

INTERNATIONAL NUCLEAR GRAPHITE SPECIALISTS' MEETING INGSM-25

28th September – 3rd October 2025
at

IAEA Building M

Kaisermühlen, Vienna, Austria

PROGRAMME AND ABSTRACTS

FINAL EDITION CHECK ON THE DAY FOR ANY AMENDMENTS



Organised in
Co-operation with:

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IAEA

International Atomic Energy Agency

Atoms for Peace and Development

Organising Committee

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Mr F. Reitsma, IAEA

Manager:

Prof A.J. Wickham, Nuclear Technology Consultancy, Laugharne, UK
and The University of Manchester Nuclear Graphite Research Group, UK

Local Coordination:

Dr Alina Constantin, IAEA

Finance and Website:

Mr Roy Bellingan, Ireland

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Dr Nidia Gallego, ORNL, USA

Dr Richard Harris, Nuclear Restoration Services, UK

Dr Miriam Knebel, JEN, Jülich, Germany

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PREFACE

On behalf of The International Atomic Energy Agency, we are delighted to welcome everyone to Vienna once again on the occasion of the 25th International Nuclear Graphite Specialists' Meeting. These meetings began in September 2000 at the suggestion of members of the technical steering committee of what was then the IAEA International Database of Nuclear Graphite Properties. Having visited the USA, Japan, Republic of Korea, China, Germany, France, South Africa, The Netherlands and The United Kingdom (in some cases, several times), the meeting comes once again to its 'spiritual home' here in Vienna.

As this is the 'silver' anniversary of the birth of INGSM, it is useful to review the previous occasions and locations where the meetings have taken place:

2000	Oak Ridge National Laboratory, Tennessee, USA (sponsored by USDoE)	INGSM-1
2001	Lech-am-Rain, Germany (sponsored by SGL Carbon)	INGSM-2
2002	Parma, Ohio, USA (sponsored by Graftech)	INGSM-3
2003	Marugame, Japan (sponsored by Toyo Tanso)	INGSM-4
2004	Plas Tan-Y-Bwlch, UK (sponsored by British Carbon Group, RSC, IOP, SCI)	INGSM-5
2005	Chamonix, France (sponsored by SGL Carbon)	INGSM-6
2006	Oak Ridge National Laboratory (sponsored by USDoE)	INGSM-7
2007	Bakubung, South Africa (sponsored by PBMR Co)	INGSM-8
2008	Egmond-aan-See, The Netherlands (sponsored by NRG Petten)	INGSM-9
2009	West Yellowstone, USA (sponsored by USDoE)	INGSM-10
2010	Eastbourne, UK (sponsored by British Carbon Group, RSC, IOP, SCI)	INGSM-11

2011	Jeju Island, Republic of Korea (sponsored by KAERI)	INGSM-12
2012	Meitingen, Germany (sponsored by SGL Carbon)	INGSM-13
2013	Seattle, Washington State, USA (sponsored by USDoE)	INGSM-14
2014	Hangzhou, China (sponsored by Sinosteel)	INGSM-15
2015	Nottingham, UK (sponsored by British Carbon Group, RSC, IOP, SCI)	INGSM-16
2016	Vienna, Austria (sponsored and hosted by IAEA)	INGSM-17
2017	Baltimore, USA (sponsored by USDoE)	INGSM-18
2018	Shanghai, China (sponsored by SINAP and others)	INGSM-19
2019	Bruges, Belgium (sponsored by SCK•CEN)	INGSM-20
2020	<i>... was planned for Chicago, USA, but Covid intervened</i>	<i>No meeting</i>
2021	Small on-line event hosted from SINAP, Shanghai, China	INGSM-21
2022	On-line meeting hosted by North Carolina State University, USA	INGSM-22
2023	Aachen, Germany (sponsored by FHA and BGZ)	INGSM-23
2024	Berkeley CA, USA (hosted by The University of California [Berkeley])	INGSM-24

An innovation this year is the decision to go for a full week. This is principally to accommodate a specific request from the United Kingdom's Nuclear Restoration Services (NRS), a relatively new organisation within the orbit of the UK Nuclear Decommissioning Authority but derived largely from the former Magnox companies which operated those power reactors in the UK and now have responsibility for the sites. Our one-and-a-half days on the topic of managing irradiated graphite wastes now forms the third part of a series of NRS events on the subject leading eventually to a proposal for a more formal approach to dismantling existing reactors more expediently than originally envisaged. In the words of Dr Richard Harris, who joined the organising committee to assist with this part of the programme, these sessions *"...have been organised ... in support of progressing development of technologies for retrieval and treatment of irradiated graphite to allow timely, affordable and sustainable decommissioning of sites in preparation for their next intended use."*

It is interesting that around 10 additional international delegates will join us for those waste-management sessions commencing after lunch on Thursday, reflecting the growing importance of not only dealing with the legacy waste but also of planning for appropriate disposal as part of the licensing process for new-built plants.

Many of you visit Vienna regularly for various IAEA activities in the nuclear graphite or waste-management fields: others will be making their first visit. Vienna is a small capital city (the entire country has fewer inhabitants than London, UK!) and you will find everyone to be welcoming and courteous – sometimes, even, perhaps a little formal. Many people here speak English and amongst young people of secondary school age it is almost universal.

Make sure you stay long enough to enjoy the history. The magnificent Stephansdom cathedral, the Hofburg Palace, The Rathaus (city hall), Opera and other landmarks are situated within or on the 'Ringstrasse' which surrounds three sides of the old city ('Innenstadt') with the Donau Kanal marking the northern edge at Schwedenplatz. We shall be visiting the famous Schönbrunn Palace together on Wednesday afternoon to enjoy the gardens, a guided tour of the Castle, dinner and a concert of typical Viennese music. A walk/tram/U-Bahn tour taking in the Prater Park (the famous big wheel of 'Third Man' fame), Hundertwasser Haus (and also his famous Fernwarmwien power-station design at Spittelau!), The Belvedere, and Naschmarkt is highly recommended. You can also visit the sewers (yes, 'Third Man' theme again) – guided tours are available on Friday evenings starting from a manhole adjacent to the Stadtpark.

We hope that you enjoy your time with us here in Vienna!



The famous 'Hundertwasserhaus' apartment building in Lowengasse: the arcade opposite, built in his style is well worth a visit too (the basement toilets are a particular attraction!). The nearby Kunst Haus Wien museum in Untere Weißgerberstraße has an exhibition of the Hundertwasser village and numerous other attractions.

Not to forget his power station too, at U4/U6 Spittelau!



The Hofburg Palace houses the Sisi Museum and is also home to the performances of the famous Lipizzaner 'dancing' horses which offer daily shows



Naschmarkt – between Karlsplatz and Ketterbrückengasse, U4



Traditional 'fiaker' outside Stephansdom. Fiakers offer unique (but expensive) tours of the Innenstadt

Notes on Public Transport in Vienna



Public transport in Vienna works on an honesty system, but the penalties for ticketless travel (if detected) are extremely costly. *Except for journeys between the Airport and the city boundary at Schwechat*, one inner-zone ‘weekly’ ticket will serve for all U-Bahn, S-Bahn, buses and trams. These are purchased from machines at each U-Bahn station or can be purchased at the airport ‘VOR’ office (if open). There is also an office at Praterstern station. They must be validated before first use (insert in the ‘Entwerter’ and make sure it is stamped). After that, just carry it with you and hop on to any service you wish. No need to show it to bus drivers or anything like that... just the (very) occasional ticket inspection during the journey.

The journey to and from the airport requires an additional ticket. There is a ‘City Airport Train’ which is outrageously expensive and goes only to one destination – Wien Mitte – which is OK if that is where you want to go! We recommend the S-Bahn S7 (direction ‘Floridsdorf’, approx. every 30 mins, with some regional ÖBB trains also available) which also serves Wien Mitte and Praterstern (change for IAEA to U1) – when you arrive it just easier to purchase the inexpensive ÖBB ticket from their office or a machine in the airport to central Vienna (any inner-city station is the same cost) and then organise your weekly ticket afterwards.

When going back to the airport, ask the ÖBB machine at the S-Bahn station for an inexpensive add-on ticket valid from Schwechat – all the machines ‘speak’ English – and carry it alongside your weekly ticket. Ticket inspections on S-Bahn are more frequent.

Note that U2 (purple line) is undergoing changes at the city end as part of the creation of U5... see information at stations.

INTERNATIONAL NUCLEAR GRAPHITE SPECIALISTS MEETING INGSM-25

A Conference at IAEA (Press Room, Building M)
Vienna, Austria

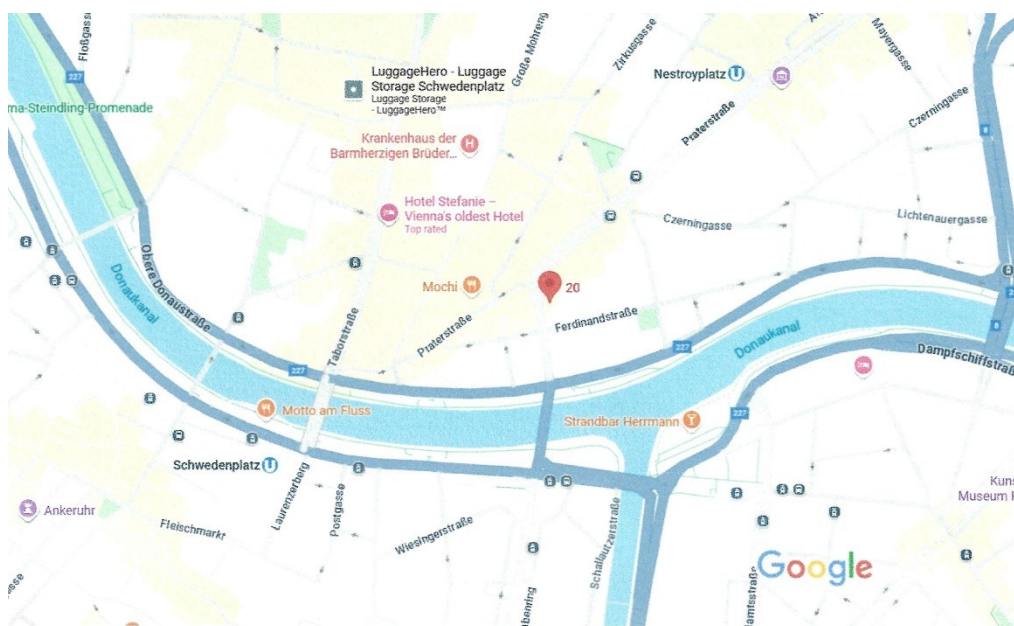
28th September – 3rd October 2025

*Organised in cooperation with:
The International Atomic Energy Agency
Nuclear Restoration Services, United Kingdom*

Sunday 28th September 2025

17h30 until 20h30 (just drop in!):

**WELCOME RECEPTION
“Family and Friends” Restaurant, Praterstraße 20**



The nearest U-Bahn stations are Nestroyplatz (U1, bus 5A) and Schwedenplatz (U1/U4, trams 1,2)



All delegates and accompanying persons are invited to attend the WELCOME RECEPTION at 'Family and Friends' on SUNDAY 28th SEPTEMBER and presenters are also requested to **bring their presentations on a USB drive** for uploading to the presentation computer. This is especially important if your presentation is on Monday morning!

PRESENTATIONS & AUDIO/VIDEO FACILITIES

Presentations will be 20 minutes inclusive of discussion with certain exceptions previously agreed. Aim for 15 minutes of presentation and 5 minutes for discussion. PLEASE KEEP WITHIN YOUR ALLOTTED TIME.

Presenters are encouraged to bring their presentations on a USB memory stick and to upload them ahead of the session in the Conference Room if you were not ready to do so on the Sunday evening. PowerPoint is requested format, not pdf. 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.

ALL presentations will be made available to delegates via the conference website for a limited period, and will also be made available to the INGSM archive on the IAEA Knowledge Base on Nuclear Graphite where they can be accessed by suitably authorised persons via NUCLEUS.

POSTERS

Six poster presentations will be on view throughout the conference in the Press Room. More details (abstracts) are provided below in the sessions to which they relate. Please feel free to contact their authors at any time during the conference.

CONFERENCE VISIT, DINNER AND CONCERT (Wednesday October 1st)

This year we celebrate the 200th anniversary of the birth of Johann Strauss II. We felt that it was appropriate to repeat an event we did in 2016 (INGSM-17) when we last came to Vienna for the meeting.

This will take place on **WEDNESDAY EVENING**.

We shall enjoy private tours (four groups) of Schönbrunn Palace, with time to enjoy the gardens at leisure, a dinner at Restaurant Gerstner, and then a concert of Viennese music in The Orangery with (no doubt) audience participation in the Radetzky March! Dress code is smart casual.



Please make your way via U4 subway line to **Schönbrunn**. The approximate journey time from IAEA is 35 minutes: take U1 and change to U4 at Schwedenplatz (direction 'Hütteldorf'). Arriving at Schönbrunn station, cross the bridge over the tracks before leaving the station. Turning to your right

you will see a large coach park and just beyond it a new **VISITOR ARRIVAL CENTRE**. There is an obvious walkway alongside the coach park. We will assemble outside the Centre **no later than 16h45** before moving off to meet our guides. ***Please do not be late*** because the visit schedule is determined by the timing of tours, dinner and the concert – we cannot await latecomers!



Schönbrunn Group Arrival Center (Meeting Point) looking towards the coach park. The U4 station is in the distance far left



CONFERENCE TIMETABLE AND ABSTRACTS

Monday 29th September 2025

Opening Session

Chairman: Prof A.J. Wickham

09h00

Opening Addresses:

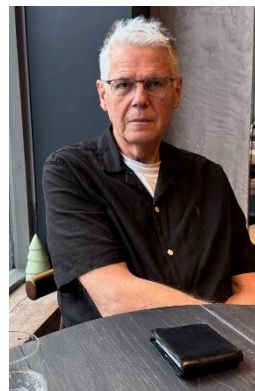
Mr. Frederik Reitsma

International Atomic Energy Agency
Conference Chairman

09h15

In Memoriam: Dr Martin Metcalfe

Tribute led by Ms Nassia Tzelepi,
UK National Nuclear Laboratory



KNOWLEDGE TRANSFER: LEGACY REACTORS to NEW BUILD

Chairman: Prof A.J. Wickham

09h20

INVITED LECTURE

Lessons from the 20th-Century Graphite Reactor Designs for Those of the 21st Century

M W Davies (to be presented on Mike's behalf by Dr Stephen Davis)
Amentum, UK

Many reactors with graphite cores have been designed, constructed and operated over the past ~80 years. The first large scale graphite cores were built at ORNL and Hanford, in the US. The UK followed these with the two Windscale piles, the four reactors at both Calder Hall and Chapelcross, eleven civil Magnox stations, the Windscale prototype Advanced Gas-cooled Reactor (AGR), followed by seven twin AGR stations. Many others graphite cores have been built around the world, for example in mainland Europe, Japan and Russia. Several High Temperature Reactors (HTRs) have also been built, in the UK, US, Germany, Japan and China.

A number of challenges were faced by the core designers along the way, such as the fast neutron induced dimensional change behaviour of graphite, the large difference in thermal expansion rates between graphite and steel, and how to interconnect components both vertically and horizontally to provide column continuity and core stability. These were largely overcome by their ingenuity, but with varying degrees of success. As such, many lessons have been learnt and the core designers of future HTRs need to be aware of them. The presentation covers this journey in terms of graphite data requirements, core design, construction and operational/monitoring feedback.

09h50

What is Sufficient Information on Graphite Behaviour to Support Operation of a Nuclear Reactor?

Jim Reed

EDF Energy Generation

Graphite has been used as moderator in many reactors over the last 70 years, and it is intended to be used in the next generation of high temperature Advanced Modular Reactors that will operate 2030+. As this next generation of reactors are at the final design stage – is it clear what information is required (*i.e.* essential) and what information is just too difficult to gather? Can the AGR experience help to set this context?

Advanced Gas-cooled Reactor (AGR) cores are composed of thousands of graphite bricks. After many years of service, non-uniform irradiation, temperature and weight loss fields gradually modify both the microstructure and internal stresses of the bricks. These modifications may lead to crack initiation and propagation. There are plenty of examples of the cracking in AGR fuel bricks.

10h10

HTR applications of AGR Graphite Testing Methods

M. Kirkham, M.W. Davies, S. Primavesi and S. Sarkar

Amentum UK

(To be presented by Jonathan Melia, Amentum)

The graphite core structures within the UKs Advance Gas-cooled Reactors (AGRs) are the life limiting parts of the reactors. The individual graphite components in the core structures are subjected to continuous degradation due to fast neutron irradiation and radiolytic oxidation. This results in components distorting and potentially leading to component cracking, both of which contribute to whole core distortion. Distortions may eventually reach a level that could impede control rod entry, fuel cooling, and refuelling operations.

Through-life assessments of individual AGR core components, and the core structure as a whole, are carried out using finite element modelling, and a number of bespoke computer models specifically developed for the AGRs. These require inputs from test rigs, such as failure loads, which are used to determine failure stresses. Test rigs, both full-scale and small-scale, are used to validate the computer models. Future High temperature Reactors (HTRs) will also require various test rigs to substantiate their core designs.

This presentation discusses two examples of modelling/testing/assessment programmes used for AGRs which can be readily adapted for HTR applications: component feature testing and modelling for determining failure loads and critical stresses for component integrity assessments.

10h30

Extending MOFEM Graphite Component Crack Modelling from AGRs to HTGRS

E. Largie-Poleon and P. Loughnane

Amentum UK

Advanced Gas-cooled Reactors (AGRs) and High Temperature Gas-cooled Reactors (HTGRs) contain graphite components exposed to significant non-uniform irradiation, thermal loads, and other external loads. The quasi-brittle nature of graphite coupled with evolving stresses in the graphite components can lead to cracking. For AGRs, where cracking is widespread, understanding and predicting crack behaviour including crack propagation, crack arrest, and crack/fragment morphology is key to maintaining their safe operation. HTGRs are not expected to have widespread cracking, however, it is still important to understand the extent of cracking when, or if, it should occur.

MoFEM (an energy based finite element solver developed by the University of Glasgow), has been used to model crack propagation, predicting crack morphology and stability, in the AGR graphite fuel bricks. A new solver has been developed which has significantly reduced solve times allowing for a greater number of analyses to be conducted, and improved scalability, allowing for larger more complex models to be investigated.

A comparison of the new and old MoFEM solvers will be presented in relation to the AGR programme. Based on learnings from the AGR MoFEM models, a new crack propagation model has been developed for a typical HTGR fuel block.

10h50

Coffee break

11h20

Geometric Aspects of Reactor Core Assessment

S. Sarkar R. Crawford and M. W. Davies

Amentum UK

Stress analysis is central to assessing the through-life structural integrity of nuclear graphite components *i.e.* the likelihood of crack initiation at any given time. However, for Advanced Gas-cooled Reactors (AGRs) and High Temperature Gas-cooled reactors (HTGRs), stress analysis cannot be the sole focus of assessment of a reactor core's ability to meet its fundamental nuclear safety requirements: shut-down, hold-down, and fuel cooling.

These functions can be compromised by fast neutron irradiation-induced dimensional changes in graphite components, and hence the structure as a whole, which can adversely affect channel shapes. Therefore, the UK approach, developed for the AGRs, incorporates geometric aspects.

