Operating Facilities – Operating Reactors

Fault Studies Assessment of the Hunterston B R4 Return to Service Safety Case Following Core Inspection Results in 2018

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EXECUTIVE SUMMARY

As the graphite core of an Advanced Gas-cooled Reactor ages the effects of neutron irradiation lead to stresses building up within the graphite bricks from which the core is constructed. Stress within the graphite bricks can lead to cracks developing, and as the core ages more cracks develop. Each Advanced Gas-cooled Reactor is periodically shutdown so that inspection of the graphite core can take place, and its state is determined. Hunterston B reactor 4 was shut down for its period inspection in October 2018, and EDF NGL have presented a safety case in order to justify operation of the reactor up to a core irradiation of 16.025 TWd.

EDF NGL has requested agreement from the Office for Nuclear Regulation under the arrangements made by the licensee under Licence Condition 22(1) to the modifications described in NP/SC 7785 Return to Service Safety Case for R4 Following Core Inspection Results in 2018.

Graphite brick cracking has the potential to challenge the ability of the graphite core to fulfil its safety functions, and therefore EDF NGL's safety case is focussed on demonstrating that the safety functions of the graphite core are fulfilled over the limited 4 month operating period proposed.

This fault studies assessment report has focussed on making a judgement on whether or not EDF NGL have adequately demonstrated that the safety functions of the graphite core will be fulfilled over the proposed 4 month operating period. Whilst this report has considered the effects of graphite brick cracking on all of the safety functions of the graphite core, the 2 safety functions with the greatest potential to be affected by graphite brick cracking are, allowing unimpeded movement of the fuel and control rods, and directing coolant flow to ensure adequate cooling of the fuel and core components; this report has therefore focussed on these 2 safety functions.

Allow unimpeded movement of the fuel and control rods

Cracking of the graphite core has the potential to increase the freedom of movement of the graphite components within the core and thus potentially lead to greater core distortion. The control rods and the fuel are inserted into the graphite core through channels in the graphite bricks and thus significant core distortion could lead to impeded movement of the control rods and the fuel as the channels become increasingly distorted.

The safety case presented by EDF NGL presented arguments and evidence to demonstrate that the movement of control rods and fuel would not be impeded in normal operation, in faults, or following a seismic event, during the proposed 4 month period of operation.

Consideration of the impairment of control rod movement in normal operation, and in seismic events has been made in other assessment reports on EDF NGL's safety case (NP/SC 7785). This assessment report considered the free movement of control rods following plant faults and concluded that EDF NGL has presented adequate arguments and evidence to demonstrate that the free movement of control rods will be maintained following plant faults, such that the shutdown function is not threatened.

EDF NGL concedes that there may be an increase in the risk of fuel becoming snagged whilst it is moved. This report examined a sensitivity study presented by EDF NGL which demonstrates that there are large margins to the risk target limits for dropped fuel caused by fuel snagging. This report additionally challenged EDF NGL to demonstrate that there were no further improvements which could be made to reduce the risk associated with fuel snagging. This report concluded that the arguments and evidence presented by EDF NGL adequately demonstrated that the risks associated with fuel snagging over the proposed 4 month operating period have been reduced so far as is reasonably practicable.

This report concluded that EDF NGL has demonstrated that the nuclear safety function of the graphite core to allow unimpeded movement of control rods and fuel, will be adequately fulfilled over the proposed 4 month operating period, and that EDF NGL has taken all reasonably practicable measures to reduce the risk associated with impaired movement of fuel and control rods.

Direct coolant flow to ensure adequate cooling of the fuel and core components

The safety case presented by EDF NGL presented arguments and evidence to demonstrate that the disruption to gas flow paths within the core would not significantly affect the cooling of fuel and core components such that temperature limits were threatened, during the proposed 4 month period of operation.

Increased cracking of the graphite bricks has the potential to change the gas flow paths within the core; this has the potential to reduce the cooling of core components and fuel. This report examined the arguments and evidence presented by EDF NGL, and concluded that it had been demonstrated that the changes to gas flow paths due to graphite brick cracking would not present a threat to the fuel and core component temperature operating limits over the proposed 4 month operating period.

