NUCLEAR SAFETY COMMITTEE

NP/SC 4807 Addendum 6
Wylfa Power Station

TITLE - **Safety Case for the Integrity of the Graphite Cores: A Probabilistic Assessment**

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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: **Wylfa Power Station: Safety Case for the Integrity of the Graphite Cores: A Probabilistic Assessment**

TARGET NSC SUBMISSION DATE: July 2010

AUTHOR

Name [redacted]

Signature [redacted] Date 5th July 2010

VERIFIER

I confirm that this submission has been verified in accordance with MCP/021/001 and is fit for purpose. The verification plan is provided as Appendix A and an auditable record of the verification process is retained in EWST Task file RG7428. A verification statement is given overleaf.

Name [redacted]

Signature [redacted] Date 05.07.2010.

SAFETY CASE OFFICER

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

Name [redacted]

Signature [redacted] Date 5/07/10

ENDORSED FOR ISSUE BY

Name [redacted]

Signature [redacted] Date 6/7/10

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 the verification plan reproduced in Appendix A and is fit for purpose. An auditable record has been made of the verification process. These records and the completed verification plan will be retained in EWST Task File No RG7428

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

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

[redacted] Date: 05. 07. 2010

- The descriptions of the plant and operational arrangements are correct.

[redacted] Date: 03.07.10.

SUMMARY

The current safety case for the Wylfa graphite cores is set out in NP/SC 4807 and its Addenda 1 through 5 and covers operation up to September 2014. The safety case is based on four legs: inspection and sampling, monitoring, structural integrity assessment and consequences of failure analysis. It was noted in Addendum 5 that a probabilistic assessment of the risk of clad melt due to graphite brick cracking was also being undertaken, following the methodology developed for Oldbury, to provide further support to the safety case. The probabilistic assessment has now been completed and this Addendum presents the results obtained and shows how they strengthen the existing safety case.

Key input parameters and modelling assumptions are summarised, highlighting (as appropriate) differences from the Wylfa deterministic assessment and the Oldbury probabilistic assessment. The results indicate that the nuclear safety risks associated with graphite brick crack initiation, crack progression, brick splitting and consequential coolant gas flow diversion are well within the Broadly Acceptable region for operation up to September 2014. This conclusion is not sensitive to changes in the input parameters.

The opportunity is also taken to consider the implications of recent developments emerging from the ongoing programme of work in support of the safety cases for the Wylfa and Oldbury reactor graphite cores.

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

1. NOTE that the results of a probabilistic assessment confirm the risks associated with graphite brick crack initiation, crack progression, brick splitting and consequential coolant gas flow diversion to be Broadly Acceptable, thereby providing additional support to the safety case reviewed and summarised in Addendum 5.
2. NOTE the proposal to present a further Addendum to the NSC prior to the end of 2010.

CONTENTS

1 INTRODUCTION

The safety case for continued operation of the graphite reactor cores at Wylfa Power Station was most recently reviewed in NP/SC 4807 Addendum 5 (Reference 1) which was presented to the December 2009 Nuclear Safety Committee (NSC). The safety case is based on a demonstration that the geometry of the control rod channels is maintained such that there is no impediment to rod insertion during normal operation or faults, and a demonstration that core flow geometry is maintained sufficient to ensure adequate fuel cooling. The demonstration of these key functions is based on four main safety case elements; inspection & sampling, monitoring, structural assessment and consequences. The 2009 review considered the safety case implications, for these four elements, of recent inspection, sampling and monitoring activities as well as developments in assessment methodologies and data. The review closely followed the format of previous submissions but was extended to provide a safety case for operation of the graphite cores to September 2014.

Results obtained from inspection and monitoring activities completed since the previous review were discussed; it was shown that the results from the NOREBORE programme and TV inspections were consistent with expectations although some difficulties were initially experienced with the re-inspection of a previously reported defect in Reactor 1 channel 0315/10. Nevertheless the defect was subsequently identified and an audit of the inspection quality arrangements confirmed them to be satisfactory albeit with some minor issues to be addressed prior to the 2010 inspections. A statement was made under Matters Arising at the March 2010 NSC (Reference 2) reporting the resolution of these issues; although the camera system met the requirements of the safety case, a number of potential improvements were made to the system, and its operation and maintenance, prior to its deployment during the 2010 statutory outage.

The most recently available weight loss measurements obtained from trepanned samples were presented and, with the exception of those obtained for a single channel (1223/15) in Reactor 1, which showed anomalously high weight losses at layers 4 and 7, the results obtained were within expectations. Anomalously high weight losses associated with channel 1923/15 on Reactor 1 had been reported in previous reviews but it was shown that strength utilisation factors (UFs) were not substantially affected and that the safety case remained secure. This position was reviewed in the light of the 2009 measurements and it was confirmed (Reference 1) that the safety case remained robust against such anomalies, even if they occurred at locations of peak hoop stress.

Other monitoring activities undertaken since the 2008 review, including refuelling, control rod movement checks, CGOT monitoring and operation of the BCD system, did not identify any challenge to the maintenance of the key safety functions of the graphite core.

The structural integrity assessment methodology and the calculation of UFs were unchanged from the 2008 review and no further assessments were undertaken. However it was noted that emergent data indicated improved flexural strength values from installed sets withdrawn in 2008. The integrity assessment also considered the implications of thermal loads arising during refuelling activities and argued that the associated transient is benign compared to a shutdown transient and that the effect on core risk is therefore low.

Thus the 2009 review identified no substantive issues which presented a significant challenge to the safety case, and it was concluded that the graphite cores of both reactors remain acceptable for continued operation up to Mean Core Irradiations (MCIs) of 34.1GWd/t for

Reactor 1 and 32.4GWd/t for Reactor 2, these being unlikely to be exceeded prior to the end of September 2014.

It was noted that a probabilistic assessment of the risk of clad melt due to graphite brick cracking was being undertaken, following the methodology developed for Oldbury, to provide further support to the overall safety case. The probabilistic assessment was developed, initially for Oldbury, to address concerns that the deterministic assessment might not be sufficiently conservative to provide a robust safety case alone. The methodology and results for Oldbury were presented to the December 2008 NSC (Reference 3).

A summary of progress with the principal elements that make up the Wylfa probabilistic assessment was presented to the December 2009 NSC (Reference 1), but insufficient progress had been made at that time to provide any indication of the anticipated outcome. The Wylfa probabilistic assessment has now been completed (Reference 4) and it has been possible to demonstrate very low clad melt probabilities. The purpose of this submission is to present the results obtained and to show how they underpin the safety case reviewed in NP/SC 4807 Addendum 5 (Reference 1).

2 PLANT DESCRIPTION

2.1 Wylfa Graphite Cores

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 reflector and the top and bottom reflectors are made from Pile Grade B (PGB) graphite, while the more central bricks are made from Pile Grade A (PGA) graphite. The core is made up of 13 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 are short and do not span the interfaces between vertically adjacent bricks.

The brick columns comprising the active core are centrally bored to form 6156 vertical fuel channels per reactor, each being 98mm diameter. There are 8 fuel elements stacked in each fuel channel; they extend approximately from Layer 2 to Layer 12 of the active core. The safety case concentrates on these core layers.

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.

2.2 Differences from Oldbury

There are some key differences between the Wylfa and Oldbury graphite cores that have a bearing on the outcome of the probabilistic assessment to be reported below. The most notable differences are:-

- *Core Size*: The Wylfa cores are much larger than those of Oldbury which have only 3308 fuel channels per reactor and 10 moderator brick layers per column. Hence, more bricks contribute to the overall risk of clad melt at Wylfa.
- *Brick Size*: The fuel channel pitch is the same at both sites. However, the square bricks at Wylfa are smaller than those at Oldbury and, to maintain the same pitch, the reverse is true for octagonal bricks. The magnitude of thermal stresses developed within bricks is influenced by brick geometry and thickness and hence the relative likelihood of crack initiation and progression within square and octagonal bricks is different for Wylfa.
- *Keying Arrangements*: At Wylfa, the keyways and inter-brick graphite keys are confined to a short length at the top of each brick and do not span horizontal layer interfaces. This arrangement results in a very open core from the flow perspective. Once coolant has escaped from the fuel channel there would be little flow resistance and therefore all bricks in the Wylfa cores will contribute to the clad melt risk, irrespective of whether they have interstitial cut-outs or not. This contrasts with the keying arrangements at Oldbury, where the keyways and inter-brick graphite keys are continuous over the full length of the fuel channel and only those bricks where cracking could discharge by-pass flow into an interstitial channel contribute significantly to the overall clad melt risk.
- *Spigots*: At Wylfa, the spigots at the upper end of each brick are substantial and have very small diametrical clearances (0.7mm) to the counter-bore machined in the brick above. The graphite bricks at Oldbury also have spigots but these are less substantial than the spigots at Wylfa and a relatively large spigot to counter-bore clearance (17.5mm) exists. The more substantial spigotted joints at Wylfa have the potential to prevent crack face separation at the upper end of a doubly cracked brick.
- *Graphite Material Properties*: Weight loss in the graphite cores affects a number of material properties that collectively determine stress and strength within the moderator bricks. It has been observed that the predicted end of generation (EoG[1]) fuel channel wall (FCW) weight losses and the rate of weight loss with irradiation dose are lower at Wylfa; peak predicted weight losses at Wylfa are generally about half of those predicted for Oldbury. Hence significant differences in the values and trends of material properties relevant to the prediction of stress and strength are to be expected.

3 PROBABILISTIC ASSESSMENT APPROACH

3.1 Scope and Objective

The safety case for the Wylfa cores, developed through NP/SC 4807 and its addenda, has demonstrated that the likelihood of axial brick splitting, with the potential to lead to subsequent fuel overheating, is dominated by the risk of crack initiation at the FCW during the temperature transient following a reactor trip. It has been concluded that the associated nuclear safety risks lie in the Broadly Acceptable region as defined in the Safety Review Guidebook (SRG, Reference 5). The probabilistic assessment has therefore similarly been focussed on evaluating the likelihood of clad melt as a result of axial splitting of the graphite bricks, initiated by cracking

[1] EoG is taken to be September 2014 throughout this submission.

at the FCW during a shutdown transient (SDT). For completeness, the assessment has also addressed the likelihood of cracking during normal operation of the reactors.

The overall objective of the probabilistic assessment, reported in this current submission, is therefore to provide a quantitative estimate of the risks associated with graphite brick cracking. This estimate has been provided in accordance with Nuclear Safety Principle (NSP) 3 from the SRG, whilst retaining pessimisms within the assessment.

3.2 Method

The probabilistic assessment, including methodology, assumptions and results, is reported in Reference 4 and summarised in Appendix B of this current submission. As for Oldbury, the Wylfa assessment employs a 'three stage' methodology as follows:-

<u>Stage 1: Calculation of the probability of crack initiation</u>

An estimate is made of the probability of axial crack initiation at the FCW; that is the probability that the FCW peak hoop stress exceeds the perpendicular strength during a shut-down transient. In evaluating the probability of crack initiation, account is taken of the uncertainties in those parameters that determine the loading and strength of the graphite bricks such as: weight loss, Young's modulus and coefficient of thermal expansion (CTE).

<u>Stage 2: Calculation of the conditional probability of crack progression and brick splitting</u>

A conservative estimate is then made of the conditional probability of crack progression and ultimately of the brick splitting. Two alternative models are employed. The "*crack propagation model*" assigns probabilities to the crack being diverted to become a circumferential crack, being blocked by an existing circumferential crack, being arrested or continuing to grow axially and finally progressing to a split brick; this model relies on an expert elicitation approach. In the "*double initiation model*" it is assumed that, if an axial crack initiates, it will immediately progress to a full length through-wall crack. This full length crack causes an increase in the peak hoop stress at the opposite side of the FCW which may result in the initiation of a second crack. The probability of initiation of this second crack is calculated using the same method as for the first crack, taking into account redistribution of the stresses in the brick. As for the first crack, it is assumed that this second crack grows through the brick with a probability of unity.

<u>Stage 3: Calculation of the conditional probability of fuel clad melt</u>

The third and final stage is to determine the conditional probability of fuel clad melting in the event of axial brick splitting for various brick geometries and elevations, using estimated crack gapes and coolant gas flow analysis. The melt probabilities for cracks at various locations in the core are calculated using the PANTHER/PREDICT2 assessment route.

It is assumed that crack initiation and progression (stages 1 and 2 above) occur during the SDT. However, as the reactor is being shut down, melting of the fuel cladding would not occur immediately; if fuel clad melting were to occur, it would happen when the reactor was returned to power following the shutdown. Thus, as for Oldbury, the relevant condition for considering the conditional probability of clad melt (given brick splitting) is the normal operating condition.

The brick clad melt probabilities derived as outlined above, are combined for all bricks to obtain the probability that fuel clad melting will occur somewhere within the FZ under the specified operating condition. Finally, from the FZ clad melt probability, an estimate is made that fuel clad melt will occur somewhere within the entire core as a result of graphite brick cracking.

3.3 Input Data and Modelling Assumptions

For the probabilistic assessment, the data are expressed in the form of probability distributions, rather than discrete values as in the deterministic assessment. It is also important to note that some of the detailed modelling assumptions in the Wylfa probabilistic assessment differ from those employed in the Oldbury probability assessment; this is largely a direct result of the plant differences highlighted in Section 2.2 above. The following is a summary of the key input data and modelling assumptions made in the Wylfa probabilistic assessment, highlighting differences from the Wylfa deterministic and Oldbury probabilistic assessments as appropriate.

3.3.1 Mean Core Irradiation Predictions

The 2008 graphite review (Reference 6) introduced revised forecasts of EoG MCI for generation up to the end of September 2014. Values of 34.1GWd/t and 32.4GWd/t were forecast for Reactors 1 and 2 respectively; these values were confirmed in the 2009 review (Reference 1) and have been used to derive crack initiation and progression probabilities and also in the derivation of crack gapes in the current assessment. The most recent MCI predictions (Reference 7) are slightly lower at 33.8GWd/t and 32.0GWd/t respectively; the assessment is therefore conservative in this respect. However, for the conditional clad melt probability calculations (Reference 8), a representative MCI of 33GWd/t has been assumed. Nevertheless, it is noted in Reference 8 that changes to core state over time have only a modest effect on graphite risk of clad melt studies; as the core state becomes more onerous with respect to risk, the reactor operating temperature constraints are already adjusted to compensate, maintaining an acceptable level of risk, for varying core states, in the event of the limiting symmetric reactor fault. It is therefore judged that the outcome of the probabilistic assessment is not significantly affected by this assumption.

3.3.2 Shut Down Transients

As for Oldbury, two SDTs of different severities are considered; a "pessimised transient", which has been selected to bound any likely future trip, and a "measured transient" based on a typical, hard reactor trip. The SDTs have been conservatively derived via a formal review of the key parameters affecting the SDT model (Reference 9) and a revised PANTHER model of the pessimised transient was developed (Reference 10) such that it bounded the severity of any reactor trip that is expected to occur in the remaining years of operation. This new pessimised transient (Reference 10) is similar to, but slightly less onerous than, the previous pessimised transient (Reference 11) which was used in the deterministic assessment reported in NP/SC 4807 Addendum 4 (Reference 6) and reviewed in Addendum 5 (Reference 1); the assumed channel gas outlet temperature (CGOT) reduction in the 5 minutes following a trip is within 10°C of that previously assumed.

The review also proposed that the trip which occurred on Reactor 2 on 13th May 2009 should be used to define the measured transient (Reference 9); this transient is slightly more onerous than the measured transient previously used in the deterministic assessment.

(It is intended to use these updated bounding SDTs for a revised deterministic assessment to be presented, by the end of 2010, in Addendum 7.)