This presentation will cover the geometric aspects of assessments of both AGRs and prismatic block type HTGRs. In both cases, whole-core modelling is informed by more detailed modelling of dimensional changes of components and the displacements from whole-core modelling are analysed for their potential to reduce the ability of the core to meet its fundamental nuclear safety requirements. While the overall approach is consistent between AGRs and HTGRs, this presentation will also discuss the differences in the assessments, due to differences in design and operational requirements.

11h40

Core Inspection Data Management and Model Validation in AGRs

J. Taylor¹, R. Gray¹, G. Horne¹, M Joyce¹, J Reed²

¹ Frazer-Nash Consultancy Ltd

² EDF Energy Generation

EDF's Advanced Gas-cooled Reactor inspection and monitoring provides a rich dataset, including dimensional measurements, internal video footage, eddy current measurements, thermocouple measurements and graphite samples from selected channels. These data play a vital role in understanding core behaviour, substantiating structural integrity calculations and validating the models that underpin safety cases for the ageing graphite cores. It is reassuring to see procedures such as ASME BPVC Section XI highlighting the importance of inspection and monitoring for evidence-based integrity assessment of future plant.

This presentation summarises the management of this inspection data from receipt through to structured validation. Our approach ensures traceability, consistency, and suitability of the data for downstream use. We describe how validated datasets are used to inform updates to graphite behaviour models, ensuring that predictions reflect the observed condition of the core and maintain alignment with safety assessment expectations. Drawing on years of experience, we highlight key lessons and insights from supporting inspection and modelling activities across multiple reactor sites. This work strengthens the feedback loop between inspection,

modelling, and operational assessment, supporting informed decisions on monitoring strategies, safety case updates, and reactor operation within regulatory expectations.

12h00

ENLIGHT - Enabling a Lifecycle Approach to Graphite for AMR

A.N Jones¹, C. Sharrad¹, G. Hall¹, M. Mallon¹, A. Theodosiou¹, X. Zhang¹, P. Mummery¹, K. Jolley², K. Jones³, D. Liu⁴, and T. J Marrow⁵

¹Nuclear Graphite Research Group, Department of Mechanical and Aerospace Engineering, School of Engineering, Henry Royce Institute, The University of Manchester, M13 9LP, UK

²Department of Chemistry, Loughborough University, Leicestershire, LE11 3TU, UK

³School of Geography, Earth and Environmental Sciences, Faculty of Science and Engineering, University of Plymouth, PL4 8AA, UK

⁴Department of Engineering Science, University of Oxford, Parks Road, Oxford, OX1 3PJ, UK

⁵Department of Materials, University of Oxford, Parks Road, Oxford OX1 3PH, UK

The UK aims to deliver 24 GW of new nuclear power by 2050, including a large proportion from new Advanced Modular Reactors, with UK commitments to develop High Temperature Gas-cooled Reactor (HTGR) technology and worldwide interests in Molten Salt Reactors. The UK's historical leadership in graphite reactor technology, exemplified by the Advanced Gas-cooled Reactor fleet, is now at risk without renewed investment and the move from AGR to AMR. Our new 5-year Programme grant; ENLIGHT offers a pathway for the UK to repurpose and upskill its unique position as a global knowledge and innovation centre for graphite.

The overall aim of ENLIGHT is to design and produce sustainable graphite materials for AMR deployment and transform irradiated graphite from a waste stream into a valuable resource. ENLIGHT addresses these challenges through three integrated research programmes: (1) *Sustainable Graphite*—developing decontamination and recycling pathways; (2) *Graphite Selection & Design*—engineering materials for AMR-specific degradation modes; and (3) *Graphite Performance*—evaluating behaviour under AMR-relevant irradiation and coolant environments.

Our programme of research, collaboration, and skills development will expand and advance the UK graphite research community, nurturing the next generation of graphite engineers and scientists, essential to AMR technology advancement. We aim to provide a critical framework for data generation, regulatory engagement, and pathways to material qualification essential to future advanced reactor systems.

Support to Operating Reactors

Chair: Prof. Liu Dong, University of Oxford, UK

12h20

Finite Element Modelling of Crack Propagation in AGR graphite bricks

X. 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, UK

Gilsocarbon graphite has been used as a structural material and neutron moderator in the UK AGRs and it is well documented that during reactor operation, all properties of the graphite evolve. These changes can potentially lead to cracking in the graphite bricks, and progression can degrade the structural integrity of the reactor core and potentially limit the reactors' lifetime. In this research, finite element modelling (FEM) has been used to simulate the propagation of cracks that have been observed in the reactor inspections. Specifically, the propagation of keyway root crack (KWRC) was first simulated using cohesive zone model for a single-layer AGR graphite brick. The influence of KWRC opening on the development of KWRC in the neighbouring graphite brick was investigated using a two-layer model. The results showed that the influence of KWRC opening on the crack development in a neighbouring brick is dependent on the fluence, temperature, and weight loss. For the reactors' peak rated layers, the internal stresses in the brick build up fast enough such that cracking occurs prior to any interactions due to KWRC opening. Conversely, when loads on the brick

are relatively low, the opening KWRC will be constrained by the neighbouring bricks, leading to stress building up which can promote the earlier crack development in the neighbouring bricks. This research also investigated the propagation of a secondary KWRC and found that following the onset of the main KWRC, the stress in the brick was relaxed partially. However, with further irradiation, the stress can build up. Moreover, graphite naturally contains defects which will grow as a result of radiolytically induced weight loss. The onset of a secondary KWRC is found to be sensitive to the size of the defects at the keyways.

12h40 **Predicting Keyway Root Cracking Locations through the integration of Finite Element and Statistical Modelling**

A. Farrokhnia, G.N. 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, UK

Graphite components play a crucial structural role and act as moderators in nuclear reactors. In the UK, these components experience degradation from fast neutron irradiation, temperature, and radiolytic oxidation, which generate stresses and changes in material strength that can lead to the initiation of cracks. Traditional finite element analyses have relied on stress distributions to predict cracking; however, such deterministic approaches typically assume homogeneous material strength and fail to capture the probabilistic nature of defect-driven failure in graphite.

This study presents a new methodology developed to support the nuclear safety regulation for the continued operation of Advanced Gas-cooled Reactors (AGRs), integrating finite element analysis with a statistical model of strength variability to quantify the probability of crack initiation and progression within multicomponent graphite cores. The method accounts for operation-induced changes in stress and material properties, and models both pre- and post-cracking states. The framework employs Monte Carlo simulations to represent the stochastic nature of material strength, enabling the identification of both expected and rare cracking scenarios. This provides important insight into cracking sequences and support reactor life-extension decisions. This is especially important within AGRs, where cracking in one brick can redistribute stresses and influence cracking in adjacent bricks. By moving beyond peak-stress assumptions, this novel approach enables comprehensive predictions of cracking behaviour.

13h00 *Lunch. Cafeteria, ground floor F Building (cards or cash); snack bars M building ground floor and C building 4th floor – cards only*

13h50 **Examining the Dependence of Coefficient of Thermal Expansion on AGR Weight Loss using Finite Element Analysis**

V. Zolotarevskiy¹, E. Tan², N. Warren², G.N. Hall¹ and A.N Jones¹

¹ Nuclear Graphite Research Group, Department of Mechanical and Aerospace Engineering, University of Manchester, UK

² Health and Safety Executive Science and Research Centre, Harpur Hill, Buxton, UK

It is well documented that AGR Graphite properties change over time due to changes in temperature, fast neutron irradiation, and ionising radiation combined with carbon dioxide coolant that leads to graphite weight loss. Consequences of these are evidenced in formation of local stress concentrations and distortion of the bricks. Accurate prediction of graphite behaviour, therefore, is vital for ensuring the continued safe operation of the AGRs.

Coefficient of thermal expansion (CTE) has a significant effect on stresses that develop in graphite. The present study focuses on a CTE model that incorporates statistical methods to fit the available material test reactor sample measurements as well as trepanned data. The independent variables in this model feature irradiation temperature and dose, while dependence on weight loss was not originally considered. However, examination of recent

trepanned data shows a modest but statistically significant association between CTE and weight loss.

This work examines the dependence of CTE on weight loss using the finite element (FE) analysis. The empirical CTE model was revised by introducing weight loss dependence. Simulation results are presented in terms of maximum in-plane principal stresses calculated through the life of an AGR brick. This work is concluded with a discussion and recommendations for the choice of CTE model for future FE analyses.

14h10

Hybrid Approaches for Modelling Graphite Oxidation within the UK AGR Fleet

J. Taylor¹, R. Lincoln¹, D. Kent¹, A Young¹, M Joyce¹, M Bradford², M Griggs²

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 mechanistic FEAT-DIFFUSE base to ensure interpretability and adequately conservative projections in data-sparse regions, while empirical scaling improves 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.

Oxidation

Chair: Dr. Nidia Gallego, ORNL, USA

14h30

Evolution of Our Understanding of Thermal Oxidation in Nuclear Graphite

W. Windes¹, R. Smith² and C. Contescu³

¹ Idaho National Laboratory

² Idaho National Laboratory (retired)

³ Oak Ridge National Laboratory (retired)

For the past 20+ years the USA has conducted extensive research in thermal oxidation behaviour of nuclear graphite. As is well known, oxidizing graphite is exceedingly simple but quantifying the oxidation process is extremely difficult. To achieve any kind of control over graphite oxidation the extrinsic parameters (oxidizing environment) must be controlled to an exacting level while the intrinsic parameters (the unique microstructural attributes) yield different results for each grade. Significant strides have been made in both precision control of the extrinsic oxidation parameters and understanding the material response from intrinsic differences which has led to a progression in understanding thermal oxidation behaviour in graphite.

This paper will highlight the accomplishments achieved over the past 20+ years and how they have affected practical engineering applications, applied science, and fundamental research in graphite oxidation. Areas of practical interest on the development of new testing standards to provide precision control for intrinsic oxidation response, steam oxidation behaviour, graphite burning and dust explosions, irradiation effects on oxidation rate, practical considerations of Kinetic- versus Diffusion-controlled oxidation, sample size effects, and others will be highlighted.

14h50

Characterisation of Nuclear Graphite Oxidation Behaviour for Advanced High-Temperature Reactors

S. Zaila, A. Theodosiou and A.N. Jones

Nuclear Graphite Research Group, Department of Mechanical and Aerospace Engineering, School of Engineering, Henry Royce Institute University of Manchester, M13 9LP, UK

Nuclear graphite is a promising material for use as a moderator, reflector, and structural components in the next generation nuclear systems, particularly High Temperature Gas-Cooled Reactors (HTGRs) and the Molten Salt Reactors (MSRs). Both HTGRs and MSRs will operate at significantly higher temperatures than current systems, therefore, understanding the behaviour of new nuclear graphite grades, especially in potentially oxidising environments is crucial for predicting material lifespan and informing reactor design.

This work provides important insights into the characterisation and comparative oxidation behaviour of the selected nuclear graphite grades, with a particular focus on understanding how the microstructure influences oxidation behaviour. To achieve this, we have used multi-instrument approach including Polarised Optical Microscopy, X-ray Computed Tomography, Thermogravimetric Analysis and Electron Microscopy to thoroughly characterise the oxidation behaviour of the nuclear graphite materials under investigation.

15h10

Investigation into the Effects of Irradiation on Thermal Oxidation Behaviour

J. Dinsdale-Potter, A. Tzelepi and M. Jordan

United Kingdom National Nuclear Laboratory (UKNNL)

Thermal oxidation describes the process through which a material is reduced in mass through exposure to high temperatures and oxidising agents. Oxidation of graphite is significant as the resultant weight loss leads to a reduction of its effective strength, as well as changing other properties of the material.

Thermal oxidation is of interest to the development of future High Temperature Gas-cooled Reactors (HTGRs), where there is the potential for slow thermal oxidation to occur from impurities in the helium coolant. It is also of concern during air-ingress accident scenarios given the high operating temperatures of HTGRs.

To date, studies on the relation between the effects of neutron damage induced changes to graphite microstructure and resultant thermal oxidation behaviour have not shown a clear correlation between the two properties. This work combines thermogravimetric analysis (TGA) data from multiple studies to investigate the oxidation behaviour of material irradiated in both inert and oxidising environments.

15h30

Impact of Low-Temperature Oxidation on the Thermal Properties of Three Grades of Nuclear Graphite: NBG-17, NBG-18, and IG-110

M. Saoudi, D. Cluff, H. Wu, J. Mouris, C. Boyer and J. Battersby

Advanced Fuels and Reactor Physics Branch, Canadian Nuclear Laboratories, Chalk River, Ontario, Canada

The thermal properties of graphite used as a moderator in high temperature gas cooled reactors (HTGRs) need to be accurately quantified to assess the capacity of a graphite core to remove decay heat. In particular, during an air-ingress accident scenario, oxygen introduced into the

core, exposes the graphite to oxidation which can affect its thermal properties. Assessment of thermal conductivity of oxidized graphite in HTGRs is a critical parameter in the demonstration of effective passive heat removal from the reactor core in air-ingress accident conditions.

In this work, three types of nuclear grade graphite were thermally oxidized at 550°C in flowing air. Disc and cylindrical specimens were machined following standards dimensions for thermal diffusivity and thermal expansion measurements and oxidized up to 20% weight loss. This presentation discusses the changes in the coefficient of thermal expansion (CTE), thermal diffusivity, and thermal conductivity of the oxidized graphite specimens, which depend on the graphite microstructure.

15h50 *Coffee break*

16h20 **Thermal Oxidation of IG-110 Nuclear Graphite using Time-lapsed X-ray Computed tomography**

X. Zhang, A. Theodosiou, 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, UK

The thermal oxidation characteristics of IG-110 nuclear grade graphite have been investigated using time-lapse X-ray micro-computed tomography (μ CT). Specifically, the same location of IG-110 graphite samples have been tracked after thermal oxidation at 650°C, 700°C and 750°C. Our analysis shows that at 650°C, thermal oxidation has caused microstructure degradation over a deeper region whilst at 750°C thermal oxidation was confined near the surface of the specimen. High-resolution X-ray nano-computed tomography (nano-CT) is employed to examine the graphite microstructure in what appeared to be the dense regions in the μ CT virtual slices. The results show that these regions contain fine, lenticular-shaped cracks which can also serve as paths for oxygen diffusion into the specimen. Based on these observations, a pore network model was developed incorporating the nano-CT and μ CT data. This model enables the steady state oxygen distribution to be predicted and the relative role of oxygen diffusion through the pores of different length scales on the thermal oxidation response to be evaluated.

16h40 **Porosity Characterization of Nuclear Graphite Oxidized at 550 °C**

H. Wu, M. Saoudi, R. Osmond, C. Boyer, M. Gertig, J. Mouris, V. Szlachta and J. Budgell

Canadian Nuclear Laboratories

This study investigates the evolution of open and closed porosity in nuclear graphite IG110 and NBG17 upon air oxidation at 550 °C using Optical Microscopy (OM) and X-ray Computed Tomography (XCT). Both graphite grades were oxidized in air until approximately 20% mass loss was achieved. Surface porosity was quantified via OM image analysis using FIJI's Trainable Weka Segmentation, while open and closed porosity in NBG17 was analyzed using both XCT and fluorescent image analysis. The iso-pressed, fine-grained (20 μ m) IG110 exhibited markedly different oxidation behavior compared to the vibro-molded, medium-fine-grained (800 μ m) NBG17. At comparable weight loss levels, IG110 developed higher total surface porosity than NBG17. Oxidation in NBG17 was relatively uniform, whereas IG110 displayed non-uniform oxidation characterized by larger pores within the bulk. Pore size distribution analysis indicated that while most pores in both materials remained below 500 μ m², IG110 uniquely developed macropores exceeding 2000 μ m² post-oxidation. XCT analysis of NBG17 also revealed a reduction in closed porosity correlating with increased mass loss. These findings highlight the role of microstructural differences and oxidation conditions in governing the porosity evolution and oxidation behavior of nuclear graphite grades.

17h00

**General Discussion on the Day's Presentations – Leader: Dr. J. Reed, EdF
Nuclear Operations**

Tuesday 30th September 2025

Irradiation Behaviour and Related Research

Chairman: Dr M Srinivasan, USA

08h30

**Determining the Accuracy of Ion Transport Models for Ion Beam Implantation
in Nuclear Graphite**

J. Daw, A. Theodosiou and A. N. Jones

Nuclear Graphite Research Group, Department of Mechanical, and Aerospace Engineering, School of Engineering, Henry Royce Institute University of Manchester, M13 9LP, UK

Graphite has played a vital role in nuclear reactors and remains a key component in the Molten Salt Reactors (MSRs) and High Temperature Gas-cooled Reactors (HTGRs). These advanced systems expose graphite to higher temperatures, greater neutron flux, and corrosive environments, potentially altering its behaviour under irradiation. Whilst materials test reactor studies provide essential data, these are costly and time-consuming. Ion beam implantation offers a faster, more controlled alternative to simulate radiation damage and introduce relevant chemical species for studies of fission products and contaminants.

Traditionally, ion transport modelling utilise a programme called SRIM - Stopping and Range of Ions in Matter due to its accessibility, however its static target approximation limits accuracy. New software, TRIDYN, by contrast, dynamically updates the target's structure and tracks displaced atoms, offering improved predictions. Currently published studies have not applied TRIDYN to model ion implantation in nuclear-grade graphite.

This study compares the accuracy of TRIDYN and SRIM models for ion implantation in IG-110 graphite. Ge²⁺ ions were implanted at 25 keV between doses of 1×10^{15} and 1×10^{17} ions·cm⁻² and analysed using Secondary Ion Mass Spectrometry (SIMS). This work investigates if TRIDYN can be used to predict the ion distribution in implanted graphite and compares how both models approximate the damage received by the graphite sample.

08h50

**Evaluation of Irradiation-Induced Dimensional Changes in Nuclear Graphite
Based on Ion Beam Irradiation-Induced Bending**

He Zhoutong

Shanghai Institute of Applied Physics, Chinese Academy of Sciences, Shanghai 201800, China

Nuclear graphite development has advanced rapidly in recent years, driven by demand. Evaluating irradiation behavior in nuclear graphite, particularly irradiation-induced dimensional changes, continues to be a major bottleneck in its development due to the scarcity of facilities, high costs, and lengthy durations associated with fast neutron irradiation test programs. Developing a quick evaluation method for irradiation-induced dimensional changes in nuclear graphite could significantly accelerate the development of new nuclear graphite. This method could also be used to assess the uniformity of irradiation behavior in nuclear graphite blocks and the consistency between different graphite blocks and batches, addressing the final challenge in quality control.

Ion beam irradiation has become a significant technique for simulating fast neutrons in irradiation materials, owing to its similar mechanism, facility accessibility, time efficiency, low cost, and precise temperature control. Presently, irradiation methods to assess dimensional

changes (such as irradiation swelling and growth) in alloys and ceramic materials have been well-established, including step-height measurements, micro/nano-cantilever bending techniques, and bubble statistics methods. However, these methods are not suitable for nuclear graphite due to the porous nature and local inhomogeneity of nuclear graphite.

In recent years, we have developed a method to evaluate the dimensional changes in nuclear graphite induced by irradiation, based on ion beam exposure. The method utilizes an ion beam to irradiate thin graphite foils, causing the foils to bend due to the varying strains produced by ion beam irradiation at different depths. By correlating the strain with the curvature change of the foil sample using a model, we can estimate the dimensional change of graphite under irradiation.

