NUCLEAR SAFETY COMMITTEE

NP/SC 4927 Rev 1 Addendum 6
Oldbury Power Station

TITLE - **A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation: Proposal to Operate Reactor 1 to a Mean Core Irradiation of 32.8GWd/t**

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

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TITLE: **Oldbury Power Station: A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation: Proposal to Operate Reactor 1 to a Mean Core Irradiation of 32.8GWd/t**

TARGET NSC SUBMISSION DATE: June 2011

AUTHOR

Name [redacted]

Signature Date 16th May 2011

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 EF Task file RG7369. A verification statement is given overleaf.

Name [redacted]

Signature Date 16 May 2011

SAFETY CASE OFFICER

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

Name [redacted]

Signature Date 17/5/11

ENDORSED FOR ISSUE BY

Name [redacted]

Signature .... Date 17/5/11

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 EF Task File No RG7369

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

- The presentation of the arguments is clear and the layout, grammar and spelling are satisfactory.
- The information from the references is correctly quoted and 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] ate: 16 May 2011

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

[redacted] Date: 17/5/11
Oldbury Power Station

SUMMARY

The safety case for the integrity of the Oldbury graphite cores was developed in NP/SC 4927 Rev 1 and its Addenda 1 – 5 to support operation of both reactors up to June 2011.

NP/SC 5059, presented to the August 2010 NSC, outlined a strategy for extending generation on Reactor 1 beyond June 2011, based on the transfer of partially irradiated fuel from Reactor 2 following its final shutdown, by 30th June 2011.

This strategy required re-validation of the Periodic Safety Review (PSR) to December 2012 and revisions to the fuelling and criticality safety cases, the Reactor 1 graphite core integrity safety case and the Post-operational and De-fuelling safety case (PODSC). The current submission presents the safety case for operating the Reactor 1 graphite core to an MCI of 32.8GWd/t which is not anticipated to be reached before December 2012.

The graphite core safety case for extended generation on Reactor 1 is based on demonstrating that:-

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

Demonstration of these two key safety functions is achieved, as for previous Addenda, via the four safety case elements of inspection & sampling, monitoring, structural assessment and consequences analysis. It is recognised, however, that extended generation will lead to further graphite weight losses and hence a real increase in the likelihood of brick cracking, coolant gas diversion from the fuel channel and fuel clad melt. These additional risks are therefore to be mitigated during the period of extended generation by progressively reducing the severity of the "trigger" trips at which action (graphite core inspections or safety case review) is required prior to returning Reactor 1 to service. In order to minimise the likelihood of any reactor trip leading to a more severe transient than the revised upper trigger trip level, operational targets will be introduced for the period of extended generation.

Consideration has also been given to the risks associated with on-load re-fuelling activities and it has been concluded that the likelihood of crack initiation during the period of extended generation will not be significantly higher than that for operation to 31.5GWd/t. Nevertheless, a number of additional practicable measures are proposed to enhance safety and further reduce the risks throughout this period.

It is concluded that a satisfactory safety case for the Reactor 1 graphite core has been established for continued operation up to an MCI of 32.8GWd/t, this being unlikely to be exceeded before December 2012.

CONTENTS

1 INTRODUCTION

The safety case for the Oldbury graphite cores is set out in NP/SC 4927 Revision 1 and its Addenda 1-5 (References1-6). Essentially, the case is based on demonstrating that the graphite bricks that make up the core retain sufficient strength to support the loads imposed during the various reactor operational conditions, in particular the transient loads that are experienced during shut-down. The safety arguments are focussed on the effects of progressive reduction of core graphite density, due to radiolytic corrosion, with increasing core irradiation.

The safety case for both reactors was reviewed in NP/SC 4927 Rev 1 Addendum 4 (Reference 5) which was presented at the December 2008 Nuclear Safety Committee (NSC). Deterministic and probabilistic assessments were reported, assuming end of generation (EoG) mean core irradiations (MCIs[1]) of 31.5GWd/t and 32.7GWd/t for Reactor 1 and Reactor 2 respectively. Based on the results of these assessments, together with the satisfactory results obtained from reactor inspection and monitoring, it was concluded that the risks associated with continued operation up to these limits were at least Tolerable and ALARP but also judged to be Broadly Acceptable. A comprehensive probabilistic assessment was carried out specifically for Reactor 1, although sufficient assessment of Reactor 2 was undertaken to indicate that the assessment of Reactor 1 comfortably bounded that of Reactor 2 (Reference 5). As explained in previous Addenda, the risks arise largely from crack initiation at the fuel channel wall (FCW) and the predicted EoG weight losses at the FCW were greatest for Reactor 1, despite the lower EoG MCI for that reactor.

In order to secure operation of both reactors to June 2011, the periodic safety review (PSR) date, the safety case for Reactor 2 was subsequently updated in NP/SC 4927 Rev 1 Addendum 5 (Reference 6), which was presented to the November 2009 NSC, to allow operation up to an EoG MCI of 34.0GWd/t. The revised safety case for Reactor 2 included a full probabilistic assessment of the risks associated with crack initiation at the FCW. It was concluded that the increase in nuclear safety risk associated with operation of Reactor 2 to the revised limit of 34.0GWd/t was small and that the risks were bounded by those associated with operating Reactor 1 to its limit of 31.5GWd/t; the risks for Reactor 2 were therefore at least Tolerable and ALARP and were judged to be Broadly Acceptable. Generation from Reactor 2 will cease by 30th June 2011 at which time the MCI is not expected to exceed 33.7GWd/t.

A strategy for generating from Reactor 1 beyond 30 June 2011, based on the transfer of partially irradiated fuel from Reactor 2 following its final shut-down, was outlined in NP/SC 5059 (Reference 7) which was presented to the August 2010 NSC. The strategy required a re-validation of the PSR to December 2012 as well as revisions to the fuelling and criticality safety cases, the graphite core integrity safety case and the Post-operational and De-fuelling safety case (PODSC). It was noted that the revision to the graphite safety case for Reactor 1 proposing an EoG MCI no greater than 32.8GWd/t would be presented to the NSC early in 2011. It was anticipated that this MCI limit would not be reached before December 2012 although cessation of generation on Reactor 1 was planned for July 2012

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[1] MCIs referred to in this Addendum are all *cumulative* MCIs. MCI *limits* refer to safety case bounds rather than operational or safety limits.

in order to meet de-fuelling commitments. The purpose of this Addendum 6 is to present the revised safety case for the integrity of the Reactor 1 graphite core to an EoG MCI of 32.8GWd/t. The associated effects of the inter-reactor fuel transfer strategy on graphite core state and re-fuelling operations are included.

2 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. The side and top reflectors are made from Pile Grade B (PGB) graphite, while the active core and bottom reflector are made from Pile Grade A (PGA) graphite. The core is made up of 12 layers of bricks, numbered from the bottom and arranged in columns. The upper end of each brick is spigotted to locate in a counter-bore machined in the brick above. Alternate columns are composed of octagonal and square bricks respectively. Figure 1 illustrates the layout.

In the active core, the bricks are sized to provide Wigner gaps between the bricks to allow for irradiation induced dimensional changes. The bricks are keyed together, by loose keys set in vertical keyways, in such a way as to maintain their relative positions under the effects of core expansion (Figure 1). The keys locating adjacent bricks are continuous over the full core height.

The brick columns comprising the active core are centrally bored to form 3308 vertical fuel channels per reactor. There are 8 fuel elements stacked in each fuel channel; they extend from approximately mid-way up Layer 1 to Layer 11 of the active core. The safety case concentrates on core Layers 2 - 11.

315 interstitial channels are each centred at the junction between four adjacent columns (Figure 1); these contain control rods or are allocated to other purposes (absorbers, flux scanning, graphite sampling etc.). The brick pattern may therefore be regarded as blocks of 4 fuel channels adjacent to an interstitial channel, alternating with blocks of 4 fuel channels remote from an interstitial. This layout is of significance in assessing the coolant leakage from a fuel channel through a cracked brick.

The radial flux profile comprises a flattened zone (FZ), with the power in the outer channels (the un-flattened zone) falling off radially towards the edge of the active core. The axial flux profile peaks around the core mid height and falls off axially towards the top and bottom of the core. Graphite weight loss in service is related to the cumulative irradiation experienced by each brick.

3 BASIS OF CURRENT SAFETY CASE

The safety case for the integrity of the Reactor 1 graphite core up to an MCI of 31.5GWd/t (Reference 5) is based on a demonstration that:-

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

The evidence underpinning the absence of widespread core damage is based on satisfactory results from control rod drop and freedom of movement tests, gas and graphite temperature monitoring, monitoring of re-fuelling difficulties, NOREBORE inspections, visual inspection of channel bores and graphite trepanning and density measurements. Avoidance of widespread damage is also assured by demonstrating that an adequate margin exists between the predicted maximum weight loss at 31.5GWd/t and the percolation limit weight loss, i.e. the weight loss at which the strength of the graphite is predicted to fall to zero.

The evidence underpinning the low likelihood of brick splitting is based on the satisfactory results obtained from TV inspections of the flattened region together with the outcomes of both probabilistic and deterministic assessments of the structural integrity of the graphite cores and the associated risks of clad melt and radiological release. Risks are mitigated by specifying two trip transients, which if exceeded, trigger either 10% inspection of the flattened region fuel channels or a review of the safety case prior to the return to service of the reactor.

4 RECENT DEVELOPMENTS

The most recent review of the Reactor 1 graphite core integrity case was presented to the NSC in December 2008 (Reference 5). Since that time a programme of work aimed at improving understanding of graphite behaviour and refining associated assessment methodologies has continued in support of the safety cases for the Oldbury and Wylfa reactor graphite cores.

This programme has identified potential refinements associated with the prediction of the temperature derivative of dimensional change which influences graphite brick hoop stresses, as well as a potentially increased temperature dependence of the Coefficient of Thermal Expansion (CTE) which also influences graphite brick thermal stresses. Predicted Utilisation Factors (UFs, stress/strength) and crack initiation probabilities are therefore potentially affected. These two issues were considered recently in relation to the safety case for the Wylfa reactor graphite cores (Reference 8); scoping calculations were reported which indicated that the combined effect would be neutral or beneficial at the dominant brick layers and it was judged that there were no significant safety case implications. The programme has also identified refinements to the by-pass flow modelling approach employed to predict channel flow impairments and conditional clad melt risk in the event of brick splitting. These refinements have been employed in the assessments supporting the recent graphite safety case reviews for Wylfa (References 8 and 9). They have resulted in lower predicted by-pass flows and higher channel outlet flows, significantly reducing the calculated risk of clad melt.

The programme of materials testing and weight loss measurements on trepanned samples removed during outages continues to provide additional data, influencing the predicted structural properties, and hence UFs and crack initiation probabilities, for the graphite bricks. Since the Oldbury 2008 review (Reference 5), weight loss predictions have been revised (see Section 7.1) to take account of the results from measurements on samples trepanned from Reactor 1 during 2009. Flexural strength predictions have also been revised (Section 9.1) and revisions have been made to the estimated percolation limit weight loss following compressive strength testing of samples trepanned from Reactor 2 during 2010 (Section 7.2).

The probabilistic and deterministic assessments for the Oldbury graphite cores have now been updated to reflect these refined assessment methodologies and data (see Sections 8 and 9 below). The overall effect of these recent developments has been a reduction in the predicted likelihood of clad melt due to graphite brick splitting and hence an enhancement of the extant safety case for operation to the current EoG MCI of 31.5GWd/t.

5 BASIS OF SAFETY CASE FOR EXTENDED GENERATION

The safety case for extended generation will continue to provide a demonstration that the two key safety functions identified in Section 3 above are met based on the four safety case elements (inspection & sampling, monitoring, structural assessment and consequences).

Current inspection, sampling and monitoring activities are described in Section 6 below and Section 12 presents the proposals for their continuation.

Revised assessments of the likelihood and consequences of graphite brick cracking, to take account of the developments identified in Section 4 above, as well as the proposed increase in MCI, are presented in Sections 8 and 9.

It is recognised, however, that extended generation will lead to further graphite weight losses, and hence a real increase in the likelihood of brick cracking, coolant gas diversion and clad melt. To underpin the safety case for extended generation, and reduce reliance on results from the revised assessments, a conservative reduction in the severity of the 'trigger trips' for the period of extended generation is therefore proposed (Section 11 below) to mitigate this increase in risk.

Consideration is also given to the reasonable practicability of implementing additional measures to reduce risks and enhance safety (Appendix B).

6 INSPECTION, SAMPLING AND MONITORING ACTIVITIES

TV inspections of fuel and interstitial channels, carried out during reactor outages, provide direct evidence of the condition of the graphite cores following a period of operation. Addendum 4 (Reference 5) reported the completion of the inspection of all flattened region fuel channels in Reactor 1, with no significant findings that could pose a threat to either fuel cooling or the shut-down and hold-down capabilities of the reactor. Inspections during statutory outages have continued; channels are targeted for inspection, for example, as a result of anomalies detected during routine channel gas outlet temperature (CGOT) monitoring (see also Section 10.5 below), previous inspection findings or from any observed re-fuelling difficulties (Section 10.4).

