NUCLEAR SAFETY COMMITTEE

NP/SC 4807 Addendum 4, Revision 1
Wylfa Power Station

TITLE - **Annual Review of the Safety Case for the Integrity of the Graphite Cores: 2008**

Prepared by: [redacted]

Verified by: [redacted]

Presented by: [redacted]

NOVEMBER 2008

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: .... Date:
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TITLE: **Wylfa PS: Annual Review of the Safety Case for the Integrity of the Graphite Core: 2008**

TARGET NSC SUBMISSION DATE: **19th November 2008**

AUTHOR

Name [redacted]

Signature [redacted] Date 10th Nov 2008.

VERIFIER

I confirm that this submission has been verified in accordance with MCP/021/001 and is fit for purpose. The verification plan is provided as Appendix A and an auditable record of the verification process is retained in EWST Task file RG7225. A verification statement is given overleaf.

Name [redacted]

Signature [redacted] .... Date 10/11/2008

SAFETY CASE OFFICER

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

Name [redacted]

Signature .. [redacted] .... Date 10/11/2008

ENDORSED FOR ISSUE BY

Name [redacted])

Signature .... Date ....

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 in accordance with a copy of the verification plan reproduced in Appendix A and is fit for purpose. An auditable record has been made of the verification process. These records and the completed verification plan will be retained in EWST Task File No RG7225.

Key elements of the scope of verification have been to confirm that:

- The presentation of the arguments is clear and that the layout, grammar and spelling are satisfactory;
- All issues that have been identified in earlier reviews for consideration in this paper are addressed appropriately;
- The information from the references is correctly quoted and that the interpretation and conclusions reached on the basis of this information are sound;
- All relevant aspects of nuclear safety are addressed;
- The safety arguments are logically based and sound;
- All significant judgements are identified and reasonable;
- The conclusions and recommendations are supported by the arguments presented.

Lead Verifier: [redacted] Date: 10/11/2008
[redacted]

- The description of the plant, operational arrangements, outage activities, and monitoring and inspection procedures are correct.

[redacted] …………………… Date: ……………..
SEC, Wylfa Power Station

SUMMARY

This is the fourth annual review of the Wylfa graphite safety case NP/SC 4807, originally presented in 2004. This meets the requirement that annual reviews of the graphite integrity safety case be prepared and presented at Category 1 to the NSC. This review presents relevant developments and additional data which have become available in the last calendar year.

The basic structure of this review remains the same as previously and is based on four 'legs' or elements: inspection, in-service monitoring, structural integrity assessment and consideration of the consequences of significant brick damage.

The previous reviews have all addressed core integrity up to the planned end of generation in December 2010. This review extends the analysis for the first time by exploring the possibility of optimised generation beyond that date, as a sensitivity study.

The following specific issues are addressed:-

- The results of the continuing inspection, sampling and NOREBORE measurements carried out during the Reactor 2 statutory outage in 2008 including operating experience with monitoring the two reactor cores. These continue to be consistent with previous reviews;
- Revisions to the BEST prediction of graphite weight-loss based on the results of measurements that have been obtained since the last review. Other than the consideration of increased mean core irradiations associated with optimised generation, there is no significant change;
- A revised flexural strength relationship is introduced that addresses the encountering in April 2008 of an error in the assignment of the orientation of strength-test specimens that underpin the modelling of graphite strength. Progress with the follow-on actions arising from that error is reported. The proposed strength relationship is based on that advocated to the NSC in May 2008 but includes the quantification, for the first time at Wylfa, of the stressed area factor that had hitherto been pessimised in previous strength relationships;
- Reactor specific re-calculation of the graphite utilisation factors under normal operation and shut-down transient conditions based on predicted 'end of generation' doses in December 2010 and September 2014 has been undertaken and is presented in summary form in this review. The combined results of the revised strength relationship and the re-assessment of UFs indicate no significant change to the axial utilisations and a significant reduction in the more safety significant hoop utilisations;
- A significant improvement to the assessed reliability of rapid operator action in response to the BCD alarms is reported.

Overall it is concluded that the Wylfa cores are judged to be fit for operation until the planned end of generation in December 2010. Furthermore, no issues have been identified that preclude optimised generation up to September 2014.

CONTENTS

1 INTRODUCTION

NP/SC 4807, Reference 1, was the first revised safety case for the Wylfa graphite cores following the last PSR. Thereafter annual reviews of the graphite safety case have been prepared at Category 1 and presented to the NSC. NP/SC 4807 Addenda 1, 2 and 3 (References 2, 3 and 4) provided the equivalent for 2005, 2006 and 2007 respectively. The current paper presents a further review of the safety case for 2008 addressing relevant developments and additional data which have become available since the last submission.

As for all previous reviews, this submission addresses the core integrity safety case for operation of both reactors until the planned end of generation (EoG) in December 2010. For the first time, this review also addresses the sensitivity of the graphite safety case to the possibility of optimised generation beyond that date.

2 OUTLINE OF THE SAFETY CASE

2.1 The Existing Safety Case

The key functional requirements for the graphite core are:

- Maintenance of the geometry of the control rod channels such that there is no obstruction to control rod drop during normal operation and faults;
- Maintenance of reactor core flow geometry so that fuel cooling remains adequate during normal operation and faults, and post trip;
- The safety case recognises that degradation of the core could potentially lead to cracking or crushing of the graphite bricks. Cracking, if sufficiently severe, could permit substantial flow by-pass from the associated fuel channel, degrading the cooling of the fuel in normal operation and in any subsequent fault transients. Brick crushing could lead to debris or collapse of the bricks resulting in flow blockage or impairment of control rod entry.

In demonstrating that the key core functions would be maintained a safety case was presented based on four main elements or legs:

- Inspection;
- Monitoring;
- Structural assessment;
- Consequences arguments.

2.2 Current Review of the Graphite Safety Case

This further review of the safety case follows closely the format that was established in the previous submissions. Whereas the previous reviews considered the case up to and including cessation of generation in December 2010, this review considers the potential for optimised generation beyond that date. The justification for any such proposal is outside the scope of this submission. Nevertheless, it is judged prudent in this review to consider the potential effect of increased graphite irradiation beyond the forecasts employed in previous reviews. In doing so, the assumption has been made that each reactor might operate up to 30th September 2014.

The revisions to each of the legs of the safety case are identified below.

Inspection:

- Results from the R2 2008 NOREBORE inspections;
- Results from the R2 2008 TV inspections;
- Report of graphite sampling activities during the R2 2008 outage;
- The inspection programme up to the planned end of generation in December 2010.

Monitoring:

- An update of operational experience with routine monitoring in both reactors;
- The monitoring programme up to the planned EoG.

Structural Integrity:

- An update of the predicted graphite irradiations comprising revised estimates at December 2010 and new estimates at September 2014;
- Revisions to the forecast weight-losses for both the planned and postulated EoG dates, including consideration of the results of sample testing from R1 in 2007;
- A revised strength relationship is introduced that addresses the error in the assignment of orientation for flexural strength specimens. The underlying strength relationship is that advocated to the NSC in May 2008. However, for the first time at Wylfa this review also includes the quantification of the stressed area factor that had hitherto been pessimised in previous strength relationships.
- Recalculation of the brick stresses and utilisations for both EoG dates;
- Progress on the implementation of a revised shut-down transient assessment procedure, based on the principles outlined in the 2007 graphite review;
- Further structural integrity consideration of the low measured graphite density in R1 Channel 1923/15, bricks 6 and 8.

Consequences:

There are no significant changes to the arguments for the prevention of a single channel fire arising from the consequence of brick failure. However, the current review addresses the combination of deterministic and probabilistic considerations in setting target acceptance criteria for this fault and its escalation to multiple channels. It also reports improvements achieved in the reliability of rapid operator action in response to the BCD system alarms.

3 PLANT DESCRIPTION

The graphite moderator structure comprises the active core, which contains the fuel, and the reflector around it which reduces neutron leakage from the core, Figure 1. The moderator consists of 13 layers of bricks numbered from the bottom and arranged in columns. The upper end of each brick is spigoted to locate in a counterbore machined in the brick above. The columns are composed alternately of octagonal and square bricks. Figure 2 illustrates the layout of the 16 channels associated with one standpipe. In the active core the bricks are sized to provide Wigner gaps between bricks to allow for irradiation induced dimensional changes and keyed together in such a way as to maintain their relative positions under the effects of core expansion. The keyways illustrated in Figure 2 occupy approximately the top 5.5 inches (140mm) of the graphite brick; each brick being approximately 31 inches (790mm) tall.

The columns forming the outer boundary make up an effectively solid graphite shell which is not keyed and has no Wigner gaps and is known as the graphite arch (Figure 1). The arch is held in hoop compression by the core restraint structure. The moderator and arch structure has sixteen sides in plan, with a width across the flats of approximately 61 ft (18.6m) and a height of approximately 33.75 ft (10.3m).

Brick columns comprising the active core are centrally bored to form 6156 vertical fuel channels per reactor each being 3.862 inches (98mm) diameter. The top of each fuel channel is counter-bored to accept channel extension tube assemblies which connect each of the 16 channels associated with a standpipe to the guide tube assembly. In addition to the fuel channels, there are around 767 interstitial channels per reactor, each formed by scalloping the corners of four adjacent brick columns. These holes are used for control rods, flux flattening elements, neutron sources and graphite specimens. The side reflector is approximately 2ft. (610 mm) wide in a radial direction. The side reflector and the top and bottom reflectors of the active core are made from Pile Grade B (PGB) graphite, while all the more central bricks are made from Pile Grade A (PGA) graphite.

The compression in the graphite arch is maintained by pre-tensioned tie bars which are part of the core restraint structure. There are 16 tie bars in a core restraint band and each tie bar is the length of one of the 16 sides of the core. There is one restraint band for each of the 13 layers of graphite bricks. The inner faces of the arch bricks are keyed to the reflector bricks, some of which are specially shaped to fit within the 16-sided moderator structure. The rest of the reflector and the active core is made of standard octagonal and square bricks maintained within a 7.75 inch square lattice. They are laterally keyed together by loose keys set in keyways in the upper 5.5 inch section of the vertical faces of the bricks, as illustrated in Figure 2. The bricks in the bottom layer, which form the bottom reflector, are butted together over the bottom five inches to form a solid disk of graphite. This disk is maintained in position by a restraint band with a double ring of tie bars acting through the bottom core restraint beams unlike the higher brick layers where only one restraint ring is used. The tie bars are a thermally compensated design which expands at a similar rate to graphite and hence minimises the change in the compressive pre-load on the arch as the core and restraint structure warm up. The design of the bottom and side reflectors and the core restraint system ensures the core as a whole expands and contracts as graphite.

Gaps are provided between columns of bricks to allow for Wigner growth of the graphite. The gap is 0.025 inches on all layers except 2 and 3 where it was increased to 0.030 inches to allow for anticipated increased Wigner growth at low temperatures.

Gas leakage from between the layers of bricks in the fuel channel is kept to a minimum by inter-brick seals. These are fabricated from thin Magnox strip and fit into the spigots provided between all active core brick interfaces, except the top two layers where the pressure drop between the channel and the Wigner gap is negligible.

There are 8 fuel elements stacked in each fuel channel. They extend approximately from layer 2 to layer 12 of the active core. This safety case concentrates on these layers in the active core. There are considered to be no significant integrity concerns for the top and bottom layers or the graphite arch because of the relatively low irradiation dose at these levels.

4 INSPECTION AND SAMPLING

This section describes the inspection activities that occur at each statutory outage to support the graphite integrity safety case. These activities comprise inspections using the NOREBORE equipment which provides detailed measurements of brick and fuel channel geometry and TV inspections of fuel and interstitial channels to identify any signs of cracking or brick damage. Removal of graphite samples by trepanning or installed set recovery is also carried out during statutory outages and is discussed in Section 4.3.

4.1 NOREBORE Results from the R2 2008 Outage

The NOREBORE measurement programme for 2008 comprised six fuel channels. Four channels were selected to be on a radius extending over the flattened and un-flattened regions to provide information on the progressive nature of channel profiles. Two additional fuel channels, one in each region, were selected adjacent to an interstitial channel to give information on brick bow. The selected locations included sites that have a previous history of fuel snagging.

A review of the 2008 results is presented in Reference 5. An assessment of the significance of the results is presented in Reference 6. This adopts the extended approach that was presented in the 2006 review of the graphite case as part of a generic review of the contribution that NOREBORE makes to the safety case. The results and assessment show (as for R1 in 2007) that brick bore shrinkage, brick bow and channel bow are continuing to progress as expected and do not indicate any anticipated difficulties for control rod entry, fuel element snagging or other safety related issues.

In 2007 the equivalent review of NOREBORE, Reference 7, reported some evidence that could be interpreted as 'kinking' in the bow profiles of three R1 channels. However the significance with respect to indication of keyway pinching was not clear. The relative misalignment of the bricks was judged to be very small as were the overall channel bows. Similar observations have been made in the review of NOREBORE data for two R2 channels in 2008 although, as for R1 in 2007, no conclusion could be drawn that this was evidence of keyway pinching. Reference 1 argued that keyway pinching would have to be widespread to affect the graphite integrity significantly. Therefore it is judged that the evidence from NOREBORE on both reactors to date is that keyway pinching is not a significant issue for core integrity. Hence the NOREBORE results to date do not indicate any emergent issues that have a significant adverse effect on nuclear safety.

