The 8th EdF Nuclear Graphite Conference

Graphite Expertise in Support of Current Operation and the Next Generation of Nuclear Reactors

22nd – 24th September 2026

at

QUEENS COLLEGE
High Street
OXFORD OX1 4AW

PROGRAMME AND ABSTRACTS

Sponsored by

EdF Nuclear Operations
Amentum
American Ceramics Society
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INTRODUCTION

Dr. James Reed, Chief Graphite Engineer, EdF Nuclear Operations, Gloucester

Welcome to the 2026 UK Nuclear Graphite Conference at the University of Oxford. This conference brings together experts from across industry, academia and the regulatory community to share developments in graphite science and engineering. The programme reflects the key challenges facing the UK nuclear sector today: supporting the safe operation of the remaining AGR fleet, preparing for graphite decommissioning, and applying decades of graphite knowledge to future reactor technologies.

For the AGRs, continued understanding of graphite ageing remains fundamental to safe operation. Irradiation damage and radiolytic oxidation continue to drive changes in material properties and core behaviour, and many of the papers presented address advances in inspection, modelling and safety assessment.

As the industry plans for the future, the management of irradiated graphite remains a major decommissioning challenge, while Advanced Modular Reactors offer the opportunity to build on the knowledge and experience gained from more than four decades of AGR operation. I would like to thank all authors, presenters and delegates for their contributions, and I hope the conference provides an excellent forum for technical discussion, collaboration and innovation.

This conference is the most recent in a series planned to showcase the technical work supporting the operation of the UK AGRs and, latterly, decommissioning and technology transfer to the next generation of Advanced Modular Reactors. The first seven were held in Cardiff, Nottingham, Aston, again in Nottingham, then Southampton, Kendal and finally Manchester in 2005, 2008, 2011, 2014, 2016, 2018 and 2024 respectively. Papers derived from presentations at 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. RSC Publishing (Royal Society of Chemistry, Cambridge) (ISBN 978-1-84973-390-8)
4. Flewitt P.E.J. and Wickham A.J. (Editors) (2015). Engineering Challenges Associated with the Life of Graphite Reactor Cores. EMAS Publishing (Warrington) (ISBN: 978-0-9576730-5-2)
5. Flewitt P.E.J. and Wickham A.J. (Editors) (2017). Science and Engineering in Collaboration to Support Safe Operation of the Graphite Reactor Cores. EMAS Publishing (Warrington) (ISBN: 978-0-9576730-4-5)
6. Wickham A.J. (Editor) (2020). Achieving the Right Balance between Conservatism, Complexity and Confidence to Secure a Safe and Extended AGR Lifetime. FESI Publishing (Preston) (ISBN: 978-0-9935485-2-9)
7. Wickham A.J. (Editor) (2025). Managing the Full Life Cycle of a Reactor with a Graphite Core. FESI Publishing (Preston) (ISBN: 978-0-9935485-6-7)

PROGRAMME AND ABSTRACTS

GRAPHITE EXPERTISE IN SUPPORT OF CURRENT OPERATION AND THE NEXT GENERATION OF NUCLEAR REACTORS

A Conference at Queens College, Oxford

22nd – 24th September 2026

REGISTRATION

Registration will be available from 16h00 until 20h00 on Monday 21st September in the Queens College cloisters and then again at 08h30 *sharp* in the Shulman Auditorium, which is our conference venue.

ALL NECESSARY INFORMATION ABOUT ARRIVAL INTO OXFORD, ACCESS TO THE COLLEGE, ACCOMMODATION, AND A COLLEGE MAP MAY BE DOWNLOADED FROM THE CONFERENCE WEBSITE – "EdF Conference Delegate Guide" – kindly compiled by Robin Scales (member of Prof. Dong's research group).

Please note that any issues arising in connection with your accommodation must be taken up directly with the college: the conference organisers accept no responsibility in this regard.

TRAVEL INFORMATION

There are frequent trains to Oxford from London Paddington and London Marylebone stations, and also from the direction of Birmingham and Hereford. Other journeys may require a change at Didcot. From London Heathrow airport, travel to Paddington via the Elizabeth Line. From Gatwick airport, take a Southern or Gatwick Express train to Victoria, then London Underground Circle Line (direction Edgware Road) to Paddington.

All further enquiries regarding the conference arrangements should be directed to the registration desk (cloisters on the Monday evening), to participating members of the College, or from the Conference Manager, Tony Wickham, wickhamtony1@gmail.com, (07785 567 577), who will probably be found inside the meeting room once sessions have commenced.

PRESENTATIONS & AUDIO/VIDEO FACILITIES

Presentations will mostly be 15 minutes plus 5 minutes for discussion, with extended discussion sessions timetabled at intervals. All presentations will take place in the SHULMAN AUDITORIUM. Presenters are requested to bring their presentations on a USB memory stick *in PowerPoint format* 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 made available to delegates *via* the website for a limited period after the conference.

CATERING ARRANGEMENTS & CONFERENCE DINNER

Coffee breaks will be taken adjacent to the conference room. Delegates should retain their conference badges at all times, including when taking breakfast, lunch and dinner, in order to identify themselves as registered delegates.

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

A Conference Dinner will be held on the Wednesday evening (23rd September, 7.15pm for 7.30pm). Non-resident delegates are invited to join this event. Dress code is business casual.

Tuesday 22nd September 2026

08h30 Registration Desk Open

08h50 Welcome, Housekeeping *Professor Dong Liu, University of Oxford*

09h00 Introduction *Dr Jim Reed, EdF Nuclear Operations*

Stepping into the Future — Session Chair: Dr Jim Reed

09h10 **Development of Graphite for China's HTRs Core Support Structures**

Libin Sun
Institute of Nuclear and New Energy Technology (INET), Tsinghua University, P.R. China

Development of HTRs in China has progressed through the research reactor (HTR-10), the demonstration plant (HTR-PM) and the commercial plant (HTR-PM600). The nuclear graphite used in the core support structures has successively included imported IG11 and IG110 of Toyo Tanso, as well as domestically developed SNG742 (SIAMC) and NG-CT-01 (Chengdu Carbon), followed by new developments graphite from WXXC. The planned service life of HTRs has also been continuously extended, from 20 years (HTR-10) to 40 years (HTR-PM) and 60 years (HTR-PM600).

The HTR-PM has been in operation for several years, during which some issues with graphite selection and structural design have been identified, including exceeding the estimated bypass flow and local hotspots. The HTR-PM600 designers have reduced chamfer of graphite keys, added/modified flow restriction strips in top reflection at several gaps in the core support structure. Each of these design optimisations has undergone independent tests. In system-level all implemented optimisations effectiveness depends on the engineering practice results from recent HTR-PM600, including cold and hot commissioning as well as the process from critical loading to power operation.

HTR2.0 and VHTRs will continue to be improved, with the goal of improving economic efficiency, operating power, inlet/outlet temperatures, lifetime etc. C/C composite restraint are also under consideration and within the scope of design verification. China's nuclear safety regulatory authority has proposed new regulatory requirements that are suitable for situation based on existing operational and management experience, fully drawing on new trends in international safety regulation.

09h40 **Bio. v AI(-o) (Carbon v Silicon, Sums v Sims, or "How things used to, and still can, be done")**

N. McLachlan
EDF Nuclear Operations

Over time, progress with digital computational power has enabled simulations that are beyond the dreams of engineers and scientists only a few decades ago. However, there is still, and likely always will be, great value in calculations that can be performed "by hand" or using methods that were available to previous generations, including simplicity, transparency, understanding, speed and the re-assurance of similar results from diverse approaches.

This paper reviews a number of examples from supporting the lifetime ambitions of the Advanced Gas-cooled Reactor cores, and includes topics such as reference books; libraries of computer program subroutines; "snapshot" structural analyses; generalised plane strain; BERSAFE Graphite; Linear Programming; the difficult 3 Cs: Creep, Contact and Cracks; Segment Bowing and Simplified Linear Elastic Mechanics.

10h00 **Foundation Vision Models and Physical AI Frameworks for Nuclear Graphite Characterization**

O. Stevens, and D. Liu
Department of Engineering Science, University of Oxford, UK

Characterising microstructural degradation, pore distribution, and micro-crack evolution in reactor-grade graphite is fundamental to assessing Advanced Gas-Cooled Reactor (AGR) core integrity. Traditional manual annotation and classical image processing workflows have long established the rigorous benchmark for material qualification; however, scaling these techniques across large, high-resolution microscopy datasets presents significant throughput bottlenecks and analytical variability.

In this work, we present an initial exploratory evaluation of promptable foundation vision architectures—specifically Segment Anything Models 2/3 (SAM2/3)—for interactive graphite microstructure segmentation. While spatio-temporal foundation vision models offer promising acceleration in feature isolation, we explicitly examine their domain-specific limitations, including boundary ambiguity across low-contrast grain interfaces, post-irradiation sample noise, and the absence of intrinsic physical constraints.

Using these initial empirical observations as a case study, we outline a prospective Physical AI roadmap. We discuss how zero-shot segmentation telemetry can be integrated with physics-informed generative World Models, bridging passive diagnostic imaging with dynamic, predictive simulation of time-evolving material degradation in safety-critical nuclear infrastructure.