Since Addendum 4, further TV inspections of Reactor 1 have been carried out during its 2009 statutory outage (Reference 10). 66 fuel channels and 3 control interstitial channels were inspected; previously observed defects were unchanged except for one small indication, judged to be a feature of heavy deposit, rather than a crack, and considered to be of no structural significance. An 8mm long crack-like indication was observed in brick 12 of fuel channel L17D4 but, noting that the indication was in a region of very low predicted weight loss, the Outage Inspection Panel considered it unlikely to be irradiation-induced and of no nuclear safety significance (Reference 10). No new defects were identified in the control interstitial channels.

In addition to inspections undertaken during statutory outages, Addendum 4 introduced requirements for TV inspections and channel temperature probing following a reactor trip, depending on the trip severity. There have been no Reactor 1 trips that have necessitated the implementation of these requirements. It may be noted, however, that implementation of these requirements has been necessary twice on Reactor 2, following trips occurring on 11th November 2010 (Reference 11) and 17th March 2011 (Reference 12); the inspections identified no graphite core related safety issues preventing return to service of the reactor on either occasion.

In order to confirm that core geometry is being adequately maintained, measurements of the diameters and inclinations of fuel channels are carried out using NOREBORE; channels are inspected, during statutory outages, from a range of locations across the core. The NOREBORE measurements also provide dimensional change data which are used in the structural integrity assessments reported in Sections 8 and 9 below. In Addendum 1 (Reference 2) it is reported that NOREBORE inspections on both reactors indicate the absence of unexpected channel distortion or fuel channel shrinkage. In Addendum 4 (Reference 5), it was reported that NOREBORE inspections of two further Reactor 1 fuel channels had been carried out since those reported previously in Addendum 1 (Reference 2); no abnormalities were revealed by the NOREBORE inspections (Reference 5). Results from further NOREBORE inspections, carried out during the Reactor 1 2009 statutory outage, are reported in Reference 13; ten channels were inspected and the observed changes in channel geometry were not considered to be significant in terms of fuel and control rod movements.

Graphite samples have been routinely trepanned from the reactor cores during statutory outages to provide information on the progression of weight loss and to facilitate forward predictions (see Section 7 below). During the 2009 Reactor 1 outage, 85 samples were trepanned from eight flattened region fuel channels and one interstitial channel (Reference 14). These were subsequently used to produce 226 density measurements for Layers 4 – 8, significantly increasing the amount of Oldbury Reactor 1 trepanning data at high doses (Reference 15). Peak weight losses of 39.4% and 37.5% were measured for the fuel channel and interstitial channel respectively (Reference 14); these values are considered to be generally in line with previous predictions.

Control rod drop testing following a trip, together with a rolling programme of control rod freedom of movement testing while the reactor is running, has continued to demonstrate that, for both reactors, there is no impairment of rod movement, thereby providing evidence that there is no damage within the graphite core. At the Reactor 1 trip for the 2009 statutory outage, no anomalies were identified in the control rod drops and no anomalies were identified in the control rod drop tests carried out on Bulk Groups 1 and 4, the Safety Rod group and the Sector (Regulating) rods immediately after the trip (Reference 10).

As part of the routine monitoring, a computer programme, CRACKFINDER, is run on a daily basis. CGOT data from instrumented fuel channels are compared with the equivalent predicted temperatures from the PANTHER thermal hydraulic modelling code; the results are analysed weekly and any channel exhibiting a significant discrepancy is subsequently investigated (for example by targeting that channel for future inspection).

A Fuel Integrity Monitoring System (FIMS) has been installed on each reactor to provide additional protection against the risk of fire spread in the event of fuel melt. FIMS provides a reliable alarm indication within about one minute of the onset of uranium melt anywhere

in the reactor core; this enables the reactor to be tripped, thereby limiting the consequences of the initial fault. In Addendum 5 (Reference 6) it was reported that FIMS was undergoing commissioning soak tests (fully operational with limited action in response to alarm enunciation). It can now be confirmed that FIMS has since entered commissioned operational service on both reactors, where it was initially used in conjunction with the Burst Cartridge Detection (BCD) system; the enunciation of two out of three FIMS alarm indications required the operator to refer to the BCD system to aid the decision on whether to trip the reactor. Following a human factors review of FIMS (Reference 16), FIMS is now operated as a standalone system under a Temporary Operating Instruction (TOI) which requires the reactor to be tripped immediately in the event of two out of three alarms enunciating.

7 PREDICTED GRAPHITE WEIGHT LOSSES AT END OF GENERATION

Extending generation of Reactor 1, as proposed in this current Addendum, will result in continued graphite weight loss throughout the core. FCW weight loss predictions and the peak whole core weight loss predictions have therefore been made at the revised EoG MCI of 32.8GWd/t.

7.1 Fuel Channel Wall Predictions

Fuel channel wall weight loss predictions are used as inputs in the deterministic and probabilistic assessments to demonstrate that the risks associated with crack initiation, leading to brick splitting, channel flow by-pass and consequential fuel overheating, are acceptable (see Section 3 above).

Table 1 presents the best estimate and upper bound Reactor 1 weight loss predictions, from layer-wise statistical fitting of the FCW weight loss data, including those obtained during the 2009 Reactor 1 trepanning campaign, for the current and proposed MCI limits of 31.5GWd/t and 32.8GWd/t respectively (Reference 15). Table 1 also shows weight loss predictions for an MCI of 34.0GWd/t, well beyond the proposed EoG MCI of 32.8GWd/t, and for comparison, previous predictions from Addendum 4 (Reference 5) at the current EoG MCI of 31.5GWd/t.

From Table 1, it may be seen that:-

- Inclusion of the 2009 data in the analysis has not significantly changed the weight loss predictions. At 31.5GWd/t, a peak best-estimate FCW weight loss of 36.3% (Layer 5, Slice 3) is predicted, compared with a previously predicted peak of 36.6% (also Layer 5, Slice 3).
- The peak best estimate FCW weight loss predicted at 32.8GWd/t also occurs at Layer 5 Slice 3 and, at 38.7%, is 2.4% higher than for the current EoG MCI limit.
- The peak best-estimate FCW weight loss predicted at 34.0GWd/t is 40.9% and also occurs at Layer 5 Slice 3. Thus it can be seen that predicted peak FCW weight loss varies approximately linearly with MCI over the range considered.

These FCW weight loss predictions have been used, as appropriate, in the revised deterministic and probabilistic assessments summarised in Sections 8 and 9 below.

7.2 Peak Whole Core Predictions

Peak weight losses are predicted to occur at the interstitial channel wall (ICW). These peak weight loss values may be compared with the predicted percolation limit weight loss, supplementing the evidence from inspection and monitoring activities and providing additional assurance that the likelihood of widespread core damage, sufficient to challenge core shut-down and hold-down capability (one of the key safety functions identified in Section 3 above) is extremely low.

Estimates for the percolation limit weight losses were initially reported in Addendum 5 (Reference 6) based on tensile, flexural and compressive strength data, obtained from both Oldbury and Wylfa reactors (a significant proportion of the higher weight loss data was provided from compressive strength data from Oldbury installed sets). In a recent Category 2 submission in support of the continued operation of Reactor 2 (Reference 17), revised estimates of the mean and lower bound percolation limit weight losses were presented. These revised estimates were based on compressive strength measurements from samples trepanned from ICW locations in the Oldbury cores and were intended to address uncertainties with the claimed percolation limit weight loss by obtaining data from the reactor graphite. Table 2 shows the best estimate and upper bound (+2σ) peak predicted weight losses (anywhere in the core) for Reactor 1 at its proposed EoG MCI limit of 32.8GWd/t (Reference 18); the revised percolation limit weight losses (best estimate and lower bound (-2σ) values) are also shown.

From Table 2, it may be seen that a minimum margin (Lower bound percolation – upper bound peak weight loss) of 0.7% is predicted at the proposed EoG MCI for Reactor 1; on a best estimate basis, the margin is 10.7%. These margins may be compared with the corresponding values of 3.1% and 13.1% for Reactor 1 at its current EoG limit of 31.5GWd/t. The margins for Reactor 1 at the proposed EoG are lower than the corresponding EoG margins reported for Reactor 2 (2.2% minimum and 12.9% best estimate) in Reference 17. However, up to an MCI of about 32GWd/t (approximately 6 months into extended generation) the minimum margin for Reactor 1 is bounded by that for Reactor 2. [The implications of results from currently ongoing work in support of the graphite safety case (see Sections 10.4 and 12 below) are to be addressed before this MCI of 32.0GWd/t is exceeded.]

7.2.1 Implications for Widespread Core Damage

Demonstration of a low likelihood of widespread core damage is provided, primarily, by the satisfactory results obtained from monitoring and inspection activities (see Section 6 above). Further confidence is provided by demonstrating that a margin to the percolation limit weight loss exists. In considering the adequacy of this margin for Reactor 1 at its proposed EoG, the following observations may be made:-

- The minimum margin of 0.7% at 32.8GWd/t is the difference between the lower bound percolation limit weight loss and upper bound peak predicted weight loss. The margin is therefore very unlikely to be lower than this at the peak weight loss location and even more unlikely at all other locations throughout the core. Since this is the smallest margin predicted in any part of any brick, the onset of local damage is not anticipated.
- Given the variability of weight loss predicted throughout the core (and throughout individual bricks), even if the margin were to be reduced to zero at a localised peak weight loss location, it is judged unlikely that the local loss of strength would have

a significant effect on the overall brick integrity or geometry; at worst, it is judged that only minor spalling would occur.

- Any significant impairment of the shut-down function would require widespread core damage as a result of progressive deterioration with increasing weight loss. However, it is evident from Table 1 that the predicted weight losses throughout the core increase only slowly to EoG. Thus, the possibility of significant progressive damage during the period of extended generation can be discounted. Even if such damage were to occur, it is anticipated that it would be detected by inspection or monitoring activities before it presented any challenge to the shut-down or hold-down functions.

It is concluded that the margin to the percolation limit weight loss, estimated for Reactor 1 at its proposed EoG, adequately supports the inspection results to date and ongoing monitoring activities in demonstrating that widespread core damage is extremely unlikely.

7.2.2 Implications for Fuel Over-heating

To investigate the implications for crack initiation, brick splitting and consequential fuel overheating, of localised loss of strength, should weight loss extend beyond the percolation limit weight loss, finite element (FE) analyses have been undertaken (Reference 19). The results have previously been used to support continued operation of Reactor 2 (Reference 17). Weight losses were progressively increased and areas of the brick that exceeded the percolation limit weight loss were allocated very low stiffness to simulate zero strength. The results indicated that there was no enhancement of the peak hoop stress at the FCW until the MCI was at least 1.8GWd/t higher than that at which the percolation limit weight loss was first reached. This conclusion was insensitive to assumptions in thermal boundary conditions employed and implies an additional margin of about 3% (or 2 years operation) to the weight loss at which there could be an increased likelihood of crack initiation at the FCW. A sensitivity study was carried out involving the axisymmetric removal, from the FE model, of elements from the outside of the brick; this resulted in a progressive reduction in stresses at the FCW. These results indicate that localised loss of graphite strength at the highest weight loss locations would not significantly affect the likelihood of crack initiation at the FCW. The finite element analyses were subject to Peer Review by Professor Burdekin (Reference 20). The review considered the adequacy of the approach adopted, the sensitivity studies undertaken and the validity of the conclusions drawn. The results and outcomes of the analyses were supported and the approach to sensitivities on weight loss was considered sensible.

It is concluded that, even in the event of substantial volumes of graphite at ICW locations reaching weight losses in excess of the percolation limit weight loss, the likelihood of crack initiation at the FCW will not be significantly increased.

8 REVISED PROBABILISTIC ASSESSMENT

As well as affecting the predicted graphite weight losses, increasing MCI will also consequentially affect brick stresses, graphite strength and crack initiation probabilities. It will also influence EoG Wigner gaps, and hence potential crack face separation (gape), channel flow impairments and conditional clad melt probabilities. The state of the core is gradually changing through life in terms of graphite properties and operational parameters. The probabilistic assessment considers the core in its projected state at the proposed EoG (this has a direct influence on the estimates of conditional clad melt probabilities).

Since presentation of the probabilistic assessment for Oldbury Reactor 1 in Addendum 4 (Reference 5), refinements and improvements to the methodology have been made and these were applied in the more recent probabilistic assessment carried out in support of the safety case for the integrity of the Wylfa graphite cores (Reference 8). In support of the current proposal to extend operation of Reactor 1, a revised probabilistic assessment of the likelihood and consequences of graphite brick cracking has been carried out using this refined methodology (Reference 21).

8.1 Methodology

The revised probabilistic assessment follows the same general approach as that previously adopted for Reactor 1 in Reference 5, Reactor 2 in Reference 6 and more recently for the Wylfa graphite cores in Reference 8. The assessment is focussed on evaluating the likelihood of clad melt following return to service as a result of axial splitting of the graphite bricks, initiated by cracking at the FCW during a shutdown transient (SDT). It is based on a three-stage methodology:-

- Stage 1: Calculation of the probability of axial crack initiation at the FCW.
- Stage 2: Calculation of the conditional probability of crack progression and brick splitting.
- Stage 3: Calculation of coolant gas flow diversion and the conditional probability of clad melt.

