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NUCLEAR SAFETY COMMITTEE

NP/SC 4927 Revision 1 Addendum 3 Issue 2 Revision 1
Oldbury Power Station

TITLE – A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation:
Update of the Safety Case for Reactor 2 Following the Re-Classification of the Orientation of Some Graphite Sample Strength Data and Discovery of Cracks in Top Reflector Bricks of Reactor 1.

Prepared by: [redacted]

Verified by: [redacted]

Presented by: [redacted]

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This paper has been subject to the in-house procedures including consideration by the Nuclear Safety Committee and the undertaking of an INSA as appropriate.

Signature: [redacted] Date: 30-10-2008

<u>TITLE:</u> **A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation:**
Update of the Reactor 2 Safety Case Following the Re-Classification of the Orientation of Some Graphite Sample Strength Data and Discovery of Cracks in Top Reflector Bricks of Reactor 1

TARGET NSC SUBMISSION DATE: 16th October 2008

<u>AUTHOR</u>

Name [redacted]

Signature Date .. ..

<u>VERIFIER</u>

I confirm that this submission has been verified in accordance with MCP/021/001 and is fit for purpose. An auditable record has been made of the verification process. A verification statement follows.

Name [redacted]

Signature Date ......17/10/08.....

<u>SAFETY CASE OFFICER</u>

I confirm that this submission fully describes the proposed modification, and satisfactorily addresses the relevant nuclear safety aspects.

Name [redacted]

Signature Date .. ..

ENDORSED FOR ISSUE BY

Name [redacted]

Signature Date ..21.10.08

All formal correspondence relating to this document should be addressed to the Site Director and marked for the attention of, or copied to, the Safety Case Officer.

VERIFICATION STATEMENT

This paper has been verified and is fit for purpose. An auditable record has been made of the verification process and is contained in the EWST Graphite Group Task File No. RG7280. The verification has confirmed that:

- the presentation of the arguments is clear and that the layout, grammar and spelling are satisfactory,
- the descriptions of the plant, operational arrangements, and the monitoring and inspection procedures are correct,
- the claims made with respect to monitoring and inspections (including TV inspection of R1 and matters related to brick 12 cracking) are correct,
- the graphite integrity assessment is satisfactorily reported,
- the size factors utilised are appropriate for the present case,
- the information from the references is correctly quoted and the conclusions reached on the basis of this information are appropriate and sound,
- the safety arguments are logically based and sound,
- all relevant aspects of the safety case are addressed,
- all significant judgements are identified and are reasonable
- the conclusions and recommendations are supported by the arguments presented.

A copy of the Verification Plan is attached as Appendix 1.

SUMMARY

The prime purpose of this submission is to address two recent issues: the re-classification of the orientation of a number of graphite samples used to provide strength data in a previous submission, and observed cracking in 2 top reflector bricks in Reactor 1. In addition, developments in monitoring and inspection are briefly considered. The submission is focussed on Reactor 2, which is currently operational. Reactor 1 is expected to be out of service for some months, and will be the subject of a further submission before its return to service.

The sample orientation issue was addressed in a statement under Matters Arising to the May NSC meeting and this submission formalises and updates that statement for Reactor 2.

Upon discovery that data had been misinterpreted an interim strength model was developed, based on information available at that time. This model predicts flexural strength significantly lower than the values predicted by the erroneous model. The significant reduction in strength has been offset by removing a previously unquantified pessimism in the representation of the effect of stressed area on strength. A further improvement has arisen from using a more accurate prediction of the graphite weight loss.

The interim strength model has now been shown to be a reasonable representation of data which has subsequently become available from an extended test programme.

On the basis of these results, it is concluded that the limiting values of UF quoted in the previous Oldbury graphite safety submissions are conservative, and the safety case remains valid.

The March NSC meeting was apprised of the 2 cracked top reflector bricks found during TV inspections of Reactor 1. At that time it was considered likely that the observed cracking was caused by interference between the affected bricks and the steel fuel sleeve entering the top of each brick. Intervening measurements on a sample of 20 sleeves have shown a wide range of oxide thickness, thought to be caused primarily by variations in the silicon content of the steel.

It has been concluded that this oxide layer is responsible for the cracked top reflector bricks. The variation in as-manufactured clearances between the sleeves and the bricks, and the additional oxide thickness needed to generate the hoop stress necessary to crack a brick have been evaluated. It has been found that, at minimum as-manufactured clearances and maximum oxide thickness, brick cracking is a clear possibility. However, in view of the very small incidence of observed cracking (2 cracked bricks in over 2500 channels inspected in both reactors), together with the limited potential for further oxide growth within existing core irradiation limits, it is concluded that the future incidence of Layer 12 brick cracking is likely to be small, and will not jeopardise reactor safety.

This view is supported by the very low incidence of top reflector bricks considered to be lifted as a result of interference between the brick and the oxide layer on the sleeve.

The damage mechanism is such that the risk of cracking is confined to top reflector bricks, which are above the top of the uppermost fuel element and the lower tip of the control rods. It is also noted that none of the other possible mechanisms considered has the potential to cause damage to bricks lower down the fuel channels. Hence, the risks of adverse effects on fuel cooling or control rod entry are judged to be insignificant.

On the basis of these arguments it is concluded that the existing graphite safety case for Oldbury Reactor 2 remains valid, and the reactor remains fit for continued operation and subsequent start up in the event of a trip.

CONTENTS

1 INTRODUCTION

1.1 Purpose

The prime purpose of this submission is to address two recent issues: the re-classification of the orientation of a number of graphite samples used to provide strength data in a previous submission, and observed cracking in 2 top reflector bricks in Reactor 1. In addition, developments in monitoring and inspection are briefly considered.

The submission is focused on Reactor 2, which is operating. Reactor 1 is currently out of service, and will be the subject of a further submission before its return to service.

The incorrect assignment of the orientations of a number of installed set samples used for graphite strength measurement led to the production of an erroneous strength model (Reference 1) which was used in the most recent Oldbury graphite safety case submission, Reference 2. As a result, a Matter Arising Statement was presented to the May 2008 NSC meeting (Minutes 8-22 refer). This Statement updated the deterministic hoop utilisation factors (UFs) for Oldbury, based on an interim strength model derived using the available re-classified strength data.