10h20 *Coffee break*

10h50 **Structural Integrity Assessment of the Nuclear Graphite Components Using Machine Learning**

Yiqin Ning, Yaxi Li, Graham Hall, Abbie N. Jones and Xun Zhang
The Nuclear Graphite Research Group (NGRG), Department of Mechanical and Aerospace Engineering, Henry Royce Institute, The University of Manchester, Oxford Road, Manchester, M13 9PL, UK

The structural integrity of the nuclear graphite components is important for the safe operation of the graphite-moderated gas-cooled reactors including AGR and HTGR. A central aspect of structural integrity assessment is the determination of the stresses developed within graphite components throughout their service life. This requires an accurate representation of the mechanical and physical properties of nuclear graphite and their evolutions under reactor service conditions. Experiment data of these kinds can be obtained from graphite samples tested in test reactors.

Given the complex geometries of graphite components and the spatially and temporally varying service conditions within a reactor, stress assessment for these components is usually carried out using numerical methods such as finite-element modelling (FEM). While FEM can predict the complex stress distributions within individual components, high-fidelity simulations can be computationally expensive, particularly when large numbers of components and multiple service conditions need to be considered. This presents a significant challenge for the development of large-scale, whole-core models for graphite structural integrity assessment. Recent advances in machine learning (ML) provide new opportunities to develop computationally efficient models that can directly predict stress values from the relevant geometric, material, and service-condition parameters.

In this work, a machine-learning-based framework is developed to establish the relationship between nuclear graphite component geometry, service conditions, and the resulting stress state. FEM simulations are first used to generate a comprehensive dataset covering the relevant ranges of reactor operating conditions and the resultant stress states. A machine learning model is then developed to predict the stress states. Moreover, the model is used to identify the relationship between the reactor service condition with the resultant stress states.

11h10 **Automated Nuclear Graphite Characterization Using Advanced Deep Learning Segmentation and Image Analysis Workflows**

A. Stratulat
MIPAR Image Analysis Software

Artificial Intelligence (AI) is rapidly transforming materials science by enabling faster, more accurate, and reproducible analysis of complex imaging data. Among these advances, deep learning has emerged as a powerful tool for automating image segmentation, object detection, and quantitative characterization of material microstructures. This presentation will highlight the latest AI-powered image analysis capabilities available in MIPAR (www.mipar.us), with examples including the characterization of nuclear graphite and other challenging materials systems.

The talk will explore MIPAR's deep learning architectures, including SegNet, U-Net, YOLOX, and Spotlight, discussing their strengths, limitations, and appropriate applications. Practical guidance will be provided on selecting and optimizing models to improve segmentation accuracy, reduce annotation effort, and accelerate analysis.

In addition, the presentation will demonstrate how MIPAR combines no-code workflow development, automated batch processing, quantitative measurements, and standardized report generation to deliver robust, reproducible results. These capabilities enable researchers and engineers to streamline materials characterization, support process optimization, and accelerate AI-driven research and industrial decision-making.

11h30 **UKNNL's Experimental Capability for Current and Future Reactors**

R. Smith[1], J. Dinsdale-Potter[2] and S. Wilkinson[2]
[1]United Kingdom National Nuclear Laboratory (UKNNL), Preston Laboratory, Springfields, Preston.
[2]United Kingdom National Nuclear Laboratory (UKNNL), Central Laboratory, Sellafield, Cumbria.

Graphite is both a moderator and a structural component of reactors such as Advanced Gas Cooled Reactors (AGRs) and future reactor designs. For safe operation and easier decommissioning, the structural integrity and functionality of the core need to be ensured. Hence, core assessments need to include accurate date of physical, chemical and mechanical property measurements of the graphite components.

The properties of samples trepanned from AGR reactor cores are measured at the Active Handling Facility (AHF), Windscale through a variety of experimental apparatus. Continuous improvement of these techniques is key to supporting the current fleet of AGRs and extending their operational life.

As part of the government's Coated Particle Fuel (CPF) programme, UKNNL is installing a TRI-Structural ISOtropic (TRISO) fuel manufacturing pilot plant at Preston. The matrix graphite will be fabricated at UKNNL Workington, and a new non-active graphite characterisation laboratory at the same location will support the CPF programme by conducting quality assurance on the manufactured graphite matrix for future reactor designs.

This presentation will cover recent improvements to measurement techniques relevant to the AGR campaigns and provide insight into how UKNNL is applying this knowledge and experience to future reactors.

11h50 **Multi-scale Microstructural Characterisation and Planned Investigation of Coupled Irradiation and Fission Product Interactions in Nuclear Graphite**

Emily Aradi, Xun Zhang, Alex Theodosiou, and Abbie Jones
Nuclear Graphite Research Group, Department of Mechanical and Aerospace Engineering, School of Engineering, Henry Royce Institute, University of Manchester, M13 9LP, UK

Advanced reactor technologies such as High Temperature Gas-cooled Reactors (HTGRs) and Molten Salt Reactors (MSRs) are designed to operate at elevated temperatures and extended service lifetimes, placing increasing demands on nuclear graphite performance. Under these conditions, graphite undergoes microstructural evolution driven by irradiation-induced defects, which may alter transport pathways for fission products, influencing their retention, migration, and release behaviour. Despite its importance, the relationship between graphite microstructure, radiation damage, and fission-product transport remains poorly understood. Developing a mechanistic understanding of this relationship is therefore critical for qualifying graphite materials in next-generation reactor systems.

This study investigates four graphite materials, including HOPG, IG-110, 2114, and PCEA, using a multi-scale characterisation approach comprising XCT, SEM, TEM, STEM, and the novel four-dimensional STEM (4D-STEM, which provides local strain and defect information at the nanoscale) to establish baseline microstructural and crystallographic characteristics prior to irradiation. Building on this dataset, a systematic experimental programme has been developed to examine irradiation-induced defect evolution and its influence on the transport and retention of representative fission products (Ag, Sr, Kr, and Eu). Comparative analysis across the graphite grades will provide new insights into microstructure-dependent transport mechanisms, supporting predictive modelling for the safety of future HTGR and MSR applications.

12h10 **Progress on the UK ENLIGHT Programme - <u>En</u>abling a <u>Li</u>fecycle Approach to <u>Gr</u>ap<u>hit</u>e for AMR**

A.N Jones[1], C. Sharrad[1], G. Hall[1], M. Mallon[1], A. Theodosiou[1], X. Zhang[1], P. Mummery[1], K. Jolley[2], K. Jones[3], D. Liu[4], and T. J Marrow[5]
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The UK's aspiration to expand new nuclear capacity as a basis for clean energy growth includes a significant role for Advanced Modular Reactors (AMRs). While the UK has historically led in graphite reactor technology through the Advanced Gas-cooled Reactor fleet, this capability risks erosion during the transition to next-generation systems.

ENLIGHT, an ambitious 5-year UK programme, provides a coordinated pathway to repurpose and strengthen UK expertise, focusing on graphite innovation. Now one year into the programme, significant experimental and modelling capabilities have been established, collaborative research is underway, and initial data on recycled and AMR-relevant graphite systems are emerging. Early engagement with industry and regulators is shaping pathways toward material qualification.

The overarching aim of ENLIGHT remains to design and deliver sustainable graphite for AMR deployment, whilst aiming to transform irradiated graphite waste into a resource. This is achieved through three integrated themes: (1) Sustainable Graphite: retrieval, treatment and recycling; (2) Graphite Selection & Design to understand degradation modes; and (3) Graphite Performance underpinning behaviour under representative AMR environments. Through its combined programme of research, collaboration and skills development, ENLIGHT is contributing to the expansion of the UK graphite research community to support the evidence base required for future deployment.

12h30 *SESSION SUMMARY: Dr James Reed*

12h40 *Lunch*

Pore Structure and Weight Loss — Session Chair: Prof A.J. Wickham

14h00 **Advances in the Simulation of the Void Structure of Nuclear Graphite by Inverse Modelling of a Range of Experimental Characterisation Methods**

G.P. Matthews[1], G.M. Laudone[1], B. Moresby-White[2] and K.L. Jones[1]
[1]University of Plymouth and [2]University of Manchester, UK.

Monitoring the change in the void structure of nuclear graphite under intense radiation is an important aid in predicting the overall safe generation life of a reactor. However, the void structure of nuclear graphite is very difficult to characterise experimentally. Gilsocarbon graphite has a highly heterogenous void structure. Next generation graphites are typically more homogeneous, but nevertheless have void structures that collapse under the higher applied pressures of mercury used in mercury porosimetry. In this presentation, we show how density measurements, mercury porosimetry, gas adsorption and image analysis of electron micrographs can all be used to provide input data for inverse modelling to produce a simulated fully three-dimensional void structure covering the full range of void sizes. None of the experimental characterisations give a complete picture, and an important finding from our recent work is that the range of applicability of each measurement must be accurately known before input to the inverse modeller, which combines all the partial findings into a complete void structure which matches them all. Once constructed, the simulated void network can be used to predict a range of properties, for example absolute permeability and the extent of unwanted intrusion of non-wetting phases such as molten salt.

14h20 **Hybrid Approaches for Modelling Graphite Oxidation within the UK AGR Fleet**

J. Taylor[1], D. Kent[1], R. Lincoln[1], A Young[1], M Joyce[1], M Bradford[2], M Griggs[2] and E. Young[2]

[1] Frazer-Nash Consultancy Ltd
[2] EDF Energy Generation

Forecasting graphite weight loss in Advanced Gas-Cooled Reactors involves two primary modelling approaches: mechanistic models that simulate underlying physical and chemical processes, and statistical models that infer trends from operational data. Mechanistic models offer interpretability but can struggle to align with observations in complex systems and often rely on unverified assumptions. Statistical models align well with data but may lack transparency and can be unreliable when applied beyond their training range. Each approach has trade-offs, particularly where credible, well-evidenced modelling is essential for regulatory confidence.

