage bricks at HNB R4. Stress models had long predicted the occurrence of KWRCs although the precise onset time was uncertain, and the predicted cracking rate was generally thought to be conservative.

Over the following two years, an approach was developed to forecasting numbers of KWRCs and comparing these to operational allowances derived from damage tolerance assessments of the whole reactor. The aim was to acknowledge the uncertainty inherent in such forecasts while providing a basis for deriving a justified period of safe operation (JPSO) which showed that it was safe to operate the reactor up to and beyond the next planned inspection.

With the benefit of hindsight, the assumptions, forecasts and JPSOs made for the 2017 and 2018 HNB R3 inspections are discussed. The way that various sources of uncertainty were accounted for are reviewed. Reviewing these historical forecasts provides valuable information relevant to the confidence that can be placed in current forecasts of core ageing.

14h10: **SESSION: Modelling Graphite Properties and Brick Evolution**
Chairperson: Ms Athanasia Tzelepi (NNL)

14h10: **Regulatory Considerations when Dealing with Graphite Materials-Property Models**
Poulter L. and Teifi M.

A knowledge of the expected behaviour of the graphite, as it ages, is central to the preparation and assessment of safety cases for the Advanced Gas Cooled Reactors (AGRs). The Licensee, EDF Energy, has developed a detailed model of graphite materials properties, using data from various sources including the AGRs themselves, but also several material test reactor (MTR) experiments. It is claimed that this model, currently termed the EDF Integrated Model (EIM), provides a sufficient description of behaviour such that there is adequate confidence that likely phenomena associated with ageing can be identified in advance.

From a regulatory point of view, The Office for Nuclear Regulation (ONR) is keen that future behaviour is predictable, whilst recognising that multi-parameter models are inevitably limited by the available data. In the case of graphite, materials properties, which include dimensional change, coefficient of thermal expansion (CTE), Young's modulus and strength, are often assumed to be functions of irradiation, weight loss and temperature. Inevitably the datasets are limited and mathematical descriptions are created which purport to interpolate between regions where the actual data is limited. However, at least for the AGRs that are leading in terms of irradiation, a degree of extrapolation is currently needed for some properties, e.g., dimensional change, as existing MTR data does not completely cover the high fluence region. Even for properties such as CTE, where there is applicable MTR data at high fluence, uncertainties in the precise MTR conditions can lead to questioning of the expected behaviour. ONR regards the need for extrapolation as undesirable, but recognises the obvious reality that only the leading reactors will provide data that confirms or denies the predictions based on models. ONR's guidance recognises this difficulty, whilst expressing both the desirability of mechanistic explanations for observed behaviour and also the need for caution in applying models, particularly when extrapolation is involved.

To aid ONR's work in providing a constructive challenge to Licensee's safety cases and ultimately to facilitate regulatory judgements about whether future operation is permissible, ONR has commissioned alternative analysis of the available data. This work, performed by the University of Manchester, the HSE Health and Safety Laboratory and consultants, including members of ONR's graphite technical advisory committee (GTAC), has several purposes. Firstly it is intended to show whether an alternative analysis to that produced by the Licensee is credible and expose any uncertainty in the EIM, perhaps caused by insensitivity of the proposed relationships to the data used in their calibration. Secondly, it facilitates independent analysis, including sensitivity studies, by which the effect that small changes in the models have on stress analyses of the graphite bricks can be investigated.

14h30: **EdF-Energy Integrated Methodology 1.2: Supporting Evidence and Impact on Brick Cracking**
Cathcart H., Joyce M., Davies M. and Baylis S.

The EDF Energy Integrated Methodology (EIM) material model predicts the dimensional change and material properties of Gilsocarbon graphite subjected to fast neutron irradiation and radiolytic oxidation. Recently, the formulation of the Coefficient of Thermal Expansion equations in the EIM has been updated, with this updated formulation termed EIM v1.2. This presentation firstly explains the evidence, from both deep-cutter AGR trepanned samples and the ACCENT material test reactor programme, which suggested and supported the change in formulation. Secondly, this presentation describes the impact of the formulation change AGR fuel brick stress analyses and cracking forecasts.