To verify the method, 30 MeV Ni ion beams were used to irradiate thin graphite samples at 400°C and 600°C. The changes in curvature of the samples were measured, and a model that relates these curvature changes to dimensional changes was employed to predict irradiation-induced dimensional changes and irradiation lifetime. Additionally, post-irradiation microstructural evolution was analyzed to compare the differences between ion and fast neutron irradiation.

By comparing the results with fast neutron irradiation data, it was demonstrated that the ion beam method can effectively evaluate irradiation-induced dimensional changes across different graphite grades. This approach is a promising tool for advancing the development of new nuclear graphite grades.

09h10

Annealing Behaviour and Ion Mobility of IG-110 Graphite in HTGR Environments

F. Altamimi^{1,3}, M. Coke², B. F. Spencer², S. Sheraz², E. Aradi³, M. Lindley⁴, N.P. Lockyer², R. J. Curry⁵, S. J. Haigh⁴, C. A. Sharrad¹, and A. N. Jones³

¹Department of Chemical Engineering, The University of Manchester, Oxford Road, Manchester M13 9PL, UK

²Photon Science Institute, Faculty of Science and Engineering, The University of Manchester, Oxford Road, Manchester M13 9PL, UK

³The Nuclear Graphite Research Group (NGRG), Department of Mechanical and Aerospace Engineering, The University of Manchester, Henry Royce Institute, Manchester M13 9PL, UK

⁴Department of Materials, University of Manchester, Manchester M13 9PL, U.K

⁵Photon Science Institute, Department of Electrical and Electronic Engineering, The University of Manchester, Oxford Road, Manchester M13 9PL, UK

Low-energy cobalt ion irradiation was performed on IG-110 graphite to investigate the mobility of cobalt in irradiated graphite. Ion doses ranged from 6.35×10^{14} to 1.27×10^{16} ions/cm². Subsequent thermal treatment in a helium environment at temperatures up to 1100 °C showed recovery of graphitic features starting from 750 °C, indicating healing of severe irradiation damage in HTGR environments. The changes of the graphite morphology and structure are assessed via Raman spectroscopy, Transmission Electron Microscopy (TEM), and Scanning Electron Microscopy (SEM). Additionally, Electron Energy Loss Spectroscopy (EELS) is utilised to quantify the damage profile from the graphite surface.

Cobalt ion mobility at high temperatures was investigated using several complementary high resolution techniques. *In situ* Hard X-ray Photoelectron Spectroscopy (HAXPES) identified the onset diffusion temperature. *In situ* SEM heating provided visual evidence of ion mobility. Thermal treatment in helium was carried out for durations ranging from 1 second to 4 hours at four different temperatures (550, 750, 950, and 1100 °C). Depth profile changes measured by Time-of-Flight Secondary Ion Mass Spectrometry (ToF-SIMS) revealed distinct diffusion mechanisms across the temperature range. A slow diffusion toward the surface appears likely at typical HTGR operating temperatures.

09h30

Hydrogen Uptake in Graphite Matrix at High Temperature

G.S. Rakib, J. Liu, E. Yelmenoglu and L. Vergari

University of Illinois Urbana-Champaign, USA

Graphite matrix (GM) is employed as fuel element in fluoride-salt-cooled high-temperature reactors (FHRs) and high-temperature gas-cooled reactors (HTGRs). In these systems, chemical interactions between tritium (produced via neutron activation of the gas and salt) and graphite components can affect reactor safety and operation. While there is evidence that nuclear graphite uptakes tritium, only limited data is available for GM, preventing a quantitative estimate of its uptake capacity and kinetics, especially in salt and under irradiation.

An experimental campaign is underway at the University of Illinois to address this gap. A gas sorption analyzer is being used to collect adsorption isotherms of hydrogen (a tritium surrogate) on A3 GM and to perform thermal desorption spectroscopy. Isotherms have been collected on pristine A at temperatures of relevance to FHRs and HTGRs and with variable equilibration times, helping decouple the thermodynamics and kinetics of the interaction. A diffusion-with-trapping code has been developed to extract uptake capacities, trapping site energies, and diffusivities. Our results highlight the multi-site nature of hydrogen trapping and suggest that the literature may underestimate uptake capacity because of short equilibration times and incomplete degassing. Further experiments are planned to elucidate the effects of salt chemistry and irradiation on hydrogen-graphite interactions.

09h50

Graphite Irradiation Testing at the Idaho National Laboratory

K. Kleimenhagen, M. Fanning, G. Hawkes and A.M. Cruz

Idaho National Laboratory, USA

Nuclear graphite is a key component in many proposed advanced reactor designs, wherein it is used as a neutron moderator, a structural material in the core, or both. Many grades of nuclear graphite exist today as possible options for reactor designers; however, no grades of graphite currently manufactured are qualified for use in commercial reactors. Part of the data required to qualify a given grade of graphite for use is creep behaviour under irradiation at various temperatures, irradiation doses, and mechanical loads. To that end, the Idaho National Laboratory (INL) designed and executed the Advanced Graphite Creep (AGC) and High Dose Graphite (HDG) graphite irradiation experiment campaigns over the past 20 years intended to provide the nuclear graphite industry with irradiation data required for qualification. The legacy AGC/HDG design experienced challenges maintaining temperature control during irradiations; the in-core section has been updated with lessons learned from past tests for the upcoming HDG-2 experiment. Additionally, the INL designed a new Vendor Irradiation Capsule (VIC) experiment intended to provide data for temperatures, doses, or graphite grades not found in the original AGC/HDG campaign. The VIC design will be presented and discussed alongside the improvements made to HDG-2.

10h10

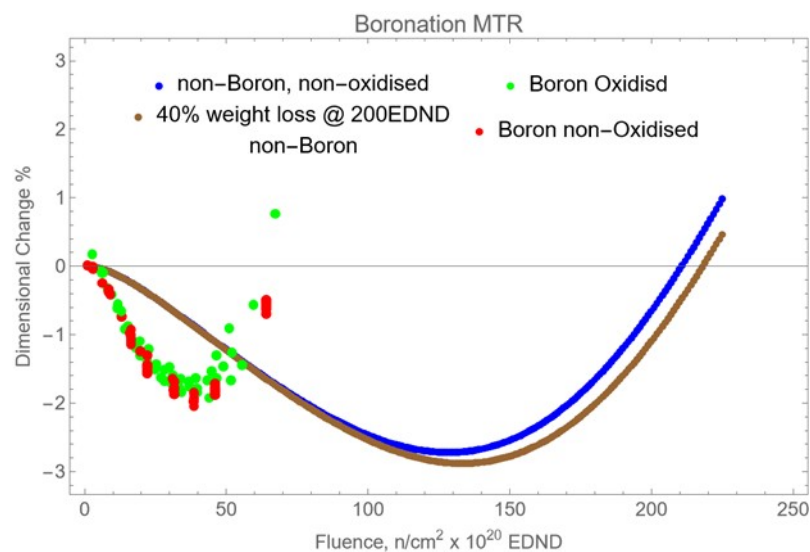
Revisiting the Acceleration of Irradiation Damage in Nuclear Graphite using Boron Doping

B. J. Marsden, G. Hall and A. N Jones

Nuclear Graphite Research Group, Department of Mechanical and Aerospace Engineering, Royce Hub Building, The University of Manchester, M13 9PL, UK

During microstructural investigations of irradiated graphite, Thrower [1] observed that the density of irradiation damage loops was higher in some graphite grades than expected. This was later found to be due to contamination from boron during preparation by heat treatment. Subsequently it was shown that substitutional boron doping may be used to accelerate damage in polycrystalline graphite, thus potentially reducing the time and cost of MTR graphite irradiation programs by about 67% [2]. Recognising this benefit during the 1980s, the UKAEA undertook an extensive irradiation programme to provide graphite irradiation data for the last

two AGR NPPs to be built in the UK. This involved both boronated and non-boronated samples irradiated in inert and radiolytic oxidising (CO₂) atmospheres. The boronated part of the programme was successful, but unfortunately the non-boronated part was cut short due to the closure of the UKAEA MTRs. The results from the boronated oxidised samples revealed some unexpected results when compared to the boronated inert samples. No discernible change in dimensional change turnaround was observed and the fall expected fall in the coefficient of thermal expansion at high dose was significantly reduced. Recently an extensive irradiation programme aimed at extending the lifetime of the AGRs was carried out by EDFE providing irradiation data on radiolytically oxidised graphite on non-boronated samples to a high fluence. Interestingly. These new MTR experiments confirmed the boronated results that had discredited the earlier work. Graphite boronation is discussed, highlighting the use of boronation as a useful tool for obtaining and comparing the graphite irradiation behaviour for new reactor designs, even in an oxidising atmosphere.



References

1. Impurity Nucleation of Irradiation Damage in Graphite. Thrower P. A. Thrower Nuclear Materials, 12, 56 (1964)
2. The Effect of Substitutional Boron on Irradiation Damage in Graphite. Brockelhurst J.E., Kelly B.T., Gilchrist K.E. Chemistry and Physics of Carbon Vol. 17, Chapter 3, Marcell Dekker Inc. 1981.

10h30

Coffee break

11h00

Modelling the Thermal Conductivity of Nuclear Graphite under Irradiation

F.J. van Zanten^{}, M. Saitta², S. Robinson², S. Baylis¹*

¹ X-energy, Rockville, MD

² MPR Associates, Alexandria, VA

Many nuclear reactor designs utilize graphite due to its excellent thermal and neutronic properties. Understanding the thermal conductivity of graphite is important for analyzing several key scenarios: conducting heat from reactor core during an accident, insulating exterior components during normal operation, and evaluating the stress experienced by a graphite core component due to thermal gradients.

Neutron irradiation can induce order of magnitude reductions in the thermal conductivity by generating defects in the crystal lattice. The thermal conductivity of irradiated graphite is dependent on the amount of fast neutron irradiation and the temperature both during and after irradiation. Modelling the conductivity of irradiated graphite is especially challenging over a wide range of neutron doses.

In this work, the authors propose a semi-empirical model to calculate the change in the thermal conductivity based on dose and temperature. This model is a combination of equations developed for both low and high dose applications. Model coefficients are determined by fitting to experimental data from a variety of graphite grades, and the quality of the fit is evaluated through error plots relating the model to the experimental data.

11h20

Measuring Nuclear Graphite Thermal Conductivity by using Thermoreflectance Techniques

M. Jiang¹, M. Mowat², J. Pomeroy³, Z. Abdallah³, C. Densham⁴, M. Kuball³, and D. Liu¹

¹ Department of Engineering Science, University of Oxford, Oxford, OX1 3PJ, UK

² TherMap Solutions, HH Wills Physics Laboratory, University of Bristol, Bristol, BS8 1TL, UK

³ Centre for Device Thermography and Reliability, School of Physics, University of Bristol, Bristol, BS8 1TL, UK

⁴ High Power Targets Group, STFC Rutherford Appleton Laboratory, Harwell, Oxford, OX11 0QX, UK

Contact: ming.jiang@eng.ox.ac.uk, and dong.liu@eng.ox.ac.uk

Fine-grained nuclear graphite is a critical structural and neutron moderation material used in high-temperature gas-cooled reactors (HTGRs) and Generation IV fission systems. It also plays a vital role in proton accelerator targets for particle physics experiments exemplified by the Deep Underground Neutrino Experiment led by US FermiLab. Understanding how nuclear graphite's thermal conductivity evolves with temperature and neutron and/or proton irradiation gradients is essential for predicting heat conduction and thermal strain development yet remains a challenge. This study employed state-of-the-art thermoreflectance techniques to measure thermal conductivities in various fine-grained nuclear graphites at pristine condition as a foundation step, including IG-430U, IG-510U, IG-110U, G347A, POCO ZXF-5Q, Mersen 2020, Mersen 2114, as well as HOPG graphite.

Time-domain thermoreflectance (ns-TDTR) showed promising results from HOPG graphite having highly anisotropic thermal conductivities (up to 2000 W/m/K within *a*-axis and up to 10 W/m/K in *c*-axis). Thermal transient curves showed significant local variation of thermal transport properties in fine-grained graphites measured at micrometre scale. Furthermore, Frequency-domain thermoreflectance (FDTR) has been used to measure bulk thermal conductivities (beam diameter ~100 µm and probe depth ~ a few millimetres) of these graphite materials and the results showed reasonably good agreement with manufacturers' reported values. Thermoreflectance techniques are considered promising for measuring irradiated graphite materials with damage gradient due to their non-contact nature, relatively high spatial resolution, and minimal sample geometry requirements.

11h40

Experimental Study on the Irradiated Performance of Carbon Composites at High Temperatures

Josina W. Geringer¹, Jose D. Arregui-Mena¹, Lianshan Lin¹, Anne A. Campbell¹, Takaaki Koyanagi¹ and William E. Windes²

¹ Oak Ridge National Laboratory, TN, USA

² Idaho National Laboratory, USA

Advanced Gas-cooled Reactor (AGR) cores are composed of thousands of graphite bricks and some technologies envisioned that these bricks will be held together by tie rods or restraint straps made from a carbon ceramic composite material. The ASME BPV-III-5 code introduces a pathway to qualify these materials for nuclear reactors. Ceramic composite materials are proven technology used for various high temperature applications outside of nuclear environments and with an established supply chain. However historical data indicates that these materials do not tolerate high dose neutron environments well. The complex nature of the ceramic composites together with the dimensional instability common to irradiated carbon materials, results in cracking of the graphite matrix and carbon-based laminates.

With the development of the new AGRs and micro-reactors which requires lower dose environments, together with advancement in material technologies, there is an increased interest using these carbon ceramic composite materials for core applications. Vendors are increasingly more interested in using these materials for their unique strength properties and high temperature tolerance. A study is underway to perform some initial comparisons of various ceramic matrix composite materials with specific focus on irradiation behaviour. This talk focusses on background of the study, test matrix, irradiation vehicles and initial test results of the various materials.

12h00

Modelling Dislocation Induced Buckling and Dimensional Change in Graphite

D. Nasartdinova, K. Jolley and P. Mouratidis

Dept. of Chemistry, Loughborough University, UK

Neutron radiation induces significant dimensional changes in graphite, which are driven, at least in part, by microstructural defects. One such proposed defect mechanism is the Ruck and Tuck phenomenon, proposed by Malcolm Heggie in 2011 which has since been experimentally observed in irradiated graphite samples. The Ruck and Tuck mechanism involves the pileup of basal dislocations, leading to the formation of an atomic-scale ripple, which eventually folds over forming the Ruck and Tuck defect. In our work, we explore the feasibility of modelling the Ruck and Tuck process using Density Functional Theory (DFT) and Molecular Dynamics (MD) to examine defect stability and energetics. We successfully conducted several calculations investigating atomic-scale ripples (the initial stage of the Ruck and Tuck defect) in both bulk and surface layer graphene and found that basal dislocation pileups (above a critical magnitude) will result in ripples rather than an array of flat partial basal dislocations. Machine Learning models like MACE reduce computational costs while preserving the accuracy of ab-initio methods, enabling larger-scale simulations of irradiated graphite. Although the pre-trained MACE models were not optimized for these specific defects, we benchmarked their performance and propose fine-tuning strategies to improve their applicability in radiation damage modelling.

POSTER

Laser-Based Ultrasonic Resonance Measurements of Nuclear Graphite Elastic Modulus at Elevated Temperatures

E. Berry¹, J. Arregui-Mena², L. Lin², C. Contescu^{1,2}, N. Gallego² and J. Spicer¹

¹ Johns Hopkins University, Baltimore, MD, USA

² Oak Ridge National Laboratory, Oak Ridge, TN, USA

The microstructural characteristics of nuclear grade graphites affect the elastic modulus of these materials and can greatly impact their performance in nuclear environments. In order to observe changes in modulus when in use, appropriate sensing techniques may be employed, but results must be interpreted in relation to the baseline values obtained using reference materials. Measuring elastic modulus using sonic/ultrasonic resonance techniques is a standard approach, however related ASTM standards are not compatible with small sample sizes that might be associated with various reference materials. This work focused on using laser-based methods to extract modulus from resonance data collected on samples of PCEA in two different cylindrical geometries that were incompatible with ASTM standards. Results agreed with literature values and showed a range of behaviours that provided broader insight into these materials. Current work aims to relate these measurements to those obtained on samples that conform to ASTM standard specifications and to obtain measurements at elevated temperatures in order to demonstrate the utility of laser-based methods for assessing elastic modulus in high temperature environments.

12h20

General Discussion on the Irradiation Behaviour Session - Leader – Prof Zhou Xiangwen, Tsinghua University, China

12h50

Lunch

13h40

Synthetic Brittleness Characteristics of Irradiated Nuclear Grade Graphites*Makuteswara Srinivasan*

Materials Matter, Germantown, MD. U.S.A.

Quantifying irradiation-induced brittleness in nuclear-grade graphite enables safe reactor operation, timely component replacement, and informed decommissioning strategies. This study introduces a methodology for computing normalized synthetic brittleness indices (NSBIs) as dimensionless figures-of-merit (FOMs) that reflect microstructural degradation across a wide range of operating conditions.

Due to sparse experimental data, property trends were interpolated and extrapolated using regression models that couple dose-dependent polynomials with temperature-dependent Arrhenius functions. Two independent AI models were employed to ensure robustness and assess model sensitivity to baseline data fidelity.

Four indices were formulated: $NSBI_1$ (hardness/toughness), $NSBI_2$ (modulus/strength), $NSBI_3$ (strain-to-failure), and NHSBI (a hybrid index). These indices capture characteristic embrittlement trends—increasing brittleness at low dose, peak values at intermediate dose, and softening or plateauing at higher doses, especially under elevated temperatures. Grade-specific regression equations were calibrated to reflect these behaviours consistently, with baseline values anchored in literature and normalized across all models.

Eight graphite grades—IG-110, ATR-2E, Gilsocarbon, H-451, NBG-17, NBG-18, PCEA, and EU-10—were analysed due to their extensive characterization and historical or ongoing use in HTGRs, MSRs, and AGRs. Computed NSBIs demonstrate physically plausible trends that enable comparative assessment across grades and reactor environments.

This synthetic framework provides a risk-informed, lifecycle-relevant tool for anticipating graphite embrittlement, supporting decisions on component longevity, inspection intervals, and safe end-of-life handling.

14h00

Graphite Fatigue Evaluation Using the Modified Goodman Diagram and Miner's Rule*J. Cano, T. Ligon, F.J. van Zanten and S. Baylis*

X-energy, Rockville, MD, USA

Graphite core components (GCCs) are subjected to cyclic loadings from both normal operations and during transient events. Temperature changes during reactor start-up, shut-down, reactor trips, and load-following events are some examples of cyclic loads during a reactors' operational life. The ASME Code Section III Division 5 paragraph HHA-3144 lacks guidance on how to evaluate graphite fatigue. Emerging methodologies to assess fatigue life using statistical models, tolerance limits, and traditional fatigue methods (*e.g.*, Modified Goodman, Miner's Rule, *etc.*) can be candidates to evaluate GCCs. Despite the development of these methodologies', questions remain on the applicability in real components using realistic fluence and temperature distributions. In this study, a representative design is assessed using the Modified Goodman diagram developed from the Price Model and tolerance limits. Additionally, the applicability of the Miner's rule is assessed for cumulative fatigue damage in the case where multiaxial loading conditions are present. This work presents a technical basis for future fatigue evaluation guidance and provides confidence that the current methodologies are sufficient for fatigue life qualification.