This work presents results from the Hybrid DIFFUSE Approach (HDA), designed to combine the benefits of both approaches. It retains a physical FEAT-DIFFUSE base to aid interpretability and adequately conservative projections in data-sparse regions, while empirical scaling improves the fit to trepanned sample measurements where available. A Gaussian Process Regression approach captures spatial variation and model fitting uncertainty is accounted for *via* bootstrapping techniques.

The HDA provides a balanced and credible method for forecasting graphite behaviour, enhancing confidence in the models that underpin operational safety cases. These forecasts contribute to more informed decisions around core monitoring strategies, inspection planning, and longer-term investment in the continued operation of AGR assets under varying reactor conditions.

14h40 **Beyond Improved Fit: Lessons from the Hybrid DIFFUSE Approach**

E. Tan and N. Warren

HSE Science and Research Centre, Harpur Hill, Buxton, SK17 9JN

Reliable prediction of graphite weight loss in Advanced Gas-Cooled Reactors (AGRs) underpins safety case decisions. The UK regulator has requested an independent review of the Hybrid DIFFUSE Approach (HDA), developed by Frazer Nash Consultancy on behalf of

EDF Energy. The HDA modifies the FEAT-DIFFUSE model by replacing a uniform oxidation parameter (the Attack Rate Multiplier, ARM) with a spatially and temporally varying surface derived using Gaussian Process Regression (GPR).

Introducing a spatially varying, data-informed ARM improves agreement between model predictions and measured weight loss where inspection data exist. This demonstrates the practical value of combining physics-based models with data-driven adjustments to capture spatial effects that are otherwise difficult to model.

The review shows that key modelling choices, particularly around data treatment, parameter selection, and validation, can substantially influence results. There should therefore be careful treatment of any conservatisms from the outset.

The key message is that the HDA, and the use of GPR more generally, can substantially improve predictive performance, but confidence in their outputs relies on more than improved agreement with data. Confidence in their application in safety cases is further supported by understanding of model behaviour, sensitivity studies, treatment of uncertainties, and clear justification of modelling choices.

15h00 **Use of Eddy Current and Trepan Data to Build Understanding of AGR Graphite Cores**

R. Newson(1), J. Nicholas(1), E. Young(2), T. Taylor(2) and M. Brown(2)
[1]Quintessa Ltd., The Hub, 14 Station Road, Henley-on-Thames, Oxfordshire RG9 1AY
[2]EDF Energy, Javelin House, Building 1420, Charlton Court, Gloucester Business Park, Gloucester GL3 4AE

The graphite bricks comprising the cores of the UK's Advanced Gas Cooled Reactors (AGRs) are subject to loss of mass due to chemical processes enhanced by the high levels of irradiation. This "weight loss" is important from a safety perspective as the graphite is both the moderator and the structural material of the reactor core.

The two main inspection techniques of importance to monitoring graphite weight loss are eddy current scans and trepans. These have been used routinely to monitor weight loss in fuel channels. Recent development of a new trepan tool has enabled samples to be taken from control rod channels at Hartlepool (HRA) and Heysham 1 (HYA). Previously, the only direct information on graphite weight loss in these channels was from eddy current scans.

Eddy current scans cover the whole brick surface and are non-destructive but have relatively shallow penetration into the graphite and require careful calibration. Trepan samples give a more precise measurement of weight loss at the trepan locations, including deeper into the bricks, but are more spatially limited. Here we discuss how information from these two sources can be effectively combined to reduce uncertainties in each dataset and build understanding of the graphite weight loss.

Structural Integrity (1): Methods and Observations

Session Chair: Prof Dong Liu, University of Oxford

15h20 **Continuous Validation of the EDF Integrated Methodology (EIM) Material Properties Model**

D. Kent[1], R. Gray[1], G. Horne[1], M. Joyce[1], M. Bradford[2], E. Young[2]
[1] Frazer-Nash Consultancy Ltd
[2] EDF Nuclear Operations

The EDF Integrated Methodology (EIM) material properties model is a physically-informed framework used to predict the evolution of graphite material properties under irradiation and radiolytic oxidation within the cores of the UK's fleet of Advanced Gas-cooled Reactors. Forecasts from this model are used to support cracking and damage tolerance assessment safety cases, so it is important that the model is continuously reviewed to maintain confidence in these safety cases.

This is principally achieved through inspection and monitoring of the reactors, which provide a steady stream of new operational data, including dimensional measurements and trepanned sample properties, to test the extant model and ensure its forecasts remain appropriate beyond the domain of original calibration, such as for interstitials. These validation exercises are also used to inform where further model development may be required to maintain alignment with the core condition. The EIM model together with any development updates is periodically calibrated to this data, which provides a basis for further analysis to support future safety cases.

This presentation provides a background of the EIM model and the available data, followed by a summary of the routine validation exercises performed and the insights into graphite behaviour arising from long term reactor service.

15h40 *Coffee break*

16h10 **Using Survival Analysis Techniques to Predict the State of AGR Graphite Cores**

C. Watson[(1)], D. Bailey[(1)], P. Robinson[(1)] and M. Bradford[(2)]
[1]Quintessa Ltd., The Hub, 14 Station Road, Henley-on-Thames, Oxfordshire RG9 1AY
[2]EDF Energy, Javelin House, Building 1420, Charlton Court, Gloucester Business Park, Gloucester GL3 4AE

The graphite bricks comprising the cores of the UK's Advanced Gas Cooled Reactors (AGRs) are subject to radiation and thermal stresses that lead to damage over time. This damage includes full and partial height axial cracks associated with the interlocking keyways, of which there may be multiple within a single brick, and horizontal cracks associated with the seal ring groove wall. Where cracks intersect, or meet edges of bricks, fragments can form.

The cores are subject to safety cases that determine how many of each feature can be tolerated during operation. Regular inspections monitor the graphite, with observations feeding into models of the core that are used to predict the evolution of the damage over time. Models based on the techniques of survival analysis have recently been developed and applied to Heysham 2 (HYB) Reactor 7 and Torness (TOR) Reactor 1, where the largest numbers of cracks have been observed to date across the current fleet. Bootstrapping and sampling are combined with interval-censored survival curve fits to incorporate uncertainty and variability in the predictions. Techniques such as trending of parametric fits, multi-state analysis and survival regression have been applied to deal with limited data and investigate dependencies.

16h30 **Predicting Graphite Fuel Brick Bore Condition from Routine Fuel Grab Load Traces**

Callum Sharp[1], Thomas Storey[1], Stephen McArthur[1], Julia Wareham[2], Alexander Price[2], and Mary Rajeswaran[2]
[1]Bellrock Technology, Level 5, 9 Haymarket Square, Edinburgh, Scotland, EH3 8RY
[2]EDF Energy, Javelin House, Building 1420, Charlton Court, Gloucester Business Park, Gloucester, GL3 4AE

Per-brick bore measurements are critical to managing the UK's Advanced Gas-cooled Reactor (AGR) fleet: graphite cores cannot be replaced and set each reactor's operating lifetime, yet in-channel inspections are limited in number and frequency. We present two complementary models that infer per-brick bore geometry from Fuel Grab Load Traces (FGLTs) - the load on the refuelling grab as it moves the fuel through a channel during routine refuelling across multiple AGR reactors, with matched charge–discharge pairs. The first predicts inter-brick step changes (top, middle and bottom diameter discontinuities between adjacent bricks, indicative of open axial cracks) using a Ridge linear trend, RFECV feature selection, bagged XGBoost residuals and a gradient-boosted residual head, achieving 0.2967 mm MAE and 0.6354 R^2 on a chronologically held-out test set. The second predicts absolute top, mid and bottom brick diameters using the same backbone with BorutaSHAP selection on out-of-fold TreeSHAP attributions, a Yeo-Johnson residual transform and independently modelled inner (4, 5) and outer (3, 6–10) layers, achieving 0.4831 mm MAE and 0.7319 R^2. Together the models turn data the reactors already produce into a fleet-wide picture of core condition, informing inspection planning and supporting operation.

16h50 **Experimental Investigation of Sleeve–Debris Snag Behaviour and Outcomes in AGR Fuel Channels (HY2/TOR)**

A. McKenna, M. W. Davies
Amentum

As fuel brick degradation progresses within Advanced Gas-cooled Reactors (AGRs), the prevalence of Keyway Root Cracks (KWRCs) is expected to increase. Within Heysham 2/Torness (HY2/TOR), specific fuel-brick end-face geometry can facilitate the formation of Seal Ring Groove Wall (SRGW) fragments following KWRCs. If mobilised into the channel annulus between the fuel stringer and the fuel channel bore, SRGW debris could entrain with the fuel stringer and obstruct fuel retrieval.

When a fuel stringer is concentric within a fuel channel, there is insufficient space for SRGW debris to entrain with a fuel sleeve, and it can only entrain with the anti-gapping unit (AGU). However, in cases of extreme stringer bowing and/or channel distortion, it is conceivable that a SRGW fragment, if fully detached from its parent brick, could mobilise fully into the eccentric channel annulus, as debris, and entrain with a fuel sleeve or brick bore during fuel movement. A multi-stage testing programme was completed using the "5-Layer Fuel Channel Debris Rig" to investigate fuel movement outcomes when debris entrains with a fuel sleeve or brick bore. This programme has enabled sleeve–debris snag characteristics and subsequent consequences to be analysed; the paper describes the rig, methodology, and analysis.