It is intended that NOREBORE inspections will continue at the current scope of six channels at each statutory outage on both reactors until the end of generation, irrespective of its timing. The data will continue to be reviewed with respect to the implications of brick and channel distortion. The specific proposals for NOREBORE inspections at subsequent outages will be identified in the relevant Outage Intent Documents and be subjected to INSA in the normal manner.

4.2 TV Inspections

4.2.1 R2 2008 TV Inspection Scope

A total of 50 channels, including 4 Television Access Stool (TVAS) channels, were planned for inspection during the R2 2008 statutory outage. The inspections combined both targeted and speculative inspections. The targeting of channels is routinely based on a number of factors:-

- Sites recommended for re-inspection by previous Outage Assessment Panels, although there were no specific re-inspections during 2008;
- Trepanned channels (fuel and interstitial) since the trepanning cavities are possible regions of increased local stress;
- Channels with higher pre-shut-down channel gas outlet temperatures (CGOT). These were included since high CGOTs could enhance the shut-down transient graphite stresses;
- Channels subjected to NOREBORE inspection;
- Selected channels which had experienced fuel withdrawal problems;

- TV Access Stool (TVAS) sites where inspection is a condition of the safety case for the revised TV access stool design. This inspection employs the downward viewing 'Insight' camera to facilitate comparison with previous inspections.

With the exception of the TVASs, all of the graphite inspections were carried out using the 'GraphIC' camera which has a conical viewing mirror and high-magnification side viewer built in. It provides a conical view of 100% of the channel wall to a resolution of 0.5 mm or better with the lateral view being used for higher-resolution inspections of specific features as required. The criteria for reporting defects were as follows:-

- All crack-like indications above 10mm long;
- Graphite detachments greater that 1 cubic centimetre;
- Any debris or channel bore obstruction, including protruding Magnox sealing strips;
- Any bricks lifted more than 3mm.

4.2.2 Results of Channel Wall Inspections

The detail of the scope and results of inspections is provided in Reference 8. A total of 8 defects were reported. There were no requirements for re-inspections of items observed during previous R2 outages. The nature of all defects was consistent with the type of observations from TV inspections during earlier outages and consisted of:-

- An instance of a linear gouge mark that is orientated vertically along the channel - judged to have been present since construction;
- Surface features indicative of rub or scratch marks that are judged to be the normal result of in-service fuelling and inspection activities;
- Instances of minor protrusions of the inter-brick gas interface seals at various heights into two control rod interstitial channels. In each case the extent of protrusion was small and the seal was observed to be held captive between the relevant bricks. Protruding gas seals have been observed in previous inspections and are considered to have been present since construction. Impairment of control rod entry is considered to be improbable.;
- One instance of a residual detached piece of trepanned sample resting in the sample cavity, added to the reactor debris list;
- One instance of an un-detached trepanning sample;
- A debris item consisting of a flux scanning bob weight located in the bottom of the control rod interstitial channel at Standpipe 6419, also added to the reactor debris list.

All reported defects were presented to the Outage Assessment Panel. Reference 9 concludes that none has nuclear safety significance.

4.2.3 Proposals for TV Inspections in 2009

The proposals for TV inspections in Reactor 1 at the 2009 statutory outage will follow the established principles for graphite inspection strategy that have been adopted in recent years. The specific channels to be inspected are not yet selected but the criteria for selection will be included in the Outage Intent Document for the 2009 outage which will be subjected to INSA.

The principles of the TV inspection programme include the inspection of around 50 channels including re-inspection of the TVAS channels; the selection being based on the factors listed in Section 4.2.1 and including a significant speculative element.

4.3 Graphite Sampling

4.3.1 R2 Sampling in 2008

Sample Trepanning:

A considerable number of graphite samples have been trepanned from the Wylfa reactors over recent years to meet the data requirements for the structural integrity safety case. During the 2008 outage of Reactor 2, samples were trepanned from bricks 2 to 12 in 4 fuel channels and 4 interstitial channels equally divided between the flattened and un-flattened regions. Following the encountering of abnormal weight-losses at standpipe 1923 in Reactor 1, Reference 4 committed to undertaking further investigations into in-brick variability and all trepanning was undertaken in 2008 by removing specimens from the fuel and interstitial channel walls of the same bricks. The height and orientation of the samples and the post trepanning inspection procedure were designed to ensure that cutting two samples in close proximity was not possible.

The preliminary results of measurement on the trepanned samples obtained in 2008 and the comprehensive results from samples trepanned from R1 in 2007 are discussed in Section 6.

'Installed Set' Withdrawal:

The flexural strength relationships employed in the structural integrity assessment are based on four point bending tests on 'installed-set' graphite samples. Historically these have been obtained on high weight-loss installed sets withdrawn from the Oldbury reactors in 2004, 2005 and 2006. In early 2008 it was judged appropriate to obtain installed-set material from Wylfa. Therefore four out of twelve of the PGA grade installed sets in Reactor 2 were withdrawn from standpipe location 6620 during the 2008 outage. These comprise all the relatively high irradiation specimens at that standpipe. They will be tested to provide flexural strength and other data in support of the safety case.

Supplementary inspections above the fuel channel were undertaken in response to difficulties encountered in the withdrawal of graphite 'installed-set' samples at standpipe 6620. These concerned the recovery of graphite debris and the assessment of the potential for consequential damage following the dropping of the top installed-set sample carrier, Reference 10. All debris was recovered and no core damage was observed.

4.3.2 Proposals for Future Graphite Sampling

The programme of graphite trepanning is expected to continue at future outages, noting that a reduced scope of sampling may be adopted in the final outage due to the proximity of the planned end of generation whenever that may occur. Further installed sets are available as set out in Reference 11. Consideration will be given to the withdrawal of further installed sets in future outages, depending on the safety case requirements and the results of the analysis of the samples withdrawn in 2008. The specific proposals will be identified in the relevant Outage Intent Documents and be subjected to INSA in the usual manner.

In this context it is noted that, due to a mechanical failure during refuelling in August 2008, a withdrawal has been made of the top sample carrier from the R1 1023 location, Reference 12. This material has been retained pending consideration of options regarding its use for material property testing.

4.4 Inspection Programme to the Planned End of Generation

Wylfa currently plans for the cessation of generation in December 2010. The history at other Magnox sites has been to argue a deferral of the final reactor outage as they have approached end of generation at the calendar year-end. A separate submission at the appropriate category will be presented to justify such a deferral at Wylfa should the site intend to adopt a similar strategy. Subject to the option still being considered, it is judged appropriate to address these factors in the relevant outage intent documents that will be subject to INSA in the usual manner.

A strategy for generation beyond 2010 has been prepared, Reference 13. The justification of this initiative is outside the scope of this review. Should it be adopted, the effects on the graphite safety case will be addressed appropriately.

5 MONITORING

The routine core monitoring arrangements are unchanged from those described in the previous safety case reviews and consist of:

- Monitoring of channel gas outlet temperatures (CGOT) on all fuel channels using the Single Channel Scan (SCS) facility on the BCD system to detect the effect of flow bypassing which might be caused by cracked bricks;
- Monitoring of control rod movements and refuelling operations to help identify distortion of the core or the formation of graphite debris.

Following the previous reviews of options to improve monitoring, some enhancements to the CGOT monitoring were made as described in References 2 and 3. They involve the comparison of CGOT, measured via the BCD SCS system, with PANTHER predictions. A set of criteria have been adopted to alert the operator to the potential for evidence of channel blockage or coolant by-pass. These improvements have been in place now for around three years. A ‘fitness for purpose’ review has recently been undertaken of the PANTHER monthly ‘snapshot’ calculational route for all its roles, Reference 14. So far as the graphite monitoring capability is concerned the review is supportive of the arrangements but notes a number of potential improvements which are currently under consideration.

Reference 15 reviews operating experience with the core monitoring arrangements on both reactors from October 2007 to August 2008 and reports that:-

- The continued use of the PANTHER based graphite monitoring tool has demonstrated that there has been no unexplained occurrence of a CGOT temperature difference at the action level(s) on either reactor. This evidence suggests that no serious graphite brick cracking has occurred;
- The monitoring of control rod insertion times during trips and during the R2 2008 outage have been conducted as specified in site procedures and has shown no signs of abnormal drop times or insertion profiles on either reactor;
- On-load control rod freedom of movement checks have been carried out and no problems have been reported;
- The monitoring of difficulties encountered in fuel element removal during re-fuelling has continued. It has shown only a low incidence of difficulty with occurrences being confined to the lowest two fuel element positions and most incidences being related to the bottom fuel elements. Around 1.5% of the channels refuelled in R1 during the period of the review have exhibited snagging. The equivalent proportion in R2 is around 0.75%. These results are consistent with the previous history of fuel movement difficulties that have arisen from the fuel element retention scheme that was suspended some years ago. The incidence of snagging continues to show a general reduction, as expected from the progressive removal of the affected fuel, and do not give rise to any specific concerns relating to channel bore shrinkage. It is noted that the fuel element retention arrangement has been recently re-instated in anticipation of the end of generation in December 2010. The loadings imposed on

the graphite due to fuel withdrawal difficulties were assessed in the 2006 review of the graphite safety case and shown to be less than the bounding shut-down transient loads;

- It is considered that the monitoring procedure and actions limits(s) in use remain appropriate;
- The report also reviews the two shut-down transients that had been encountered at the time of the report (August 2008). A subsequent shut-down in September 2008 has been assessed separately. Section 6.6 considers recent trips and the revised arrangements for assessing their severity.

It is concluded that the monitoring regime is providing useful additional supporting evidence on core condition and it is intended to continue it at its current scope subject to consideration of potential improvements suggested in Reference 14.

6 STRUCTURAL INTEGRITY ASSESSMENT

The basic methodology for the SI assessment and the calculation of UFs is unchanged from the previous review apart from the following revisions to some of the inputs:-

- Updated EoG MCI and weight-loss predictions: these are calculated on a reactor specific basis under two scenarios - cessation of generation in 2010 or at the next PSR review date in September 2014;
- A revised strength relationship for flexural strength. This takes into account the encountering of an error in the assignment of installed set test material to ‘perpendicular’ and ‘parallel’ extrusion directions. This was the subject of a ‘Matters Arising’ statement to the Wylfa NSC in May 2008;
- A systematic re-calculation of the stresses for all brick types under normal operation and shut-down transient conditions. This employs FEAT finite element modelling, as in previous assessments, but is used for the first time at Wylfa to determine the actual stressed area of the bricks. This is employed to reduce the hitherto un-quantified pessimisms for this parameter that were embodied in previous strength relationships.

A report is also made of progress in the implementation of an administrative control to prevent the start-up of a reactor that has been subjected to a shut-down transient (SDT) that is more severe than the pessimised SDT that was presented in the 2007 review.

The predictions for EoG on 30th September 2014 (the next periodic safety review date) are based on a hypothetical model of operation that assumes that either reactor could operate to this date. It is intended to represent a bounding envelope to maximise the flexibility of operation rather than to reflect an expectation that this would be achieved with both reactors. The structural integrity calculations have been carried out using the optimised MCIs and are presented in this submission for information. Any justification for optimised generation would be the subject of a separate submission at the appropriate category that would address the relevant ageing issues for all safety related plant systems but would draw upon the judgments and conclusions of this review in relation to graphite core integrity.

6.1 Mean Core Irradiation Predictions

6.1.1 Planned End of Generation in December 2010:

The previous structural assessment was based on an EoG mean core irradiation (MCI) of 31.5 GWd/t in December 2010. This bounded the then predicted MCIs for both Wylfa reactors. The planned date for EoG for both reactors is unchanged at December 2010 but there are now slightly revised estimates of the EoG MCI which are 30.7 GWd/t for Reactor 1 and 29.0 GWd/t for Reactor 2, Reference 16. These figures assume conservatively that the R2 outage in 2010 is deferred. The structural assessment would therefore continue to be valid if that outage were not to be deferred. The reductions are due to the slight pessimisms that are applied to such predictions, the extension of outages in recent years and the unplanned T/A1 outage in 2007.

6.1.2 Postulated End of Generation in September 2014:

Should a proposal to optimise generation beyond 2010 be agreed, the estimates of EoG MCI in Reference 16 assume reduced full-power operation at 1730 MW(th) for R1 and 1680 MW(th) on R2. These figures compare with the previous value of 1750 MW(th) for both reactors used in earlier predictions. The revised power levels are as currently employed in fuel utilisation predictions, do not apply the effects of possible reduction in time-averaged T2 over the remaining years, and assume statutory outages of 3 months duration, except for the final planned year of operation when it is assumed that a safety case would be made for deferment of the final outage until after EoG.

Based on this model the predicted EoG MCIs at the end of September 2014 are 34.13 GWd/t for R1 and 32.39 GWd/t for R2. Current judgments indicate that there may not be sufficient fuel to achieve generation on both reactors up to the end of September 2014. Therefore the estimates of core MCI for September 2014 are judged to be suitably conservative. Generation beyond this date would require a further periodic safety review for generating reactors and therefore the generation optimisation project strategy, in agreement with the NII, is that generation will not be progressed beyond that date.

6.2 Weight-Loss Predictions

6.2.1 Review of the Basis of Weight-Loss Prediction:

During 2008, consideration has been given to the appropriateness of the basis for predicting graphite weight-loss at Wylfa. This has been done by comparing the predictions of weight-loss made by BEST v2.1 at EoG doses with equivalents obtained by a layer-wise statistical fit method as adopted for Oldbury.

