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AGR CORE SAFETY ASSESSMENT METHODOLOGIES

N. McLACHLAN, J. REED, M.P. METCALFE
Nuclear Electric plc.,
Berkeley Technology Centre,
Berkeley, Gloucestershire,
United Kingdom

Abstract

To demonstrate the safety of its gas-cooled graphite-moderated AGR reactors, nuclear safety assessments of the cores are based upon a methodology which demonstrates no component failures, geometrical stability of the structure and material properties bounded by a database. All AGRs continue to meet these three criteria. However, predictions of future core behaviour indicate that the safety case methodology will eventually need to be modified to deal with new phenomena. A new approach to the safety assessment of the cores is currently under development, which can take account of these factors while at the same time providing the same level of protection for the cores. This approach will be based on the functionality of the core: unhindered movement of control rods, continued adequate cooling of the fuel and the core, continued ability to charge and discharge fuel.

1. INTRODUCTION

To demonstrate the safety of its gas-cooled graphite-moderated AGR reactors, the designers of the graphite cores adopted a criterion that they would accommodate through-life distortions and displacements without adversely affecting control rod movements, refuelling operations and coolant flow. Furthermore, the cores should remain intact over their design life. That is, neither brick components nor keys should fail. However, properties will vary between components and it was acknowledged in safety cases for the earlier AGR cores that there existed some low probability of component failure. A restriction on the extent of graphite radiolytic weight loss was also introduced to ensure that component properties remained within the graphite properties database used in the core design and did not suffer an unacceptable reduction in strength.

All AGRs continue to meet these three criteria. Assessments by Nuclear Electric, in conjunction with UK Atomic Energy Authority and National Nuclear Corporation, have shown that for externally applied loads safety margins associated with these criteria are very high. However, more sophisticated analyses have shown that the fuel bricks may eventually fail due to internally generated stresses. Furthermore, graphite in some of the cores is oxidising at a faster rate than anticipated, and the limits on weight loss will be exceeded sooner than originally predicted.

A new approach to the safety assessment of the cores is currently under development, which can take account of these factors while at the same time providing the same level of protection for the cores. This approach will be based on the functionality of the core: unhindered movement of control rods, continued adequate cooling of the fuel and the core, the continued ability to charge and discharge fuel.

This paper is composed of three parts:

- the existing methodology for core safety cases is briefly outlined,
- the applicability of the existing methodology is discussed in the light of predictions of the future condition of the cores,
- the approach to a new methodology is discussed in general terms.

2. BACKGROUND

2.1 AGR Core Design

Although there are other presentations at this conference giving details of the AGR core design, some of the significant features will be described here. The AGR cores are built from columns of hollow graphite bricks, stacked to form vertical fuel/control-rod channels, and linked together into a lattice using a horizontal keying arrangement, as illustrated in Fig 1. This arrangement encourages uniform core movement while allowing a degree of local freedom. It is important to maintain the mechanical integrity of the lattice, since this ensures the correct geometry for adequate coolant flow, refuelling operations and control-rod insertion.

The core structure rests on, and is restrained by, steelwork which, during power transients, can undergo different thermally induced movements, with the consequence that the differential displacements are transmitted to the outer graphite components through the restraints and brick keying arrangement, and the brick keyways are subjected to external loading. The geometry of the bricks is such that the highest load induced stresses arise in the vicinity of the keyway roots.

2.2 Effects of Irradiation Exposure

Core graphite components undergo fundamental changes to their shape and physical properties due to neutron irradiation. The amount of change is dependent upon the accumulated irradiation dose. The dose is dependent upon the location of the component within the core and on the local distribution of fuel. Therefore the changes expected within the core are neither uniform nor constant and assessments have to be carried out at various stages in core life.