3.3.3 Predicted Weight Losses

In NP/SC 4807 Addendum 4 (Reference 6) it was judged appropriate to base weight loss/dose predictions on the BEST model rather than the layer-by-layer statistical fitting method that has been used in recent Oldbury assessments. In Addendum 5 (Reference 1), it was judged appropriate to continue to employ the BEST model and it was reported that the Reactor 2 2008 trepanned weight loss data had been incorporated into a revised weight loss prediction; inclusion of these data resulted in little change to the weight loss forecasts. The weight loss probability distributions within the probabilistic assessment are consistent with these most recent weight loss predictions from Addendum 5.

Addendum 5 (Reference 1) also discussed the significance of the anomalously high weight losses that had been observed in a small number of Reactor 1 bricks (Section 1 above). The probabilistic assessment has included consideration of the implications of higher weight loss bricks (see Section 4.2 below).

3.3.4 Strength Model

The relationship between graphite strength and weight loss is established by statistical fitting of flexural strength measurements made on installed set samples. It was reported in Addendum 5 that flexural strength data obtained from installed sets, withdrawn from Reactor 2 during the 2008 statutory outage, indicated consistently higher values than those used in the deterministic assessment. The strength model has now been updated to include these data (Reference 12) and the revised relationship, shown in Figure 2, has been employed in the probabilistic assessment. Recently obtained flexural strength data, from installed sets withdrawn from Wylfa Reactor 1 during the 2009 statutory outage, have been shown to mostly lie above the revised relationship shown in Figure 2 (Reference 13), thereby indicating the conservative nature of the revised strength relationship.

3.3.5 Coefficient of Thermal Expansion

CTE is a primary parameter in determining SDT stresses and, in order to account for offsets between measured and predicted values of CTE, correction factors of 1.22 and 1.44 have previously been applied, to perpendicular and parallel CTE values respectively. A review of structural integrity parameters (Reference 14), carried out in support of the Wylfa probabilistic assessment, has taken account of the most recent Wylfa-specific measurements and recommended revised correction factors of 1.15 and 1.22 respectively. It is noted that predicted thermal hoop stresses are influenced by perpendicular CTE and the revised correction factors will have resulted in a modest decrease in these stress predictions.

3.3.6 Dynamic Young's Modulus

Revised correction factors including a new dose-dependent term have been derived (Reference 14) to account for offsets between Wylfa-specific measurements and predicted values of both perpendicular and parallel Dynamic Young's Modulus (DYM). The effect has been to increase the predicted perpendicular FCW DYM in most of the core layers which will have resulted in an increase in predicted displacement controlled stresses within the Wylfa assessment.

3.3.7 Crack Gape

In the event of double axial cracking of a brick, the pressure differential between the fuel channel and the Wigner gap could cause separation of the crack faces. However, providing the spigot/counterbore interface remains intact, the narrow diametral clearance of this interface would constrain the separation at the top of the brick to be essentially zero (Reference 4). The two brick sections would then pivot about the outside of the spigot and form a crack gape of triangular cross section with maximum separation at the base of the brick (Figure 3a). It has been demonstrated by testing (Reference 15) that the spigot/counter-bore arrangement at the top of the brick would remain intact under the maximum applied load during the SDT; a reserve factor to failure of more than 3 has been indicated[2], thereby justifying the assumption of triangular cracks in the analysis.

Movement under the separation moment would be resisted by the self weight of the split brick sections and the weight of the column of bricks above and, if the gapes were to become large enough, through engagement with adjacent moderator bricks. An equilibrium position would eventually be reached where the (decreasing) separation moment would be balanced by the resistive moments and the crack faces would move no further apart. The limiting crack gapes at which this equilibrium position is reached for both octagonal and square bricks in core layers 2-12 are evaluated in Reference 16, conservatively neglecting the effects of friction between graphite core components.

It is recognised (Reference 17) that a doubly cracked brick would cause the spigot to impart opposing radial loads against the counter-bore of the brick above. In order to assess whether this loading could result in axial cracking of the brick above, a finite element assessment of the scenario has been conducted (Reference 17). The results of this work have shown that, although the peak hoop stress would increase by up to ~3%, the increase is not large enough to initiate axial cracking; there is therefore no significant risk of cascade failure as a result of double axial cracking at any core layer. This possibility has therefore been discounted in the probabilistic assessment.

3.3.8 Coolant By-pass Flow

Coolant by-pass flow in the presence of a double-cracked brick has been evaluated (Reference 18) using a flow network approach similar to the corresponding Oldbury assessment (Reference 3). However, whilst maintaining a conservative approach, this assessment has introduced refinements to the evaluation of hydraulic losses at the crack-to-fuel-channel junction, resulting in a significant reduction in the predicted leakage through the crack.

In addition to crack gape, which is discussed in Section 3.3.7 above, predicted leakage rates and hence channel flow impairments are also dependent on the radial crack path. The by-pass flow calculations assumed a radial crack length of ~30mm for all bricks, corresponding to the shortest possible crack through a square brick in the cardinal orientation; this represents a conservative approach, particularly for the thicker octagonal bricks. [Note however that, for the purposes of deriving crack separation forces, and hence gape (Section 3.3.7), this simplification was not adopted since it would have been non-conservative.]

4 PROBABILISTIC ASSESSMENT RESULTS

4.1 Base Cases

Reference 4 presents the results of the probabilistic assessment for each of the following "base cases" for each reactor:-

1. Full power normal operation loading condition + crack propagation model
2. Full power normal operation loading condition + double initiation model
3. Pessimised shut-down transient loading condition + crack propagation model

[2] The assessment presented in Reference 15 is essentially deterministic and does not include uncertainty in strength or weight loss variability. For completeness, an evaluation of spigot/counter-bore failure probability, taking account of strength variability, has been carried out (Reference 4) which supports the assumption that failure may be discounted for the purposes of the probabilistic assessment.

4. Measured shutdown transient loading condition + double initiation model

These cases are the same as those presented for Oldbury (Reference 3) and generally assume "best estimate" probability distributions for input parameters but include a number of pessimisms within the methodology - see Appendix B. The results are summarised in Appendix B where it may be seen that, for the most onerous cases (3 and 4), the predicted whole core probability of single channel clad melt is $\leq 10^{-7}$(per hard trip). It may also be seen (Appendix B) that the risks associated with cracking under full power operating conditions (cases 1 and 2) are negligibly small and that, for all cases, the risks for Reactor 2 are bounded by those for Reactor 1.

The following sub-sections consider the various contributions to these base case estimates of whole core clad melt probability and discuss the implications of the results from each of the analysis stages. The discussion is focussed on the bounding cases (3 and 4) for Reactor 1.

4.1.1 Likelihood of Crack Initiation

Crack initiation probabilities are presented in Reference 4 and summarised in Table 1 of this current submission. The results indicate that brick layers 7, 8 and 9 exhibit the highest crack initiation probabilities for all brick geometries. This layer range lies slightly above the mid-height of the core and the dominance of these layers is broadly the result of the predicted weight loss reaching a peak around layers 5-7, together with the higher thermal hoop stresses during the SDT experienced by the upper core layers.

From Table 1 it may also be seen that, for octagonal bricks, the peak crack initiation probabilities at the critical layers 7-9 are approximately 3 orders of magnitude higher than the corresponding values for square bricks. This significantly higher probability of crack initiation for octagonal bricks is a consequence of octagonal bricks being subject to significantly higher thermal hoop stresses during the SDT as a result of their different geometry. This is consistent with the results of previous deterministic assessments which have indicated significantly higher UFs for octagonal bricks than for square bricks; for example, the peak FCW hoop UFs for Reactor 1 at EoG were predicted to be 0.77 for octagonal bricks and 0.53 for square bricks (Reference 19).

4.1.2 Crack Progression and Brick Splitting

As indicated in Section 3.2 above, two different approaches to the modelling of crack progression and brick splitting have been adopted. For both approaches, the probability of brick splitting is derived from the product of the crack initiation and crack progression probabilities.

For the crack propagation model, the probability that crack initiation progresses to a split brick is a judgemental, fixed value. Based on examination of the predicted strain fields within peak rated bricks, expert elicitation (Reference 20) has derived conditional probabilities of 2.7×10^{-3} and 1.7×10^{-3} for square and octagonal bricks respectively; a bounding conditional crack progression probability of 10^{-2} was proposed and used in the assessment for all brick geometries[3].

In the double initiation model, it is assumed that, given the first initiation, the crack will propagate immediately through one side and along the length of the brick with a probability of unity. Following this, the probability of a second crack initiation in the opposite side of the brick, under the re-distributed stress field, is estimated. As for the first crack, progression of the

[3]The equivalent probability derived for Oldbury bricks is 10^{-3} and this was used in the probabilistic assessment for all brick geometries at Oldbury (Reference 3). The lower probability for Oldbury is a result of the different brick geometry, particularly wall thickness.

second crack is then assumed to occur, during the same transient, with a probability of unity. It is evident that if the probability of crack initiation is relatively high, the double initiation model will be more conservative than the crack propagation model, whilst for lower crack initiation probabilities, this model is less conservative.

4.1.3 Coolant Flow Diversion and Fuel Clad Melt

Given double axial cracking of a brick, the conditional clad melt probability due to the channel flow reduction above the crack depends largely on the crack gape, the resistance to by-pass coolant flow and the brick layer in which the failure occurs (Reference 4).

Table 2 presents the limiting values of crack gape, calculated for both square and octagonal bricks, for each of the core layers. For the square bricks, the split brick sections are predicted to separate until they engage with the bricks in the adjacent columns (Figure 3b) and crack gapes are therefore limited by the Wigner gaps at EoG (Reference 16); a maximum gape value of 5.9mm, at the base of the brick, occurs at core layer 6. For octagonal bricks, restraint is provided via the corner keys which react the load[4] through the adjacent column of bricks (Figure 3c) and the corresponding crack gapes are predicted to be lower for all layers (Reference 16); a maximum gape of 4.2mm occurs at layer 3.

Channel coolant flow impairment and consequential risk of clad melt are evaluated in Reference 18 for a range of postulated crack gapes and brick locations. The predicted flow impairments associated with the limiting crack gapes at each core layer are also presented in Table 2. For square bricks, a maximum flow impairment of 49% is predicted for a crack at core layer 6 corresponding to the location of the widest gape. As the predicted gapes are smaller for octagonal bricks then, as expected, flow impairments are lower than for square bricks. Conservatively assuming the shortest radial crack path (i.e. 30mm associated with a square brick – see Section 3.3.8 above), a maximum flow impairment of 40% is predicted for a crack in an octagonal brick at layer 3 (Table 2).

The conditional fuel clad melt probabilities, given a split brick, have been calculated for all brick elevations and crack gapes up to 10mm (Reference 18). Using the predicted limiting values of gape (Reference 16) and flow impairment (Reference 18), the conditional clad melt probabilities at each core layer have been calculated (Reference 4); these probabilities are also presented in Table 2. As would be expected, the conditional clad melt probabilities are consistently lower at all layers for the octagonal bricks. The maximum conditional clad melt probability predicted is 3×10^{-3} for square bricks in core layer 5.

4.1.4 Whole Core Clad Melt Probability

For each brick geometry and layer, the probability of clad melt has been derived by combining the probabilities of crack initiation, crack progression and brick splitting with the conditional probability of clad melt in the event of brick splitting. The estimates of single brick clad melt probability have then been combined over all core layers and fuel channels to provide an estimate of whole core clad melt probability. The predicted whole core probabilities of single channel clad melt are $\leq 10^{-7}$ for all base cases. However, the assessments include a number of pessimistic or bounding assumptions, such as:-

- Assuming a bounding conditional crack progression probability of 10^{-2} for the crack propagation model (Section 4.1.2).

[4] By considering the potential stresses in the key it is argued in Reference 16 that the corner key will remain intact under the predicted loading.

- Neglecting the effects of friction between graphite core components and hence over-predicting crack gapes (Section 3.3.7).
- Assuming a radial crack path of only ~30mm and hence over-predicting gas leakage flows, particularly for octagonal bricks (Section 3.3.8).
- Assuming a pessimistically high peak CGOT of 421°C when calculating conditional clad melt risks (Reference 4).
- Assuming that the risks associated with cracking of bricks in the un-flattened zone are the same as those for the corresponding bricks in the FZ (Reference 4).

As a result of the pessimistic assumptions, it is considered that the predicted fuel clad melt probabilities could be conservative by up to several orders of magnitude (Reference 4).

4.1.5 Conclusions

The following conclusions and observations may be made from the base case probabilistic assessment:-

- The probability of crack initiation, and hence the probability of brick splitting, is significantly higher for octagonal bricks than for square bricks; this is because the thermal stresses induced in the octagonal bricks are significantly higher than those induced in square bricks.
- For all brick geometries, the probability of crack initiation is greatest at layers 6–9; this is broadly the result of predicted weight losses which reach a peak around layers 5-7 and the magnitude of thermal stresses which increase with brick height.
- In the event of brick splitting, the probability of clad melt is significantly lower for octagonal bricks than for square bricks; this is largely due to the smaller crack gapes predicted for octagonal bricks. Octagonal brick gapes are limited as a result of the restraint provided, through corner keys, by the adjacent columns of bricks; for square bricks, gapes are generally limited only by the Wigner gap at EoG.
- For all brick geometries, the conditional probability of clad melt, in the event of brick splitting, is highest for the lower layers 3–6; for these layers, the flow impairments are slightly greater and affect more fuel elements.
- Moderator layers 6–9 make the highest contribution to the overall whole core risk of fuel clad melt.

4.2 Revised Base Cases (Inclusion of Higher Weight Loss Bricks)

The 2007 graphite safety case review (Reference 21) reported the observation of anomalously high weight losses in Reactor 1 channel 1923/15, bricks 6 and 8. The 2009 review (Reference 1) reported a further instance of high weight loss observed in Reactor 1 channel 1223/15, bricks 4 and 7. It was noted that the most safety significant location was the FCW and the results for channel 1223/15 at bricks 4 and 7 showed respective weight losses of 1.44 and 1.76 times previous predictions. The corresponding factors for channel 1923/15 bricks 6 and 8 were 1.41 and 1.72 respectively. Despite comprehensive investigations, no firm conclusion had been reached on the cause of the observed high weight losses; the most plausible explanation was considered to be poor impregnation of these particular bricks, resulting in a low initial density, but this remained unproven.

Scaling calculations were performed to determine the potential effect on predicted UFs and it was confirmed that the UFs were not unduly sensitive to anomalously high weight losses, irrespective of core location (Reference 1).

Nevertheless, it was considered appropriate to revise the base cases, discussed in Section 4.1 above, by including an allowance for these high weight loss bricks within the probabilistic assessment. This has been achieved by including a second population of higher weight loss bricks within the assessment. For simplicity, and commensurate with the scarcity of high weight loss data, the mean weight loss of the higher weight loss population is taken as a constant multiplier of that for the "normal" weight loss population. The multiplier has been assigned a value of 1.6 which bounds the mean of the individual anomalies given above. The weight loss probability distributions for the high weight loss population are assumed normal with identical standard deviations to those of the lower weight loss distributions.

Figure 4 shows how the predicted whole core clad melt probability varies with the proportion of bricks in the core that are assumed to exhibit anomalously high weight loss. The observation of such high weight loss in only 4 bricks represents less than 4% of the bricks from which samples have been trepanned. An "adjusted base case" has therefore been defined in which the higher weight loss population contains 10% of the bricks in the core; this is judged to provide a conservative probabilistic assessment, adequately encompassing the observations of anomalously high weight loss bricks. It can be seen from Figure 4 that this results in a predicted whole core clad melt probability of $\underline{\leq 3 \times 10^{-6}}$ per hard trip. The sensitivity of this result to variations in the key parameters used in the probabilistic assessment is considered in Section 4.3 below.