17h10 **Automated Inspection and Measurement of Graphite Cracks**

Felipe F. Freitas[1], Innes Simon[2], Jonathan Wright[2] and Jordan Murkin[1]
[1]EDF UK R&D
[2]EDF Energy Nuclear Generation Ltd.

As internal brick stresses evolve across the operational lifetime of AGRs, it is expected that the frequency of cracks and critical defects will rise, and as such, the ability to precisely locate, quantify, and monitor crack development through successive inspections is vital for understanding graphite moderator performance. To aid the analysis of graphite inspection across the AGR fleet, we have developed a robust methodology utilising computer vision and image processing to automate the detection, labelling, and quantification of cracks within chanoramas. This approach enhances the ability to record and monitor fracture progression. Our model achieves an accuracy of up to 95% in crack identification and offers subpixel-level measurement precision.

Our crack detection framework integrates multiple specialised architectures to identify and mark the cracks for further processing. After a crack is detected the measurement of its dimension is carried out by our measurement system, which can achieve subpixel precision levels by leveraging a sequence of image filtering and adaptive thresholding algorithms designed to isolate crack boundaries and their medial axes used for estimate the pixel intensity gradients and calculate the distance between opposing edges. This approach will allow engineers to rapidly and accurately identify and measure cracks across the AGR fleet.

17h30 *SESSION SUMMARY: Prof Dong Liu*

Dinner

Wednesday 23rd September 2026

08h30 Welcome and housekeeping: **GROUP PHOTOGRAPH**
(Dress code: business casual)

Structural Integrity (2): Prediction and Modelling

Session Chair: Dr A Harker, UCL

08h40 **Modelling Framework for Fracture in Nuclear Graphite**

Chris Pearce, Andrei Schvarts, Callum Runcie, Annaya Bijaya, and Ł. Kaczmarczyk
James Watt School of Engineering, University of Glasgow

This paper presents a finite element framework for modelling brittle fracture in nuclear graphite. Crack propagation is governed by the Griffith fracture criterion, with the propagation direction determined by the maximum energy-release-rate criterion, consistent with the principle of maximum energy dissipation. The predictive capability of the framework is demonstrated through comparisons with experimental results, while its robustness is illustrated through simulations of representative reactor-brick geometries. The formulation has also been extended to account for material heterogeneity, irradiation-induced internal stresses under reactor conditions, and the coupling of brittle fracture with contact. This FE technology has been deployed within Amentum.

The presentation focuses on computational scalability and recent developments that enable the robust simulation of large systems, including arrays of graphite bricks. These developments include a hybridised mixed solver, novel approaches to contact, and the evaluation of J-integrals in heterogeneous materials.

We also present the integration of the solver with Amentum's UMAT material model, enabling high-fidelity simulations of crack propagation throughout reactor operation, including both at-power and shutdown conditions.

Finally, we introduce MoFEM, the finite element solver underpinning this work (https://mofem.eng.gla.ac.uk). MoFEM has been developed at UofG with support from EDF and is used by Amentum.

09h00 **Handling Complexity in CrackSim Modelling of AGR Graphite Cores**

P. Robinson(1), R. Newson(1), M. Bradford(2) and A. Price(2)
[1]Quintessa Ltd., The Hub, 14 Station Road, Henley-on-Thames, Oxfordshire RG9 1AY
[2]EDF Energy, Javelin House, Building 1420, Charlton Court, Gloucester Business Park, Gloucester GL3 4AE

The graphite bricks comprising the cores of the UK's Advanced Gas Cooled Reactors (AGRs) are subject to radiation and thermal stresses that lead to damage over time. This damage

includes full and partial height axial cracks associated with the interlocking keyways, of which there may be multiple within a single brick, and horizontal cracks associated with the seal ring groove wall at HYB and TOR. Where cracks intersect, or meet edges of bricks, fragments can form.

The cores are subject to safety cases that determine how many of each feature can be tolerated during operation. Regular inspections monitor the graphite, with observations feeding into models of the core that are used to predict the evolution of the damage over time. CrackSim has been used to simulate the evolution of the core, enabling predictions at inspections and longer term forecasts to be made to support safety cases for continued operation. Over time, the complexity of the processes involved has increased, threatening the feasibility of undertaking CrackSim simulations in a timely fashion. The major sources of complexity are discussed along with the tactics that have been used, or are under consideration, to handle these efficiently within the CrackSim code.

09h20 **Simulation of Graphite Fuel Brick Cracking using an Explicit Finite Element Model**

S. Bounds, J. Wilde and S. Ashton
AtkinsRéalis

An explicit finite element modelling approach has been developed to support the assessment of the risks associated with cracking of graphite fuel bricks. This combines a material model for simulating the irradiation induced deformation and stress with a methodology for predicting crack initiation and propagation. This is implemented in a framework that allows assemblies of graphite bricks to be modelled to capture the complex contact interactions that can exist following cracking.

This paper presents the use of the model to simulate cracking in an assembly of bricks. The work has sought to understand the mechanisms contributing to the particular crack patterns observed during inspections of reactor cores.

09h40 **Finite Element Simulation of Graphite Crack Initiation and Growth using an Erosion Based Material Model**

S. Bounds, J. Wilde and K. King
AtkinsRéalis

An explicit finite element modelling approach has been developed to support the assessment of cracking of the graphite fuel bricks in the EDF Advanced Gas-Cooled Reactor fleet. The material model simulates the dimensional change and stress in the graphite bricks due to irradiation and uses a method based on element erosion to predict the initiation and growth of cracks. The erosion method benefits from ease of implementation in a model that also accounts for the other complexities involved in simulating the core behaviour.

Development of the erosion method has made use of an extensive range of rig tests on graphite specimens. These tests provide a variety of data for validation, including force-displacement histories and fracture surface morphology, which have been compared against models using the erosion method.

This paper presents a summary of the erosion method, together with details of the model validation. The validation demonstrates the method to be capable of predicting the timing and progression of cracks across a range of features and loading applicable to graphite fuel bricks.

10h00 **Influence of Restraint by Surrounding Core on the Formation of Cracks in AGR Graphite Bricks: An AGRIGID-Informed uCBNA Study**

R. Shaikh, S. Foster, C. Minors
Amentum

Graphite cores of Advanced Gas-cooled Reactors (AGRs) contain several thousand fuel and interstitial bricks that interact throughout reactor life. The Unified Cracked-Brick-Neighbourhood Array (uCBNA) methodology is used to assess local crack development within fuel brick arrays; however, simplified boundary condition assumptions may not adequately represent the restraint imposed by the surrounding core. A methodology is presented for incorporating displacement boundary conditions derived from the lower-resolution whole-core AGRIGID "stick-and-spring" (global) model into detailed "solid-body" (sub-model) uCBNA finite element analyses.

A series of 1x1x5 column models were analysed to investigate crack evolution for aligned, 90° offset and 180° offset primary crack configurations in vertically adjacent bricks. AGRIGID-derived displacement histories were applied through a user-defined boundary condition framework, enabling the influence of surrounding core restraint on crack initiation to be assessed.

The results demonstrate that the detail of restraint by the surrounding core has a measurable effect on crack progression, influencing both the timing and location of crack initiation. The methodology provides a practical means of transferring surrounding core behaviour into local fracture assessments and is readily applicable to other (sub-model) finite element frameworks, including the Mesh-Oriented Finite Element Method (MoFEM), enabling representative surrounding core restraint to be incorporated into crack propagation analyses.

10h20 *Coffee*

10h50 **A Whole-Core Model of Keyway-Root Cracking and Damage Progression Fitted to Inspection Findings**

Timothy Yates, Nicholas Warren, Emma Tan
HSE Science and Research Centre, Harpur Hill, Buxton, SK17 9JN

EDF's regular inspections of AGRs provide a snapshot of the current condition of the graphite core. Whole core forecasts of keyway root cracking and subsequent damage are required to provide confidence in acceptability of continued operation. However, predictions made by statistical models fitted to inspection data will always be uncertain and the UK regulator has requested independent advice on the source and impact of these uncertainties. Hence, we have developed a whole core model, fitted to inspection findings at HYB/TOR using Markov Chain Monte Carlo methods, to make predictions of the core state and better understand the uncertainties. To predict the times and locations of primary and subsequent cracks the model uses brick level parameters to represent the potential crack initiation times at every keyway of each brick. These crack times are modified through interactions with neighbouring bricks, and to represent the stress relaxations at the intact keyways after the first and each subsequent occurrence of a full height axial crack: both depend on crack locations. Partial and full height secondary cracks are modelled, as well as seal ring groove wall related damage processes. We present the model structure and predictions of inspection findings to demonstrate model validity.

11h10 **Strength of Graphite Components with Corners**

A.G. Steer
EDF Nuclear Operations, Javelin House, 1420 Charlton Court, Gloucester Business Park, Gloucester, GL3 4AE, UK

One of the less endearing features of structural integrity assessments of graphite components is the lack of a general measure for graphite strength with the resulting need to test full-size components with representative geometry and loading. This paper reviews available experimental data on components with notches and corners from a fracture mechanics perspective. The findings are combined with a review of available measurements of stress-strain characteristics to propose a two-stage approach combining a non-linear stress analysis using a statistically-based, graphite damage model with a fracture mechanics analysis.