This submission places on record the basis of the Matter Arising Statement, and presents results for Reactor 2. It also gives a preliminary indication of the latest findings of ongoing graphite strength testing.

The discovery of long axial cracks in two top reflector (Layer 12) graphite bricks was the subject of a Matter for Special Report made to the March 2008 NSC (Minutes 202 - 237 refer). This statement reported the initial findings of preliminary investigations, that the cracks were probably caused by interaction between the top brick and the mild steel fuel sleeve attached to the charge pan. Damage would therefore be restricted to Layer 12 bricks, above the fuel, and also above the lower tip of fully withdrawn control rods. Thus, fuel cooling was assured and control rod entry would not be significantly impeded. On this basis, continued operation of Reactor 2 was justified.

The investigations into the cause of the cracking have now been completed. The findings, which are in line with the expectations indicated in the Matter for Special Report, are reported below.

This submission is focussed on these two particular issues. It is intended to present a complete review of other aspects of the case for both reactors as part of a forthcoming submission reporting the development of a probabilistic assessment to support the structural integrity leg of the safety case. Hence, reporting of general progress on the monitoring and inspection legs is confined to noting recent findings. Findings from Reactor 1 are included since these have potential implications for Reactor 2. The consequences leg of the case remains unchanged, and is not discussed.

The revised structural integrity assessment for Reactor 2 is given in Section 2, and the Inspection and Monitoring legs of the safety case are briefly updated in Section 3. The implications for both reactors of the cracked Layer 12 bricks are discussed in Section 4.

1.2 Background to the sample orientation issue

The existing core graphite safety case for the Oldbury reactors is embodied in NP/SC 4927 and its Addenda 1 and 2 (References 3, 4 and 2 respectively). Over the period of these submissions the structural integrity analysis has developed in a number of areas including strength relationships and prediction of weight loss. The latest submission, Addendum 2, updated the shutdown transient (SDT) load case for both reactors and was the only submission to incorporate the strength relationship based on the erroneous data. Normal operation utilisations were presented in NP/SC 4927 (Reference 3) for Reactor 2 and in Addendum 1 (Reference 4) for Reactor 1, but these were not updated using the erroneous strength relationship because to do so would clearly have produced an improvement in what were already considered to be low values. However both of these submissions used conservative estimates of the fuel channel wall (FCW) weight loss predictions which have now been revised.

Graphite bricks are extruded before firing and impregnation. This produces an anisotropic structure and many properties exhibit different values in the parallel (to extrusion) and the perpendicular directions. Strength in virgin graphite is one such property with higher strength in the parallel direction than the perpendicular. However, the erroneous strength relationship noted above (Reference 1), showed no orientation effect. In particular, the perpendicular strength was higher than had previously been derived. This relationship was based on data derived from Oldbury installed set samples with weight loss above 30%.

A subsequent review of materials properties studies led to a concern that the categorisation of certain specimens as parallel or perpendicular might be unreliable. As a result, an investigation has been carried out into the categorisation of the installed set samples used in Reference 1. This has concluded that most of the original 20 perpendicular samples were incorrectly categorised. The strength relationship of Reference 1 is therefore invalid. The NSC has been informed of the investigation into the error at the July 2008 Meeting (minutes 11-22 refer).

2 RE-ASSESSMENT OF UF USING INTERIM STRENGTH MODEL

In this section, the development of an interim strength model is described, including an important change in the use of a size factor to scale from results derived from small test specimens up to full-size reactor bricks. New weight loss predictions have also been developed by re-fitting all of the available data. In References 2, 3 and 4, existing models were conservatively adjusted to take the latest data into account.

Revised estimates of the hoop utilisations have now been calculated (Reference 5) for both the normal operational and SDT load cases using an interim strength model (Reference 6).

Since the submission of Reference 4, a major development programme has been under way to provide a probabilistic graphite structural integrity assessment methodology. The discovery of the misinterpretation of some of the strength data has delayed completion and presentation of a probabilistically based integrity assessment. However, two important enhancements of this methodology, completed several months ago, have now been used alongside the interim strength model to provide updated deterministic values of UF. These enhancements are revised weight loss predictions and a refined assessment of stressed area size effects as described below.

2.1 Re-assessment of strength data

The installed set data used in Reference 1 included 34 flexural tests on beams machined from Oldbury installed sets. Of these samples, 20 were originally categorised as perpendicular and 14 were parallel.

The visual examination of the orientation of the graphite grains on the surface of the remnants of the test sample has been carried out independently by three SQEP staff and the conclusions were broadly consistent. (Reference 7 describes the process for 2006 data). These investigations indicated that only 6 of the original 20 'perpendicular' samples were in fact perpendicular (Reference 6).

The visual tests have been augmented by ultrasonic time-of-flight measurements made in orthogonal directions across the square cross-section of the samples. The ultrasonic velocity is different in the parallel and perpendicular directions. The results from the visual tests and the time-of-flight tests were consistent. The physical examinations combined with investigations into the assignment of sample orientations have revealed the cause of the error to be an inadequately documented assumption that the 'chemical' installed sets were a mixture of parallel and perpendicular specimens, with the orientation of each specimen identified by its position in the carrier. This is the convention for 'mechanical' test samples. In fact the chemical samples in the current dataset were all parallel, (Reference 8).

As a result, it was concluded that only six flexural strength measurements were available from the irradiated 'perpendicular' specimens used in the preparation of Reference 1. This data set is too small be relied upon to provide a statistically meaningful perpendicular strength relationship directly. Conversely, re-assignment of the data increases the number of 'parallel' specimens. This increased data set has been subjected to the same statistical analysis process as had originally been used and a modified 'parallel' strength relationship has been developed.