4.3 Sensitivity Studies

Allowance for random uncertainty in input parameters is incorporated into the assessments. To quantify the influence of potential future *systematic* input parameter adjustments (arising as a result of new data or understanding, for example), the sensitivity of the adjusted base case whole core clad melt probability to weight loss, peak stress, strength and crack gape has been investigated (Reference 4). The results are illustrated in Figures 4 through 7 and the following observations may be made:-

- If the mean weight loss of the higher weight loss population is increased by 10% (i.e. increasing the multiplier from 1.6 to 1.76), the maximum predicted whole core probability of single channel clad melt would increase to ~4 x 10^{-5} per hard trip (Figure 4). A similar result would be obtained if the anomalously high weight loss population (with a multiplier of 1.6) were to be extended to 100% of the core.
- Increasing the peak stress in all bricks by 20% increases the maximum predicted whole core probability of single channel clad melt to ~4 x 10^{-5} per hard trip (Figure 5).
- Decreasing the strength in all bricks by 20% increases the maximum predicted whole core probability of single channel clad melt to ~10^{-4} per hard trip (Figure 6).
- Increasing the crack gape for all bricks by 30% increases the maximum predicted whole core probability of single channel clad melt to ~3 x 10^{-5} per hard trip (Figure 7). A similar result would be obtained if the crack gape for all bricks were to be increased by 1.3mm (Reference 4, Figure 10b).

The core states considered in the base case assessment are representative of those anticipated to be reached at EoG in September 2014. To understand how the overall whole core clad melt probabilities may be changing with time, the sensitivity of the (base case) whole core clad melt probability to increasing MCI levels has been evaluated (Reference 4). It is shown in Reference 4 that the predicted clad melt probability increases only slowly up to and beyond the EoG MCI and that the probability of clad melt remains very low over this range. Increasing the MCI from 34.1GWd/t to 35GWd/t increases the maximum whole core probability of a single channel melt by about a factor of 2. It is judged that a similar increase in whole core probability of a single channel clad melt would apply to the revised and adjusted base case assessments discussed in Section 4.2 above.

4.4 Comparison with Oldbury

For the adjusted base case, the maximum whole core single channel clad melt probability of $<3 \times 10^{-6}$ per hard trip (Section 4.2 above) is significantly lower than the equivalent value of $<2 \times 10^{-2}$ previously evaluated for Oldbury and reported in Reference 3.

There are a number of features of the graphite cores and associated analyses which have collectively resulted in these different outcomes for the two probabilistic assessments. Plant and assessment differences between Oldbury and Wylfa which have a significant beneficial effect on the predicted clad melt probability for Wylfa include:-

- Despite having similar MCIs, the prevailing weight losses associated with bricks in the Wylfa graphite cores are approximately half those associated with the Oldbury cores.
- In the event of brick splitting at Wylfa, crack gapes are constrained by the substantial spigotted joints which prevent crack face separation at the upper end of the brick. This limits the extent of coolant bypass flow, reducing the likelihood of clad melt.
- Refinements have been introduced to the bypass flow modelling approach previously employed for the Oldbury assessment whilst maintaining significant overall conservatisms. These refinements have taken full account of the hydraulic losses at the crack-to-fuel-channel junction and have resulted in lower predicted by-pass flows and higher channel outlet flows, significantly reducing the calculated risk of clad melt.

Plant differences which have a significant detrimental effect on the predicted clad melt probability for Wylfa include:-

- The cores at Wylfa are much larger than those of Oldbury and hence more bricks contribute to the overall risk of clad melt.
- The keying arrangements at Wylfa result in a more open core with little flow resistance and hence all bricks contribute to the clad melt risks, whilst, at Oldbury, only those bricks where cracking could result in by-pass flow into an interstitial channel contribute.

The net effect of all the plant and assessment differences is a significant reduction in the predicted probability of clad melt at Wylfa compared with Oldbury. Note, however, that application of the refined flow modelling approach would also be expected to result in a significant reduction in predicted clad melt risk at Oldbury (Reference 4).

5 RADIOLOGICAL CONSEQUENCES OF A SINGLE CHANNEL CLAD MELT

It was noted in NP/SC 4807 Addendum 5 (Reference 1) that the assessed public dose due to a single channel clad melt, 0.98mSv, was close to the borderline between the 0.1–1mSv and 1-10mSv dose bands defined in NSP3 of the SRG (Reference 5). It was reported that a review, conducted into the underlying assumptions and methodology of the assessment, had identified a variety of factors and standards which could alter contributions to the assessment. It was concluded, on balance, that the assessed dose would remain in the 0.1–1mSv band. Nevertheless, in view of the potential significance of the issue, a commitment was given to re-evaluate the public dose, taking into account all relevant methods and criteria, and to evaluate the probabilistic assessment results against the revised dose predictions.

The revised dose assessment has now been completed, in line with current methods and data, and the outcome is reported in Reference 22. The offsite dose to the critical individual has been evaluated as 0.44mSv which is comfortably within the 0.1-1mSv dose band. This value is used in the risk assessment in Section 6 below.

The potential for a single channel clad melt fault to spread to adjacent fuel channels has been considered previously in NP/SC 4807 Addendum 4 (Reference 6); the assessment was based on the assumption that a single channel fire escalates to the eight surrounding channels and that the dose increases accordingly into the 1–10mSv dose band. Scaling the single channel value of 0.44mSv (see above) suggests an offsite dose of 4mSv for the escalated fault. Protection against such escalation is provided by operator action, within 5.5 minutes, in response to the Burst Cartridge Detection (BCD) system alarms. Significant improvements have been implemented to increase the reliability of operator response as reported in NP/SC 4807 Addenda 4 and 5 (References 6 and 1). It was noted in Addendum 5 that a retrospective specialist review of the final outcome of the improvement would be undertaken to confirm that the Human Reliability Analysis (HRA) assessed value of 1.3×10^{-2} failures per demand had been achieved and to consider the continuing applicability of the "unlearning" factor of 1.5 included in this HRA figure. This review has now been completed and the outcome is reported in Reference 23; an improved reliability figure for operator response of 8.4×10^{-3} failures per demand is estimated (under conditions of full BCD system availability). Thus the conditional probability of escalation of a single channel clad melt may be conservatively taken as 10^{-2}.

6 RISK ASSESSMENT

The adjusted base case results discussed in Section 4.2 above have indicated conservative whole core single channel clad melt probabilities of $<3 \times 10^{-6}$ per hard trip. As noted in Section 3.3.2 above, the assessments modelled two transients of different severities. The "pessimised" transient was selected to be sufficiently severe to bound any future reactor trip, whilst the "measured" transient was chosen to be representative of a more typical hard trip. It is considered conservative to assume that one hard trip occurs per reactor each year; the whole core single channel clad melt probability may then be taken as $<3 \times 10^{-6}$ per reactor year (pry). Based on this value, together with the associated radiological consequences and potential escalation discussed in Section 5 above, the nuclear safety risks are illustrated, for this single class of faults, in the staircase diagram of Figure 8. It may be seen that the risks are well within the Broadly Acceptable region. From the results of sensitivity studies, summarised in Section 4.3 above, it is also evident that this conclusion is not sensitive to changes in the assessment's input parameters.

7 SAFETY CASE IMPLICATIONS

NP/SC 4807 Addendum 5 (Reference 1) considered that, taken together, the four legs of the graphite integrity safety case (inspection and sampling, monitoring, structural assessment and consequences) demonstrated that the risk of crack initiation, crack propagation, brick splitting and consequential clad melt was sufficiently low that the radiological risks associated with continued operation to September 2014 were Broadly Acceptable. The probabilistic assessment reported in this current submission has confirmed the risks associated with graphite brick cracking and coolant flow by-pass to lie well within the Broadly Acceptable region, thereby strengthening the safety case for the integrity of the Wylfa graphite cores.

8 ASSESSMENT AGAINST THE NUCLEAR SAFETY PRINCIPLES

The 2009 annual review reported in NP/SC 4807 Addendum 5 (Reference 1) provided the most recent assessment of the graphite safety case against the NSPs. That review did not identify any changes to the basis or principles on which the graphite safety case was made, but reported a number of improvements that had been completed in support of the safety case.

There are no changes in compliance with the NSPs as a result of this current submission. However, the graphite safety case has been further strengthened by the probabilistic assessment reported here. In particular, the results of the assessment confirm that the radiological risks associated with graphite brick cracking and coolant flow by-pass lie in the Broadly Acceptable region (NSP 3). Improvements in the reliability of the operator to manually trip the reactor in response to BCD alarms (NSP 2.2 (e)) have also been confirmed (Section 5 above and Reference 23); this improved operator effectiveness is based on plant and procedural modifications and operator training and awareness.

NSP 3 states that best estimate methods and data should preferably be used for transient and other plant analyses, for radiological analysis and for frequencies and probabilities. Where this is not practicable, reasonably conservative assumptions should be made but the accumulation of pessimisms should be avoided. As noted in Section 4.1.4 above, the probabilistic assessment presented in this current submission has included a number of pessimistic assumptions and hence the predicted clad melt probabilities could be conservative by up to several orders of magnitude. Nevertheless, even with these conservatisms, the results obtained demonstrate a very low level of risk and hence provide robust support to the safety case.

9 CONCLUSION

The results of a probabilistic assessment indicate that the nuclear safety risks associated with graphite brick crack initiation, crack progression, brick splitting and consequential coolant gas flow diversion are well within the Broadly Acceptable region for operation up to September 2014. This conclusion is not sensitive to changes in input parameters.

10 RECENT DEVELOPMENTS

A programme of work is currently ongoing to support the safety cases for the Wylfa and Oldbury reactor graphite cores. The work is aimed at improving understanding of graphite behaviour and refining associated assessment methodologies.

As previously reported in the most recent Wylfa and Oldbury graphite safety cases (References 1 and 24 respectively), the axial variations in fuel channel bore radius, as measured by the NOREBORE system have been used to deduce the gradients of the graphite brick dimensional changes; these dimensional change data were then used to infer strain and stress in the bricks (Reference 25). This work resulted in a more accurate prediction of the temperature derivative of dimensional change which very strongly affects brick hoop stresses. The revised dimensional change equations (Reference 25) predicted hoop stresses caused by dimensional change (accounting for approximately 30% of the total internal stresses during the SDT) to be less than half those predicted by the previously used equations. However, the revised equations have not been used in any stress analyses in support of the graphite safety cases for either Wylfa or Oldbury, hence, in this respect, the assessments contain significant conservatisms.

The CTE is also an important parameter employed in the derivation of SDT stresses in support of both Wylfa and Oldbury safety cases. CTE measurements are routinely undertaken, over a temperature range of 20-200°C, on graphite samples removed from the reactors by trepanning. The stress analyses use relationships derived from Materials Test Reactor (MTR) CTE data obtained over the temperature range 20-120°C, enhanced by a multiplicative factor to take account of differences between MTR and station data. However, for the SDT analyses, CTE values are required over the temperature range 270-370°C and a further correction factor is applied using an expression derived from data obtained on un-irradiated graphite. As part of the ongoing programme of work to reduce uncertainties in stresses, CTE measurements have recently been made, at temperatures up to 370°C, on a small number of irradiated graphite samples removed from the Wylfa reactors. The results indicate a higher temperature dependence of CTE than previously assumed; consequentially, the thermal stresses employed in the graphite safety cases could have been underestimated, introducing non-conservatisms.

In order to address the safety case implications of the above recent developments, scoping calculations have been undertaken to estimate their combined effect on Wylfa graphite brick stresses (Reference 26). The results indicate that the combined effect on SDT hoop stresses is neutral or beneficial at core layers 6-9 (the dominant layers); at the peak stress location (octagonal brick layer 9) a net reduction of about 8% is indicated. At the lower and higher layers, some increases in hoop stresses are indicated. However, in view of the lower nuclear safety significance of brick failures at the higher layers and the low likelihood of crack initiation at the lower layers, it is judged that, overall, there are no significant safety case implications associated with these developments. Nevertheless, a further programme of work is being developed to provide a more comprehensive assessment of the temperature dependence of CTE.

11 FURTHER WORK

NP/SC 4807 Addendum 5 (Reference 1) identified ongoing/further work in support of the safety case and completion of some of the identified work is reported in this current submission. A further review of the safety case will be presented to the NSC in Addendum 7 later this year in accordance with the commitment given in Addendum 5. Addendum 7 will report on the completion of all outstanding activities identified in Addendum 5 and report progress on the further assessment of the temperature dependence of CTE (Section 10 above). It will also include a summary of the graphite safety case together with proposals for further reviews of the case as appropriate.

In order to optimise continued generation at Wylfa, consideration is being given to the possibility of inter-reactor fuel transfer (Reference 27). It is not anticipated that the effect, on predicted core state, of the implementation of such a strategy for a limited period would have a significant effect on the outcome of the probabilistic assessment reported here. Nevertheless, prior to implementing any strategy of inter-reactor fuel transfer, the associated implications for the graphite safety case will be addressed.

12 RECOMMENDATIONS

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

1. **NOTE** that the results of a probabilistic assessment confirm the risks associated with graphite brick crack initiation, crack progression, brick splitting and consequential coolant gas flow diversion to be Broadly Acceptable, thereby providing additional support to the safety case reviewed and summarised in Addendum 5.
2. **NOTE** the proposal to present a further Addendum to the NSC prior to the end of 2010.

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4. Magnox North Report MEN/EWST/WYA/REP/0021/10 Issue 1, Probabilistic Assessment of Wylfa Graphite Core Integrity, [redacted] June 2010.
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6. NP/SC 4807 Addendum 4, Revision 1, Wylfa Power Station: Annual Review of the Safety Case for the Integrity of the Graphite Cores: 2008, [redacted] November 2008.
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14. Magnox North Report MEN/EWST/WYA/REP/0057/09 Issue 1, Choice and Definition of Structural Integrity Input Parameters and their Variability for Use in the Wylfa Graphite Core Probabilistic Assessment, [redacted] December 2009.

