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Damage tolerance in the graphite cores of UK power reactors and implications for new build

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ABSTRACT

The United Kingdom has been operating graphite-moderated gas-cooled commercial nuclear power reactors for more than 60 years. Both Generation I Magnox reactors and Generation II Advanced Gas-cooled Reactors have operated beyond their nominal design lifetimes and the safety cases for continued operation have had to address the issue of damage tolerance of the graphite cores. This review describes the designs of the reactors and the methodologies employed for their assessment. Emerging issues arising from extended operation and ageing of the cores are discussed together with the evolution of assessment methodologies to address predicted and observed damage. The UK nuclear regulator perspective is summarised and some conclusions are drawn on the implications for new build.

1. Introduction

The United Kingdom has been operating commercial nuclear power reactors since 1956, when the world's first commercial power-producing reactor was commissioned, up to the present day. The Generation I Magnox-type plant have all now ceased power production but the Generation II Advanced Gas-cooled Reactors (AGRs) continue to operate. Whilst significant advances have been made since then in the design of new graphite-moderated gas-cooled reactors, the experience accumulated and the challenges faced by both the UK operators and the UK nuclear regulator provide vital insights into lifetime management and nuclear safety when compiling design codes for new plant. In the case of the UK designs, the graphite cores were regarded as one of the life-limiting features of the plant and the approach to component integrity and the tolerance to failures has been and continues to be an extensively researched and critical issue.

The objective of this paper is to describe the UK approach to damage tolerance in graphite cores. To provide context, the Magnox and AGR core designs are first described. Approaches to damage tolerance are reactor design dependent and there is no generic statement that can be made about acceptable levels of damage. The methodologies for predicting the behaviour of components and arrays of bricks are then described. Emerging issues for both reactor designs and safety case approaches are discussed. The UK nuclear regulator perspective in terms of nuclear safety and damage tolerance is presented and finally some observations are presented on the implications for new build.

In providing this assessment of damage tolerance, it must be appreciated that much of the documentation around this topic remains restricted. In preparing this review, the author has cited published work and recovered some documentation through the UK Freedom of Information Act, but also has had to refer back to his own experience over 40 years working in the UK nuclear industry working principally on graphite core safety case management. So, while some information presented cannot be verified against published and peer reviewed documents, the issues and opinions raised may provide a useful focus for designers and assessors of new nuclear plant containing graphite components.

2. Design of Magnox and AGR cores

The descriptions provided in this section have been extracted in part from an earlier document, compiled by the author (Banford et al., 2009) and references therein (Hart et al., 1972), (Davies, 1996), (Steer, 2007). Nonboel (1996) also provides a comprehensive description of AGR reactor design.

2.1. Magnox core design

Only a limited amount of graphite property data was available at the time that the first power-producing UK Magnox reactors were designed. In particular, the understanding of anisotropic dimensional change behaviour of graphite subjected to fast neutron irradiation was based

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upon a relatively small number of materials testing reactor experiments. Consequently, as more information was gained, improvements to reactor core design were made. However, the same basic design principles were maintained through the evolution of the plants, comprising eleven paired reactors. No two reactor pairs were identical, presenting complex and varied approaches to plant safety and operational life.

The entire fleet of Magnox reactors was graphite-moderated and carbon dioxide cooled with natural unenriched uranium fuel elements contained within Magnox alloy cans. Vertical channels carried the coolant gas from the bottom to the top of the core over the fuel elements, sometimes referred to as a ‘once through’ design. A schematic showing the general arrangement of a Magnox reactor is shown in Fig. 1. While it is beyond the scope of this paper to provide details of the design evolution, a broad description is provided to highlight the most important features of the graphite cores.

The primary graphite used for the construction of the Magnox reactor cores was Pile Grade A (PGA) graphite. The graphite produced for these cores was based upon a ‘needle’ coke, this term referring to the needle like appearance of the grains after crushing. The basic shape of the graphite blocks was produced by an extrusion process, which had the effect of preferentially aligning the grains and hence the crystallite basal planes in a direction parallel to the direction of extrusion. As a result, the properties of the graphite are anisotropic (or more specifically orthotropic, meaning that properties in the plane perpendicular to the extrusion direction are isotropic). A less pure graphite produced in an earlier stage of the same manufacturing process (Pile Grade B) was used for reflector components (side reflector and upper and lower reflector layers, depending upon design).

In the earliest Magnox reactors, Calder Hall and Chapelcross, the cores were made up of layers of square cross-section bricks of side-dimension slightly smaller than the lattice pitch of 203 mm, with the extrusion direction along the brick height. Details of the Calder Hall core design can be found in Hart et al., 1972, including an isometric view of the Calder Hall reactor. The bricks were approximately 800 mm in length, separated in a column by two layers of rectangular tiles. The extrusion direction for each tile was aligned horizontally, parallel to the long sides of the tile. Pairs of tiles were orthogonally aligned; the long side of the upper tile being rotated 90° to the long side of the lower tile (Fig. 2). The bricks and tiles in each column were located relative to one another by means of cruciform keys on the upper and lower faces. This resulted in moderator bricks being located in their correct lattice position by the tiles. Because they had sides which were slightly shorter than the lattice pitch, a gap around their vertical sides accommodated the predicted lifetime growth of the graphite.

Although allowances were made for the different expansion/shrinkage rates between tiles and bricks and between parallel and perpendicular expansion to the extrusion axis, it emerged later that PGA graphite actually shrank in the direction parallel to the extrusion axis. This would have the effect of opening up gaps between the tiles, permitting lateral movement of individual columns, which could impact cooling and control rod insertion. At the time that this behaviour became apparent, the manufacture of graphite bricks for Berkeley, Bradwell and Hinkley Point A cores was at an advanced stage. To address lateral alignment of brick columns, core designs were modified to incorporate zirconium pins in deep horizontal holes drilled into the bricks and/or tiles linking components. Zirconium has a low neutron absorption cross-section and the small reactivity reduction caused by its use was deemed acceptable. The arrangement adopted for Bradwell, in which the zirconium pins were located in the bricks, is illustrated in Fig. 2.

The Trawsfynydd design, the next member of the Magnox fleet, was less advanced and the keying between adjacent columns with the maintenance of column geometries was achieved by including a single tile at each brick junction incorporating radial keys on the 90°and 45° axes. After Trawsfynydd, the brick and tile method of maintaining channel lattice alignment was changed in favour of a radial keying arrangement between bricks with the elimination of tiles as illustrated in Figs. 3 and 4, the latter showing the Oldbury core at the time of construction.

The radially-keyed core design was adopted for the twin-reactors at Sizewell A, Dungeness A, Oldbury and Wylfa power stations. All designs maintained the same lattice pitch. Sizewell A, and Dungeness A cores comprised 12 graphite layers (including top and bottom reflector layers), approximately 3800 fuel channels and an operating gas pressure

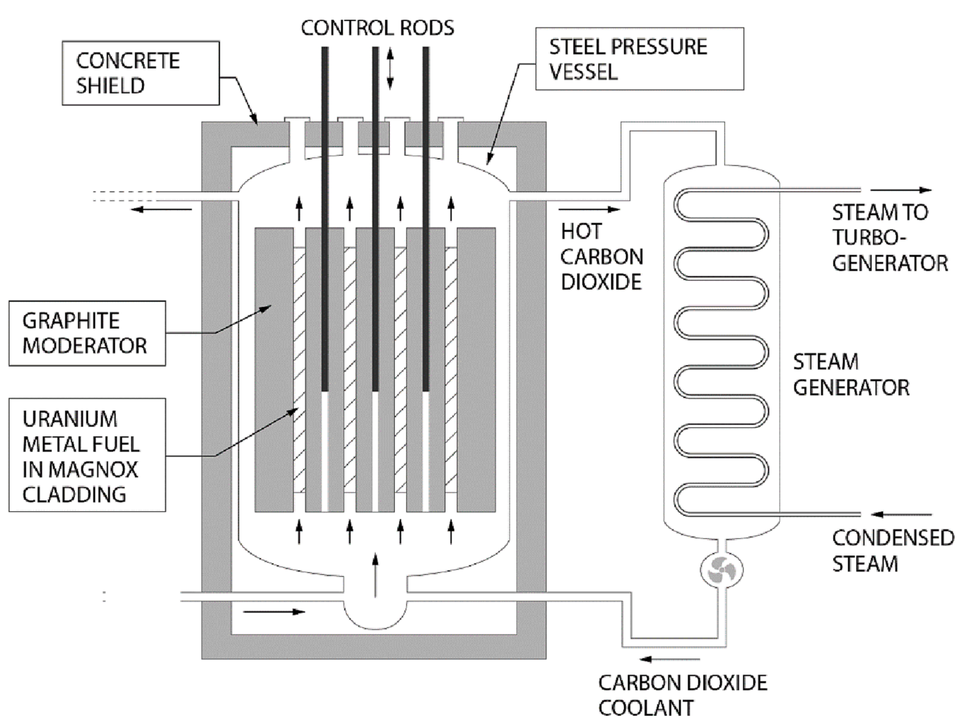

Fig. 1. General design arrangement for a Magnox-type reactor.