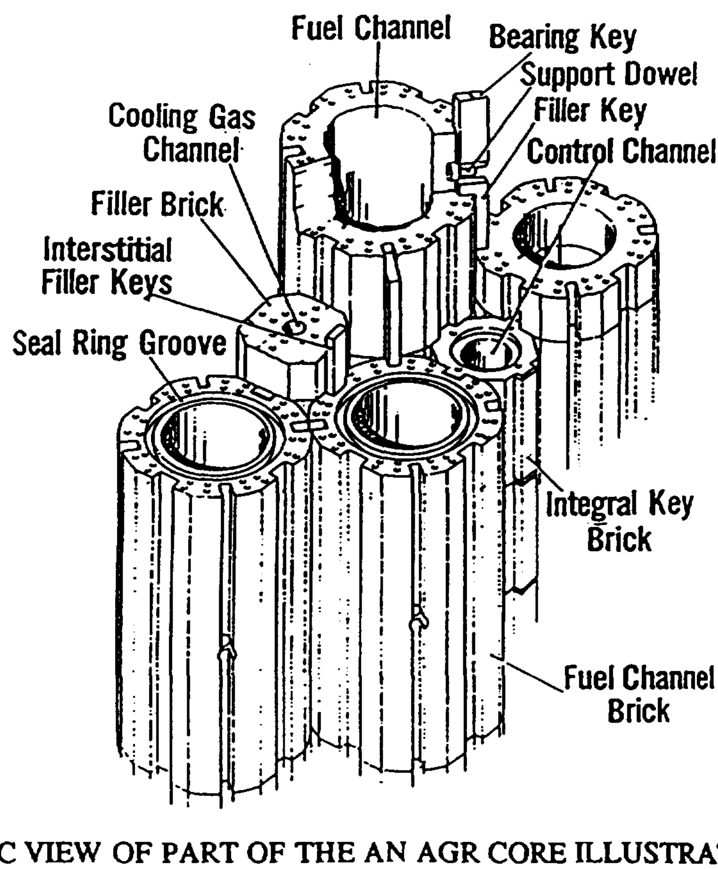


Fig 1 ISOMETRIC VIEW OF PART OF THE AN AGR CORE ILLUSTRATING THE KEYING ARRANGEMENT

Shrinkage in fuel bricks experiencing an uneven dose distribution can result in brick barrelling and keyway closure. Bricks at the core periphery, which are subjected to a nett dose gradient, can experience so-called brick bowing. In this case, the individual components curve towards the core centre. All of these deformations may result in a reduced degree of local freedom, with the consequence that transient displacements are transmitted further into the core, and external loading of components is increased.

Dimensional changes can also generate internal shrinkage stresses. Each brick in the active core region has a non-uniform distribution of irradiation dose, irradiation temperature and weight loss. Consequently, each brick will have a complex distribution of shrinkage strains over its cross-section. For a fuel brick, the dose at the bore is around twice the value at the periphery. Initially the bore of the brick will shrink faster than the periphery, with the bore becoming stressed in tension and the periphery in compression. The stress field will continue to grow until around the time at which shrinkage turnaround occurs, when the graphite tries to grow. The stress field then reverses with the bore now in compression and the periphery in tension (Fig 2).

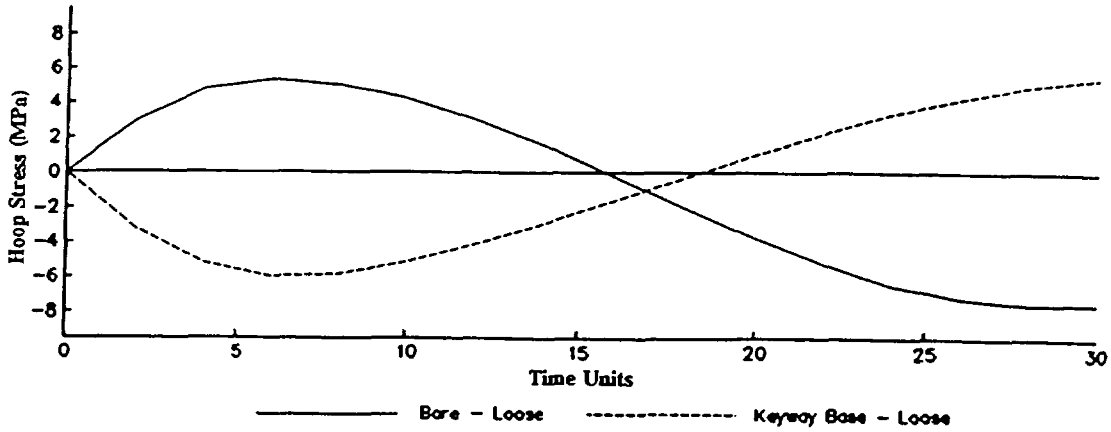
Differential shrinkage strains over the brick section will also produce distortion of the keyways, causing a reduction in initial key/keyway clearances. The design intent was for clearances to be present over the reactor life. However the use of more sophisticated finite element analysis and improved data has shown that the clearances will reduce to zero, potentially locking-up the core structure. Furthermore, this process introduces tensile pinching stresses at the keyway root.

A further source of internal stress, not fully appreciated at the design stage of the early AGRs, arises from irradiation-induced changes in the coefficient of thermal expansion (CTE). At power, these thermal stresses undergo irradiation creep relaxation. However, since CTE changes are non-uniform due to the local variation in dose, additional thermal stresses will arise during thermal transients unaccompanied by irradiation, in particular during shutdown of the reactor. These thermal shutdown stresses grow increasingly tensile at the keyway root with time (Fig 2).