Core distortion has the potential to impinge upon fuel stringers leading to gaps opening up between the sleeves of the fuel elements in the fuel stringer and disrupting the flow of coolant within the fuel stringer. This report examined the arguments and evidence presented by EDF NGL and concluded that the effects of sleeve gaps would not threaten fuel clad temperature limits within the proposed 4 month operating period.

Graphite brick cracking has the potential to produce debris, which could cause a partial blockage within a fuel stringer leading to impaired fuel cooling. This report has examined the evidence presented by EDF NGL and concluded that although EDF NGL did not demonstrate that all barriers to a radiological release were preserved, there was sufficient evidence to determine that the resultant mitigated risk was acceptable as assessed against ONR's risk targets, provided that the risks had been reduced so far as is reasonably practicable.

This report concluded that there is significant uncertainty associated with the point at which fuel clad melt would occur following a partial blockage in a fuel stringer, and made a recommendation in this regard:

Recommendation 1: For inclusion in future safety cases justifying the operation of the Hunterston B Reactor 4 graphite core, NGL should perform further analysis of the effects of a blockage at the element 1 support grid in order to establish the point at which fuel clad melt temperatures would be reached.

This report considered EDF NGL's analysis of whether there were any reasonably practicable improvements which could be made to reduce the risk associated with graphite debris, and concluded that EDF NGL has considered all potential measures to reduce risk, and demonstrated that there were no reasonably practicable measures which could be taken.

This report made a recommendation that EDF NGL should implement the technical specification changes proposed by the generic failed fuel safety case prior to the restart of reactor 4:

Recommendation 2: The changes to Technical Specification 8.1.3 proposed in NP/SC 7653 should be implemented at Hunterston B prior to restart of Reactor 4.

This report has considered the effects of graphite brick cracking and core distortion on the capability of the core to fulfil its safety function of directing gas flows to ensure cooling of the fuel and core. This report concluded that EDF NGL has presented adequate arguments and evidence to demonstrate that the safety function will be fulfilled over the proposed 4 month operating period.

Within EDF NGL's safety case consideration of cooling following a seismic event was limited to consideration of the integrity of the fuel sleeve, and did not consider the effects on fuel sleeve gapping. This report concluded that EDF NGL should include consideration of fuel channel distortions and hence fuel sleeve gapping following a seismic event in future graphite safety cases, and make a recommendation in this regard.

Recommendation 3: NGL should include consideration of fuel channel distortions following a seismic event and its effect on fuel sleeve gapping in future graphite safety cases.

Conclusion

This report concludes that EDF NGL has demonstrated that the nuclear safety functions of the graphite core to:

- allow unimpeded movement of control rods and fuel,
- to direct gas flows to ensure adequate cooling of the fuel and core,
- to provide neutron moderation and thermal inertia,

will be adequately fulfilled over the proposed 4 month operating period, and that EDF NGL has taken all reasonably practicable measures to reduce the risk associated with cracking of the graphite core.

Following this fault studies assessment of EDF NGL's safety case for the return to service of Hunterston reactor 4, this report concludes that ONR should agree to the modifications to the safety case described in NP/SC 7785 (Ref. 1) once Recommendation 2 of this report has been addressed.

LIST OF ABBREVIATIONS

AGR	Advanced Gas Cooled Reactor
ALARP	As low as is reasonably practicable
BCD	Burst Cartridge Detection system
BSL	Basic Safety Level
BSO	Basic Safety Objective
CFD	Computational Fluid Dynamics
CGOT	Channel Gas Outlet Temperature
DBA	Design Basis Analysis
EDF NGL	EDF Nuclear Generation Limited
FEAT	Finite Element Analysis Tool
HNB	Hunterston B Power Station
HPB	Hinkley Point B Power Station
JPSO	Justified Period of Safe Operation
KWRC	Keyway Root Crack
MCB	Multiply Cracked Brick
ONR	Office for Nuclear Regulation
PANTHER	PWR and AGR Neutronic and Thermal Hydraulic Evaluation Route
pa	per annum
pry	per reactor year
PSA	Probabilistic Safety Analysis
PVCW	Pressure Vessel Cooling Water
R#	Reactor number #
SAP	Safety Assessment Principle(s)
TAG	Technical Assessment Guide(s) (ONR)
VBA	Visual Basic for Applications

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1 INTRODUCTION

1. EDF Nuclear Generation Limited (NGL) has requested agreement from the Office for Nuclear Regulation (ONR) under the arrangements made by the licensee under Licence Condition 22(1) to the modifications to the safety case described in NP/SC 7785 Return to Service Safety Case for R4 Following Core Inspection Results in 2018 (Ref. 1).