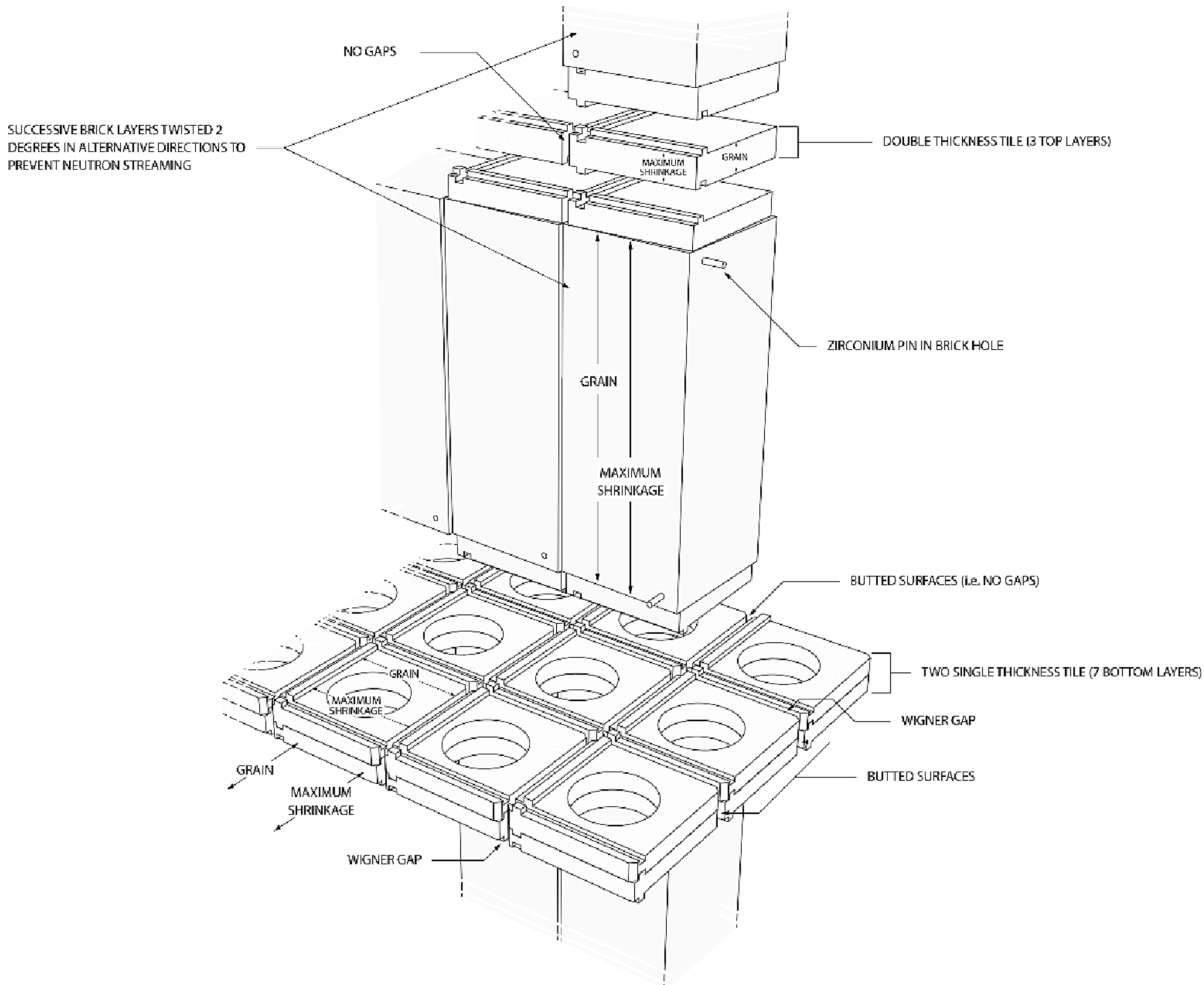

Fig. 2. Arrangement of bricks and tiles in Bradwell graphite core.

of approximately 1.8 MPa. The Oldbury core also had 12 graphite layers, a slightly reduced number of 3308 fuel channels and an increased operating pressure of 2.56 MPa. The final design in the evolution of the Magnox stations was Wylfa, with a significantly larger core comprising 13 layers of bricks, 6156 columns and over 80,000 graphite bricks and an operating pressure similar to that of Oldbury at 2.62 MPa.

Finally, it is worth briefly mentioning the design of the steel core restraint cages used in the design of the Magnox reactors to maintain geometric stability of the graphite structures. In the case of the early brick and tile core designs, compensating beams placed around the periphery at the same height as the tile layers were designed to constrain the structure such that the lattice as a whole would expand as graphite in the long tile direction (in effect radially). The beam arrangement was designed to have a coefficient of thermal expansion close to that of graphite thereby providing the necessary restraining forces at the core boundary. Furthermore, by mounting the columns on ball-bearings, the core could expand at a different rate relative to its steel support structure. This design concept (where the core restraint structure expands as graphite) was changed as the core design evolved from brick and tile arrays to bricks with a radial keying system. These later designs employed a graphite core and side restraint system where it was the thermal expansion of the steel restraint cage that maintained vertical columns of bricks when the reactor was at power. This was achieved by engineering axially varying radially inward offsets on the cage when cold that became straight and vertical at power. This revised core restraint cage design was challenged later in reactor operation when gas outlet temperatures were reduced to bring mild steel oxidation rates within acceptable limits, with the consequence of a cage not fully expanding as designed.

2.2. Magnox core design lifetimes

The author has previously had access to unpublished Magnox reactor design documentation and core case histories. These were never prescriptive about core lifetimes, rather making reference to lifetimes for planning purposes of 20–25 calendar years. These lifetimes were based upon rudimentary (by today's standards) stress analyses and assessments of core component changes due to fast neutron irradiation and radiolytic oxidation. It was recognised that these notional lifetimes were very conservative given the intentional robustness of core component design. Because of the perceived large margins on graphite component integrity and the significant economic benefits of lifetime extension, considerable effort over several decades was devoted to improving understanding of the behaviour of the cores. This behaviour, combined

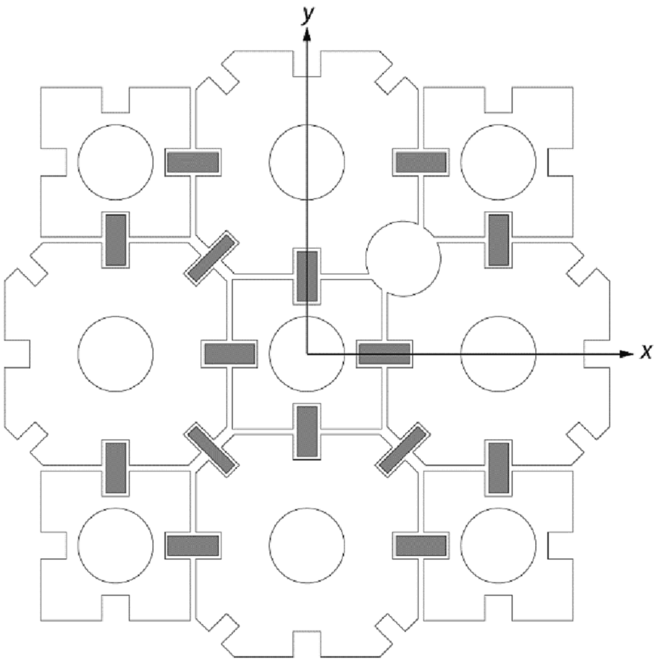

Fig. 3. Radial keying arrangements in later Magnox core design (octagon and square bricks).

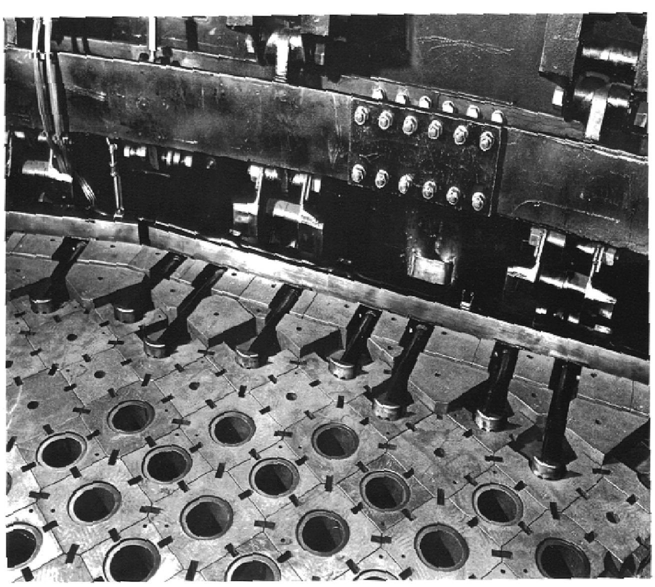

Fig. 4. View of the Oldbury reactor core during construction showing the radially-keyed octagon-square brick arrangement (Banford et al., 2009).

with outputs from predictive models and data from core monitoring and inspections, was assessed against the design function and nuclear safety requirements of the cores to maximise plant lifetimes.

In fact, all Magnox stations were required by the nuclear regulator to undergo 'long-term safety reviews' in order to have licenses granted for continued operation beyond 30 calendar years.

Lifetime extension of the Magnox reactors was remarkably successful with seven of the eleven reactor pairs operating for periods of 40 calendar years or more. Of the other four reactor pairs, three operated for 26–27 calendar years and one for 34 calendar years. This group of four stations ceased operation not because of degradation of the graphite cores but because of issues associated with mild steel oxidation. For the other seven reactor pairs, the ageing of the graphite cores featured prominently in safety assessments for continued reactor operation.

2.3. AGR core design

The AGR reactor design that evolved from the Magnox reactor design (a general arrangement of the reactor design is shown in Fig. 5) achieved higher power densities by increasing channel gas outlet temperatures and reactor gas pressure. The AGR fleet comprises seven pairs of reactors. Like the Magnox stations, no single design was adopted for the entire fleet. The first AGR to be constructed, Dungeness B, evolved from the prototype Windscale AGR (WAGR) and designs for the remainder of the fleet evolved from this. The remaining six stations formed three sets of twins, each station pair having broadly identical designs.

The higher power densities achieved with the AGRs were engineered in part through a 're-entrant flow' design that maintained graphite and steel support structure temperatures within acceptable limits. Gas circulators draw cooled gas from the bottom of the boilers and discharge it into the space below the core. Around half this cooled gas flow passes directly to the fuel channel inlets, with the remainder, known as the re-entrant flow, being drawn up the annulus surrounding the core then returning downwards through the core within annuli between the bricks and the graphite sleeves surrounding the fuel as well as through arrowhead passages between the bricks. The re-entrant flow then rejoins the main coolant flow at the bottom of the fuel channels to flow upwards over the fuel within the sleeves. This re-entrant flow arrangement cools the graphite bricks, the core steel restraint system and the gas baffle. The graphite sleeves, which form part of the replaceable fuel element assembly, shield the graphite bricks from the hotter gas which transfers the heat from the fuel.