The derived clad melt probabilities are then combined for all bricks to obtain the probability that fuel clad melt will occur somewhere within the core as a result of graphite brick cracking.

Stage 1 is based on comparing SDT stresses with graphite strengths; by combining the probability distribution for peak hoop stress at the FCW with the probability distribution for perpendicular flexural strength, the probability of the stress exceeding the graphite strength is estimated at each location within the core. For Stage 2, two alternative approaches are employed, the "crack propagation model" and the "double initiation model", as described in previous Addenda (References 5 and 6). In Stage 3, the conditional probability of clad melt is dependent on brick location, crack gape, coolant by-pass flow and the channel axial power profile.

There have been a number of detailed changes to the input parameters and assumptions since the previous probabilistic assessment of Reactor 1 (Reference 5) to reflect recent developments and data acquisition as well as the proposed extended generation period. These changes are summarised in Table 3 and their impact on the outcome of the probabilistic analysis is discussed in Section 8.2 below. The effects from the proposed re-fuelling strategy, which includes the transfer of partly irradiated fuel from Reactor 2 (Reference 7), are also included in the revised assessment. There have also been a number of enhancements to computer modelling efficiency, consistent with those adopted for the recent Wylfa probabilistic assessment (Appendix B of Reference 8).

8.2 Results

The results of the revised Reactor 1 probabilistic assessment are reported in Reference 21 and are summarised in Table 4. As for the previous probabilistic assessments, presented in Addendum 4 (Reference 5) and Addendum 5 (Reference 6), the overall, whole core single channel clad melt probability has been estimated for the measured SDT in

combination with the double initiation model and also for the pessimised SDT in combination with the crack propagation model.

It may be seen from Table 4 that, for both of these modelling approaches, the revised assessment (Reference 21), based on the parameters and assumptions presented in Table 3, predicts lower whole core single channel clad melt probabilities than the previous assessment (Reference 5). For operation up to the current EoG MCI of 31.5GWd/t, the revised probabilistic assessment has indicated whole core single channel clad melt probabilities of approximately 1 x 10^{-6} and 2 x 10^{-4} (per trip) for the crack propagation and double initiation models respectively. These probabilities are significantly lower than the corresponding values of 3.5 x 10^{-3} and 2.1 x 10^{-2} previously derived in support of the safety case for operation of Reactor 1 to the current EoG MCI.

These changes to the estimates of whole core single channel clad melt probability are a direct result of the refined methodology. The crack initiation probabilities reported in Reference 21 (Stage 1 as discussed above) are consistently higher than those previously estimated in Reference 5; this is the outcome of a number of changes to the values of input parameters including CTE, Dynamic Young's Modulus (DYM) and dimensional change. However, the refinements to the modelling of channel by-pass flows (Stage 3), as summarised in Table 3 and discussed further in Section 9.3 below, have resulted in higher estimated channel flow rates above the cracked brick and hence lower estimates of conditional clad melt probabilities. These lower conditional clad melt probabilities more than compensate for the higher predicted crack initiation probabilities.

The risk associated with operation to the proposed EoG MCI of 32.8GWd/t remains low; whole core single channel clad melt probabilities of less than 5.1 x 10^{-6} and 1.5 x 10^{-3} are indicated for the crack propagation and double initiation models respectively, significantly below the values previously derived in support of operation to the current EoG MCI limit.

8.3 Sensitivity Studies

Section 7.2 above reported that compressive strength measurements had been made on samples removed from ICW locations on Reactor 2; these measurements were considered appropriate for the prediction of percolation limit weight losses since peak weight losses were predicted at the ICW locations. However, it is noted in Reference 22 that the measurements indicated median compressive strengths approximately 12% lower than those derived from measurements on installed set samples (although the lower bound compressive strength data were similar for the two datasets). Noting that the assessment employs flexural strength data derived from measurements on installed sets, this result implies a possible non-conservatism within the analysis (this applies equally to the deterministic assessment discussed in Section 9 below). However, the observation of reduced strength at the ICW is consistent with previous analyses of diametral compression tests (measuring tensile strength) on trepanned slices which have indicated that samples extracted from the ICW are ~10% weaker and FCW slices ~10% stronger than predicted from fits to installed set data (Reference 23). The probability of crack initiation during a thermal transient is dominated by tensile hoop stress and flexural strength at the FCW. It is therefore judged that the potential non-conservatism associated with the assumed strength at the ICW is offset by the potential conservatism associated with the assumed strength at the FCW and that the overall assessment retains an appropriate degree of conservatism.

Nevertheless, a sensitivity study has been carried out (Reference 21) using a modified tensile strength relationship, consistent with the median strength reduction and lower bound strength values implied by the compressive strength tests undertaken on samples

removed from ICW locations on Reactor 2 (i.e. with a 12% median strength reduction and appropriately revised standard deviation). This resulted in an increase in predicted whole core single channel clad melt probabilities by factors of about 1.25 and 1.5 for the crack propagation and double initiation models respectively. These results demonstrate that the outcome of the probabilistic assessment is insensitive to such changes in strength data.

Reference 21 also presents the results of additional sensitivity studies on the effects of lower strengths and higher stresses. A 10% reduction in median strength has been shown to result in just over a factor of 2 increase in whole core single channel clad melt probability assuming the crack propagation model and just over a factor of 3 assuming the double initiation model. A 10% increase in mean peak hoop stress (and peak hoop stress standard deviation) has been shown to result in an increase in this probability by a factor of about 2 for both crack development models. These results provide further confirmation that the outcome of the probabilistic assessment is relatively insensitive to such changes in strength and stress data.

As discussed in Section 9.3 below, the coolant by-pass flow through a split brick depends on crack gape which is determined, in turn, by the local core geometry (Wigner gap sizes and key/key-way clearances) and the constraints provided by the balancing of pressure forces as the halves of the split brick move apart. Reference 21 reports a sensitivity study on the effects of graphite permeability and Wigner gap pressure, the main factors which determine the "equilibrium" position at which there will be no driving force to separate the brick halves. A 20% reduction in permeability causes a factor of just over 2 increase in overall clad melt probability. With the most extreme and unrealistic assumption that Wigner gap pressurisation is removed altogether, the overall clad melt probability is predicted to increase by slightly greater than one order of magnitude for the crack propagation model and by a factor of 2 for the double initiation model. It is concluded (Reference 21) that overall clad melt probabilities are not particularly sensitive to graphite permeability or Wigner gap pressure.

Finally, for notional operation up to an MCI of 34.0GWd/t, whole core single channel clad melt probabilities of less than 2.1×10^{-5} and 9.9×10^{-3} are indicated for the crack propagation and double initiation models respectively (Table 4). These values remain less than previously claimed in support of operation to the current EoG MCI of 31.5GWd/t and demonstrate that there is no "cliff-edge" associated with operating up to the proposed EoG MCI of 32.8GWd/t.

8.4 Conservatisms within the Probabilistic Assessment

Within an ideal probabilistic assessment the probability distribution for each input parameter would be rigorously characterised without bias. Where this has not been possible (because of sparse data, for example) bounding assumptions have been made. The more significant conservatisms within the present assessment are discussed in Reference 21 and, although it was not possible to quantify the overall effect of these, it was considered that the predicted fuel clad melt probabilities discussed above may be conservative by several orders of magnitude.

It is also noteworthy that, although both Oldbury cores have experienced hard trips (including two recent occasions on Reactor 2, see Section 6), extensive visual inspections of fuel channels have not identified any signs of moderator brick damage attributable to the mechanisms of interest in this assessment. These satisfactory results independently imply a degree of conservatism within the assessment of the likelihood of crack initiation.

8.5 Conclusion

The revised probabilistic assessment has indicated that the risk associated with operating Reactor 1 up to the proposed EoG MCI of 32.8GWd/t is low and that the probability of whole core single channel clad melt remains significantly lower than that previously predicted for the current EoG MCI of 31.5GWd/t.

9 REVISED DETERMINISTIC ASSESSMENT

The deterministic assessment of the integrity of the graphite cores, supporting the extant safety case, was developed in References 1 through 6. The assessment has been revised to support extended generation of Reactor 1 as discussed in Sections 9.1, 9.2 and 9.3 below. As for the probabilistic assessment, the deterministic assessment can be considered in three stages.

9.1 Crack Initiation

Historically, the deterministic assessments have been focussed on this stage of the assessment which involves the evaluation of UFs using a combination of conservative and best estimate graphite data and analysis. UFs are derived by comparing graphite brick stresses at locations throughout the core with the graphite strength at the corresponding locations. It has been established in previous addenda that the dominant risk of brick splitting and consequential clad melt arises from axial crack initiation at the FCW of channels in the flattened region during a SDT. The revised assessment has therefore focussed on the evaluation of tensile hoop UFs at the FCW during the SDT.

The results of calculations for Reactor 1 at the current EoG MCI limit of 31.5GWd/t were most recently presented in Addendum 5 (Reference 6, with further detailed results provided in Reference 24). The revised assessment closely follows the previous approach, but utilises revised EoG weight loss predictions, an updated flexural strength model as shown in Figure 2 and revised values for CTE, DYM and dimensional change (all of these changes being consistent with those adopted within the probabilistic assessment, see Table 3). The assessment (Reference 25) has been carried out at the current and proposed EoG MCI limits of 31.5GWd/t and 32.8GWd/t respectively.

A summary of the revised UFs from Reference 25 is presented in Table 5. UFs are presented for both octagonal and square bricks with interstitial cut-outs, these being either bounding of or approximately equal to those of the bricks without interstitial cut-outs for all layers. Table 5 also includes, for comparison, the previously derived UFs at 31.5GWd/t (Reference 24). It may be seen that, the current UF values are in most cases higher than those previously predicted indicating an increased likelihood of crack initiation. This increase is consistent with the results from stage 1 of the probabilistic assessment reported in Section 8 above and is a consequence of the revised input parameters and modelling assumptions as set down in References 21 and 25 and summarised in Table 3. The peak UF (octagonal brick, Layer 8) is increased from 0.67 to 0.71. However, the UFs at all layers remain below unity, indicating that the likelihood of crack initiation is low.

The revised assessment at the proposed EoG MCI of 32.8GWd/t indicates only a small increase in the UF values compared with those calculated at 31.5GWd/t; the peak UF at 32.8GWd/t is 0.74 at Layer 8 for the pessimised transient, compared with a peak of 0.71 at Layer 8 for the same transient at 31.5GWd/t. Table 5 also shows the results of an

assessment carried out at 34.0GWd/t (Reference 25), in order to investigate trends beyond the proposed EoG MCI; the peak UF value in this case is 0.78 at Layer 8.

In common with previous assessments, these results were derived from assessments using graphite strength measurements made on installed set samples. As noted in Section 8.3, it is judged that the potential non-conservatism associated with the assumed strength at the ICW is offset by the potential conservatism associated with the assumed strength at the FCW and the overall assessment retains an appropriate degree of conservatism. It may be noted however that, even if the assumed strength were to be reduced by 10%, the peak UF at 32.8GWd/t would be comfortably below unity at 0.81.

In conclusion, the results in Table 5 indicate that the likelihood of crack initiation during the pessimised SDT for operation to an MCI of 32.8GWd/t is not substantially higher than that for operation to 31.5GWd/t and that the likelihood increases only slowly for operation beyond the proposed EoG MCI.

9.2 Crack Progression and Brick Splitting

Crack initiation does not, itself, result in coolant flow diversion; even if a crack were to extend from the FCW to the outside of a brick over its full height there would be no significant leakage path. Brick splitting is required before the crack gape is sufficient to allow significant by-pass flow. It has previously been judged that progression from crack initiation to brick splitting is unlikely and this position remains unchanged. This judgement is underpinned by the "expert elicitation" carried out as part of the crack propagation model within the probabilistic assessment (Reference 21) which indicates a bounding crack progression probability of 10^{-3}.

In order to estimate the extent of coolant flow diversion (Section 9.3 below), it is necessary to understand the various ways in which bricks could potentially split. The most likely mode of brick splitting is for a diametral (180°) crack to form between the side key-ways on opposite sides of the brick, resulting in two equal brick halves (Reference 21). Alternatively, where there is an adjacent interstitial channel, cracks could develop between the fuel channel and a side keyway on one side of the brick and between the opposite side of the fuel channel and the interstitial channel; this would result in two unequal brick segments (135° splitting pattern). Both of these brick splitting modes are credible for octagonal bricks, although the 180° splitting pattern is more likely than the 135° split, due to the number of keyways and the relative sizes of the ligaments between the fuel channel/keyways and the fuel channel/interstitial channel (Figure 1). For square bricks, the 135° splitting pattern is extremely unlikely (Reference 21) as the ligament between the fuel channel and interstitial channel is much wider than those between the fuel channel and key-ways.