11h30 **Probabilistic Modelling of Potential Cracking Pathways in AGR Graphite Cores**

A. Farrokhnia, A. Jones and G. Hall
Nuclear Graphite Research Group, School of Mechanical and Aerospace Engineering, The University of Manchester

In UK Advanced Gas-cooled Reactors, graphite bricks are utilised for both structural components and neutron moderators/reflectors. During operation, bricks undergo degradation due to fast neutron irradiation, high temperatures, and radiolytic-oxidation, resulting in stresses and material degradation that can lead to crack-initiation. Finite-element analyses are widely used to assess core integrity. However, such deterministic approaches assume

homogeneous material strength and do not capture the probabilistic nature of any defect-driven failure.

This work presents a probabilistic methodology for evaluating cracking scenarios, accounting for stress redistribution following crack formation. Such assessments are essential because cracking within a graphite core is not an isolated event; individual cracks can influence the location, likelihood, and sequence of subsequent cracking, leading to multiple potential pathways of core evolution.

Our results demonstrate that the location and likelihood of cracks are dependent on the sequence of preceding cracks. Changes in load transfer can alter cracking pathways, increasing the likelihood of alternative crack sequences and resulting in a broader distribution of possible cracking states. The analysis highlights low-probability cracking scenarios that would not be identified through conventional assessments. The methodology provides a means of quantifying uncertainty in crack progression and supports improved prediction of core condition throughout operation.

11h50 **ON-LINE PRESENTATION**

Peridynamic Damage Evolution in Nuclear Graphite: Simulation and Validation

S. Carpio[1], M. Jiang[1], M. Davies[2], D. Liu[1], and E. Madenci[3]
[1]University of Oxford, UK
[2]Marad Ltd, UK
[3]University of Arizona, USA

Nuclear graphite is a critical structural material in advanced nuclear reactors, where its integrity is governed by complex damage and fracture processes. This study presents a peridynamic modelling framework for simulating damage evolution in Gilsocarbon graphite and validating the predictions against experimental observations. The nonlocal nature of peridynamics enables the natural representation of crack initiation, propagation without requiring supplementary crack-tracking techniques or remeshing.

The proposed framework is applied to a double-notched four-point bending specimen made of Gilsocarbon graphite to investigate progressive damage development and fracture behaviour. Damage patterns, and load–displacement responses are compared with experimental measurements. The simulations accurately capture the onset of cracking, crack propagation from the notch roots, and the final failure mode observed in the experiments.

The results demonstrate good agreement between numerical predictions and experimental data, highlighting the capability of peridynamics to model progressive damage and fracture in quasi-brittle nuclear graphite materials. The validated framework provides a robust tool for assessing structural integrity and predicting failure in nuclear graphite components subjected to complex loading conditions.

12h10 **A Phase-Field Framework for Predicting Fracture in Irradiated AGR Graphite Components**

G. Rodríguez-Vellando Soriano[1], D. Liu[1], M. Davies[2] and E. Martínez-Pañeda[1]
[1] Department of Engineering Science, University of Oxford, Oxford, UK
[2] EDF Energy, UK

Predicting fracture in nuclear graphite components requires a method capable of capturing crack initiation and growth through complex geometries without prior knowledge of the crack path. This work presents an Abaqus phase-field framework in which cracks are represented by a continuous damage field and evolve through the competition between stored elastic energy and fracture energy. The formulation predicts crack nucleation, propagation, deflection, branching and coalescence on the original finite-element mesh, making it particularly suited to AGR moderator bricks containing complex stress concentrators such as keyways and methane holes. The framework has been validated against a broad set of unirradiated nuclear graphite experiments covering different grades, specimen sizes and loading configurations. The simulations reproduce the principal features of measured load-displacement responses and the experimentally observed crack paths. An irradiation-dependent graphite UMAT has also been developed to account for dimensional change, primary and secondary irradiation creep, thermal strain and their interactions. Coupling both developments enables three-dimensional simulations of ageing AGR fuel bricks, providing a physics-based tool for assessing crack initiation and propagation and supporting graphite-core integrity and lifetime evaluations.

12h30 *SESSION SUMMARY: Dr James Reed*

12h40 *Lunch*

Irradiated Graphite Retrieval and Disposal

Chair: Dr David Bradbury, Sirius Analysis

14h00 **Enabling Evidence-Based Decisions: An Integrated Technical Framework for Irradiated Graphite Management**

Chris Hutson, Amritpal Agar, David Bradbury, Anthony Wickham, Richard Harris
Sirius Analysis, Portsmouth UK

Extensive operational experience, inspection and research have provided a strong understanding of graphite behaviour, degradation mechanisms and radionuclide inventories. As AGRs enter decommissioning, the challenge is to apply this knowledge to support decisions on dismantling, treatment, waste management and disposal.

This paper presents a systems-based framework for evaluating irradiated graphite management options across degradation, characterisation, dismantling, treatment, waste classification, packaging and disposal. Integrating characterisation, inventory assessment, degradation modelling and waste management analysis, the framework identifies opportunities to reduce uncertainty, preserve future options, improve disposability and support evidence-based decision-making.

A central principle is decision-quality characterisation, where characterisation is driven by the decisions it must support rather than by material understanding alone. The aim is to generate evidence of sufficient quality and confidence to enable timely, defensible decisions and progressively reduce programme risk.

A key finding is that the greatest opportunities for optimisation lie at the interfaces between individual processes. Examples show how graphite cracking can influence dismantling approaches, while understanding carbon-14 distribution and retention can affect treatment selection, waste classification and long-term disposal requirements.

14h20 **Retrieval of Pressure Vessel Reactor Graphite Waste by Robotics and Subsurface Transferal and Interim Storage**

Martin Knights
LBA Ltd

Many options are being assessed for the retrieval of graphite waste from decommissioned UK Magnox Nuclear Power Stations. Recently a study was commissioned by NRS to identify, categorise and optioneer methodologies for removing the MLW and LLW pressure vessel waste using robust and remotely controlled robotic equipment and transfer the retrieved waste via shafts, adits and chambers constructed under the reactor building for interim processing and onward to interim underground storage chambers. The study considered existing and future technology that would enable such methodologies to successfully and safely retrieve reactor waste by underground means. The benefits of doing so were compared with current NRS baseline surface removal options. The study also considered the environmental, societal, cost and time benefits and `value ` of doing so.

14h40 **Short-List Options Development for Alternative Retrievals of Graphite from Reactor Cores**

R Harris, I Haigh, E Hankinson
Nuclear Restoration Services, UK

Nuclear Restoration Services' Graphite Innovation Programme is evaluating alternatives to the baseline for managing Magnox reactor graphite, aiming to reduce timescales and whole-life costs without compromising subsequent dismantling or safe-storage options. This paper describes Phase 1 feasibility work to identify and screen credible routes for early graphite retrieval allowing targeted removal of material with throughput that is aligned to potential treatment options.

The study consolidates information on graphite core configurations, reactor pressure vessels, access penetrations, in-core furniture, radiological conditions and waste-routing constraints across the Magnox fleet. Six retrieval concepts were developed, using access through the pile cap, primary gas ducts, thermal columns, standpipes, lower or side penetrations, and burst-can-detection ducts. These combine remotely operated or autonomous equipment with cutting, size reduction, vacuum, conveyor and packaged-export methods.

Initial coarse screening suggests that a small pile-cap opening offers a robust route aligned with established dismantling practice, while primary-duct access may be promising for selected reactor designs. Key uncertainties include core accessibility, in-core furniture removal, mixed-material handling, graphite particulates, tritium and dust control, interim waste routes, and the effects of graphite removal on residual dose and containment. The work establishes a basis for targeted characterisation, research and concept development before formal optioneering and preferred-option selection.

15h00 **Retrieval Pathways for Sustainable Nuclear Graphite Management**

N. Edwards, A.N. Jones, and A. Theodosiou
Nuclear Graphite Research Group, The University of Manchester, Henry Royce Hub, Oxford Road, Manchester, M13 9PL

ENLIGHT, *Enabling a Lifecycle Approach to Graphite for AMRs*, is developing technologies to support sustainable graphite lifecycle management in advanced modular reactors. Within Theme 1.2A, there is a focus on bulk retrieval and pre-processing of graphite from current and future reactor systems.

Graphite retrieval is important for both legacy decommissioning and future reactor operation. Existing graphite-moderated reactors contain large, irradiated graphite inventories that must be removed and managed, with potential opportunities for decontamination, recycling and treatment. In future high-temperature gas-cooled reactor concepts, graphite fuel or moderator blocks may require removal and replacement during service. For these systems, retrieval is expected to require intact block or component removal, making graphite handling a lifecycle challenge rather than only an end-of-life waste issue.

This work package will assess retrieval routes according to the form and future use of the recovered graphite. For legacy systems, intact retrieval remains the assumed baseline, while *in situ* size reduction may provide an alternative where intact removal is impractical or where particulate material can be integrated directly with selected treatment or secondary waste-management processes. By assessing retrieval route, material form, containment and recovery requirements, this work will define technical requirements for scalable graphite recovery across current and next-generation reactor systems.