This design concept could be accommodated by a smaller graphite structure compared to a Magnox reactor core (typically 11 m diameter, 9.8 m in height for an AGR compared with 18.6 m diameter, 10.3 m in height for the final Magnox design Wylfa). An AGR core typically comprises an inner cylinder of moderator graphite (in 10 or 11 layers) surrounded (top, bottom and side) by reflector graphite (Fig. 6). The larger bricks, which contain the fuel, are approximately circular in plan with a diameter of about 460 mm and a height of about 850–900 mm. They are arranged in columns on a square lattice. The small bricks are about 190 mm square, arranged in columns in interstitial positions (Fig. 6 and Fig. 7). The re-entrant flow design is illustrated in Fig. 8.

Unlike the Magnox reactor fuel, AGRs have fuel assemblies comprising a cluster of 36 stainless steel fuel pins containing slightly enriched uranium (2–3%).

Two types of graphite are used in AGR cores, noting that there were three suppliers (BAEL, UCAR, AGL) with methods of manufacture and the graphite grades changing over time. The permanent components (moderator bricks, filler bricks and keys) are made from vibro-moulded Gilsocarbon graphite, which uses natural Gilsonite coke as filler

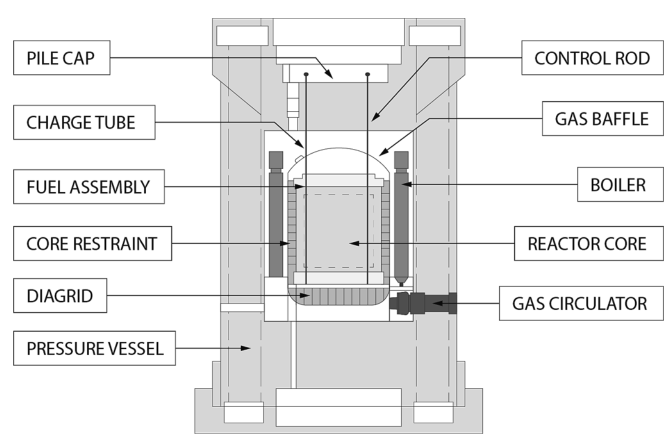

Fig. 5. General design arrangement for an AGR-type reactor.

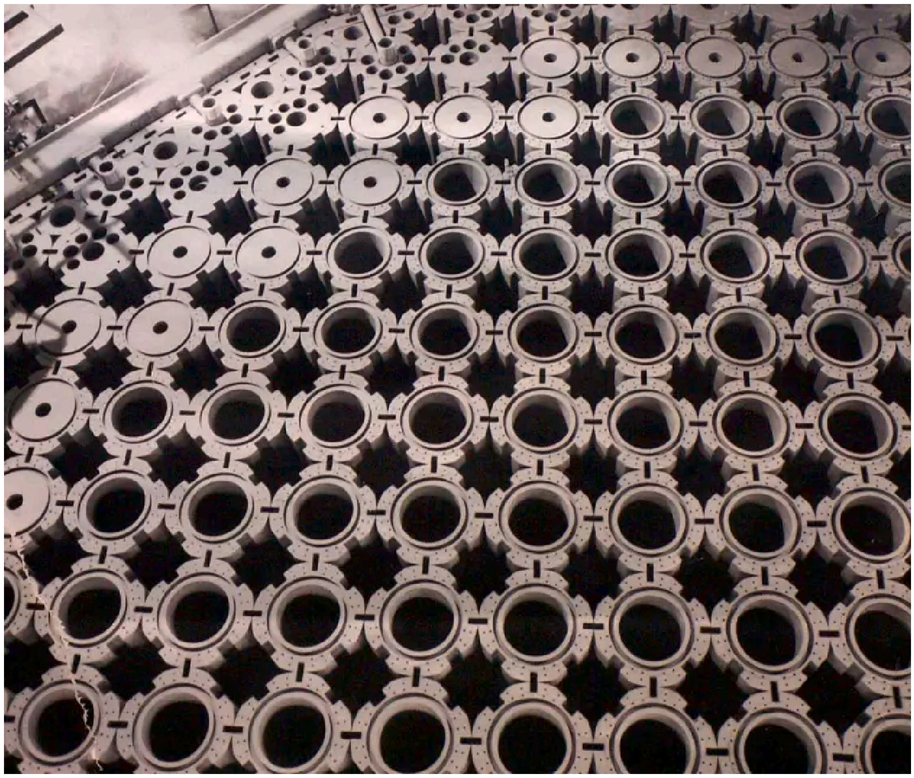

Fig. 6. Side reflector arrangement in AGR core design. Interstitial bricks and keys illustrated in Fig. 7 are below and out of sight (https:/www.edfenergy.com/graphite).

particles and flour (fine particle coke) impregnated with pitch to produce near-isotropic properties. The moderator bricks use graphite that is double-impregnated, while the reflector and shielding bricks use graphite that is not necessarily the same grade or have the same number of impregnations. The removable fuel assembles contain graphite fuel sleeves made using needle coke particles double-impregnated with pitch to produce slightly denser, near-isotropic graphite.

While the designs for the first AGR Dungeness B and the subsequent station pairs differ, the progression in design is not as significant as that for the Magnox fleet. An explanation of detailed design differences between the AGRs is not necessary for this review of damage tolerance.

As with the later generation of Magnox reactor designs, the AGR cores sit within a steel core restraint cage with an axially varying inward radial offset when cold that thermally expands to verticality when the reactor is at power.

2.4. AGR core design lifetimes

In contrast to the Magnox stations, where the earliest reactors accumulated data, experience of graphite and a deeper understanding of core behaviour while the plant was in service, AGR designers could build on this knowledge base such that cores could tolerate higher temperatures, higher levels of fast neutron irradiation, and radiolytic oxidation could be better controlled using more exotic carbon-dioxide based coolants and gas conditioning plant.

The top-level criteria for a licensable design were that the cores should act as a moderator of fast neutrons and provide a stable geometric structure which would provide passages in which fuel could be freely loaded and unloaded, control rods could be freely inserted and removed, and the coolant gas flow was directed over the fuel to remove heat, as well as being capable of supporting their own weight. It was recognised that there would be distortions due to dimensional changes resulting from fast neutron damage; the cores would experience corrosive gasification due to radiolytic reactions with coolant; core components would lose strength as a consequence of oxidation, leading to an increasing possibility of fracture due to externally applied loads. Fracture could also arise from excessive internal stresses generated by differential rates of dimensional change under irradiation, and by differential thermal strains, within individual components. A reduction in thermal conductivity leading to increased temperature gradients could, under certain conditions, accelerate one or more of the above modes of deterioration. Although not a damage tolerance-related issue, the accumulation of stored energy from irradiation damage leading to the possibility of a subsequent uncontrolled release was also recognised (but not considered to be an issue for higher temperature AGR cores compared to Magnox cores).

It is noteworthy that only the final station pair of Torness and Heysham 2 were designed in accordance with specific seismic criteria. In the earlier designs, the core and diagrid (intersecting steel beam support structure) were supported from the bottom of the pressure containment. In the case of Torness and Heysham 2, the diagrid and core were suspended from the top of the boiler shield wall and the possibility of fracture during a seismic event was reduced by strengthening linkages between adjacent fuel bricks using thicker keys. However, for all designs, the intent was that fracture of core components would not occur during normal operation. In the case of a seismic event, it was acknowledged that cracking could occur but that the reactors would safely shut down and hold down safely. Continued operation after the event would be dependent upon an assessment of the condition of the cores and whether they continued to meet the safety criteria. Designers of the AGR fleet estimated that their reactors would operate for at least 30 calendar years, with an accounting life of 25 calendar years. As with the Magnox stations, there are significant financial and national strategic benefits from life extension. A combination of research, core monitoring and inspection continues to support the continued operation of the AGR fleet. The first AGR station to cease operation was Dungeness B in 2021, with Hunterston B closing in early 2022 after ~ 44 years supplying power to the national grid. While the station operators believe