14h20

Some Thoughts on the ‘Linear Inelastic’ Behaviour of Nuclear Graphite

Dong Liu and Ming Jiang

Department of Engineering Science, University of Oxford, Oxford, OX1 3PJ, UK

When a bulk sample of nuclear graphite is loaded in tension or bending, the load-displacement curves often show a linear trend approximately beyond 70%-90% failure load. The stress in the region of interest can be derived by analytically normalising the load applied, and the strains can be measured, either by strain gauges, digital image correlation method or digital volume correlation analysis. The generated stress-strain curves in tension, compression, bending, disc compression showed similar behaviour as their load-displacement relationship where the stress values increase proportionally to the strains until a very high percentage of failure load is reached. When neutron diffraction or X-ray diffractions are used to probe the change of elastic crystalline lattice strains (usually in terms of basal plane lattice spacing change) in the strained volume of the material with increasing load, the crystalline lattice strains mostly showed linear increase to about 30-40% of the failure load. This is then followed by a plateau where the crystalline lattice ceases to increase, i.e., the ‘true’ elastic deformation that is translated to crystalline lattice level only occurs up to 30-40% of the bulk failure load. Beyond this range, even though the bulk load-displacement/stress-strain curves appeared as linear, ‘inelastic’ mechanisms must have been activated in the bulk graphite material; hence this phenomenon in nuclear graphite is termed as ‘linear inelastic’ behaviour here. Based on our own data collected over the last 12 years as well as literature work, a deformation model for nuclear graphite will be presented. The impact of stress state on this will be discussed.

14h40

Inverse Identification of Damage-Fracture Behaviour in Nuclear Graphite

H.N. Chen^{1,2} and J. Shen^{1,3}

¹ Research Center of Space Structures, Guizhou University, Guiyang 550025, China.

² Key Laboratory of Green Building and Intelligent Construction of Guizhou Province, Guiyang 550025, China.

³ College of Civil Engineering, Guizhou University, Guiyang 550025, China

Accurate identification and simulation of nuclear graphite damage mechanisms and fracture behaviour are essential for graphite core structure design. To characterize damage evolution and crack propagation in graphite, disc compression tests and three-point bending tests on centre-notched beams were performed on fine-grained graphite (CDI-1D and IG-11). Digital image correlation (DIC) and electronic speckle pattern interferometry (ESPI) were used to measure full-field surface displacements of the specimens. A segmental finite element inverse analysis method was developed to characterize graphite damage evolution by quantifying Young's modulus degradation with tensile/compressive strains in disc specimens and propose a damage law accordingly. Fracture energy and the bilinear tension-softening curve (TSC) were determined by comparing experimental and simulated load-displacement responses from bending tests. The damage law was integrated with the TSC to establish a damage-fracture model which was applied to simulate tensile tests on L-shaped specimens with varying fillet radii. Simulation results were in good agreement with previous experimental results, verifying the rationality of present model. Numerical results further showed that larger radii expand the fillet damage zone and this blunting effect enhances load-bearing capacity.

15h00

Effect of Sample Thickness on the Splitting Tensile Strength of Coarse- and Fine-Grained Graphites

L. Lin¹, C. Contescu¹, J. B. Spicer² and N. C. Gallego¹

This presentation will be given by Tomos Grejtak)

¹ Oak Ridge National Laboratory, Oak Ridge, TN, 37831, USA

² University of Johns Hopkins, Baltimore, MD, 21218, USA

The ASTM D8289, Standard Test Method for Tensile Strength Estimate by Disc Compression of Manufactured Graphite, was developed to provide an alternative means for testing tensile

strength on smaller specimens. ASTM D8289 specifies that acceptable specimen diameter can range from 6 to 12.7 mm and that the maximum allowed sample thickness should be half of the diameter, which are compatible with available irradiation capsule volumes, and other measurements. However, information is limited with respect to the effect that thickness can have on the measured splitting tensile strength.

We will report on our efforts to understand the effect of sample thickness on splitting tensile strength of both coarse and fine grain graphites. We tested graphites samples from grades 2114 and IG-110, PCEA, NBG-17 and NBG-18; samples were Ø12.7mm diameter and of different thicknesses (6.35, 5, 4, and 3 mm). The digital image correlation (DIC) method and acoustic emission (AE) were applied along with the ASTM D8289 standard on samples to help interpret the measured results.

15h20

Coffee break

15h50

Evolution of Elastic Modulus in Nuclear Graphite

B Cowen¹, R Gray¹, M Joyce¹, J Reed² and R Miller²

¹ Frazer-Nash Consultancy Ltd, UK

² EDF Energy Generation, UK

It is well known that prolonged service within a nuclear reactor causes the material properties of the graphite moderator to evolve. An appropriate characterisation of these changes is essential to allow meaningful structural integrity calculations to be performed for core components.

Physically motivated material property models like EDF Energy's EIM, often attempt to decompose the evolution of a given property into multiple contributions arising from the irradiation environment. For elastic modulus, this typically considers fast neutron induced dislocation pinning and structural effects; together with void formation arising from radiolytic weight loss. Whilst this approach is attractive, the decomposition is challenging in situations where multiple mechanisms are thought to be operating.

This paper focusses on the evolution of elastic modulus in the low dose and weight loss regime. A re-interpretation of recent inspection and test reactor data suggests that the relationship between elastic modulus and weight loss maybe more complex than is considered by many models. It is hypothesised that this is linked to inhomogeneous oxidation of the filler and binder phases in this low weight loss regime and an alternative model is proposed. This approach may have implications for reactor designs where low levels of chronic oxidation are feasible.

16h10

Studying Crack Initiation in Fine-Grained Nuclear Graphite using Double-Notch Four-Point Bend Configuration

M. Jiang¹, A. Graham¹, H. Wu², C. Densham³ and D. Liu¹

¹ Department of Engineering Science, University of Oxford, Oxford, OX1 3PJ, UK

² Sinosteel Advanced Materials Corporation, Zhejiang, China

³ High Power Targets Group, STFC Rutherford Appleton Laboratory, Harwell, Oxford, OX11 0QX, UK

With the prevalence of fine-grained nuclear graphite materials in advanced nuclear fission reactors and proton accelerator beamline target systems, a mechanistic understanding of the deformation and fracture behaviour of fine-grained graphite is considered important, since it depicts their damage tolerance capability in these extreme environments. In particular, the crack initiation process in nuclear graphite is still less well understood. For instance, it is not clear if crack initiation is controlled by critical strain or stress locally. This question is further complicated by the distribution of defects (voids, grain boundaries, filler-binder interfaces) ahead of cracks and notches. This work employs a four-point bending configuration for testing graphite beams with two identical blunt notches on the same side of the beam, i.e., double-

notch 4-point bend (DN-4PB), to study the crack initiation process in fine-grained SNG742 and T220 graphite. This method has been successfully applied to other quasi-brittle materials [1]. The sample surface is imaged by a camera at each loading step enabling displacement and strain fields to be computed by digital image correlation (DIC). Scanning electron microscopy (SEM) has been used to identify the presence and location of microcracks as supplementary evidence for the crack initiation. The developed experimental methodology sets the foundation for future tests at high temperatures for studying crack initiation at elevated temperatures.

Reference:

[1] Nalla RK, Kinney JH, Ritchie RO. Mechanistic fracture criteria for the failure of human cortical bone. *Nature Materials*. 2003 Mar 1;2(3):164-8

16h30

Understanding Wetting Behaviour of Graphite by Water Contact Angle Measurement

Wenqi Li¹, Gang Seob Jung², Jose' David Arregui-Mena³, Lianshan Lin³, Jisue Braatz⁴, Cristian Contescu³, Jun Qu³ and Nidia Gallego¹

¹ Chemical Sciences Division, Oak Ridge National Laboratory, Oak Ridge TN 37831, USA

² Computer Sciences & Engineering Division, Oak Ridge National Laboratory, Oak Ridge TN 37831, USA

³ Materials Science & Technology Division, Oak Ridge National Laboratory, Oak Ridge TN 37831, USA

⁴ Isotope Science & Enrichment Division, Oak Ridge National Laboratory, Oak Ridge TN 37831, USA

In this study, we use Highly Oriented Pyrolytic Graphite (HOPG) as a model of graphitic material. A goniometer equipped with an environmental chamber was employed to measure the water contact angle of HOPG. The chamber enables precise control of temperature and humidity during measurements, allowing accurate assessment of wetting behaviour under well-controlled conditions. Based on our contact angle measurements and comparison with previous studies, we believe the intrinsic wettability of graphitic materials is hydrophobic. Additionally, the initially hydrophilic nature observed on freshly cleaved HOPG surfaces can be attributed to oxygen-containing functional groups formed at surface defects during exfoliation, as well as wetting transparency effect driven by van der Waals forces from the underlying graphene layers. Molecular dynamics (MD) simulations reveal that exposing a fresh surface will trigger an irreversible structure defect, such as ripples and wrinkles, on the top graphene layer, driven by defect relaxation at grain boundaries and the release of interlayer confinement. As exposure time increases, these structural deformations grow and the distance between the top graphene layer and the underlying layers expand, which leads to significant increase in water contact angle over time and the transformation of surface behaviour from hydrophilic to hydrophobic.

16h50

Measurement of Gas Permeability in Graphite under Steady Flow

D.K.L. Tsang

Melody R&D

Gas permeability is an important physical property which is required for correct assessment of graphite oxidation and to understand interactions between gas and graphite components in reactors. Such interactions include thermal and radiolytic oxidation, as well as the escape of gaseous fission by products through the graphite components.

A new novel method has been developed to measure the gas permeability in graphite. The new method is easy to setup. Different specimen sizes can be used in the method. Moreover, the specimen can be retrieved after the testing.

The new method has been used to study the gas permeability in graphite. Results show the gas permeability vary enormously within the same grade of graphite. The size effect of specimens on the gas permeability measurement has also been study and will be discussed.

POSTER

Determining the Shear Modulus and Young's Modulus of Irradiated Nuclear Graphite from the ACCENT MTR Programme

M. Forber^{1,2}, W. Bodel², C. Sharrad², M.A.L. Laot³, and A. N. Jones¹

¹ Nuclear Graphite Research Group, Department of Mechanical and Aerospace Engineering, School of Engineering, Henry Royce Institute, The University of Manchester, M13 9LP, UK

² Department of Chemical Engineering, The University of Manchester, Oxford Road, Manchester M13 9PL, UK

³NRG PALLAS, Westerduinweg 3, 1755 LE Petten, Nederland

Nuclear graphite is a key material within many nuclear reactors and undergoes degradation due to coolant and irradiation conditions. An understanding of this degradation is therefore of great importance to understand how the behaviour of graphite may change within a reactor and how the properties will be affected. An area of focus is the mechanical properties of nuclear graphite; however, measurement methods often require techniques that can be destructive. Dynamic modulus testing provides a non-destructive alternative to traditional techniques, through the measurement of the time-of-flight of an ultrasonic pulse through specimens. The use of this technique on highly irradiated and highly oxidised nuclear graphite specimens is difficult to operate and reproduce. Within this work a new methodology and experimental set-up has been developed for the measurement of irradiated materials taken from the EDF MTR Programme ACCENT which have been subject to high levels of fluence and oxidation.

In order to reduce dose exposure and provide confidence in the new methodology, thermal oxidation has been performed on surrogate AGR graphite to understand the difficulty in irradiated modulus measurements and obtaining reproducible results in specimens with increased weight loss, this will also provide a comparison in modulus between thermal oxidation and radiolytic oxidation.

17h10

General Discussion on Mechanical/Physical Properties - Leader: Dr. W. Windes

Wednesday 1st October 2025

Graphite For SMR

Chairman: Dr Chichi Li, NRG PALLAS, The Netherlands

08h30

NRG's Graphite Qualification Programme for Kairos Power

M.A.L. Laot¹, J. McGladdery², G. Meric²

¹ NRG Pallas, Westerduinweg 3, 1755 LE, Petten, The Netherlands.

² Kairos Power, 707 W Tower Ave Suite A, Alameda, CA 94501, United States

Kairos Power is developing a Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR). Graphite is used as reflector and structural material in the KP-FHR, and future license applications rely on the qualification of the selected graphite grade; Ibidem's ET-10. The graphite qualification program pursued by Kairos includes both materials testing and modelling.

NRG Pallas possesses a series of nuclear facilities that enables characterisation of nuclear material properties under neutron irradiation and has been contracted to characterize the behavior of the ET-10 graphite following the requirements of Kairos Power. NRG Pallas is developing two irradiation facilities, ATHENA and ATLAS. ATLAS is a program focusing on graphite creep, and ATHENA is a program focusing on the baseline material properties. Both

ATLAS and ATHENA are multi-step irradiations where graphite specimens are irradiated in several irradiation steps to obtain datapoints at key fluences.

In this presentation, an overview of the program will be presented. Basic design features for the qualification projects ATHENA and ATLAS will be shared in this presentation and status of both projects will be communicated.

08h50

Updates on the Qualification and Testing of ET-10 Graphite for Use in the Hermes KP-FHR

J. McGladdery, R. Dieter, L. Cohen, T. Gerendasy, J. Choi, P.-A. Juan and G. Meric
Kairos Power, USA

Kairos Power's mission is to enable the world's transition to clean energy, with the ultimate goal of dramatically improving people's quality of life while protecting the environment. To accomplish this mission, Kairos Power is focused on the integrated design, licensing, and demonstration of advanced reactor technology, the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor (KP-FHR). As part of its iterative development process, Kairos builds 1:1 scale non-nuclear prototypes of the KP-FHR. The Engineering Test Unit (ETU) 1.0 built in 2023 has been successfully decommissioned, and completion of ETU-2.0 is scheduled to finish construction in 2025. Both ETUs provide invaluable knowledge for the construction of Hermes 1.0 test reactor: the first graphite moderated reactor to begin construction in the USA for over half a century.

The Hermes reactor uses ET-10, an iso-moulded graphite for structural graphite reflector blocks. This presentation will discuss Kairos' latest advances in graphite testing for supporting the iterative design process. This includes qualification of the ET-10 grade, aspects of oxidation behaviour unique to KP-FHR, and how empirical data can support damage tolerance design strategies. The presentation will also discuss progression of Kairos' in-house graphite machining capability, and lessons learned from the ETU 2.0 test iteration.

09h10

High-Resolution Porosity Characterization in Graphite Using Large-Volume Oxygen Plasma FIB-SEM Tomography

J. D. Arregui-Mena¹, S. Gonzalez-Calzada¹, P. D. Edmondson^{1,2} and N. C. Gallego¹

¹ Oak Ridge National Laboratory, TN, USA

² University of Manchester, UK

Graphite serves as a neutron moderator in nuclear reactors, where its porosity plays a crucial role in determining mechanical properties and irradiation response. This study highlights the application of oxygen plasma focused ion beam-scanning electron microscopy (O-PFIB-SEM) tomography in characterizing the three-dimensional pore architecture of ETU-10, a superfine nuclear graphite grade. The method not only facilitates detailed analysis of pore morphology but also provides a framework for evaluating molten salt intrusion into small pores and assessing the irradiation effects in accommodation cracks. High-resolution reconstructions spanning over 425,000 μm^3 were obtained with ~50 nm voxel resolution, effectively capturing thermal cracks, gas evolution porosity, and other pores—features that are challenging to resolve using conventional X-ray computed tomography. To mitigate milling artifacts associated with plasma ion milling, optimized preparation techniques were developed to minimize curtaining effects and streamline image acquisition. Additionally, AI-based segmentation methods were implemented to enhance pore detection and reduce acquisition times. The extended imaging volume and superior resolution achieved with O-PFIB-SEM enable a more comprehensive assessment of neutron irradiation effects in nuclear graphite, offering critical insights into pore connectivity and morphology that are less accessible through conventional Ga^+ FIB-SEM or TEM.

09h30

Preliminary Hardness Testing Results for Various Nuclear Reactor Graphite Grades and a Novel Hardness Testing Method for Fuel Pebble Testing

S. Hamilton^{1}, N. Montrose², A. Cunningham² and William Windes²*

¹ Nuclear Science & Technology, Idaho National Laboratory, Idaho Falls, ID

² Energy and Environment Science & Technology, Idaho National Laboratory, Idaho Falls, ID

Advanced gas-cooled nuclear reactors rely heavily on synthetic graphite for reflector blocks, moderators, and fuel matrix material. A critical concern within the Pebble Bed HTR design is the potential for wear and impact performance of the fuel pebbles in relation to the reflector blocks within the reactor core. This issue is particularly crucial for large pebble bed reactors, where large quantities of fuel pebbles are consistently cycled through the core, leading to significant friction and wear potential between the pebbles and the structural graphite material. While tribological studies are necessary to measure the quantitative wear behaviour of graphite, simple hardness testing can provide a much faster and easier method to estimate wear. Hardness tests will not substitute for tribological testing but are useful where less sophisticated data and less expensive techniques are sufficient. This study investigates the hardness of NBG-18, IG-110, and ET-10 grades of graphite according to ASTM C748 standards and discusses the development of a new hardness testing method for A3-3 matrix material in prototypic fuel pebble geometries. This work compares the hardness values of the structural graphite materials to the fuel matrix material to estimate how the different graphites will interact.

09h50

Friction and Wear Behaviour of Graphite in Fluoride-Salt-Cooled and High-Temperature Gas-Cooled Pebble-Bed Reactor Environments

T. Grejtak¹, N.C. Gallego², J.R. Keiser¹ and Jn Qu¹

¹ Materials Science & Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, USA

² Chemical Sciences Division, Oak Ridge National Laboratory, Oak Ridge, TN, USA

Fluoride salt-cooled and high-temperature gas-cooled pebble-bed reactors contain thousands of densely packed spherical fuel elements (pebbles), which circulate through the reactor core multiple times before permanent discharge. As pebbles circulate through the reactor, they undergo repeated sliding and rolling interactions with neighbouring pebbles, graphite structural components, and metallic elements of the containment assembly. These repeated interactions can lead to abrasion and structural degradation of both the fuel pebbles and the graphite wall components, and generation of graphite dust contaminated with fission products, posing a significant radiological hazard. ORNL has developed capabilities for tribological characterization of graphite in fluoride salt-cooled and high temperature gas-cooled pebble-bed reactor environments. Friction and wear characterization was performed on samples fabricated from nuclear-grade graphite using a pin-on-disk configuration in both molten FLiNaK salt and an inert argon environment to investigate the effects of temperature, contact load, sliding speed, and surface roughness. The outcomes of this research could provide guidance for evaluating the integrity and service life of pebbles and ensuring safe reactor operation.

10h10

Whole-core Transient Simulations with Novel Graphite Thermal Annealing Model

M.C. van den Berg, J.L. Kloosterman and D. Lathouwers

Delft University of Technology, The Netherlands

A depressurised loss of forced cooling (DLOFC) is one of the most severe incidents in a high-temperature gas-cooled reactor (HTGR). To remain inherently safe, the decay heat must dissipate to the environment through a thick layer of irradiated graphite. The thermal conductivity of graphite decreases sharply as a function of fluence, but can be recovered by post-irradiation thermal annealing. However, no general predictive model exists to quantify recovery across varying irradiation and annealing conditions.

