Transactions, SMiRT-23
Manchester, United Kingdom - August 10-14, 2015

Division I, Paper ID 160

REGULATORY APPROACH TO THE CHALLENGE OF AGR GRAPHITE CORE AGEING

Matthew Bamber[1], Laurence Poulter[2], and Kevin McNally[3]

[1] Inspector, Office for Nuclear Regulation, UK
[2] Principal Inspector, Office for Nuclear Regulation, UK
[3] Senior Statistician, Health and Safety Laboratories, UK

ABSTRACT

The UK fleet of seven graphite moderated Advanced Gas-cooled Reactors (AGR) are the only power producing nuclear reactors in the world that have a graphite moderated core cooled by carbon dioxide gas operating at > 500°C.

The graphite core cannot be replaced and during its lifetime the graphite core is subject to the combined ageing effects of irradiation and oxidation. These ageing mechanisms change the mass, dimensions and material properties of the graphite bricks within the core. The mechanisms are complex, inter-related, and difficult to accurately predict and can potentially effect safe operation. As such they pose unique challenges to the reactor operator.

Therefore, the challenge presented to Office for Nuclear Regulation (ONR) as the UK regulator is how to adequately assess the safety claims made by the licensees of these unique reactors, particularly when the technical community is largely within or supports the licensee's organisation. Furthermore, there are few comparators in terms of operational experience, experimental data or codes and standards relative to water cooled/moderated reactors.

In order to assess the degree of conservatism in the licensee's safety claims, ONR has funded a number of technical work streams to provide independent, robust challenge to the licensee's safety claims. These include average graphite core weight loss, the rate of graphite dimensional change and the development of internal stresses within graphite reactor components.

In this paper I will review some of the engineering challenges posed by the graphite cores of the AGRs and the results of some of the work ONR has commissioned as part of its regulatory activity. It demonstrates how the work commissioned by ONR on nuclear graphite serves to develop an independent pool of expertise, separate from the licensee, to provide diverse thinking on problems posed by the UK's unique graphite moderated carbon dioxide cooled reactors. It is important that ONR funds such work to maintain a pool of independent experts who can advise the ONR and help it reach informed regulatory decisions about the adequacy of the licensee's safety cases.

INTRODUCTION

On the 1st April 2014 the Office for Nuclear Regulation (ONR) was established as an independent statutory corporation under the Energy Act 2013 which provides the legal framework of responsibilities and powers of the organisation to regulate the nuclear industry in the UK. In particular, ONR is responsible for enforcement of the Health and Safety at Work Act 1975 and the licensing of all sites in

Great Britain that are to be used for the installation or operation of nuclear reactors (Nuclear Installations Act 1965 as amended).

The safety of nuclear installations is secured primarily through the site licence and the conditions which ONR attaches to that licence. Currently, there are 36 licence conditions attached to the nuclear site licence granted by ONR. Condition 14 requires the licensee to produce a safety case that justifies safety during design, construction, manufacture, commissioning, operation and decommissioning of the installation. Responsibility for meeting this condition lies with the licensee; that is the station operator. ONR does not prescribe detailed standards or codes of practice for nuclear plant. However, ONR will assess the adequacy of the licensee's design safety criteria and standards, and the construction, commissioning, operating arrangements and procedures that will be used in determining compliance with this licence condition.

There are 14 AGRs currently operating on 8 of the 37 nuclear licensed sites around the UK. The first AGRs began operation in 1976 and the last in 1988; they are all still operating and likely to continue operation for many years.

The AGRs are all operated by a single corporate body. They are the only power producing nuclear reactors in the world that have a graphite moderated core cooled by carbon dioxide gas operating at >500°C. By comparison, there are in the region of 273 pressurised water reactors (PWR) and 84 boiling water reactors (BWR) operating in several countries across the world by a number of different operators. Therefore, the AGR technical community is small in relation to that for water cooled/moderated reactors and there are fewer comparators in terms of operational experience, experimental data or codes and standards. The graphite core cannot be replaced and during its lifetime the graphite core is subject to the combined ageing effects of irradiation and oxidation which cause complex degradation mechanisms that must be adequately understood to support safe operation of the plant.