15h20 **The Role of Impurity Concentrations in UK AGR Graphite Waste Radiological Inventories**

E. Murray[1], A. McNulty[1], S. Shaw[2], G. Whiley[2], A. Jones[3] and L. Hughes[1]*
[1] Cyclife UK Limited
[2] EDF Energy Nuclear Generation Limited, UK
[3] University of Manchester, UK

At Final Site Clearance (FSC), Advanced Gas-cooled Reactor (AGR) graphite will contain a range of radionuclides. These wastes will largely be classed as Intermediate Level Waste, of which the preferred disposal option in the UK is in a Geological Disposal Facility (GDF). However, uncertainties in radionuclide concentrations, particularly C-14, impact the accuracy of the UK Radioactive Waste Inventory and, consequently, the GDF inventory.

Graphite waste will constitute approximately 33% of total GDF waste volume, making accurate end-of-life inventory assessment crucial. Enhanced understanding may enable reclassification and alternative disposal approaches. An initial MCBEND study was performed to investigate the influence of graphite impurities on specific activities at FSC. Long-term decay-weighted fission neutron sources were used based upon EDF Hunterston B Core Follow data, enabling the neutron flux from the startup to end of generation to be simulated.

The results highlight the significant role that initial impurity concentrations play in AGR graphite waste radiological inventories and demonstrate the sensitivity of key radionuclides to assumptions regarding precursor impurity levels. Building on this initial study, opportunities to apply the methodology in support of wider UK graphite characterisation programmes, inventory development and future waste management approaches are discussed.

15h40 *Coffee*

16h10 **Short-listed Alternative Graphite Treatment Processes**

E Lancaster, I Haigh, R Harris
Nuclear Restoration Services, UK

The management of irradiated graphite remains a significant challenge during decommissioning of the UK's graphite-moderated reactors. This is due to the large inventory of graphite, the presence of long-lived radionuclides, and the current strategy of managing core graphite as intermediate-level waste, requiring disposal in a Geological Disposal Facility. Developing alternative treatment options could reduce disposal volumes, improve graphite and radionuclide resource efficiency, and support consideration of alternative decommissioning processes and site end states.

This study evaluates a range of treatment pathways for irradiated graphite, including direct disposal, thermal treatment, chemical decontamination, and waste recycling. Each option was assessed against key technical, environmental, safety, and socio-economic criteria to determine feasibility and identify a shortlist of options for further evaluation under the NRS Graphite Programme.

The findings indicate that thermal treatment offers the most promising options to reduce waste volumes and/or divert graphite waste from the Geological Disposal Facility. They may also enable radionuclide separation and reuse, creating potential commercial opportunities.

Overall, the study provides a structured basis for decision-making and supports the development of a graphite-focused research and development programme at NRS.

16h30 **Comparative Life Cycle Assessment Study of UK Baseline and Alternative Baseline Graphite-Management Strategies**

J. Kirk, J. Smith and J. Goodwin
Cyclife UK

This study evaluates the life cycle environmental performance of two waste management strategies for irradiated reactor graphite in the UK: the current Baseline Strategy of deferred disposal to a Geological Disposal Facility (GDF) and an Alternative Baseline Strategy involving early retrieval, thermal treatment, and diversion of carbon dioxide to Carbon Capture and Storage (CCS). A gate-to-grave life cycle assessment (LCA) was undertaken using one kilogram of reactor graphite processed and permanently disposed of as the functional unit.

Results indicate a climate change impact of 25.07 kg CO_2-Eq for the Baseline Strategy and 15.94 kg CO_2-Eq for the Alternative Baseline Strategy per functional unit, representing a reduction of approximately 36%. Disposal infrastructure was identified as the major hotspot in both scenarios (approximately 91 - 97% of total impacts), with reinforcing steel used in disposal infrastructure driving the majority of burdens. The Alternative Baseline Strategy also demonstrated lower impacts for material resource use, freshwater eutrophication, freshwater ecotoxicity, and land use. Sensitivity analyses indicated that the relative performance of the two strategies remained unchanged under variations in GDF material requirements and CCS performance. The assessment therefore indicates that the Alternative Baseline Strategy could be the environmentally preferable option of the two strategies evaluated.

16h50 **From R&D to Industrial Application: Proposal for the Basic Design of a Pilot Plant for Graphite Processing**

G. Laurent
IN Solutions, *in cooperation with* Deep-Geo (USA)

Graphite is a unique type of waste because it is very pure, it contains volatiles RNs (which pose major challenges for containment in disposal), but which can, If separated, become recyclable materials with high recovery value.

Extensive, high-quality R&D and summaries by international bodies (EC, IAEA) authorise immediate implementation of the basic design of a pilot plant .

This pilot plant must be highly flexible (it could operate in batch mode or continuously) and modular to fulfill the required waste-acceptance criteria of each country (the goal could be decontamination or total destruction of the matrix), with collection of volatiles and their recycling.

This pilot will help to improve understanding of C-14 and Cl-36 behavior by processing larger quantities of graphite. An in-depth knowledge of overall performance and potential for improvement of the initial design could be reached by testing different gas composition, temperatures, potential use of CECE and PSA.

The future design of the full-scale facility will be prepared by establishing the set of process and instrumentation diagrams, the mass and energy balances and a rough cost estimate.

The combination of these capabilities provides: confidence to improve C-14 decontamination levels, possibility to recycle a part of the extracted C-14, collect ashes and the Cl-36, capture tritium and recycling it.

The target of this pilot is to make the demonstration that the original matrix could be "break down" into 4 separately manageable/recyclable components: Volatiles, Ash, Extraction gases, Recirculating water.

17h10 **Cl-36 Distribution in Magnox Reactor Systems Derived from Irradiated Graphite**

Alan Fisher, Joshua Weatherill
Nuclear Restoration Services, Hinton House, Birchwood Park Avenue, Risley, Warrington, WA3 6GR

Chlorine-36 (Cl-36) is a key radionuclide in the long-term management of irradiated nuclear graphite. It is generated predominantly through neutron activation of Cl-35 impurities in reactor core graphite and is significant because of its long half-life, environmental mobility and bioavailability. Reliable prediction and measurement of Cl-36 inventories are therefore important to the safety assessment of graphite treatment, storage and disposal options.

Radiochemical analysis of samples collected during Magnox reactor and primary circuit characterisation provides evidence of the release, transport and re-deposition of Cl-36. Analysed levels of Cl-36 in Trawsfynydd reactor 1 core graphite are an order of magnitude lower than those predicted by activation modelling, while Cl-36 is also observed throughout the primary circuits and boilers for reactors at Dungeness A, Hunterston A and Wylfa. These observations align with international studies on the thermally-driven release of chlorine from irradiated graphite, resulting in partial loss and redistribution of the Cl-36 inventory during reactor operation.

Furthermore, there is evidence of at least two mechanisms for the deposition of Cl-36 across reactor primary circuits associated with different chemical speciation. Some Cl-36 is intrinsic to graphitic dust carried within the stream of coolant gas, but it also appears to be present in a different chemical form whose deposits are inversely correlated with temperature gradient across reactor boilers.

Different approaches to Cl-36 determination in graphite are also described, with commentary.

17h30 **Validating a Null Result: Wigner Energy Assessment of Irradiated FRJ-2 Reflector Graphite**

D. Boya[1], A. Wickham[2], H-J. Steinmetz[3], M. Grimm[4], E. Langegger[5], M. Knebel[6], S. Merz[1], G. Steinhauser[1]

[1] TU Wien, [2] Nuclear Technology Consultancy, UK, [3] FH Aachen, [4] DMT Group, [5] JUNO Nuclear Engineering & Consulting GmbH, [6] Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH

For the safe management and disposal of irradiated reactor graphite, it is essential to determine whether stored Wigner energy could pose a risk during storage, disposal or accident conditions. This work investigates ARS reflector graphite from the FRJ-2 (DIDO) research reactor, obtained from a segmented drill core representing relevant neutron exposure. Measurements by differential scanning calorimetry using a Mettler Toledo TGA/DSC 3+ showed no meaningful Wigner energy release up to 800 °C. Since no distinct exothermic signal was observed, central challenge was to verify that this result was caused by the material itself rather than by limitations of the measurement setup. The method was therefore validated against two independent reference materials: BEPO reactor graphite with established Wigner energy content, enabling comparison with published release-rate curves [1] and in-house produced graphite standards by neutron irradiation at the TRIGA Mark II reactor in Vienna. Additional confirmation measurements are planned at Nuclear Engineering Seibersdorf using a pure DSC instrument previously applied in Wigner energy investigations [2]. All reference materials produced reproducible exothermic responses, whereas the FRJ-2 samples followed the baseline, indicating no safety-relevant Wigner energy content.

17h50 *SESSION SUMMARY: Dr Richard Harris, Nuclear Restoration Services*

19h15 *for* 19h30 *Conference Dinner*

Thursday 24th September 2026

08h50 Welcome, housekeeping

PLEASE NOTE: ROOMS MUST BE VACATED AND KEYS RETURNED BY 10h00

Graphite at the Molecular Scale — Chair: Dr Peter Minshall

09h00 **Strain-Driven Defects and Ripple Networks in Bilayer Graphene: From Atomistic Mechanisms to Mesoscale Behaviour**

Kenny Jolley, Paul Mouratidis, Daria Nasartdinova
Dept. of Chemistry, Loughborough University Epinal Way, Loughborough LE11 3TU, United Kingdom

Layered 2D materials such as bilayer graphene exhibit a distinctive mechanical response arising from the interplay between strong in plane covalent bonding and comparatively weak interlayer van der Waals interactions. This competition gives rise to a rich spectrum of deformation pathways, in which in plane strain is accommodated not only through dislocation climb and glide but also via out of plane buckling, delamination, and the formation of complex ripple structures. Using large scale molecular dynamics (MD) simulations, we access deformation regimes that bridge atomistic and mesoscale behaviour.