We introduce a novel, physics-informed model that predicts the recovery of graphite thermal conductivity during post-irradiation annealing as a function of irradiation history (fluence, temperature, graphite grade) and annealing conditions (time, temperature, graphite grade). The model is validated against experimental data. Next, the in-house PHANTOM-OPERA system code is used to simulate a DLOFC in a prismatic micro-HTGR to assess the effect of graphite annealing on the maximum fuel temperature.

10h30 *Coffee break*

11h00 **Source Term Research on the Irradiated Graphite in HTGRs**

F. Xie^{}, T. Ma, L. Wei, S. Zhou, X. Liu, F. Wang, D. Ding*

Institute of Nuclear and New Energy Technology, Collaborative Innovation Center of Advanced Nuclear Energy Technology, Key Laboratory of Advanced Reactor Engineering and Safety of Ministry of Education, Tsinghua University, Beijing 100084, China

A large amount of graphite materials are used in high-temperature gas-cooled reactors (HTGRs), including graphite reflector layers and fuel element matrix materials. The study of radioactive source terms in graphite materials is a prerequisite for determining the treatment and disposal strategy for irradiated graphite. The HTR-10 has a total of 27,000 fuel elements. The total weights of the graphite reflector layer and the carbon insulation layer in HTR-10 are approximately 60 and 42 tons, respectively.

We have completed the measurement and analysis of the source terms of four irradiated graphite spheres from the HTR-10 core, determining the typical nuclides of H-3, C-14, Co-60, Cs-137, Eu-152 and Eu-154, and measuring their specific activities and radial distributions. We have combined the first-principle method and thermal desorption theory to determine the quantitative relationship between the thermal desorption peak temperature of hydrogen isotopes from the graphite and the adsorption configuration, and explained the mechanism of tritium enrichment on the surface of graphite spheres.

Finally, we discuss the future research work related to irradiated graphite, such as the properties and behaviour of radioactive graphite dust, the interaction between different chemical states of radionuclides and the graphite surface, and so on.

11h20 **Research on Volume Inspection of Graphite and Carbon Components in High-Temperature Gas-Cooled Reactors**

Cong Peng^{1,2}, Liu Ximing^{1,2}, Wang Liqiang^{1,2}, Zhang yanmin^{1,2}, Huang Yibin^{1,2}, Zhang Yuai^{1,2}, Sun Yuewen^{1,2}, Chen Liu^{1,2}, Tong Jianmin^{1,2}, Wang Xuqing^{1,2}, Zheng Jian^{1,2}, Tan Chunming^{1,2}, Zeng Tianchen^{1,2}, Fu Jintao^{1,2}, Chen Yilun^{1,2}

¹ Institute of nuclear energy and new energy technology, Tsinghua University

² Beijing Key Laboratory on Nuclear Detection & Measurement Technology

Different from other reactor types, the HTGR uses a large number of ceramic materials as core support structures. During the production and manufacturing of porous carbon materials, defects such as pores, cracks, inclusions, and looseness are inevitably generated and may expand under long-term high-temperature and neutron flux irradiation, and even cause cracking and shedding, thereby affecting the safety of the reactor. Considering that the core support structure cannot be replaced, it is necessary to conduct a quality inspection before it is put into the reactor.

For the volume detection of large-scale graphite/carbon components, there are no effective detection standards. The INET of Tsinghua University has first proposed a pipeline-type multi-slice spiral CT volume detection method for graphite components. We use a cobalt-60 isotope source to emit γ -rays to penetrate the detected components, and high-performance multi-row array detectors are used to collect signals. Through high-quality real-time image reconstruction algorithms and an AI network for automatic defect recognition and measurement, the detection reports can be obtained.

INET has completed the design and manufacturing of several volume detection systems, which have been applied in several domestic nuclear graphite production facilities and inspected a large number of graphite/carbon components to ensure the safety of HTGRs.

POSTER

Determining the Behaviour and Migration of Metallic Fission Products in Nuclear Graphite

G. Blom, A.N. Jones¹, B. Spencer and A. Theodosiou

Nuclear Graphite Research Group, Department of Mechanical and Aerospace Engineering, School of Engineering, Henry Royce Institute University of Manchester, M13 9LP, UK

Next generation reactors may use TRISO fuels which under operational conditions could release fission products into the graphite matrix. Therefore, in order to quantify the migration and retention of metallic fission products in the graphite grades we will investigate fission products implantation in two different nuclear grades of graphite (IG-110 and NBG-18) using ion irradiation. The distribution of deposited ions has been analysed using X-ray photoelectron spectroscopy (XPS) to determine differences in chemical distribution profiles between grades. Additionally, the effect of ion irradiation on the graphite matrix has been investigated using Raman spectroscopy and Scanning Electron Microscopy (SEM) to characterise any damage or migration effects. This work hopes to further investigate migration of fission products under high temperature reactor conditions (750 – 950 °C) using *in-situ* heating XPS with the aim to provide important insights into the distribution, transport and retention of fission products within the graphite components for future reactors.

POSTER

Application of SiC-Coated Graphite in Sodium-Cooled HTR

Nadia Dempow¹, Matteo Monegaglia²

¹ Blue Capsule Technology, Playground Paris-Saclay, 3 Boulevard Thomas Gobert, 91120 Palaiseau, France

² Blue Capsule Technology, Technopole de l'Environnement Arbois Méditerranée, Avenue Louis Philibert, 13290 Aix-en-Provence, France

Blue capsule Technology is developing a High Temperature Reactor (HTR) that uses TRISO fuel, with fuel kernels made of UO₂ (LEU). The reactor uses graphite as moderator and sodium as the coolant.

Graphite plays a key role in the Blue Capsule fuel design : it is used both in the compact fuel form, which consists of TRISO particles embedded in a graphite matrix, and in the fuel assembly barrel, which is coated with a thin layer of silicon carbide (SiC).

The moderator is a critical component in HTR, as it serves to slow down fast neutrons, making the fission process more efficient. In the Blue Capsule reactor, the moderator is made of graphite coated with a thin SiC-CVD layer.

Extensive studies have been conducted on various types of nuclear grade graphite to determine which material is most suitable for HTR conditions and compatible with SiC. The selected graphite must exhibit high density, good thermal and mechanical properties, and dimensional stability under irradiation. For the Blue Capsule concept, two graphite grades, Mersen grade 2114 from France and IG-110 from Japan, appear to meet these requirements, showing minimal densification under irradiation and fulfilling the key criteria for nuclear grade graphite.

The selection of Mersen 2114 and IG-110 graphite grades appears to be well suited for the Blue Capsule HTR design, offering both thermal and mechanical stability, as well as compatibility with SiC. Blue Capsule envisages experiments aiming at confirming properties of Mersen 2114 (bare and coated), in particular at high temperatures and in contact with sodium. A high-burnup irradiation campaign will be carried out by Mersen to confirm the long-term performance of Mersen 2114 graphite.

11h40

General Discussion - SMR Issues - Leader: Mr Frederik Reitsma, IAEA

Irradiation Creep

Chair: Prof Barry Marsden, The University of Manchester, UK

12h00

A General Discussion on Irradiation-Induced Graphite Creep Behaviour and Model Comparison with Recently Obtained Data on SNG742

C.Li¹, T.O. van Staveren¹, H. Wu² and H. Zhang²

¹ NRG Pallas, The Netherlands

² Sinosteel advanced materials (SIAMC), China

The effect of irradiation induced graphite creep is an essential input for the stress analysis of graphite core components. Irradiation creep can relieve stresses that arise from thermal gradients and graphite dimensional change, and thereby extend the lifetime of core components.

Irradiation creep experimental designs are generally complicated in order to achieve target temperature conditions and loading conditions on the specimens. Both need to be well controlled and determined by in-pile and out-of-pile measurements. Only limited data of graphite irradiation creep is available publicly, and this holds especially for the high dose regime (>10 DPA).

Based on historical irradiation creep data, a dislocation pinning-unpinning model (Kelly's model) was proposed, which is currently widely used as assessment method for modern graphite grades to obtain the secondary creep rate and as input for core component modelling. However, discrepancies can be observed when comparing this model with ATR-2E graphite data in the high dose regime. Several researchers have published different models, trying to describe this discrepancy and some models obtain better fits to the ATR-2E data. Nevertheless, more data at high dose regimes is needed to further validate these models.

The SINOGRAPH project is a set of irradiation experiments conducted at NRG PALLAS to study the irradiation creep behaviour of a fine-grain graphite type SNG742 aiming for application in HTR (high temperature gas cooled reactors). It has completed two phases of irradiation with an accumulated DPA of ~6 DPA. The third phase of irradiation is on-going. In this presentation, graphite irradiation creep data available in the public domain and data obtained from the SINOGRAPH experiment is compared to different creep models and observation of the relationship between data and model will be discussed.

12h20

A SLS Viscoelastic Model for Irradiation Induced Creep in Graphite

M.A. Davies¹ and J Reed²

¹ Marad Limited, UK

² EDF Energy Generation, UK

Irradiation induced creep in graphite and its subsequent effect on other material properties, especially CTE, is one of the most important factors in the determination of component cracking in HTGR fuel blocks and reflector blocks which ultimately determines the safe operating lifetime of HTGRs. However, despite multiple investigations including several creep experiments over many decades and an IAEA CRP, no universally accepted model of irradiation induced creep in graphite exists today.

This paper presents a standard linear solid viscoelastic model of irradiation induced creep in graphite and its validation against a recent EDF Energy compressive creep experiment, ACCENT conducted at Petten in the Netherlands. The paper will also apply the model to wider creep programmes conducted in support of US and German HTGR programmes to examine irradiation induced creep behaviour at high dose under both tension and compression.

12h40

Irradiation Creep and Dimensional Change in Gilsocarbon Graphite, Measured by Digital Volume Correlation of X-ray Computed Tomographs

M. Gregory¹, T. Bhudia^{1,2}, M. Brown³, G. Hall⁴, A. Jones⁴ and T.J. Marrow¹

¹ University of Oxford, UK

² Imperial College, London, UK

³ EDF Energy Generation.UK

⁴ University of Manchester, UK

Irradiation creep is dimensional change that occurs when graphite is irradiated by fast neutrons whilst under load. Creep is important for the relaxation of the stresses that develop in graphite components due to thermal gradients and neutron dose gradients. Without irradiation creep at a sufficient rate, this stress could cause the graphite to fracture. Experimental data for irradiation creep, used to predict its effect on stress evolution in reactor cores, are quite limited. The 'ACCENT' creep experiment, conducted by NRG Petten for EDF Energy, studied Gilsocarbon graphites used in the UK AGRs (Advanced Gas-cooled Reactors). Its materials characterisation included high resolution X-ray computed tomography (μ XCT) of paired samples (loaded in compression and a reference without stress) at intervals of irradiation dose.

This study has applied digital volume correlation (DVC) to some of the ACCENT μ XCT data. DVC is an image analysis technique that tracks features, such as pores, to obtain the relative three-dimensional displacement field between tomographs. The high spatial resolution of DVC found a systematic difference in the dimensional change of the coarse filler and matrix regions in the Gilsocarbon graphite microstructure. Measurements of the displacement gradients within the samples gave the bulk strains to obtain the irradiation creep strain rate and showed that swelling of unstressed graphite at low irradiation dose is an intrinsic behaviour of the microstructure.

13h00

Lunch

TRISO and Matrix Material

Chair: Dr Monika Grimm, DMT GmbH, Germany

14h00

A Strategy to TRISO Fuel Licencing in the United Kingdom

M. W. Owen¹, R. S. Aldred¹, T. Taylor¹, C. Davidson², N. Rhodes² and K. Atkinson²

¹ EDF Energy R&D UK Centre, Gloucester, GL3 4AE, UK

² EDF Energy, Nuclear Services, Gloucester, GL3 4AE, UK

High Temperature Gas-cooled Reactors (HTGRs) are one of the potential Advanced Modular Reactor (AMR) designs to be pursued by the UK Government under the AMR Research, Development and Demonstration (RD&D) Programme. Some HTGRs use Coated Particle Fuel (CPF), often referred to as TRI-structural ISO-tropic (TRISO) fuel. TRISO fuel is encompassed in graphite and then embedded in a larger graphite pebble or prismatic block, dependent on the reactor core. TRISO fuel has not been licenced in the UK, therefore, work is required to understand the licencing process of this novel nuclear fuel.

This work will understand how conventional UO_2 nuclear fuel is licenced in the UK, and whether this knowledge is transferrable to TRISO licencing. It will consider the differences in the regulatory regime from e.g. the US, and the possible implications for the licensing of TRISO fuel in the UK. The potential TRISO lifecycle will be discussed and compared to conventional UO_2 . Important parameters for conventional UO_2 licencing will be discussed, alongside how parameter limits are determined within the safety case. The requirement for experimental and modelling work to support the safety case will be understood, and whether more work is required for TRISO qualification.

14h20

HTGR Graphite Matrix Compact Production R&D

S. Wilkinson, M. Crelling, J. Dinsdale-Potter, R. Smith and A. Tzelepi
United Kingdom National Nuclear Laboratory (UKNNL)

High temperature gas-cooled reactors (HTGRs) are one of the next generation civil reactors undergoing research and development within the UK currently. Over a number of years UKNNL has been funded through the UK government's Coated Particle Fuel (CPF) programme to increase the technology readiness level (TRL) of HTGR fuel through an extensive research and development programme. This has included developing TRISO (Tri-structural ISOtropic) fuel manufacturing pilot scale capability and a matrix graphite research facility.

This presentation details the latest research and development work conducted at the matrix graphite research facility at UKNNL Workington. This will include elemental analysis, and particle size analysis, as well as the high temperature techniques for characterising thermal and mechanical behaviour.

14h40

Evaluation of In-Reactor Performance of TRISO Fuel: A Materials Perspective

Makuteswara Srinivasan

Materials Matter, Germantown, MD. USA

TRISO (**TR**istructural **ISO**tropic) fuel, known for its multi-layered structure and high-temperature resilience, is central to both prismatic and pebble-bed reactor designs. This study presents preliminary findings from a materials-based evaluation of TRISO fuel performance under irradiation and thermal stress, drawing on operational data from Fort St. Vrain (USA), AVR (Germany), HTR-10 and HTR-PM (China), and HTTR (Japan). Post-irradiation examination (PIE) of FSV and AVR fuels reveals that while TRISO coatings maintain their integrity, surrounding graphite and matrix materials may deform or fragment, posing handling challenges. To investigate these effects, the study models key mechanical and thermal properties—Young's modulus, strength, hardness, fracture toughness, and thermal conductivity—at both the composite compact level and individual TRISO layer level (buffer PyC, IPyC, SiC, OPyC, and graphite matrix).

Dose- and temperature-dependent degradation behaviours are quantified through simplified models calibrated against empirical data, including validation with the 2021 PNNL report *TRISO Fuel: Properties and Failure Modes* (Wells et al.). A companion spreadsheet-based tool was developed to estimate neutron fluence, dose, and material degradation trends across reactor conditions.

This framework supports durability assessment for TRISO-fuelled high-temperature gas-cooled reactors (HTGRs), such as the KRONOS MMR, which uses TRISO in a fully ceramic matrix (FCM®). The work also informs hybrid designs like Hermes, a molten salt-cooled reactor using TRISO pebbles. Understanding TRISO's thermo-mechanical resilience is essential to guiding fuel qualification and safety for next-generation advanced reactors.

15h00

The Wear Behaviour of A3-3 Matrix Graphite for Pebble-Bed HTR

Xi Tong, Xiangwen Zhou and Bing Liu

Institute of Nuclear and New Energy Technology, Tsinghua University, Beijing, China

The pebble-bed HTR can achieve non-stop refuelling and high burning consumption of the fuel elements, while the inter-pebble wear and generated graphite dust become an important issue for the safety analysis of pebble-bed reactors. The complex motion of fuel pebbles in the pebble bed makes it difficult to evaluate the wear behavior and graphite dust generation, and also lacks systematic and in-depth research. In this work, a roller wear device is used to test the wear behavior of A3-3 matrix graphite pebbles. The mass loss and wear rate of matrix graphite pebbles and the morphology of graphite dust at different wear times (0-100 h) were obtained

and analysed. The worn surfaces of graphite pebbles were observed by SEM and a surface profiler. The results indicated that the first hour of wear was the most severe, with mass loss accounting for approximately 25 wt% of the total quality loss. Graphite dust showed a lamellar structure. With the progress of wear, pits and lubrication films formed on the worn surface. The lubricating films were in a state of dynamic formation and destruction due to the collisions, rolling and sliding between pebbles. The roughness of the worn surface gradually increased.

15h20 **Announcement of INGSIM-26**

Depart for Schönbrunn: tour, conference dinner, and concert

Details given earlier: meeting point is the VISITOR ARRIVAL CENTRE adjacent to the coach park at U4 Schönbrunn station **NO LATER THAN 16h45**

Dress code: business casual

Thursday 2nd October 2025

Molten-Salt Reactor Issues

Chair: Dr Mouna Saoudi, Canada Nuclear Laboratory

08h30 **Hi-Fidelity Characterisation of Molten Salt – Graphite Pore Interactions through Experiments and Embedded Modelling**

A. Theodosiou, X. Zhang and A. Jones

Department of Mechanical and Aeronautical Engineering Division L5, University of Manchester, M13 9PL, UK

Many MSR designs utilise graphite as a moderator material, however one of the key challenges to molten salt reactor commercialisation is the known lifetime limitations of graphite in a reactor core; even with design enhancements, graphitic components are subjected to fast neutron flux in a MSR for several years and there is a potential for salt or fuel salt infiltration into the bulk graphitic regions.

This project is a US/UK collaboration looking primarily at the interactions of molten salts with graphite through advanced characterisation techniques and a range of fuel-salt infiltration experiments. Both FLiBE and K/LiCl salts will be investigated, the subsequent effects on mechanical properties of the graphite due to infiltration, through XCT, microscopy and hi-fidelity data analytics and modelling and comparisons of these effects across a range of candidate graphite grades with varying microstructure.

08h50 **The *in situ* Imaging of Molten-Salt Infiltration of Nuclear Grade Graphite using Synchrotron X-ray Computed Tomography**

F. E. Brooks-Ward¹, X. Zhang², F. Altamimi², A. Farrokhnia², N. Edwards², V. Zolotarevskiy², T. Zillhardt³, C. A. Sharrad¹ and A. N. Jones²

¹ Department of Chemical Engineering, The University of Manchester, Manchester M13 9PL, UK

² The Nuclear Graphite Research Group (NGRG), Department of Mechanical and Aerospace Engineering, The University of Manchester, Henry Royce Institute, Manchester M13 9PL, UK

³ I12 Beamline, Diamond Light Source, Diamond House, Harwell, Science and Innovation Campus, Didcot, Oxfordshire, OX11 0DE, UK

The next generation of reactors (Gen IV) includes a variety of designs for improved reactor operation, one of which is the molten salt reactor (MSR). There are several designs of MSRs which utilise either a fluoride or chloride-based salt eutectic, and many include the use of graphite to provide structural components and nuclear moderation. Nuclear graphite has approximately 20% porosity, which may be susceptible to infiltration of the fuel salt when in direct contact. If the salt is allowed to penetrate into the pore network, neutron poisons such as ¹³⁵Xe could be transferred and retained. There are additional important implications for the structural integrity of the graphite, as the two phases undergo dimensional changes under elevated temperatures and throughout the reactor lifetime. By gaining a deeper understanding of any infiltration mechanism, there can be improved optimisation of the MSR design.