THE EFFECTS OF IRRADIATION ON GRAPHITE CORE COMPONENTS IN A CARBON DIOXIDE ATMOSPHERE

The core of an AGR is composed of several thousand interlocking brick components made from Gilsocarbon graphite. The components have been designed to fit together to provide a neutron moderating structure that can accommodate either fuel or control rods within designated channels (Figure 1). The structure is of a loose construction and has been designed to allow certain degrees of movement between components in order to accommodate the difference in thermal expansion between the steel core restraint system and the graphite core.

There are three fundamental safety requirements of an AGR core:

- Allow the unimpeded movement of control rods and fuel
- Direct the flow of coolant gas so as to ensure adequate cooling of the fuel and the core structure
- Provide neutron moderation and thermal inertia

The ability of the core to meet these safety requirements during its operational lifetime may be compromised by the degrading effects of irradiation and oxidation of graphite components.

Development of Internal Stresses within Graphite Components under Irradiation

Cumulative neutron irradiation of graphite changes its density. In AGR components this is generally observed as a change in dimensions of a component and referred to as dimensional change. The component will shrink initially by as much as 3 - 4% before reaching a plateau, referred to as turnaround, after which the component will then expand indefinitely (Kelly and Brocklehurst 1977). There is a further complicating aspect to the bulk dimensional change behaviour of graphite. If an additional load is applied during irradiation the dimensional change will be modified e.g. application of a compressive load will increase the measured shrinkage (Cundy, Von Der Hardt and Loelgen 1977). This phenomenon has been termed graphite creep, although a more accurate description would be strain modified irradiation induced dimensional change since it is not the same mechanism as creep in metals.

Dimensional change in graphite also changes the bulk coefficient of thermal expansion (CTE) and the effect of creep further changes the CTE response of the material (Kelly, Martin and Nettley 1966).

The operational dose in some of the AGR reactors is sufficiently great that some components closest to the fuel will now be experiencing turnaround, namely fuel bricks. The dose gradient across the fuel bricks induces differential dimensional change strains and consequently, internal stress. A further complicating factor is that the differential strains are modified by creep making it difficult to determine the stress within the fuel brick. During early operation all parts of the fuel brick are shrinking, albeit at different rates. This generates tensile stresses toward the bore of the fuel brick and compressive stresses at the outer circumference. As dose increases a point in time is reached when material close to the bore passes through turnaround and begins to expand. Therefore, the stresses at the bore become compressive and those at the circumference become tensile. The fuel bricks feature keyway channels along the axial length of the outer circumference (Figure 1), these act as stress raisers and could potentially serve to initiate fracture. This phenomenon is generally referred to as keyway root cracking (KRC) and has been observed in two high shrinkage bricks in Hunterston B (HNB) Reactor 4. In theory, this mechanism could lead to the cracking of a large number of fuel bricks in the core and has the potential to increase the freedom of movement of the core. If allowed to proceed unchecked it may challenge the functionality of the core and must therefore be addressed and bounded by the safety case.

Effects of Radiolytic Oxidation on Graphite Component Properties

Irradiation in the presence of carbon dioxide coolant causes oxidation of the Gilsocarbon graphite components in the AGR core. The microstructure of Gilsocarbon graphite is generally characterised by spheres of filler particles typically 0.5 – 1.0mm in diameter dispersed within a matrix of binder phase (Figure 2). The entire structure contains pores from a few tens of nanometres up to 500 microns in size. Some of these pores are open to the gas phase and thus radiolytic oxidation can occur throughout the structure at the surfaces of the open pores. The oxidation product is gaseous carbon-monoxide therefore the mass of the graphite component decreases and has led to the process being generally referred to as graphite weight loss. The loss of mass of the core results in deterioration in neutron moderation and a decrease in component strength and modulus (Brocklehurst *et al* 1970). In combination these material changes have the potential to challenge the fundamental safety requirements of the core.

Regulatory Consequences of the Degrading Effects of Radiolytic Oxidation on the Graphite Core

In order to satisfy the requirements of licence condition 14 it is necessary for the licensee to demonstrate through their safety case that they have adequate understanding of graphite material changes and their rate of progression to justify safe operation of the plant. In order to do this they need to understand the current state of the core and conduct suitable experiments to provide leading data upon which predictions can be made and the claims of the safety case substantiated to support continued operation. This requirement is particularly acute for the operators of graphite core reactors because world knowledge and experimental research is predominantly focused on water cooled/moderated reactors and the AGR design of graphite moderator reactor is singly unique to the UK.