An interim perpendicular flexural strength relationship has then been derived by applying the ratio of virgin parallel to perpendicular strength to the new 'parallel' relationship throughout the weight loss range. The derivation of this interim model is presented in Reference 6. The effect of this revised relationship is to reduce the perpendicular strength values from those used in the current safety case by a factor which varies with weight loss. The maximum reduction (by a factor of 0.73) occurs at weight losses at or above 30%. For Reactor 2, this weight loss is representative of that at the FCW at the location of the peak UF in the existing safety case (Reference 2).

This interim model was used to derive the results presented in the May 2008 Matters Arising statement, and has also been used in the present submission. The results are discussed in Section 2.4 below.

To supplement the perpendicular strength data set, Magnox have accelerated the testing of perpendicular samples from the Oldbury 2006 installed set test programme. Subsequent to the development of the interim strength model described above, this programme has provided a substantial number of perpendicular samples spread across a weight loss range from about 12% to 42% (Reference 9). The results of this test programme, together with the existing data points are plotted alongside the interim strength curve from Reference 6 in Figure 1. It can be seen that the interim model (solid green line), although developed in the absence of the main body of the data, provides a reasonable representation. (It should be noted that the green line is representative of the small test samples used to provide the data points. The values of strength used in the analysis of square bricks are substantially smaller, as illustrated by the red broken line, and are not to be compared with the data points. A smaller strength is applicable to reactor bricks because they have a larger high stress area than the test samples.)

2.2 FCW Weight Loss Predictions

For the fuel channel wall, statistical predictions of weight loss have been made by fitting a polynomial equation in dose and height to layer-specific measurements for all samples taken to date. Layer specific fits to the Reactor 2 data, developed in Reference 10, provide better fits than the previous approach of providing a single fit for all core heights.

In preparing the previous submissions (References 3, 4 and 2) the weight loss predictions were not based on fitting new weight loss curves to the entirety of the available data. Instead, a conservative approach was adopted whereby previously available predictions were adjusted using enhancement factors, E_f. These factors were used to force the fitted curves through the latest available data, which had not been used in the original fitting process. In the case of Reactor 2, this was a novel approach and an additional level of pessimism was prudently introduced by arbitrarily applying an enhancement of $2E_f$.

Adoption of the data fits from Reference 10 has reduced the predicted FCW weight loss somewhat. The magnitude of the effect on UF in the shutdown transient is considered in Section 2.4.2 below.

The predicted weight losses away from the FCW continue to be based on BEST and are unchanged from those used in the previous structural integrity analyses (e.g. Reference 2).

2.3 Stressed Area Size Factors

Both the erroneous strength relationship described in Reference 1 and the interim strength relationship (Reference 6) include a dependency on the size of the stressed region to take account of the fact that small test specimens are likely to exhibit a higher failure strength than reactor bricks because they are less likely to include initial defects of a limiting size. In previous assessments, the magnitude of this size effect has been conservatively based on the entire brick bore area.

The significant pessimism embodied in the previous value of the reduction factor was highlighted in Reference 4 (Section 5.4.3), where it was observed that the region of high tensile hoop UF was restricted to the fuel channel wall in a small region close to each end of the brick. Subsequent to the submission of Reference 4, further quantitative analysis of the tensile hoop stress distribution, undertaken as part of the development of a probabilistic methodology, has shown that smaller reductions are appropriate (Reference 11). Specifically, the localised area of high stress has been quantified in a conservative manner. This process results in the assessed effective perpendicular flexural strength (relevant in the context of the tensile hoop stresses) of octagonal bricks being increased by approximately 28%, and by 38% for square bricks, compared with using a factor based on the full fuel channel wall area.

2.4 Results

2.4.1 Normal Operation

For compressive strength and parallel flexural strength, the changes to the statistical strength curves are minor and therefore the compressive UFs (which are in any case small) and tensile axial FCW UFs (which are structurally less significant than the hoop UFs) have not been re-calculated.

The 'previous' values in the table below have been taken from Reference 3, which used earlier strength relationships which were not based on the erroneous data. These relationships did incorporate a reduction in the perpendicular strength compared to the parallel strength. The corresponding reduction in strength given by the present interim model is moderate, and is more than offset by the revised size factor and predicted weight loss at the FCW. The maximum tensile hoop UF at the FCW in normal operation is therefore reduced.

Maximum hoop tensile UF at FCW	
Previous (Reference 3)	New analysis (interim model) (Reference 5)
0.46	0.27

Table 1: Reactor 2: Comparison of previous and new maximum hoop tensile UFs in Normal Operation (square bricks)

2.4.2 Shutdown Transient

Previous and new hoop tensile UFs for the shutdown transient are presented in Table 2. The peak values for square bricks with interstitial cut-outs are shown, since these were the calculated peak values in Reference 2. The new results are calculated on the same basis as the previous (erroneous) ones, except the interim strength model described in Reference 6 has been used, together with the revised weight loss and revised size factor based on a reduced stressed area, as discussed above.

For Reactor 2, the combined effect of these changes is to reduce the peak UF by about 22%. As noted in Section 2.1 above, at weight losses typical of the FCW at the core elevation of peak tensile hoop UF in Reference 2, the strength predicted by Reference 6 is about 73% of the erroneous value used in Reference 2. However, the resulting increase in UF is offset by the effect of using the revised size factor (increasing strength by a factor of 1.38 in square bricks) discussed above. The reduction in UF can be explained by the reduction in predicted FCW weight loss used in the new studies leading to significant increases in predicted strength.

Previous UF: Erroneous strength (Additional case from Reference 2, Section 3.3.2)	New UF (Reference 5): Interim strength model (Reference 6), updated size factor and Wt loss
0.65 (layer 6)	0.51 (layer 7)

Table 2: Reactor 2 Peak shutdown transient hoop tensile UFs in square bricks with interstitial cut-outs for various cases.

As noted above, the previously calculated peak FCW UF occurred in a square brick. In the latest analysis, the application of a brick-specific size factor leads to a smaller reduction in assumed strength for square bricks than for octagonal ones, as explained in Section 2.3 above. This feature contributes to the peak calculated UF in an octagonal brick now exceeding that in the square brick. The octagonal peak is now 0.57, somewhat larger than the value for square bricks, but still well below the peak values presented in Reference 2.