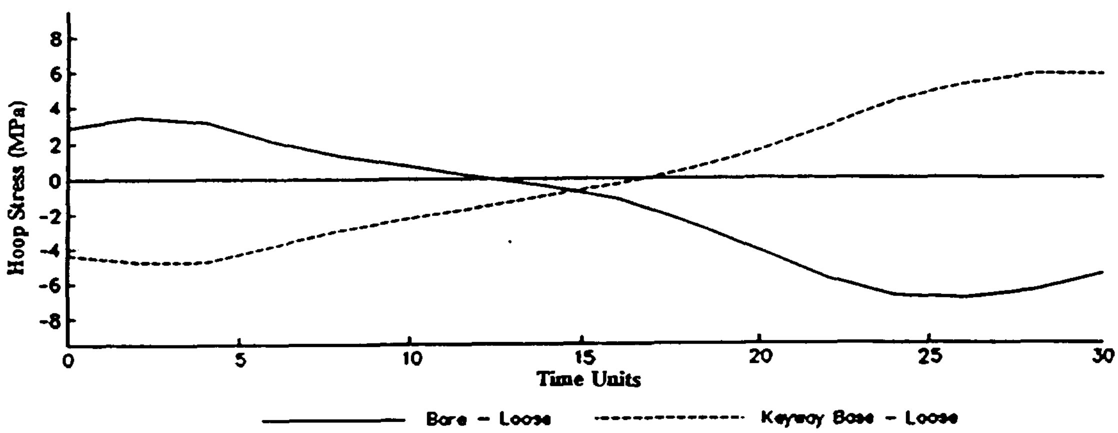
Finally, gamma radiation activation of the CO_2 coolant results in radiolytic weight loss from the graphite, the carbon reacting chemically with the CO_2 to form CO. This oxidation process tends to reduce the material strength and Young's modulus exponentially. Fortunately, irradiation-induced changes to the structure of individual graphite crystals, and consequential changes to their arrangement in the nuclear graphite, have the effect of offsetting these strength and modulus reductions, particularly in early life. The dependency of Young's modulus on weight loss and irradiation dose is illustrated in Figs 3 and 4 respectively.

3. EXISTING CORE SAFETY CASE METHODOLOGY BASED UPON DESIGN

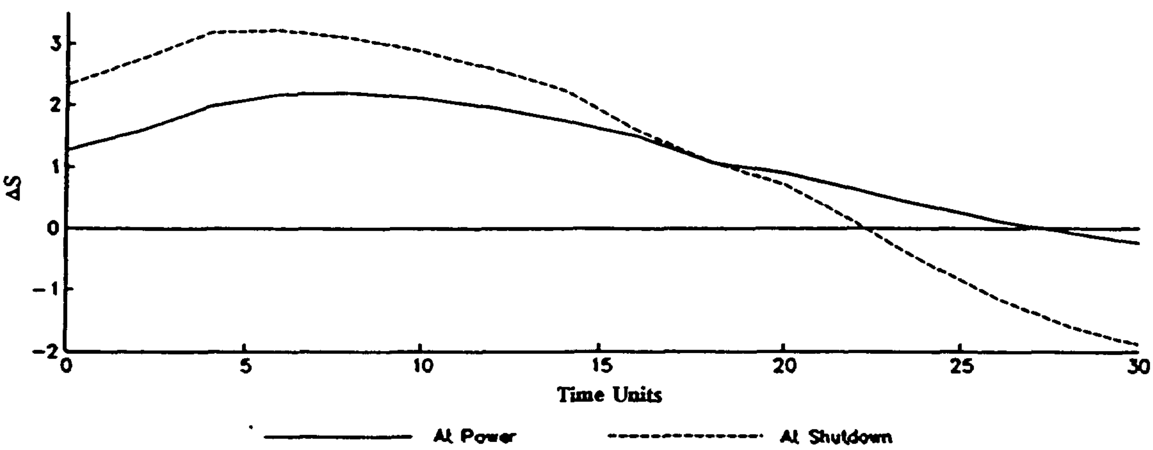
A simplified schematic flowchart showing the individual elements of the methodology and their inter-relationships is given in Fig 5. The primary aim of the process is to demonstrate that no core component will fail over the design life of the reactor. This is accomplished by establishing the time in life at which the first component will fail, and demonstrating that sufficient margins are available to avoid this eventuality over the period of assessment. In addition to this, the methodology provides methods for assessing weight loss and core geometry. The way in which the predicted condition of a graphite core is assessed against these criteria is described below. Several other papers at this conference expand on some of the topics ("Application of a Method for Assessing the Probability of Graphite Core Brick Failure" by R C B Judge, and "Radiolytic Graphite Oxidation Revisited" by P C Minshall).