Much of the understanding of radiolytic oxidation of graphite was discovered as a result of materials test reactor (MTR) experiments conducted during the 1960's up to the 1980's, during the height of graphite core development in the UK and parts of Europe. More recently experiments have been conducted by EDF Energy at MTRs in the Netherlands and America on graphite weight loss and creep. These have been combined with data gleaned from sampling of the AGR cores to provide improved predictive models.

The challenge presented to ONR as the UK regulator is how to adequately assess the safety claims made by the licensee of these unique reactors when the technical community is largely within or supports the licensee's organisation. Furthermore, there are few comparators in terms of operational experience, experimental data or codes and standards. Therefore, the ONR has for several years focused on creating and developing a number of independent sources of subject expert advisors that can conduct research to develop independent models.

Following the findings of a Technology Review in 2003 (Marsden 2003) ONR established a Graphite Technical Advisory Committee (G.T.A.C.) to provide advice to ONR. The members of the committee are subject matter experts on issues relating to graphite core structural integrity and have backgrounds in academia and industry. Furthermore, ONR has contracts with the universities of Manchester and Birmingham funding academics and post-doctoral research to develop models and understanding of irradiated graphite phenomena. ONR also has a contract with the Health and Safety Laboratories (HSL) to look at statistical trends in AGR graphite material data to develop diverse and independent predictive models of graphite material behaviour and trends in AGR component degradation.

AN EXAMPLE OF THE INDEPENDENT ADVICE PROVIDED BY THE HEALTH AND SAFETY LABORATORIES TO ONR

AGRs have operational limits relating to graphite core weight loss. There are peak weight loss limits, which address graphite structural integrity, and there are average core weight loss limits which relate to possible reactivity faults as a consequence of decreasing core moderation. These operational limits are put in place by the licensee to demonstrate that a safety margin is maintained at all times. The licensee's weight loss predictions are primarily determined using a mechanistic model of graphite radiolytic oxidation called FEAT-DIFFUSE. This model is based on MTR experiments and is tuned using AGR core sampling data. It is important that ONR has confidence in the licensee's determination of the progression of graphite core weight loss and that it does not transgress its operational limits.

At the request of the ONR, HSL has developed a statistically based model of temporal trends in through brick density of fuel bricks based on samples that the licensee has taken from fuel bricks in AGRs at Hinkley Point B (HPB).

Reliable core sampling data on weight loss at HPB dates back to 1999. The data is derived from cylindrical samples of approximately 10mm diameter and 45mm in length that are trepanned from the wall of the fuel brick closest to the fuel stinger in layers 4 - 9. However, since 2011 the licensee has trepanned deeper into the fuel bricks removing samples approximately 75mm in length. Furthermore, since 2012 samples have been taken from layer 3. There is no sampling of components at the periphery of the fuel bricks, such as the interstitial bricks or the keys. Each sample trepanned from the core is cut into several slices and the density, as well as other material properties, is measured.

The fuel in the fuel channel imposes a through wall flux gradient which decreases as a function of distance from the fuel. Since weight loss is controlled by gamma decomposition of the CO_2 within the porous structure of the graphite; the graphite core components exhibit a weight loss profile similar to that of the flux gradient. Therefore, peak weight loss is observed at the bore of the fuel channel closest to the fuel. Furthermore, sampling is restricted to the central part of the core which is subject to the greatest irradiation; therefore sampling is conducted in the highest weight loss regions of the core.

The average weight loss limit applies to a core volume that includes layers 2 to 11. Therefore, the average graphite weight loss requires consideration of the weight loss in components that lay radially and axially outside of the sampled region. This means prediction of average core weight loss is subject to greater uncertainty than the peak weight loss and requires greater scrutiny.

HSL took the following approach to the problem:

- Develop a radial brick density model based on the trepanned sample data from HPB AGR power station.
- Extrapolate the model to non-sampled parts of the core defined by the average core weight loss.
- Develop an approach to convert density to weight loss accounting for graphite shrinkage behaviour.
- Make temporal predictions for the progression of average core weight loss at HPB.