2.5 Ongoing Work

The entire data set of available perpendicular flexural strength measurements on installed sets is currently being subjected to statistical analysis to provide a comprehensive model with quantified uncertainties suitable for subsequent use in the probabilistic assessment of fuel clad melt.

The new model will also be compared with the conservative interim model, to confirm that the deterministic results remain acceptable. Subject to a satisfactory outcome, it is not proposed to carry out any additional deterministic analysis.

2.6 Summary

The erroneous model for perpendicular flexural strength has been replaced with a model shown to be a reasonable representation of the additional data which has recently become available. The significant reduction in strength has been offset by removing a previously unquantified pessimism in the representation of the effect of stressed area on strength. A further improvement has arisen from using a refined prediction of the graphite weight loss. On the basis of these results, it is concluded that the limiting values of UF quoted for Reactor 2 in the previous Oldbury graphite safety submissions are conservative, and the safety case for Reactor 2 remains valid.

3 INPECTION AND MONITORING

3.1 Inspection

The inspections of fuel and interstitial channels carried out during a reactor outage provide direct evidence of the condition of the graphite cores following a period of operation. A summary of the current position on inspection is given below. A full review of the results is beyond the scope of this submission, but is planned for inclusion in the forthcoming submission presenting the probabilistic structural integrity analysis noted above.

Reactor 2 is currently operating, and no new inspections have been undertaken since the 100% flattened region inspection programme was completed prior to return to service in May 2007.

In Reference 2, the status of TV inspections in Reactor 1 as of 8th November 2007 was reported. 772 flattened region fuel channels had been inspected (59.2%). Three reportable defects had been identified, none of which was of nuclear safety significance. Since that time inspection has continued and, at 3rd September 2008, >1200 flattened region fuel channels (>92%) had been inspected.)

Since Reference 2 was issued, four reportable defects have been detected. Two of these were long axial cracks observed in two Layer 12, top reflector bricks which are discussed in detail in Section 4 below. The other two were minor damage to the spigot at brick interfaces, similar to defects seen elsewhere, in the past.

3.2 Monitoring

The status of monitoring results on the two reactors was reviewed in Reference 4 (April 2007), when both reactors were shutdown, and TV inspection of Reactor 1 was under way. The basic core monitoring leg of the safety case remains as presented in Reference 4.

Since submission of Reference 4, Reactor 2 has been returned to service (in May 2007), and undergone two controlled shutdowns, 2 unplanned trips from high power and 1 unplanned trip from low power. In each of these cases, the fall in bulk T2 during the first 3 minutes of the transient was within the 110°C limit set out in Reference 2. Since May 2007, no events have occurred which degrade this leg of the case. As noted in Section 1, it is planned to include a full review of monitoring experience in the next Addendum, in which it is planned to present a probabilistic assessment.

4 CRACKS IN LAYER 12 BRICKS

Long axial cracks have been observed in two top reflector (Layer 12) PGB graphite bricks in Reactor 1. The discovery of these cracks and the basis for continued operation of Reactor 2 were the subject of a Matter for Special Report at the March 2008 meeting of the NSC (minutes 202 – 237 refer).

Since that time, extensive investigations and analysis have been undertaken, and justification for return to service of Reactor 1 in the presence of the observed cracked Layer 12 bricks has been issued at Category 2 (Reference 12).

In the following subsections the broader issues, including the likely future extent of this type of cracking and the implications for Reactor 2 are addressed. Although this submission relates only to operation of Reactor 2, the arguments presented in this section are applicable to both reactors.

4.1 Description of Layer 12 bricks and sleeves.

A diagram of the various sleeve arrangements is shown in Figure 2. Above the Layer 12 top reflector bricks is a complex arrangement of horizontal steel plates, suspended from the standpipes. These plates are perforated above the fuel channel and charge channel locations. To form an interconnection between the plates and the Layer 12 bricks, vertical steel fuel sleeves are attached to the plates, and penetrate into the top of each fuel channel.

The external diameter of the fuel sleeves is greater than the bore of the main part of fuel channel, and the top section of each Layer 12 brick has an enlarged counterbore to accommodate the sleeve. The length of the counterbore is such that relative axial expansion and contraction of the core and the above core steel structure can be accommodated. The sleeve is suspended from a spherical mounting, to accommodate small horizontal misalignments between the charge plates and the graphite bricks resulting, for example, from differential thermal expansion.

The diametral clearance between the sleeve and the counterbore is relatively large over most of the length of a standard fuel sleeve, but there is a flared section near the base where the clearance is small.

Similar sleeves penetrate the interstitial channels. Note, however, that these channels are formed by the cut-outs at the corners of 4 adjoining bricks.

4.2 Description of damaged channels

The two channels consist of square bricks, and are in the flattened region, close to the centre of the core.

4.2.1 Channel L11Q3[1]

Two axial cracks have been found in the Layer 12 brick of this fuel channel. The larger crack was visible at the bottom of the fuel sleeve, extending downwards for about 390mm, stopping about 260mm above the lower end of the brick. Its maximum observed gape was 0.8mm, where it disappeared behind the fuel sleeve. The gape was much smaller at the lower end. The crack appeared to deviate from vertical by about 15°.

The smaller crack was diametrically opposite, extended about 190mm below the sleeve, and had a maximum gape of 0.3mm. The brick was lifted above the layer 11/12 brick interface by about 40mm. There were no other notable features in the channel and no debris was observed.

4.2.2 Channel L11C1

Only one axial crack has been found in the Layer 12 brick in this channel, though it is similar in nature to the pair of cracks found in L11Q3. The crack was approximately 190mm long with a maximum gape of 0.2 mm at the top, adjacent to the fuel sleeve. The brick was not lifted above its normal position.