We examine the role of partial basal dislocations of mixed screw and edge character, introduced as dipoles within bilayer systems. Structural optimisation reveals that the edge component of these dislocations plays a dominant role in driving out of plane buckling, with 60° mixed partials adopting energetically favourable buckled configurations consistent with experimental observations. Building on this, we explore how dislocation pile ups lead to surface ripple formation under increasing compressive strain, identifying a critical threshold beyond which flat configurations become unstable and ripple structures emerge as the preferred state. The insights gained provide a predictive basis for tailoring mechanical and structural properties in 2D heterostructures.

09h20 **Investigating Irradiation Creep in Gilsocarbon Graphite using X-ray Diffraction**

Matthew Harrison [a], Ming Jiang [a], Robin Scales [a], Mark Bradford [b] and Dong Liu [a]
[a] Department of Engineering Science, University of Oxford, Oxford, UK
[b] EDF Nuclear Operations, Gloucester, UK

Project ACCENT (AGR Carbon Creep Experiment) was a five-year neutron irradiation campaign conducted by NRG (now NRG PALLAS) on behalf of EDF Energy to extend the irradiation creep dataset for Gilsocarbon graphite. X-ray diffraction (XRD) measurements were carried out at NRG to: (i) examine changes in lattice spacing and strain with irradiation damage; and (ii) compare reference and stressed specimens to investigate possible changes in the crystal structure associated with irradiation creep.

In this study, XRD data acquired by NRG after each irradiation phase are examined. The diffraction peaks are analysed in terms of their position, width, and asymmetry using a range of peak-fitting functions. The results are reported with respect to two aspects: (i) the estimated lattice parameters, coherence lengths, and lattice strains as a function of radiation fluence; and

(ii) differences between reference and stressed specimens. The latter indicate that the applied load has no discernible effect on the XRD data. The observed crystallographic changes therefore appear to arise entirely from irradiation damage.

Structural Integrity (3): Crack Behaviour

Chair: Mr Mike Davies, Amentum

09h40 **Studying Crack Initiation, Permanent Damage, and Residual Strain in Fine-Grained Nuclear Graphite using Double Notch Four-Point Bend**

Ming Jiang[1], Aaron Graham[1], Sophia Carpio[1], Mark Davies[2], Jim Reed[3], Houzheng Wu[4], Dong Liu[1*]*
1 Department of Engineering Science, University of Oxford, Oxford, OX1 3PJ, UK
2 Marad Limited, UK
3 EDF Nuclear Generation Limited, Gloucester GL3 4AE, UK
4 SIAMC Advanced Materials Co., Ltd, Zhejiang, China

Fine-grained nuclear graphite materials are important for advanced nuclear fission reactors and proton accelerator beamline target systems. Mechanistic understanding of the crack initiation and fracture behaviour of fine-grained graphite is considered important. In the present study, a novel double notch four-point bend (DN-4PB) test method has been implemented to study the crack initiation, permanent damage, and residual strain in fine-grained nuclear graphites including SNG742, T220, IG-430U, IG-450U, and IG-510U. *In-situ* two-dimensional digital image correlation (2D-DIC) was used to monitor the deformation process and to quantify the maximum principal strain fields at both notches. When fracture occurred at one notch, the maximum principal strain at the remaining notch relaxed by approximately 80% in magnitude for both grades of graphite. Complete unloading of the beam specimens yielded 8 – 20% residual maximum principal strain at the remaining notch. Further DN-4PB tests have been conducted inside a scanning electron microscope (SEM) for *in-situ* SEM imaging. During loading, multiple microcracks have already been initiated and propagated at both notches at very low load levels (at most ~ 26 % of failure load (P_f) for SNG742 and at most ~ 24% P_f for T220). As the load increased, stable growth of multiple microcracks was observed up to 93.7% P_f for SNG742 and 92.5% P_f for T220 prior to fracture instability despite the short crack extension. The development of permanent damage and residual strain, as well as implications to nuclear graphite components will be discussed.

10h00 **Mixed mode I/II fracture of Gilsocarbon graphite at 1000°C**

Ming Jiang[1], Sophia Carpio[1], Jim Reed[2], Dong Liu[*]
1 Department of Engineering Science, University of Oxford, Oxford, OX1 3PJ, UK
2 EDF Nuclear Generation Limited, Gloucester GL3 4AE, UK

Gilsocarbon graphite has been long used in the UK's advanced gas-cooled reactor (AGR) fleet. Despite the research and development of next generation high temperature gas cooled reactors and Gen IV designs which use fine-grained nuclear graphite, studying the crack initiation and fracture of coarser-grained Gilsocarbon graphite can be beneficial to the mechanistic understanding of the cracking of nuclear graphite components under complex thermo-mechanical stress state and knowledge transfer to the new builds. In this work, semi-circular discs with inclined sharp notches of Gilsocabron graphite (IM1-24 grade) have been tested under three-point bend at both room temperature and high temperature at 1000°C to study the mixed I/II mode fracture behaviour, at the Advanced Light Source of U.S. Lawrence

Berkeley National Laboratory. *In-situ* three-dimensional observation of the deformation and crack tip strain field quantification have been conducted by using micro-X-ray computed tomography in conjunction with digital volume correlation. The mixed I/II mode fracture toughness in terms of linear elastic stress intensity factor, K_I and K_{II} has been quantified and compared with modelling results. The mixed mode K_I/K_{II} dependence of the mode mixcidity has also been investigated by varying the angle of the inclined sharp notch at both temperatures.

10h20 **Effect of Neighbouring Bricks on Stress Relief in a Cracked Nuclear Graphite Bricks**

J. Zhang, G. Hall, and A.N. Jones
The Nuclear Graphite Research Group (NGRG), Department of Mechanical and Aerospace Engineering, The University of Manchester, Henry Royce Institute, Manchester M13 9PL, United Kingdom

In the graphite cores of AGR reactors, keyway root cracking relieves some of the bricks internally generated stress, and quantifying this relief is important for assessing the continued core integrity. However, a cracked brick is constrained by its neighbouring bricks and radial keys, which are expected to limit the crack opening and stress relief. The objective of this work is to investigate the extent of stress relief in an AGR graphite brick due to cracking.

Finite element analysis was performed using the University of Manchester material subroutine (ManUMAT). Six models were developed around the peak-irradiation layer; single-layer cracked, uncracked, and cracked-with-radial-keys configurations, together with equivalent three-layer arrays. For each analysis, the maximum in-plane principal stresses (MIPPs) at the keyways and the strain energy in the central brick were evaluated.

The results highlight two key findings. First, the effect of the radial key constraints can be a significant factor on the extent of stress relief since constraints from neighbouring bricks prevent the release of elastic deformations. Consequently, analyses based on isolated bricks overestimate the stress relief because they neglect constraint from the wider system. Second, the brick directly keyed to the cracked brick develops stress concentration at its keyway following cracking suggesting that keyway route cracking may promote the initiation of further cracking in neighbouring bricks.

10h40 *Coffee*

11h10 **MoFEM: The Incorporation of an Irradiated Material Model and Development Towards a Multi-Component Array**

P. Loughnane, C. McElvaney and S. Sarkar
Amentum

Mesh-oriented Finite Element Method (MoFEM) is an open-source finite-element library developed by the University of Glasgow. Aspects of MoFEM have been developed in collaboration with Amentum and EDF R&D for modelling crack propagation of Advanced Gas-cooled Reactors (AGRs) graphite bricks. To handle more complex and larger models, capable of addressing the interactions between bricks, their neighbourhood, and the surrounding core (see *R. Shaikh, et al.*), a semi-implicit solver has been developed.

In AGR cores, mechanical and thermal graphite properties continuously evolve due to fast neutron irradiation, temperature, and radiolytic oxidation. Previously, irradiation strains/stresses were calculated using Code_aster, incorporating an irradiated-material-properties subroutine developed by Amentum, and "snapshot" instances of the time-evolving strains/stresses "projected" into MoFEM to model instantaneous brittle fracture. However, to properly understand the contribution of contact stress evolution and fracture evolution with increasing core burn up and associated evolving and inhomogeneous irradiated material properties, the irradiated-material model must be incorporated directly into MoFEM. The "user material subroutine", originally developed by Amentum to apply the material property changes predicted by the EDF Integrated Methodology into Abaqus, and then Code_aster, finite-element models, has been incorporated into MoFEM. These updates will ultimately allow the development of array models in MoFEM which will consider the contributions of multi-component contact stress, irradiation stress and creep, and fracture propagation.

This paper presents the development of MoFEM to achieve the goal of a large array, starting with the implementation of the irradiated-material model.

11h30 **MoFEM Support to Brick Segment Bowing and associated Part-Height Crack Propagation including Validation against Rig Tests**

P. Loughnane, S. Primavesi and E. Largie-Poleon
Amentum

Recent inspections of Advanced Gas-cooled Reactor (AGR) cores have identified radial outward bowing of graphite brick segments formed between axial cracks in fuel bricks, referred to as Segment Bowing. Segment Bowing occurs when the shape of the fuel brick changes from "cylindrical" to "wheatsheaf" due to a spatially-radially varying dimensional change (shrinkage and thermal strains). When fuel brick "wheat-sheafing" is coupled with multiple full or part-height axial cracking, segmentation of the fuel brick can become an issue, substantially increasing the "wheat-sheafing" and outward radial displacement effect due to loss of hoop stiffness. Therefore, lateral contact of the bowing segments with neighbouring bricks and its effect on the segment's partial height cracks must be considered.