In considering this issue, it was judged prudent to compare the methods of determining the cumulative local graphite irradiation that have been adopted by Oldbury and Wylfa. A calculation was undertaken of the aggregate of the individual fuel element discharge irradiations over the reactor lifetime (using the method that had been employed for Oldbury dose assessments) for 128 flattened zone channels and compared the results with the mean core irradiation and form-factor method that had been adopted historically for the local dose predictions to date at Wylfa. The results, Reference 17, showed that both methods are comparable but that the cumulative irradiation method provides a slightly higher dose than the flattened zone mean irradiation method. However, it was judged that the deviation between the two methods is not sufficient to affect the current predictions of brick properties. It was also judged that no significant improvement would be gained from refitting the BEST predictions to revised cumulative channel irradiations.

Once the validity of the cumulative irradiation assessment had been established adequately, a comparison was made of the BEST predictions of weight-loss with individual layer-wise fits and all-layer (aggregated) fits that were based solely on statistical criteria, Reference 18. Stepwise regression analysis was employed to generate a fit to the trepanned sample results taking into account all the density measurement results up to and including the R2 2006 data. It concluded that there is insufficient data to enable a statistically meaningful layer-wise fit to be adopted using individual layer-wise fits. However, an all-layer fit, when applied to each layer subset does provide consistent weight-loss predictions when the fitted curve is constrained to an accepted virgin density.

The results when compared with the BEST v2.1 model show that, irrespective of the EoG date, the BEST model consistently predicts higher weight-losses in the upper regions of the channels where the shut-down hoop stresses peak. Since this is the region where the predicted utilisations are highest it is judged appropriate and prudent to continue to employ the BEST model at this time.

6.2.2 Review of Recent Trends in Weight-Loss:

The 2007 review of the graphite safety case reported the results obtained for all weight-loss measurements from the trepanning undertaken on R2 in 2006 and the 'fast track' measurements from the R1 2007 outage. It noted the possibility of an upward trend in the data from the R2 interstitial channel wall (ICW) whilst observing that the ICW predictions remain lower for R2 than for R1. Since then a comprehensive review of the Wylfa weight-loss data has been undertaken with this feature considered. The review has comprised the following:-

- An assessment of layer-wise statistical fits compared with BEST that has concluded that BEST v2.1 continues to be the preferred method of performing weight-loss predictions, as discussed in Section 6.2.1;
- A detailed review of the results of all weight-loss measurements from the trepanned samples obtained from R1 in 2007, Reference 19. This concludes that although there is a slight tendency for the fuel channel wall (FCW) results to lie above the mean BEST predictions there is no global trend of elevated weight-loss. This factor contributes to a slight increase in the predicted FCW weight-loss in R1 when the R1, 2007 results are included in the BEST model. Furthermore it reports that the scatter of interstitial channel data is evenly distributed around the BEST v2.1 predictions. It also concludes that although high weight-loss anomalies exist at R1 channel 1923, as discussed in the 2007 graphite review, this is not indicative of a global increase in the rate of weight-loss with increasing dose;
- A preliminary assessment of the 'Priority 1' trepanned sample results from R2 in 2008, Reference 20, has compared the distribution of fuel and interstitial channel wall sample measurements with the BEST v2.1 predictions. Allowing for scatter, the weight-loss measurements are consistent with historical trends of measured data and with BEST predictions.

Based on the conclusions of the above investigations it is judged appropriate to continue predicting weight-losses using BEST v2.1 but to maintain continued vigilance on monitoring the results of sampling to ensure that no early indications of significant divergence of weight-loss from expected behaviour are overlooked.

6.2.3 Revised Weight-Loss Predictions:

As discussed in Section 6.1, two sets of predictions are provided using BEST v2.1. The R1 2007 trepanned weight-loss data has been included into a revised weight-loss prediction, Reference 21, and the results are summarised, on a best estimate basis, in Tables 1 and 2 for the two respective EoG dates.

The tables represent the FC Slice 1 and the IC Slice 3 values for the previous and revised BEST fits at the predicted and bounding MCIs for the bricks with the highest predicted weight-loss. Although BEST predictions are made throughout all bricks and locations in the flattened zone in each reactor, the peak FC slice 1 and IC slice 3 results are usually presented in safety case reviews since the fuel channel wall (FC slice 1) is the most important location for hoop stress induced cracking and the IC Slice 3 position is the maximum weight-loss location. This enables an overview of the revised position to be assimilated rapidly.

Whereas in the 2007 review the BEST predictions were only updated for R2 (the bounding reactor) the current review updates the predictions for both reactors for EoG in December 2010 and in September 2014. For the previous review the Reactor 2 weight-losses at the bounding MCI (31.5 GWd/t) were used in the analysis because the FC slice 1 weight-loss was higher than for Reactor 1. The revised predictions show that the R1 and R2 fuel channel wall weight-losses are very similar at both December 2010 and September 2014. The ability to make a direct comparison of the current predictions with previous equivalents is restricted due to the EoG MCIs in December 2010 being slightly reduced. However, the broad position is that the rate of graphite oxidation in both reactors is not deviating significantly from previous predictions.

The revised BEST weight-loss data has been employed in a FEAT analysis of the cores at the two MCIs, reflecting EoG in 2010 or in 2014. The calculations have been undertaken on a reactor specific basis and are discussed further in Section 6.4.

6.3 Revised Strength Relationship

6.3.1 Previous Strength relationships

The 2007 graphite review presented an overview of the strength relationships that have been employed over the last four years in reviewing the structural integrity leg of the safety case. Specific relationships are employed for ‘parallel’ and ‘perpendicular’ strengths based on the alignment in relation with the direction of extrusion of the graphite, i.e. the channel axis. The relationships employed in 2007 were based on flexural strength data obtained from the installed sets removed at high weight-loss (>30%) from the Oldbury reactors, Reference 22. A statistical stepwise regression model was adopted that predicted flexural strength throughout the intermediate weight-loss range that is more representative of the Wylfa reactors. The model employed the flexural strength data measured from installed sets at high weight-losses and the virgin material properties at zero weight-loss. Intermediate flexural strength values were interpolated using the form of the curve derived from the analysis of compressive strength and weight-loss data obtained from the wealth of historic test results on installed set graphite samples.

In view of the lack of flexural strength data for weight-losses less than about 30%, the shape of the derived strength curve in the range 0-30% depends principally on the compressive strength data. Furthermore, graphite is known to undergo hardening in the early stages of irradiation which significantly increases its strength. Although this effect will be included in the compressive strength measurements taken at low weight-loss, the virgin strength data has not been enhanced to make allowance for irradiation hardening. Hence the strength curve obtained from the overall analysis will be pessimistic in the low weight-loss range.

Throughout all structural assessments the strength relationships are reduced by a ‘biaxiality’ factor of 0.8 to account for the combination of orthogonal stress fields in situations where significant combinations of such stresses are present. This is judged to be a realistic approach in such circumstances. When the results of the assessment indicate uniaxial stress fields the biaxiality factor is withdrawn as appropriate.

The 2007 relationship also included a ‘stressed area factor’ that was derived from the stepwise regression statistical modelling method. This reduces the measured strength results obtained from testing by a factor that represents the larger stressed area of the bricks compared with the stressed area of the test specimens. This approach takes account of the fact that small test specimens are likely to exhibit higher failure strength than reactor bricks because they are less likely to include initial defects of a limiting size. The 2007 relationship made the assumption that the stressed area of a brick is the entire fuel channel wall area. This maximises the 'stressed area factor' reduction in strength to be employed in the structural assessment and is considered to be a significant conservatism in the previous assessments.

6.3.2 'Installed set' Test Specimen Orientations and their Implications

Discovery of the issue:
During 2008 an investigation was undertaken into the flexural strength test results that had been obtained from the installed sets withdrawn from Oldbury in 2004, 2005 and 2006. This was initiated since there appeared to be no significant difference in the parallel and perpendicular flexural strength measurements. These had all been obtained at high weight-losses (> 30%). The investigation was initiated since other graphite material properties continue to exhibit orientational differences at high weight-loss. The investigation revealed that there had been an error in the identification of the extrusion orientation of the installed set material, designated chemical samples, prior to their machining and testing. These chemical samples had not previously been used to provide flexural strength data. A revised assessment process was implemented to validate the orientations of the samples that had been tested. It transpired that, unlike dedicated mechanical test samples, chemical samples were not provided in each orientation and the convention used to allocate orientations to mechanical samples was not appropriate for chemical samples. The implication of this error was that the predominance of the high weight-loss flexural strength tests that had contributed to the previous perpendicular strength relationship had been performed on parallel orientated material. Furthermore, insufficient perpendicular tests had been conducted for a statistically significant perpendicular relationship to be derived. The implication of this is the strength relationship employed in the 2007 review had a likely overestimate in the perpendicular flexural strength and a slight underestimate of the parallel flexural strength.

Justification of continued operation:
A justification of continued operation was prepared immediately and the issue was reported as a 'matters arising' statement to the Wylfa NSC in May 2008, Reference 23. Pending further investigations and additional testing, to be conducted on an urgent basis, the safety justification was based on a re-evaluation of the parallel strength relationship through use of the full dataset of flexural and compressive strength measurements made on the parallel specimens. Thereafter the ratio of virgin perpendicular to parallel strength measurements is applied to the parallel strength to provide a perpendicular strength relationship over the entire weight-loss range, recognising that the most relevant range at Wylfa is up to around 20% weight-loss. The relevant strength relationships are presented in Figure 1 of Reference 24, noting that the full 'stressed area' reduction factor is retained, together with the 0.8 biaxiality factor in this figure.

Since the parallel strength relationship was judged to have been slightly overestimated the axial utilisation factors were considered to have been unchanged or improved. The revised perpendicular strength relationship was used to deduce revised hoop utilisation factors at the fuel channel wall under normal operation and shut-down transient loading conditions. It also produced a scaled assessment of the significance of the perpendicular strength change for the channel 1923/15, Bricks 6 and 8 locations in Reactor 1 where anomalous high weight-losses have been observed. The relevant assessments are presented in Reference 25. They conclude that although small increases in hoop utilisation factors are predicted, the revised utilisations remain bounded by those agreed in earlier safety case reviews.

Continuing work programme:
The source of the original error has been traced to the chemical specimens. Since these had only been used in the safety case to provide flexural strength data there has been no

adverse effect on graphite material properties other than perpendicular strength. Arising from the orientation issue, an action plan was implemented to test further perpendicular 'installed set' graphite on an urgent timescale. Furthermore the Wylfa R2 2008 outage intent document had undertaken to remove installed sets. The intent was that these installed sets be tested to supplement the database of evidence for perpendicular flexural strength, but it was noted at that time that this data was not expected to be available until after the 2008 review of the safety case. The opportunity to employ a reduced 'stressed area' factor at Wylfa, as was included in the equivalent statement to the Oldbury NSC, was noted. Reference 23 undertook to review progress on these issues in this safety case review.

Progress statement

Since the May statements, further testing of installed set perpendicular specimens removed from Oldbury has been undertaken. The results are recorded in Reference 26. A considerable number of perpendicular samples have been tested although their weight-loss range is, as expected for Oldbury installed sets, strongly biased into the high weight-loss (>20%) region. Only four measurements have been obtained in the weight-loss range that is more relevant to Wylfa (up to 20%). This is insufficient to enable an improved statistical analysis to be made of the perpendicular flexural strength relationship over the intermediate weight-loss range. The results are plotted in Figure 3 along with the relationship derived for Reference 23, using the stressed area of the test specimens, for comparison.

The Wylfa installed sets have been obtained from R2 during the 2008 outage, see Section 4.3.1, and they have been transported to Sellafield for testing. An inadvertent withdrawal of the top installed set carrier has been made from R1, see Section 4.3.2, and this is being retained at Wylfa pending consideration of the provenance of the material and its suitability for testing. Currently, work is in progress to improve the arrangements for specimen preparation in the expectation that a less aggressive process can be established. This is judged to be significant since the current method of machining the specimens could be weakening the material. A significant review point that will determine the testing timescale is the resolution of whether a revised machining process should be adopted. The key issue is the need to avoid undue delay whilst ensuring that the optimum specimen preparation and testing regime is employed. This is not expected to be resolved until early in 2009.

6.3.3 Revised Strength Relationship for 2008

Following further work to obtain additional perpendicular flexural strength results, consideration has been given to the form of strength relationship to be adopted for the future, pending the securing of further data from the Wylfa installed sets that were withdrawn from R2 in 2008. As detailed in Figure 3, the available data provides the ability to make reasonable fits in the high weight-loss region (>20%) since most data resides in this range. However, there is still insufficient data available to enable statistically meaningful regression analysis to be undertaken as the sole means of determining a strength relationship over the entire weight-loss range. The key issue is that, without constraining the freedom of the intercept at the zero weight-loss axis, a wide range of statistically valid fits can be generated that diverge substantially in the low weight-loss region.

Valid unconstrained statistical fits can be generated to describe the data in the >20% range, provided their range of application is controlled to ensure that inappropriate assumptions are not made for low or intermediate weight-loss material. However, this is considered to be inappropriate for a core-wide Wylfa assessment since the predominance of the core graphite lies in the weight-loss region where data is scarce.