4.3 Summary of Findings Presented in Matter for Special Report

In the statement made under Matters for Special Report at the March 2008 NSC, it was explained that investigations made at that time indicated that the cracking was associated with

[1] L11Q3 is the extended reference to the channel whose normal reference is L13D3

a loading related to loss of clearance between the fuel sleeve and the graphite, and that the problem would therefore be restricted to Layer 12, with no implications for bricks lower down the core. Because the fuel is located below the brick 11/12 interface, out-of-channel coolant leakage caused by brick failure or lifting at Layer 12 has no consequences for fuel cooling. Furthermore, the nature of the observed cracking strongly indicates that neither channel cooling nor control rod entry will be seriously affected by debris. The recent inspection of 100% of the flattened region of Reactor 2 have revealed no similar cracks, giving assurance that there are unlikely to be a significant number of cracked Layer 12 bricks in that reactor.

In addition, Norebore had shown nothing of significance in the affected channels, no debris had been found, and inspection of the associated charge pans had produced no adverse findings.

It was noted that work was in hand to determine oxide thickness and to calculate whether loss of clearance was a plausible mechanism for failure. That work has now been completed, and is reported in Reference 13 and discussed below.

4.4 Summary of Latest Investigations

Reference 13 shows that, at the most onerous tolerances, the oxide thickness required to take up the minimum as-manufactured clearance between the fuel sleeve and the brick at operating temperatures is in the range 260 – 480 μm, depending on the sleeve design. For nominal clearances, the oxide thicknesses are substantially larger, being 937 μm for a standard fuel sleeve.

Finite element analysis representing a square brick has shown that an additional 220 μm of oxide thickness (beyond that required to fill the clearance) is required to produce hoop stresses capable of initiating cracking (Reference 12). Hence, it can be seen that cracking of square bricks might occur with oxide layers in the range 480 – 700 μm for the minimum as-manufactured clearances, rising to over 1100 μm for nominal values. Somewhat greater thicknesses would be required to crack octagonal bricks, because of their larger through-wall ligaments.

Oxide thickness on the inner wall of 20 fuel sleeves has been measured using an eddy current probe. Two sleeves were found to have thicknesses of > 700 μm. (700 μm is the limit of calibration of the eddy current measuring device.) One of these sleeves was in channel L11Q3, the double cracked brick, and the other was in a stronger octagonal brick, which was found to be lifted by 60 mm, but showed no signs of cracking. Lifting of a brick by interference between the sleeve and the brick is considered to be a potential precursor of brick cracking.

For the other 18 measured sleeves, the thickness was in the range 150 – 475 μm. The channel observed to contain a single-cracked brick (L11C1) was not accessible to the eddy current probe because this channel contains a stainless steel repair sleeve inserted after repairs to its fixed CGO thermocouple pocket.

High oxide thickness is influenced to some extent by channel lifetime average outlet temperature However, Reference 13 notes that the specification of the steel used to manufacture the sleeves does not preclude the use of semi silicon-killed steels, which could exhibit oxide growth rates 50% greater than those of fully killed steels. It is therefore considered that the observed cases of very high oxide thickness are more probably due primarily to low silicon content of a batch of fuel sleeves, rather than an extremely high lifetime average temperatures.

From these findings it is evident that, for sleeves with oxide thickness in the typical range (150-475 μm), even the most onerous clearances are unlikely to give rise to cracking. For cases of very high oxide thickness, cracking remains unlikely except in cases towards the minimum of as-manufactured clearances. The risk of cracking for the stronger octagonal bricks will be smaller than that for square bricks.

The likely future extent of Layer 12 brick cracking is discussed in more detail in Section 4.5 below.

Alternative cracking mechanisms considered

Several alternative mechanisms for causing the cracks have been considered. Misalignment of the channels is considered unlikely because Norebore inspections indicate no abnormalities in ovality, tilt or bow in the affected channels. Charge-pan distortion or misalignment are also unlikely causes because TV inspections showed no abnormalities, and other channels using same charge-pan structures are undamaged. Note that the two affected channels are associated with different charge-pan structures.

If the spherical bearing at the top of the sleeve were to seize, side loads could be transmitted to the top brick. However, the two affected channels are near the centre of the core, where differential rates of radial expansion are likely to be at a minimum.

Finally, there is the possibility that debris has entered the clearance between the sleeve and the wall of the graphite brick. There is no visible evidence of debris, but inspection between the sleeve and the graphite wall is impractical. A graphite sample was lost in this vicinity in 1982. The 'C' ring debris from the oxidised FCGO pocket in channel L11C1 was recovered in 1993. Although it is impossible to be certain, it is concluded that if debris is involved, it is only in a contributory capacity.

None of these alternative mechanisms is judged to be the primary cause of the brick cracking for the reasons outlined, but this does not eliminate them as possible contributory factors. However they all relate only to the Layer 12 bricks and the loading from the fuel sleeve. Therefore there is no reason to suppose that damage at other core elevations would be induced.

Implications for Core Restraint Rod Anchor Locations

The discovery of graphite damage caused by steel oxidation has prompted a review of other potentially similarly affected areas. In particular, the core restraint rods (also known as puller rods) are anchored in cylindrical recesses in solid bricks in the side reflector. The potential for damage to these bricks by oxide growth of the steel anchor has been considered (Reference 14).

It was found that, even if cracking were to occur, the extent would be limited and the anchor would be expected to remain effective. There is a high degree of redundancy on the restraint system, and if the anchor point were to become ineffective, its function would be transferred to its neighbours. There are no debris or fuel cooling implications with the postulated cracking.

It is concluded that oxide growth of the core restraint anchors will not jeopardise safe operation of the reactors.

4.5 Significance of Layer 12 Cracks

On the basis of the probable mechanism identified above, it is concluded that loss of diametral clearance is the primary cause of the observed cracks in Layer 12 bricks, and that the phenomenon is confined to Layer 12. This is strongly supported by analysis of clearances and measurements of oxide thickness discussed in the previous section.

As noted in Section 4.3 above, cracking in Layer 12 bricks is very unlikely to affect core cooling or control rod entry. Reactor temperature protection would not be affected because the trip thermocouples are located on the fuel, below the height of the Layer 12 bricks.