9.3 Coolant By-pass Flow and Clad Melt Risk

Given a split brick, the coolant gas could be diverted from the fuel channel, resulting in impaired cooling of the fuel elements above the affected brick. The risk of clad melt in the affected channel depends on the magnitude of the coolant by-pass flow through the split brick, the channel inlet flow rate, channel power and channel axial power profile. A detailed assessment of the consequences of brick splitting was presented in support of the probabilistic assessment reported in Addendum 4 (Reference 5). This assessment has been updated to include the refined approach to calculating channel by-pass flows, initially developed in Reference 26 and underwritten by Peer Review in Reference 27. This approach was previously adopted in support of the Wylfa probabilistic assessment (References 8 and 9). Although the results of the consequences assessment were

employed in the context of the probabilistic assessment (Reference 21), they were derived on a conservative basis and are therefore also applicable to this deterministic assessment.

The highest potential by-pass flows would result from splitting of bricks with interstitial cut-outs as the gas diverted from the fuel channel would have a direct path to the interstitial channel which itself would offer very little flow resistance. In general, the 135° splitting mode offers the shortest leakage path into the interstitial channel and, in the absence of any other influencing factors, would therefore be expected to be associated with the highest clad melt risks. The splitting of a brick without an interstitial cut-out would result in very little impairment of coolant flow in the fuel channel; the continuous keys would provide a restriction to the leakage flow within the Wigner gaps and very little of the gas would be diverted into the interstitial channel (Figure 1). Splitting of such bricks therefore makes no significant contribution to the overall risk of clad melt (References 5 and 21).

The coolant by-pass flow also depends on crack gape. Given double axial cracking of a brick, the pressure differential between the fuel channel and the Wigner gap would cause separation of the crack faces. As the crack faces moved apart, the reducing gas forces inside the crack would be opposed by the pressure in the Wigner gap and an equilibrium position would be reached which determined the crack gape. In the absence of such constraint to the movement of the two brick "halves", this separation would only be limited by engagement with bricks in the adjacent column; in which case the gape of the crack would be determined by the Wigner gap and key/key-way clearances.

The calculated crack gapes and conditional clad melt probabilities (Reference 21), for bricks with interstitial cut-outs and for both the 180° and 135° brick splitting modes, are presented in Table 6 for each core layer. The results are conservative in that the shortest potential crack length (corresponding to the highest out-of-channel leakage rate) is assumed to apply to all brick splitting modes. It may be seen that for the upper Layers, 9 - 11, the Wigner gap gas pressure is such that no crack face separation is expected. For the lower Layers, 2 - 4, the brick halves would separate until engagement with the adjacent brick occurred. For the remaining Layers, 5 – 8, the crack would open but the equilibrium position would be reached before brick engagement. These are significantly different results from those derived in the previous assessment (Reference 5) which assumed that the brick halves at all layers would separate to the extent permitted by the Wigner gaps. It is also confirmed in Table 6 that the highest conditional clad melt probabilities are those associated with the 135° brick splitting mode.

Overall, the refined by-pass flow modelling (as described in Table 3) and risk analysis has resulted in smaller estimates of crack gape and lower conditional clad melt probabilities than previously predicted. The risks are highest for the lower layers where the predicted gapes are greatest and more fuel elements are affected by the channel flow impairment.

9.4 Conclusions

1. As a consequence of the revised input parameters and modelling assumptions used in the assessment, UFs higher than those previously established for the current EoG MCI of 31.5GWd/t have been evaluated. Nevertheless, the UFs remain well below unity and do not increase significantly during the proposed period of extended generation. The results provide further assurance that the likelihood of crack initiation and hence the probability of brick splitting, in the event of a severe SDT during extended generation, is very low.

2. Although low for all bricks, the likelihood of crack initiation and hence the probability of brick splitting is higher for octagonal bricks than for square bricks; this is because the thermal stresses induced in the octagonal bricks are higher than those induced in the (smaller) square bricks.
3. For all brick geometries, the likelihood of crack initiation is generally higher in Layers 6 – 9; this is broadly a result of the thermal hoop stresses which are highest in the upper layers and predicted weight losses which reach a peak in the lower core layers.
4. Only splitting of those bricks with interstitial cut-outs could result in sufficient coolant by-pass flow to give a significant risk of fuel clad melt.
5. Given brick splitting, conditional clad melt probabilities are highest in the lower layers and extremely low in the upper layers.
6. The core layers at which the likelihood of crack initiation and brick splitting is highest are those associated with low conditional probabilities of clad melt.

These observations are consistent with the outcome of the probabilistic assessment summarised in Section 8 above. They provide additional support to the conclusion reached in Section 8.5 that the risk associated with operating Reactor 1 up to an EoG MCI of 32.8GWd/t is low and that the probability of whole core single channel clad melt is low.

10 RE-FUELLING ACTIVITIES DURING EXTENDED GENERATION

10.1 Thermal Stresses during Re-fuelling

As reported in Addendum 5 (Reference 6), on-load re-fuelling operations can lead to elevated thermal stresses in graphite moderator bricks. As fuel elements are removed, the amount of heat transferred to the channel coolant gas reduces whilst, at the same time, the reduced flow resistance in the channel increases the gas mass flow rate. Both of these effects lead to significantly lower fuel channel gas temperatures than for normal full power operation. The resulting effect on the graphite bricks is to cool the FCW thereby increasing the thermal gradient in the brick, leading to elevated tensile stresses at the FCW and hence increasing the likelihood of crack initiation.

Addendum 5 (Reference 6) reported an assessment of the effect of on-load re-fuelling operations on graphite stresses. It was concluded that the peak FCW hoop stresses for a fully de-fuelled channel were bounded by both the pessimised and measured trip hot channel stresses used in the deterministic and probabilistic assessments at that time. Addendum 5 also noted that re-fuelling plans included the retention of Fuel Element 1 (FE1) when re-fuelling channels in the flattened and inner intermediate regions of the core and also FE2 retention in the outer regions; the effect of fuel element retention is to reduce the tensile stresses associated with re-fuelling operations. Taking into account these fuel element retention schemes, it was argued that the risk of cracking at the FCW during re-fuelling was lower than the risk of cracking during a normal shutdown. Noting that 20% of the channels are re-fuelled each year, it was further argued that more than 5 years of re-fuelling would be required before the total reactor risk from re-fuelling was equivalent to a single reactor trip (Reference 6).

The effect of re-fuelling on graphite brick stresses has been considered further in Reference 28 where re-fuelling stresses were compared with those arising during a trip of

severity equal to that of the current inspection trigger trip (Reference 6 and see also Section 11.1 below). It was shown that stresses in a fully de-fuelled channel would be bounded by those associated with the current inspection trigger trip transient except for Layers 8 -11 (where the conditional clad melt probability is very low). With FE1 and FE2 retention, the de-fuelled channel stresses at all layers are bounded by those in the trigger trip. Thus, for all re-fuelling activities, thermal stresses, apart from those in the upper brick layers, are bounded by those associated with the inspection trigger trip SDT.

10.2 Clad Melt Risks Associated with Re-fuelling

From the results discussed in Section 9.3 above, the conditional risk of clad melt in the event of brick splitting is predicted to be highest for brick Layers 2 – 6, and negligibly low for the upper Layers 9 and 10 (see Table 6). Since the re-fuelling thermal stresses at the lower layers are bounded by those associated with the inspection trigger trip (Section 10.1), it may therefore be deduced that the nuclear safety risks associated with on-load re-fuelling activities over a five year period are no more onerous than those arising from a single inspection trigger trip. The on-load re-fuelling activities, undertaken without TV inspection of the graphite bricks, are therefore considered to be consistent with the logic of the post trip inspection criterion.

It was also concluded in Section 9.1 above that the likelihood of crack initiation during a SDT for operation to an MCI of 32.8GWd/t is not substantially higher than that for operation to 31.5GWd/t. Taking into consideration the discussion in Section 10.1 above, it may also be concluded that the likelihood of crack initiation during re-fuelling activities over the period of extended generation is unlikely to be substantially higher than that for operation to 31.5GWd/t.

10.3 Re-fuelling Risk Mitigation Measures

As indicated in Section 5 (and proposed in Section 11 below), it is intended to mitigate increased risks during the period of extended generation by introducing revised trigger trip levels. However, the implementation of revised trigger trip levels will not mitigate any increase in risk associated with re-fuelling activities during extended generation. Although the increased risk associated with re-fuelling is expected to be small, it is nevertheless appropriate to consider whether there are any reasonably practicable risk mitigation measures which could be implemented.

Appendix B identifies a number of reasonably practicable safety enhancements for implementation in support of extending generation on Reactor 1; the following risk mitigating measures, associated with re-fuelling, will be implemented:-

1. A requirement will be introduced to maintain the lock-on of the BCD close monitoring system for at least 10 minutes following the completion of channel re-fuelling, for on-load whole channel re-fuelling activities (i.e. those involving removal of FE1, approximately 140 channels in total). Identification of an anomaly could be indicative of a split brick. This requirement will be implemented via an amendment to the appropriate Plant Operating Instruction (POI).
2. If an opportunity arises, due to an unforeseen reactor shut-down for example, consideration will be given to bringing forward re-fuelling and priority will be given to those channels that are due for whole channel re-fuelling. This would avoid the on-load re-fuelling of some of those channels with the highest re-fuelling stresses.

10.4 Fuel Element Snagging

The effects of fuel element snagging on graphite brick integrity during re-fuelling activities are not explicitly addressed in the graphite safety case. It has been implicitly assumed that the likelihood of such incidents is low and that, in the event of snagging, the fuel hoist overload limit would prevent the application of any significant load increase. However, following discussions arising from an optioneering workshop (Reference 29), it is acknowledged that there remains some uncertainty over the validity of these assumptions. Work is therefore underway to establish the frequency of snagging events and the effectiveness of the fuel hoist overload limit (see Appendix B). This will facilitate an assessment of the effects of snagging during re-fuelling operations on graphite brick integrity. The outcome of this work will be reported and any safety case implications will be addressed in an appropriately categorised submission before operating beyond an MCI of 32GWd/t. In the interim, the associated risks are judged to be acceptably low on the following basis:-

- In the absence of snagging, the nuclear safety risks associated with on-load re-fuelling activities over a 5 year period are no more onerous than those arising from a single inspection trigger trip (see Section 10.2 above).
- It is judged that the fuel hoist overload limit is likely to prevent any significant load increase in the event of snagging.
- 5 channels, where re-fuelling difficulties had been recorded, were inspected during a recent unplanned outage of Reactor 2 in November 2010 and no evidence of damage due to re-fuelling was observed.
- As a prudent measure, further inspections will be undertaken, in channels where re-fuelling difficulties have been experienced, prior to the start of extended generation on Reactor 1 (see Section 12 below).

10.5 Inter Reactor Fuel Transfer

The strategy for extending generation from Reactor 1 includes the transfer of partially irradiated fuel from Reactor 2 once that reactor has been finally shut down, thereby making optimum use of the overall availability of fuel on the site (Reference 7). It is recognised that this re-fuelling strategy (together with the fuel retention schemes referred to in Section 10.1 above) will affect the Reactor 1 EoG core state and this has been reflected in the structural integrity assessments reported in Sections 8 and 9 above (see also Table 3).

The safety assessment of the physical processes involved in inter-reactor fuel transfer is presented in Reference 30. The partially irradiated fuel elements will be loaded into Reactor 1 in the same axial positions that they occupied in Reactor 2 so that previously retained elements will not be over-irradiated and changes in core reactivity will be minimised. Noting the limited period of extended generation, the inter-reactor transfer process will therefore have little or no effect on graphite weight loss distributions within the core. By monitoring and assessing any discharge difficulties, the loading of significantly distorted elements into Reactor 1 will be avoided (Reference 30). Even if attempts were made to load such fuel into Reactor 1, completion would be prevented by interference and the fuel element would be withdrawn; there would be no adverse effect on brick integrity. Any subsequent bowing or swelling of transferred fuel elements would not present a challenge to graphite core integrity (see Section 10.4 above) as there would be no requirement to remove such elements during the period of extended generation.

As described in Section 6 above, CRACKFINDER is used daily to compare CGOT data with the equivalent predicted temperatures from PANTHER. As the PANTHER model is

reset each month, based on observed temperatures, the effects of inter reactor fuel transfer will be automatically accommodated and this monitoring capability will therefore remain unaffected.

11 REACTOR TRIP RISK MITIGATION MEASURES

11.1 Current Arrangements

The risks associated with crack initiation, brick splitting, coolant flow diversion and fuel clad melt are currently mitigated by specifying two trip transients which, if exceeded, trigger actions prior to returning a reactor to service. The required actions, following a trip, were specified in Reference 6. They depend on the severity of the SDT that follows the trip and this is specified in terms of the rate of fall of bulk gas outlet temperature (T2). If the bulk T2 falls by more than 75°C (but less than 110°C) within 3 minutes, then TV inspections of 10% of the fuel channels in the flattened region are required prior to return to service. If T2 falls by more than 110°C within 3 minutes of the trip, then the safety case must be reviewed, via an appropriately categorised safety submission, prior to returning the reactor to service. Compliance with these requirements is intended to ensure that the likelihood of returning to service with significant graphite brick cracking is acceptably low.