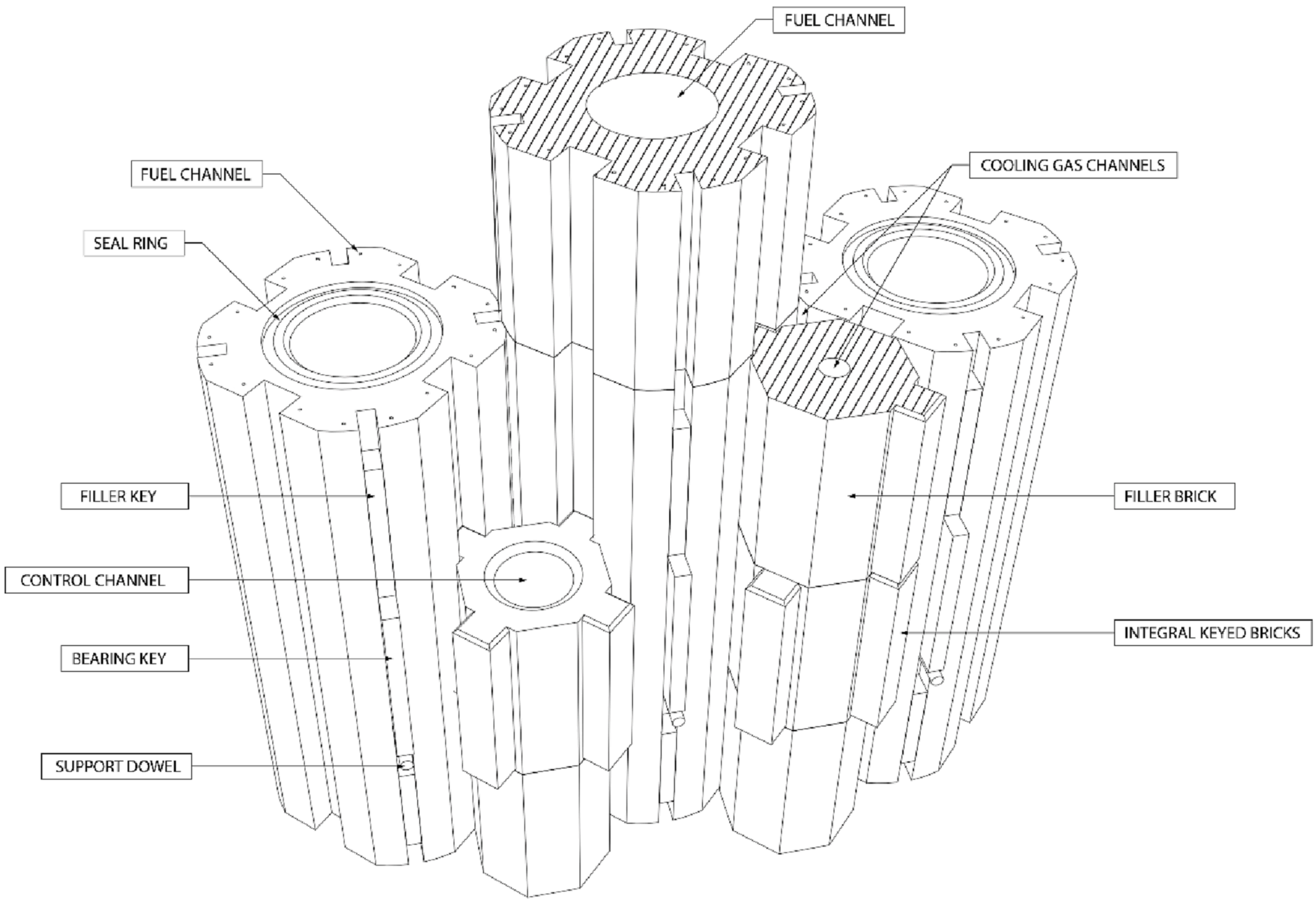

Fig. 7. Schematic of AGR brick keying arrangement.

they can make a satisfactory safety case for further operation beyond this date with cracked graphite bricks, the increasing complexity of such cases and the timescales for securing regulatory approval have led them to reach this decision. The final stations in the AGR fleet are expected to cease generation around 2028, although these dates are continually under review.

3. Methodologies for predicting/assessing the integrity of core components

Assessment methodologies are complex and continually evolving. However, the design of AGR cores was a natural progression from Magnox reactor core designs, and the fundamental nuclear safety requirements for the two reactor types are the same:

- To allow the unimpeded movement of control rods and fuel
- To direct the flow of coolant gas so as to ensure adequate cooling of the fuel and the core structure both in normal and under fault conditions

Both safety requirements require the core geometries and specifically fuel and control rod channels to remain adequately straight and free from significant discontinuities and obstructions. The principles behind all methodologies are broadly the same, namely that continued safe operation requires the ability to manage reactivity for 'shut down' and long term 'hold down' purposes. In addition, the integrity of the fuel must be maintained so as to prevent contamination of the core and gas circuits and to minimise the potential for a radioactive release in the event of a depressurisation. It is important to note that these requirements apply specifically to current UK gas-cooled reactor designs. While these may be broadly typical for most gas-cooled reactor types, it will be the responsibility of the designer to define the requirements for their specific design.

The management of Magnox graphite cores, which includes core integrity safety cases, has been described in Ellis and Staples, 2007 and an overview of the AGR methodology is provided in Bradford, 2007. Neighbour (2010) provides a review of AGR plant life management. While these papers are not recent, the principles and safety requirements remain unchanged.

In the case of Magnox cores, Ellis and Staples, 2007 outlines a multi-legged safety case comprising:

- Inspection leg – forewarning of severe cracking
- Monitoring leg – deterioration of the graphite and impact on the overall condition
- Structural assessment leg
- Consequences leg – consideration of postulated brick cracking and keyway pinching

The inspection and monitoring legs included temperature monitoring, channel geometry measurements, visual inspections and graphite sampling. The consequences leg is discussed separately below.

The structural integrity leg comprised three elements: assessment of the changes either predicted or measured in physical properties of the graphite (including weight loss, the reduction in material density due to radiolytic oxidation), calculation of the various components of loads and stresses arising in the graphite as a result of individual internal and external mechanisms, and assessment of the integrity against a suitable

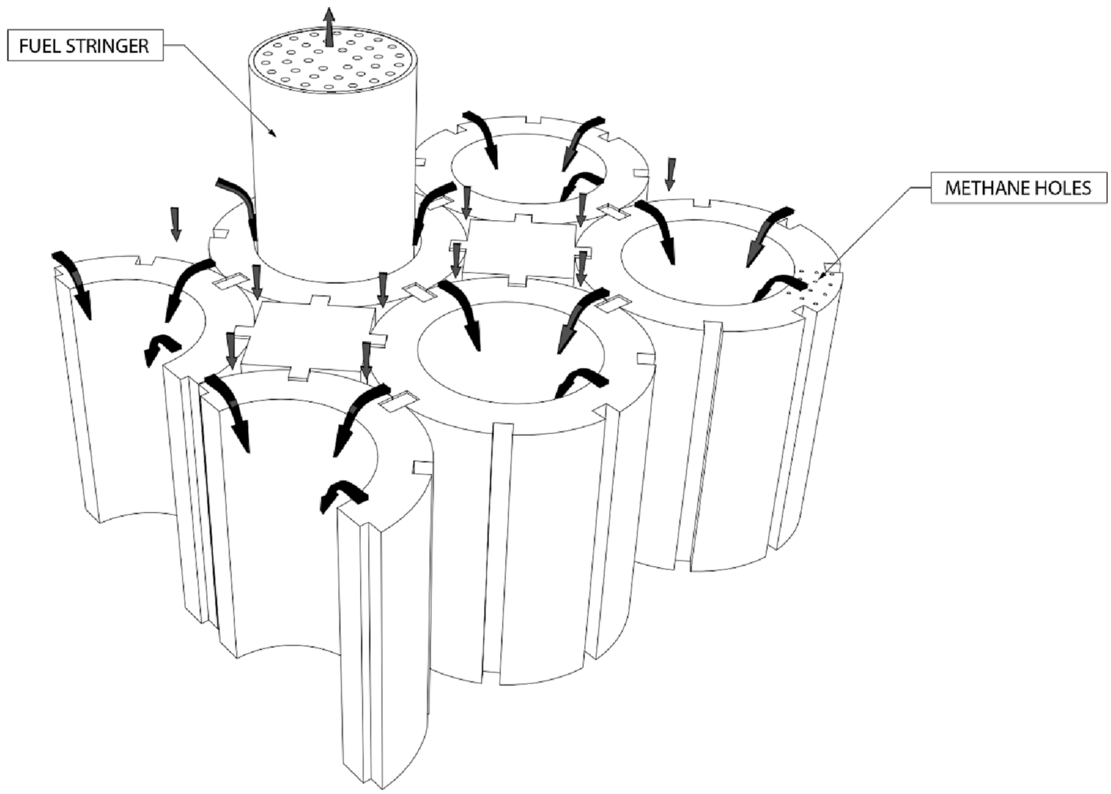

Fig. 8. Re-entrant flow arrangement in an AGR core showing upward gas flow within fuel stringer and downward flow between the stringer and moderator bricks and within brick gaps.

failure criterion.

It is important to appreciate that at the time the first Magnox reactors commenced generation (1956), data on the effects of fast neutron irradiation and radiolytic oxidation on graphite properties were sparse. The concept of a materials database covering the projected lifetime of the cores was not considered a prerequisite for operation nor had there been the time or facilities to generate such a database. As noted above, the effects of fast neutron irradiation on the dimensional change of PGA graphite were not adequately researched leading to major changes to later core design. The extent to which graphite would undergo radiolytic oxidation was also poorly understood and more importantly the behaviour of defects and the potential for the release of Wigner energy had not been adequately characterised. From the late 1950 s, the United Kingdom Atomic Energy Authority embarked on a programme of accelerated materials testing to address these issues and to establish a graphite database which formed the basis for assessment of core component behaviour. The operator of the Magnox fleet also undertook an extensive research programme to characterise radiolytic oxidation from which it developed oxidation models for predicting mass loss. To validate the use of this database and oxidation models, the physical properties of the in-service graphite were monitored by periodically withdrawing pre-characterised graphite test specimens designed for specific property measurements, installed in the cores during construction. Following initial operation of the reactors, it was recognised that these 'installed specimens' were insufficient in number to provide adequate regular property monitoring and database validation. Furthermore, their location at off-lattice positions away from the fuel channels limited the information they could provide on the condition of bricks at peak dose positions. Therefore, trepanning devices were introduced that could sample fuel channel walls in temporarily defueled channels during statutory maintenance outages. While this method of property monitoring significantly improved the knowledge of the condition of the cores, there were inherent uncertainties in property changes due to graphite manufacturing variabilities and the necessity to assume 'representative' as-manufactured properties. It was not possible to assign manufacturers' heat certificate data (where it existed), test data information for manufacturing batches, to specific components in the cores: the original nuclear programme within the UK was both civil and military in nature, details of graphite supply were classified and records for the early reactors were sparse; the production of PGA graphite at three production sites ceased with the closure of all three manufacturing companies, exacerbating any subsequent attempts to compile a database. To summarise, graphite properties from Materials Testing Reactor programmes and predictive models validated against core sampling provided the data employed in component and core assessments to support safety cases for continued operation of the Magnox fleet.

Calculation of loads and stresses developed significantly in the level of detail and sophistication over the years. Computational whole core models quantified component external loads and stresses under normal operation and postulated fault conditions. Finite element models quantified internal stresses in components generated by temperature transients, dimensional change and within-brick distributions of properties. Component integrity based upon these combined internal and external loads and stresses was assessed deterministically against a brick failure criterion. The failure criterion was defined from mechanical testing to failure of whole bricks and cross-sectional brick slices prepared from unirradiated as-manufactured PGA graphite. Probabilistic methods of assessment were introduced in parallel, in later years.