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Core Layer	Pessimised Transient		Measured Transient	
	Square Bricks Ref 4 Tables 24a&b	**Octagonal Bricks** Ref 4 Table 24d	**Square Bricks** Ref 4 Table 26a	**Octagonal Bricks** Ref 4 Table 26d
12	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$
11	$< 10^{-8}$	5.3×10^{-6}	$< 10^{-8}$	2.8×10^{-7}
10	1×10^{-8}	7.9×10^{-4}	$< 10^{-8}$	6.1×10^{-5}
9	$3.8\text{x}10^{-7}$	8.4×10^{-3}	1×10^{-8}	1.1×10^{-3}
8	7.5×10^{-7}	9.2×10^{-3}	2×10^{-8}	1.4×10^{-3}
7	1.3×10^{-6}	4.4×10^{-3}	1×10^{-8}	6.4×10^{-4}
6	$< 10^{-8}$	2.4×10^{-5}	$< 10^{-8}$	1.2×10^{-6}
5	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$
4	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$
3	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$
2	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$	$< 10^{-8}$

Table 1: Maximum Crack Initiation Probabilities (per hard trip for individual bricks)

Core Layer	Square Bricks			Octagonal Bricks		
	Limiting Gape (mm) Ref 4 Table 17	**Flow Impairment[5] (%)**	**Conditional Clad Melt Probability** Ref 4 Table 19	**Limiting Gape (mm)** Ref 4 Table 17	**Flow Impairment[5] (%)**	**Conditional Clad Melt Probability** Ref 4 Table 19
12	3.5	26	1.1×10^{-11}	2.2	17	2.9×10^{-13}
11	3.5	26	1.1×10^{-11}	2.2	17	2.9×10^{-13}
10	4.4	35	4.6×10^{-9}	2.6	24	8.9×10^{-11}
9	5.1	41	1.8×10^{-7}	2.9	29	1.6×10^{-9}
8	5.6	45	6.6×10^{-6}	3.1	32	1.2×10^{-8}
7	5.9	48	2.2×10^{-4}	3.2	34	7.2×10^{-8}
6	5.9	49	1.5×10^{-3}	3.4	36	4.7×10^{-7}
5	5.6	48	2.9×10^{-3}	3.6	38	4×10^{-6}
4	4.9	45	1.2×10^{-3}	3.9	39	3.2×10^{-5}
3	4.5	42	3.6×10^{-4}	4.2	40	1.3×10^{-4}
2	3.4	33	7.7×10^{-7}	2.3	22	3.3×10^{-9}

Table 2: Variation of Limiting Crack Gape, Flow Impairment and Conditional Clad Melt Probability with Core Layer

[5] Interpolated values using Reference 18 Tables 2A, 2B, 2C and 5

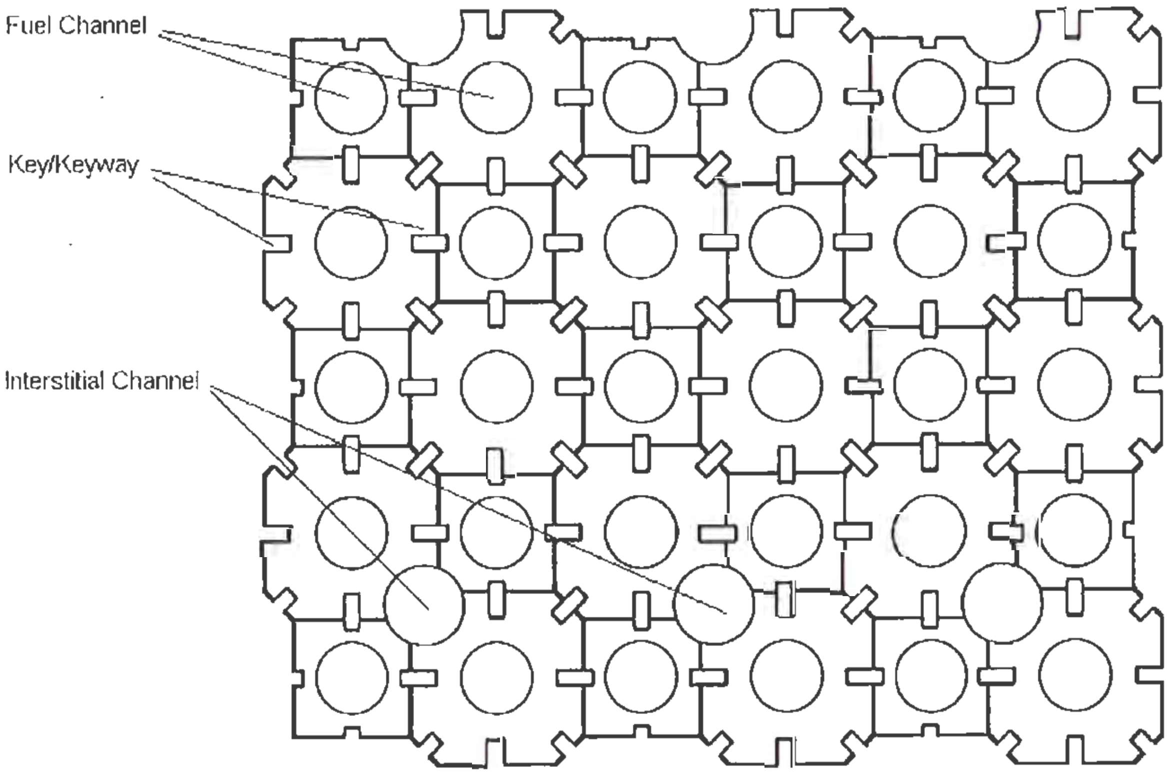


Figure 1: Plan Arrangement of Wylfa Graphite Bricks

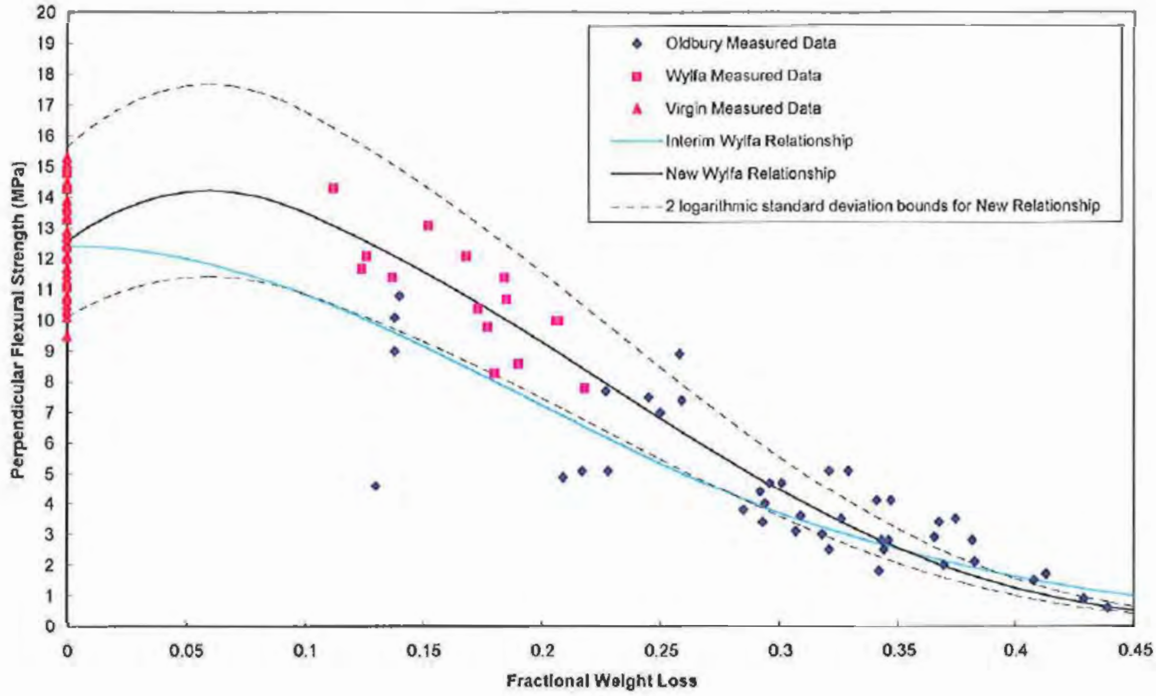
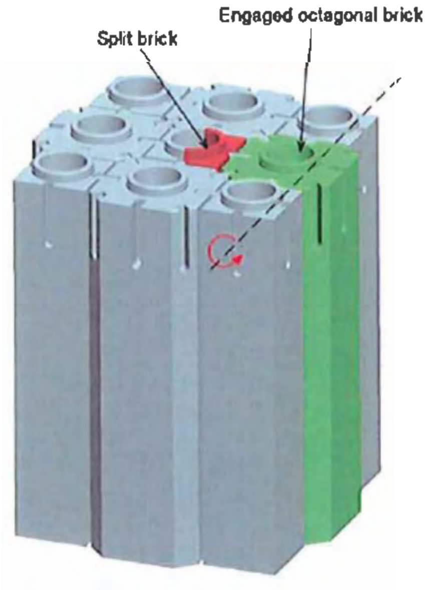


Figure 2: Comparison of the Updated Perpendicular Strength Relationship used in the Probabilistic Assessment with the Previously Used Strength Relationship

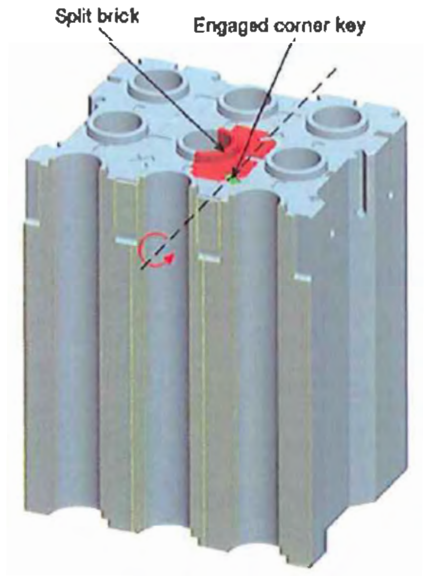
Notes

1) Derivation of the interim Wylfa relationship shown above (and used in NP/SC 4807 Addendum 5, Reference 1) did not include the Wylfa measured data.
2) The new Wylfa relationship (derived in Reference 12) utilises both Wylfa and Oldbury measured data (all >10% weight loss) and employs a separate polynomial fit between 10% weight loss and the virgin graphite data. Note that, although the Interpolation below 10% is purely a mathematical fit, it is not inconsistent with the expected process of irradiation hardening of the graphite at low accumulated doses.
3) Recent Wylfa data (reported in Reference 13 but not shown above) have been found to lie mostly above the new Wylfa relationship.
4) The data points for the Wylfa installed set weight-loss measurements are based on contact mensuration. In their endorsement of the new relationship, the GTIG considered that this method is sufficiently accurate, largely due to the superior surface finish produced by the diamond cutting method of preparation for the Wylfa specimens. The Wylfa data have, therefore, not been corrected to be equivalent to the waxed immersion method employed on the Oldbury specimens.

a) Formation of Triangular Shape Crack



b) Split Square Brick (Red) Acting on Adjacent Octagonal Brick (Green)



c) Split Octagonal Brick (Red) Acting on Corner Key

Figure 3: Development of Crack Gape in Split Graphite Bricks

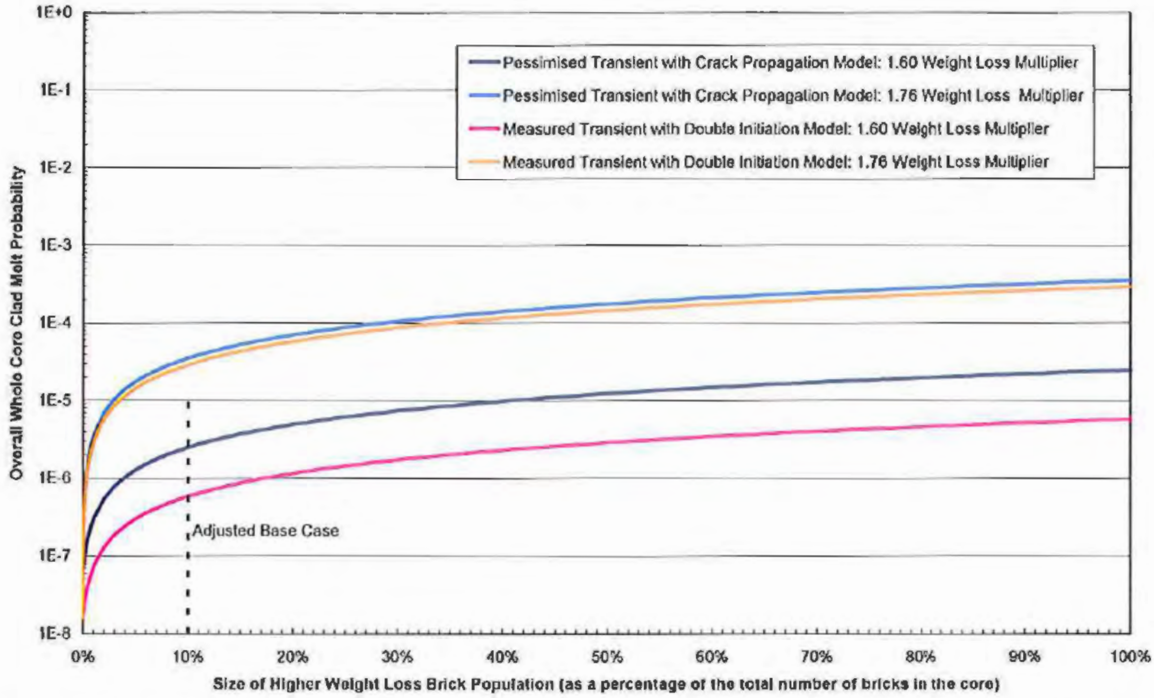


Figure 4: Variation of Reactor 1 Whole Core Clad Melt Probability with Proportion of High Weight Loss Bricks

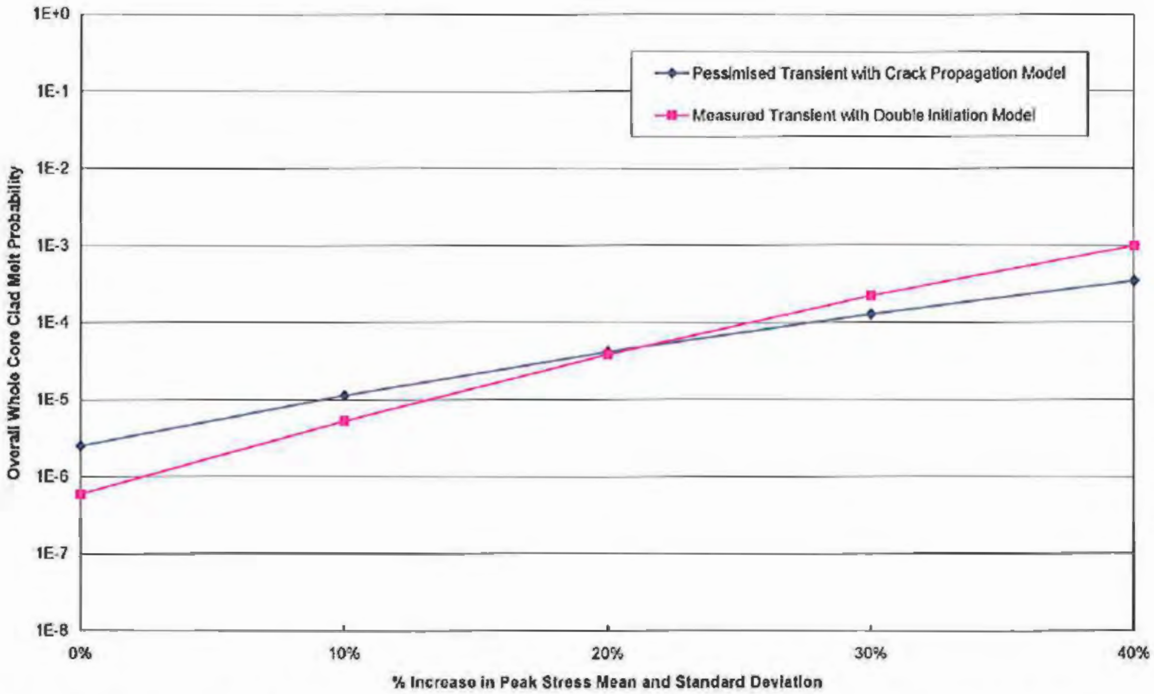


Figure 5: Variation of Reactor 1 Whole Core Clad Melt Probability with Increase in Peak Stress (Adjusted Base Case)