2. This report presents the findings of the fault studies assessment of the Hunterston B Power Station (HNB) return to service safety case for reactor 4 (R4) following core inspection results in 2018 (Ref. 1) and supporting documentation provided by NGL. Assessment was undertaken in accordance with the requirements of the ONR How2 Business Management System guide NS-PER-GD-014 (Ref. 2). The ONR Safety Assessment Principles (SAP) (Ref. 3), together with supporting Technical Assessment Guides (TAG) (Ref. 4) informed this report. The methodology for the assessment follows How2 guidance on mechanics of assessment within the Office for Nuclear Regulation (Ref. 7).

3. The graphite moderator core in an Advanced Gas Cooled Reactor (AGR) is made up of fuel channel bricks and interstitial bricks. The fuel channel bricks form the pathways in which the fuel stringers are inserted, and the interstitial bricks form the pathways in which the control rods are inserted (Figure 1).

4. Over the lifetime of the reactor the neutron doses to the graphite bricks leads to stresses being induced within the fuel channel bricks which can lead to cracks forming. A particular type of crack in the fuel channel bricks called Keyway Root Cracks (KWRC) are the dominant form of cracking late in reactor life, and were first observed in the main population of fuel channel bricks at HNB in October 2015 in reactor 3 (R3) and September 2017 in reactor 4 (R4).

5. Since Keyway Root Cracking was observed the safety cases for the operation of the HNB and Hinkley Point B (HPB) graphite cores have been based on determining a Justified Period of Safe Operation (JPSO); this approach has been accepted by ONR for previous safety cases (Ref. 44). The JPSO is determined from knowledge of the state of the cores from core inspections, and predictions of the rate of future cracking, and a demonstration that the graphite cores can fulfil their safety functions at the predicted levels of cracking with adequate margin.

6. Cracking in fuel channel bricks means that the population of fuel channel bricks can be categorised as either intact bricks, singly cracked bricks, doubly cracked bricks, or multiply cracked bricks. No multiply cracked bricks have yet been observed in either reactor at HNB (or any other AGR), however fuel bricks have been observed which are close to being a multiply cracked brick (MCB) and which are likely to progress to being MCBs.

7. The HNB R3 graphite core inspections in March 2018 identified cracking in excess of that predicted, but within the demonstrated safety margin. HNB R3 remains shutdown whilst a revised safety case is developed which addresses the effects of multiply cracked bricks.

8. HNB R4 was shut down in October 2018 so that graphite core inspections could be performed. NP/SC 7785 Return to Service Safety Case for R4 Following Core Inspection Results in 2018 (Ref. 1) seeks to justify the return to service of R4 following the inspections of the graphite cores, and operation of HNB R4 up to a core burn up of 16.025 TWd which corresponds to a JPSO of 4 months of operation at 80% reactor power.

9. As well as providing neutron moderation the graphite bricks are designed to ensure unimpeded movement of the fuel stringers and control rods, and to direct the gas flow to ensure adequate fuel cooling. This fault studies assessment report has focussed on ensuring that the graphite core can adequately fulfil its safety functions with sufficient confidence in the presence of the predicted level of graphite brick cracking, including the potential effects of multiply cracked bricks (MCBs).

10. The safety case presented by EDF NGL (Ref. 1) does not propose any physical modifications to the plant, but simply seeks to justify that the aging effects of the graphite bricks does not challenge the capability of the reactor core to fulfil its nuclear safety functions.

2 ASSESSMENT STRATEGY

11. The intended assessment strategy for NP/SC 7785 Return to Service Safety Case for R4 Following Core Inspection Results in 2018 (Ref. 1) is set out in this section. This identifies the scope of the assessment and the standards and criteria that have been applied.

12. This fault studies assessment is focussed on the ensuring that NGL has presented an adequate safety case to justify that the nuclear safety functions of the graphite reactor core are maintained in the presence of graphite brick cracking over the next 4 month Justified Period of Safe Operation (JPSO).