The AGR assessment route is structured in terms of 'Aspect', 'Methods' and 'Data'. 'Methods' forms the basis of the safety case, 'Aspect' comprises material behaviour, component behaviour, core behaviour and consequences and 'Data' comprises data trending, rig tests, inspection and monitoring. This structure broadly aligns with that adopted by the Magnox reactor licensees. It is important to recognise that the emphasis within these methodologies can change with the accumulation of new operational data and improved methods of predicting material, component and core behaviours. For example, McLachlan (1995) presented the position on AGR core methodologies back in 1995 and how it was speculated that the assessment approach would need to change. Up to this time, it had been predicted that core components would remain intact over the lifetime of the cores. The deterministic criterion used in the design safety cases for assessing core integrity was the comparison of the strength of a graphite component against the internal stresses and stress generated by external loads. However, evolving methods for quantifying component stresses were beginning to show that component failure, by definition, could not be dismissed. Other factors such as excessive radiolytic weight loss and component distortions due to differential irradiation-induced dimensional change could also impact on component integrity and core geometry. Furthermore, a more considered approach to statistical variability inevitably requires assessors to accept and consider the consequences of component failure. As a result, it was proposed in McLachlan et al. (1996) that the assessment methodology should shift from core design to core functionality. In other words, could the cores continue to meet their nuclear safety requirements (effectively functionality) with failed components present? By ‘core design’, assessment methodologies quantified component stresses to demonstrate that failure would not occur over the period of assessment. This contrasted with the Magnox approach where the potential presence of brick failures was viewed with much more serious concern, as will be discussed below in the emerging issues section. It also raised many new questions that the AGR methodology needed to address. How do components fail, where in the core do they fail, what level of failures are expected and what level of failures can be tolerated? These issues have dominated AGR core assessors subsequently, leading to improvements in the interpretation of material behaviour (and associated uncertainties), refinements to stress analyses, more detailed assessment of operating environments (monitoring) and fault conditions, increased and more targeted levels of inspection and the adoption of probabilistic methods of assessment. As will be discussed in the emerging issues section below, there has been a further change in emphasis in AGR core assessments. This has arisen because the level of confidence in predictions of component and core behaviour has been undermined by instances of poor correlation between prediction and observation. If predictions of the core condition do not align with inspection, how meaningful are functionality assessments based upon upper limits on failures derived using numerical methods that may not adequately define uncertainties and may not correctly represent all relevant physical processes? Is the modelling of material and component behaviour too unreliable to form the basis of safety cases for continued operation? While such studies are justified as part of a holistic approach to AGR core assessment, it has been necessary over recent years to refocus analysis towards more extreme and bounding core states using probabilistic methodologies.

4. Emerging issues with graphite cores and safety case approaches to damage tolerance

4.1. When is a crack not a crack?

Before reviewing the nature of and approaches to damage tolerance in UK nuclear reactor graphite cores, it is very important to understand the morphology of as-manufactured graphite and its behaviour under irradiation and oxidation. It is impossible to discuss damage tolerance in graphite structures without some appreciation of graphite micro- and nano-structure and how this relates to stress-induced crack initiation in core components.

The graphite manufacturing process has been extensively documented (see for example historical Kelly, 1981; Kelly, 1981 and more recently Sang-Min Lee et al., 2015) and is not detailed here. The morphology of the final product is determined principally by the blend and nature of coke particles and binder, and the final stage of manufacture involving graphitisation at a temperature range between 2600 °C and 3000 °C. This results in a material with a very high defect population within the nano- and microstructure spanning multiple length scales. The raw materials and the method for forming the final product have a major influence on the graphite properties and its behaviour in-service. In the case of the graphites used for the UK reactor graphite cores, the Magnox reactors used PGA graphite and the AGRs used Gilsocarbon graphite. PGA graphite contains needle-like coke particles and the green articles (first blend of coke and binder) were formed by extrusion. This resulted in an orthotropic material with the extrusion axis along the length of the bricks. The near-isotropic properties of Gilsocarbon graphite arise from the use of coarse-milled Gilsonite coke grains consisting of small contiguous crystallites that are misaligned to form circumferential patterns within the material. These differences lead to markedly contrasting component behaviours that need to be carefully modelled to predict in-service stress development and potential crack initiation and propagation. What is common to both graphite types is a large and varied defect population with similar characteristics. During calcination at ~ 1200 °C, weak bonding between basal planes causes cracks within the coke particles and generates porosity. The release of volatile gases within the binder phase during this manufacturing phase also generates larger porosity. More importantly, during the final high temperature graphitisation and subsequent cooling, small cracks are produced within both the coke particles and binder phase due to anisotropic contraction between basal planes within the matrix. These cracks, termed Mrozowski cracks (Mrozowski, 1956, Freeman et al., 2016), can typically have widths in the tens to hundreds of nanometres size range with lengths hundreds of times larger. They play a significant role in matrix dimensional change and thermal expansion behaviour. All these features continue to be the subject of investigation and characterisation as techniques for their examination develop, most recently in Liu et al. (2017).

In the context of damage tolerance, can it be claimed that these defects, microcracks and porosity can be regarded as inherent damage and therefore tolerable by virtue of the choice of graphite as a structural moderator material? Or is it misleading to describe them as cracks? Can these features be compared or grouped with cracks that propagate through the matrix terminating at component surfaces and produce loose fragments or fractured bricks? In fact, these features are often initiators of crack growth leading to such failure but, without some stress or strain driver, they are benign.

It is important to note that during the construction of both the Magnox and AGR cores, graphite components were visually inspected and accepted (or rejected) for assembly based upon a list of allowable material defects. These included cracks, flowlines (striations), soft spongy areas and porosity, hard areas of porous graphite, blow holes (voids), chips, gouges and contamination (such as with metallic foreign objects). The criteria also included tolerances on allowable size and location of individual defects plus tolerances on multiple defects within the same brick. The 100% visual inspection prior to assembly was claimed to provide powerful evidence in mitigation against surface breaking flaws in the graphite bricks. In the case of AGR bricks, it has been reported that only 0.2% of bricks had significant flaws relating to damage, heavily dominated by chips, gouges etc. rather than cracks. Non-destructive techniques for detecting sub-surface defects were not available at the time. However, it is worth noting that some AGR bricks were tested to verify their quality (Preston, 1989).

In the case of UK graphite core safety case management, these stable defects, porosity and small cracks are never regarded as cracks when assessing component damage. Furthermore, technical discussions about cracking strictly adhere to an accepted terminology that excludes them in a component damage context, to avoid misunderstanding. As will be discussed below, even large surface defects or cracks that are stable and are known not to run through the brick ligament can be regarded as benign and can be excluded from damage statistics. But this presupposes that the crack mechanism is known and the morphology of the feature has been correctly characterised.

4.2. What is meant by failure?

Just as a clear common understanding is needed of the term 'crack' in the context of damage tolerance, it is equally important to be careful with terminology associated with graphite components and core structures where damage has been initiated. Some texts describe various states of cracking in components as 'failure' while others refer to 'component integrity' and 'structural integrity'.

Mechanical strength tests, where the applied load falls sharply before the test specimen actually breaks, are regarded as having been loaded to failure. In contrast, although there are no strict guidelines on the use of such terms when describing graphite components, failure in this context should be regarded as 'loss of design function'. In the case of UK graphite cores, design function comprises two elements, (i) moderation of neutrons to sustain nuclear chain reaction and (ii) providing a structural framework for flow of gas coolant, location of fuel, control rods and instrumentation and unimpeded movement of fuel and control rods. It is the latter function that may be affected by damage to core components. A graphite component may continue to fulfil its design function even with extensive cracking, distortion or loss of fragments and therefore has not failed. The same may be true for an assembly of components where individual component integrity may be challenged by the presence of damage but the assembly itself may continue to perform its design function. It is good practice to qualify the meaning of all these terms when addressing damage and damage tolerance in reactor graphite cores.

4.3. Magnox reactor cores

Over roughly the first twenty years of operational life, the Magnox fleet safety cases for continued operation focused principally on mild steel oxidation issues and the risks of a breach of gas pressure circuits. Little attention was given to the graphite cores themselves which were regarded as being over-engineered and benign. However, it was standard practice periodically to monitor graphite properties and core geometry. In these 'early' years, core geometry monitoring involved lowering a device containing tilt and channel measurement instrumentation down a small selection of temporarily defueled channels during a statutory maintenance outage. Such measurements provided evidence to support arguments for unimpeded control rod entry and free movement of fuel. However, the continuous traces, as the device was pulled out of the channel, could also provide information on movement between bricks and potentially displacements or obstructions that could be associated with brick failure as well as dimensional change data.

As part of the preparation for Long Term Safety Reviews (safety cases for continued operation beyond nominal design lifetimes) and also with the availability of improved analytical and numerical methods for quantifying internally and externally generated stresses in core components, a programme of structural integrity assessments for each of the Magnox reactor cores was undertaken. These assessments covered both normal operation and fault events such as a rapid depressurisation. They also took account of more sophisticated predictive models of radiolytic oxidation (caused by the interaction of gamma irradiation and carbon dioxide coolant with the graphite pores) and the associated property profiles within a brick. What became apparent from these assessments was that component cracking or 'failure' could not be dismissed as the cores aged and that certain regions of the cores were more vulnerable to such events. It was recognised that these integrity assessments contained uncertainties as well as built in conservatisms and that predictions of brick conditions could be both inaccurate and overly pessimistic. It was therefore decided in 1997 that remote visual inspections of channels should be undertaken to test the representativeness of predictions at one of the first stations scheduled for a Long-Term Safety Review (Hinkley Point A). The 100% inspection of two entire fuel channels using a downwards facing camera with circular reflecting mirror showed an absence of any anomalies. This exercise became the blueprint for regular, but limited in number, visual inspections at all Magnox stations at statutory outages from 1998 onwards. While some very minor damage at brick interfaces was observed in a very small number of inspections, there was no evidence for brick cracking, highlighting the very conservative nature (or incomplete modelling) of component integrity assessments at the time.

Notwithstanding the encouraging findings from these inspections, the finite probability that bricks will crack at the tails of distributions and the consequences of cracking must be assessed as part of any programme aiming to maximise plant lifetime.