This first of a kind experiment, conducted at the UK's Diamond Light Source, used high powered synchrotron X-ray Computed Tomography (XCT) to track the real time infiltration of molten salt at elevated operational environments. Various fine, medium and large grain graphite grades have been studied, and all have been subjected to salt inflation at temperature with *in situ* analysis. The results of this experiment will also contribute to the development of future models which can be used to predict the evolution of nuclear graphite properties exposed to MSR environments.

09h10

NCT Visualization of Molten FLiNaK Intrusion in Graphite Pores at Low Pressure and Extended Exposure Time

N.C. Gallego, J. Moon, C.I. Contescu and Y. Zhang

Oak Ridge National Laboratory, Oak Ridge, TN, USA

Molten salt infiltration in the pore network of nuclear graphite may cause unwanted changes to graphite's local structure and mechanical and thermal properties. A detailed and comprehensive understanding of molten salt intrusion (distribution across sample cross section and penetration depth) is needed to assess its effects. In this work, we report on an improved methodology for the use of neutron imaging (computed tomography) to evaluate salt penetration and distribution of a wide range of graphite grades with diverse microstructures. Neutron tomography data were acquired on the same graphite sample before and after salt intrusion; the 3D reconstructed volumes were digitally co-registered and subtracted. The difference in neutron attenuation coefficient represents direct visualization of FLiNaK (LiF–NaF–KF) salt distribution in the salt-impregnated graphite samples. This improved methodology was applied to investigate the effect of exposure times (12 h and 336 h) and of graphite microstructure when exposed to FLiNaK at 750 °C and 3 bar (gauge) pressure, starting from flowing argon at near atmospheric pressure. The results show that medium-grained and fine-grained graphites evolve to equilibrium at significantly different rates: fast salt uptake in medium-grained graphites produces salt deposits throughout the volume of graphite specimens, whereas salt infiltration in fine-grained graphites is much slower and limited to exposed surfaces.

09h30

Investigation of Tritium Desorption from IG-110 Graphite within FLiBe

P. Boca^{1,2} and R.O. Scarlat²

¹ Ecole Polytechnique, France

² Nuclear Engineering Department, University of California, Berkeley, California, USA

Neutron activation of the lithium and beryllium leads to the production of tritium within molten salt Li_2BeF_4 (FLiBe) that is used as coolant in fluoride-salt-cooled-high-temperature reactors (FHRs) and as in prospective fusion breeding blankets. Moreover, graphite has shown uptake capacity for tritium, which evolves with neutron irradiation of the graphite. The interaction between tritium and graphite has been well understood under vacuum and cover gas conditions.

In this work, hydrogen and graphite interaction will be investigated within molten FLiBe. The desorption mechanisms are investigated *in situ* using electrochemical techniques to probe surface interactions at a IG-110 graphite coupon. Both cyclic voltammetry (CV) and electrochemical impedance spectroscopy (EIS) are employed using a three-electrode cell setup, initially with a gold working electrode and subsequently with graphite. And hydrogen uptake is modulated by the addition of LiH which acts as a hydrogen donor.

The electrochemical data are interpreted using a desorption-reaction model to extract kinetic and thermodynamic parameters. Additionally, elemental analyses are performed on FLiBe samples before and after the experiments to assess the baseline composition and quantify potential contaminants that could interfere with electrochemical measurements.

POSTER**Exploration of Molten Salt Simulant-Graphite Interfaces Using Laser-Based Ultrasonic Techniques**

E. Berry¹, J. Arregui-Mena², L. Lin², C. Contescu^{1,2}, N. Gallego² and J. Spicer¹

¹ Johns Hopkins University, Baltimore, MD

² Oak Ridge National Laboratory, Oak Ridge, TN

Sodium Polytungstate (SPT) is a salt solution used predominantly for making buoyancy measurements, but it shares important characteristics with molten salt, FLiNaK, which make it a suitable candidate for simulating molten salts under various conditions. Properties including acoustic impedance, bulk modulus, and density can be tuned to match those of FLiNaK at different operating temperatures. This work explores the ultrasonic behaviours of SPT salt-graphite interfaces using laser-based techniques to understand the wave modes that are supported by the interface under a variety of environmental conditions. The understanding these interfaces will support design of sensing systems for examining the interfaces between graphite and molten salts. In particular, the measurements with simulant will assist with the interpretation of ultrasonic signals associated with molten salt intrusion into nuclear graphite. By exploring these kinds of interfaces, there is the potential to assess intrusion depth of the molten salt into graphite surfaces using ultrasonic techniques under varying environmental conditions.

POSTER**Computational Modelling of Graphite Degradation due to Molten-Salt Infiltration and Wear**

Veerappan Prithivirajan¹, Benjamin Spencer¹, Som Dhulipala¹, Daniel Schwen¹, Joseph Bass²

¹ Idaho National Laboratory, USA

² Nuclear Regulatory Commission, Washington DC, USA

Molten-salt reactors (MSRs) represent promising next-generation designs that utilize graphite as a moderator and reflector. However, the structural integrity of graphite in molten salt environments remain uncertain due to limited data and operational experience. This work evaluates graphite degradation in MSRs through modelling and simulation, focusing on salt infiltration in fuel salt designs and surface wear in pebble-bed reactors.

The first study investigates the role of molten salt infiltration on the structural integrity of graphite. Using Grizzly software, stress induced by internal heat sources due to infiltration is evaluated. Finite element analysis reveals increased stress with higher infiltration, and the parallel subset simulation framework is used to obtain the distribution of inputs corresponding to a failure metric. In the second study, wear mechanisms in pebble bed MSRs are assessed using a multi-physics modelling framework based on Grizzly software. Surface defects are assumed to be present as grooves and pits. The simulations evaluate the structural integrity of graphite with varying sizes and configurations of the surface defects, considering radiation and temperature effects.

POSTER**Computational Modelling of Molten Salt Infiltration and Oxidation in Nuclear Graphite**

Veerappan Prithivirajan¹, Shantanu Vachhani¹, Joshua Kane¹, Krishna Pillai², William Windes¹

¹ Idaho National Laboratory, USA

² University of Wisconsin-Milwaukee, USA

Graphite serves as a critical moderator and structural material in advanced nuclear reactor designs due to its excellent thermal conductivity, neutron moderation capability, and radiation damage resistance. However, its long-term performance is challenged by degradation mechanisms such as molten salt infiltration and oxidation, which can compromise structural integrity and reduce component lifespan. This work presents the development of physics-based

computational models within the MOOSE (Multiphysics Object Oriented Simulation Environment) framework to analyse these degradation processes and support the design and performance evaluation of graphite components in next-generation reactors.

A coupled Navier-Stokes and phase-field framework is developed to model molten salt infiltration into the interconnected pore network of graphite. The model is first verified using benchmark two-phase flow scenarios and then applied to realistic graphite microstructures obtained via X-ray tomography. Simulations reveal how pore-scale features, including porosity, connectivity, and tortuosity, govern infiltration pathways and progression.

For oxidation, the study focuses on pore-scale mass and heat transport behavior, capturing multi-component gas diffusion, reaction kinetics, and thermal effects within graphite's pore structure. These insights help elucidate how localised physical mechanisms contribute to graphite degradation.

Together, these modelling efforts advance predictive capabilities for graphite degradation and provide a foundation for improved material design and qualification in high-temperature nuclear reactors.

- 09h50 **General Discussion on MSR Issues - Leader: Prof James Spicer, Johns Hopkins University, USA**
- 10h10 *Coffee break*

Codes, Standards, Benchmarking

Chair: Mr. Gwennael Beirnaert, MPR Associates

- 10h40 **Graphite Requirements for Reactor Vendors and Graphite Manufacturers for Material and Component Manufacturing**
A. Appleton¹, J.W. Geringer² and Mark Mitchell³
¹ Appleton Quality Concepts
² Oak Ridge National Laboratory
³ Valar Atomics

Various Advance Gas Reactor (AGR) technologies in the United States are in a position where they need to navigate through the requirements presented in ASME Section III Division 5 to qualify, certify and license the design and construction processes of these reactors.

Over the years there has been a lot of emphasis on collecting data to characterize various graphite materials used by designers, but it is not always clear how to apply the rules through an established quality program navigating through the technical complexities presented.

The duties and responsibilities, quality assurance, marking and authorized inspection discussed in Division 5 General Requirements provide guidance to support these activities. This talk will focus on the process stipulated in Subsection HAB and what is expected from designers, graphite and component manufacturers. It will elaborate on the roles and duties of the different Certificate Holders as well as the Authorized Inspectors within the graphite qualification process. Moreover, an overview of the recent findings, code changes and current actions to simplify these processes will be discussed.

11h00

Benchmarking of Irradiated Graphite Material Models for Graphite Core Component Design

D. Joshi¹, FJ van Zanten¹, S. Baylis¹, M. Saitta², A. Young³, C. McInnes³, B. Spencer⁴ and T. Ligon¹

¹ X-energy, LLC, USA

² MPR Associates, USA

³ Frazer-Nash Consultancy Ltd., UK

⁴ Idaho National Laboratory, USA

An independent code-to-code comparison between three nuclear graphite material models: X-energy's IGNIS, Idaho National Laboratory's Grizzly and Frazer-Nash Consultancy's GUMAT is presented. This virtual round-robin test allows benchmarking of independently developed codes, each using independently calibrated constitutive models, building confidence in the tools used for graphite core component design. The comparison considers graphite under fast neutron dose and temperature conditions representative of a hypothetical high temperature gas reactor component. Simulation results from the three codes are compared by probing several quantities such as dimensional change, thermal, and creep strain and principal stresses at the locations of interest as function of fast neutron dose and temperature. The resulting comparison quantifies the agreement between the models for the conditions considered. Identifying the regimes where the codes predict similar responses and quantifying discrepancies between the codes builds confidence in and helps development of these tools for reactor component design and assessment.

11h20

Reliability and Integrity Management of Graphite Components in Advanced Reactors

A. Tzelepi^{1} and M. P. Metcalfe^{†2}*

¹ United Kingdom National Nuclear Laboratory

² Nuclear Graphite Research Group, The University of Manchester, UK

Drawing on the extensive experience in the operation of graphite-moderated gas-cooled reactors in the UK, ASME Section XI is now in the process of being expanded to include supplements for Reliability and Integrity Management (RIM) of non-metallic components. As a first step in what will be a major revision to this part of the code, RIM requirements for graphite components in High Temperature Gas Reactors are being identified. This presentation describes the RIM process and highlights the differing approaches for metallics and non-metallics, the former being highly prescriptive with the latter identifying options that will be design-dependent. The established regulatory framework for operating gas-cooled reactors in the UK and the published draft code for operating the High-Temperature Engineering Test Reactor in Japan are described, with best practices identified. The presentation also highlights the importance of a link to the Design code to ensure that adequate provision is made for monitoring and inspection of components that impact the nuclear safety of the facility over its operational lifetime.

Data Management

Chair: Dr Alina Constantin, IAEA

11h40

Nuclear Data Management and Analysis System for the Idaho National Laboratory Graphite Technology Development and Worldwide Collaboration

C. Otani (This presentation will be given by William Windes, INL)

Idaho National Laboratory, USA

Data is crucial for developing advanced nuclear reactors to meet global nuclear energy goals. The Nuclear Data Management and Analysis System (NDMAS) supports this by managing

data from Idaho National Laboratory's nuclear research programs, including the Graphite Technology Development (GTD) program. NDMAS data scientists create and maintain databases and web tools for analysing and displaying data. For the GTD program, it handles data from the Baseline program and Advanced Graphite Creep and High Dose Graphite experiments, including characterization, irradiation, and post-irradiation examination. NDMAS also produces statistical analyses and reports. A major report produced for each experiment is the Data Qualification Final Report. Data is qualified under an American Society of Mechanical Engineers Nuclear Quality Assurance (NQA-1) compliant program and is tracked within NDMAS. These data and related programme reports are shared via NDMAS' SharePoint and SAS Visual Analytics portals. These portals also reference the International Atomic Energy Agency Nuclear Graphite Knowledge Base to link global nuclear graphite data resources. Together, these datasets can cover graphite material property data with a wide range of irradiation conditions, increase statistical significance, and therefore further the ability for graphite qualification and use in new reactors.

12h00

Updates to the Graphite Microstructure Library: Insights from Historical and Modern Nuclear Grades

J. D. Arregui-Mena¹, C. Contescu¹, P.D. Edmondson^{1,2}, J.B Spicer³, T.J. Gerczak¹ and N. C. Gallego¹

¹ Oak Ridge National Laboratory, TN, USA

² University of Manchester, UK

³ John Hopkins University, USA

Since the development of Chicago Pile I, graphite has been extensively utilized as a neutron moderator in nuclear reactors and is now considered a candidate material for next-generation advanced reactors. The microstructure of graphite plays a crucial role in defining its mechanical and thermo-physical properties, particularly under irradiation. To comprehensively assess graphite microstructure, it is essential to characterize porosity across both macro- and micro-scales, focusing on the morphological connectivity of voids and the two primary phases: filler and binder.

This study presents an updated and expanded version of ORNL's comprehensive library of graphite grades, now incorporating historical and modern nuclear grades, including those designed for molten salt reactor applications. Additionally, the library has been further enriched by the inclusion of pyrolytic carbon extracted from tri-structural isotropic (TRISO) particles, carbon-carbon composites, and glassy carbon. Advanced microscopy techniques were employed to provide new insights into the evolution of graphite microstructures, tracing developments from the early Chicago Pile I to contemporary reactor-grade materials.

The expanded framework established in this work offers a robust methodology for assessing the microstructure of diverse graphite grades, supporting the down-selection of materials for specific applications in graphite-moderated reactor designs and advanced nuclear systems.

12h20

Lunch

Progressing Irradiated Graphite Management - Session 1

Chair: Dr. David Bradbury, Sirius Analysis

This session, and those following on Friday, have been organised in association with Nuclear Restoration Services, UK, and Sirius Analysis, UK, in support of progressing development of technologies for retrieval and treatment of irradiated graphite to allow timely, affordable and sustainable decommissioning of sites in preparation for their next intended use.

13h10 **Progress in Identifying Credible Options for Alternative Retrieval and Processing of the UK Irradiated Graphite Inventory**

R. Harris, I. Haigh, E. Hankinson and E Lancaster

Nuclear Restoration Services, UK, Graphite Innovation Programme

As a result of power generation stretching back more than 60 years, the UK has around 100,000Te of irradiated graphite that will need to be managed to allow decommissioning to be completed. The baseline route for this material is disposal in the planned UK Geological Disposal Facility (GDF), but this will not be available until the latter half of this century. Due to this strategy, a conservative but technically mature approach has been assumed of top-down dismantling with removal of the pilecap followed by extraction of graphite blocks.

Recent work within Nuclear Restoration Services (NRS) has shown that the cost and duration of this Programme is likely to challenge what was envisaged in the early 1990s when this strategy was first adopted. As a result, NRS have established a Graphite Innovation Programme to evaluate alternative technologies that could be used to remove and manage graphite material.

Much of this work has been focussed on a more sustainable approach to decommissioning that identifies opportunities for reduction and recycling of material – options which are both further up the Waste Hierarchy than disposal. Initial outputs of this work will be presented along with an outline plan for how options might be developed further.

13h30 **Regulation of Advanced Nuclear Technologies and Innovation**

G. Black and J. Surman

Environment Agency, UK

The Environment Agency is the lead environmental regulator in England responsible for protecting people and the environment from the impact of disposal of radioactive wastes. The Environment Agency regulates operational and decommissioning sites as well as waste management repositories. We have been working with the UK Government and other regulators, including the Office for Nuclear Regulation (ONR), to build our capability in readiness for the deployment of advanced nuclear technologies (ANT), including High Temperature Gas Reactors (HTGRs) to ensure that we enable new technologies and innovative processes.

We will describe our work to date to build capability and describe how we have improved our processes to ensure that these are fit for purpose and proportionate for new technologies. We will describe our approach to innovation and how we are influencing to ensure impacts on people and the environment from radioactive wastes are minimised. We will highlight regulatory risk for innovative waste management proposals and new technologies including HTGRs.

13h50

The Role of Impurity Concentrations in Predicting AGR Graphite Waste Radiological Inventories

E. Murray¹, A. McNulty¹, S. Shaw², G. Whiley², A. Jones³ and L. Hughes¹

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 including ^{14}C , ^{36}Cl and tritium. 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. MCBEND has been used to perform detailed calculations on 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.

Based on nitrogen, chlorine and cobalt impurity concentrations identified in Anglo Great Lakes data sheets and relevant literature, specific activities were calculated for ^{14}C , ^{36}Cl and ^{60}Co . The activities were then decayed for 85 years to generate predicted activities at FSC. The results highlight the significant role that initial impurity concentrations play in AGR graphite waste radiological inventories.

14h10

HTGR Graphite Waste Management

C. McElvaney, A. Tzelepi

United Kingdom National Nuclear Laboratory (UKNNL)

High Temperature Gas-cooled Reactors (HTGRs) of the prismatic-type use fuel blocks made of graphite as the moderator and main structural component in the core. The UK has extensive experience in operating gas-cooled reactors, and this has resulted in 100,000 tonnes of existing irradiated graphite in the UK. Compared to Advanced Gas Cooled Reactors (AGRs), HTGRs experience a much lower fluence, some graphite components are replaceable and resulting radioactivity levels in the graphite are lower due to the combination of lower fluence with higher purity materials such as IG-110. The radioisotopes of most concern for existing gas-cooled reactors are ^{14}C and potentially ^{36}Cl due to their abundance and long half-life. For the case of the prismatic fuel blocks, short-lived nuclides will also have to be examined.

Due to the lower expected activities, there is a strong incentive to develop a reuse/recycle option. The best solution for the HTGR waste stream depends on a number of factors including disposal cost/tonne, waste processing cost and new graphite cost. This presentation outlines current work including cost assessments, ^{14}C measurements, neutron activation data and GDMS measurements.

14h30

A Compact Multimode Detector for the Decommissioning of Irradiated Graphite

N. Hunter¹, B. Seitz¹, F. Thomson¹, R. Harris², A. Tzelepi³ and M. Morisson²

¹ University of Glasgow

² Nuclear Restoration Services

³ United Kingdom National Nuclear Laboratory

The decommissioning of aging nuclear reactors is a global challenge. The processing and disposal of radioactive material from these reactors is a massive endeavour with far reaching environmental and economic implications. The classification and characterisation of these materials is vital not only for reducing the environmental impact of long-lived radionuclides but also in processing them in a fiscally responsible way. One material of particular interest is graphite used in construction of aging Magnox and AGR reactors. The long-lived radionuclides present in irradiated graphite, particularly ^{14}C and ^{36}Cl , present unique issues in their characterisation, processing, and disposal. A novel scintillation detector for this purpose was developed and tested in a series of experiments at Trawsfynydd, a twin Magnox reactor site undergoing decommissioning works in North Wales. The prototype detectors were deployed in fuel channels from the reactor charge face, and a radiation survey was conducted at a series of fixed points throughout the channel. In this presentation, the properties of the prototype detectors are examined, particularly their efficiency, noise characteristics, and gamma sensitivity. The results from the experiment are presented and evaluated in comparison to a trepanning study which was conducted in July 2022. Preliminary analysis demonstrates that the detector system is capable of measuring beta activity as a function of depth inside the fuel channel and provide a pulse height spectrum at each position.