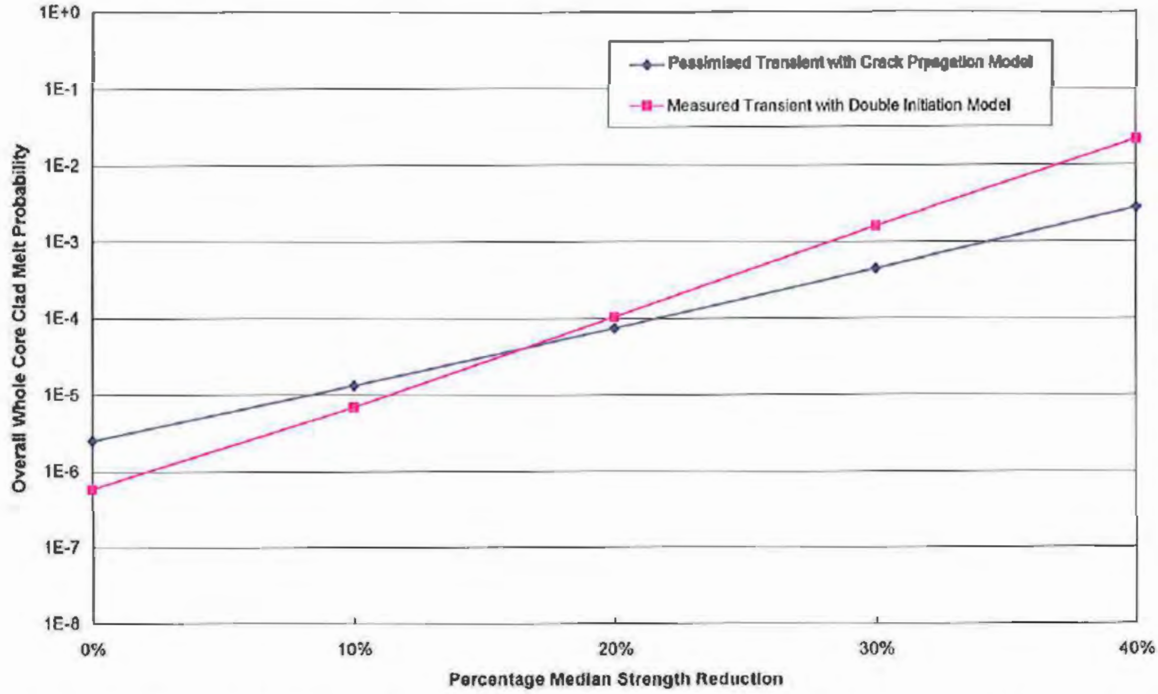


Figure 6: Variation of Reactor 1 Whole Core Clad Melt Probability with Reduction in Median Strength (Adjusted Base Case)

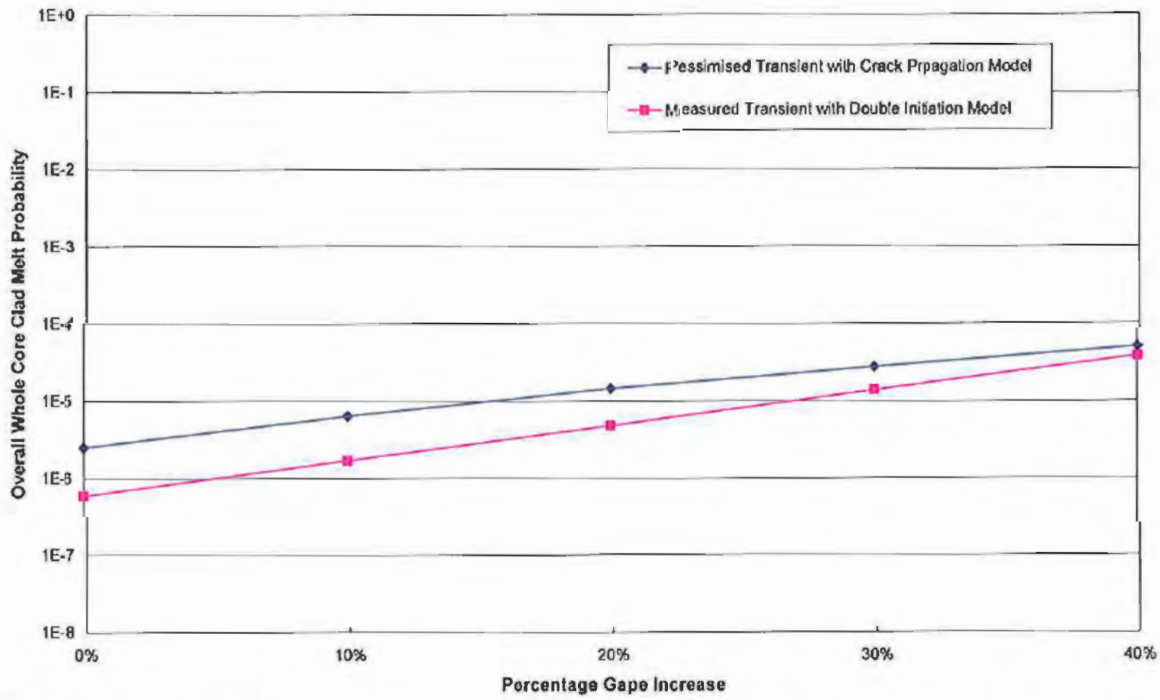


Figure 7: Variation of Reactor 1 Whole Core Clad Melt Probability with Increase in Crack Gape (Adjusted Base Case)

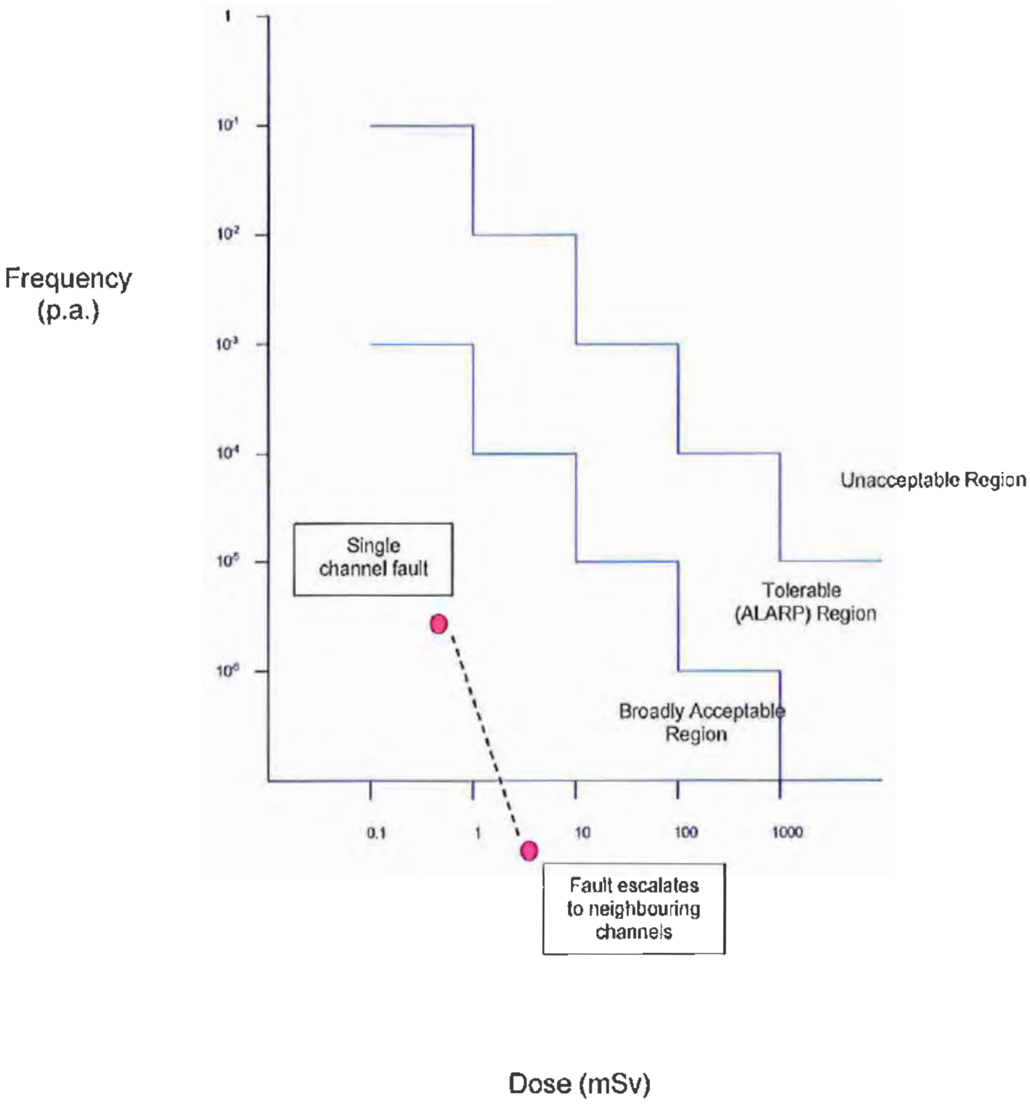


Figure 8: Radiological Risks - Frequency vs Effective Dose for Single Class of Faults (Adjusted Base Case)

APPENDIX A: VERIFICATION PLAN

VERIFICATION PLAN	No: RG7428_VP_001 Issue: Issue 1		PF009
Authors: [redacted]	Approved: [redacted]	Date: 14/05/10	
Document Ref: NP/SC 4807 Addendum 6 Document Title: **Wylfa PS: Safety Case for the Integrity of the Graphite Cores: A Probabilistic Assessment** Date Verification required by: 30th June 2010			Issue: 1

VERIFICATION RISK ASSESSMENT

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

NOTE FOR LEAD VERIFIERS

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

SELF VERIFICATION CHECKS

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

INDEPENDENT VERIFICATION

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

Initial to confirm pre-verification discussion has taken place		Document Section	Scope, input document, acceptance criteria	Risk No(s)	Verification Statement Required
Originator	Verifier				
1, 2	1, 2	All	Strategy Document for 2010 review.		The safety arguments are logical, clear and soundly based. The paper is complete and consistent with Graphite Programme decisions.
1,2	1	All	The Cat 1 paper and all references		Information in references has been accurately cited.
1,2	1	All	Safety issues and commitments in previous submissions or NSC statements	2	All relevant nuclear safety issues and commitments have been adequately addressed and judgements are supportable.
2	1	Appendix B – Probabilistic Assessment	Appropriate citation of PA methodologies, input data and results.	1	PA methodologies, input data and results have been checked for consistency with supporting references.
2	3	Section 4.1 (Clad melt probability aspects)	Citation and interpretation of supporting references	3	The data and interpretations in the consequences sections are accurate.
1,2	4	All: Plant data and operating conditions	Data relating to plant and its operation	4	The plant data and operating conditions are correctly described.
1	1	Public Dose Assessment	Off site release re-assessment		Accuracy of citation and correctness of NSP 3 assessment
1	5	All	Overall adequacy of safety case		The safety arguments are clear and adequately presented and the safety case is supportable.
	1	All	Comments by all supporting verifiers.		Comments by all supporting verifiers have been adequately shared and addressed.

Use form PF010 to record verification activities.

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: SUMMARY OF PROBABILISTIC ASSESSMENT

by [redacted] (EWST OTC)

B1 INTRODUCTION

The purpose of this appendix is to summarise technical aspects of the probabilistic structural integrity assessment for the Wylfa graphite cores undertaken in support of the present safety case addendum. This summary mirrors the comprehensive description presented in Reference B1.

A probabilistic integrity assessment for the Oldbury graphite cores was completed in December 2008 (Ref. B2). The Wylfa probabilistic assessment has been modelled on that earlier Oldbury assessment. It has been recognised, however, that in some important respects the Oldbury and Wylfa graphite cores differ substantially. Adjustments to the probabilistic assessment methodology have been incorporated to accommodate these differences. Nevertheless, the underlying principles and assumptions remain unaltered.

B2 OUTLINE OF THE DAMAGE MODEL

The aim of the assessment is to estimate the probability that melting of the Magnox fuel cladding will occur in the Wylfa reactor cores as a result of graphite moderator brick cracking. For clad melt to occur, at least one moderator brick must crack in such a way that bypass coolant flow through the crack starves the fuel adjacent to, or above, the crack of adequate cooling. Such cracking requires the brick to have been subjected to stresses sufficient to exceed the brick strength. Furthermore any resulting localised cracking must develop to a geometry that allows substantial crack opening.

Horizontal, circumferentially oriented cracks would be held closed by the weight of bricks and other components above. Thus, as for Oldbury, the significant graphite brick damage scenario at Wylfa is considered to involve the initiation of a vertical, axially oriented crack on the fuel channel bore (the region of highest tensile hoop stress). That axial crack might propagate to the entire length of the brick and fully through the brick wall. However, as a result of the internal stresses within the brick, any opening of a single full-length through-thickness axial crack will be negligible. For significant bypass coolant flow to ensue, therefore, a second distinct axial crack must form in the same brick and the two portions of brick thus created must separate. It is this damage mode that is examined in the present probabilistic assessment.

The damage process within an individual brick has been broken down conceptually into three precursor stages:

i) Axial failure initiation - the onset of localised axial cracking at the fuel channel wall of the brick, defined mathematically as the occurrence of a state in which the maximum hoop stress on the fuel channel wall exceeds the flexural strength perpendicular to the brick extrusion axis;

ii) Progression to double axial cracking - development from localised axial cracking to through-thickness axial cracks on both sides of the brick giving an axially split brick;

iii) Coolant flow bypass and fuel clad melting - the cracks open and a sufficient proportion of the normal coolant flow exits from the fuel channel through the cracks to starve fuel adjacent to, or above, the cracks of adequate cooling leading to an increase in fuel temperatures and to clad melting.

On the basis of operational, inspection and monitoring evidence from both Wylfa and Oldbury reactors, a fundamental assumption of the probabilistic assessment methodology is that the moderator bricks are in a substantially undamaged condition prior to the initiating event that causes a brick to crack.

There is substantial variability in graphite moderator brick geometries, material properties, loading parameters and potential coolant flow conditions. The conditional probabilities associated with each of the above precursors to fuel clad melting are functions of this variability.

B3 OVERVIEW OF THE ASSESSMENT PROCESS

The key steps in the assessment process for a specified loading condition are summarised below.

Step A) For each brick in the flux-flattened region of the core:

- Step A1) Estimate the probability distribution for the peak hoop stress associated with each potential initiation site on the fuel channel wall under a given loading condition;
- Step A2) Estimate the probability distribution for graphite flexural strength perpendicular to the extrusion direction at the location of maximum stress;
- Step A3) From the probability distributions for stress and strength, calculate the axial failure initiation probability for each initiation site;
- Step A4) Estimate the conditional probability, given failure initiation at a particular initiation site, that two distinct axial cracks will form within the brick;
- Step A5) Estimate the conditional probability, given two distinct axial cracks, that fuel clad melt will occur as a result of coolant flow bypass;
- Step A6) Combine the above probabilities across all initiation sites to obtain a fuel clad melt probability associated with cracking in the brick being considered.

Step B) Combine the individual brick clad melt probabilities to obtain the probability that fuel clad melting will occur somewhere within the flattened region of the core as a result of graphite moderator brick cracking under the specified operating condition.

Step C) From the flattened region clad melt probability estimate the probability that fuel clad melt will occur somewhere within the entire core as a result of graphite moderator brick cracking under the specified operating condition.

When the initiating event that may cause brick cracking is the thermal transient occurring during a reactor shutdown it is implicitly assumed in the present assessment that failure initiation and progression to double cracking (Steps A1 to A4, above) will occur during that transient. Clearly, however, as the reactor is being shutdown, melting of the fuel cladding will not occur immediately. Rather, if fuel clad melting is going to occur it will happen when the reactor is returned to power following the shutdown. Thus the relevant reactor state for evaluating the conditional probability of clad melt given double brick cracking is, in all cases, the normal operating condition. As in the Oldbury probabilistic assessment, no benefit is taken in this assessment from inspections that may be conducted during the shutdown period or channel gas outlet temperature monitoring that is conducted during reactor start-up.

B4 NOTABLE DIFFERENCES BETWEEN OLDBURY AND WYLFA REACTOR CORES

A comprehensive investigation has been undertaken to identify differences between the Oldbury and Wylfa reactors that impinge upon the probabilistic assessment methodology and input parameters. The most notable differences are highlighted below.

B4.1 Core Size

The Wylfa cores are much larger than those at Oldbury. Each Wylfa core contains 6156 fuel channels (3568 flattened region, 2588 unflattened region) with moderator bricks arranged in 11 layers (brick layers 2 to 12). By contrast, Oldbury cores each have 3308 fuel channels (1304 flattened region, 2004 unflattened region) with 10 moderator brick layers (layers 2 to 11). Thus more bricks contribute to the overall risk of clad melt at Wylfa than at Oldbury.