(a) Shrinkage Stress



(b) Thermal Shutdown Stress



(c) Fractional Remanent Strength associated with Keyway Base Stresses

Fig 2 THE VARIATION IN SHRINKAGE STRESS, THERMAL SHUTDOWN STRESS AND FRACTIONAL REMANENT STRENGTH WITH ACCUMULATED IRRADIATION DOSE

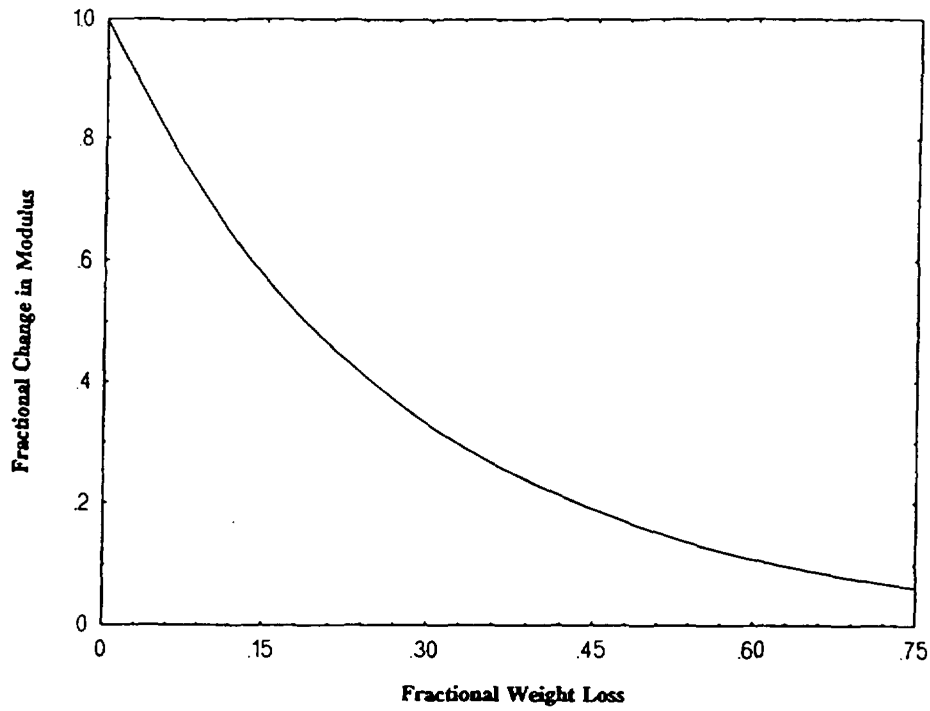


Fig 3 DEPENDENCY OF YOUNG'S MODULUS ON RADIOLYTIC WEIGHT LOSS FOR ISOTROPIC AGR GRAPHITE

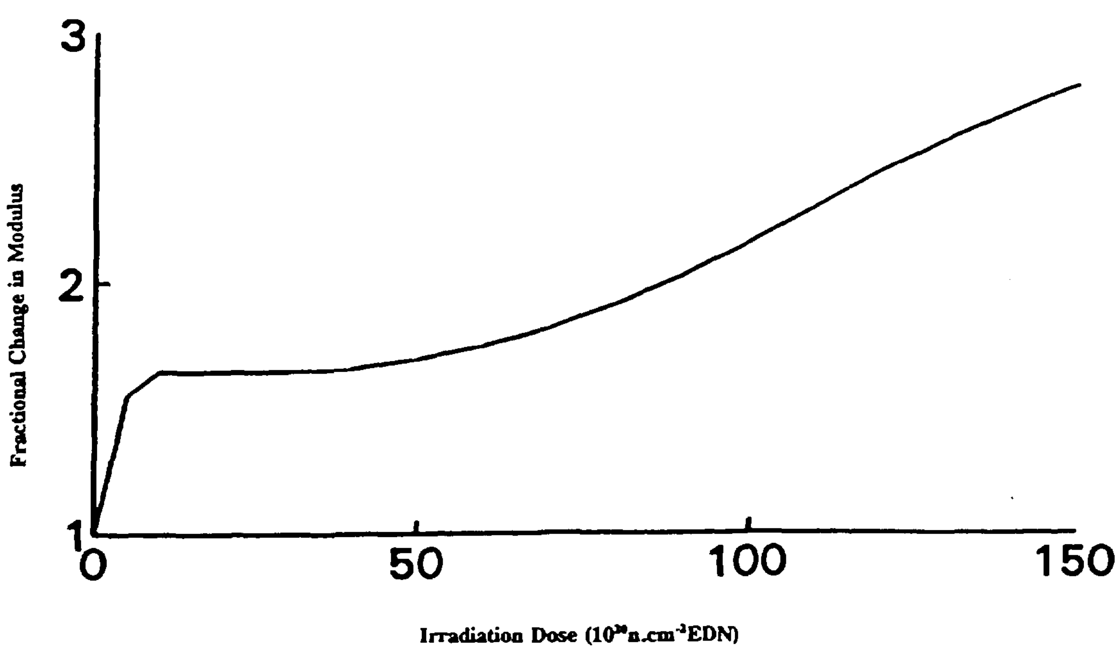


Fig 4 THE DEPENDENCY OF YOUNG'S MODULUS ON IRRADIATION DOSE FOR ISOTROPIC AGR GRAPHITE

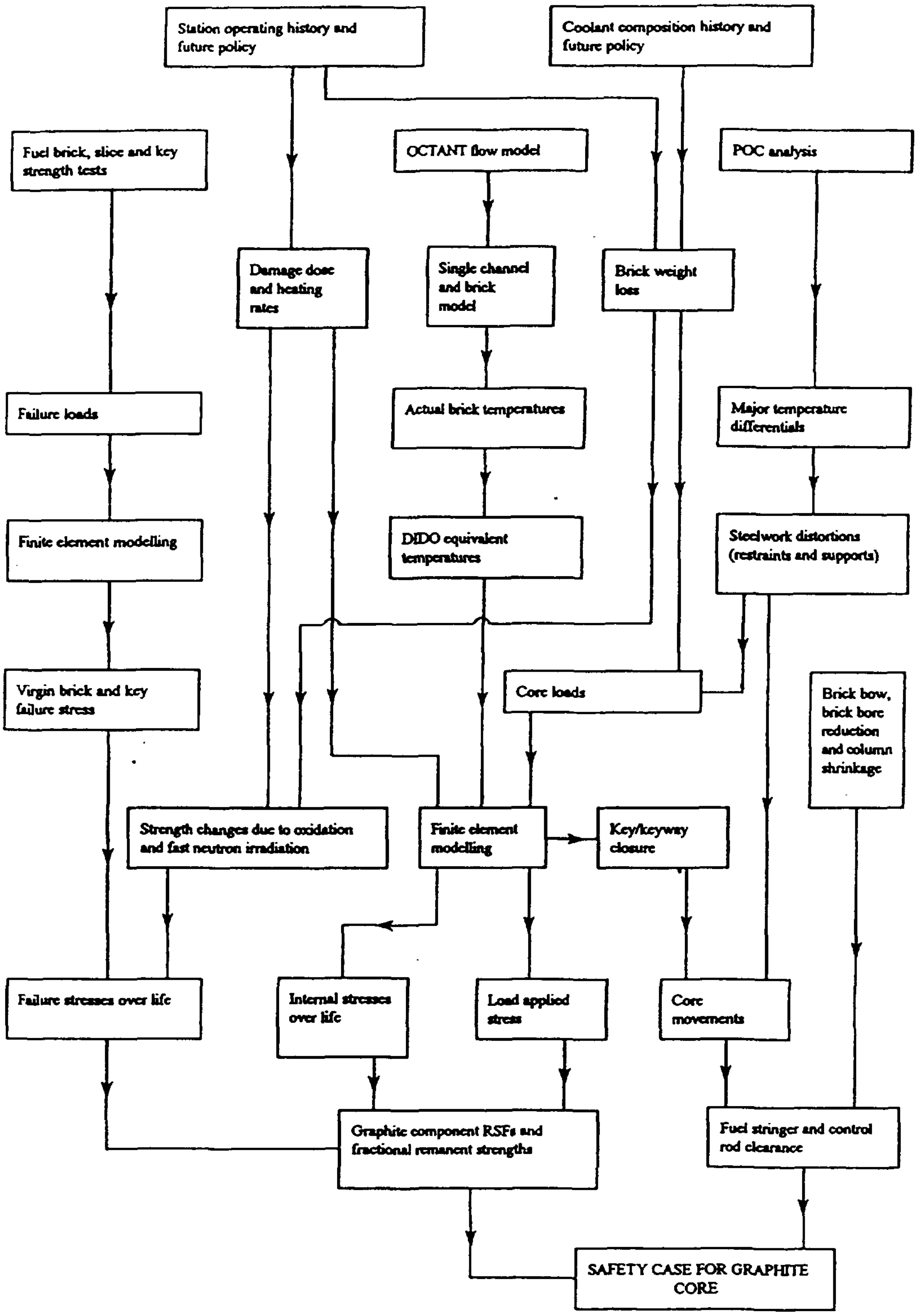


Fig 5 FLOW CHART DESCRIBING AGR GRAPHITE CORE INTEGRITY METHODOLOGY

3.1 Reserve Strength Factor and Fractional Remanent Strength

The criterion originally used in the design safety cases for assessing core integrity was the comparison of the reserve strength of a graphite component against the stress generated by external loads. The reserve strength took account of the strength modifying effects of shrinkage and thermal stresses. The strength ratio was termed the Reserve Strength Factor (RSF):

$$\mathrm{RSF} = \frac{\text{irradiated strength} - \text{internal stress}}{\text{load applied stress}} \qquad (1)$$

This factor was required to exceed certain lower bounds (greater than unity) under various reactor operating conditions, with the value reflecting the likelihood and severity of the event. Some account had also to be taken of the variability in graphite material properties. An RSF greater than five was required for normal operation, reducing to two for infrequent faults.