11.2 Proposed Arrangements

As discussed in Sections 8 and 9 above, the risks associated with axial crack initiation, crack progression, brick splitting, flow diversion and clad melt for operation to 32.8GWd/t are predicted to be low. It is nevertheless recognised that extended generation will lead to further graphite weight loss and hence a real increase in the likelihood of brick cracking, coolant flow diversion and clad melt. To enhance safety, underpin the safety case for extended generation and reduce reliance on the revised assessments summarised in Sections 8 and 9, it is therefore proposed to mitigate this additional risk by reducing the severity of the specified trigger trips. Revised trigger trip levels are specified in Section 11.2.1 below to provide assurance that the risk of returning to service with significant graphite brick cracking is not increased throughout the period of extended generation.

It is also intended to introduce operational targets for the period of extended generation to minimise the risk of any reactor trip leading to a more severe transient than the revised upper trigger trip level, see Section 11.2.2 below.

11.2.1 Revised Trigger Trip Levels

The objective of introducing revisions to the specified trigger trip levels is to ensure that, following any Reactor 1 trip during the period of extended generation, the risk of returning to service with cracked graphite bricks is no greater than that associated with returning to service following a trip during current operations.

The hoop UF (stress/strength) at the FCW provides an indication of the likelihood of axial crack initiation during a SDT; the likelihood of crack initiation increases with increasing UF and decreases with decreasing UF. Although thermal stresses during a SDT are, to a first approximation, proportional to the reduction in T2 during the first 3 minutes following the trip, they are also proportional to Young's Modulus which decreases with increasing weight loss. Thus, to some extent, the reduction in strength due to increasing weight loss is offset by the reduction in stress due to decreasing Young's Modulus. This is illustrated by the

results discussed in Section 9.1 above and summarised in Table 5 which show only a modest increase in UFs with increasing weight loss.

A potential approach to risk mitigation (i.e. limiting the likelihood of crack initiation) throughout the period of extended generation is therefore to progressively reduce the current trigger trip levels in proportion to the predicted reduction in graphite strength. Referring to Table 1, it can be seen that the increase in predicted weight loss, between MCIs of 31.5GWd/t and 32.8GWd/t, is no more than 2.5 percentage points at any location. The corresponding reduction in strength may be derived using Figure 2 where it can be seen that, over the weight loss range of interest (20 – 40%), a weight loss of 2.5% corresponds to a strength reduction factor of approximately 0.8. Hence, risk mitigation could be provided by progressively reducing the current trigger trip levels of 110°C and 75°C (at 31.5GWd/t) by a factor of 0.8 to 88°C and 60°C respectively (at 32.8GWd/t).

For the upper core layers, the UF values and hence the likelihood of crack initiation are dominated by the thermal stresses during the SDT whereas, in the lower layers, the temperature changes are smaller and thermal stresses are therefore less dominant. Hence a progressive reduction in the trigger trip levels to 80% of their current values at the proposed EoG would not offer the same level of risk reduction across all layers. To ensure that there will be no significant increase in the likelihood of crack initiation at any brick layer, it is therefore proposed to progressively reduce the current trigger trip levels of 110°C and 75°C to 80°C and 55°C respectively by the EoG MCI of 32.8GWd/t (representing a reduction of 73% over the period of extended generation). Figure 3 shows the upper and lower trigger trip levels against which any trip of Reactor 1 during the period of extended generation will be assessed.

The deterministic assessment discussed in Section 9 above and reported in Reference 25 included the evaluation of UFs, assuming a SDT equivalent to the revised upper trigger trip level at an MCI of 32.8GWd/t. Table 7 presents the FCW hoop UF values for the revised upper trigger trip SDT (i.e. a drop of 80°C within 3 minutes of a trip at 32.8GWd/t). In the upper layers, where the likelihood of cracking is greatest, it may be seen that the UFs at 32.8GWd/t are considerably lower than those at 31.5GWd/t. This indicates that, for the upper core layers, the likelihood of brick cracking will decrease throughout the period of extended generation. For the lower layers, where UFs are lower, the values at 32.8GWd/t are similar to those at 31.5GWd/t, indicating that the likelihood of brick cracking will not increase for these layers. The probabilistic assessment, discussed in Section 8 above and reported in Reference 21, has also indicated that the whole core single channel clad melt probabilities at 32.8GWd/t will be similar to those at 31.5GWd/t if account is taken of the revised trigger trip levels. Further confirmation of the effectiveness of this proposed measure is provided in Section 11.3 below.

11.2.2 <u>Operational Targets</u>

In order to minimise the risk of any reactor trip leading to a more severe transient than the upper trigger limit shown in Figure 3, it is proposed to introduce operational targets for key reactor parameters during the period of extended generation.

Using historical reactor trip data, a simple linear relationship has been derived between the reduction in bulk gas outlet temperature ($T2_{bulk}$) at 3 minutes post-trip and the pre-trip value of ($T2_{bulk}$ –T1), where T1 is the bulk gas inlet temperature (Reference 31). This relationship is shown in Figure 4 and is used to establish the operational targets for Reactor 1.

From Figures 3 and 4 it may be deduced that, by introducing a target maximum for ($T2_{bulk}$ - T1) of 150°C at an MCI of 31.5GWd/t, reducing to 110°C at 32.8GWd/t, a margin of

approximately 10°C between the expected severity of potential trips and the upper trigger trip limit will be maintained.

Although compliance with this operational target will reduce the likelihood of a trip exceeding the upper trigger level, it is important to recognise that it does not guarantee this for all trips since the actual severity will also depend on the precise nature of the event leading to the trip.

11.3 Supplementary Structural Integrity Assessment

The deterministic assessment of crack initiation reported in Section 9.1 above focussed on the evaluation of UFs using a combination of conservative and best estimate graphite data and analysis. In particular, best estimate weight loss predictions and graphite strength values were assumed. The results of this assessment indicated that the risks associated with crack initiation during the pessimised transient during extended generation were not substantially higher than those for operation to 31.5GWd/t.

In order to provide additional confidence that the likelihood of crack initiation during the proposed period of extended generation will be low, further analyses have been undertaken (Reference 18). These take into account the revised upper trigger trip levels (Section 11.2 above) and conservatively employ upper bound ($+2\sigma$) weight loss predictions and lower bound (-2σ) strength predictions.

The results of this supplementary bounding assessment are summarised in Table 8. Up to the current EoG MCI of 31.5GWd/t, the limiting trip transient is characterised by 110°C drop in bulk T2 at 3 minutes post trip (i.e. the pessimised trip). For extended generation, the limiting transient is progressively reduced such that, at an MCI of 32.8GWd/t, it is characterised by a drop in T2 of 80°C in 3 minutes (see Section 11.2.1 above). The UFs presented in Table 8, for MCIs of 31.5GWd/t and 32.8GWd/t, are therefore based on these limiting transients.

It may be seen that, even with the very conservative assumptions on strength and weight loss, the UFs remain below unity, indicating that the likelihood of crack initiation is low during the period of extended generation. Again, UFs at the upper layers reduce significantly with increasing MCI and do not increase with MCI in the lower layers. These results therefore provide additional assurance that the likelihood of crack initiation during the period of extended generation is low and no greater than that associated with operation to the current EoG MCI.

12 INSPECTION AND MONITORING STRATEGY

No further statutory outages are planned for Reactor 1 up to its current EoG date of no later than 30th June 2011; the next statutory outage is due on 16th December 2012 and operation of the reactor will not continue beyond that date. This submission justifies the continued operation of Reactor 1 up to an MCI of 32.8GWd/t, a limit which is not anticipated to be reached before 16th December 2012. Reactor 1 was taken off load on 10th May 2011, before its current MCI limit of 31.5GWd/t was reached and a programme of graphite inspections and trepanning will be completed to provide further support for continued generation.

TV inspections will be undertaken of approximately 5% (i.e. 65) of the fuel channels in the flattened region and also of 6 interstitial channels from which samples will be trepanned

(see below). The fuel channel inspections will target sites where re-fuelling difficulties have been experienced as well as channels exhibiting high pre-trip CGOTs. The results of these inspections will be reported in an appropriately categorised safety submission prior to return to service of the reactor.

In view of the small minimum margin to the percolation limit weight loss predicted at the proposed EoG (see Section 7.2 above), a relatively large trepanning campaign will be carried out, focussed on the acquisition of samples from peak weight loss locations at the ICW. It is proposed to trepan 100 samples from interstitial channel locations and 20 samples from two flattened region fuel channels. The intention is to provide 20 whole samples from the interstitial channels for compressive testing to underwrite percolation limit weight loss predictions for Reactor 1. Samples from the remainder will be subject to slicing and density measurement for comparison with existing weight loss predictions. The implications of the results obtained from the compressive testing and density measurements will be addressed in a further submission at the appropriate category prior to operation of the reactor beyond an MCI of 32GWd/t.

The current requirements for TV inspections following a trip and for temperature probing following return to service are specified in Addendum 5 (Reference 6, Table 7). During the period of extended generation, the trigger trip level for inspection will be progressively reduced with increasing MCI in accordance with the proposals in Section 11.2 above. Table 9 specifies the inspection, monitoring and safety case review requirements for operation beyond an MCI of 31.5GWd/t.

Other monitoring activities (control rod drop and freedom of movement tests, analysis of re-fuelling difficulties and CRACKFINDER analysis) will continue in accordance with the extant safety case requirements (Reference 5).

13 FURTHER WORK

13.1 Implications of Fuel Element Snagging

Work is underway to establish the frequency of fuel snagging events during on-load re-fuelling activities and to assess the effectiveness of the fuel hoist overload limits as discussed in Section 10.4. The outcome of this work will be reported and any safety case implications will be addressed in an appropriately categorised submission prior to operation of the reactor beyond an MCI of 32.0GWd/t.

13.2 Flexural Strength Tests of Samples from Reactor 2

In common with previous assessments, the structural integrity analyses reported here employed flexural strengths derived from measurements made on installed set samples. It was judged in Sections 8.3 and 9.1 that the potential non-conservatism associated with the assumed strength at the ICW would be offset by the potential conservatism associated with the assumed strength at the FCW and that the overall assessment would therefore retain an appropriate degree of conservatism.

In order to provide further support to this judgement, consideration is being given to the removal of sufficiently large samples from the Reactor 2 core (following its final shut-down in June 2011) to enable flexural strength measurements to be undertaken. Initially, a feasibility study into the practicability of removing such samples is being undertaken. If it is

shown to be reasonably practicable, an appropriate sampling and testing programme will be developed and implemented.

13.3 Monitoring Assessment Panel

A Monitoring Assessment Panel will be set up to regularly review plant data in order to determine if there are any trends or anomalies that indicate potential damage within the graphite core (Appendix B). This will follow a similar approach to that already implemented by British Energy.

13.4 Transfer of Articulated Control Rods from Reactor 2

Following the final shutdown of Reactor 2, the 8 Articulated Control Rods (ACRs) from that reactor will be transferred to Reactor 1 (Appendix B). This will improve the likelihood of an adequate number of control rods entering the Reactor 1 core in the (unlikely) event of widespread core disruption. This transfer will be implemented via an appropriately categorised safety submission.

14 OVERVIEW OF THE SAFETY CASE

The safety case for the integrity of the Reactor 1 graphite core is based on a demonstration that:-

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

The core has been designed and constructed taking into account the graphite ageing mechanisms of radiolytic corrosion and dimensional change and its operation has been shown to be compatible with the above safety case requirements (as outlined below).

TV inspections are carried out at every outage providing direct evidence that the core remains in good condition; all of the Reactor 1 flattened region fuel channels have been inspected. Trepanning campaigns, conducted during outages, have produced numerous samples for off-site analysis to determine the rate of radiolytic weight loss throughout the core and other material property trends; the data have been used in structural assessments to assure the ongoing integrity of the core. In support of the proposal to extend operation to an EoG MCI of 32.8GWd/t, a further campaign of inspection and trepanning activities will be completed during the current Reactor 1 outage.

Channel bore profile measurements are made during statutory outages to ensure that dimensional changes are within expectations. Further practical checks are routinely carried out to confirm the continuing adequacy of the core geometry; re-fuelling operations are monitored and the unrestricted movement of control rods is regularly checked by freedom of movement tests during power operation and rod drop time measurements during reactor trips.

It is recognised that extending generation on Reactor 1 will result in continued radiolytic corrosion throughout the graphite core. However, the predicted upper bound peak weight loss at the proposed EoG has been shown to be lower than the lower bound percolation

limit weight loss. This provides confidence (supporting the results from inspection and monitoring activities) that the likelihood of widespread core damage, sufficient to challenge core shut-down and hold-down capability, is extremely low.

Structural integrity assessments have been undertaken that provide an understanding of the key factors that influence the potential for graphite cracking in a manner that could affect safety. This has shown that the potential for cracking during normal operation can be discounted and that the most onerous loading condition occurs during the first few minutes following a hard trip. The principal contributory factor is the thermal stress in the brick caused by the rapid reduction in coolant temperature, brought about by a successful shut-down. The assessment has shown that significant coolant flow reduction in the affected fuel channel can only occur if a brick splits into two parts that subsequently separate to create a substantial gape. In such circumstances the safety implications only arise during the subsequent return to power.