HSL found that density measurements to a depth of 45mm could be adequately fitted using a linear model. However, the deeper 75mm deep density data from 2011 – 2014 suggested that the trend was non-linear deeper into the brick. This was consistent with the expected flux profile, therefore HSL opted to use a non-linear model. The model allows for temporal differences in the rate of density change through the depth of the brick and features random effects parameters to account for sample to sample variability.

The through brick density profile was extrapolated beyond the extent of the dataset to a depth of 90mm, the full depth of the fuel bricks. It was found that fitting the model exclusively to the 45mm deep data and extrapolating it to the brick periphery gave a reasonable prediction of the 75mm data. This provided confidence in the form of the model and the final extrapolation to the brick periphery using both the 45mm and 75mm deep data (Figure 3).

To extrapolate the model and hence predict the densities of components beyond the fuel bricks, such as interstitial bricks, HSL assigned the density predicted at the radial extremity of the fuel brick to all components beyond this point. Since flux decreases with distance from the fuel, it is reasonable to assume that weight loss will decrease as well; therefore, assuming a constant density beyond the outer circumference of the fuel brick is likely to lead to a conservative conclusion. It was assumed that the dataset accounted for any angular variability in density profile.

In order to extrapolate the model to non-sampled axial heights within the description of the average weight loss limit (layers 2 -11), it was necessary for HSL to scale their radial model to axial heights beyond the dataset. In order to determine a reasonable basis for the scaling factor HSL consulted graphite oxidation experts from GTAC. It was assumed that the gas chemistry would be relatively uniform over the height of the core and that the dose variation with height, particularly for the peripheral layers, would have the greatest effect on the density. It was found that the random effects parameters within the radial density model adequately described the variation in density of graphite in all layers of the sampled dataset (layers 4 to 9). It was assumed that the dataset adequately described any variability in axial dose over this height and its resultant effect on density. This assumption may not be reasonable if the sampling was biased towards axial heights of either high or low dose e.g. fuel end dose depressions and will need further investigation. If it is discovered that the dataset does not adequately describe the variability in density as a result of axial height differences in dose it may be necessary to include a height related damage metric in the model. However the current model assumes uniform behaviour over layers 4 – 9.

To extend the model to layers 3 and 10 HSL assumed that the flux remains similar to that across layers 4 - 9. This is likely to be conservative as the flux will begin to reduce in these peripheral layers and therefore the oxidation and dimensional change rate will decrease. Assuming the same flux profile for layers 2 and 11 as layers 3 and 10 was considered too onerous given that the flux would be considerably lower than in layers 4 – 9. Therefore, HSL decided to restrict the current form of the density model to layers 3 – 10.

To convert the output from the density model to graphite weight loss it was necessary to account for the volume change attributable to irradiation induced dimensional change. HSL consulted experts at Nuclear Graphite Research Group (NGRG) based at the University of Manchester. ONR funds the Nuclear Graphite Research Group based at the University of Manchester to develop independent models to determine the development of internal stresses within graphite core components as a result of dimensional change. The NGRG finite element (FE) model of dimensional change is tuned to inspection data of the measured changes in fuel channel bore diameter as a function of irradiation. Using this model NGRG supplied HSL with data describing the dimensional changes of graphite within the core as a function of core irradiation. HSL used the NGRG dimensional change model to determine weight loss from predicted density; however, the model is restricted to layers 3 – 10.

As stated, layers 2 – 11 are subject to significantly lower flux and would therefore be expected to experience less dimensional change and radiolytic oxidation. In the absence of sample data upon which to base satisfactory models, HSL used the output from licensee's mechanistic model of graphite weight loss to determine a scaling ratio that could be applied to layers 2 and 11. HSL compared the licensee's predicted weight losses at the midpoint of bricks in layer 7 and layer 11. The average ratio of the two was then applied to the HSL's predicted weight loss results from layer 7 to determine weight loss in layer 11; layer 2 was taken to be equivalent to layer 11.