The earliest brick and tile design reactors only exhibited moderate levels of radiolytic oxidation and ceased generation due to factors other than those relating to degradation and potential cracking of graphite components. One incident of brick cracking was discovered during routine inspections of the upper region of the core. The cracking in this top reflector brick was attributed to stresses exerted by an oxidising mild steel chute located in the upper region of the brick. Following permanent defueling of that channel and a 100% inspection of similar locations on both reactors, both reactors were deemed fit for continued operation. The later brick and tile designs (Hinkley Point A and Trawsfynnd) had higher radiolytic weight losses and hence lower component strengths but a combination of early closure and design meant that there were no credible challenges to core functionality and safety.

The later radially-keyed octagon-square brick designs all faced the same challenges. Rather than trace the evolution of the final safety cases and the approach to damage tolerance, it is more instructive to take the last two operating plant Oldbury and Wylfa as examples. The Oldbury reactors were considered to be the reactors that had suffered most from graphite degradation (radiolytic oxidation). The Wylfa reactors were the last Magnox reactors in the fleet to cease generation. Although the Wylfa cores are much larger than those of Oldbury, presenting more bricks that contribute to the risk of a fuel clad melt, the safety case methodologies were broadly the same Magnox North (2010 and 2011).

The safety case for the reactor cores at both stations was required to demonstrate that:

- The likelihood of widespread core damage of an extent and severity sufficient to challenge core shut-down and hold-down capability is extremely low, and that
- The likelihood of graphite brick splitting is sufficiently low to ensure that the nuclear safety risks associated with channel flow by-pass and consequential fuel overheating are acceptable.

The evidence that underpinned the absence of widespread core damage was based on control rod drop tests, temperature monitoring, monitoring of refuelling activities, core visual inspections, channel geometry measurements and graphite sampling and characterisation. Also, key to the safety case was the concept of a 'percolation limit weight loss', the weight loss above which the graphite strength was predicted to be zero.

The evidence that underpinned the low likelihood of brick splitting was based upon visual inspections and both probabilistic and deterministic assessments of structural integrity. This was combined with the associated risks of fuel clad melt and radiological release to provide arguments for continued operation. Furthermore, trip transients were assessed to identify those transients leading to the most severe brick thermal stresses. Of these, two were classed as sufficiently onerous to require either a 10% inspection of the flattened region of the core or a review of the safety case prior to return to service. The safety case strictly adhered to the multi-legged approach described in Section 3 above.

Both the deterministic and probabilistic methodologies of assessment followed the same three stages, crack initiation, crack progression and brick splitting, and coolant by-pass and clad melt risk. It should be noted that a potential consequence of cracking and fragmenting of bricks is the production of debris. This can present significant problems in terms of coolant gas flow and associated heating of the fuel as well as impairment to control rod and fuel movements.

In the case of the deterministic approach, emphasis was placed on the likelihood of crack initiation. This involved the evaluation of Utilisations Factors (UFs), derived by comparing brick stresses at locations throughout the core with the predicted graphite strength at each location. The predicted failure mechanism was crack initiation at the fuel channel wall due to thermal tensile hoop stresses during a shut-down transient. Provided UFs were below unity, with an added margin to cover uncertainties in their evaluation, the likelihood of crack initiation was low. But it should be noted that a UF of unity corresponds to 50% of bricks being cracked. Crack initiation, even with propagation, over the full height of the brick, was not expected to lead to coolant flow diversion. Propagation of a crack at the fuel channel wall through to the outside of the brick was considered to be highly unlikely because of the presence of compressive hoop stresses at this outer region. A single axial crack was shown from flow modelling studies to provide insufficient coolant gas leakage flow to cause fuel temperatures to rise. For this to happen, a second full length axial crack diametrically opposite or at 135° from the first crack (depending upon type of brick) was necessary, effectively splitting the brick into two segments. However, even then, it was argued that the crack gape would need to be sufficiently large to allow leakage flow and brick segment movement would be hindered by the magnitude of any 'lost motion' (available clearances to accommodate displacements) local to the failure. Such movement would be determined by pinching of keys in keyways (see Fig. 3 for Magnox keying arrangement) due to irradiation-induced dimensional change holding bricks in place and the size of Wigner gap clearances between bricks which would again be modified due to dimensional change. The judgement of the safety case authors was that brick splitting was unlikely. If a brick were to split, diverted coolant gas through the crack could impair cooling of the fuel elements above the affected brick. This deterministic approach was regarded as subservient to probabilistic assessments, with judgements on the likelihood of brick splitting being imported from those assessments.

As would be expected, the probabilistic methodology quantifies the probabilities of each of the three stages occurring. The derived fuel clad melt probabilities were then combined to obtain the probability of a melt somewhere in the core as a result of brick cracking. Stage 1 was based upon shut-down transient stresses and graphite strengths using appropriate probability distributions for both hoop stresses and graphite flexural strength. Stage 2 utilised both a 'crack propagation model' and a 'double initiation model' as alternative approaches to crack development. Stage 3 evaluated conditional clad melt probabilities dependent upon brick location, brick gape, coolant by-pass flow and the channel axial power profile. Whole core single channel clad melt probabilities of less than 5.1×10^{-6} and 1.5×10^{-3} were indicated for the crack propagation and double initiation models, respectively. Based upon these results combined with sensitivity studies, it was claimed that the risks were Tolerable and ALARP (As Low As Reasonably Practicable) based upon the UK nuclear regulator probabilistic principle for nuclear safety cases, Office for Nuclear Regulation (2014).

Summary: An outline of the most advanced safety cases for the Magnox reactors highlights the shift in approach from deterministic assessments effectively leading to loss of component functionality to probabilistic assessments that quantify the likelihood of graphite brick cracking with an approach to damage tolerance based upon functionality, nuclear safety and environmental impact. The cores were designed with the intent that core components would remain intact, broadly based upon deterministic assessments of component integrity. The drive to operate the Magnox stations well beyond nominal design lifetimes coupled with the more advanced methods of assessment being employed for the next generation AGR stations necessitated a change in assessment approach that acknowledged that brick cracking could occur. As will be seen in the next section that summarises the AGR position, the consequences of cracking for Magnox reactor and AGR cores are significantly different due to differences in reactor design. The starting point for the Magnox cores was that, despite extensive inspections and monitoring, there was no evidence for brick cracking. However, distributions around analysis parameters applied to probabilistic assessments showed that there was always some probability of cracking. The potential chain of events following crack initiation, namely crack propagation, brick splitting, coolant gas diversion, fuel clad melt and radiological release, forced assessors to demonstrate that the probability of each event was low and the combined probability either insignificant or at least tolerable. The level of tolerance, although never specifically addressed, was very low.

4.4. AGR cores

A similar summary of actual AGR safety cases has been limited as direct public access to safety case submissions is not possible while the plants are still in operation. However, the regulator does make their assessment documents available on their website which provides insights into safety case arguments associated with damage tolerance. The discussion below does not attempt to capture all aspects of these safety cases but rather provides some insight into their evolution in response to emerging issues.

Early assessments focused on Reserve Strength Factors and Fractional Remanent Strength (analogous to UFs in Magnox deterministic assessments) to demonstrate that margins to brick cracking were large. Stress analyses of ageing graphite bricks showed that brick cracking would be expected relatively late in life (following dimensional change turnaround from shrinkage to growth), initiated by tensile stresses at the roots of keyways. However, early visual inspections of fuel channels revealed unexpected and unwelcome cracks in the bricks at the fuel channel wall, well in advance of the predicted onset of keyway root cracking. There has been a variety of crack shapes in different AGR reactor cores at this relatively early stage in the plant lifetimes, including some axial, some circumferential, some 'lasso' (axial linked to circumferential), some cases of bricks with two axial cracks and less common 'interesting' configurations.

For the purposes of classification, features discovered at the bore during inspections of reactors were defined by the licensee, as follows:

- Type I – Features that will not act as initiators for bore cracking irrespective of stresses within the material.
- Type II – Potential crack initiators – features that could grow into cracks if stresses are sufficient and cracks less than 50 mm in length.
- Type III – Cracks of a length that imply in service growth.
- Type IV – Outliers (i.e., defects that lie outside of current experience, expectation and safety case assessment).

An example of an early life bifurcating crack observed in an AGR brick is shown in Fig. 9. Repeat inspections of cracks have shown that some can be stable while others have developed progressively with time.

The sentencing of these features can be difficult as illustrated in Fig. 10. Here, two full-length axial cracks were observed with the measurable crack widths or gape suggesting that the brick may have split into two segments.

While the licensee has soundly-based methodologies for assessing brick stresses and material strengths, it has not been able to predict the observed bore-initiated cracking. There has been speculation that such cracking may have been initiated from pre-existing manufacturing defects at stresses lower than those predicted to cause cracking in 'normal' material. However, even with material without such defects, uncertainties in input data lead to low confidence in the prediction of timing of cracking and number of cracks across a core.

The licensee for the AGR fleet, EDF Energy, has set out its methodology for damage tolerance assessments (DTA) of graphite cores in

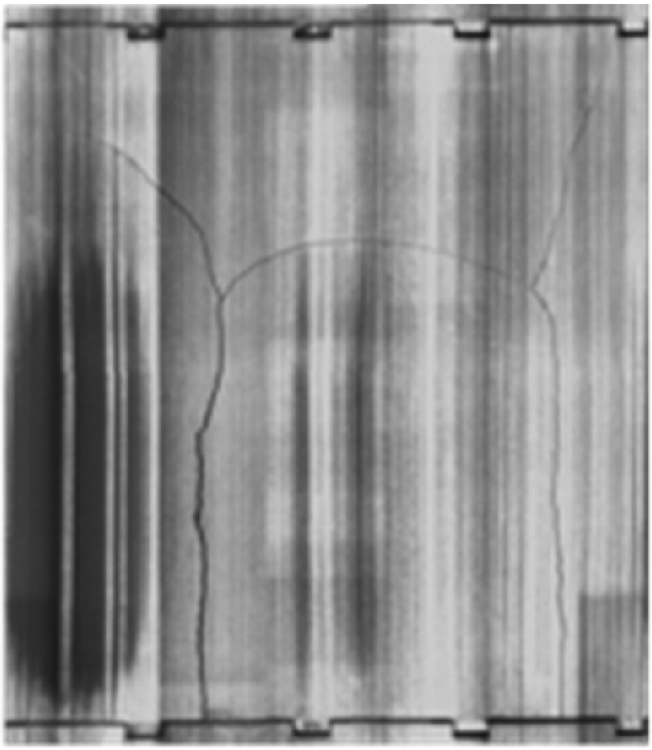

Fig. 9. Composite image showing entire fuel channel wall surface of an AGR brick exhibiting early life cracking (Source EDF Energy).