The deterministic and probabilistic assessments of the fault sequence have been based on conservative assumptions and cover the operating period up to the revised EoG MCI of 32.8GWd/t. These have shown that the likelihood of crack initiation during a hard trip is low and that the likelihood of progression to splitting and separation of the brick halves is also low. In the highly unlikely event that a brick does split, the assessment shows that the conditional likelihood of clad melt at full power is low.

Consideration has also been given to the risks associated with on-load re-fuelling and it has been concluded that the risks associated with crack initiation during the period of extended generation will not be significantly higher than those for operation to 31.5GWd/t.

The risks associated with increasing weight loss throughout the period of extended generation are to be mitigated by progressively reducing the severity of the trigger trips at which action (inspection or safety case review) is to be taken prior to returning Reactor 1 to service. In order to minimise the risk of a reactor trip exceeding the (revised) upper trigger trip level, operational targets will be introduced to maintain a margin between the expected severity of potential trips and the upper trigger trip level. As a further practicable measure, the BCD close monitoring system will be left in place for 10 minutes, during on-load whole channel re-fuelling, to monitor for anomalies that could indicate a split graphite brick. Hence the risks associated with extending operation to a revised EoG MCI of 32.8GWd/t, including those associated with re-fuelling activities, are expected to be no greater than the risks associated with operation to the current EoG MCI of 31.5GWd/t.

15 ASSESSMENT AGAINST THE NUCLEAR SAFETY PRINCIPLES

NSP2: Deterministic Principles

NP/SC 4927 Revision 1 Addendum 4 (Reference 5) presented a comprehensive review against the deterministic NSPs, demonstrating full compliance. Following this, Addendum 5 (Reference 6) concluded that the operation of Reactor 2 to an EoG MCI of 34.0GWd/t was also fully compliant with the deterministic NSPs. The deterministic safety arguments and assessments presented in Section 9 of this current Addendum 6 show that the increased risk of graphite brick cracking during a SDT, associated with operating Reactor 1 to an MCI of 32.8GWd/t, is very small. Furthermore, mitigating measures are proposed (Sections 11 and 13) to provide assurance that the risk of brick cracking associated with operation of Reactor 1 to an EoG MCI of 32.8GWd/t is no greater than that for current operation. It may

therefore be concluded that the proposal in this current Addendum, to extend generation on Reactor 1, is also fully compliant with the deterministic NSPs.

NSP3: Probabilistic Principle

Reference 5 considered the likelihood of clad melt in Reactor 1 and the consequential radiological releases. It was demonstrated, on the basis of a probabilistic assessment that the risks were at least Tolerable and ALARP. Furthermore, by taking into account the satisfactory results obtained from inspection of the flattened regions of both reactor cores, it was concluded that the risks were likely to be in the Broadly Acceptable region. A revised probabilistic assessment has been reported in Section 8 of this current Addendum 6, based on the updated modelling approach outlined in Table 3. The revised assessment has shown the risks of fuel clad melt associated with operation of Reactor 1 up to an EoG MCI of 32.8GWd/t to be lower than those previously estimated in Reference 5 for operation of Reactor 1 to an MCI of 31.5GWd/t. It may therefore be concluded that the risks associated with the proposed extension of generation on Reactor 1 are also at least Tolerable and ALARP and likely to be Broadly Acceptable.

16 CHANGES TO PROCEDURES

There are no changes to Operating Rules required as a result of this submission.

POIs will be amended to include the requirement to maintain the lock-on of the BCD close monitoring system for at least 10 minutes following completion of channel re-fuelling for all on-load whole channel re-fuelling activities (Section 10.3).

The Station Physics Departmental Instruction for analysis of graphite thermocouple trip data will be updated to incorporate revised trigger trip levels in accordance with the proposals in Section 11.2.1 and Figure 3.

The Station Physics Departmental Instruction for the issue of reactor trimming target values will be updated to incorporate the provision of advice on reactor trimming consistent with the revised operational targets (Section 11.2.2) during the period of extended generation.

17 TRAINING

Operator training will be conducted to provide an awareness of the safety case requirements for the revised operating targets and BCD close monitoring system lock-on during the period of extended generation.

18 QUALITY ASSURANCE

This submission has been prepared in accordance with the requirements of Site Management Control Procedure MCP21 and the EF (formerly EWST) Quality Management System for the production of safety cases and their supporting references.

19 INDEPENDENT NUCLEAR SAFETY ASSESSMENT

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

20 CONCLUSION

A safety case for the Reactor 1 graphite core has been established for continued operation up to an MCI of 32.8GWd/t, this being unlikely to be exceeded before December 2012.

21 RECOMMENDATIONS

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

1. **AGREE** the proposal to operate Reactor 1 to an increased EoG MCI limit of no more than 32.8GWd/t.
2. **NOTE** that, with respect to the integrity of the graphite core, the nuclear safety risk associated with the proposal is low.
3. **NOTE** that, in support of the proposal, revised trigger trip levels have been specified to provide assurance that the risk of returning to power with significant graphite brick cracking is not increased throughout the period of extended generation.
4. **NOTE** that operational targets will be introduced for the period of extended generation to minimise the risk of any reactor trip leading to a more severe transient than that associated with the revised upper trigger trip.
5. **NOTE** that a requirement will be introduced to maintain the lock-on of the BCD close monitoring system for at least 10 minutes following the completion of channel re-fuelling, for all on-load whole channel re-fuelling activities.
6. **NOTE** that, in support of the proposal, a campaign of inspection and trepanning activities will be completed during the current outage.
7. **NOTE** that the implications of the results obtained from density measurements and compressive testing of the trepanned samples removed during the Reactor 1 outage will be addressed in an appropriately categorised safety submission prior to operating Reactor 1 beyond 32.0GWd/t.
8. **NOTE** the programme of further work identified in Section 13.

22 REFERENCES

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18. Magnox North Report MEN/EWST/OLA/REP/0049/10, Oldbury Reactor 1 Upper Bound Deterministic Utilisation Factors with Young's Modulus Correlated to Flexural Strength, [redacted] May 2011.

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Slice	Layer	Weight Losses (%) Predicted in Add 4 (Reference 5)		Revised Weight Loss Predictions (%) from Reference 15					
		31.5 GWd/t		31.5 GWd/t		32.8 GWd/t		34.0 GWd/t	
		BE	UB	BE	UB	BE	UB	BE	UB
1	4L	*30.7*	*35.6*	31.0	35.7	33.2	37.9	35.4	40.1
	4U	*32.6*	*37.5*	32.8	37.5	35.3	40.0	37.6	42.3
	5L	*31.6*	*36.4*	30.9	35.9	33.1	38.1	35.2	40.2
	5U	*31.4*	*36.2*	30.9	35.9	33.1	38.1	35.2	40.2
	6L	*31.5*	*35.4*	31.1	35.1	33.3	37.3	35.4	39.4
	6U	*29.9*	*33.8*	29.5	33.6	31.6	35.6	33.6	37.6
	7L	*28.7*	*33.6*	28.6	33.1	31.0	35.4	33.2	37.7
	7U	*26.4*	*31.3*	26.4	30.9	28.6	33.0	30.7	35.1
	8L	*26.6*	*30.9*	26.2	31.0	28.3	33.1	30.3	35.1
	8U	*23.5*	*27.8*	23.3	28.1	25.2	30.0	27.0	31.7
2	4L	*32.3*	*37.5*	33.4	38.6	35.2	40.4	37.0	42.1
	4U	*34.5*	*39.7*	35.8	41.0	37.8	43.0	39.7	44.8
	5L	*34.7*	*39.0*	35.2	40.0	37.4	42.3	39.6	44.4
	5U	*35.2*	*39.6*	35.8	40.6	38.1	42.9	40.3	45.1
	6L	*35.8*	*39.3*	35.9	39.8	38.0	42.0	40.1	44.0
	6U	*35.2*	*38.7*	35.3	39.2	37.4	41.3	39.4	43.3
	7L	*33.1*	*41.3*	33.7	41.1	36.2	43.6	38.6	46.0
	7U	*31.8*	*39.9*	32.3	39.7	34.7	42.1	37.0	44.4
	8L	*32.5*	*36.6*	31.9	36.4	33.7	38.3	35.5	40.0
	8U	*30.4*	*34.5*	29.9	34.4	31.6	36.1	33.2	37.8
3	4L	*32.3*	*37.5*	33.4	38.4	35.2	40.2	36.9	41.9
	4U	*33.8*	*39.0*	35.1	40.2	37.1	42.1	38.9	43.9
	5L	*33.2*	*37.2*	34.7	39.2	37.0	41.4	39.1	43.6
	5U	*36.6*	*40.6*	36.3	40.8	38.7	43.1	40.9	45.3
	6L	*36.1*	*40.6*	36.2	40.5	38.2	42.5	40.1	44.4
	6U	*35.0*	*39.4*	34.6	38.9	36.5	40.8	38.3	42.6
	7L	*32.9*	*39.5*	33.1	39.5	35.3	41.7	37.4	43.8
	7U	*31.3*	*37.9*	32.5	38.9	34.6	41.0	36.6	43.0
	8L	*32.8*	*37.2*	31.9	36.4	33.7	38.2	35.4	40.0
	8U	*30.4*	*34.8*	29.8	34.4	31.5	36.1	33.1	37.6

Table 1: Reactor 1 FCW Flattened Region 2010 Predicted Weight Losses showing Best Estimate (BE) and 2σ Upper Bound (UB) Values (corrected for volumetric change, deposits and immersion). 2007 Predicted Values shown at 31.5GWd/t for Comparison.

	Maximum Weight Loss at 31.5GWd/t (%) (Reference 18)	**Maximum Weight Loss at 32.8GWd/t (%)** (Reference 18)	**Percolation Limit Weight Loss (%)** (Reference 17)
Best Estimate	38.9	41.3	52
Upper Bound (2σ)	43.9	46.3	
Lower Bound (2σ)			47

Table 2: Maximum Predicted and Percolation Limit Weight Losses

Parameter	Comparison with Previous Assessment
EoG MCI	Changed from 31.5GWd/t to 32.8GWd/t to reflect the proposal for extended generation.
EoG Weight Loss Predictions	Updated to include data from the 2009 Reactor 1 trepanning campaign.
Flexural Strength Model	A revised flexural strength model (Reference 32 and Figure 2) has been adopted. This statistical regression model is the same as that used in recent deterministic assessments (References 6 and 23) and includes virgin graphite weight loss data to provide better predictions of flexural strengths at low weight losses.
Trip Transients	The pessimised and measured SDTs are unchanged.
CTE Temperature Dependence	In the previous assessment, correction factors were applied to account for the offsets between measured values (from virgin graphite material) and predictions (largely based on Materials Test Reactor (MTR) data) for both parallel and perpendicular CTE. A similar approach has been adopted for the current assessment but with updated correction factors that reflect the data obtained from recent measurement campaigns (Oldbury and Wylfa specific data for irradiated material, Reference 33). With the removal of uncertainties, the CTE measurements at high temperatures indicate higher temperature dependence than previously assumed (Reference 8) hence the current probabilistic assessment is more onerous than the previous assessment with respect to CTE assumptions.
Dimensional Change	Changed, following analysis of reactor measurements using NOREBORE and consequential revised estimates for perpendicular dimensional change temperature dependence. (The outcome of scoping calculations was previously reported at the time of presentation of the Wylfa probabilistic assessment (Reference 8) but the revised estimates were not used in that assessment). With the removal of uncertainties, the revised dimensional change equations result in lower predicted hoop stresses, hence, although considered bounding, the current probabilistic assessment is less onerous than the previous assessment with respect to dimensional change assumptions.
Young's Modulus (DYM)	DYM values have been changed to reflect revised correction factors to the MTR data, following a review of all available measurement data including those from recent trepanning campaigns (Reference 33). The revised correction factors are 1.16 for perpendicular DYM and 1.29 for parallel DYM (previously 1.15 for both directions).
Core Geometry	The core geometry, including estimates of Wigner gaps, is unchanged (both previous and current analyses pessimistically use Wigner gaps estimated for an MCI of 33GWd/t).
Crack Progression	The modelling of crack progression is unchanged.
Limiting Crack Gapes	Changed. Updated analyses make allowance for Wigner gap pressurisation in calculating crack gapes.
By-pass Flow Analysis	A refined approach to calculating channel by-pass flows (Reference 26) was adopted in support of the Wylfa probabilistic assessment (Reference 8). The revised evaluation of hydraulic loss coefficients at the crack-to-fuel-channel junction resulted in reduced out-of-channel leakage rates and higher predicted channel flow rates above the cracked brick. The same approach has been applied in the current Oldbury assessment.
EoG Core State	The revised Probabilistic assessment at 31.5GWd/t uses the same EoG core state as the previous assessment. Assessments at 32.8GWd/t and 34.0GWd/t utilise an updated Reactor 1 core state at an MCI of 33.0GWd/t (Reference 34) which takes account of the effects of loading enriched fuel, fuel element retention and inter-reactor fuel transfer.