Impairment of fuel cooling or control rod entry by debris are considered to be very unlikely because of the retaining effect of the steel sleeves at the top of the fuel and control rod interstitial channels, the small Wigner gaps at Layer 12 and the likely shape and size of any graphite fragments. The probable nature of the loading would suggest that cracking is likely to cause bricks to separate into two halves, consistent with the observed defects, rather than to form smaller fragments or segments which might become dislodged. Large fragments would be located by the fuel sleeves and the small brick clearances.

Surrounding Layer 12 bricks would be held in position by their own fuel sleeve, and by the keying and spigotting to other bricks. Furthermore, the risk of cracking is lower in octagonal bricks than square ones, and groups of adjacent bricks are therefore unlikely to be affected. Thus, significant disruption of Layer 12 bricks may be discounted, even in the extreme case of fragmentation of an affected brick.

Likely future extent of Layer 12 brick cracking

The above arguments that Layer 12 cracking will not affect fuel cooling due to out-of-channel leakage, fuel cooling or control rod entry due to debris, or control rod entry due to disturbance of core geometry are not considered to be sensitive to a substantial increase in the incidence of this mode of cracking.

The investigations of oxide growth and as-manufactured tolerances discussed above indicate that cracking will be limited to channels with very high sleeve oxide thickness arising from a low silicon content combined with clearances below the nominal values. This indicates that the incidence of cracking is likely to be small. However, the oxide thicknesses are based on a very limited survey, and the distribution of as-manufactured clearances is unknown.

Additional information on current clearances can be gained by considering the number of lifted Layer 12 bricks observed at the recent outages. The majority of observed cases of lifted bricks have been associated with interference between the brick and the thermocouple pockets in FCGO channels, which is an entirely separate mechanism. Only those cases unassociated with FCGO channels are considered.

In the current Reactor 1 TV inspection programme, at the time of writing, fewer than 20 lifted non-FCGO Layer 12 bricks have been observed. In the 100% inspection of the flattened region of Reactor 2, fewer than 10 such lifted bricks were observed. Two of these were adjacent to an FCGO channel with a lifted brick.

It is also noted that no cracked Layer 12 bricks were observed in the 100% inspection of the flattened region of Reactor 2.

Given the very low incidence of failures and lifted bricks, it is evident that the affected channels are in the extreme tail of the distribution of relevant parameters. It is also estimated in Reference 13 that the extent of oxidation will not increase by more than 5% in the next 2 years of operation, supporting the conclusion that there will be no rapid increase in the incidence of cracking during the remaining period of generation.

4.6 Top Reflector Brick Cracking: Conclusions

From the above arguments it is concluded that:

1. The observed cracking on Reactor 1 is primarily caused by oxide growth on the steel fuel sleeves, and the damage mechanism will be confined to Layer 12 bricks. The extent of brick-sleeve interaction in Reactor 2 is expected to be similar to that in Reactor 1.
2. This mode of cracking does not prejudice safe operation of either reactor.

3. The future incidence of Layer 12 brick cracking is likely to be small, and will not jeopardise reactor safety within the current MCI limits for either reactor.

5 ASSESSMENT AGAINST THE SAFETY REVIEW GUIDEBOOK

5.1 Nuclear Safety Principles

Appendix B of NP/SC 4927 Rev.1 (Reference 3) provided a review of compliance against the Nuclear Safety Principles (NSPs). This compliance was considered to remain valid for the subsequent Addendum 1 (Reference 4). Since the present Addendum does not materially change the basis of the safety case, the existing review is considered to remain valid.

6 FURTHER WORK

6.1 Structural integrity analysis methodology

As noted in Section 2.5 above, the perpendicular flexural strength measurements on installed sets are currently being used to provide a comprehensive strength model suitable for use in the probabilistic assessment of fuel clad melt. Work on this assessment for the Oldbury graphite cores is well advanced. The results will be reported the Committee as soon as practicable.

7 IMPACT ON OTHER SAFETY CASES

The impact of the revised graphite safety case on other safety cases was found to be satisfactory in Reference 4 and that discussion is not repeated here. The position remains satisfactory.

8 CONCLUSIONS

1. Utilisation factors for Reactor 2 based on an interim strength model described in this submission remain bounded by the peak values reported in the previous presentation of the graphite safety case (NPSC 4927 Add. 2). Therefore the structural integrity leg of the Reactor 2 safety case is not degraded by the error in assigning an orientation to the installed samples. The position on Reactor 1 will be confirmed prior to its return to service.

2. The cause of the observed cracking in Layer 12 bricks is primarily due to oxide growth on the mild steel fuel sleeves. This damage mode will be confined to Layer 12 bricks, and does not jeopardise core cooling or reactor trip and shutdown for either reactor.

3. The existing graphite safety case for Oldbury Reactor 2 remains valid. The reactor remains fit for continued operation and subsequent start up in the event of a trip. Start up will, of course, be subject to the requirements set out in NP/SC 4927 Rev.1 Addendum 2 Rev. 1.

9 CHANGES TO STATUTORY DOCUMENTATION

No changes to statutory documentation are necessitated by this submission.

10 TRAINING

No training requirements arise as a result of this submission.

11 QUALITY ASSURANCE

This paper has been prepared in accordance with the requirements of Oldbury Site Management Control Procedure MCP21.

12 INDEPENDENT NUCLEAR SAFETY ASSESSMENT

An Independent Nuclear Safety Assessment of this submission is being undertaken in accordance with Magnox procedures.

13 RECOMMENDATIONS

Members of the Nuclear Safety Committee are recommended to advise the Chairman to:

(i) NOTE that revised modelling of graphite strength and weight loss has demonstrated that, in spite of the previous error in the derived graphite perpendicular flexural strength, the Reactor 2 safety case is not degraded. The position on Reactor 1 will be confirmed prior to its return to service.

(ii) NOTE that the cause of the observed cracks in Layer 12 bricks in Reactor 1 has been identified and such cracks pose no significant threat to the cooling, trip or shutdown of either reactor.

(iii) AGREE the continued operation of Reactor 2 and subsequent start up in the event of a trip (subject to the requirements set out in NP/SC 4927 Rev.1 Addendum 2 Rev. 1.)