McLachlan (2018). Tolerance to damage in the form of cracked graphite components arising during normal operation, faults or seismic events is regarded as an integral part of the safety cases for the graphite cores. Unlike assessment methodologies adopted for the Magnox fleet and historical assessment approaches for the AGRs involving the prediction of brick cracking, and also due to the observation of a number of multiply-cracked (albeit part-length) bricks, the current DTAs accept that cracked components will be present in the cores and that it is the behaviour of the entire core structure which should be assessed against nuclear safety principles. This has led to the following developments:

- Core behaviour based upon computational models of whole cores has been developed and validated against scaled arrays of bricks (rather than actual models/predictions of an irradiated and oxidised core).
- Particular emphasis has been given to seismic loading with detailed assessments of seismic motion histories and the loading modes on the cores. While the most onerous loading direction is considered to be 45° to the array, the probability of seismic consequences arising has been deemed better represented by uniform sampling of directions with the averaging of results.
- Careful consideration has been given to brick load capacities and the role of stress reversal on effective keyway load capacity which governs through-brick cracking (Fig. 11).
- Traditional 'stick and spring' models of keyed brick arrays have neglected friction which may be a reasonable assumption for normal operation and long-term effects. However, for seismic events where the period of interest is only a few seconds, some allowance for friction needs to be built in to models.
- Seismic end-of-event distortions and the margins for control rod entry may be as important as those for the seismic event itself and require evaluation against those for normal operation.
- Core behaviour cannot be assessed from any one computational model in isolation. To attain the most complete understanding of core behaviour, a multiplicity of models including 'stick and spring', solid-body and hybrid models have been developed. This allows

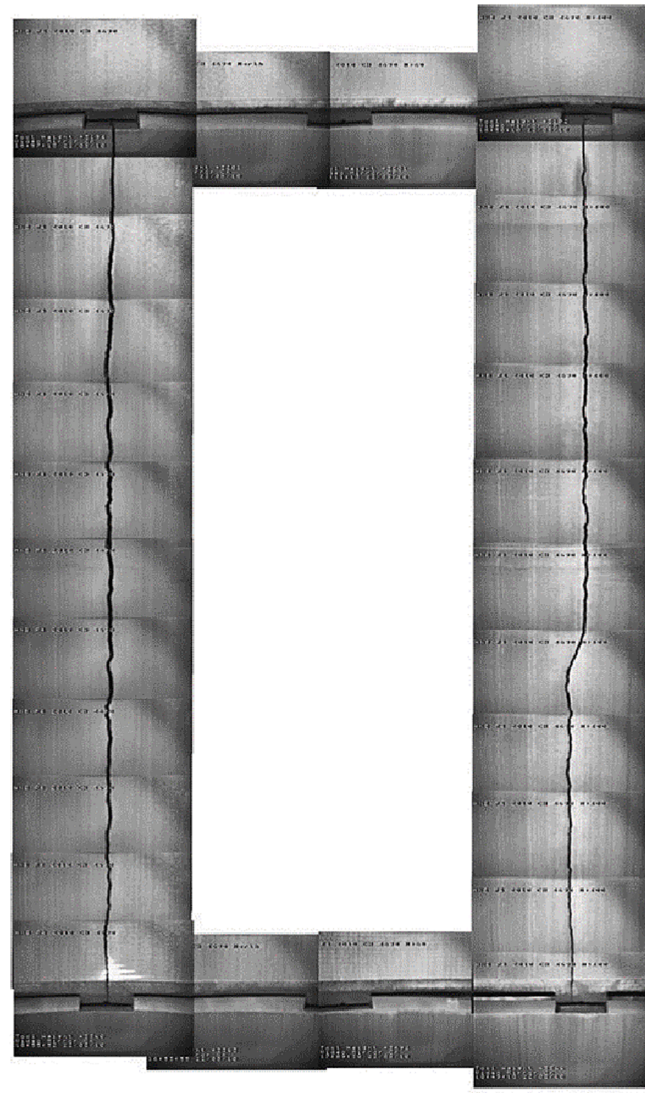

Fig. 10. Composite image showing fuel channel wall surface of an AGR brick (images for uncracked region omitted) exhibiting two full length axial cracks (Source EDF Energy).

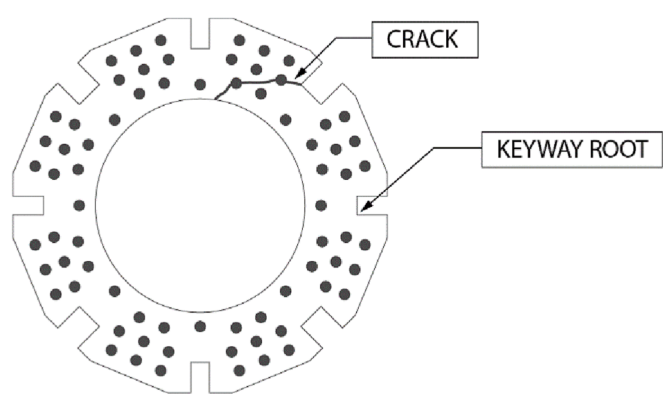

Fig. 11. Cross-section of an AGR fuel brick illustrating a keyway root crack running through to the fuel channel.

more accurate analysis of geometric details such as discrete keys, key/keyway disengagement and various cracked-brick geometries.
- Singly-axially cracked bricks will perform differently from intact bricks, with changes in brick width and diameter potentially leading to circumferential keyway displacement and the loading of end-face keyways between axially adjacent bricks. Such local effects have been incorporated in computational models.
- Loading events have been assessed with simulated core data. For a core with a defined number and distribution of cracked bricks (based upon inspection), control rod arrays can be shifted by increasing numbers of lattice pitches or rotated to 'control rod quasi-channels'. This has the effect of 'multiplying' a single model run to get 16 or 32 core's worth of standard or sensor rod information, thereby adding significant value to statistics.

The models referred to in this list of DTA developments may also be used to give an appreciation of the current state of the whole core by incorporation of inspection data that is collected during routine reactor maintenance periods. These data comprise brick tilt measurements and numbers and locations of cracked bricks. Furthermore, uninspected channels represented in the models are assumed to have cracked layers and crack characteristics similar to those for inspected channels.Fig. 9: Composite image showing entire fuel channel wall surface of an AGR brick exhibiting early life cracking (Source EDF Energy).

To illustrate the way in which station operators have addressed core degradation, it is instructive to examine the approach used for the lead reactors: the twin Hunterston B and Hinkley Point B stations. Hunterston B Reactor 3 has shown the highest level of degradation across the four reactors and was shut down for assessment in 2018. Fuel channels were visually inspected and cracks categorised as 'within or below expectation', 'broadly within expectation' and 'outside of expectation'. While the majority of bricks with keyway root cracks (Fig. 11) were found to lie within the first two categories, some of these bricks had induced full height cracking in a significant number of axially adjacent bricks which had to be placed in the third category. This phenomenon of induced brick cracking required the operators to revise their crack prediction methodologies and also to revise their assessments to include significantly larger numbers of cracked components. Furthermore, observed crack morphologies indicated the need for more rigorous assessments of the consequences of debris production. The current safety case approved by the regulator, Office for Nuclear Regulation (2019) addresses uncertainties in the number and type of cracked components by defining a bounding case (currently established damage tolerance limit) for the central region of the core (graphite layers 3–7, rings 1–9) comprising approximately 1300 bricks. Of these, a combined total of 400 bricks are either doubly or multiply cracked (with no more than 200 multiply cracked) and the remainder contain full length single cracks. The anticipated number of doubly and multiply cracked bricks is 100. This core state is then assessed against the fundamental nuclear safety requirements of the core, namely straight enough channels to allow fuel and control rod movement and no challenge to gas flows to change fuel temperatures. As stated above, Hunterston B was finally ceased generation in 2022. In the case of the Hinkley Point B reactors, the UK nuclear regulator approved return to service in 2021 after shut down in 2020 for inspection and assessment, for operation up to separate specified core cumulative power outputs for each reactor with the requirement for approximately 6 monthly inspections, Office for Nuclear Regulation (2021). The approach to damage tolerance has been similar to that for the Hunterston B reactors with a number of further refinements including: (i) updated ground motion cases for seismic assessments, (ii) updated representations of multiply axially cracked bricks, (iii) adding friction to the core seismic model, (iv) a defined limit of normal operation which bounds the damage tolerance boundary, and for which there is high confidence in the graphite core not impeding control rods, (v) a seismic end-of-event cracking assessment, which accounts for in-event damages, and for which there is high confidence in the graphite core not impeding control rods, and (vi) updated keying system capacities. The noting of these refinements provides insights into how safety assessments are being developed but more comprehensive details are beyond the scope of this paper. It is noteworthy that while the seismic event typically presents the greatest challenge to nuclear safety requirements, it is believed that significant cracking of components takes place at shutdown due to a combination of changed temperature gradients and irradiation-induced dimensional change.