13. This assessment report has considered the nuclear safety function of the graphite core to allow unimpeded movement of control rods and fuel in section 4.1. In this section I have assessed NGL's safety case for the unimpeded movement of control rods following plant faults and challenged them on the lack of treatment of core distortion following depressurisation faults. The movement of control rods in normal operation and following a seismic event have been assessed by an ONR graphite specialist, and an ONR civil engineering specialist. Section 4.1 of this report also considers NGL's safety case for the movement of fuel in the presence of graphite brick cracking.

14. Assessment of the requirement to direct gas flows to ensure adequate cooling of the fuel and core is contained in section 4.2. In this section I have assessed NGL's safety case for the effects of disrupted flow paths due to graphite brick cracking, the temperature effects of core distortion, and the effects of gaps in the fuel sleeves. I have also assessed the NGL's arguments and evidence seeking to justify the acceptability of the risks associated with graphite debris.

15. In section 4.3, I have briefly assessed the capability of the core to fulfil its requirement to provide neutron moderation and thermal inertia in the presence of graphite brick cracking.

2.1 Standards and Criteria

16. The relevant standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAP) (Ref. 3), internal ONR Technical Assessment Guides (TAG) (Ref. 4), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites. The key SAPs and any relevant TAGs are detailed within this section. National and international standards and guidance have been referenced where appropriate within the assessment report. Relevant good practice, where applicable, has also been cited within the body of the assessment.

2.1.1 Safety Assessment Principles

17. The key SAPs applied within the assessment are included within Table 2 of this report.

2.1.2 Technical Assessment Guides

18. The following Technical Assessment Guides have informed my assessment strategy (Ref. 4):

- ONR-TAST-GD-034 Transient Analysis for DBAs in Nuclear Reactors
- ONR-TAST-GD-075 Safety of Nuclear Fuel in Power Reactors
- ONR-TAST-GD-042 Validation of Computer Codes and Calculation Methods
- ONR-TAST-GD-005 Guidance on the Demonstration of ALARP

2.1.3 National and International Standards and Guidance

19. The following international standards and guidance have been used as part of this assessment (Refs 5, 6):

- IAES SSG-2, Deterministic Safety Analysis for Nuclear Power Stations
- IAEA SSR-2/1 Rev 1, Safety of Nuclear Power Plants: Design

2.2 Use of Technical Support Contractors

20. No technical support contractors have been used in support of this assessment.

2.3 Integration with Other Assessment Topics

21. Assessment reports have been produced on this safety case on the topics of Fault Studies (this report), Graphite, External Hazards, and Civil Engineering. There are no interfaces with this report and either the civil engineering or external hazards assessments. This assessment report has interfaces with the graphite assessment as follows:

- The reliability of the Primary Shutdown System is claimed to be unaffected by the potential core distortion due to graphite brick cracking in normal operation or in seismic events as the core distortion is not significant enough to impede control rod movement. The graphite assessment report has examined the predicted levels of core distortion in normal operation and following seismic events, and its effect on the freedom of movement of the control rods, and concluded that the claim can be supported (Ref. 12 & Ref. 45); as there is no impact on the reliability of the Primary Shutdown System there was no requirement for the assessment of the effects of a change in the reliability of the shutdown function in normal operation or following seismic events.
- The potential for graphite debris to partially obstruct the coolant flow through a fuel stringer has been considered in the safety case (Ref. 1). The graphite assessment (Ref. 45) has considered the likelihood of debris partially obstructing the coolant flow (as determined by NGL) while this fault studies assessment has considered predicted consequences, and how these, together with the likelihood, support the claims and arguments put forward by EDF NGL to support the JSPO.
- The potential for graphite debris to impede free movement of the fuel stringers during refuelling operations has been considered in the safety case (Ref. 1). The graphite assessment (Ref. 45) has considered the likelihood of the free movement of fuel stringers being impeded (as determined by NGL) while this fault studies assessment has considered predicted consequences, and how these, together with the likelihood, support the claims and arguments put forward by EDF NGL to support the JSPO.
- The potential for gaps to arise between fuel element sleeves due to core distortion or the effects of graphite debris has been considered in the safety case (Ref. 1). The assessment of the maximum size of the inter-element sleeve gap due to core distortion or graphite debris is covered in the graphite assessment report, and the assessment of the consequences of inter-element sleeve gaps is contained in this fault studies assessment report.