In support of analysing Segment Bowing and associated partial height crack propagation, a Mesh-Oriented Finite Element Method (MoFEM) assessment has investigated the nature of partial height crack growth, as a mechanism to reduce segment stiffness, in preference to an excessive increase in lateral displacement and brick-to-brick contact load.

This paper presents the modelling of crack propagation, for a constrained bowing segment, from a partial height keyway root crack to a longer crack, and the associated brick-brick contact load, in MoFEM. To improve confidence in these results, a test rig experiment has been developed, with design informed by initial MoFEM modelling. This test rig involved externally loading a modified graphite fuel brick to reproduce the segment bowing and partial height crack propagation behaviour.

11h50 **Development and Validation of a Vectorised User Material Subroutine (ManVUMAT) for Explicit Finite Element Analysis**

V. Zolotarevskiy, G.N. Hall, and A.N. Jones

Nuclear Graphite Research Group, Department of Mechanical and Aerospace Engineering, Henry Royce building, the University of Manchester, UK

The later life of the Advanced Gas-cooled Reactors (AGRs) is associated with complex cracking patterns and distortion of the graphite components, the understanding and prediction of which are crucial for safe operation.

This work presents a novel in-house predictive modelling capability to analyse transient effects on AGR cores and its components, e.g. prompt cracking and seismic events. This is done by means of a Vectorised User MATerial subroutine (ManVUMAT) used in the commercial finite element software Abaqus/Explicit. The subroutine implements empirical constitutive laws describing the evolution of nuclear graphite material behaviour. The explicit method offers several advantages over the implicit (static) method: it avoids convergence issues, it is robust in contact-dominated problems, it supports damage and fracture, and both dynamic and quasi-static problems can be simulated within a single framework.

Quasi-static ManVUMAT simulations have been verified against our well-established Abaqus-Standard (implicit) ManUMAT subroutine and nearly identical results in terms, of displacements and strains for graphite cuboids as well as irradiation creep experiment samples, were obtained. The validation of ManVUMAT has been extended to the simulation of a Torness reactor brick, showing excellent agreement of maximum in-plane principal stresses and radial displacements at-shutdown compared to the results of the implicit method.

12h10 **Modelling Material Behaviour at the Notch Tip Region of a Keyway Corner Specimen Test with ABAQUS CDP**

Jonathan Wright

EDF Energy Generation, Building 1420, Charlton Court, Gloucester Business Park, GL3 4AE

Developing the capability to model and understand crack initiation and propagation in graphite component features is important for refinement of structural integrity assessments; graphite specimen tests are a potential source of information which can enable development of knowledge about cracking.

In an un-irradiated Gilsocarbon graphite specimen test, the 'keyway corner test', a characteristic non-linear load vs displacement response is measured near to peak loading. The quasi-brittle behaviour of graphite influences the formation of localised variations in stresses and strains and micro-cracking in the vicinity of the specimen notch corner and the formation of a small initial crack, prior to rapid crack propagation through the bulk of the specimen.

A study has been done to understand whether a finite element model can reproduce the load displacement response near peak load and the formation of initial cracking of the specimen test. The study uses the Abaqus Explicit model with the CDP (Concrete Damage Plasticity) material model with set parameters aiming to simulate the effects of graphite micro-cracking. Several sensitivities are presented and the model outcomes are reviewed. The paper proposes a way forward to improve refinement of the modelling and understanding.

12h30 *Lunch*

Structural Integrity (4) – Segment Bowing and Strain

Chair: Mr Mark Davies, Marad

14h10 **Balancing Complexity and Conservatisms in AGRIGID Modelling of Brick Segment Bowing**

C. Minors, R. Jennings, E. Washbrook and J. Melia
Amentum

AGRIGID is a whole-core stick-and-spring modelling tool used to model graphite component loading and channel distortions owing to graphite dimensional change from ageing and irradiation. Its results underpin safety case assessments supporting continued operation of the UK AGR fleet. Recent inspections at Heysham 2 and Torness power stations identified radially-outwards bowing of graphite brick segments starting at the ends of some fuel bricks. This phenomenon, termed segment bowing, is not represented in previous AGRIGID methodologies. Simplified approximations and slight adjustments to a small number of springs in AGRIGID to represent some of the largest effects due to segment bowing were initially considered but ultimately rejected. This work shows how the use of approximate changes would fail to capture the complexities of the potential consequences of segment bowing by showing the limitations of that approach. It also shows how the developments that have been made will allow better representation without compromising on conservatism. Additionally, confidence in these methods is supported by the creation of a new visualisation tool imAGR which allows for easy-to-understand presentation of results from AGRIGID.

14h30 **AGRIGID Modelling of Observed and Postulated Segment Bowing**

D. Broadbent, N. Solanki, J. Trotter, L. Clarke, J. Melia and C. Minors
Amentum

Assessing the safety of an Advanced Gas-cooled Reactor (AGR) graphite core relies on accurate modelling of graphite brick distortions and the resulting changes in core geometry. This work shows how the enhanced AGRIGID model represents the effects of segment bowing on brick loading and channel distortion, with results validated against observations from fuel and control rod channels inspections. The results demonstrate that segment bowing has a measurable influence on channel distortions but remains bounded by current safety limits. This work demonstrates the extended capability of AGRIGID discussed by Minors et al. In particular the improved confidence in graphite core safety assessments for continued reactor operation by considering the observed, predicted and postulated continuation of brick segment bowing.

14h50 **Modelling Induced Graphite Brick Cracking in GCORE Seismic Analysis**

J. Roff and R. Dickenson
AtkinsRéalis

Inspections of graphite cores within the UK Advanced Gas-cooled Reactor (AGR) fleet have identified fuel bricks containing one or more full-height cracks. Such defects present challenges to the seismic safety case, particularly regarding control rod channel distortion. Cracked bricks show greater irradiation-induced deformation, such as crack opening, and the presence of multiple cracks can degrade the effectiveness of the keying system responsible for maintaining core geometry.

GCORE is an Abaqus/Explicit finite element model developed for efficient seismic analysis of AGR cores, with particular focus on control rod channel distortion. Graphite core components are represented as rigid bodies, with component interactions governed by bespoke user-defined elements (VUELs) implemented in Fortran. The current architecture allows full-height cracks to be induced by overloads occurring during the analysis, whether from crack opening or seismic excitation.

This paper describes the implementation of induced cracking within GCORE. Example results are presented to illustrate the influence of cracking on core response during seismic events.

15h10 **Modelling Graphite Brick Segment Bowing in GCORE Seismic Analysis**

R. Dickenson and J. Roff
AtkinsRéalis

Inspections of the Advanced Gas-cooled Reactors (AGRs) at Heysham 2 and Torness have found bowed segments in cracked fuel bricks. There is a risk that this deformation could deform control rod channels, presenting a challenge to the seismic safety case.

GCORE is an Abaqus/Explicit finite element model developed for efficient seismic analysis of AGR cores, with particular focus on control rod channel distortion. Graphite core components are represented as rigid bodies, with component interactions governed by bespoke user-defined elements (VUELs) implemented in Fortran. Historically, GCORE has relied on the prismatic geometry of the graphite components to deliver an efficient parametric idealisation. Recent developments have introduced partial-height cracks and curved surface contact algorithms to provide an efficient idealisation of the bowed geometry seen in inspections.

This paper describes the approach and challenges to the implementation of partial height cracking and segment bowing within GCORE.

15h30 **High Strain-Rate Testing of Nuclear Graphite**

Sophia Carpio[1], Aaron Graham[1], Ming Jiang[1], Muhammad Fahad[2] and Dong Liu[1]
[1]Department of Engineering Science, University of Oxford, Oxford, OX1 3PJ, UK
[2] EDF Nuclear Generation Limited, Gloucester, GL3 4AE, UK

Graphite core components in civil nuclear reactors are subjected to dynamic loading events, including stress wave propagation following keyway root crack (KWRC) formation, resulting in secondary damage to graphite fuel bricks which may impair safety critical functions. Understanding dynamic mechanical behaviour of nuclear graphite is important for optimising predictive models on crack development and supporting material selection and design of advanced fission reactor systems. However, limited data exists on the influence of strain rate on the mechanical behaviour of nuclear graphite. This study investigates the dynamic compressive strength of Gilsocarbon and fine-grained graphites SNG623, SNG742 and T220 using Split Hopkinson Pressure Bar (SHPB) testing combined with high-speed imaging and 2D Digital Image Correlation (2D-DIC). Strain rates between 176 – 662 s^{-1} were achieved, and compressive strength and failure behaviour were characterised from the dynamic stress-strain response and high-speed imaging. Compressive strength was found to increase with strain rate, while the magnitude of the strain rate effect was understood to correlate with differences in bulk density and grain size. Displacement and strain fields obtained by DIC were used to monitor the dynamic loading process and validate calculated specimen displacements and strains. High-speed imaging revealed fragmentation of graphite following axial compression and splitting.

15h50 *SESSION AND CONFERENCE SUMMARY: Prof Dong Liu and Dr Jim Reed*

16h10 *Safe travel home and thanks for your participation!*

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As an editorial suggestion, where more than one presentation has been given by an organisation, combining some or all into a single paper may be beneficial in putting the work into context and simplifying the writing.

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