The development of improved methods for quantifying load applied stresses and internal stresses showed the RSF concept had a number of limitations. The RSF expression compares stresses which may be very different in form and which may have different critical values. Furthermore, load applied stress is treated in a separate and special manner which excludes the possibility of component failure from internal stress alone. These problems were resolved by the introduction of an improved component failure criterion termed fractional remanent strength or ΔS. This criterion ranks with equal merit the various mechanisms of component failure, ie. shrinkage, pinching, thermal shutdown, load-applied stresses, and associates potentially different critical values of stress with each:

$$\Delta S = 1 - \frac{\text{shrinkage stress}}{\text{critical shrinkage stress}} - \frac{\text{thermal shutdown stress}}{\text{critical thermal shutdown stress}}$$

$$- \frac{\text{pinching stress}}{\text{critical pinching stress}} - \frac{\text{load applied stress}}{\text{critical load applied stress}} \qquad (2)$$

The ratio of applied stress to failure stress is summed over all failure mechanisms and compared with unity, so that the average component will fail when ΔS has the value zero. Failure may be caused by any mechanism. Like RSF, suitable lower bounds may be imposed on ΔS which must be exceeded under the various operating conditions, and these may be established by performing a detailed analysis of their statistical variability throughout life. The formulation of ΔS has been subject to considerable experimental validation, and been proven to behave consistently.

3.2 Weight Loss Criterion

As with RSF, there has been some evolution of the definition of the weight loss criterion. Originally a limit of 20% was set on the mean weight loss of any component. By choosing such a value, it was argued that the peak weight loss in the component would remain within the materials property database weight loss limit of 40%. The criterion was later refined to take account of the different cross-sectional weight loss distributions within components. As the critical stress position within a radially keyed component is at the keyway root, "effective weight loss" was introduced, defined as the average of the mean component weight loss and the weight loss at the keyway root. The same limiting value was adopted to determine coolant composition strategy, the coolant chemistry for an AGR being optimised to maintain a balance between carbon deposition on the fuel and boilers and graphite oxidation. The DIFFUSE6 computer code has been developed to model radiolytic graphite oxidation in AGRs. This code, which has been validated against measurements from the AGR graphite monitoring programme, provides detailed information on the radial and axial distribution of weight loss within components at any position within the core at any time in reactor life.

3.3 Core Geometry

In addition to the two criteria discussed above (ΔS and % weight loss), there is also the additional requirement to maintain fuel stringer and control rod clearances within acceptable limits. This involves the analysis of transient boundary displacements of the core and irradiation-induced distortion of core components. Boundary displacements include thermal transients associated with normal operational and fault conditions, and their analysis is based upon the basic assumption that:

- core boundary displacements are determined by the movement of the core restraint
- the central channels in the core are vertical
- the envelope of intermediate channels is determined by the key/keyway clearances

This set of rules allows the kinking of columns to be determined together with the associated externally applied loads at the brick keyways. The assumption that the central channels are vertical and hence core displacement is axi-symmetric, termed the "ring theory", is fundamental to the analysis of both core geometry and load applied stress.

Imposed upon this are the component and channel distortions arising from irradiation-induced dimensional changes. A channel bore monitoring unit (CBMU) provides information on brick bows and tilts, from which off-load channel configurations can be derived. These are compared against predictions from dimensional change data.

4. LIMITATIONS OF EXISTING SAFETY CASE METHODOLOGY

Three fundamental limitations have been identified in the existing methodology:

- brick failure is predicted to occur
- weight loss attack rates in two AGRs are greater in magnitude than those determined from the Materials Testing Reactor data, giving rise to higher predicted weight losses
- core monitoring data have shown that channels in central regions of the core are bowed to a greater extent than those at the boundary, an indication of the difficulties in predicting the detail of future core behaviour.

4.1 Brick Failure

As discussed above, bricks may be subjected to a combination of shrinkage, pinching, thermal shutdown and load applied stresses over the life of the reactor. The characteristic variation of shrinkage and thermal shutdown stress with time is illustrated in Fig 2. These stresses have been used to evaluate ΔS also shown in Fig 2. For this example, ΔS based upon the combination of shrinkage stress and thermal shutdown stress falls to zero before ΔS based upon shrinkage stress alone. This indicates that bricks are more susceptible to failure at shutdown due to internal stress. The primary nature of failure will be a crack running from the keyway through to the brick bore.

Following primary cracking, the brick will still be in one piece, albeit "C" rather than "O" shaped, and will be constrained by the surrounding radial keying system. It will continue to serve its structural function. Neither production of debris nor any significant change to the bore geometry which could restrict coolant flow is expected. Cracking will release internal stresses, resulting in the fractional remanent strength for the component being higher after failure. However, the change in geometry will give the component a reduced stiffness under applied loading, particularly in tension perpendicular to the cracked keyway. Analysis has also shown that the propagation of the primary crack enhances stresses at diametrically opposite keyway roots. Secondary cracking of fuel bricks may therefore occur, either spontaneously or under applied load. Secondary cracking will cause the brick to separate into two "C" shaped pieces. Some restriction on this separation is offered by the presence of the axial cruciform keying system, which will almost certainly prevent horizontal shear of the cleft brick surfaces, and significant changes to the fuel bore geometry. However, the likelihood of snagging of the fuel stringer during refuelling will be increased.