2.4 Out of Scope Items

22. This fault studies assessment report has focused on the potential effects of graphite cracking on the fuel including the implications in terms of fuel cooling, and free movement of the fuel in refuelling operations; these aspects are covered by Arguments 1.4 of the safety case (Ref. 1) and some aspects of Argument 2.2. Argument 1.3 of Reference 1 covers the free movement of control rods which has been discussed in

this report. All other aspects of the safety case are outside the scope of this assessment report, but are considered in other ONR assessment reports, primarily the graphite assessment report (Ref. 45). EDF NGL's safety case (Ref. 1) is for Hunterston (HNB) Reactor 4 (R4) only, and thus this assessment is limited to HNB R4 only.

3 LICENSEE'S SAFETY CASE

23. NP/SC 7785 Return to Service Safety Case for R4 Following Core Inspection Results in 2018 (Ref. 1) makes the following claims:

Claim 1: Graphite core degradation over the proposed JPSO will not undermine the required reliability of the Primary Shutdown (PSD) system for shutdown and holddown during normal operation and plant faults and the seismic hazard or prevent the graphite core from meeting its other fundamental nuclear safety requirements (fuel movement and fuel cooling).

Claim 2: All reasonably practicable measures have been taken in order to ensure that the risk associated with continued operation of HNB R4 is As Low As Reasonably Practicable (ALARP).

24. The structure of EDF Nuclear Generation Limited's (NGL) safety case (Ref. 1) is laid out below with a summary of the supporting evidence. The level of detail I have provided for each argument below is proportionate to the relevance to this fault studies assessment report. Some topics are discussed in greater detail in Section 4 of this report where I have assessed the arguments and evidence presented.

25. **Claim 1: Graphite core degradation over the proposed JPSO will not undermine the required reliability of the Primary Shutdown (PSD) system for shutdown and holddown during normal operation and plant faults and the seismic hazard or prevent the graphite core from meeting its other fundamental nuclear safety requirements (fuel movement and fuel cooling).**

26. **Argument 1.1**: The current state of the core has been conservatively established through recent reactor monitoring and inspections.

27. NGL argues that the ongoing core monitoring arrangements provide evidence that the condition of the graphite core is within expectations.

28. NGL presents the results of the R4 2018 graphite core inspections and compares them with the pre-inspection expectations to conclude that R4 remained within the operational allowance on the number of cracked bricks during the previous operating period. Argument 1.1 additionally presents evidence that the opening of cracks was within expectations.

29. **Argument 1.2**: The predicted state of the core at the end of the proposed JPSO has been conservatively established.

30. NGL argues that the mechanisms leading to the development of fuel brick cracks are well understood, and that the predictions of the number and size of cracks are consistent with observations. NGL states that the number and size of the fuel brick cracks in R4 are sufficiently similar to those observed in R3 such that the crack morphologies can be considered equivalent, and that the future crack opening rate can be bounded with high confidence.

31. NGL gives the predicted number of multiply cracked bricks (MCBs) currently in the R4 core at HNB as fewer than 15 in the main population to a 99.9% confidence, based on the observations of the latest inspection campaign. Noting that the main population excludes the population of 30 high shrinkage bricks which may be more prone to cracking.

32. NGL states that the number of MCBs predicted at the end of the proposed 4 months justified period of safe operation (JPSO) is 3-10, and that when taking account of

uncertainties the 99.9th percentile prediction is of 30-40 MCBs, with a worst case scenario given as 50 MCBs including the contribution from high shrinkage bricks.

33. **Argument 1.3**: At the end of the proposed JPSO, core distortion will not prevent successful insertion of the control rods during normal operation, plant faults or following a seismic event.

34. NGL states that the extent of core distortion is related to the number of cracked bricks of the different types (singly, doubly, or multiply cracked bricks), the extent of crack opening, and the strength and clearances of the graphite components.