Since the emergent perpendicular flexural test strength data-set is not adequate to provide a statistically meaningful fit over the entire weight-loss range it is proposed to employ the underlying strength relationship that was applied in the Matters Arising statement. The use of this strength relationship is judged appropriate since:-

- Measured virgin graphite strength properties have been incorporated into the derived relationship without any enhancement arising from irradiation hardening. Hence the strength curve will be pessimistic in the low weight-loss range;
- At high weight-loss, although the method of defining the perpendicular strength is by factoring the measured parallel strength, the result in this region is consistent with the measured perpendicular properties that have been obtained during 2008, as illustrated by Figure 3;
- In the range 0-30% weight-loss, which is applicable to the Wylfa assessment, the strength relationship is based on the compressive strength data obtained from installed sets as employed in the statistically derived relationship used in the previous annual graphite safety case review. There is evidence from the analysis of other strength measurements (tensile strengths on installed set specimens and diametral compressive tests on trepanned samples) that the relationship is likely to significantly underestimate the flexural strength in this weight-loss range;
- The use of this strength relationship for Wylfa has been considered and endorsed by the Graphite Technical Issues Group in October 2008, Reference 27.

This review includes the use of a revised stressed area factor which is less pessimistic than that used previously. The use of the actual stressed area of the Wylfa bricks is based on a comprehensive assessment of the area for all brick types through FEAT modelling in accordance with an established method as set out in Section 6.4. This approach removes a hitherto unquantified conservatism associated with the stressed area factor that was present in previous assessments. It is consistent with the safety case methodology employed on the equivalent deterministic safety case for Oldbury, Reference 28, and is therefore not novel.

The perpendicular flexural strength is more significant in the safety case since this influences the potential for axial brick cracking, which can lead to by-pass of coolant flow. A comparative graph of the perpendicular strength with the equivalent in the 2007 safety case review is shown in Figure 4. This shows an increase in strength for the new statistical relationship compared with the previous model. The net increase is due to a reduction, compared with the 2007 relationship, brought about by the orientation issue that is offset by an increase due to the use of the more realistic stressed area.

The re-assignment of data from the hitherto perpendicular to the parallel data set has enabled a slight increase to be made to the parallel strength via the statistical methodology. The calculated stressed area for axial loads is taken as the geometric area of the fuel channel wall. Therefore there is little change to the parallel strength relationship used for this assessment.

6.4 Revised Utilisation Factors

As noted in Sections 6.1.1and 6.2.3 the revisions to the MCI and weight-loss predictions for EoG in December 2010 are small. However, the possibility of optimising generation has introduced a further set of MCIs and weight-loss predictions for assessment. In addition, FEAT analysis is required to determine the stressed area to be employed in the 2008 strength relationship, Section 6.3.3. It has therefore been judged prudent to extend the calculation scope to a repeated analysis of all brick types, for the flattened zone of each reactor, employing the revised weight-loss predictions at the two target EoG core irradiations. The normal operating condition and shut-down transient (using the pessimised transient introduced in the 2007 graphite review) have been modelled, Reference 29. Since the previous structural assessments have shown that the compressive strength utilisations are consistently low the FEAT assessment in this review is restricted to evaluation of axial and hoop tensile utilisations.

The inputs to the FEAT modelling include the weight-loss and strength data as discussed in preceding sections. The model itself has been updated to provide a more refined mesh structure at the fuel channel wall; the fast neutron flux and nuclear heating have been adjusted to reflect the mean graphite density in later life, rather than early life; the initial gas temperature for the shut-down transient has been adjusted to maintain full consistency with the normal operation value; the locations at which the fuel channel slice weight-losses are provided have been revised slightly to improve the physical representation and axial interpolation has been applied to the BEST predictions of weight-loss data to align with FEAT inputs at the brick mid-height position. Other mechanical properties of graphite including the Young's Modulus and thermal expansion coefficients are based on MTR relationships, trended with reactor data and modified where appropriate; for example DYM is increased by 15%. These have been reviewed in the light of the database of results obtained from trepanning campaigns at Wylfa up to and including the results from the 2007 R1 outage, Reference 30. This has concluded that the emerging trepanned data is consistent with previous monitoring data from Wylfa. It is therefore judged that the ageing effects are not exhibiting any significant time or irradiation based changes that are not addressed in the FEAT model. The results of the FEAT analysis are therefore judged to be valid for forecasting the core structural integrity to EoG, irrespective of its timing, subject to the core inspection/sampling programme continuing to confirm this position.

The method of determination of stressed area follows an established technique that has been developed for Oldbury. It has been subject to an independent method of calculation and has also been assessed via manual checking of the stress visualisation plots that are available from FEAT. The hoop stressed areas at Wylfa have been determined as 2000mm^2 for octagonal bricks and 900mm^2 for square bricks. These results are comparable with the equivalent stressed areas at Oldbury (2850mm^2 and 740mm^2 respectively). The axial stressed areas are unchanged from the previous assessment, being the geometric area of the fuel channel wall. There is therefore no significant change to the applied parallel strength relationship.

The results of the FEAT analysis are presented in Reference 29. The analysis has shown that there is no significant change to the axial utilisations. However, the calculated hoop stresses have increased since the last assessment and the reasons for this are recognised and understood. They originate from the adjustments to the FEAT model discussed above that, broadly, have reduced the already low NOP stresses but increased the imposed additional stress due to the SDT. The effect is a net increase in hoop stress of around 10% during the SDT. This is offset by the net increase in perpendicular strength discussed in Section 6.3.3 and leads to an overall reduction in predicted hoop utilisation factors compared with previous years.

A comprehensive data set of UFs has been produced that confirms the result of previous analyses that the octagonal bricks have significantly higher stresses and utilisations than the square bricks in both normal operation and during the shut-down transient. Therefore, the summary information in this paper addresses only octagonal bricks.

Reference 29 confirms the position argued in all previous safety case reviews that the normal operation hoop utilisations, being less than 0.20, are considerably lower than during the shut-down transient. The analysis uses the 'pessimised trip' SDT that was defined in the 2007 review. The most significant SDT data relates to the peak utilisations at the fuel channel wall and this is summarised as follows:-

Peak Shut-down Transient UFs using the 2008 FEAT model and revised strength relationship, weight-loss predictions and EoG MCIs								
EoG	**R1: Peak FCW UFs**				**R2: peak FCW UFs**			
	Hoop		**Axial**		**Hoop UF**		**Axial UF**	
December 2010	0.73	Layer 10	0.73	Layer 4	0.72	Layer 10	0.74	Layer 4
September 2014	0.77	Layer 9	0.73	Layer 4	0.76	Layer 9	0.76	Layer 4
31.5 GWd/t	0.73	Layer 10	0.73	Layer 4	0.75	Layer 9	0.75	Layer 4

The peak axial UFs listed take account of the absence of any significant hoop stress in Layers 2 and 3 and the biaxiality factor has therefore been removed from the axial utilisations at these layers. This approach is consistent with that employed in previous graphite safety case reviews. The results show that there is little difference between the two reactors for peak FCW utilisations, irrespective of the timing of the end of generation. The more safety significant hoop UFs are slightly higher for R1 although the R2 axial stresses are higher than for R1. In either case the differences are relatively small and are not considered to be significant. The 2007 review used a bounding MCI of 31.5GWd/t for December 2010 and the equivalent results at this MCI are also listed to provide a comparison with the previous assessment. Since the EoG MCI is higher for R1, irrespective of its timing, and the effects of optimised generation are slightly more marked on R1, the summary information relates to that reactor. There are some features of the distribution of utilisations between the various core layers where R2 has slightly higher UFs but the differences are always small and are not considered to affect the safety judgements advanced in this review.

Figures 5a and 5b summarise the fuel channel wall axial and hoop utilisations respectively for all core layers in Reactor 1 under the shut-down transient condition. A comparative plot of the equivalent data from the 2007 review is also provided. Key features of the results are as follows:-

- The axial utilisations are little changed from those presented in 2007 and only show a minor increase if the operational life of the reactors were to be optimised up to September 2014;
- The hoop utilisations show a reduction at all brick layers when compared with the equivalent assessment in 2007. This is predominantly due to the net increase in perpendicular strength discussed in Section 6.3.3, albeit slightly offset by the increase in stress indicated by the recent FEAT analysis;
- The revised hoop utilisations remain bounded by those presented in 2007 even if generation were to be optimised to September 2014;
- The profiles of the utilisations throughout the various layers of the core are not significantly changed by the recent analysis.

These factors add support to the conclusion that although minor changes to the stresses are introduced by the recent analysis they do not indicate a radical change to the understanding of the core ageing mechanism. Furthermore they indicate that the cores can accommodate a significant increase in MCI beyond the analytical limits employed in the more recent graphite safety case reviews.

6.5 Implications of High Weight-Loss Bricks at Location R1:1923/15

The 2007 review discussed the implications of high weight-losses observed in trepanned samples obtained at location 1923/15 and 1923/IC. Two sets of measurements have been taken, the first in the interstitial channel during the 2005 outage that showed an anomalously high weight-loss in brick 6 and the second in the 2007 outage when sampling was repeated at both the interstitial and fuel channel walls of each accessible brick. The 2007 sampling confirmed that the high weight-loss observed in 2005 at brick 6 was a genuine weight-loss rather than a measurement anomaly. A similar proportional increase in weight-loss above BEST predictions was observed at the FCW as had been observed at the Interstitial Channel Wall (ICW). The 2007 sampling also observed a second anomaly in brick 8 where there is a similar high weight-loss result for the FCW of brick 8, particularly at slice 1, but unlike brick 6, the weight-loss at the ICW, as sampled in 2005, was lower than the BEST prediction.

The possible causes of these anomalously high measurements have been explored and the potential for other bricks to be similarly affected was considered in the 2007 review. The most plausible explanation was considered to be that the brick started life with a low density. This could be explained by a lapse in quality control in the manufacture of PGA graphite or substitution of a PGA brick with PGB graphite somewhere in the manufacturing and core construction route. A second possibility that the brick had experienced an accelerated weight-loss was considered less plausible as the weight-loss in adjacent bricks in the channel was in line with predictions. Investigation of the manufacturing and construction route suggested that substitution during core assembly was very unlikely due to geometric differences in the two types of brick and the more plausible explanations were PGB machined as PGA or poor impregnation of PGA.

A justification of continued operation with these high weight-loss bricks was presented in the 2007 submission. It was based on a scaling assessment of changes in Young's Modulus and strength for bricks with anomalously high, but uniform, weight-losses. The 2007 review concluded that there was significant tolerance to anomalously high weight-losses.

6.5.1 FEAT Analysis

Since the 2007 review, a more detailed FEAT analysis has been undertaken of the stress and utilisations for the 1923/15 bricks 6 and 8. It was considered prudent to undertake this analysis since the previously used scaling assessment method did not accurately reflect the different trend of density variation that had been observed in these bricks, particularly the layer 8 brick. Therefore it was judged appropriate to confirm that this did not have any adverse influence on the judgments. Furthermore it was considered necessary to extend the analysis to the higher MCIs that are expected in the event that optimised generation is agreed at Wylfa, see Section 6.1.2. This work was accomplished before the revised BEST predictions had been fully reported and therefore the predicted weight-losses at EoG in December 2010 and September 2014 for the anomalous bricks are slightly at variance with those presented in Section 6.1. However, in view of the underlying structural integrity argument that reductions in strength are largely offset by similar reductions in Young's modulus, indicating that weight-loss is not a strong determinant of UF, it is judged that the slight deviation in predicted weight-loss for the assessment is not significant. The more significant issue is whether the unusual distribution of weight-loss within the brick changes the stress field sufficiently to affect the UF that is derived.

The results of the calculations are presented in Reference 31. These employ the revised strength relationship that is discussed in Section 6.3.3, including the revised stressed area factor. The effect on the FCW tensile hoop UF in these bricks is summarised in Table 3.

The results show that for brick 6 the UFs remain reduced when compared with the revised predicted UF for a normal brick and this is consistent with the scaled assessment undertaken in 2007. For brick 8, however, a marginal increase is indicated. The FEAT analysis has shown that the stress distribution throughout brick 8 is different from normal arising from the different weight-loss distribution: it has relatively low density at the fuel channel wall and high density at the interstitial (a reversal of the usual pattern). In brick 6 the distribution of weight-loss is similar to the norm, although with increased weight-loss. In this instance the FEAT analysis shows a more usual stress distribution and yields a UF that is more consistent with that obtained from the scaled assessment method applied in the 2007 review after taking into account the changed strength relationship.

The conclusion of this assessment is consistent with the understanding of the effects of increases in weight-loss when the brick stresses are predominantly internal. It shows that the core continues to be tolerant to anomalously high and untypical weight-losses even if optimisation of generation were to be achieved.

6.5.2 Continuing Work on High Weight-Loss Anomalies

A working group was established within the Graphite Project to investigate the cause of the anomalous weight-losses and possible areas for further work were identified in the 2007 review. The following results have been achieved:-

- Trepanned sampling from the fuel and interstitial channel walls of the same brick to gather more data on the within-brick variations of weight-loss has been undertaken during the 2008 outage at Wylfa. No significant deviations from the BEST predictions have been observed from the priority test results available to date, see Section 6.2.2. The paired sampling approach will be carried forward into future outage based inspections;
- A retrospective review of historic trepanning locations has been undertaken to identify other instances where multiple samples have been obtained from the same bricks. This has addressed the trepanning campaigns at Wylfa and Oldbury. The results are reported in Reference 32. Two channel pairs have been identified at Oldbury and four additional pairs have been identified at Wylfa. A number of observations have been made that include:-
  i) At Oldbury the fuel channel Slice 1 samples are of consistently higher density than other samples in the bricks. The variations in density at the equivalent interstitial channel are less pronounced;
  ii) At Wylfa there is a significantly less pronounced increase in density at the fuel channel wall. The variations in density at the interstitial channel show marginally lower variations of in-brick variability;
  iii) There are no other instances of significant density anomalies at either site that compare with the observations in bricks 6 and 8 at the 1923/15 channel in Wylfa R1. In making this observation it is noted that the intervening brick at layer 7 displays a significantly higher density (lower weight-loss) than the normal axial trend would suggest.