Based on the HSL model the earliest central estimate for when the average core weight loss would reach 15% in the reactors at HPB was estimated at 21.5 TWd. However, it should be noted that this prediction is a significant extrapolation beyond the available dataset both spatially and more significantly temporally. The licensee's own predictions were made using a mechanistic model and consider not only the progression of weight loss at HPB but also its sister station at HNB. The licensee's current prediction on the progression of average core weight loss for HPB and HNB is that the earliest core irradiation at which the average core weight loss could be reached is 17 TWd, somewhat earlier that the HSL prediction based solely on HPB inspection data.

DISCUSSION

HSL's work has shown that spatial and temporal extrapolation of a statistically based model beyond the dataset incorporates increasing uncertainty. This is because there are a number of influencing variables and how these variables change and their effect on weight loss is not fully described by the dataset. Temporally, accounting for the rate and nature of dimensional change becomes increasingly difficult particularly at the point of dimensional change turnaround. The NGRG FE model of dimensional change uses leading data from historical MTR experiments to describe the future dimensional change behaviour. Differences in the prediction of the onset and rate of this behaviour between the licensee and NGRG are factors which would result in differences in the predicted weight loss. Determination of the true dimensional change behaviour of the graphite core can only be confirmed by continued measurement of fuel channel bore shrinkage. However, extrapolating this to other parts of the core introduces its own is uncertainty as creep will act to modify the dimensional change behaviour from that of the bore.

Spatially, deeper sampling improved the through brick density profile of HSL's model and reduced uncertainty of extrapolation to the periphery of the brick. Extrapolation to layers where there is either limited data (layers 3 and 10) or none (layers 2 and 11) introduces significant uncertainty as a statistical approach is no longer valid. HSL performed a crude scaling to account for the weight loss in these layers in order to provide an estimate of weight loss comparable with the average core weight loss limit and it was accepted that this could introduce significant uncertainty and would lead to differences in predictions of the rate of average weight loss. The work also highlighted that it is the relatively low average weight loss in peripheral layers that has a significant effect in reducing the overall average core weight loss. Therefore, accurate prediction of the weight loss in peripheral layers is important in predicting the overall average weight loss and sampling of these layers would be beneficial to improving the accuracy.

HSL are now working to understand the differences between the two predictions. The work performed thus far has been valuable in highlighting the areas of uncertainty in predicting average core weight loss based solely on sampled data and that uncertainty increases outside the bounds of the dataset both spatially and temporally. It demonstrates that a soundly based mechanistic model is valuable in making predictions of average core weight loss to reduce uncertainty. It has also highlights the importance of regular inspection and sampling to confirm or retune forward predictions.

In this respect ONR has influenced the licensee to significantly increase the inspection and sampling of all its AGR cores. Over the past 15 years the number, frequency, depth and layers sampled have all increased and resulted in a significant body of data relevant to the operating AGRs with which to compare predictions. Furthermore, the licensee has in the last ten years embarked on further MTR experiments which include graphite taken from the AGR's to show that the basis of a number of their mechanistic models including ones that effect determination of weight loss is sound.

CONCLUSION

The example discussed demonstrates ONR's regulatory approach to graphite moderated AGR's in the UK. The approach is to introduce and develop independent and diverse thinking on problems posed by the UK's unique graphite moderated CO_2 cooled reactors by contracting suitable organisations to carry out programmes of work both individually and in collaboration. The justification for this approach to graphite moderated reactors is that without this funding the majority of expert knowledge and scientific interest would lie predominantly within the licensee's organisation. Therefore, without ONR's creation of separate pools of expertise, the licensee's group would not be subject to an appropriate level of scientific challenge when compared with other areas of nuclear power such as PWRs, where the scientific body is much larger and composed of groups with differing agendas. Therefore, it is important that ONR funds such bodies to stimulate fresh thinking and reduce the likelihood of 'group think' that may lead to possible risks to nuclear safety being ignored or going unchallenged. Furthermore, the availability of a pool of independent expertise and functional models assist the ONR understand where uncertainty lies and reach informed regulatory decisions about the adequacy of the licensee's safety cases.