B4.2 Brick Dimensions

The fuel channel pitch is the same at both sites. However, square bricks at Wylfa are smaller than those at Oldbury and, to maintain the same channel pitch, the reverse is true for octagonal bricks. The primary effect of brick size is on stresses within the bricks. Consequently, the relative likelihoods of crack initiation and propagation within square and octagonal bricks will differ between Wylfa and Oldbury.

B4.3 Inter-brick Connections

At Oldbury the keyways and inter-brick graphite keys are continuous over the full height of the moderator region of the core and, in particular, span the interface between vertically adjacent moderator bricks. At Wylfa the keyways and keys are confined to a short length at the top of each brick and do not span horizontal layer interfaces. The full length Oldbury keys provide a significant impediment to bypass flow should a cracked brick occur, particularly if the cracks are through keyway locations. As a consequence, within the Oldbury probabilistic assessment it was demonstrated that only bricks where cracking could discharge bypass flow directly into an interstitial channel (that is, bricks with interstitial cut-outs) contribute significantly to the overall risk of fuel clad melt. The partial length keys in the Wylfa cores, on the other hand, give little flow impediment once coolant has escaped from the fuel channel. As a result all bricks will contribute to the Wylfa clad melt risk irrespective of whether or not they have interstitial cut-outs.

Keyway geometry also influences the stress pattern within the bricks, which will differ, therefore, between the two stations. A positive aspect of the unrestricted flow path outside the moderator bricks is that differential pressures will not exist across the partial length Wylfa keys. As a consequence key/keyway loads on the moderator bricks are negligible at Wylfa.

Horizontal alignment between vertically adjacent Oldbury moderator bricks is assured by the interface-spanning keys. At Wylfa the keys do not span the interface between vertically adjacent bricks and alignment is assured by a spigot/counterbore arrangement at that interface. While Oldbury bricks also have spigots and counterbores, the spigots are small and the diametral spigot/counterbore clearance is relatively large. The Wylfa spigots are more substantial with very small diametral clearance. These features influence the geometry of potential crack face separation that can be achieved should a double-cracked brick occur.

B4.4 Graphite Material Properties

The highest best estimate end of generation (September 2014) fuel channel wall graphite weight loss in the Wylfa cores is projected to be approximately 17% (including waxed immersion correction). This is considerably lower than the equivalent Oldbury weight loss of about 33%. Weight loss affects many other graphite properties that collectively determine stress and strength within the moderator bricks. Furthermore, the rate of weight loss with irradiation dose is lower at Wylfa than Oldbury. Thus the balance between competing irradiation and weight loss effects is different between the two stations and this will be reflected in the trends in material properties. This is particularly influential in the lower weight loss range pertinent to Wylfa. Thus significant differences in the values and trends of material properties relevant to prediction of stress and strength are to be expected between Wylfa and Oldbury.

B5 PROBABILISTIC ASSESSMENT METHODOLOGY

For the present Wylfa assessment a number of detailed changes have been made to the assessment methodology employed previously for Oldbury. Many of these changes have been incorporated to accommodate the differences between the Oldbury and Wylfa graphite cores described above. The Wylfa assessment methodology is summarised briefly below with particular emphasis being given to these changes.

B5.1 Step A1 - Derivation of Peak Hoop Stress Probability Distributions

To define the probability distribution for peak hoop stress at each potential axial failure initiation site on the fuel channel wall of a given moderator brick, a series of finite element stress analyses is undertaken. The specific input parameter values employed in an individual stress analysis are selected by stratified hypercube sampling from the probability distributions for each variable input parameter. From each analysis peak hoop stress values in the region of potential failure initiation sites (corresponding to regions of raised tensile hoop stress) are extracted. By design, the set of peak hoop stress values accumulated over the analysis series is representative of the overall peak hoop stress probability distribution. The final stage is to fit an analytical generalised extreme value probability distribution to the peak stress data corresponding to each identified initiation site.

B5.1.1 Initiation Sites

Typically, several distinct failure initiation sites (stress peaks) are predicted to occur on the fuel channel bore of a brick. Within the Oldbury assessment, based on the evident brick symmetries, all initiation sites were conservatively assumed to have the same peak hoop stress probability distribution. Wylfa bricks, however, have clear asymmetries arising because of the short keyways and circumferential variations in brick wall thickness. As a result, within the present assessment, peak hoop stress probability distributions have been derived individually for each initiation site.

B5.1.2 Loading Conditions

Naturally, peak hoop stress probability distributions depend upon loading condition. As for Oldbury, previous assessments have established that the most onerous loading condition challenging Wylfa moderator brick integrity occurs as a result of the thermal transient during a reactor shutdown. This condition is, therefore, the primary focus of the probabilistic assessment although the normal full-power operating condition is also assessed for completeness. As in the Oldbury assessment, two thermal transients are considered: a deliberately pessimised trip transient and a more realistic, but still severe, transient. The pessimised transient has been modelled to give a fall in bulk channel gas outlet temperature that will bound any trip likely to be experienced during remaining operation. This pessimised transient has been specifically characterised for the present assessment and differs from that used in previous deterministic assessments. The more realistic, or "measured", transient is based on a hard trip that occurred on Reactor 2 on 13th May 2009. The way in which conditional failure initiation probabilities calculated for each of these loading conditions are combined with conditional probabilities for crack progression (Step A4) is discussed below (Sections B5.4 and B6).

B5.1.3 Characterisation of Stress-determining Parameters

A necessary pre-cursor to the stratified hypercube sampling parameter selection process is the characterisation of variability in stress-determining material properties (weight loss, Young's modulus, coefficient of thermal expansion, Poisson's ratio etc) and other parameters. The significance of these properties and parameters to predicted stresses in Wylfa moderator bricks has been thoroughly investigated leading to identification of the most important properties and parameters. The finalised set of such properties and parameters is identical to that developed for the Oldbury assessment with two exceptions:

- thermal conductivity has been explicitly included in the Wylfa assessment because its variability is of greater significance at the lower weight losses relevant to the Wylfa reactors;
- key/keyway contact length is not considered in the Wylfa assessment as differential gas pressure loads applied through the inter-brick keying are negligible as a consequence of the partial length keying.

In common with the Oldbury assessment, two important quantities: dimensional change and secondary creep coefficient, are included as conservatively chosen fixed parameters because of the difficulty in defining appropriate probability distributions. Suitable probability distributions for all other (variable) properties and parameters have been derived by examination of Wylfa-specific measured data where possible.

Many of the stress-determining material properties depend on weight loss. Definition of weight loss variability is, therefore, crucial to defining the total variability of such dependent properties. For the present assessment weight loss probability distributions are predicted using the BEST model (taking account of Wylfa measured weight losses up to and including the Reactor 1 2007 and Reactor 2 2008 trepanning campaigns). There are insufficient reactor-specific measured data to allow the layer-by-layer statistical analysis approach employed for Oldbury. Critical examination has confirmed preferred use of the BEST model predictions within the present assessment. As in the Oldbury assessment, the effects of channel to channel dose variability are accounted for by enhancing the variance of the assumed weight loss probability distributions.

The choice and definition of these stress-determining parameters has been endorsed by the Graphite Technical Issues Group.

B5.1.4 Stratified Hypercube Sampling Stress Analyses

Since the Oldbury assessment, considerable improvements have been made to the efficiency with which stress analyses may be completed. This together with advances in available computing power has enabled full programmes of stress analyses to be undertaken for all moderator brick geometries and layers within the present assessment regardless of the level of risk posed by each brick. It has not been necessary, therefore, to employ the approximate extrapolation technique used for Oldbury to derive peak stress probability distributions for the less significant core layers (although it should be noted that this did not influence the overall calculated clad melt probability).

B5.2 Step A2 – Derivation of Flexural Strength Probability Distributions

In common with Oldbury moderator bricks, predicted stress fields within Wylfa moderator bricks exhibit approximate through-wall bending profiles. It is therefore most appropriate to compare predicted hoop stresses with flexural strength perpendicular to the brick extrusion axis. Within a given brick this flexural strength is assumed to be uniform across the fuel channel wall, consistent with the assumptions made regarding spatial distribution of material properties in the stress analysis models. Flexural strength depends on the degree of irradiation and radiolytic corrosion. Weight loss is used as a single proxy variable that parameterises both of these influences.

As for Oldbury, a relationship between flexural strength and weight loss has been derived by statistical analysis of flexural strength measurements made on specimens manufactured from installed set samples extracted from the Oldbury and Wylfa reactors. Of particular importance to the present assessment are measurements obtained from samples withdrawn from Wylfa Reactor 2 during the 2008 statutory outage. These strength data have substantially increased the quantity of data in the lower weight loss range relevant to the Wylfa reactors. An improved sample machining technique has also been deployed to prepare test specimens from these samples. The resulting strength measurements have been found to exhibit greater consistency.

The weight loss/flexural strength relationship used in the present assessment (see Figure 2 of the main paper) has been endorsed by the Graphite Technical Issues Group.

As in the Oldbury assessment the assumed flexural strength has been reduced to account for the influences of stress multi-axiality and stressed area. The method for estimating stressed areas for each initiation site remains unchanged in the present assessment but specific stressed areas have been evaluated for each initiation site in each brick geometry in each layer rather than adopting a limited number of typical values as employed in the Oldbury assessment.

B5.3 Step A3 – Calculation of Failure Initiation Probabilities

The method for calculating failure initiation probabilities for each initiation site is unchanged from that in the Oldbury assessment: Monte Carlo integration of the correlated joint peak hoop stress/flexural strength probability density function. Correlations between stress and strength via weight loss and Young's modulus are applied exactly as previously. The only substantive change made in applying this method is a global increase in the Monte Carlo sample size from 10^7 to 10^8 to address the lower level of initiation probability found for Wylfa (see Section B7, below).

B5.4 Step A4 – Progression to Double Axial Cracking

In the absence of specific empirical data relevant to crack progression in Wylfa moderator bricks the conditional probability that failure initiation will progress to a fully double-cracked brick must, of necessity, be estimated theoretically. In common with the Oldbury assessment, a prudent approach is adopted whereby two diverse analysis models are used to estimate the crack progression probability: the "crack propagation model" and the "double initiation model".

B5.4.1 Crack Propagation Model

An expert elicitation approach has been adopted to assign judgemental conditional probabilities to an exhaustive set of potential outcomes at each stage in progression to double cracking. The supporting judgements are based on mechanistic understanding of cracking in quasi-brittle materials. Probabilities assigned to outcomes leading ultimately to double cracking are aggregated, giving an estimate of the overall conditional probability that double cracking will occur.

Based on examination of the predicted strain fields within peak-rated Wylfa square and octagonal bricks, the expert elicitation assessment has derived upper bound conditional double cracking probabilities of 10^{-2} covering all brick geometries. This bounding value has been applied to all bricks in all layers. The equivalent probabilities derived for Oldbury bricks are 10^{-3} for all brick geometries. These differences between Wylfa and Oldbury arise largely because of the influences of brick geometry, particularly wall thickness.

B5.4.2 Double Initiation Model

As for Oldbury, the double initiation model of crack progression equates first failure initiation to the complete and immediate development of a single full-height through-thickness crack. The second full-height through-thickness crack is assumed to develop under the same loading condition that caused the first initiation with a probability determined by a second failure initiation. This second initiation is independent of the first except for the effects of stress enhancement resulting from the existence of the first crack.

This model is implemented within the present Wylfa assessment in a similar way to that applied in the previous Oldbury assessment. For Wylfa, cracked-brick stress enhancements have been calculated explicitly for all brick geometries and all layers whereas in the Oldbury assessment only enhancements representative of the most significant brick layers were calculated and applied by extrapolation to all layers. Again, this greater level of detail has been enabled by improvements in the efficiency with which stress analyses may be completed and advances in computing power. In practice, however, for the most significant brick geometries and layers, the Oldbury and Wylfa approaches are entirely consistent.

B5.4.3 Double Crack Geometries

A significant feature of the Oldbury assessment is consideration of circumferential location for the two postulated axial cracks. Of particular importance is whether or not one of the cracks is directly through to an adjacent interstitial channel. For Wylfa the location of the cracks is immaterial because of the connected flow path external to the brick that results from the partial length keys. Regardless of the double crack geometry there would be little resistance to subsequent bypass flow beyond the boundary of the doubly-cracked brick. Within the Wylfa present assessment, therefore, it is not necessary to evaluate relative double crack geometry conditional probabilities.

B5.5 Step A5 – Coolant Flow Bypass and Clad Melt

Given the existence of a doubly-cracked brick there is a conditional probability that this will cause fuel clad melt adjacent to, or above, the cracks. This conditional probability depends upon the amount by which the two parts of the brick can separate (crack gape) and the subsequent resistance to bypass coolant flow in the context of the local channel axial power profile.

B5.5.1 Crack Gape

The constraints on crack opening for Wylfa bricks are different from those that apply to Oldbury bricks. The analysis of Wylfa crack gapes considers the following factors that collectively determine the limiting crack opening for each brick geometry and layer:

- resistance from the spigot/counterbore interface;
- resistance from the weight of bricks above the crack;

- freedom of movement associated with Wigner gaps and key/keyway clearances;
- resistance applied by adjacent brick columns.

Of particular importance is the influence of the spigot/counterbore interface which, if it remains intact, constrains the crack opening at the top of the cracked brick to be essentially zero and the crack opening geometry to be "triangular" with greatest opening at the base of the brick. On the basis of the results of full scale strength testing, Reference B1 demonstrates that spigot/counterbore failure can be discounted.

Pessimistic estimates of crack gape have been derived for all layers and brick geometries. It is found that gapes in cracked octagonal bricks are limited by the restraint applied through corner keys and keyways. The gapes for square bricks, that lack corner keys, are limited by the freedom of movement allowed by Wigner gaps and the restraint applied by adjacent brick columns after those gaps have been taken up. The crack gapes employed in the present assessment pessimistically ignore any beneficial effects of sliding friction between graphite components.

B5.5.2 Coolant Flow Bypass

Coolant flow bypass in the presence of a doubly-cracked brick is evaluated using a flow network approach similar to that employed in the corresponding Oldbury assessment. A triangular crack opening geometry is assumed consistent with the intact spigot/counterbore. A significant feature of the present Wylfa assessment is re-evaluation of hydraulic loss data for the crack-to-fuel-channel junction. This has resulted in a significant reduction in predicted bypass flows with a substantial reduction in predicted conditional risk of fuel clad melt. The bypass flow calculations, including revised hydraulic loss data, have been critically examined. It has been concluded that, although some conservatisms have been reduced, significant pessimisms remain. Such pessimisms include using the shortest square brick through-crack path length for all brick geometries, maintaining pessimism in the hydraulic loss coefficients employed and assuming a low channel inlet flow rate.

B5.5.3 Risk of Clad Melt

The risk of fuel clad melt during full power operation assuming the presence of a doubly-cracked brick has been assessed using the established PANTHER-based PREDICT2 assessment route with a simulated September 2014 core state (which assumes the continued availability of new fuel). Maximum clad melt risks have been calculated for each combination of crack gape and crack height. The risk calculations assume a pessimistically high base peak channel gas outlet temperature. The conditional clad melt risks employed in the present assessment are, therefore, conservative and bounding.

The highest conditional clad melt risk, assuming limiting crack gapes, is a little over 10^{-3}. It is also notable that the risks associated with octagonal bricks are considerably (~3 orders of magnitude) lower than those from square bricks as a result of the greater constraint on crack opening applied by the corner key/keyway interactions of octagonal bricks.