14 REFERENCES

1. MEN/ESTD/GEN/REP/0062/07 Issue 1, Predictions of Flexural and Compressive Strength for Magnox Reactor Moderator Graphite, [redacted] August 2007.

2. NP/SC 4927 Revision 1 Addendum 2 Revision 1, A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation: Updated Utilisations for the Shutdown Transient on Both Reactors, [redacted] November 2007.

3. NP/SC 4927 Revision 1, Oldbury Power Station: A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation: Proposal for Return to Service of Reactor 2, [redacted] November 2006.

4. NP/SC 4927 Revision 1 Addendum 1 Revision 1, Oldbury Power Station: A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation: Proposal for Return to Service of Reactor 1, [redacted] April 2007.

5. MEN/EWST/OLA/EAN/0019/08 Issue 2, Oldbury R1 & R2 Structural Integrity Re-assessment for May 2008 Deterministic Safety Case Review, [redacted] October 2008.

6. MEN/EWST/OLA/EAN/0022/08, Revised Predictions of Flexural and Compressive Strength for Magnox Reactor Moderator Graphite, [redacted] May 2008.

7. MEN/EWST/OLA/REP/0028/08, Issue 2, Oldbury Reactor 1 2006 Installed Sets: Part 1 – Measurements of Flexural Strength and Dynamic Young's Modulus, [redacted] July 2008.

8. OLA/REP/AEI/0003/08 Issue 1, Assigned Engineer Investigation into an error in the orientation assigned to graphite installed set specimens used to derive flexural strength, [redacted] May 2008.

9. EWST Graphite Group Flexural Strength Database: Oldbury Flexural Strength Database Issue 5 Verified 09-09-08.

10. MEN/ESTD/OLA/REP/0091/07 Issue 1, Recalculation of Oldbury R2 Flattened-Region Fuel Channel Weight Loss Predictions Using Layer by Layer Approach, [redacted] February 2008.

11. MEN/ESTD/OLA/REP/0090/07 Issue 1, Consideration of Scale and Standard Deviation on Strength Predictions for use in Graphite Moderator Brick Probabilistic Structural Assessments at Oldbury Power Station, [redacted] December 2007.

12. OLD/MOD/9576 Cat 2 PMP, Reactor 1 – Justification for Return to service with Cracked Top Reflector Bricks, [redacted] May 2008.

13. MEN/EWST/OLA/EAN/0023/08, Oldbury PS: Oxidation Assessment of Charge Pan Fuel Entry Sleeves, [redacted] April 2008.

14. Oldbury Layer 12 Cracked Brick: Implications for Restraint Rod Anchor Locations, [redacted] 20 Feb 2008.

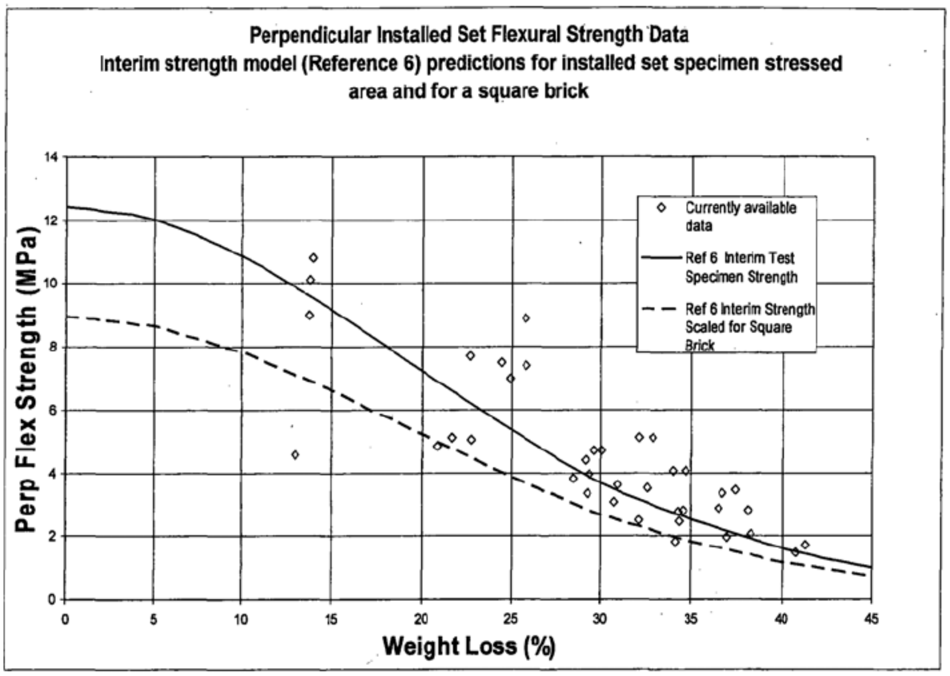

Figure 1. Comparison of Available Perpendicular Flexural Strength Data from Installed Sets (Reference 9) with Interim Strength Model (Reference 6).

The solid line represents the prediction of the interim model (Reference 6) for the test specimens, and can be compared directly with the data points.

The broken line represents the strength prediction of Reference 6 for a square brick, based on the calculated localised FCW area of high hoop stress, and is not to be compared with the data points.