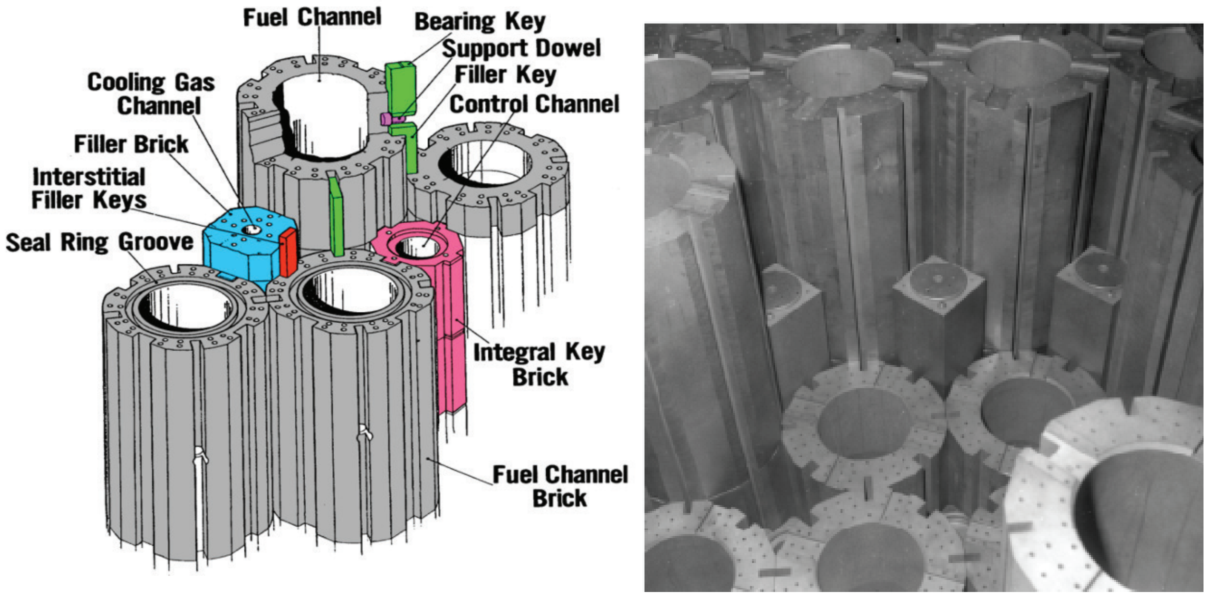

Figure 1: Description of the AGR core components (left). An AGR core in the process of being built (right) showing the graphite bricks.

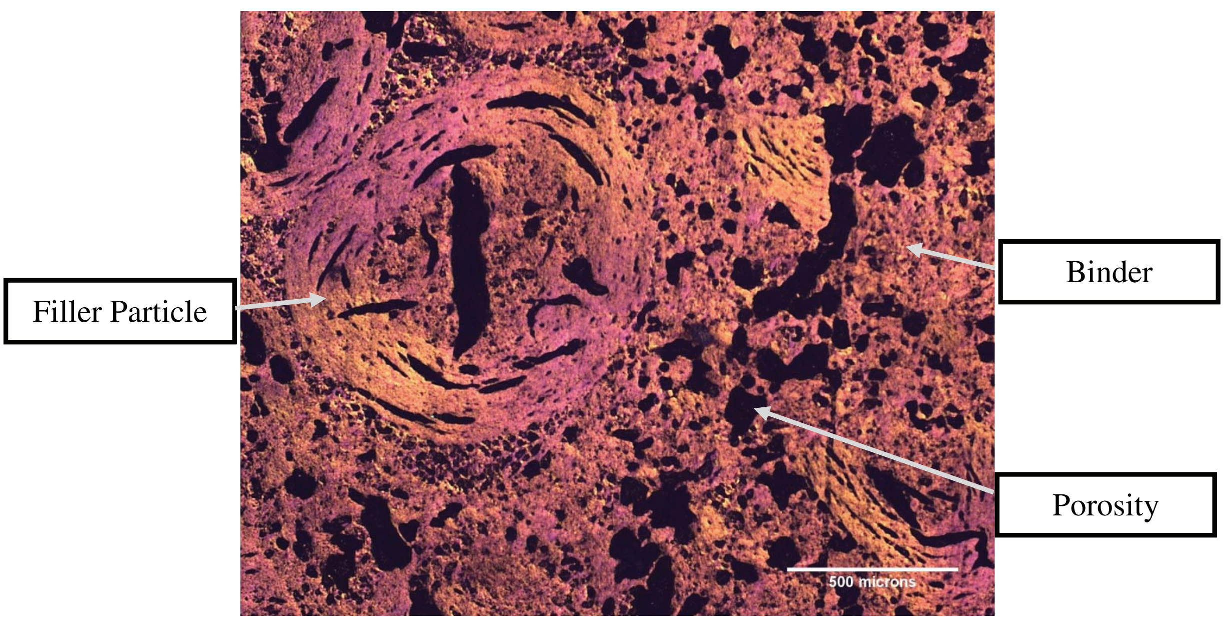

Figure 2: Microstructure of Gilsocarbon graphite.