35. The results of NGL's core distortion modelling are presented for a range of brick cracking widths and numbers of cracked bricks beyond that predicted at the end of the JPSO, and at a range of future core ages, with the most onerous outcome not predicted to impede control rod movement.

36. The results of a review of fault conditions with the potential to effect core distortion are presented. The review concludes that faults in which core distortion may occur due to loss of cooling are not a concern as the control rods would already have inserted by the time the temperatures were such that core distortion may occur. The review states that the faults which could lead to core distortion prior to the insertion of control rods are depressurisation faults, and a seismic event.

37. NGL states that the only depressurisation fault which could lead to core distortion is a failure of a standpipe as these are the only pressure vessel penetrations with a direct path for the resulting pressure wave to reach the core. NGL state that the failure of the largest standpipe penetration – a fuel channel standpipe – is beyond the design basis as the structural weld meets the standard for an incredibility of failure item. NGL additionally states that failure of the smaller interstitial channel standpipes would not lead to consequences on core distortion beyond the failed interstitial channel, and would thus not affect the reliability of the primary shutdown system.

38. NGL has employed an updated reactor building model to support the assertion that the core distortion in a seismic event would not impede control rod movement with significant margin.

39. NGL states that the number of cracked bricks modelled in the determination of the core distortion in normal operation, faults and following a seismic event were greater than the number of cracked bricks predicted at the end of the JPSO, providing safety margin in the analysis.

40. **Argument 1.4**: At the end of the proposed JPSO, core distortion will not prevent the graphite core from meeting its other fundamental nuclear safety requirements in relation to fuel movement and fuel cooling.

41. NGL argues that the potential for fuel channel distortion to lead to gaps between the fuel element's sleeves being opened up (sleeve gapping) has been analysed and shown to result in potential increase in the fuel clad temperature of less than 30°C when the reactor is operating at full power for sleeve gapping of 7mm. NGL additionally argue that as the reactors are operated at a maximum of 80% power there is significant margin to the clad temperature limit such that a 30°C increase can be tolerated. NGL state that analysis of the predicted core distortion during the JPSO gives a maximum sleeve gap of <1mm and that the risks associated with sleeve gapping are therefore tolerable.

42. NGL states that the impact of a 30°C increase in the clad temperature in channels with sleeve gapping present on the fission gas pressure safety case is insignificant. NGL argues that even if the fuel clad temperature limit were exceeded in some channels

due to sleeve gapping (which is not predicted) the fission gas pressure safety case is based on the whole core distribution of fission gas pressures and that the presence of a few high fission gas pressure fuel pins would not undermine the assumptions of the safety case.

43. NGL states that the effect of sleeve gapping on the probability of failure of the tie bar is small as the effect on the coolant gas is small and therefore that the impact on the tie bar temperature is small.

44. NGL states that the largest load induced on any fuel sleeve due to core distortion is significantly bounded by the strength of the fuel sleeve, and that as such there is no risk of failure of the fuel sleeve. NGL additionally argues that in the event of a piece of graphite debris becoming lodged in the annulus between the fuel stringer and the fuel channel wall the maximum amount of sleeve gapping which could arise is <2mm, leaving margin to the 7mm gap which is argued to be tolerable.

45. NGL states that the change in the re-entrant flow paths through the graphite cores due to the presence of significant quantities of cracks has been determined to have no safety significant effects, and that large margins to all temperature limits have been shown to be maintained.

46. NGL argues that the most significant concern relating to graphite debris is the potential for debris to become lodged at the fuel element support grid (Figure 2), and that 16.5% of the flow area of the stringer would need to be blocked before the fuel clad temperature limit was reached. NGL states that it is not considered credible for more than 16.5% of the flow area to become blocked by graphite debris.

47. The results of analysis of the effects of a seismic event on the fuel sleeve are presented, with large margins to the initiation of fuel sleeve cracking and sleeve gapping given.

48. It is conceded that pieces of graphite debris could be produced as a consequence of multiply cracked bricks, and that should they be large enough, these items of debris could impede fuel movement during raising of the fuel (snags) or lowering of the fuel (ledges). NGL however argues that a significant change to the probability of snags or ledges is not expected over the proposed JPSO, and that even if it was, a sensitivity study demonstrates that an order of magnitude increase in the probability is tolerable.