14h50

Review of Experience and Approaches Providing Graphite Removal from Uranium-Graphite Reactors

A.O. Pavlyuk^{1,2}, E.V. Bepala², S.G. Kotlyarevsky², R.I. Kan², A.E. Rif¹ and E.P. Zelenetskaya¹*

¹ National Research Tomsk Polytechnic University , Russian Federation

² Pilot & Demonstration Center for Decommissioning of Uranium-Graphite Nuclear Reactors, Russian Federation

Note: this will be a pre-recorded presentation

Nowadays, there are more than 100 power and production uranium-graphite reactors in the world. Mainly all uranium-graphite reactors (UGRs) were constructed in the Russian Federation, France, the United Kingdom, and the United States. A few are in Lithuania, Italy, Germany, Japan, China, Spain, Ukraine, North Korea, and Italy. Currently, most graphite-moderated reactors have been shut-down or are in progress toward final shutdown.

The decommissioning option considered for UGRs, except those damaged because of incidents or categorised as nuclear legacy UGRs, is "Dismantlement", which involves the complete dismantling of the reactor structures, including the graphite stack. Due to the complexity of the problem, up to the present the world has accumulated unique practical experience in dismantling only a small part of research reactors: GLEEP, DIORIT, WAGR, BGRR, HTGR and RFT. Note that the dismantling of production and power reactors, unlike research reactors, is a much more complicated task due to the high activity of structural elements and their significant mass and overall sizes. Currently, an important task is the development and testing of technical solutions for the dismantling of production and power reactors.

This work presents a review and analysis of experience and approaches to removing graphite from different types of uranium-graphite reactors, which can be used in the development of reactor dismantling technologies.

15h20 *Coffee break*

15h50 **Advanced Retrieval Techniques for Irradiated Graphite in Reactor Decommissioning: Experimental Results and Scenario Analysis**
R. Chebac^{1,3}, N. Bellingan², F. Vanoni^{1,3}, A. A. Porta³, F. Campi³, O. Gummery² and E. Hocquard²

¹ GraphiCore S.r.l., Italy

² Createc Ltd., UK

³ Politecnico di Milano, Italy

The safe and efficient handling of irradiated graphite remains a significant challenge in nuclear reactor decommissioning. GraphiCore has developed an advanced retrieval system specifically tailored for graphite-moderated reactors. This presentation details recent experimental results highlighting significant improvements in graphite lifting capabilities. The vacuum-based retrieval system demonstrated more than a sevenfold increase in lifting performance compared to traditional vacuum technologies during full-scale tests. The system was evaluated on pristine graphite blocks, axially fractured blocks, and blocks with surface damage.

Additionally, the development of a computer vision algorithm will be presented, which identifies optimal extraction paths for graphite blocks, considering scenarios involving broken or obstructed blocks.

In collaboration with Createc, a comprehensive scenario analysis was conducted to assess realistic decommissioning workflows integrating GraphiCore's technology with various manipulators and cranes. Several practical scenarios will be illustrated, providing an overview of feasible operational strategies, contextualized to specific reactor conditions and chosen decommissioning approaches.

16h10 **Optimising the 'Nibble and Vacuum' Technique to Retrieve and Minimise Graphite Waste**

N. Edwards¹, S. Wilkinson², A. Theodosiou¹, A. Tzelepi² and A. N. Jones¹

¹ The Nuclear Graphite Research Group (NGRG), Department of Mechanical and Aerospace Engineering, The University of Manchester, Henry Royce Institute, Manchester M13 9PL, UK

² The National Nuclear Laboratory, Central Laboratory, Sellafield, CA201PG, UK.

Approximately 100,000 tonnes of irradiated graphite have been produced within the UK, most of which remains stored in commercial reactor buildings. This presents ongoing challenges for decommissioning and long-term waste disposal. Traditional removal methods favour time consuming intact brick extraction to enable encapsulation in concrete for geological burial. However, this process is limited by the reactor core restraining lattice, varied graphite brick geometries, irradiation and corrosion degradation, and core struts & supports interactions.

This project explores developing the Nibble and Vacuum retrieval method to mechanically reduce graphite components *in situ* while simultaneously extracting the graphite particulate via vacuum extraction. The technique has the potential to considerably simplify and speed up retrieval and decommissioning, enable the use of universal tooling solutions, and support integration with certain disposal technologies.

A key focus is the compatibility of the nibbled graphite with Hot Isostatic Pressing (HIP) using borosilicate glass to create a high packing efficiency waste form. Graphite samples were processed using various Forstner bit geometries and rotational speeds, producing particle size distributions (PSDs) that were categorised into three size ranges. These are currently being

evaluated for HIP suitability. The study also considers tooling optimisation, vacuum system performance, and economic trade-offs, with the aim of scaling the technique for operational use. Initial results indicate that Nibble and Vacuum may offer a viable route for graphite waste volume reduction and alignment with long-term disposal strategies.

16h30

Graphite Waste Management

K. Piler, L. Qiu, G. Cota-Sanchez, K. Thaneshwor, I. Dimayuga, C. Song and M. Huang

Canadian Nuclear Laboratories

Graphite components can be found in current gas-cooled reactors and water-cooled reactors, and in next generation reactors such as advanced gas-cooled fast reactors, very high temperature reactors, and molten salt reactors. They are used in moderator, reflector, fuel matrix, fuel channel sleeve, and control-rod. During the lifetime of these reactors, graphite components are exposed to significant neutron & photon radiation doses, resulting in irradiated graphite with damaged structural integrity and internally stored energy known as Wigner energy. Structurally damaged irradiated graphite contains various irradiation-produced radionuclides, such as ^{14}C , ^{36}Cl , ^3H , and ^{59}Ni , which present safety and waste management concerns; specifically, when the irradiated graphite is directed for final disposal - because of their release into the environment.

Canadian Nuclear Laboratories (CNL), in addition to a thorough characterization study of unirradiated and irradiated graphite, are testing and developing various decontamination techniques such as thermal treatment, chemical treatment, and plasma treatment to remove radionuclides from bulk graphite. Atomistic simulations are also being carried out to study irradiation induced damages in the graphite structure. The presentation will discuss some results obtained from these studies and their significance in developing efficient graphite waste management strategies.

16h50

A Short History of the Future Development Options for ^{14}C Reduction and Recovery, and Practical Recycling, in Irradiated Graphite

A.J. Wickham¹ and W. von Lensa²

¹ Nuclear Technology Consultancy, Laugharne, UK

² formerly Forschungszentrum Jülich, Germany

With increasing awareness of the possibilities of thermally or chemically treating irradiated graphite waste in regard to ^{14}C (and other long-lived isotopes) – variously reduction of the waste category and potential recovery of the isotope by taking advantage of the observed preferential release upon heating – it is appropriate to review the archive of work covered by international programmes such as CARBOWASTE and GRAPA alongside the developing understanding of the sources of ^{14}C and its chemical form and behaviour during continued irradiation of the graphite, highlighting apparent ‘discrepancies’ for further consideration. Additionally, the possibility of complete oxidation towards CCUS as a means of disposal alongside other CO_2 sources such as the cement industry prompts a re-evaluation of earlier work such as the FZJ NACOK rig, fluidised-bed oxidation (‘incineration’) and plasma oxidation, and the plans for a joint Franco-American pilot plant.

The objective is to place in context the academic investigations as the industry moves towards potential larger-scale processing to achieve reactor-core dismantling on a shorter-timescale consistent with safety and economic objectives.

17h10

General Discussion on the Session: Leader – Dr Jose-Luis Leganes-Nieto, ENRESA

Friday October 3rd 2025

Progressing Irradiated Graphite Management - Session 2

Chair: Prof. A.N. Jones, The University of Manchester

08h30 The Importance of Collaboration in Applied R&D Across Differing National Strategies

F. Tardy¹, F. Jasserand², and J. Goodwin³

¹ DP2D, EDF SA

² Graphitech

³ Cyclife UK Ltd

EDF owns 6 Gas-Cooled Reactor in France which are currently under dismantling following a dedicated 3 step strategy:

- Development and testing by Graphitech, inside EDF Industrial Demonstrator located near Chinon, of the remote-operated technologies that will be used for the 6 reactors dismantling
- Dismantling of a first-of-a-kind reactor (Chinon A2)
- Dismantling of the 5 other reactors, benefiting from the lessons learnt and improvements coming from the first reactor dismantling

EDF have invested heavily in developing and managing reactor dismantlement activities, a key example of this is the Industrial Demonstrator which continues to be used to support both decommissioning of the UNGG fleet, as well as providing support to other national strategies.

EDF (DP2D & Cyclife Groupe) undertakes significant underpinning studies on graphite including in areas such as characterisation, packaging, reactor dismantling tooling / methodologies, size reduction, treatment and packaging. Much of this work has direct applicability to other national approaches.

Opportunities exist to leverage the skills, experience and infrastructure from across the EDF Groupe (inclusive of the Cyclife subsidiaries) to ensure the maximum value is gained from applied R&D in respect to graphite reactor dismantling with the objective of delivering a sustainable, long-term solution for graphite waste.

09h10 Innovations to Optimise Decommissioning of the Magnox Graphite Reactor at Trawsfynydd

R. Harris¹, D. Cabrera, B. Taylor and A. Thomas

Nuclear Restoration Services, UK

As a result of power generation stretching back more than 60 years, the UK has around 100,000Te of irradiated graphite that will need to be managed to allow decommissioning to be completed. The baseline route for this material is disposal in the planned UK Geological Disposal Facility (GDF), but this will not be available until the latter half of this century. Due to this strategy, a conservative but technically mature approach has been assumed of top-down dismantling with removal of the pile-cap followed by extraction of graphite blocks, then demolition of remaining structures.

Extensive studies have been undertaken at the Nuclear Restoration Services (NRS) Trawsfynydd 'Lead and Learn' site to underpin the technical feasibility of this approach, and a number of potential opportunities have been identified to optimise dismantling of reactors. These opportunities are across the whole decommissioning cycle including retrievals and materials management. Work is planned to combine the favoured options into a Detailed Design to return Trawsfynydd into a suitable site for future use.

This work should be relevant beyond the UK as many other organisations have identified top-down dismantling as their preferred strategy and will face similar challenges in delivering a sustainable and effective programme.

09h30

Danish Experiences with Graphite

Morten Dalby

Dansk Decommissioning, Denmark

A brief overview of the set up for the decommissioning of the nuclear facility at the former research centre Risø will be presented, together with explanation of how Danish Decommissioning as a state-owned company is set up to carry out this task. We have been working since 2003 and expect to be finished 2033-2035.

We will present a review of how we have removed graphite from the three reactors DR 1, DR 2 and DR 3:

- Facility characterization
- by hand tools
- remote-controlled tools
- and reversible packing for intermediate storage

09h50

Progress with DRAGON HTR Decommissioning

R. Harris, H. Bishopp, M. Newland, J. Reed, J. Simmons and T Smith

Nuclear Restoration Services

The DRAGON Reactor based at UKAEA Winfrith in the UK was the experimental reactor of the OECD High Temperature Reactor Project. It was a helium-cooled, graphite moderated reactor that was operational from 1965 until 1976. This trans-European initiative involved 13 countries as part of an international effort to study the irradiation of fuel and fuel elements for future HTRs.

Packaging of the reactor fuel was undertaken at the end of the 1990s and subsequently, designs have been developed to remove the reactor and over-building as part of the programme of work to return the Winfrith site to a natural state. This paper describes the methodology proposed for core segmentation and waste packaging in line with the current baseline of disposal in the proposed UK Geological Disposal Facility (GDF).

Challenges associated with the reactor are explored along with innovations developed by NRS to optimise the decommissioning activities, particularly the remote deployment of laser cutting equipment for non-graphitic components inside the reactor. Details of selected containers for waste packaging will be discussed (6m³ concrete container), including optimal packing of the graphite shielding blocks from the reactor and opportunities for alternative treatments.

10h10

Strategic Roadmap for Decommissioning of Graphite Reactors

T. Kilochytska¹, G. Migliore², M. Pieraccini³, R. Harris⁴, S. Boniface⁵, A. Mereznikov⁶ and P. O'Sullivan¹

¹ International Atomic Energy Agency, Austria

² SOGIN, Italy

³ EDF, France

⁴ Nuclear Restoration Services, UK

⁵ Nuclear Decommissioning Authority, UK

⁶ Ignalina Nuclear Power Plant, Lithuania

Graphite has been used as a moderator and reflector of neutrons in nuclear reactors since the beginning of the nuclear energy era, particularly in reactor designs which used natural uranium as fuel or uranium with low levels of enrichment. Historically, a deferred dismantling strategy

was widely adopted for graphite reactors that had been permanently shut down, though national policies are increasingly tending to favour an immediate dismantling approach in line with sustainability considerations. The drivers for the historic approach include the lack of a settled view on the most appropriate long-term process for management of irradiated graphite, of which there is currently about 250 000 tonnes in existence worldwide, and a lack of appropriate disposal facilities.

The very extensive cores needed in large graphite moderated reactors — each of which may contain 3 000 tonnes or more of irradiated graphite — and the associated very large and complex reactor systems and structures is also an important consideration in most programmes. Despite the constraints to decommissioning graphite reactors, many Member States are currently actively considering how to move forward with retrieval of graphite from reactor cores and the subsequent dismantling of the reactors. To support this process a number of States with significant inventories of irradiated graphite have been assisting the IAEA Secretariat in the development of a strategic roadmap outlining the main steps to be undertaken in decommissioning reactors containing graphite and the associated sequencing of the activities involved.

The work undertaken to develop the strategic roadmap followed several other international collaborative initiatives which took place over the past two decades — facilitated by the IAEA and also by the European Commission — which focussed particularly on options for management of the irradiated graphite, in particular, the International Project on Irradiated Graphite Processing Approaches (GRAPA). The roadmap presents the major tasks and activities to be considered in defining the decommissioning strategy of graphite reactors, identifies the interlinkages between the tasks and describes the overarching aspects that should be taken into account for planning associated activities such as for definition of decommissioning strategy.

10h30

Disposal of Irradiated Graphite by Oxidation to Carbon Dioxide and Co-Sequestration Underground with Industrial Carbon Dioxide

C. W. Forsberg¹, A. J. Wickham,^{2,3} and D. Bradbury³

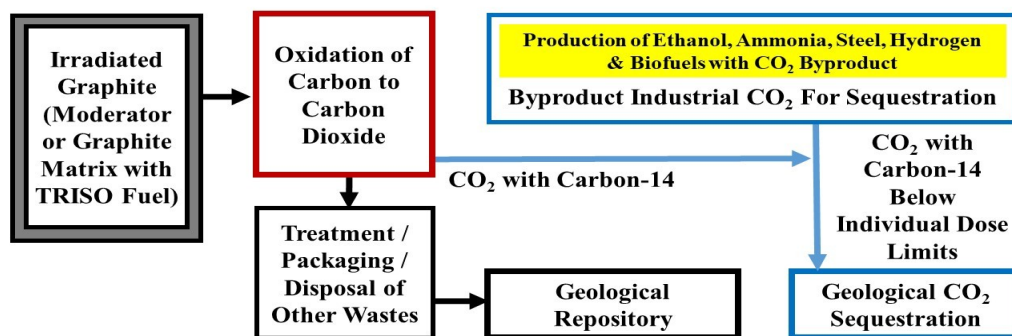
¹ Massachusetts Institute of Technology, Cambridge, MA. USA (cforsber@mit.edu)

² University of Manchester (UK) Nuclear Graphite Research Group

³ Sirius Analysis

Most high-temperature reactors use graphite as a moderator and structural material. This includes HTGRs and FHRs—both with TRi-structural ISOtropic (TRISO) fuel particles embedded in graphite. Thermal-spectrum MSRs have graphite moderator and fuel dissolved in the salt. Irradiated graphite impurities (fission products, actinides, TRISO particles and ¹⁴C) imply much of this waste will require expensive geological disposal. This is the largest volume radioactive waste stream from these reactors.

We conclude¹ that the low-cost lowest-impact disposal option for irradiated graphite will likely be oxidation to carbon dioxide (CO₂) and co-sequestering with other CO₂. Management of graphite by oxidation and sequestration is compatible with recovery of useful values from the graphite, particularly ¹⁴C, ³H and salt with ⁷Li from salt-cooled reactors. There has been considerable recent progress at small industrial scale with recovery and concentration of these isotopes from waste materials to provide useful products. CO₂ capture from dilute gas streams is expensive but underground sequestration is inexpensive. It is unclear whether CCS of dilute CO₂ from power plants or air will ever be economic. Co-sequestration offers multiple advantages. CO₂ disposal costs are 10s of dollars per ton. Any ¹⁴C from irradiated graphite upon mixing with large-scale industrial CO₂ is diluted to below individual radiation dose limits. Sequestration provides long-term geological isolation. The low cost may make it the preferred option for most irradiated graphite.



¹C. Forsberg, Roadmap of Graphite Moderator and Graphite-Matrix TRISO-Fuel Management Options, *Nuclear Technology*, **210**, 1623-1638, September 2024.
<https://www.tandfonline.com/doi/full/10.1080/00295450.2024.2337311>

10h50

Coffee break

11h20

Strategy and Disposal Feasibility of Irradiated Graphite from Vandellós I

J.L. Leganes Nieto, E. Gimeno Blesa and C. Correa Sainz

Empresa Nacional de Residuos Radiactivos, ENRESA, Spain

The decommissioning of the Vandellós I UNGG reactor in Tarragona has generated a significant volume of irradiated graphite waste (i-graphite), posing unique challenges in terms of treatment, conditioning, and final disposal. Spain's national strategy under development focuses on an integrated management approach, including detailed characterization, optimized treatment processes, and the potential disposal of this waste at the El Cabril low and intermediate-level waste repository.

Several treatment pathways were explored. Graphite was successfully incorporated as aggregate into cementitious and geopolymer matrices. Laboratory-scale trials demonstrated acceptable mechanical properties (compressive strength >10 MPa) for mixtures containing up to 59% graphite in mortars and 50% in concretes, depending on grain size and use of superplasticizers. Thermal treatment using rotary and catalytic furnaces was also studied, although it revealed limited release of ¹⁴C due to its firm incorporation in the graphite lattice. An advanced option consisting of fabricating Impermeable Graphite Matrices (IGM) via hot pressing with 80% i-graphite and 20% borosilicate glass. This method significantly reduces porosity and volume while achieving negligible leaching rates.

Graphite retrieval from the reactor's moderator pile requires customized methods. Techniques under consideration include mechanical, vacuum-based, and hybrid systems to extract individual bricks or to crush and vacuum graphite in situ. The choice of method depends on the intended downstream treatment and must ensure safety and operational feasibility in confined spaces.

I-graphite is classified as low-level, long-lived waste (LLW-LL), typically requiring intermediate-depth disposal. However, safety assessments for surface disposal at El Cabril indicate that limited volumes (approx. 2,000 m³) could be safely accommodated.

Radiological impact assessments, particularly for the groundwater pathway, show that in both reference and worst-case scenarios, the projected doses remain well below regulatory thresholds. El Cabril's radiological capacity for ¹⁴C comfortably exceeds the inventory from Vandellós I graphite in compliance with Waste Acceptance Criteria (WAC).