14h50: **Calibration of With- and Against-Grain Dimensional Change using AGR Moderator Brick Measurements**

McNally K., Fahad M., Hall G. and Warren N.

The irradiation-induced dimensional changes of nuclear graphite lead to identifiable changes in the bore shape of the moderator bricks when compared with their start-of-life condition. Through-life brick bore shape changes are complex due to the changes in stress states within the brick and the alignment of fuel within the channel. Furthermore there are systematic between-brick differences in the evolution of bore shape changes due to both differences in loading conditions in-reactor and the material properties of the virgin graphite.

In earlier work a methodological framework was demonstrated for calibrating a subset of sensitive parameters in an engineering model of an Advanced Gas-cooled Reactor (AGR) moderator brick, such that a measure of brick shape predicted by the model was consistent with the longitudinal trend observed in inspection data. In this work we further refine this approach and calibrate five parameters relating to the with- and against-grain dimensional change material property models using three measures of brick distortion and demonstrate the calibrated model is consistent with the longitudinal trend of brick bore shape data from layers three to nine at Hunterston B.

In a further refinement, a novel methodology that allows brick specific parameters to be estimated, thus capturing the evolution of the bore shape associated with a particular moderator brick is demonstrated. A conceptual framework that in future work will allow brick shape evolution to be directly linked to internal brick stresses is discussed.

15h10: **Validation of Dimensional Change Modelling Through Outage Expectations**

Cathcart H., Chidwick L., Joyce M. and Baylis S.

Predictions of the dimensional change of graphite core components are integral to many parts of the AGR core safety case. These predictions can be made using the EDF Energy Integrated Methodology (EIM) model, which is calibrated to fuel channel bore shapes observed during reactor outages. To gain confidence in the predictive capability of the model, channel bore measurements are compared to predictions made prior to the inspection. This presentation covers a Bayes-inspired methodology to generate expectations for future measurements of specific bricks within a reactor, considering information about the range of graphite properties seen at the station and previous measurements of the brick in question.

15h30: **Effect of Different Graphite Thermal-Conductivity Models on the Prediction of Graphite Temperature**

He J., He S. and Xu B.

In this study, simulations using coupled computational fluid dynamics (CFD) and thermal conduction in solids have been carried out to resolve the heat transfer of the coolant flow in a fuel channel and the temperatures of the graphite brick and sleeve. The main objective is to study the sensitivity in the heat transfer behaviors and temperature distributions of using two different graphite thermal conductivity models.

The two graphite thermal conductivity models investigated in this research are the Paper 28 model[1] and the newly endorsed EIM (EDF Energy Integrated Methodology) model[2]. Both provide reasonably good predictions of the graphite thermal conductivity using different correlations based on irradiation temperature, graphite weight loss and cumulative fast neutron dose.

Numerical simulations of a fuel channel in different full power years (FPY) have been carried out for Hinkley Point B, considering the upward stringer flow, the conduction and radiation heat transfer of the brick and the sleeve, the downward annular flow between the brick and the sleeve, and the arrowhead passage flow. Simulations are carried out using open-source software, Code SATURNE and SYRTHES developed by EDF, while the configurations and boundary conditions are extracted from the system code PANTHER.

[1] R T Szczepura, Endorsement Statement for Compendium of CAGR core and Sleeve Data and Methods. Core Graphite Data Sheet, CSDMC/P28, Data Sheet C5/1, March 1995.

[2] L Chidwick, Interim Model for Thermal Conductivity and Electrical Resistivity, FNC 40318-023/39124R Issue 1, November 2013.

15h50: **Closing Remarks** Ms. Athanasia Tzelepi

Coffee and close of conference

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To join The British Carbon Group, or for more information, please contact any one of the Officers or Committee Members attending all or part of the meeting:

Dr. Peter Minshall; Prof. Tony Wickham; Ms. Atanasia Tzelepi

The British Carbon Group would like to apologise that the originally advertised visit to the Lakeland Motor Museum with a different gala dinner venue could not take place owing to circumstances beyond BCG control.

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