49. Argument 1.4 presents a summary of the core restraint safety case which justified the operation of the core restraint system up to a core burnup of 17.1 TWd which is stated to be beyond the proposed JPSO and therefore already bounding.

50. **Argument 1.5**: The R4 core state (in terms of the level of cracking) at the end of the 4 month operating period (16.025 TWd) is expected to remain within the current R3 core state (16.185 TWd).

51. NGL states that the results of the 2018 graphite core inspections in R3 and R4 show that there are fewer cracks in R4 than R3, with 319 cracks predicted in R4 to 99.9% confidence, and 505 cracks predicted in R3 to 99.9% confidence. NGL additionally states that forecasts on the number of cracked bricks predict that the number of cracks in R4 at the end of the proposed 4 month JPSO is also lower than the number of cracks currently in R3.

52. Claim 2: All reasonably practicable measures have been taken in order to ensure that the risk associated with continued operation of HNB R4 is As Low As Reasonably Practicable (ALARP).

53. **Argument 2.1**: The decision to inspect and the extent of the inspections provides the necessary confidence in the core state at the end of the JPSO.

54. NGL states that the decision to inspect the core in 2018 was taken to provide an updated understanding of the state of the core following the prior inspections in late 2017. NGL argues that the 34 channels inspected were more than sufficient to allow extrapolation to the entire core state.

55. **Argument 2.2**: All reasonably practicable measures have been taken to reduce the risk associated with return to service of HNB R4.

56. NGL argues that modification of the Enhanced Shutdown System to provide protection against the potential for core distortion to impede the movement of control rods would require a reduction in the reliability of the Primary Shutdown System (PSD) as the control rod drive mechanism would need to be redesigned in such a way that it no longer failed safe upon loss of electrical supplies.

57. NGL states that no credit has been taken for the Super-Articulated Control Rods enhanced ability to insert into a distorted core, and that they provide additional benefit in terms of reliability of the PSD.

58. NGL argues that modification of the Nitrogen system to provide reactor shutdown capability rather than just reactor hold-down as at present would require significant modifications to the reactor core and pressure vessel, which would not be reasonably practicable.

59. NGL states that there are no reasonably practicable measures which could be taken to reduce the risk associated with fuel handling, and that it is not reasonably practicable to maintain Hunterston R4 in a shutdown state.

60. **Argument 2.3**: Monitoring techniques will be employed within the proposed JPSO to confirm that core degradation remains within the basis of this proposal.

61. NGL states that the use of Fuel Grab Load Traces (FGLT) provides ongoing monitoring of the fuel channel distortions, and that significant changes to the fuel channel bore will be detected.

62. NGL argues that the movement of the regulating control rods within the core during normal operation and the control rod drop tests performed on reactor shutdown provide ongoing monitoring of the freedom of control rod movement.

63. NGL state that comparisons between the calculated neutronic channel powers and thermal channel powers will reveal significant disruptions in the coolant flow due to sleeve gapping.

4 ONR ASSESSMENT

64. This assessment has been carried out in accordance with HOW2 guide NS-PER-GD-014, “Purpose and Scope of Permissioning” (Ref. 2), and NS-PER-GD-015, “Guidance on Production of Reports” (Ref. 41).

65. This fault studies assessment is focussed on the ensuring that NGL has presented an adequate safety case to justify that the nuclear safety functions of the graphite reactor core are maintained in the presence of graphite brick cracking over the next 4 month Justified Period of Safe Operation (JPSO).

66. The Nuclear safety functions of the graphite reactor core as stated by NGL are:

- Allow unimpeded movement of control rods and fuel,
- Direct gas flows to ensure adequate cooling of the fuel and core,
- Provide neutron moderation and thermal inertia.

I agree that these are the fundamental nuclear safety requirements on the graphite core, and thus my expectations are that NGL adequately demonstrate that these functions are fulfilled over the next 4 month operating period in accordance with SAPs FA.4 & ERC.1, and that there is no cliff edge beyond this period as per SAPs FA.7 & ERC.3.

67. I will structure this report into sections focussed on each of the nuclear safety requirements on the graphite core.

68. Argument 1.3 of NGL’s safety case (Ref. 1) presents an argument and evidence in support of the assertion that core distortion will not prevent successful insertion of the control rods at the end of the JPSO in normal operation, faults, or following a seismic event. I discuss this section of NGL’s safety case in section 4.1.1 of this report.