It is judged appropriate to continue to maintain vigilance for anomalous weight-loss distributions arising from measurements of trepanned samples and therefore to continue to sample both the fuel and interstitial channel walls in at least one brick column in future outages.

6.6 Effects of Other Faults: Shut-Down Assessments

The 2007 graphite review defined a revised approach to the assessment of shut-down severity before a reactor is started up. The principle is to ensure that, should any trip occur that may impose a significantly more onerous loading than the 'pessimised trip' that was defined in 2007, the implications for the graphite safety case will be reviewed prior to start-up. A Category 2 submission has been prepared to implement this arrangement and INSA is in progress, Reference 33. It adopts the monitoring proposals of Reference 34 to translate the pessimised trip transient to a change in the mean of the flattened zone channel gas outlet temperature reductions and sets an upper limit on the temperature drop in the first few minutes, post trip. The previous assessment is to be withdrawn and the revised requirement substituted into the relevant SOIs. This follows a similar approach to that adopted recently at Oldbury. The assessment requirements of Reference 34 are currently being employed along with the original SDT assessment criterion, pending formal approval of Reference 33.

There have been three shut-downs in the period since the last graphite review: the R2 outage shut-down in April 2008, a trip on R2 on 23rd August 2008 and a further trip of R2 on 16th September 2008. The CGOT changes were all less onerous than the 'pessimised trip' that was presented in the 2007 safety case review, References 35 and 36.

6.7 Implications of Top Brick Cracking Observed at Oldbury

During 2008 the continuing TV channel bore inspection programme on Reactor 1 at Oldbury observed significant cracking in two top bricks in the reflector layer. The origin of the cracking was attributed to steel oxidation in the context of the low clearances between the fuel channel sleeve in the top brick and brick bore itself, Reference 28. At the time of its occurrence, consideration was given to the possibility of a similar mechanism being present at Wylfa and this was presented in Section 8 of Reference 9. This submission argued that the layout of the top of the channel is different compared with Oldbury and considerable clearances are present at the extension tube assembly (ETA) region at the top of the Wylfa channels and the top reflector graphite bricks. It therefore judged that the likelihood of such clearances being taken up by oxidation is markedly lower at Wylfa than at Oldbury. Nevertheless, should similar cracking occur, the submission argued that the consequences would not be significant.

7 POTENTIAL CONSEQUENCES OF BRICK CRACKING

The arguments for the prevention of graphite brick failure that include the inspection, monitoring and structural integrity legs of the safety case remain unchanged in principle from those presented in previous safety case reviews. The revised structural integrity assessment presented in this paper has shown that the likelihood of brick cracking continues to remain low. The objective of the consequences leg of the safety case is to demonstrate that, in the event of brick failure, the nuclear safety risk is tolerable and ALARP.

This section presents additional consideration of the hazard/frequency targets based on deterministic as well as probabilistic considerations and reports improvements to the assessed reliability of rapid action by the operator in response to the BCD system alarms that are being achieved in response to commitments to further work in this area in the 2007 review of the safety case.

7.1 The Risk of Clad Melt

Previous addenda have reviewed the risk from clad melt. The radiological consequence of a single channel fire fault in an intact reactor is an estimated off-site dose, using deterministic methodology, of 0.98 mSv, Reference 37. The risk from a single channel fire would be in the broadly acceptable region as defined by the Safety Review Guidebook, Reference 38, under the probabilistic principle, NSP 3, for a single class of initiating events provided the probability of a significant crack in a brick, leading to a single channel fire, was 10^{-3} per reactor year (pry) or less. The upper bound of the tolerable region, under the same criterion, would be 10^{-1}pry.

Further consideration has been given to the upper tolerable targets for the single channel fault and its escalation to multiple channels in Reference 39 . This addresses the deterministic requirements, based on consideration of barriers to release of activity, as well as the probabilistic approach and has concluded that the upper tolerable target for a single channel fault is 10^{-2}pry. This is consistent with the well established target for fuel failure probabilities in faults and is more restrictive than the upper tolerable bound required through consideration of the probabilistic principle alone.

NP/SC 4807 Addendum 1 argued that the broadly acceptable frequency would be met when taking account of the probability of crack initiation, crack opening, consequences arguments and detection by the CGOT monitoring. The revised structural integrity arguments that have been presented in this review and the results of the core inspection and monitoring arrangements continue to support this position.

7.2 The Risk of Escalation of Localised Clad Melting

7.2.1 Frequency Targets

The potential for a single channel fire to spread to adjacent fuel channels was considered in NP/SC 4807 Addenda 1 and 2 and was argued to be a remote event. NP/SC 4807 Addendum 3 presented an assessment of the likelihood of a single crack in a graphite brick leading to multi-channel fuel melting. The assessment was based on the assumption that a single channel fire escalates to the eight surrounding channels and the dose increases accordingly into the 1-10mSv dose band. Under the probabilistic principle, the risk for such a hazard remains broadly acceptable at a frequency of 10^{-4} pry. However, since escalation involves the failure of the graphite barrier and the attendant physical processes that inhibit escalation, a deterministic approach to the sequential failure of two barriers (the fuel clad itself and the graphite ligaments between the faulted and adjacent channels) is interpreted as an upper target frequency of 10^{-4} pry, Reference 39. Again, this is more restrictive than the tolerable level that would be obtained through consideration of the probabilistic principle alone and is judged to be a prudent approach to ensuring that an acceptable position is established for the risk of escalation. Further spread would lead to increasing release but at all times the pressure boundary is intact

7.2.2 Prevention and Protection Arrangments

The inspection, monitoring and structural integrity arguments that largely comprise the ‘prevention’ aspects of the safety case have enabled the judgement to be made in Reference 1 and all subsequent annual reviews that a broadly acceptable position is established for the risk of failure of a fuel element due to a cracked graphite brick. This position is not derived quantitatively and, for the purposes of evaluating the adequacy of the protection arrangements, has been assumed to reside at the upper limit of this frequency range, i.e. 10^{-3}pry, as defined by the probabilistic principle, NSP3, for a single class of faults, Reference 38. Physical barriers and processes are present to inhibit the spread of fire from the initial fault channel. These include the expectation that the erosion of graphite in the faulted channel by molten uranium will be limited, that any flow of molten uranium out of the faulted channel via a crack in a brick will be restricted and that the severity of the thermal stresses to cause cracking in the surrounding graphite bricks will take significant time to be established.

During operation at power, reactor trip is a necessary and sufficient action to ensure that the fault is terminated. The principal protection against escalation of the fault is the role of the operator in responding to the alarms raised by the BCD system.

The 2007 review of these issues noted that a preliminary assessment of the reliability of the operator to trip the reactor rapidly indicated opportunities for improvements. The review committed to:-

- Improve the configuration of the BCD alarm displays;
- Undertake a comprehensive review of the SOI requirements for the BCD system alarms;
- Deliver appropriate training to the relevant staff to reinforce awareness of the safety cases that require the reactor to be tripped rapidly in response to BCD alarms;
- Provide a 'human factors' assessment of operator reliability taking into account the improvements achieved;
- Implement the changes via appropriately categorised submissions.

This has been achieved by changing the tiles of the Group1 BCD 'Bulk Count High' and 'Multiple Group Count High' to a new unique colour of blue with additional discriminatory legends and fonts, the objective being to differentiate these critical alarms from other Group 1 alarms. Appropriate training of the operators is partly implemented to ensure that the increased emphasis on the importance of the Group 1 alarms, as revealed by the human factors assessment, is clearly understood. The Station Operating Instructions (SOIs) have also been reviewed systematically and significant changes to 'Referenced' SOIs (RSOIs) C8 'Action Following Partial or Total Loss of the DP System' and C9 'Action Following Receipt of BCD System Alarms' have been proposed. The changes to RSOI C8 are to clarify the definition of failure of the Data Processing System (DPS), define the availability requirements for the Group 1 alarm system, specify the actions to be taken upon unavailability of the Group 1 alarm system, and to require that in the event that both the Group1 and DPS systems are unavailable prompt action be taken to trip the reactor. The changes to RSOI C9 are to improve the layout of the BCD immediate tripping instruction to emphasise the primary role of the BCD Group 1 alarms and to stress the essential requirement for immediate action based solely on the Group 1 BCD alarms when primed, specify the availability requirements of the BCD Group 1 alarms and the action to be taken in the event of their unavailability, and to specify the availability requirements of the BCD Multiple Group Count High Alarm processing unit. Details of the RSOI text changes are presented in Reference 40. In addition, changes are proposed to SOI C23 'Action on Receipt of Group 2 Alarms' to ensure mutual consistency with the revised RSOIs. An additional step is also proposed for introduction into SOI A1 'Preparation for Unit Start Up' to ensure that planned maintenance of the BCD system is avoided throughout the start-up and the initial period of operation.

The proposals are presented and justified in Reference 41 that argues that the commitments made in 2007 will be achieved once all the proposed work has been completed. Based on a human factors assessment and the implementation of its recommendations through the proposal the assessed probability of failure to trip the reactor within 5½ minutes of the BCD system alarms being raised is 1.3×10^{-2} per demand, under conditions of full BCD system and alarms systems availability. This includes an 'unlearning' factor of 1.5 to account for the change in the emphasis of operator reliance onto the Group 1 BCD alarms instead of the Group 2 BCD alarms. This factor is expected to reduce with time. This is a substantial improvement over the preliminary assessment of the position without the proposed modifications. The net safety achievement is slightly outside the normal expectations for an engineered line of protection but the position is justified in Reference 41 as ALARP.

Reference 41 draws on the Wylfa channel blockage safety case to justify the 5½ minute period for the requisite Human Factors assessment since this reflects the residual time after the initiating event has raised the BCD alarms while the local effects of the fault are expected to be confined within a single channel. The timescale in that instance is governed by consideration of a rapid and severe blockage rather than the likely rate of fault progression in the event of graphite brick failure. It is expected that the rate of reduction in fuel heat transfer efficiency due to coolant flow by-pass will be less severe in the case of graphite failure. Therefore it is judged that the timescale for successful action while the local effects are constrained within the fault channel is likely to be longer in the context of a graphite fault. Nevertheless the assessed reliability figures are employed at face value in Figure 6 to summarise the safety benefit achieved. This has employed an initiating event probability of 10^{-3}pry as discussed previously in this section. It is clear that a substantial improvement in the reliability of rapid trip has been secured in the immediate period after the BCD alarms have been raised and a significant margin is shown. This provides confidence that the acceptability of the safety position for the escalated fault is not critically reliant on the judgement that the risk of the initiating event is broadly acceptable.

Whereas the 2007 review of the graphite case judged that, pending the implementation of improvements to the BCD alarms, the risk of escalation was in the tolerable region it is now concluded that the risk of an escalated fault is reduced to broadly acceptable with the changes implemented.

8 ASSESSMENT AGAINST THE NUCLEAR SAFETY PRINCIPLES

The Nuclear Safety Principles (NSPs) were addressed in detail in the original presentation of NP/SC 4807 and in Addenda 1 and 2. This section summarises the key aspects of the assessment and notes any which have changed in the intervening period. Although the Safety Review Guidebook has been re-issued in July 2008, Reference 38, the changes do not affect the previous assessments.

The assessments are based on the principle that graphite brick cracking is not, of itself, an initiating event and it would require substantial coolant flow bypass and clad melt to produce an initiating event. Predictions based on accepted methodologies for the structural assessment of graphite, which are supported by SQEP judgements and review, demonstrate that the likelihood of coolant flow by-pass in normal operation is very unlikely. The assessment includes analysis of shut-down transients and is supported by appropriate inspection and monitoring activities including the provision of graphite samples for testing to provide supporting data to the structural assessments.

The consequences of brick cracking present no threat to the essential functions of trip, shut-down, hold-down and post-trip cooling even if clad melt were to occur. The hazard associated with coolant flow by-pass leading to clad melt in a single fuel channel is less than 1 ERL.

Two important improvements to the protection arrangements are being introduced:-

- Protection is provided by the operator in response to BCD system alarms. Significant reduction in the risk of escalation of a single channel fault is being achieved through improvements to the BCD Group 1 alarm fascias and specific operator training. The amendments to the relevant SOIs will be implemented on completion of the INSA of the relevant submissions.
- The criterion for the administrative control to inhibit reactor start-up if any shut-down transient exceeds the reference limits is being revised

Both of these improvements are being justified in separate safety submissions, References 41 and 33, respectively.

9 SUMMARY OF SAFETY CASE FOR CONTINUED OPERATION

The safety case for the integrity of the Wylfa graphite cores and its subsequent annual reviews, as presented in NP/SC 4807 and Addenda 1, 2 and 3, has been reviewed and updated in this submission. The basic structure of this safety case remains the same as previously and is based on four 'legs' or elements, namely inspection, in-service monitoring, structural integrity assessment and consideration of the consequences of significant brick damage.

Inspection

50 channels have been inspected by TV camera during the Reactor 2 outage in 2008. There was no evidence of defects which could be indicative of graphite degradation. In addition a further 6 channels were measured by NOREBORE. This confirmed that brick shrinkage was continuing at a rate proportional to the local irradiation and is unlikely to have a significant contribution to refuelling difficulties. Channel profiles were smooth and channel bows were small, indicating that there is no significant core distortion.