The presence of cracked components in the future must be considered in any new safety case methodology. Any additional freedom attributable to the broken components needs to be addressed in the assessment of the interaction of the core with fuel stringers and control rods. In addition, the effect on coolant flow distribution in the core and its impact on graphite component and fuel temperatures needs to be quantified.

4.2 Weight Loss

The limit on effective weight loss has been set conservatively at 20% in order to ensure that local values within a component do not exceeed 40%. Recent observations of graphite radiolytic oxidation rate at two sister AGR stations indicate that the moderator is oxidising at a higher rate than that at the other AGR stations.

The rate of radiolytic weight loss is crucial in determining the irradiation-induced shrinkage behaviour in graphite, with higher weight losses giving rise to greater overall shrinkages. For example, the fuel channel bore at the stations with the higher oxidation rate will shrink up to 15% more compared to the other stations, with a 20% increase in brick-to-brick spacing and axial shrinkage. The effects of these differences in the overall core behaviour are under investigation.

The relevance of a limiting weight loss to component integrity requires further consideration. Both strength and Young's modulus are reduced by radiolytic weight loss, described by similar exponential relationships. Consequently for strain-controlled loading, stresses are influenced in the same manner as the strength and, to first order, the likelihood of component failure is independent of weight loss. Furthermore, as weight loss affects shrinkage rates, higher oxidation rates will cause the postponement of turnaround and the change from compressive to tensile shrinkage stresses at the keyway root. Higher oxidation rates may actually reduce the likelihood of brick failure. However, for external loading, such as axial and radial brick-to-brick loading and interactions at the channel bore from fuel stringers and control rods, changes in strength will significantly alter the graphite behaviour. Although some relaxation on weight loss limits may be available in the context of internal stresses, further work is necessary to define limits for external loads. A programme of work is under way to address this issue and another paper at this conference ("Blunt Indentation Fracture of Core Graphite" by M S Hartley) will give further details.

4.3 Core Monitoring Data

Routine monitoring of the AGR core is carried out, and in particular fuel channel distortion is measured in selected channels during outages using the Channel Bore Measurement Unit (CBMU). The profile of a channel is determined by several factors: changes in channel diameter due to Wigner (irradiation-induced) shrinkage, bow of bricks due to differential Wigner shrinkage, end-barrelling of bricks due to internally-generated stresses, tilting of bricks caused by brick bowing and kinking of brick columns due to interactions with neighbouring columns. All must be accommodated within such freedom of movement of the component allowed by in-built clearances and machining tolerances on all dimensions.

The channel diameter profiles from the CBMU measurements have been assessed in detail and are progressing within expectation. The CBMU also measures brick tilt from which cumulative channel displacement can be calculated. A straight line can be drawn from the top of the highest to the bottom of the lowest measured brick. The deviation from this line is a measure of the channel tilt profile termed channel bow (although the "bow" is not necessarily a smooth arc of a circle). All the analysed CBMU data fall into two categories: base-line data taken before the start of reactor operation and data taken at various stages of operation. The points determined from these data are:

- Before the start of operation, channel bows are typically less than 1mm magnitude, in no particular direction.

- After a few years operation, channel bows are typically 2mm in magnitude, some in radial and others in circumferential directions.

- After extended operation, data from two stations of similar design have shown channel bows of up to 6mm in the central region of the core, while in the peripheral region bows are 2mm in magnitude. Data from a third station show central region bows which are essentially unchanged from those at the start of operation.

Although there are no safety implications from the observed data at the present time, the observations indicate that future behaviour of the cores may be difficult to predict with the current models.

5. NEW CORE SAFETY CASE METHODOLOGY BASED ON CORE FUNCTIONALITY

A new methodology is required for assessing the safety of the AGR cores, which can accommodate future predictions of core behaviour and guarantee a comparable or better level of safety. Any new criterion based on core functionality would still have to accommodate the basic nuclear safety requirements for freedom of control rod and fuel assembly insertion and adequate core and fuel cooling.

There are many observable core parameters which might provide the basis for a measure of core functionality. Two groups can be defined: those which may be proven to be sufficient for (or imply) core functionality, and those which may be proven to be necessary (or are implied). With such groupings, core functionality is assured if any one criterion in the sufficient group is guaranteed. However, if any one of the necessary criteria is not met, core functionality is lost.

The most likely candidates are fuel channel distortion, channel bore weight loss and effect of component cracking on cooling, although these could be replaced by more readily monitorable parameters. The safety case itself will be based on theoretical assessments of the three candidates, supported by experimental substantiation of the theoretical basis together with adequate monitoring of the reactor core and its individual components. At the same time, the assumed fundamental material properties will be validated by experimental work and an ongoing graphite monitoring programme at the stations.

5.1 Channel Distortion

In principle, the acceptable level of distortion of a fuel or control rod channel is definable from geometric assessments of the fuel assembly, control rod and charge path (including relevant components above the core). The prime difficulty arises in defining the likely and worst case channel geometries at any time in the reactor life.