Table 3: Revised Probabilistic Assessment Input Parameters and Modelling Assumptions – Comparison with Addendum 4 (Reference 5)

	Current Predictions (Reference 5)	**Revised Predictions** (Reference 21, Table 66)		
MCI	**31.5GWd/t**	**31.5GWd/t** (i)	**32.8GWd/t**	**34.0GWd/t**
Measured Transient Double Initiation Model	2.1×10^{-2}	2×10^{-4}	1.5×10^{-3}	9.9×10^{-3}
Pessimised Transient Crack Propagation Model	3.5×10^{-3}	1×10^{-6}	5.1×10^{-6}	2.1×10^{-5}

Table 4: Predicted Whole Core Single Channel Fuel Clad Melt Probabilities

(i) These values are quoted to a lower precision as they have been obtained by scaling values derived for 32.8GWd/t to reflect the reduction in conservatism inherent in the clad melt risk analysis (Section 4.3 of Reference 21)

Core Layer	**31.5GWd/t** (Reference 24)	**31.5GWd/t** (Revised predictions from Reference 25)	**32.8GWd/t** (Reference 25)	**34.0GWd/t** (Reference 25)
11	0.39	0.46	0.46	0.47
10	0.44	0.49	0.49	0.49
9	0.48	0.52	0.54	0.54
8	0.55	0.57	0.59	0.64
7	0.57	0.58	0.62	0.67
6	0.61	0.57	0.62	0.72
5	-	0.49	0.56	0.63
4	-	0.43	0.50	0.59
3	-	0.31	0.36	0.40
2	-	0.19	0.18	0.19

a) Square Bricks with Interstitial Cut-outs

Core Layer	**31.5GWd/t** (Reference 24)	**31.5GWd/t** (Revised predictions from Reference 25)	**32.8GWd/t** (Reference 25)	**34.0GWd/t** (Reference 25)
11	0.50	0.61	0.61	0.60
10	0.56	0.63	0.63	0.64
9	0.61	0.67	0.67	0.68
8	0.67	0.71	0.74	0.78
7	0.64	0.66	0.72	0.76
6	0.62	0.62	0.68	0.75
5	-	0.53	0.59	0.66
4	-	0.46	0.53	0.59
3	-	0.28	0.31	0.34
2	-	0.16	0.17	0.17

b) Octagonal Bricks with Interstitial Cut-outs

Table 5: FCW Hoop Utilisation Factors for Pessimised Shutdown Transient

Core Layer	Square Bricks with Interstitial Cut-outs				Octagonal Bricks with Interstitial Cut-outs			
	180° Crack Geometry		135° Crack Geometry		180° Crack Geometry		135° Crack Geometry	
	Gape (mm)	Clad Melt Probability	Gape (mm)	Clad Melt Probability	Gape (mm)	Clad Melt Probability	Gape (mm)	Clad Melt Probability
11	0	0	0	0	0	0	0	0
10	0	0	0	0	0	0	0	0
9	0	0	0	0	0	0	0	0
8	1.4	5.6×10^{-9}	1.0	6.1×10^{-8}	0.3	5.6×10^{-9}	0.4	6.1×10^{-8}
7	2.5	8.8×10^{-8}	1.7	3.0×10^{-7}	1.6	2.0×10^{-8}	1.1	3.0×10^{-7}
6	3.6	5.2×10^{-6}	2.4	5.0×10^{-6}	2.6	3.2×10^{-7}	1.7	1.1×10^{-6}
5	4.6	1.3×10^{-4}	3.1	2.4×10^{-4}	3.6	1.5×10^{-5}	2.4	1.4×10^{-5}
4	4.1	8.5×10^{-5}	3.9	4.9×10^{-3}	4.1	8.5×10^{-5}	3.2	6.3×10^{-4}
3	3.2	7.5×10^{-6}	3.3	1.2×10^{-3}	3.2	7.5×10^{-6}	3.3	1.2×10^{-3}
2	2.3	2.4×10^{-7}	2.7	6.5×10^{-5}	2.3	2.4×10^{-7}	2.7	6.5×10^{-5}

Table 6: Crack Gapes and Conditional Clad Melt Probabilities at MCI of 32.8GWd/t
(Reference 21: gapes derived from Tables 19 and 20, clad melt probabilities from Table 24)

Core Layer	Square Bricks with Interstitial Cut-outs		Octagonal Bricks with Interstitial Cut-outs	
	31.5GWd/t	32.8GWd/t	31.5GWd/t	32.8GWd/t
11	0.46	0.37	0.61	0.49
10	0.49	0.39	0.63	0.50
9	0.52	0.42	0.67	0.52
8	0.57	0.47	0.71	0.57
7	0.58	0.50	0.66	0.54
6	0.57	0.53	0.62	0.53
5	0.49	0.48	0.53	0.47
4	0.43	0.44	0.46	0.41
3	0.31	0.32	0.28	0.27
2	0.19	0.18	0.16	0.15

Table 7: FCW Hoop Utilisation Factors for Revised Upper Trigger Trip Shutdown Transient (Reference 25)

Core Layer	Square Bricks with Interstitial Cut-outs		Octagonal Bricks with Interstitial Cut-outs	
	31.5GWd/t	32.8GWd/t	31.5GWd/t	32.8GWd/t
11	0.52	0.43	0.68	0.55
10	0.61	0.50	0.78	0.62
9	0.68	0.54	0.84	0.65
8	0.79	0.67	0.96	0.78
7	0.81	0.68	0.93	0.80
6	0.80	0.71	0.91	0.77
5	0.72	0.68	0.80	0.69
4	0.65	0.64	0.67	0.64
3	0.40	0.40	0.37	0.37
2	0.21	0.20	0.20	0.19

Table 8: FCW Hoop Utilisation Factors from Supplementary Deterministic Assessment
(Upper bound (+2σ) weight losses and lower bound (-2σ) strengths (Reference 18))

Activity	Reactor Status	Current Requirements	Proposed Requirements
TV Inspections	Hard Trip (i)	10% of Flattened Region	10% of Flattened Region
	Second Hard Trip (ii)	None	None
Temperature Probing	Soft Trip (iii)	10% within 3 months	10% within 3 months
	Return to service after statutory shutdown with Soft Trip	Sufficient to ensure 10% combined TV inspection and probing	N/A
	Return to service after any other outage with Soft Trip	Sufficient to ensure 10% combined TV inspection and probing	Sufficient to ensure 10% combined TV inspection and probing

(i) Fall in bulk T2 greater than temperature illustrated in Figure 3 within 3 minutes

(ii) Fall in bulk T2 greater than temperature illustrated in Figure 3 within 3 minutes, but trip occurs within 1 full power month (80MWd/t) of previous trip and is less onerous than previous trip

(iii) Fall in bulk T2 less than temperature illustrated in Figure 3

Table 9: Post Trip Inspection & Monitoring and Safety Case Review Requirements

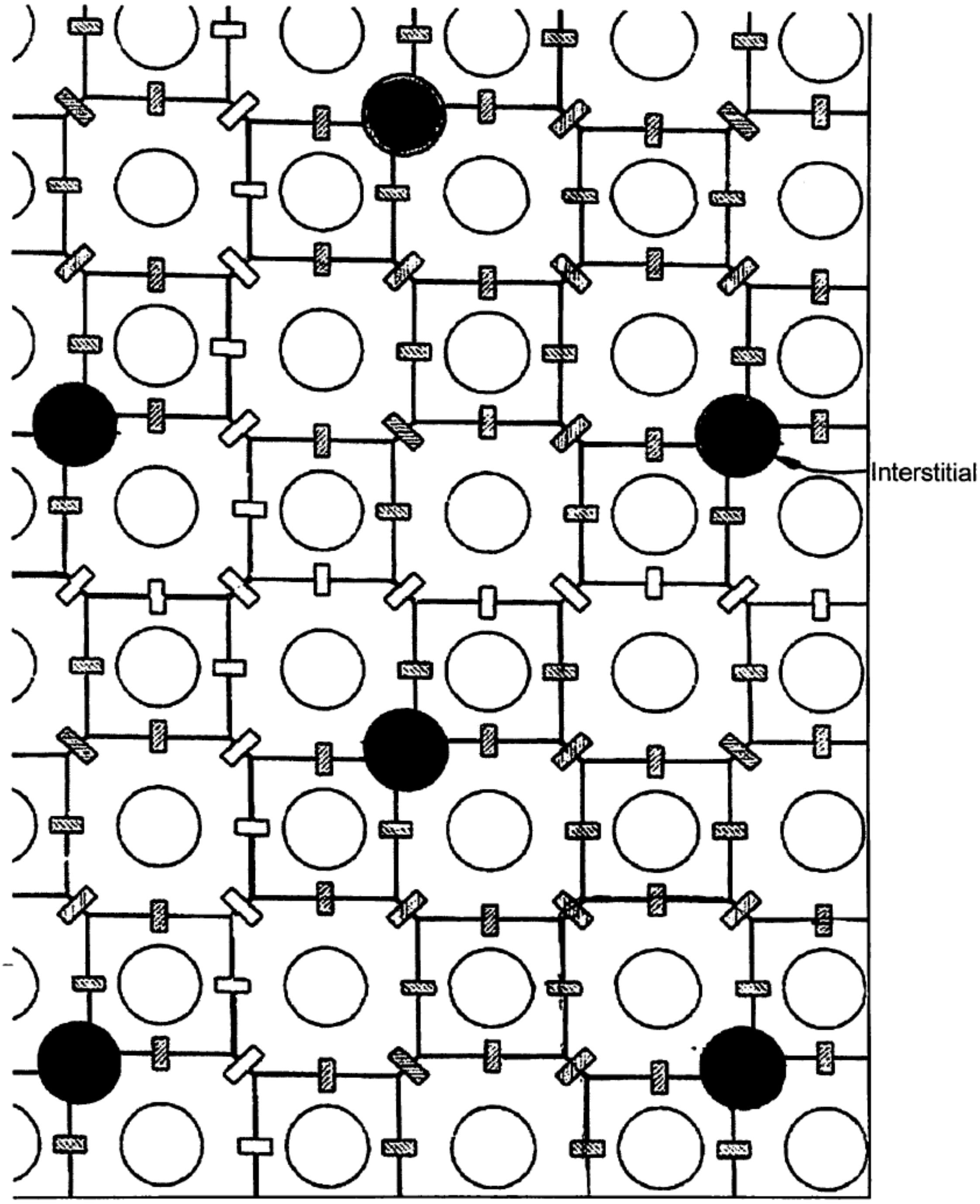

Figure 1: Plan View of Graphite Bricks and Keys, Showing Octagonal and Square Bricks, Fuel Channels, Side and Corner keyways and Interstitial Channels.
(Interstitial Channels are shaded black. Shading of keys is random and is not of any significance)

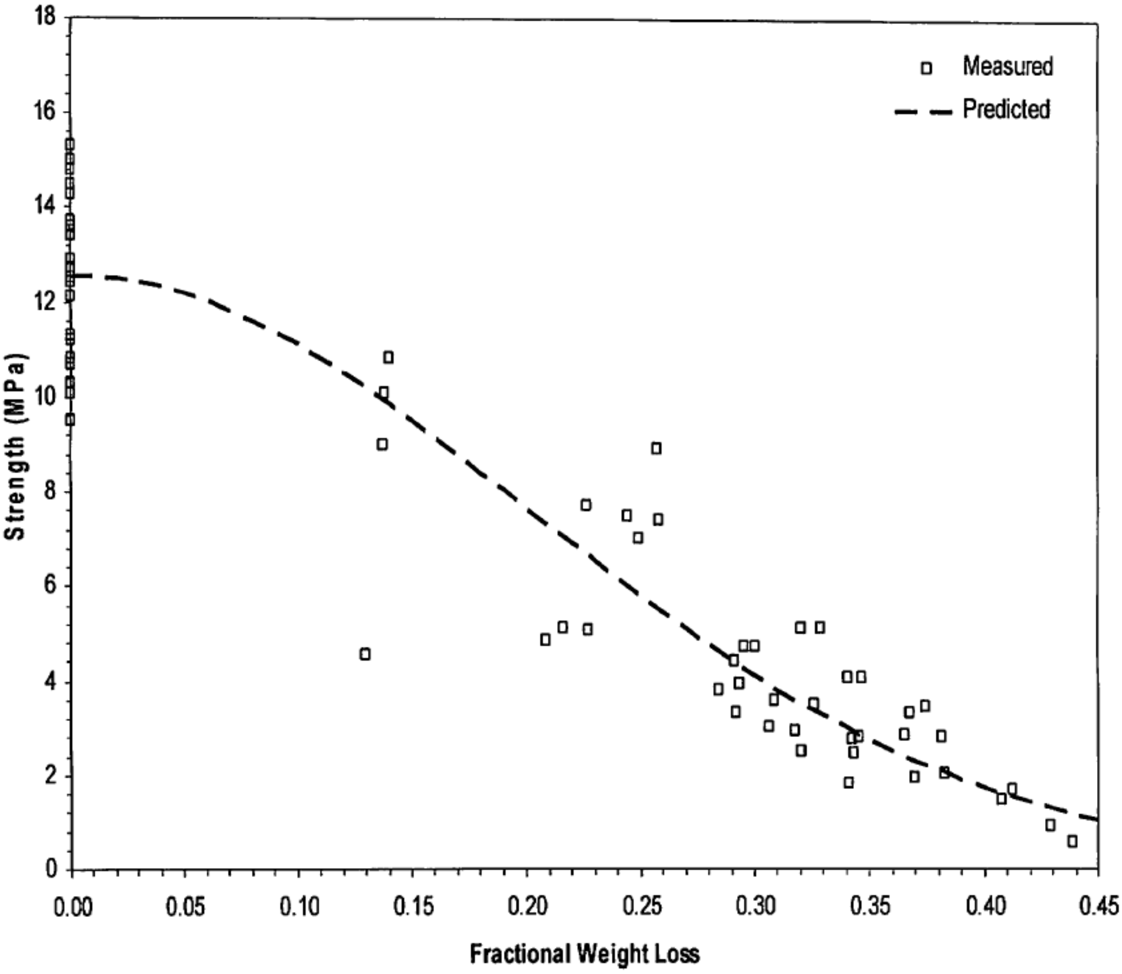

Figure 2: Perpendicular Flexural Strength Model (Reference 32)