B5.6 Steps A6, B and C – Calculation of the Overall Whole Core Clad Melt Probability

Calculation of the probability that fuel clad melt will occur somewhere in the whole core following the occurrence of a specified loading condition uses a similar probability combination logic to that employed for Oldbury. The only substantive change is the simplification resulting from not considering relative double crack geometry conditional probabilities (see Section B5.4.3, above). In outline the logic is as follows:

- Calculate the probability that clad melt will arise from each specific failure initiation site in a given brick;
- Calculate the probability that clad melt will arise from any initiation site in a given brick;
- Calculate the probability of no clad melt arising from all moderator bricks of the i-th geometry in layer j. Here i denotes one of four brick geometries: square with interstitial cut-out, square without interstitial cut-out, octagonal with interstitial cut-out or octagonal without interstitial cut-out.
- Calculate the probability of no clad melt arising from all moderator bricks within layer j.
- Calculate the probability of no clad melt arising from all moderator bricks in all layers.
- Calculate the probability of at least one instance of clad melt arising within the core.

All distinct initiation sites and all bricks are assumed to be statistically independent. This is the most onerous physically reasonable assumption as it maximises the calculated overall clad melt probability.

Detailed stress analysis has only been carried out for bricks in the flattened region of the core. To take account of the contribution from bricks outside the flattened region the simple conservative assumption is made, as in the Oldbury assessment, that the risk associated with a given unflattened region brick is the same as that for the corresponding (same layer, same geometry) bricks in the flattened region.

B6 ASSESSMENT CASES

Cases considered in the present Wylfa assessment are identical to those considered in the previous Oldbury assessment:

- Full power normal operation loading condition + crack propagation model;
- Full power normal operation loading condition + double initiation model;
- Pessimised shutdown transient loading condition + crack propagation model;
- Measured shutdown transient loading condition + double initiation model.

All of these cases have been assessed for each reactor. The base case assessment assumes "best-estimate" probability distributions for input parameters, where possible, notwithstanding the various pessimisms incorporated in the methodology (Section B5). Graphite material properties, in particular weight loss and dependent quantities are assigned values that reflect those anticipated to prevail at the end of optimised generation in September 2014: not exceeding mean core irradiations of 34.1 GWd/t on Reactor 1 and 32.4 GWd/t on Reactor 2. Thus the base case analyses provide a "pessimised best estimate" of the clad melt probability for the Wylfa cores in their highest potential weight loss condition. A comprehensive series of supporting analyses has been completed covering sensitivity to peak hoop stress, median strength, crack gape, the existence of bricks with "anomalously" high weight loss, and mean core irradiation.

B7 BASE CASE ASSESSMENT RESULTS

All predicted base case whole core clad melt probabilities are less than 10^{-7}, including both normal operation and shutdown thermal transient loading conditions.

Under the normal operational loading condition all first failure initiation probabilities are below the level of resolvability of the Monte Carlo integration despite having increased the Monte Carlo sample size to 10^{8} in the present assessment (Section B5.3). In a brick without prior damage, therefore, the likelihood of spontaneous brick cracking during normal operation is negligibly low.

For the shutdown transient load cases the highest calculated first failure initiation probability is a little less than 10^{-2} (within a Reactor 1 layer 8 octagonal brick with interstitial cut-out subjected to a pessimised transient). The equivalent highest initiation probability for the measured transient loading condition is about 10^{-3}. Square bricks typically have very much (~ 3 orders of magnitude) lower failure initiation probabilities than octagonal bricks in the same core layer. This is expected because octagonal bricks suffer significantly higher fuel channel wall hoop stresses. Octagonal bricks have higher cracking probabilities but have lower probabilities of consequent fuel clad melt (see Section B5.5.3). Conversely, square bricks with lower cracking probabilities have higher conditional clad melt risk as a result of their relatively larger crack gapes. The overall clad melt probability comprises contributions from square and octagonal bricks, from brick cracking risk and from conditional clad melt risk. The balance of those contributions varies from case to case as discussed further in Section B9.

Brick layers 7, 8 and 9 (slightly above the mid-height of the core) are found to exhibit the highest first failure initiation probabilities in all brick geometries and all shutdown transient assessment cases.

A final conclusion drawn from the base case assessments is that the likelihood of fuel clad melt occurring in Reactor 2 (at 32.4 GWd/t) is marginally lower than that for Reactor 1 (at 34.1 GWd/t) under all assessed loading conditions and assuming either crack progression model.

B8 SENSITIVITY STUDIES

Allowance for random uncertainty in input parameters is incorporated into the base case assessments. To quantify the influence of potential future *systematic* input parameter adjustments (arising as a result of new data or understanding, for example), a comprehensive series of sensitivity analyses has been completed. These sensitivity analyses are based around the bounding base case assessments: Reactor 1 following shutdown transient loading.

B8.1 Sensitivity to Peak Stress

The effect of increasing the mean and standard deviation of the derived peak hoop stress probability distributions by the same factor across all initiation sites in all bricks in all layers has been investigated. All other input parameters remain as in the base case assessments. The predicted variation of overall whole core clad melt probability with percentage increase in the mean (and standard deviation) of the peak hoop stress probability distributions is shown in Figure B1. The range of stress increase considered (up to 100%) is entirely arbitrary.

The overall probability that fuel clad melt will occur somewhere in the whole core as a result of moderator brick cracking is moderately sensitive to the magnitude of peak fuel channel wall hoop stress. A 20% increase in peak hoop stress (across all bricks) increases the clad melt probability by about two orders of magnitude.

Another perspective is given by considering the expected number of bricks, predicted by the analyses, to be suffering first failure initiation or full double cracking. As shown in Figure B2, expected numbers of damaged bricks increase rapidly with peak stress. However, detailed examination reveals that the vast majority of predicted damaged bricks are octagonal for which the conditional probability of subsequent clad melt is low (Section B5.5.3). Thus most of the expected damage is of lesser safety significance and the rise in overall clad melt probability with peak stress is slower than might be anticipated from the rate of rise in the number of damaged bricks. Reference B1 presents a detailed analysis of these results. A conclusion that can be drawn from a comparison of Figures B1 and B2 is that, following a hard shutdown transient, significant damage to around 10,000 moderator bricks would be expected before the risk of fuel clad melt is predicted to exceed 10^{-3}. Such levels of damage would be readily observable with the current outage inspection regime.

Figure B1 also exemplifies the finding that the double initiation model of crack progression becomes progressively more pessimistic relative to the crack propagation model as the whole core clad melt probability increases.

B8.2 Sensitivity to Flexural Strength

The effect of reducing flexural strength has been investigated by decreasing the median weight loss/strength relationship (see Figure 2 of the main paper) by a uniform factor across the entire weight loss range while retaining the original logarithmic standard deviation. This reduced strength is applied to all moderator bricks. All other parameters are held constant. The results of this sensitivity study (as summarised in Figure 3 of Reference B1) exhibit very similar trends to those seen in the peak stress sensitivity study discussed above. The overall clad melt probability is found to be somewhat more sensitive to strength reduction than to stress increase with a 20% reduction in strength increasing the predicted clad melt probabilities by between 2 and 3 orders of magnitude. This slightly increased sensitivity is a consequence of uncertainty in strength being greater than uncertainty in peak stress and the resulting asymmetry in the stress/strength joint probability density function.

B8.3 Sensitivity to Crack Gape

The crack gapes adopted in the base case assessments are considered to be pessimistic given key/keyway and inter-brick clearances (Section B5.5.1). Any postulated increase in crack gape is, as a consequence, entirely hypothetical. The largest predicted crack gape is approximately 5.9mm. Increasing the maximum gape across all bricks in all layers by 1mm (roughly 20% of the largest gape) increases the overall clad melt probabilities by between one and two orders of magnitude.

B8.4 Sensitivity to the Existence of Higher Weight Loss Bricks

The probability distributions for fuel channel wall weight loss adopted in the base case assessments, determined using the BEST model (Section B5.1.3), adequately encompass the vast majority of weight loss measurements from trepanned samples. A small number of measured weight losses have, however, been higher than expected as discussed in Reference B3. Such "anomalous" data are sparse and it has not been possible thus far to provide a conclusive physical explanation for the apparent low density of these samples.

In the absence of such an explanation, a simple bounding approach is taken to investigating the influence of higher weight loss bricks on the predicted overall clad melt probabilities. The existence is hypothesised of a distinct population of "higher weight loss" bricks that have their own set of weight loss probability distributions. The balance of bricks in the core (the "lower weight loss" brick population) is assumed to be adequately characterised by the base case weight loss distributions. It is assumed that all bricks in the core have the same likelihood of being in the higher weight loss population. This is merely a simple flexible model to represent whatever the true situation may be within the Wylfa reactors. In particular, although two distinct populations are modelled this should not be taken as suggesting that such populations would be distinguishable in reality.

Commensurate with the scarcity of higher weight loss data, the weight loss probability distributions for the higher weight loss population are assumed to be identical to those of the lower weight loss population in each layer except that the means are increased by a defined multiplying factor. Two multipliers have been used: a best-estimate multiplier of 1.6 and an enhanced multiplier of 1.76. The best-estimate multiplier of 1.6 bounds the average of the weight loss "anomalies" (measured/predicted weight loss) for the anomalous measurements discussed in Reference B3.

Stratified hypercube sampling stress analyses for all brick geometries and core layers have been repeated based on these enhanced weight loss distributions. All other input parameters except those directly affected by weight loss remain as defined for the base case assessments. On this basis revised failure initiation probabilities have been derived for bricks in the higher weight loss populations leading to revised individual brick contributions to the clad melt probability. Finally, overall whole core clad melt probabilities have been calculated assuming the existence of varying percentages of higher weight loss bricks ranging from 0% (equivalent to the base case assessments) to 100% (all moderator bricks having higher weight loss). The resulting variations of clad melt probability are shown in Figure B3.

Within Figure B3 the clad melt probability is plotted on a logarithmic scale giving an impression that clad melt probability is very sensitive to the existence of small numbers of higher weight loss bricks. This is misleading, however, since the variation of clad melt probability with the size of higher weight loss population is in fact approximately linear. The gradient of that variation is proportional to the difference in weighted average individual brick clad melt probabilities between the higher and lower weight loss populations. The simple model employed in this sensitive study is designed to maximise that difference in probabilities. It is considered likely, therefore, that the influence of higher weight loss bricks on the overall clad melt probability is exaggerated within this sensitivity study.

As an adjunct to this weight loss sensitivity study, Reference B1 presents the results of further analyses investigating the effects of increasing peak stress, reducing strength and increasing crack gapes on the predicted clad melt probabilities for cores with various proportions of higher weight loss bricks ("revised base cases"). The results follow very similar trends to those described in Sections B8.1, B8.2 and B8.3 for the sensitivity studies around the base case, but with elevated clad melt probabilities (consistent with Figure B3) prior to stress, strength or gape variation. As would be expected, in all cases the overall clad melt probability increases with increasing size of higher weight loss population. Typically, however, the sensitivity of the overall clad melt probability to peak stress, strength and crack gape tends to reduce as the assumed proportion of higher weight loss bricks grows.

B8.5 Sensitivity to Mean Core Irradiation

The core states considered in the base case assessment are representative of those anticipated to be reached at the planned end of optimised generation in September 2014 (a Reactor 1 mean core irradiation of 34.1 GWd/t). To understand how the overall whole core clad melt probabilities may be changing with time, the base case analyses have been replicated for two further Reactor 1 mean core irradiations: 30.7 GWd/t (reflecting a notional core state at December 2010) and 35.0 GWd/t (beyond projected end of generation).

The most significant effect of changing mean core irradiation is on graphite weight loss. For the purpose of this sensitivity study appropriate alterations have been made to input parameters related to weight loss or radiation dose including peak stress probability distributions and crack gapes. Effective stressed areas, numbers of initiation sites and cracked-brick stress enhancements have been held constant.

The resulting predicted variation of overall Reactor 1 whole core clad melt probability with mean core irradiation is shown in Figure B4. It is deduced that the overall clad melt probabilities are slowly increasing for mean core irradiations between 30.7 GWd/t and 35.0 GWd/t but that the probability of clad melt remains very low over this range.

B9 RELATIVE CONTRIBUTIONS TO THE OVERALL CLAD MELT PROBABILITY

As indicated in Section B7, the overall clad melt probability in each assessment case comprises contributions from square bricks and octagonal bricks and, within those brick geometries, from crack initiation, progression and clad melt damage process stages. The balance of these contributions varies from case to case and depends, in particular, upon the crack progression model. This variability is illustrated by representative examples below. In each case probability contributions from the damage process stages are quoted in terms of indicative orders of magnitude.

a) *Base Case, Pessimised Transient + Crack Propagation Model*

Overall whole core clad melt probability	$\sim 10^{-7}$
(Overall probability is dominated equally by square and octagonal bricks at layer 7)	
1st failure initiation probability - square	$\sim 10^{-5}$
1st failure initiation probability - octagonal	$\sim 10^{-2}$
Crack progression probability - square	$= 10^{-2}$
Crack progression probability - octagonal	$= 10^{-2}$
Conditional clad melt probability - square	$\sim 10^{-4}$
Conditional clad melt probability - octagonal	$\sim 10^{-7}$
Number of contributing bricks	$\sim 10^{4}$

$$((10^{-5} \times 10^{-2} \times 10^{-4} \text{ square}) + (10^{-2} \times 10^{-2} \times 10^{-7} \text{ octagonal}) \times 10^{4} \sim 10^{-7}$$

b) *Base Case, Measured Transient + Double Initiation Model*

Overall whole core clad melt probability	$\sim 10^{-8}$
(Overall probability is dominated by octagonal bricks at layers 7 and 8)	
1st failure initiation probability – octagonal layer 8	$\sim 10^{-3}$
1st failure initiation probability – octagonal layer 7	$\sim 10^{-3}$
Crack progression (2nd initiation) probability – octagonal layer 8	$\sim 10^{-1}$
Crack progression (2nd initiation) probability – octagonal layer 7	$\sim 10^{-2}$
Conditional clad melt probability – octagonal layer 8	$\sim 10^{-8}$
Conditional clad melt probability – octagonal layer 7	$\sim 10^{-7}$
Number of contributing bricks	$\sim 10^{4}$

$$((10^{-3} \times 10^{-1} \times 10^{-8} \text{ layer 8}) + (10^{-3} \times 10^{-2} \times 10^{-7} \text{ layer 7}) \times 10^{4} \sim 10^{-8}$$

c) *100% Stress Increase Sensitivity Study, Pessimised Transient + Crack Propagation Model*

Overall whole core clad melt probability	$\sim 10^{-2}$
(Overall probability is dominated by square bricks at layers 6 and 7)	
1st failure initiation probability – square layer 7	$\sim 10^{0}$
1st failure initiation probability – square layer 6	$\sim 10^{-1}$
Crack progression probability – square layer 7	$= 10^{-2}$
Crack progression probability – square layer 6	$= 10^{-2}$

Conditional clad melt probability – square layer 7	$\sim 10^{-4}$
Conditional clad melt probability – square layer 6	$\sim 10^{-3}$
Number of contributing bricks	$\sim 10^{4}$

$$((10^{0} \times 10^{-2} \times 10^{-4} \text{ layer 7}) + (10^{-1} \times 10^{-2} \times 10^{-3} \text{ layer 6}) \times 10^{4} \sim 10^{-2}$$

d) *100% Stress Increase Sensitivity Study, Measured Transient + Double Initiation Model*

Overall whole core clad melt probability	$\sim 10^{-1}$
(Overall probability is dominated by square bricks at layer 7)	
1st failure initiation probability	$\sim 10^{0}$
Crack progression (2nd initiation) probability	$\sim 10^{0}$
Conditional clad melt probability	$\sim 10^{-4}$
Number of contributing bricks	$\sim 10^{3}$

$$10^{0} \times 10^{0} \times 10^{-4} \times 10^{3} = 10^{-1}$$

e) *100% High Weight Loss Bricks Sensitivity Study (Weight Loss Multiplier = 1.6), Pessimised Transient + Crack Propagation Model*

Overall whole core clad melt probability	$\sim 10^{-5}$
(Overall probability is dominated by square bricks at layer 7)	
1st failure initiation probability	$\sim 10^{-2}$
Crack progression probability	$= 10^{-2}$
Conditional clad melt probability	$\sim 10^{-4}$
Number of contributing bricks	$\sim 10^{3}$

$$10^{-2} \times 10^{-2} \times 10^{-4} \times 10^{3} = 10^{-5}$$

f) *100% High Weight Loss Bricks Sensitivity Study (Weight Loss Multiplier = 1.6), Measured Transient + Double Initiation Model*

Overall whole core clad melt probability	$\sim 10^{-5}$
(Overall probability is dominated equally by square and octagonal bricks at layer 7)	
1st failure initiation probability - square	$\sim 10^{-3}$
1st failure initiation probability - octagonal	$\sim 10^{-1}$
Crack progression (2nd initiation) probability – square	$\sim 10^{-2}$
Crack progression (2nd initiation) probability – octagonal	$\sim 10^{-1}$
Conditional clad melt probability - square	$\sim 10^{-4}$
Conditional clad melt probability - octagonal	$\sim 10^{-7}$
Number of contributing bricks	$\sim 10^{4}$

$$((10^{-3} \times 10^{-2} \times 10^{-4} \text{ square}) + (10^{-1} \times 10^{-1} \times 10^{-7} \text{ octagonal})) \times 10^{4} \sim 10^{-5}$$

From these examples it can be seen that there is considerable variation in relative damage stage contributions across assessment cases. Typically, however, the conditional clad melt probability has the most significant impact on the overall whole core clad melt probability and crack progression conditional probability has least impact. It may also be observed that as the likelihood of brick cracking increases the double initiation model of crack progression becomes more conservative relative to the crack propagation model. This is evident in the results shown in Figure B1.