Monitoring

The core-wide monitoring of CGOTs against PANTHER predictions has revealed no significant discrepancies which could be indicative of brick cracking and operational experience shows a very high level of channel coverage at each scan. The provision of the CGOT monitoring and the results of the structural assessment combine to give confidence that undetected brick cracking is very unlikely.

The results of routine monitoring of control rod movements provide additional confirmation that there is no graphite debris or core distortion which could affect reactor shut-down.

Structural Integrity Assessment

A comprehensive recalculation of brick stress and utilisations has been undertaken based on revised predictions of graphite weight-loss at the planned end of generation in December 2010. This has been extended to consider the effects of a postulated optimisation of generation on either reactor to September 2014. The assessment has employed a revised strength relationship that now makes allowance for the calculated stressed area of the bricks, thereby removing one of the unquantified pessimisms in the previous analyses.

The key features have been:-

- Revisions to the forecast weight-losses for both EoG dates, including consideration of the results of sample testing from R1 in 2007, that have not shown any significant increase above expectations;
- The predicted peak tensile stresses remain less than the conservatively derived best estimate flexural strengths at each location;
- Axial fuel channel wall utilisations are broadly comparable with previous assessments and do not increase significantly with optimised core irradiation;
- Hoop fuel channel wall utilisations are reduced at the planned end of generation. If generation were to be optimised, they are predicted to remain bounded by previous assessments;
- Further structural integrity consideration of the low measured graphite density in R1 Channel 1923/15, bricks 6 and 8 continues to indicate that the Wylfa cores have good tolerance to anomalously high and untypical graphite weight-losses;

- A revised administrative control has been designed, to ensure that each trip transient is assessed to determine whether it is more severe than the pessimised trip defined in the safety case. Its technical basis is the mean of the changes in the flattened zone CGOTs immediately after a trip. In the event that the trip is more severe, the implications for the graphite safety case will be assessed prior to start-up.

The conclusion of the revised structural assessments is that the peak predicted stresses remain less then the conservatively derived best estimate strengths at each location and the risk of brick cracking is low. Hence the cores remain acceptable for continued operation.

Consequences

The consequences leg of the safety case for the risk of a single channel fire due to coolant flow by-pass caused by brick cracking is fundamentally unchanged from that in the previous review. It has concluded that the risk of a single channel fault arising due to brick failure is broadly acceptable.

This review reports improvements in the BCD alarms and associated procedures to increase the assessed reliability of rapid manual trip as protection for the escalation of the single channel fault to affect surrounding channels. These improvements have been justified in a separate submission, Reference 41. Based on these improvements and the continuing presence of the physical and chemical barriers to the escalation of a single channel fault it is now judged that the risk due to escalated faults is reinstated to the broadly acceptable range from the interim tolerable range.

Overall

Previous structural assessments have shown that utilisation factors are relatively constant with time, particularly for the key stress component of tensile hoop at the fuel channel wall. This position has been confirmed by re-analysis of the structural integrity. Only a modest increase in UFs is predicted even if the period of generation were to be optimised to September 2014. The likelihood of brick failure is therefore anticipated to change little in the remaining years of operation, irrespective of whether optimised generation is ultimately agreed at Wylfa. Brick stresses and utilisations are dominated by internal brick loadings and any further reduction in brick strength is expected to be largely offset by reducing Young's modulus. Significant tolerance to anomalies in weight-loss has been demonstrated on this basis.

The fact that utilisation factors are relatively stable with time adds to the value of recent and past inspections in terms of the justification of future operation. It is also worth noting that the utilisation factors for reactors at Wylfa are broadly comparable with those assessed for the reactors at Oldbury using essentially the same methodology. Both reactors at Oldbury have experienced 100% visual inspection of the fuel channels in the flattened region without detection of any significant cracks. These inspections provide some additional support for the judgement that crack initiation is unlikely in the reactors at Wylfa. It is also worth noting that the weight-losses at Wylfa are significantly lower than at Oldbury and, although weight-loss has only a marginal effect on the utilisations factors for the reasons discussed earlier, it demonstrates that the Wylfa reactors are operating well within the ageing of graphite properties experienced at Oldbury.

The key conclusions from each of the four legs of the safety case are:

1. The outage inspections confirm that the graphite core is behaving as expected and there are no indications that the structural integrity is threatened;
2. In-service monitoring of temperatures and control rod movements has the capability to detect serious graphite damage which could threaten shut-down or fuel cooling. In particular, core-wide monitoring of CGOTs in normal operation is capable of

detecting anomalous temperatures which might be indicative of serious cracking and coolant by-pass, and no such anomalies have been detected;

3. The refined structural assessment shows that the peak tensile stresses predicted remain significantly less than the conservatively derived best estimate flexural strengths estimated for each brick type and location;
4. Significant improvements have been made to the reliability of operator action to trip the reactor in response to the BCD alarms which indicate potential fuel failure. Consideration of the consequences of brick failure show that the risks of either a single channel fault or escalation to multiple channels are both now judged to be 'broadly acceptable'.

Taken together, the inspection, monitoring and structural assessment legs demonstrate that risk of crack initiation remains low for the remaining period of generation whether this is to December 2010 or September 2014. The risk of a double through-wall axial crack occurring to create the conditions for significant flow bypassing is significantly lower.

The probability of brick cracking remaining undetected and causing a single channel fire, or its escalation, is considered to be sufficiently low that the risk is 'broadly acceptable.

10 CHANGES TO PROCEDURES

There are no changes to procedures arising from this review of the graphite safety case. Improvements to the BCD alarms and to the assessment of reactor trips have been progressed via separate submissions, References 41and 33, respectively.

11 FURTHER WORK AND FUTURE INSPECTION PROGRAMME

11.1 Continuing Inspection and Structural Assessment Programme

Inspection, sampling and NOREBORE measurements will continue in future outages in accordance with the established principles described in Section 4.

Testing of the installed set samples removed from R2 in 2008 will be progressed and reported no later than in the 2009 review;

Graphite trepanning at the future outages will continue to sample fuel and interstitial channels from the same bricks from at least one standpipe location in an attempt to obtain more data relating to in-brick variability as encountered recently in Reactor 1.

11.2 Probabilistic Assessment

A probabilistic methodology is being established for the Oldbury cores to provide a systematic means of estimating the implications of variations in graphite material properties and loading conditions for the overall likelihood of fuel failure. Once this has been established it is intended to proceed with an equivalent approach for Wylfa. Experience with preparing the Oldbury probabilistic assessment indicates that it is time consuming and complex. It is reasonable to expect that once a method has been established some of these issues can be abated. Nevertheless, there are sufficient differences between the Oldbury and Wylfa cores for it to be judged inappropriate to give a firm commitment to a date for its completion at this time. As a minimum, a progress report and position statement will be presented in the 2009 review.

12 QUALITY ASSURANCE

This paper has been prepared in accordance with the requirements of Station MCP21 and the EWST Quality Management System for the production of safety cases and their supporting references.

13 INDEPENDENT NUCLEAR SAFETY ASSESSMENT

An Independent Nuclear Safety Assessment of this proposal is being undertaken in accordance with Company procedures.

14 CONCLUSIONS

1. The four legs of the graphite safety case have been re-assessed to take account of developments since the last review. The graphite cores of both reactors have been demonstrated to remain safe for continued operation up to at least December 2010.
2. Sensitivity studies of the effects of increased irradiation have demonstrated that the utilisation factors would only change marginally even if both reactors were to continue operating to September 2014.

15 RECOMMENDATIONS

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

1. **NOTE** that a further detailed review of the safety case for continued operation of the Wylfa reactors with respect to graphite core integrity has been completed;
2. **AGREE** that the review supports the previous safety case presented in NP/SC 4807 Addendum 3 for continued operation to the planned End of Generation in 2010;
3. **NOTE** that the structural integrity assessment has been extended to support a separate justification of generation optimisation at Wylfa;
4. **NOTE** the further work and future inspection plans identified in this paper;
5. **NOTE** that a further review of the safety case in the light of the continuing work programme will be reported to the Committee before the end of 2009.

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17. MEN/EWST/WYA/REP/0014/08, 'Wylfa Cumulative Irradiation Investigation', [redacted], July 2008.
18. MEN/EWST/WYA/REP/0019/08, Issue 2, 'Statistical Analysis of Wylfa Trepanned Sample Measurements for use in Graphite Weight Loss Prediction', [redacted], October 2008.
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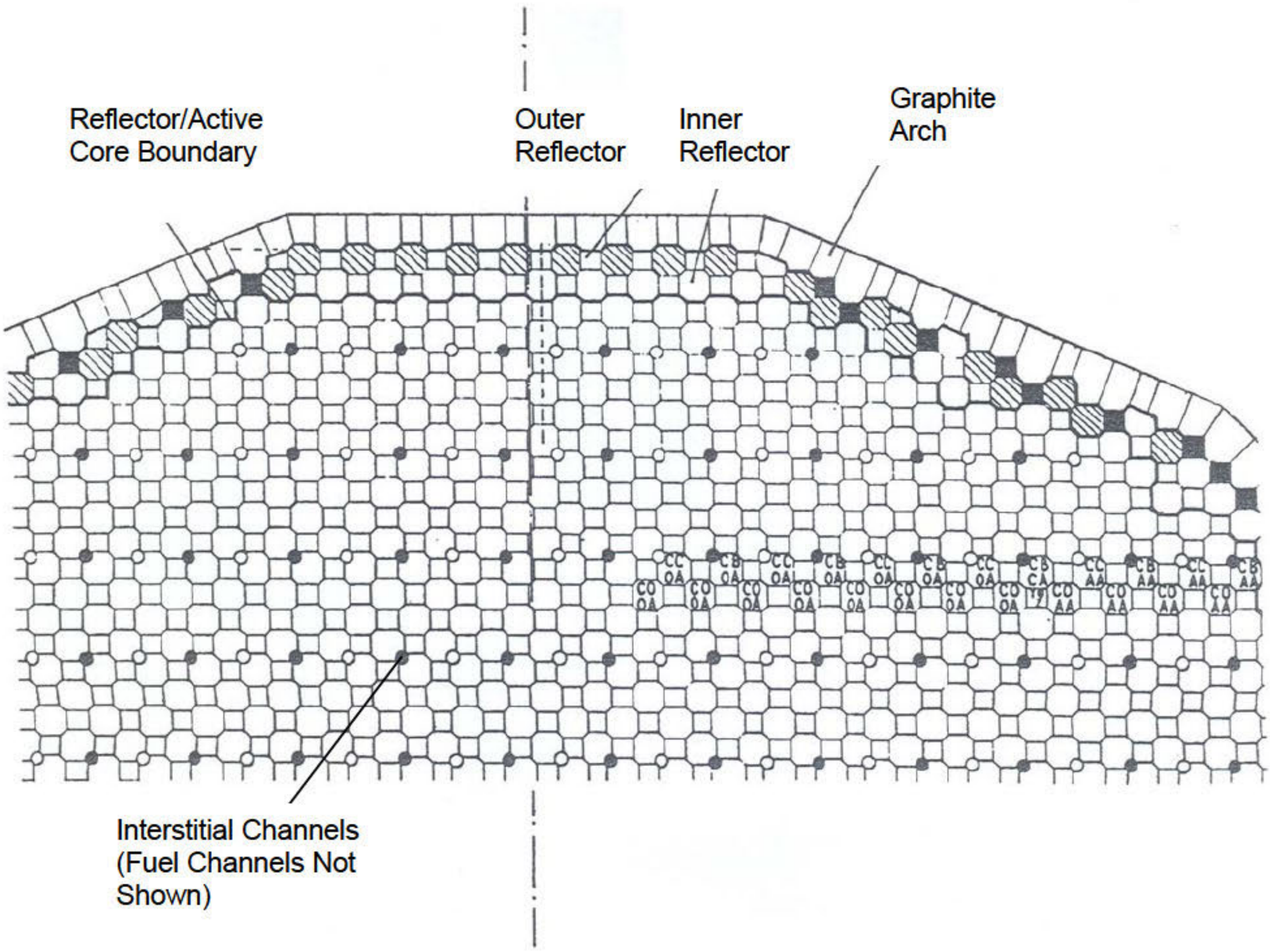