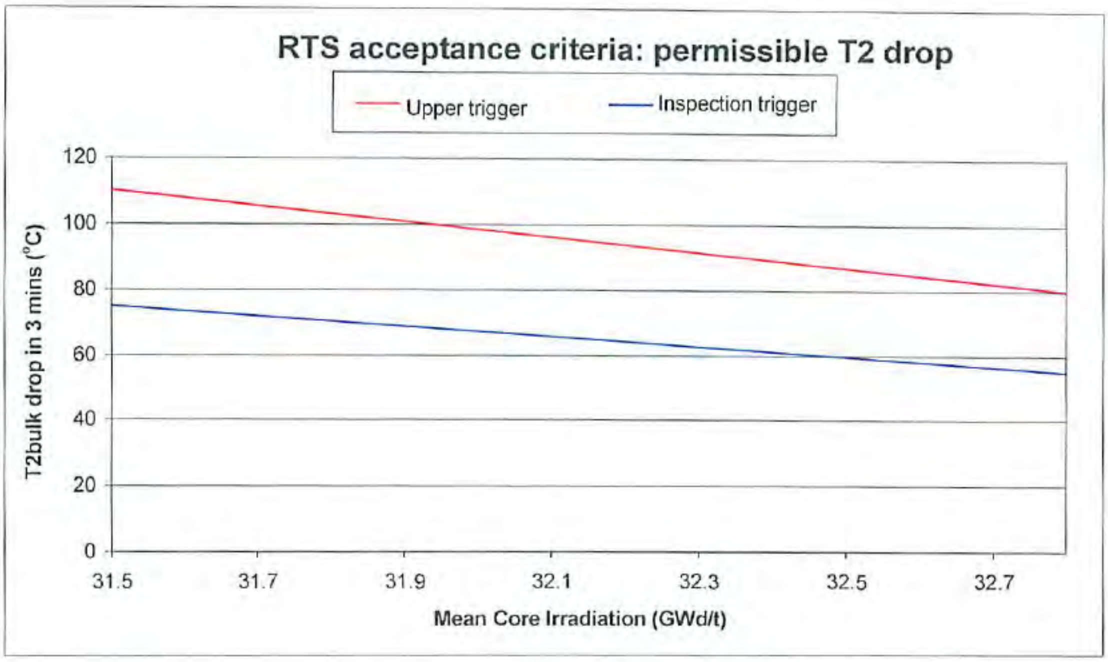

Figure 3: Proposed Return to Service Criteria following a Trip of Reactor 1

(Inspection required if: $\Delta T2_{bulk} = 559.165 - 15.385M$
Safety Case Review required if: $\Delta T2_{bulk} = 836.923 - 23.077M$
where $\Delta T2_{bulk}$ is temperature drop in 3 minutes and M is the MCI in GWd/t)

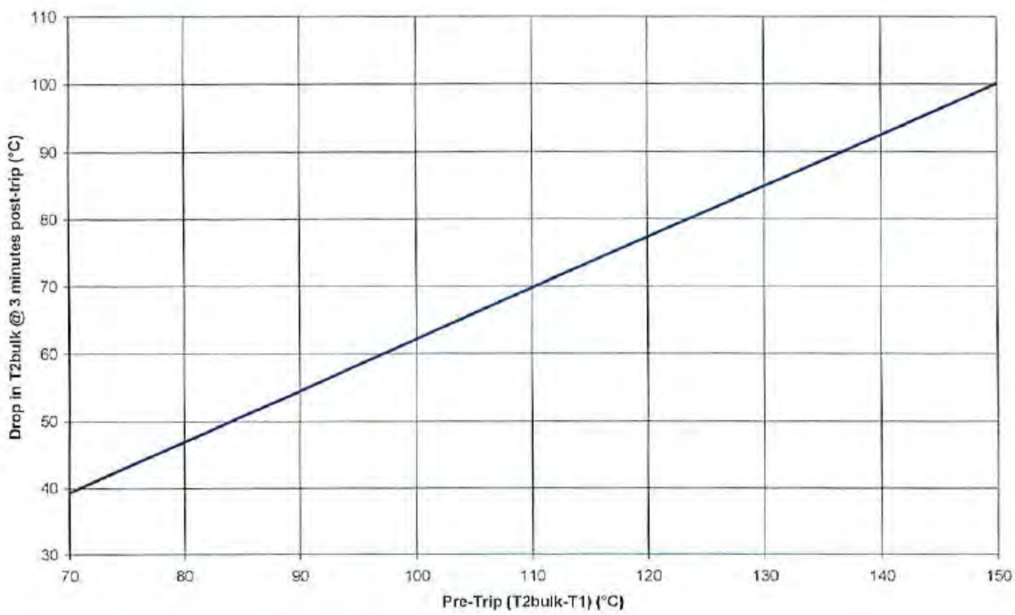

Figure 4: Predicted Variation of Post-trip Drop in $T2_{bulk}$ with Pre-trip Value of ($T2_{bulk}$ – T1)

APPENDIX A: VERIFICATION PLAN

VERIFICATION PLAN	No: NPSC4927Add6_VP_001 Issue: 1		PF009
Author [redacted]	Approved: [redacted]	Date: 11/05/11	
Document Ref: **NP/SC 4927 Revision 1 Addendum 6** Document Title: **A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation: Proposal to Operate Reactor 1 to a Mean Core Irradiation of 32.8GWd/t** Date Verification required by: May 2011			Issue: n/a

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 sequence)
		Probability	Consequence	
	Only the nuclear safety consequences of errors have been considered in the categorisation below. All verification activities satisfy QA Grade 2.			
1	Conclusion and Recommendations	**L**	**H**	These aspects will be subject to a final verification check by the lead verifier, SCO, SCA and additional review by senior Station staff.

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	Check all statements against cited references where possible.
	All	Consider all significant judgements, and confirm agreement with verifier and SCO
	All	Check completeness of addressing all comments. Retain all written comments
	All	Re-read entire document, check editorial quality of submission

Form PF010 may be used to record self verification activities.

INDEPENDENT VERIFICATION

Originator(s)	Verifier(s)
1. [redacted]	1. [redacted]

Initial to confirm pre-verification discussion has taken place: Originator	Initial to confirm pre-verification discussion has taken place: Verifier	Document Section	Scope, input document, acceptance criteria	Risk No(s)	Verification Statement Required
KAS	CPG	All	The presentation is clear and the layout, grammar and spelling are satisfactory. The information from the references is correctly quoted and the data have been accurately transcribed. The safety arguments are logically based and sound. All significant judgements are identified and are reasonable. The NSPs are appropriately addressed. The conclusions and recommendations are supported by the arguments presented in the submission. The specialist verifiers identified on this plan cover all aspects of the submission where specialist verification is appropriate. It is confirmed that no additional specialist verifiers are needed. The authors of the key references have confirmed the appropriate use of the references within the safety case. In particular:-		yes
		11.3	L.E Easterbrook – MEN/EWST/OLA/REP/0049/10		
		9.1	L.E Easterbrook - MEN/EWST/OLA/REP/0065/10		
		8, 9.2, 9.3	D.K.Anderson - MEN/EWST/OLA/REP/0001/11		
[redacted]		All	The main supporting references relating to graphite integrity have been identified in the submission. The appropriate data, information and conclusions have been extracted from the key references and are adequately presented in the submission.		written confirmation to lead verifier
		Table 3	The table represents a complete and comprehensive list of the significant changes in the assessment methodology since Addendum 4.		
		1, 2	The descriptions of the plant and its operational history are correct.		written confirmation to lead verifier
		6	Operational procedures and inspection and monitoring		

			arrangements are correct.		
		10, 11, 12, 13, 16, 17	The revised operational procedures, operational targets and further work commitments are acceptable to the Site.		
		21	The Recommendations are acceptable to the Site	1	
[redacted]		All	Overall adequacy of the safety case including consistency with other safety cases. Assessment against NSPs is acceptable.		written confirmation to lead verifier

Use form PF010 to record verification activities.

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:

APPENDIX B: REVIEW OF REASONABLY PRACTICABLE MEASURES TO REDUCE RISKS

Since 2002, numerous workshops have been held to review the overall graphite safety case methodology and "optioneering" has been employed to identify potential safety enhancements. In support of this current proposal to extend generation on Reactor 1, an optioneering workshop was held in September 2010 (Reference 29), involving a broad range of appropriate expertise drawn from the Oldbury and Wylfa sites, Engineering Waste Strategy and Technical (EWST), Project Delivery Organisation (PDO), Independent Nuclear Safety Assessment (INSA) and British Energy. The objective of the workshop was to identify reasonably practicable measures which could enhance the graphite safety case and reduce the risks associated with extended generation.

58 options were identified at the optioneering workshop but most of these were judged to be unrealistic or clearly not practicable. However, 16 options were retained for further consideration outside the workshop. Most of the retained options were subsequently rejected with the basis of their rejection being provided in Reference 29. It was also recognised that one of the retained options, reducing the severity of the trigger trips for inspection and safety case review, was already part of the safety case strategy and this is covered in detail in the main submission. On completion of the optioneering process, the following activities were identified in Reference 29 for implementation:-

1. A Monitoring Assessment Panel will be set up to regularly review plant data in order to determine if there are any trends or anomalies that indicate potential damage within the graphite core. This will follow a similar approach to that already implemented by British Energy.
2. Following the final shutdown of Reactor 2, the 8 Articulated Control Rods (ACRs) from that reactor will be transferred to Reactor 1. This will improve the likelihood of an adequate number of control rods entering the Reactor 1 core in the (unlikely) event of widespread core disruption. This option will be implemented via an appropriately categorised safety submission.
3. A requirement will be introduced to maintain the lock-on of the BCD close monitoring system for at least 10 minutes following the completion of channel re-fuelling, for all on-load whole channel re-fuelling activities (i.e. those involving removal of FE1, approximately 140 channels in total). Identification of an anomaly could be indicative of a split brick. This option will be implemented via an amendment to the appropriate Plant Operating Instruction (POI).
4. If an opportunity arises, due to an unforeseen reactor shut-down for example, consideration will be given to bringing forward re-fuelling and priority will be given to those channels that are due for whole channel re-fuelling. This would avoid the on-load re-fuelling of some of those channels with the highest re-fuelling stresses.

A further retained option identified in the workshop was associated with limiting the effects of fuel element snagging on graphite brick integrity by reducing the fuel hoist overload limit. As noted in Reference 29, the effects of snagging had not been explicitly addressed within the graphite safety case; it had been implicitly assumed that the likelihood of such incidents was low and that, in the event of snagging, the fuel hoist overload limit would prevent the application of any significant load increase. However, following further discussion, it is now acknowledged that there is some uncertainty over the validity of these assumptions. Further work will therefore be undertaken to quantify the frequency of snagging incidents during re-fuelling, determine the maximum axial force that can be applied with the current

overload limit setting and to assess the implications of any findings for the graphite safety case.

The final retained option was associated with the potential to enhance understanding of graphite behaviour by removing and testing whole bricks or large samples from Reactor 2 or testing bricks in situ in Reactor 2 after it has ceased generation at the end of June 2011. It was recognised that such activities would be unlikely to make a worthwhile contribution to the safety case for extended generation of Reactor 1 and subsequent consideration of this option (Reference 35) by the Graphite Technical Issues Group (GTIG) and the Graphite Technical Forum has concluded that such activities would not be reasonably practicable. The option will not therefore be pursued further in the context in which it was raised at the workshop. (It should, however, be noted that the feasibility of removing larger samples from Reactor 2 post shut-down, for flexural strength testing, is being investigated; see Section 13.2).

REVISION RECORD

N°	Date	Author	Reason
-	May 2011	[redacted]	First Issue