Summary: This review of AGR safety case methods highlights a fundamental shift in emphasis of approach from when the reactors were first brought on line. This change of approach, still based upon exactly the same nuclear principles, was brought about due to inconsistencies between observed damage and model predictions. Historically, assessments had been based on predictions of the time to keyway cracking and the resulting numbers and distributions of cracked bricks, in combination with assessments of the consequences of such damage. Since the presence of cracked bricks does not necessarily undermine safety cases for continued reactor operation under normal and fault conditions, assessments of different levels of damage, informed by predictions and inspections, largely eliminated dependence on the uncertainties associated with predictions of timescales to damage. Probabilistic assessments could then focus on the tolerability of selected bounding levels of damage. Unlike Magnox reactor cores, coolant gas flow disruptions due to cracked moderator bricks could not affect fuel temperatures potentially leading to radiological release. For AGR cores, the key nuclear safety issue was channel geometry impairment potentially leading to disruption to control rod movements and inadequate shutdown and hold-down capability, as well as the potential for debris from multiply cracked bricks modifying gas flows and hence fuel temperatures.

5. The UK nuclear regulator perspective

The regulator requires the licensee to demonstrate through their safety case that they have sufficient understanding of graphite behaviour and can support in a clear and evidence-based manner safe operation of the core. The regulator also requires the licensee to define conservative limits of operation based on the extent and adequacy of their understanding of graphite core ageing. It is important to appreciate that UK graphite-moderated gas-cooled nuclear plant were designed and successfully operated over more than 60 years in the complete absence of graphite-related standards, highlighting the success of the UK approach to regulation.

Specifically, within its operational safety cases, it is the licensee's responsibility to establish weight-loss and cracking safety limits for each reactor core, based on comprehensive study and routine monitoring and analysis of graphite behaviour.

The nuclear site licence stipulates that the licensee must establish and justify the operational restrictions of its nuclear plant. These limits can be revised depending on the licensee's understanding of the graphite behaviour. The licensee is expected to justify the limits based upon information obtained through surveillance and associated research, and to update the limits for weight loss and cracking defined within the safety cases for the reactors as appropriate.

The challenge to the regulator is how it can adequately assess safety case claims for these unique reactors when the technical community largely resides within or supports the licensee. The regulator undertakes its own independent research and takes advice from academics and panels of experts to allow it to make informed judgements on the validity of safety case submissions (Bamber et al., 2015). The regulator also has the power to instruct licensees to undertake particular surveillance and research to support claims in the safety cases. Regulator guidance is constantly kept under review, ensuring that it remains relevant and reflects new technologies.

The UK Office for Nuclear Regulation (2020) sets out its safety assessment principles (SAPs) for nuclear facilities. These principles provide guidance to licensees and have no legal context. They set out regulator expectations when judging if a safety case is adequate. However, if the SAPs are not met, it does not represent a breach of the law in any way and hence, adequacy may be demonstrated in other ways. This highlights the non-prescriptive nature of the UK regulation and affords some flexibility. In contrast, adequacy is a requirement of a nuclear site licence so is part of law. In the SAP section on graphite reactor cores, it is stated:

"*Due to differences in design and safety functions, graphite reactor cores may in some instances be defect tolerant, while in others, safety functions may exhibit low defect tolerance. Therefore, the application of these principles needs to cater for a spectrum of safety performance.*

Safety cases for reactor cores usually need to adopt a multi-legged approach, based on independent and diverse arguments. The rigour of application and robustness of the supporting data and information should be based upon the classification of the graphite components and structures. The multi-legged arguments, with the various elements of established engineering practice, should provide defence in depth.

Novel approaches may be acceptable provided they are supported by appropriate research and development, are tested before coming into service to demonstrate the delivery of safety functions and are then monitored during service.".

The document details a number of Engineering Principles. For the graphite core, Principle EGR.1 states that the safety case should demonstrate that either:

(a) the graphite reactor core is free of defects that could impair its safety functions; or.

(b) the safety functions of the graphite reactor core are tolerant of those defects that might be present.

To meet this principle, the safety case should adopt a multi-legged approach including consideration of design; manufacture, construction and commissioning; component and core condition assessment; defect tolerance assessment; analysis of radiological consequences of defectiveness; monitoring; examination, inspection, surveillance, sampling and testing. Furthermore, the safety case should substantiate how delivery of any safety function might be prejudiced by known or reasonably foreseeable graphite defects and identify the demands this would place on the safety case. A defect in a graphite component is regarded as a deviation from the design specification but not all defects are considered to pose a threat to safety. Analytical models used to assess tolerance to defects must be fully verified and validated. Materials data in such models should bound graphite component operational exposure conditions by an adequate and justifiable margin. Where the defect tolerance assessment is unable to demonstrate clearly that the safety functions will be met under credible conditions, a consequences case would need to be presented. Also, if component or structure degradation is shown, or predicted to occur, it must be demonstrated that such effects on safety functions are progressive with the possibility of disruptive failures, without adequate forewarning, being highly unlikely.

6. Implications for new build

To avoid recycling much of the information provided in the previous sections and also to focus on the key issues, this section on damage tolerance and its implications for new build has been structured as a series of statements or questions with an accompanying commentary. These have been compiled with cognisance of issues on this topic raised by the US Nuclear Regulatory Commission in their review of the ASME Boiler Pressure Vessel (BPV) Code Section III, Division 5 on non-metallic core components and assemblies (Srinivasan et al., 2020). The commentaries should not be regarded as definitive positions on each topic but rather a focus for further consideration and debate.

Graphite components are over-designed and will remain intact over the plant lifetime.

Designers who choose to select graphite components in new nuclear plant must accept that these materials cannot be defect-free and, depending upon the reactor environment, will progressively undergo damage and property changes. UK experience with Generation I and Generation II graphite moderated gas-cooled reactors has shown that setting out to design plant with the expectation that graphite components will not fail over the plant lifetime is unrealistic when operators will always seek to maximise power output by progressively extending operating lifetimes.

Can it be argued that a material characterised by the presence of manufacturing defects is already 'cracked' and is therefore damage tolerant?

To claim that graphite components are tolerant to damage by virtue of the fact that the manufactured product already contains defects or 'cracks' is not valid. A component can be regarded as 'failed' when it ceases to fulfil its design function. Therefore, any claims on damage tolerance must be supported by rigorous consequence assessments against nuclear safety principles. Modern methods for manufacturing graphite include visual inspections and non-destructive testing. The manufacturer and designer need to have agreed procedures on acceptance criteria, the criteria being supported by design assessments of component function and in-service conditions (irradiation conditions, chemical environment, internally and externally generated stresses). Accurate records must be maintained of accepted defects in material delivered by the manufacturer, inspection of components during reactor assembly and location within the assembly of sentenced components together with heat certificate data.

Terminology requiring agreement between designer, operator and regulator on defects, cracks, damage and failure.

UK experience showed an emerging picture of predicted and observed damage as operators sought to maximise economic returns on plant through lifetime extension. While it is not always possible to predict or explain graphite behaviour despite decades of research, it is important to establish a common language and understanding of the nature of graphite, the function of components and the type of defects or damage that may arise during operation. The emotive term failure should be used with care and with qualification on precisely what is meant.

Can generic statements be made about damage tolerance?

While a material with inherent defects will be common to all reactor designs that employ graphite components, the type, number and distribution of such defects will vary depending upon raw materials and method of manufacture. Furthermore, the design function of graphite components can be very different leading to very different approaches to damage tolerance as illustrated by UK experience. It is therefore not possible to make general statements about acceptable degrees of damage in graphite cores. However, the presence of damage does not necessarily compromise safety case assessments. Damage mechanisms need to be understood together with trends in graphite property behaviour. While it may not be possible to predict accurately timescales for changes in property behaviour and the onset of cracking, it is important to be able to align inspection and monitoring data with predicted trends and behaviours and to be able to demonstrate that there are no cliff-edges in long term trends. It is the responsibility of designers to assess the significance and consequences of component damage, to define damage criteria and to provide guidance on life-limiting features of components and arrays of components.

The behaviour of graphite components can be predicted.

The combination of more extensive better-quality materials databases, improved understanding of fracture mechanisms in graphite, the development of ever-more sophisticated models of graphite and component behaviour and real operational inspection and monitoring data has led to high confidence predictions of trends in component behaviour. However, predictions of times for the onset of cracking and predictions of crack propagation depend upon well-characterised uncertainties of property variability within the material and accurate modelling of processes such as irradiation creep. A combination of statistical models based upon inspection data and physically-based mechanistic models offer a step-wise approach to component behaviour that allows high confidence arguments for short periods of operation, particularly as the cores age. Predictive models must be subject to a continuing process of validation against reactor data. Given the known variabilities in graphite properties and uncertainties in operational data, trends in behaviour can be predicted with reasonable confidence but timescales to specific events cannot. It is therefore advisable to adopt a methodology similar to the approach used for the UK AGRs, where the focus is on the tolerability of different levels of damage rather than a reliance on predictive models.

What is the value of inspection, surveillance, sampling and testing?

Both reactor operators and nuclear regulators place a high value on component inspections, core geometry monitoring and graphite property monitoring. There are inherent limitations associated with all these activities in terms of extent inspection/monitoring, sample populations and testing methods. However, in combination and with the accumulation of data and information over time, they provide an essential validation tool for material databases, which historically have been based upon materials testing reactor experiments, and statistical/mechanistic models of core behaviour. The practice adopted by designers of the UK graphite-moderated reactors of including pre-characterised removable test specimens in the cores, making provision for channel geometry monitoring and the later implementation by operators of devices for taking samples from components, for visual inspections and non-destructive examination of core components has provided valuable validation of methodologies and significantly enhanced safety case submissions.

7. Conclusions

UK experience of operating graphite-moderated gas-cooled nuclear reactors has illustrated how the approach to damage tolerance for the graphite cores is design-dependent. Provided damage to graphite components does not lead to loss of functionality, then continued operation is not compromised subject to rigorous assessment against nuclear safety principles. However, some designs will not be damage tolerant and will rely heavily on high confidence predictions of component behaviour combined with regular inspection and monitoring for continued operation. It is important to recognise that, while designers of new plant must work to a vendor's specification for operational life, the economic benefits of life extension will lead to consideration of such a strategy as the plant ages. Designers should look beyond planned operational life to identify ageing and damage issues that could limit life extension.

CRediT authorship contribution statement

Martin Metcalfe: Conceptualization, Methodology, Writing – original draft, Writing – review & editing.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

No data was used for the research described in the article.

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