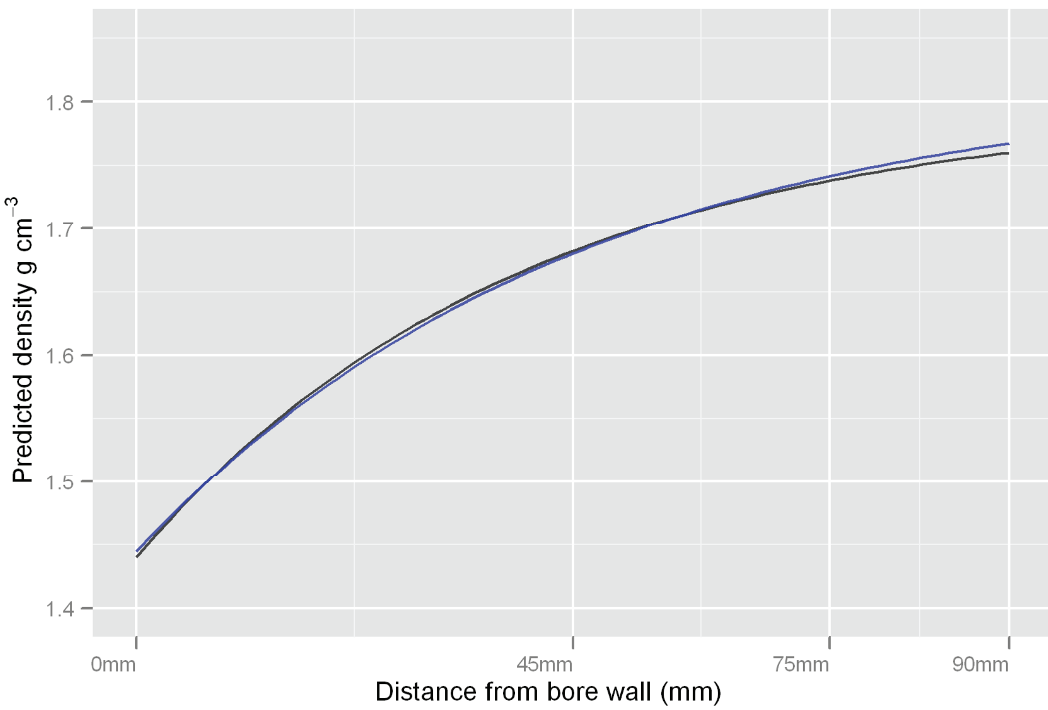

Figure 3: Comparison of through brick density profiles. Black line represents the fit using the full data; blue line represents the fit excluding the 2011 deep cutter samples.

REFERENCES

Brocklehurst, J.E., Brown, R. G., Gilchrist, K. E., Labaton, V. Y., (1970), "The effect of radiolytic oxidation on the physical properties of graphite", *Journal of Nuclear Materials*, v35(2), p. 183-194.

Cundy. M. R., Von Der Hardt. P., Loelgen. R. H., (1977) *Measurement of Irradiation induced creep in Nuclear Materials*, North Holland Publishing Company, Section II.

Kelly, B. T., Brocklehurst, J. E., (1977). "UKAEA Reactor Group Studies of Irradiation-Induced Creep in Graphite", *Journal of Nuclear Materials*, v65 p79-85

Kelly, B. T., Martin, W. H., Nettley. P. T. (1966). "Dimensional Changes in Pyrolytic Graphite under Fast-Neutron Irradiation", *Philosophical Transactions A*, v260, p1109.

Marsden, B. J., (2003) "AGR and Magnox, Nuclear Graphite Technology, Strategic Review, Report reference NGRG R117 Issue 1." Manchester School of Engineering, The University of Manchester.