B10 PESSIMISMS WITHIN THE PROBABILISTIC ASSESSMENT

Ideally within a probabilistic integrity assessment the probability distributions for all input parameters would be rigorously characterised without bias. In practice this is seldom achievable and the present assessment is no exception. Where input parameters cannot be fully defined (normally because of lack of data) bounding pessimistic substitutes have been adopted. Similarly, pessimisms have been accepted to make the analysis more tractable.

Recognised pessimisms have been highlighted within Section B5. Additionally, other pessimisms were identified within the Oldbury Assessment (Ref. B2) associated with choice of peak stress, effective stressed areas and measured weight losses. These pessimisms apply equally to the present Wylfa assessment.

Although it is not possible to quantify the overall influence of these pessimisms it is entirely credible that the predicted fuel clad melt probabilities may be conservative by several orders of magnitude. For example, the assumed conditional clad melt risks derived from the PREDICT2 analysis (Section B5.5.3) may alone be conservative by between one and three orders of magnitude in the middle layers of the core (those that contribute most to the overall clad melt probability). Moreover, comparison between predicted levels of brick damage and visual inspection of the Oldbury cores suggests that the brick cracking probability methodology yields predictions of initial crack development that are pessimistic by at least one order of magnitude (Ref. B2).

B11 CONCLUSIONS

The following conclusions have been drawn from the Wylfa probabilistic graphite integrity assessment with respect to reactor operation to September 2014.

a) The probability that fuel clad melt will occur as a result of moderator brick cracking during normal reactor operating conditions is negligibly low.

b) With best-estimate input parameter distributions the probability that fuel clad melt will occur during reactor operation following a hard shutdown transient is $<10^{-7}$ per hard trip.

c) A comprehensive range of sensitivity studies has been completed investigating the effects on the predicted fuel clad melt probabilities of changes to peak hoop stress, graphite flexural strength, postulated crack gape, weight loss and mean core irradiation.

   - Increases in peak hoop stresses or reductions in flexural strength result in increases in the overall whole core clad melt probability. That probability is more sensitive to changes in strength than changes in stress.
   - Increasing crack gapes result in increasing overall whole core clad melt probability.
   - The existence of bricks with higher than expected weight loss increases the overall whole core clad melt probability.
   - Overall whole core clad melt probability is slowly increasing with mean core irradiation over the remaining operating life of the reactors.

d) The risk of fuel clad melting as a result of moderator brick cracking for Reactor 1 bounds that for Reactor 2.

e) Moderator brick layers 6, 7, 8 and 9 make the highest contribution to the overall whole core risk of fuel clad melt.

f) Following a hard shutdown transient, significant damage to, of order, 10,000 moderator bricks would be expected before the risk of fuel clad melt is predicted to exceed 10^{-3}. Such levels of damage would be readily observable.

g) With identified pessimisms in the assessment it is considered credible that the predicted fuel clad melt risks are conservative by several orders of magnitude.

B12 REFERENCES

B1. Magnox North Report MEN/EWST/WYA/REP/0021/10, Probabilistic Assessment of Wylfa Graphite Core Integrity, [redacted] Issue 1 June 2010.

B2. NP/SC 4927 Revision 1 Addendum 4, Oldbury Power Station - A Revised Safety Case for the Integrity of the Graphite Cores to the Planned End of Generation: A Review of the Case for Both Reactors Including a Probabilistic Structural Integrity Analysis, [redacted] December 2008.

B3. NP/SC 4807 Addendum 5, Wylfa Power Station – Annual Review of the Safety Case for the Integrity of the Graphite Cores: 2009, [redacted] December 2009.

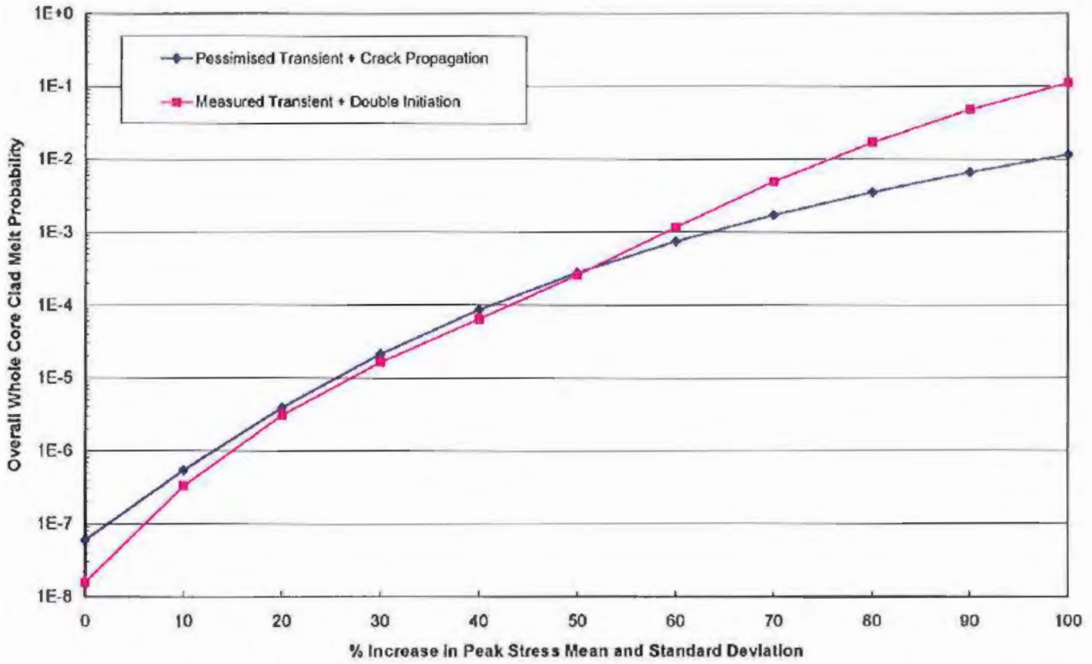


Figure B1: Peak Stress Sensitivity Study, Variation of Fuel Clad Melt Probability with Increase in Peak Fuel Channel Wall Hoop Stress (Reactor 1, 34.1 GWd/t)

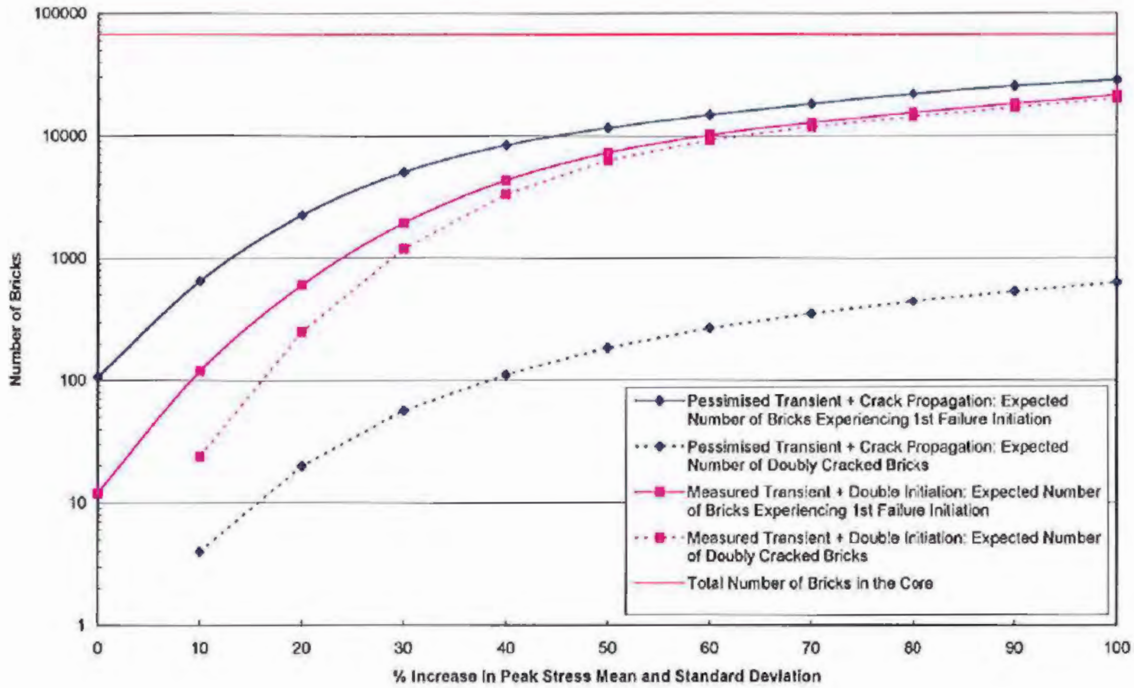


Figure B2: Peak Stress Sensitivity Study, Variation of Expected Number of Damaged Bricks in the Core with Peak Fuel Channel Wall Hoop Stress (Reactor 1, 34.1 GWd/t)

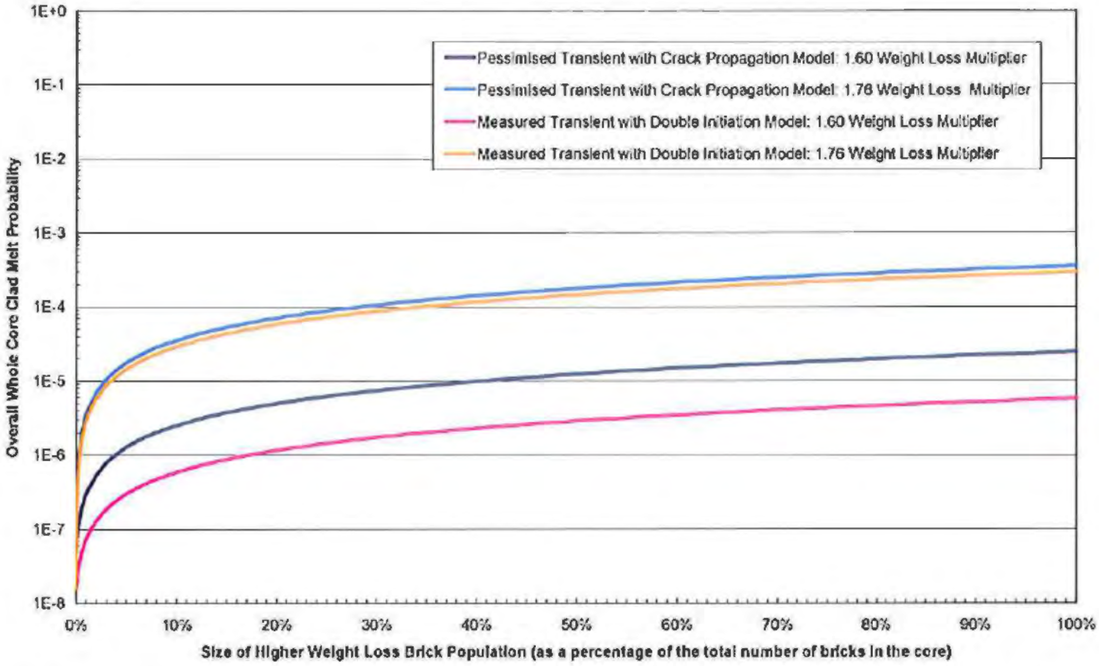


Figure B3: Weight Loss Sensitivity Study, Variation of Fuel Clad Melt Probability with Size of Higher Weight Loss Brick Population (Reactor 1, 34.1 GWd/t)

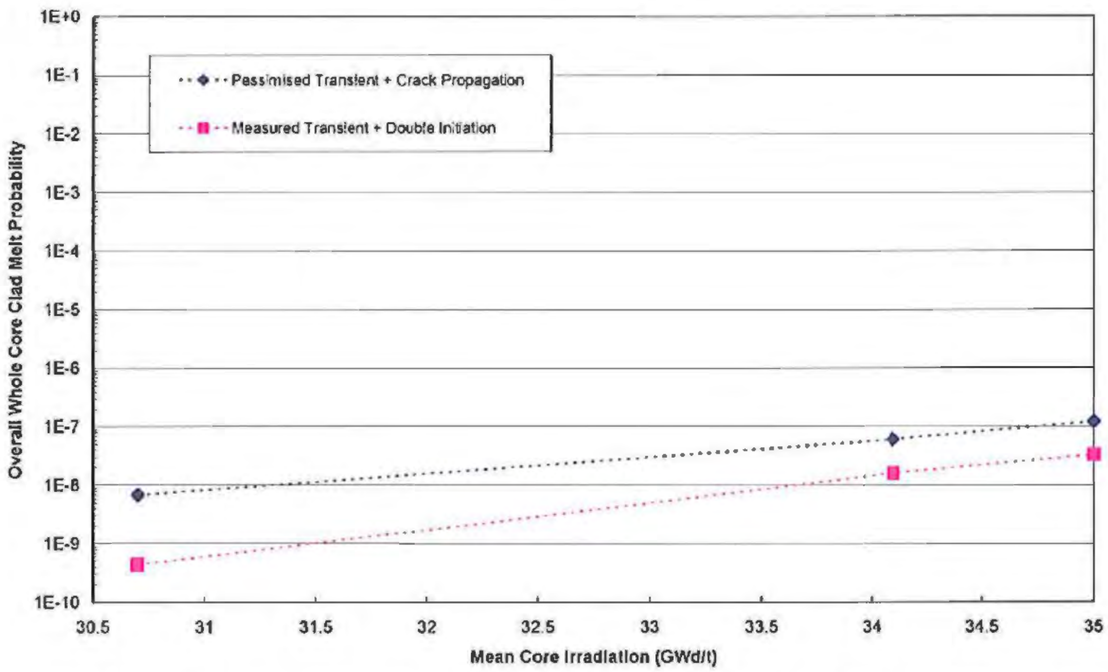


Figure B4: Mean Core Irradiation Sensitivity Study, Variation of Fuel Clad Melt Probability with Mean Core Irradiation (Reactor 1)

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