11h40

Impact of Waste Form on Disposal Performance of Irradiated Graphite Waste

L. Hines and H. Wainwright

Massachusetts Institute of Technology, USA

The deployment of Generation IV graphite-moderated reactor technologies, including High Temperature Gas Reactors and Fluoride-cooled High-temperature Reactors, is expected to produce thousands of tons of irradiated graphite waste. The disposal of irradiated graphite requires a performance assessment to quantify potential health risks and to evaluate disposal conditions and engineering designs. The most prevalent radionuclides, including ^{14}C , ^3H , and ^{36}Cl , are highly mobile in groundwater under standard modelling guidelines. However, graphite itself can remain stable under geologic timescales, up to billions of years. It is important to quantify how the graphite waste form could significantly limit the release of these highly mobile radionuclides in irradiated graphite, crediting its robustness.

This work examines the impact of leaching and oxidation in the performance of disposed irradiated graphite. A performance assessment developed for the study is based on the Energy Solutions Low Level waste repository in Clive, Utah. Our results show that crediting the graphite waste form can reduce predicted offsite groundwater concentrations of ^{14}C , the most prevalent radionuclide, by a factor of 40, from 200,000 pCi/L to 5,000 pCi/L. The lower predicted concentration approaches the Groundwater Protection Limit of 3,200 pCi/L, opening the possibility of disposing of irradiated graphite as a Class A Low Level radioactive waste by the US Nuclear Regulatory Commission standard.

12h00

Plan for the Pilot-Scale Implementation of a Laboratory-Tested Process for Producing Concrete and Geopolymers Incorporating Graphite

R. P. de Castro¹, M^a Asunción Díaz¹, M^a E. Gimeno² and J. L. Leganés².

¹ Westinghouse Electric Spain

² ENRESA – Empresa Nacional de Residuos Radiactivos, Spain

Vandellòs I Nuclear Facility, a Spanish graphite-moderated reactor, has remained in the Dormancy stage since completing Level 2 decommissioning in 2003. The design phase for Level 3 decommissioning is now underway, focusing on remaining structures, systems, and components —specifically, the reactor internals and the prestressed concrete vessel. A major challenge is managing the facility's inventory of approximately 4,000 metric tons of irradiated graphite.

In parallel with the planning of dismantling operations for the reactor internals, an implementation plan has been developed to scale up previously conducted laboratory tests. These tests involve using crushed graphite as an aggregate in mixtures designed to produce concrete and geopolymers. The ultimate purpose of these mixtures is to serve as filling material for the voids within waste packages containing other types of radiological waste, thereby replacing the use of conventional “clean” fillers.

A successful outcome from the planned pilot-scale implementation would form the design basis for industrial-scale deployment, ultimately contributing to the effective volume optimization of the final storage of this complex waste stream. However, the challenge extends beyond industrial feasibility to include the approval of regulatory and operational updates required for the final disposal of graphite-containing waste packages at the Spanish facility El Cabril.

12h20

Decontamination of Irradiated Graphite through Electrochemical Treatments

Louis Pointreau and Raluca O. Scarlat

Department of Nuclear Engineering, University of California, Berkeley, USA

The management of irradiated graphite needs to consider the presence of long-lived isotopes such as ^{14}C , shorter-lived species like ^{60}Co , and trace amounts of fission products and actinides. This study investigates the removal of these isotopes to downgrade the category of waste and enable accelerated disposal.

This work investigates the decontamination of irradiated graphite through electrochemical treatments in FLiBe (LiF-BeF_2). Two treatment approaches are explored. The first aims to selectively oxidize ^{14}C over ^{12}C by applying a constant anodic potential to a graphite electrode in oxide-saturated FLiBe. Current results suggest that graphite oxidation occurs in FLiBe, but at reduced rates.

The second treatment targets the removal of metallic contaminants such as europium and cobalt by applying varying potentials in FLiBe. Preliminary results show a significant decrease, up to 33% for ^{60}Co , in the specific activities of gamma-emitting isotopes, with an overall 10% reduction in graphite-specific activity achieved after one hour of treatment. Ongoing work focuses on optimizing the electrochemical parameters and evaluating the method's effectiveness on a broader range of irradiated graphite samples.

12h40

Vitrification as an Option to Immobilise Irradiated Graphite

K. Xu¹ and M. I. Ojovan^{1,2}

¹ State Key Laboratory of Silicate Materials for Architectures, Wuhan University of Technology, Wuhan, P.R. China

² School of Chemical, Materials and Biological Engineering, The University of Sheffield, Sheffield, UK

Glasses are solid amorphous materials which transform to liquid state (melt) above the glass transition temperature (T_g) including the families of silicate and phosphate glasses which were internationally selected to immobilise both high level [1] and low and intermediate level nuclear waste [2] due to their enhanced stability, high T_g , exceptional high chemical and mechanical durability in the environment, and radiation durability [3]. A small fraction of irradiated graphite is significantly contaminated with long-lived actinides and hence needs to be immobilised using most durable wasteforms and technologies available where vitrification is a potential option [4-6]. Between the almost fully glass and fully crystalline matrices lie glass composite (crystalline) materials (GCM) [7]. GCM can be used to immobilize long-lived radionuclides (such as actinide species) by incorporating them into the more durable crystalline phases, whereas the short-lived radionuclides are accommodated in the vitreous phase. The GCM option is currently being considered in many countries including Australia, France, Russia, South Korea, UK and USA. The processing, compositions, phase assemblages and microstructures of GCM are tailored to achieve the necessary material properties. Uses of GCM included: (i) Glass ceramics where a glassy wasteform is crystallized in a separate heat treatment, (ii) GCM in which a refractory waste is encapsulated in glass, (iii) GCM formed by pressureless sintering of spent ion exchangers with glass powder, (iv) Difficult wastes such as the HLW with U-Mo compounds, (v) GCM with "yellow phase" retaining up to 15 vol.% of sulphates, chlorides and molybdates, and (vi) GCMs with ashes from incineration of solid radioactive wastes. A wide range of technological approaches available for vitrification to produce both homogeneous glasses and GCM which was recently overviewed by J. McCloy et al [2] demonstrates flexibility of this technology and covers units capable to immobilise both fine fractions of irradiated graphite and larger size fragments.

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7. C.M. Jantzen, M.I. Ojovan. On selection of matrix (wasteform) material for higher activity nuclear waste immobilisation (Review). *Russ. J. Inorg. Chem.*, **64**, 1611–1624 (2019).

13h00 **General Discussion on Innovative Approaches to I-Graphite Management - Leader: Dr. R. Harris, NRS (UK)**

13h20 *Lunch*

Progressing Irradiated Graphite Management - Session 3

Chair: Dr. Richard Harris, Nuclear Restoration Services,

UK

14h10 **Study on the Surface Enrichment Distribution of Carbon-14 in the Graphite of the Heavy Water Research Reactor Thermal Column**

Xue-Chun Cui and Xin-Xin Zhu

China Institute of Atomic Energy, Beijing 102413, China

This study investigates the distribution characteristics of carbon-14 (one of the key challenges in spent graphite treatment) in the spent graphite from China's first heavy water research reactor (HWRR-the 101 reactor). Through stratified sampling and surface characterization techniques (SEM, Raman, EDS), the distribution pattern of ¹⁴C in the graphite was systematically analysed. The results reveal a distinct deposition layer on the outer surface of the 101 reactor's thermal column graphite, with ¹⁴C exhibiting surface enrichment. This finding provides an important basis for targeted spent graphite treatment. The D and G peak characteristics of Raman spectroscopy, combined with EDS elemental analysis, effectively distinguish the graphite matrix from the surface deposition layer. SEM observations indicate that the deposition layer is primarily distributed within the graphite pores, leading to increased thickness of the pore walls' flaky structure and blunted edges. The study also shows that samples directly exposed to air are more prone to forming a deposition layer, while the internal matrix, which relies on air diffusion through pores, exhibits lower deposition but remains detectable. Additionally, the deposition layer formation is significantly positively correlated with neutron fluence. These findings reveal carbon-14 surface enrichment and support safe spent graphite management and recycling.

14h30 **Progress on the US Integrated Research Project on I-Graphite Waste Minimization**

L. Snead¹, H.M. Wainwright², D. Sprouster¹, K. Shirvan², P.F. Peterson³, R. Scarlet³, J.A. Mena⁴, C. Tan⁵ and C.W. Forsberg²

¹Stony Brook University, USA

²Massachusetts Institute of Technology, USA

³University of California, Berkeley, USA

⁴Oak Ridge National Laboratory, TN, USA

⁵Idaho National Laboratory, USA

This presentation provides an update and results from the US multi-institutional program on developing irradiated-graphite waste reduction, mitigation, and disposal strategies, resulting in a specific primary objective of quantitative cost savings. In this program, we consider a wide range of technologies – with a combined modelling and experimental effort – in three broad areas: (1) the volume/activity reduction at the source, (2) the decontamination and recycle as an alternative to disposal, and (3) the scientific basis and regulatory framework for cost-effective and risk-informed disposal. In Area 1 we perform an in-depth characterization to develop predictive understanding of how radionuclides are created, moves, and can potentially be operationally mitigated through impurity reduction. In Area 2, we develop mitigation technologies to limit the impact of i-graphite prior to disposal such as salt decontamination and waste volume reduction through direct recycle or repurposing i-graphite as an alternative to disposal. In Area 3, we review the current low-level waste disposal regulations across the world, as well as develop a comprehensive framework to evaluate the i-graphite disposal pathways, by combining reactor physics simulations and repository performance assessment (PA) models. These new findings and technologies will be integrated into the i-graphite waste roadmap, enabling vendors, operators and regulators to evaluate multiple options for most cost-effective management strategies.

14h50

Characterisation of Irradiated Reactor Graphite from Research Reactors by using Destructive and Non-Destructive Measurement Methods

L. Meunier¹, N. Heiss¹, L. Lens¹, D. Reischwich¹, M. Knebel² and U.W. Scherer¹

¹ Technische Hochschule Mannheim, Institut für Physikalische Chemie und Radiochemie, 68163 Mannheim, Baden-Württemberg, Germany

² JEN Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH, Dekontamination und Entsorgung - Verfahrensentwicklung und Produktkontrolle Verfahrensentwicklung, 52428 Jülich, North Rhine-Westphalia, Germany

Germany holds less than 1% of the global inventory of irradiated graphite (i-graphite) [1]. Due to the activation of the matrix and its impurities, i-graphite is potentially radioactive. Several challenges arise when considering its final disposal in Germany, due to the lack of information on the radionuclide inventory as well as the strict limits for certain radionuclides. While Monte Carlo simulations can offer initial inventory estimates, experimental characterization remains necessary.

To identify and quantify radionuclides in i-graphite, both non-destructive and destructive analytical techniques are used. In order to ensure representative sampling, the spatial distribution of beta- and gamma-emitting radionuclides is determined by electronic autoradiography. The gamma-ray emitters are characterised by gamma-ray spectrometry for safety purpose, since they are responsible for the dose rate. The pure beta emitters are characterised by LSC using a destructive sample preparation technique.

The results of the study give new insights on the spatial distribution of the activated impurities in i-graphite and help to improve already existing simulations to be more accurate. The methods developed have been proven robust and effective when used on samples of varying geometry and radiation history, originating from the TRIGA reactor in Mainz and the FRJ-2 reactor in Jülich.

Reference

[1] IAEA, *Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal*, 2016, TEDCOC Nb 1790, Vienna

15h10

Radiological and Mechanical Characterization of FRJ-2 I-Graphite

M. Knebel¹, B. Haxhiu¹, H. Keller², S. Kramer¹ and T. Steinhardt-Knips¹

¹ Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH (JEN), Germany

² Wissenschaftlich-Technische Ingenieurberatung GmbH (WTI), Germany

FRJ-2 is a DIDO-type MTR currently undergoing decommissioning and dismantling after 40 years of operation. The FRJ-2 has a reflector consisting of individual blocks made from ARS graphite.

Various samples from recent years exhibit a high level of activation making the FRJ-2 i-graphite an intermediate level waste. Moreover, they show that the graphite is locally to be considered a chemically and, thus, radiologically inhomogeneous material.

With the help of the obtained samples, a 3D MCNP/ORIGEN activation model for the graphite reflector has been validated. The activation model produced an estimate for the minimum nitrogen content in the graphite pores of 50 ppm. Furthermore, the radiation damage in the graphite has reached a maximum of 1.2E-3 dpa.

Knowledge of the mechanical properties of i-graphite is necessary for the safe handling of the graphite blocks during dismantling. There is no reliable data on the ARS graphite under FRJ-2 operating conditions. However, with the help of transferable irradiation tests on comparable graphite grades, it is plausible that the mechanical properties of ARS graphite have not changed significantly due to years of irradiation.

15h30

Coffee break

16h00

Quantification of Radiochlorine Exhalations from Reactor Graphite using Short-Lived ³⁸Cl as a Proxy for ³⁶Cl

D. Boya¹, A. Wickham², H-J. Steinmetz³, M. Grimm⁴, E. Langeegger⁵, M. Knebel⁶ and G. Steinhauser¹

¹ TU Wien, Austria

² Nuclear Technology Consultancy, Laugharne, UK

³ FH Aachen, Germany

⁴ DMT Group, Germany

⁵ JUNO Nuclear Engineering & Consulting GmbH, Germany

⁶ Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH (JEN), Germany

As part of the characterisation of irradiated reactor graphite from the secondary reflector of the FRJ-2 research reactor (Jülich, Germany), a procedure is being developed to quantify the exhalation of long-lived ³⁶Cl using ³⁸Cl as a comparatively short-lived tracer. Samples from this reactor graphite are irradiated with neutrons at the TRIGA Mark II reactor in Vienna to activate ³⁷Cl via an (n,γ) reaction to form ³⁸Cl (T_{1/2} = 38 m). Heating experiments are conducted to thermally induce a release of ³⁸Cl, which is then collected under controlled conditions in a washing flask with an alkaline solution. The exhalations are later quantified *via* gamma spectroscopy. Due to the identical chemical behaviour of ³⁶Cl and ³⁸Cl, this approach allows extrapolation of long-term ³⁶Cl exhalation rates under “fire scenarios” in a final repository.

16h20

Decontamination of Irradiated Reactor Graphite by using Thermal and Supercritical-Fluid Extraction

N. Heiss^{1}, L. Meunier¹, L. Lens¹, B. Abdolla¹, S. Goedert¹, M. Knebel² and U.W. Scherer¹*

¹ Technische Hochschule Mannheim, Institut für Physikalische Chemie und Radiochemie, 68163 Mannheim, Baden-Württemberg, Germany

² JEN Jülicher Entsorgungsgesellschaft für Nuklearanlagen mbH, Dekontamination und Entsorgung - Verfahrensentwicklung und Produktkontrolle Verfahrensentwicklung, 52428 Jülich, North Rhine-Westphalia, Germany

Graphite is widely used in nuclear reactors as a moderator, reflector, or structural material. In Germany, this applies solely to research reactors. Neutron activation of trace impurities results in the formation of radionuclides such as ⁶⁰Co, ^{152/154}Eu, ¹⁴C and ³H. Among these, ¹⁴C and ³H pose particular challenges for final disposal due to their long half-lives and/or high volatility.

In Germany, the KONRAD repository is foreseen to accept the irradiated graphite (i-graphite) from these reactors. Some estimates indicate that the total estimated ¹⁴C inventory contained in the i-graphite from research reactors could consume up to 90% of the ¹⁴C capacity of KONRAD, if it were disposed without prior conditioning. This makes effective decontamination of irradiated graphite essential.

This work focuses on developing practical removal methods of radionuclides from i-graphite. Two approaches are investigated experimentally: an optimized thermal treatment and supercritical fluid extraction (SCFE). The talk will present both techniques, along with analytical methods to determine decontamination factors – key to evaluating the efficacy of these methods.

The objective is the development of a process to decontaminate samples to a level at which they are no longer classified as radioactive waste, or to reduce contamination to a degree that allows the reuse in future reactor designs, such as Small Modular Reactors (SMRs).

16h40

Filtration of Graphite for Decommissioning

M. Sauder

Krantz, Germany

Graphite plays a crucial role in nuclear applications, serving as a moderator, structural material, reflector, and fuel matrix. In Germany, irradiated graphite is classified as low and medium-level waste, posing significant challenges for decommissioning processes. This trial focuses on the filtration of graphite to address these challenges.

Background to this research is The Windscale Pile 1 project, that faces challenges due to unknown graphite dust. This potentially causes issues with cleanliness, air quality, and equipment performance. A 2008 concept was effective but costly. RHF filters are suitable but lack OPEX and performance data on graphite.

The primary goal of the graphite filtration trials was to evaluate the effectiveness of a recleanable filtration system and to determine the possibility of its application in decommissioning activities.

The experimental setup included a recleanable system (RHF-1500) equipped with a graphite dosing device, designed to facilitate the controlled introduction of different graphite materials into the filtration system. The results demonstrated the system's capability to effectively filter graphite, thereby reducing the volume of waste and enhancing the safety and efficiency of decommissioning operations.

These findings underscore the potential of advanced filtration technologies in managing irradiated graphite, contributing to more sustainable and cost-effective decommissioning strategies.

17h00 **Waste-Management Wrap-Up - Leader: Mr. R. Harris, NRS (UK)**

END OF CONFERENCE – we look forward to meeting up again in 2026!

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INGSM-25 PROGRAMME AT A GLANCE

DAY AND TIME	TECHNICAL	ADMINISTRATION
SUNDAY 28TH SEPTEMBER		
17h30 onwards		Welcome Reception 'Family and Friends'
MONDAY 29TH SEPTEMBER		IAEA Press Room, M Building
09h00 – 12h20	Welcome, Invited Lecture, Knowledge Transfer	10h50 – 11h20: Coffee Break
12h20 – 14h30	Support to Operating Reactors	13h00 – 13h50: Lunch Break
14h30 – 17h00	Oxidation	15h50 – 16h20: Coffee Break
17h00	Discussion on First Day	
TUESDAY 30TH SEPTEMBER		IAEA Press Room
08h30 – 12h20	Irradiation Behaviour and Related Research	10h30 – 11h00 Coffee Break
12h20 – 12h50	Discussion	12h50 – 13h40 Lunch Break
13h40 – 17h10	Mechanical and Physical Properties	15h20 – 15h50 Coffee Break
17h10	Discussion	
WEDNESDAY 1ST OCTOBER		IAEA Press Room
08h30 – 11h40	Graphite for SMR	10h30 - 11h00 Coffee Break
11h40 – 12h00	Discussion	
12h00 – 13h00	Irradiation Creep	13h00 – 14h00 Lunch Break
14h00 – 15h20	TRISO	
15h20	Information about INGS-26	
16h45 AT THE LATEST	Conference Tour/Dinner/Concert	Visitor Arrival Center, Schönbrunn (via subway U4)
THURSDAY 2ND OCTOBER		IAEA Press Room
08h30 – 10h10	Molten Salt Reactor Issues	10h10 – 10h40: Coffee Break
10h40 – 11h40	Codes, Standards, Benchmarking	
11h40 – 12h20	Data Management	12h20 – 13h10: Lunch Break
13h10 – 17h10	Progress in Irradiated Graphite Management (1)	15h20 – 15h50: Coffee Break
17h10	Discussion	
FRIDAY 3RD OCTOBER		
08h30 – 13h00	Progress in Irradiated Graphite Management (2)	10h50 – 11h20: Coffee Break
13h00 – 13h20	Discussion	13h20 – 14h10: Lunch Break
14h10 – 17h00	Progress in Irradiated Graphite Management (3)	15h30 – 16h00: Coffee Break
17h00	Discussion and conference wind-up	