Figure 1 – Plan View of Wylfa Graphite Core Showing Graphite Arch and Reflector

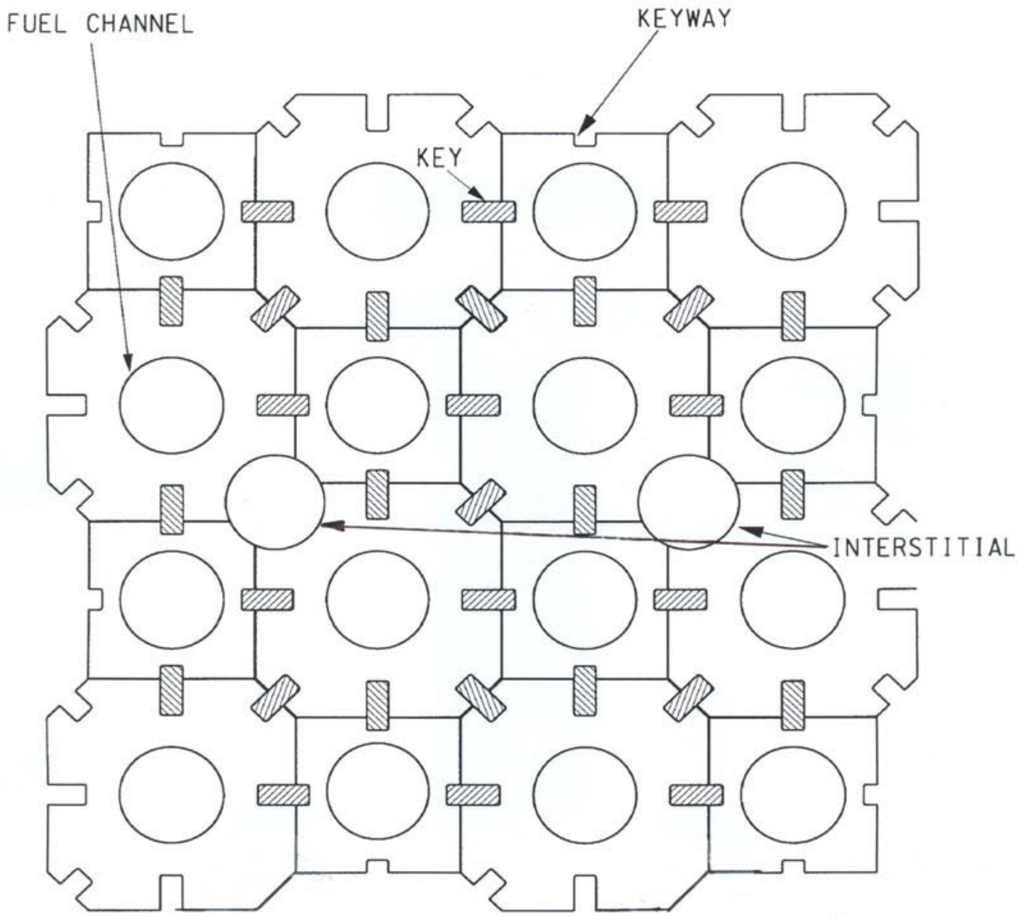


Figure 2 - Plan Arrangement of Wylfa Graphite Bricks

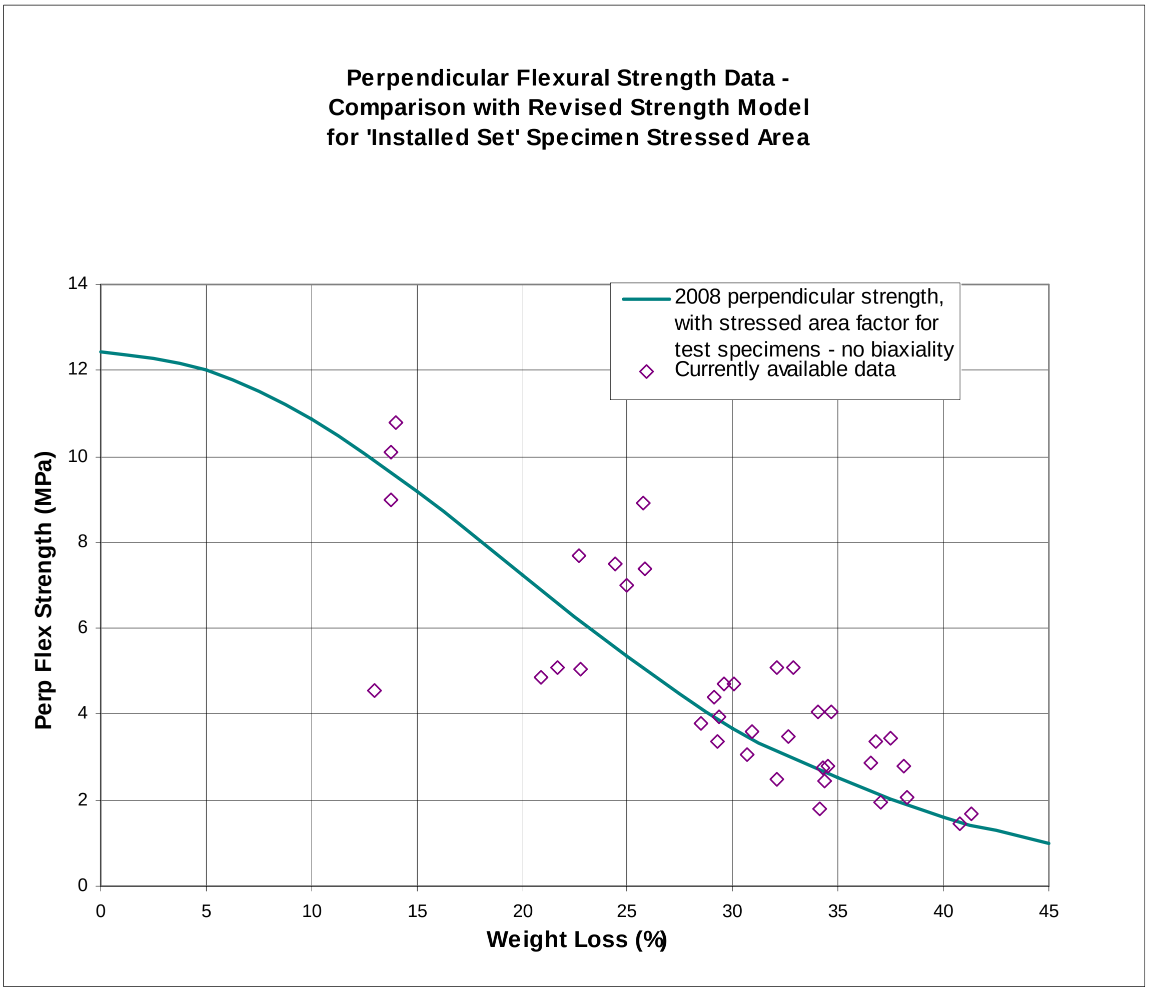
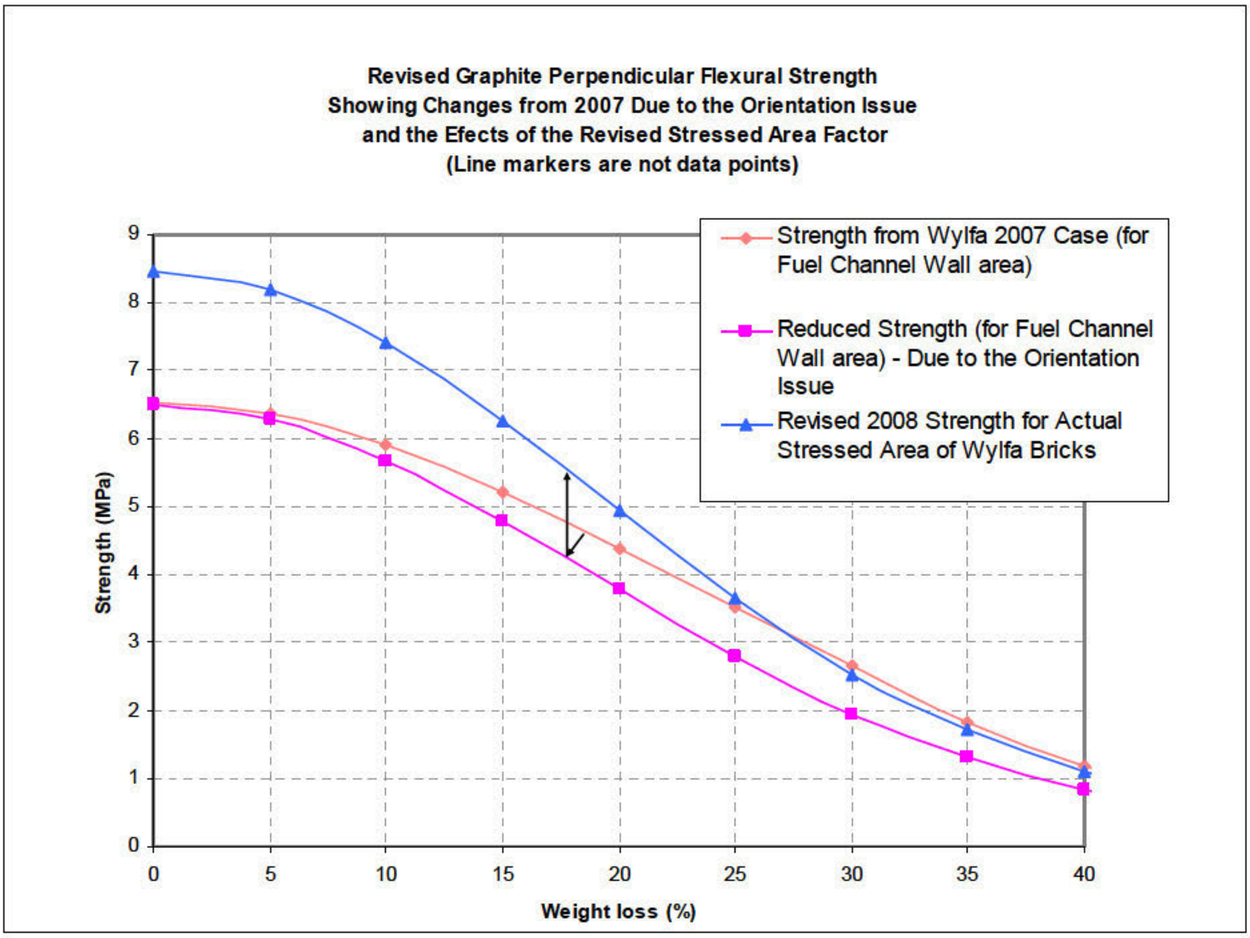


Figure 3: Comparison of the Emergent 'Installed-Set' Perpendicular Flexural Strength Data for the Test Specimens and the Strength Relationship Employed.



The revised strength relationship has a stressed area factor defined as $a^{-0.0555}$ where a is the stressed area in mm^2. The fuel channel wall area is $247000mm^2$; for octagonal bricks, the FEAT model calculates a as $2000mm^2$. A biaxiality factor of 0.8 is applied throughout.

Figure 4 – Comparison of Perpendicuar Flexural Strength Relationships

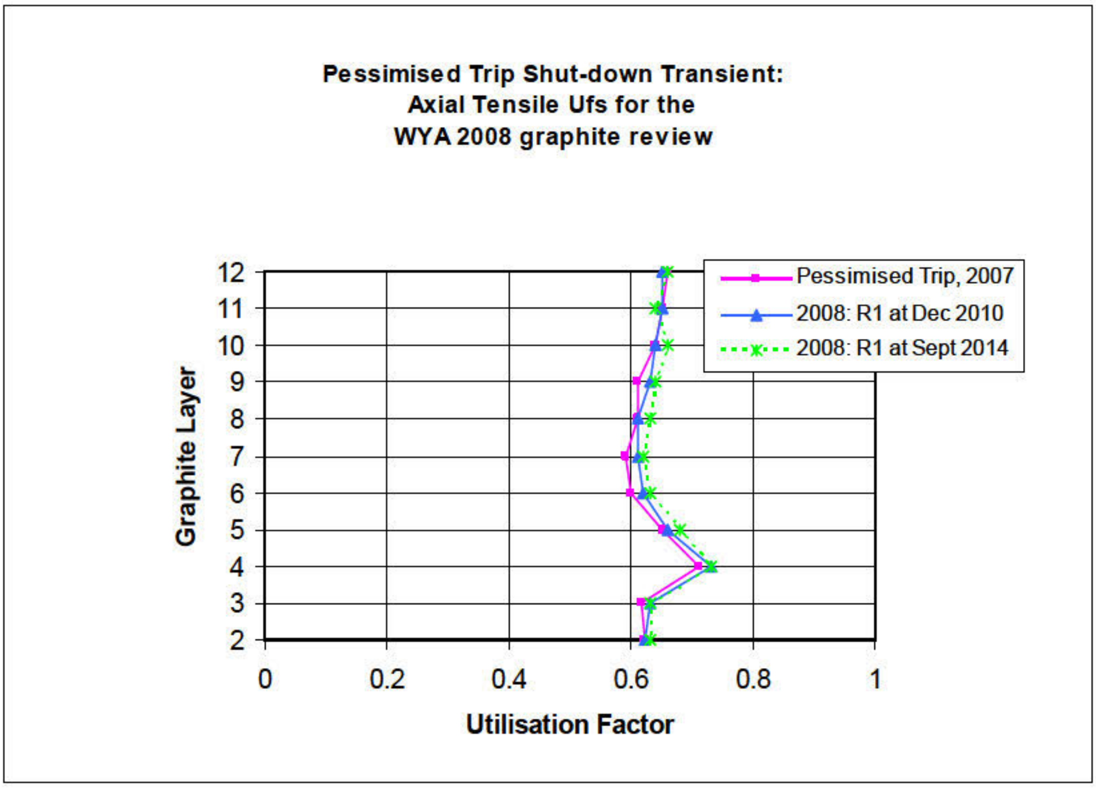


Figure 5 (a) Shut-down Transient FCW Axial Tensile Utilisations – Note that the biaxiality factor is removed from the strength for Graphite layers 2 and 3 due the absence of sigificant hoop stress