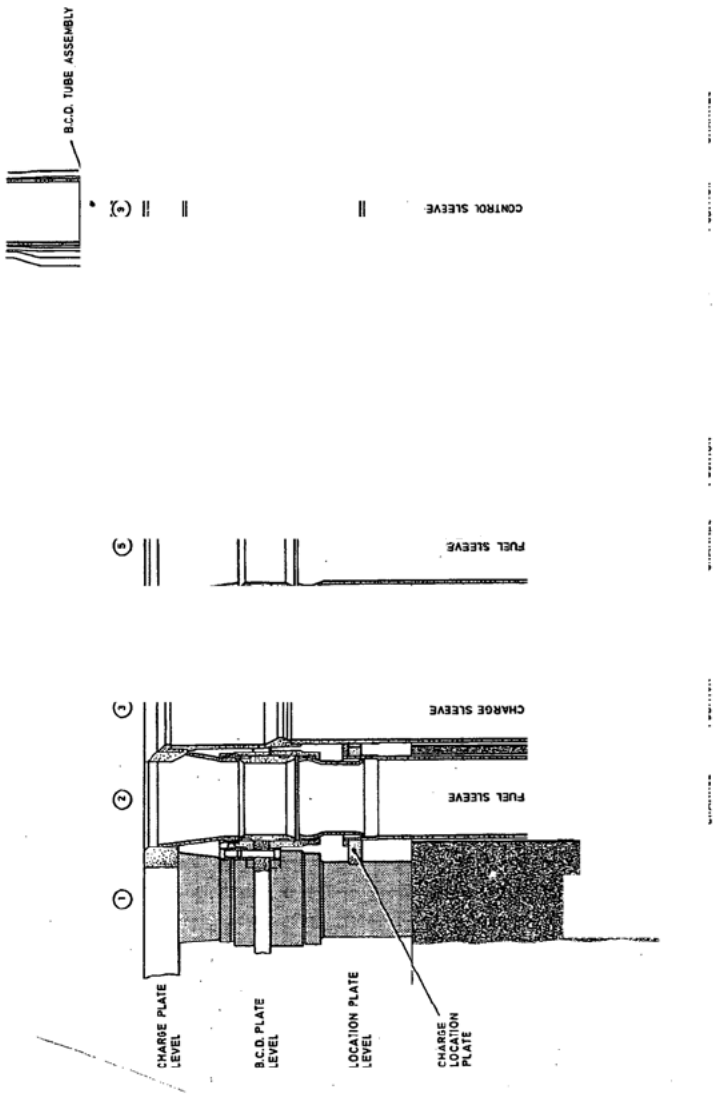

Figure 2. Diagram of a fuel sleeve and a charge sleeve.

APPENDIX 1: VERIFICATION PLAN

VERIFICATION PLAN	No: NP/SC 4927 Add 3 VP010 Issue: 2		Form No: RS/PF/009 Issue 2 Date: October 2001
Author: [redacted]	Approved: [redacted]	9/9/2008	

Document Ref: **NP/SC 4927 Rev1 Add 3 (First Issue)**

Document Title: **A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation: Update of the Reactor 2 Safety Case Following the Re-Classification of the Orientation of Some Graphite Sample Strength Data and Discovery of Cracks in Top Reflector Bricks of Reactor 1**

Date Verification required by: 11/09/08

VERIFIER	DOCUMENT SECTIONS	SCOPE, INPUT DOCUMENT, ACCEPTANCE CRITERIA	VERIFICATION STATEMENT REQ:
[redacted]	All	Please confirm that: The presentation of the arguments is clear and that the layout and spelling are satisfactory. The information from the references is correctly quoted and the conclusions reached on the basis of this information are sound. Confirm with Technical Integrator (and authors of source references if appropriate). The safety arguments are logically based and sound. All relevant aspects of the safety case are addressed. All significant judgements are identified and are reasonable. The presentational standard, grammar and spelling are satisfactory. Please confirm (with the technical integrator and authors of the source references if appropriate) that the conclusions and recommendations are supported by the arguments presented.	Yes

[redacted]	All	Please confirm that: • the descriptions of the plant, operational arrangements, and the monitoring and inspection procedures are correct. In particular, note the descriptions of the fuel sleeves and charge pan and the core restraint anchorage points. • the claims made with respect to monitoring and inspections (including TV inspection of R1 and matters related to brick 12 cracking) are correct. • no additional specialist station verifiers (e.g. authors of technical references) are needed in the areas covered • the conclusions, recommendations and commitment to address R1 in a further submission are appropriate.	Written confirmation to Lead Verifier
[redacted]	Sections 2, 4.4-4.6, 8 and 13	Please confirm that • the verifiers identified on this plan cover all aspects of the submission where specialist verification is appropriate. In particular, confirm that no additional specialist verifiers (e.g. authors of key technical references) are needed. • the graphite integrity assessment is satisfactorily reported. • the supporting reports relating to the graphite integrity assessment have been correctly interpreted and the conclusions reached are appropriate and correct • that the size factors utilised are appropriate for the present case • that it is appropriate to refer to tables of strength data which are not yet the subject of formal issued reports.	Written confirmation to Lead Verifier

Statement by Approver

(i)* I confirm that all verification comments have been incorporated or answered to the satisfaction of the verifiers as recorded on the attached .... no. of sheets or documents.

(ii)* I confirm that all verification comments have been incorporated or answered to the satisfaction of the verifier except where I have otherwise approved the action of the author and all as recorded on the attached ....no. of sheets or documents.

(iii)* I confirm that all verification statements are complete and satisfactory.

*Delete as necessary

Approved Date

VERIFICATION RISK ASSESSMENT				
Risk No.	Verification Component/Description of risk (e.g. input data, calc 1, section 1, etc.)	Error (High/Med/Low)		Specific Mitigation of Risk (mandatory for high prob/high consequence)
		Probability	Conse-quence	
1	Inappropriate citation of results of supporting evidence.	L	H	The supporting EANs have themselves been subject to separate and independent verification. The key risk is therefore restricted to the transfer of incorrect numeric results into the submission. This is readily detectable and does not require any complex processing.
2	Reference to unreported strength data in Fig1	H	L	No specific quantitative use is made of this data. Errors would need to be major to affect the conclusions.
3	Size factors in strength assessment are based on an earlier stress calculation which differ slightly from latest stress analysis	L	H	Consult technical integrator
4	Reference to unverified memo: Watson – Banahan, 20 Feb 2008	L	L	Arguments are of a general nature, based on an understanding of the anchor geometry, and of the safety case for the similar restraints used at Dungeness A. No quantitative analysis is involved.

REVISION RECORD

Revision No	Date	Prepared by	Reason
Add 3 'Issue 1'	April 2008	[redacted]	Probabilistic assessment, prepared for the April 2008 NSC but not presented.
Add 3 Issue 2	October 2008	[redacted]	Deterministic update for Reactor 2 only, pending completion of probabilistic analysis.