The development of theoretical models for interaction of channel distortions has been the subject of some investigation, Such development was led by a model based fundamentally on core reflective symmetry, subsequently shown from observed reactor data to be invalid. More recently, computational models of vertical and horizontal brick planes have been investigated. These initially took the form of optimisation programs, enabling the assessment of bounding distortions for a given unbalanced force. These have since been extended to study the extent of influence and reactions for given disturbing distortion or forces, bounding channel distortion subject to mechanical and energetic constraints. Generally theoretical modelling seeks to describe the actual core state as simply as possible, using only significant parameters. At the same time, the complexity of the model is increased by the requirement to encompass observations made on experimental models or the actual cores. Thus there is a balance between ease-of-use and validation requirements. There is a requirement to develop these models to provide conservative estimates of channel distortion.

Experimental modelling of channel distortion may provide the basis of a predictive analogue model, or be a source of validation of theoretical modelling. The nature of the experimental modelling to be performed is dependent on the core functionality criteria chosen, and the form of the theoretical modelling employed. Ideally, the experimental model would be a whole core using full-scale components machined from nuclear graphite incorporating irradiation-induced geometry changes in each component. More realistically, a simplified model would represent a reduced zone of the core exploiting symmetry or zone-of-influence. Components with only the appropriate level of detail would be employed, scaled to a more compact size using an alternative material for the components.

To complement the theoretical modelling of core distortions, it would be desirable to have a facility to monitor the existing and developing state of the cores. The changes which should be monitored would include the life-time transients of brick bowing and keyway pinching, the short term thermal transients of steel-work displacements and the permanent key/keyway loss of function through brick failure. New innovative techniques require development to complement the relatively limited CBMU measurements on cores in the static cold shutdown state. These could exploit the measurement potential of fuel channel annuli, arrow head passages, interstitial channels and even methane holes. In addition to this, detailed tie-bar load traces obtained during refuelling operations could give an indication of adverse channel straightness, particularly if there are trends in the frequency of fouling.

5.2 Channel Bore Weight Loss

The tolerable weight loss is influenced by consideration of several factors: the economic balance between moderation and enrichment; the increased shrinkage behaviour which manifests itself as reduced channel bore size, increased brick bowing and increased brick-to-brick gaps; the possibility of debris due to reduced strength; the potential for embedding of the fuel stringer in the channel bore surface. Appropriate limits are required for each of these phenomena.

Limits for weight loss need to be determined for channel bore strength, in the context of debris production and the potential for fuel-stringer interactions during refuelling. This information will ultimately arise from specific test work. Nuclear Electric is currently supervising research at Bath University into the fundamental mechanical properties of graphite subject to a range of extreme weight loss mechanisms. More direct studies of the effects of weight loss on debris and refuelling are planned. These and other relevant experimental programmes will form the basis for an acceptable criterion for safety and operational purposes.

5.3 Fuel and Core Cooling

The consequences of future component/channel distortions or component cracking on core temperatures will need to be addressed thoroughly. The only previous considerations have been for isolated specific component failures where the effects were negligible. It may be that the more extensive modifications to gas flow paths resulting from channel kinking and brick cracking will have a measurable impact on fuel and component temperatures.

Analyses of the effect of core geometry changes on fuel cooling have been limited to the presence of a single cracked brick. Calculations have been performed of the core temperatures taking component shrinkage into account, but only to reflect increases in inter-brick gaps and decrease in fuel channel annulus within a particular layer. The consequences of increased leakage due to cracked components and increased inter-layer gaps due to channel kinking will need to be addressed. Modifications to existing computational models should provide a feasible method of analysis.

If channel distortions were to become sufficient to place loads on fuel assemblies, then inter-element gaps may be opened, or, worse, fuel sleeves cracked. The need for these assessments is dependent on the outcome of the channel distortion studies.

6. CONCLUSIONS

The existing safety case methodlogy is dependent on derived criteria of graphite component integrity and limited mean component radiolytic weight loss. in the future. A new methodology is being developed which seeks to replace the existing criteria with one or a combination of proven new derived criteria for functionality. Initial considerations indicate these to be non-interference between the core and control-rods and fuel-stringers, resilience of the fuel channel wall and acceptable gas flow conditions for fuel and core cooling.

Fuel channel distortion, channel bore weight loss and the effects of component cracking on cooling are the most likely candidates for the core functionality measures, but these may be replaced by others which may be more readily monitored. The new methodology will be based on theoretical appraisal of the candidate measures, using bounding assessments of fuel channel distortion initially while less conservative approaches are developed.

Experimental substantiation of the theoretical basis for channel distortion is necessary, together with adequate monitoring of the reactor core and its individual components. The assumed fundamental material properties of the graphite will be validated by experimental work and an ongoing graphite monitoring programme at the stations.

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