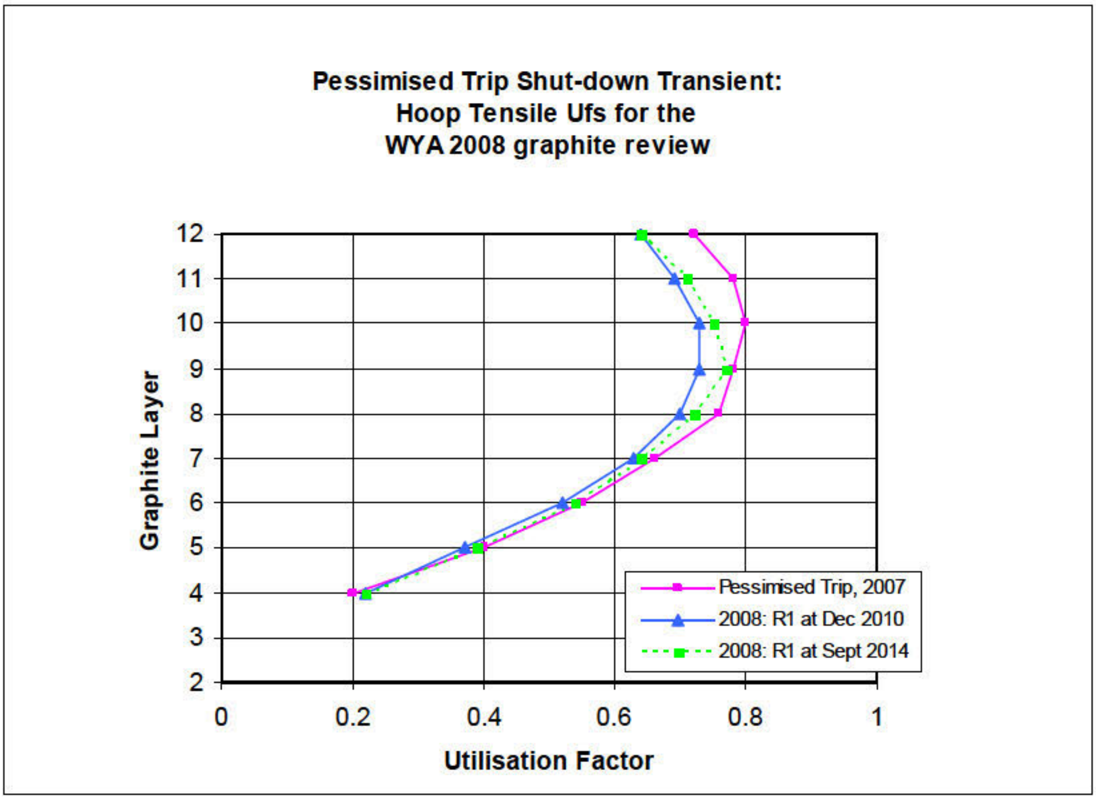
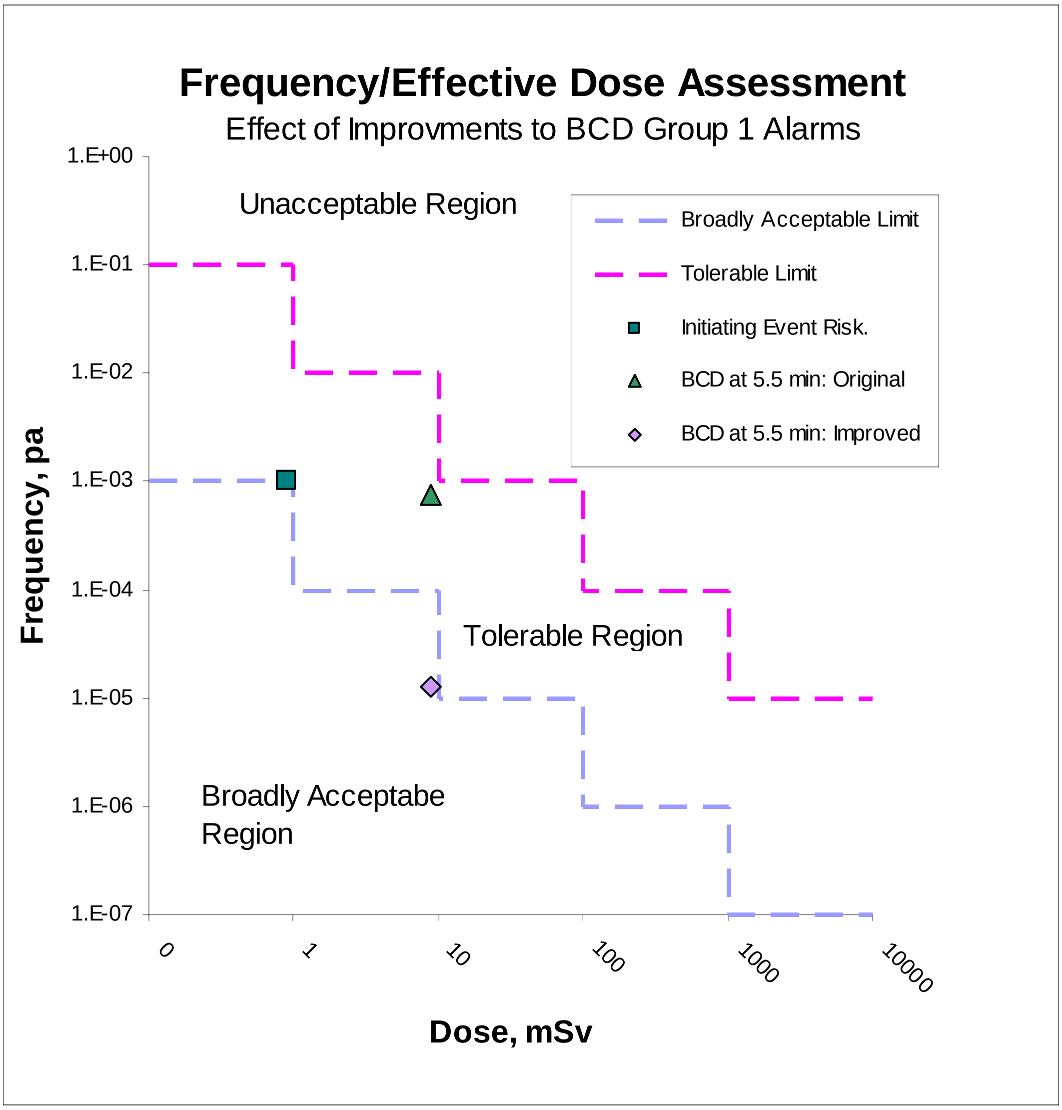


Figure 5 (b) Shut-down Transient FCW Hoop Tensile Utilisations



Notes

1 The initiating event frequency is assumed from the judgement in Section 7.2.2.

2 The ‘broadly acceptable’ and ‘tolerable’ limits are cited from the probabilistic principle, NSP3, Reference 38, for a single class of faults.

3 The improved BCD response at 5½ minutes retains the ‘unlearning’ factor of 1.5, i.e. - the operator performance is expected to improve by this factor with time.

4 The dose markers for the escalated fault assume pessimistically that all 8 surrounding channels are affected.

Figure 6: Risk Assessment for the Escalation of a Fault due to Graphite Brick Failure

Table 1: Summary Comparison of the Results of BEST v2.1 Peak Layer Weight-Loss Predictions (located at layer 6) for EoG in December 2010, with previous predictions.						
Weight-Loss %	BEST Predictions based on data up to and including R2 2006, the basis of the 2007 graphite review.			Current revised BEST predictions, including data from R1 in 2007 (Reference 21).		
	Fuel Channel Slice 1	Interstitial Channel Slice 3	EoG Dose GWd/t	Fuel Channel Slice 1	Interstitial Channel Slice 3	EoG Dose GWd/t at EoG
R1	13.8	18.8	31.0	13.9	18.4	30.7
R2	14.2	16.9	29.5	13.9	16.5	29.0
2007 BEST prediction at bounding MCI				The 2008 assessment is reactor specific		
R2	15.3	18.3	31.5			

Table 2: Summary Comparison of the Results of BEST v2.1 Peak Layer Weight Loss Predictions (located at layer 6) for EoG in September 2014, with previous predictions.						
Weight-Loss %	BEST Predictions based on data up to and including R2 2006, the basis of the 2007 graphite review.			Current revised BEST predictions, including data from R1 in 2007 (Reference 21).		
	Fuel Channel Slice 1	Interstitial Channel Slice 3	EoG Dose GWd/t	Fuel Channel Slice 1	Interstitial Channel Slice 3	EoG Dose GWd/t at EoG
R1	13.8	18.8	31.0	15.6	21.1	34.1
R2	14.2	16.9	29.5	15.8	19.0	32.4

Table 3	**Comparison of FCW Shut-down Transient UF assessments for high weight-loss bricks observed in R1 at Channel 1923/15**			
2007 assessment based on scaling 2007 strength and DYM				
	Brick 6		**Brick 8**	
	Normal	**Anomalous**	**Normal**	**Anomalous**
FCW SDT Hoop Tensile UF at 31.5 MWD/t	0.70	0.61	0.76	0.71
Wt loss at 31.5 MWD/t, %	14.1	20.3	13.2	22.5
Revised assessment using 2008 strength relationship and FEAT analysis				
FCW SDT Hoop Tensile UF at 31.5 MWD/t	0.52	0.43	0.70	0.72
Wt loss at 31.5 MWD/t, %	14.3	19.8	13.6	22.7
FCW SDT Hoop Tensile UF at 34.1 MWD/t	0.54	0.44	0.72	0.77
Wt loss at 34.1 MWD/t, %	15.6	21.6	14.8	24.7
Note:	Items in grey represent optimised generation to September 2014			

Note: The 'normal' UFs and weight-losses for 2008 are obtained from Reference 29.

APPENDIX A. VERIFICATION PLAN

VERIFICATION PLAN	No: RG7225_VP_001 Issue: Issue 1		PF009
Authors: [redacted]	Approved: [redacted]	Date: 24/9/08	
Document Ref: NP/SC 4807 Addendum 4			Issue: 1
Document Title: Wylfa PS: Annual Review of the Safety Case for the Integrity of the Graphite Core: 2008			
Date Verification required by: 24th October 2008			

VERIFICATION RISK ASSESSMENT

Risk No	Verification Component /Description of risk (e.g. input data, calc 1, section 1, etc.)	Error (High/Low)		Specific Mitigation of Risk (mandatory for high prob/high consequence)
		Probability	Consequence	
1	The graphite strength relationship is inappropriate or incorrect	M	M	QA Grade 2 applied to underpinning supporting references and endorsement by the Graphite Technical Issues Group.
2	The completeness of issues including specifically all items committed to in previous safety cases and statements to the NSC	L	M	Explicit check of previous Matters Arising statements and safety case commitments.
3	The citation of supporting references is inappropriate or introduces bias in the interpretation	M	M	Key specialists to act as supporting verifiers and/or key reference authors to be consulted by the Lead Verifier.
4	Plant configuration, operating conditions and other parameters may be inaccurately described	M	M	NRE to be asked to focus on these aspects during verification
5	Certain data central to the safety arguments is not explicitly stated	M	M	All verifiers to check that any data central to safety arguments is included.

NOTE FOR LEAD VERIFIERS

In cases where a Lead Verifier is appointed, verification comments should be made available to all Verifiers.
It is the Lead Verifier's responsibility to ensure that all verification comments are adequately closed out.

SELF VERIFICATION CHECKS

Originators	Section or Scope	Self Verification Checks
[redacted]	All	Checks against Strategy document.
		Checks against each supporting reference.
		Check, correct typographic errors – via word processor functionality and subsequent read-through.
		Checks of completeness of addressing verifiers or other consultants' comments via issue control of drafts, use of track-changes functionality and maintenance of paper records of revisions for independent check by the verifier.

INDEPENDENT VERIFICATION

Originator(s)	Verifier(s)
[redacted]	1. [redacted] 2. 3. 4. 5. 6. 7.

Initial to confirm pre-verification discussion has taken place		Document Section	Scope, input document, acceptance criteria	Risk No(s)	Verification Statement Required
Originator	Verifier				
1	1, 2	All	Strategy Document for 2008 review.		The safety arguments are logical, clear and soundly based. The paper is complete and consistent with Graphite Programme decisions.
	1	All	The Cat 1 paper and all references		Information in references has been accurately cited.
	4	All	Safety issues and commitments in previous submissions or NSC statements		All nuclear safety issues and commitments have been adequately addressed and judgements are supportable.
	2, 3	Structural integrity assessment	Appropriate use of methodologies and input data especially graphite strength.		Methodologies and input data have been checked and are appropriate.

	5	Inspection and monitoring sections	Citation and interpretation of supporting references		The data and interpretations in the inspection and monitoring sections are accurate.
	5	All: Plant data and operating conditions	Data relating to plant and its operation		The plant data and operating conditions are correctly described.
	6	Reactor analysis	Reactor analysis especially shutdown transient issues		Reactor analysis references are correctly interpreted
	7	Consequences	Description of consequences of clad melt.		The description of consequences is adequate and rigorous.
	7	NSPs	Section addressing NSPs and ALARP		The assessment against the NSPs is supportable.
	1	All	Comments by all supporting verifiers.		Comments by all supporting verifiers have been adequately shared and addressed.

STATEMENT BY APPROVER

Tick	Statement by Approver
☐	I confirm that all verification comments have been incorporated or answered to the satisfaction of the verifier as recorded on the attached number of sheets or documents.
☐	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, as recorded on the attached number of sheets or documents.
☐	I confirm that all verification statements are complete and satisfactory.
Completion Approval:	Date:

REVISION RECORD

N^{o}	Date	Author	Reason
1	6/11/2008	[redacted]	Minor typographic amendments, following original issue

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