69. NGL has presented evidence in support of the assertion that the other nuclear safety functions of the graphite core are fulfilled in Argument 1.4 of the safety case (Ref. 1), and therefore this section of the safety case is the focus of the majority of this assessment report (Sections 4.1.2, 4.2, & 4.3).

70. Argument 2.2 of NGL’s safety case states that all reasonably practicable options to reduce risk have been taken; this assessment report has considered the arguments and evidence presented in this section of the safety case as it relates to the discussion in sections 4.1.1, 4.1.2, & 4.2.3.

71. Argument 2.3 of the safety case states that monitoring techniques such as control rod freedom of movement tests and analysis of the channel power discrepancies provide confirmation that the core degradation remains within the bounds of the safety case assumptions. This claim is not a change over the existing safety case (Ref. 9), and I have therefore not focussed my assessment in this area. Assessment of this area was undertaken in the assessment report on the previous graphite safety case (Ref. 40), and I judge that the conclusions of that assessment report remain valid.

72. The scope and structure of this assessment report is as follows:

- Assessment of the requirement to allow unimpeded movement of control rods and fuel (Section 4.1)
    - Control rod movement
    - Fuel movement

- Assessment of the requirement to direct gas flows to ensure adequate cooling of the fuel and core (Section 4.2)
  - The effects of changes in coolant flow paths due to cracking
  - The effects of channel distortion – eccentric annulus
  - The effects of channel distortion – sleeve gapping
  - The potential effects of debris
- Assessment of the requirement to provide neutron moderation and thermal inertia (Section 4.3)
  - The potential effects of brick cracking on the neutron flux distribution

73. Previous graphite safety cases have been assessed by ONR, and the latest fault studies assessment (Ref. 40) was undertaken on NGL's safety case for operation following the onset of keyway root cracking - NP/SC 7716 (Ref. 9).

74. Reference 40 assessed the arguments and evidence presented by NGL seeking to demonstrate that the nuclear safety function of the graphite core to direct gas flows to ensure adequate cooling of the fuel and core was fulfilled following the onset of keyway root cracking. Reference 40 focused on the effects of graphite brick cracking on the coolant flow paths, the effects of gaps opening between the graphite sleeves, and briefly considered the potential effects of graphite debris. Reference 40 concluded that NGL had adequately demonstrated that the core would be able to fulfil its nuclear safety function up to the operational limits on the number of graphite brick cracks specified in the safety case (Ref. 9).

75. In this report I re-examine the areas assessed in Reference 40 in the context of increased graphite brick cracking and the potential for multiply cracked bricks to occur. The main area in which my assessment differs from that of Reference 40 is that the potential for debris to be produced by graphite brick cracking has increased, and thus I place a greater focus on this area.

4.1 Allow unimpeded movement of control rods and fuel

76. Cracking of the graphite core has the potential to increase the freedom of movement of the graphite components within the core and thus potentially lead to greater core distortion. The control rods and the fuel are inserted into the graphite core through channels in the graphite bricks and thus significant core distortion could lead to impeded movement of the control rods and the fuel as the channels become increasingly distorted.

4.1.1 Control rod movement

77. Should the freedom of movement of the control rods be significantly impeded then there is the risk that insufficient numbers of control rods enter the core on demand and that the reactor is not shutdown, this then would lead to a large offsite release. Normal operating procedures for the AGRs require a demonstration that the shutdown function will be secured even in the event of the most onerous two control rods failing to insert, however due to the uncertainties associated with predictions of core distortion my expectation is that NGL should demonstrate that no control rods are impeded due to core distortion in any operating state or fault scenario in accordance with IAEA SSR-2/1 such that the reliability of the shutdown function is not affected.

78. NGL's safety case (Ref. 1) seeks to demonstrate that my expectations in this regard are met with Argument 1.3 stating that, at the end of the proposed Justified Period of

Safe Operation (JPSO), core distortion will not prevent successful insertion of the control rods during normal operation, plant faults or following a seismic event.

79. Changes were made to the seismic building model in support of the safety case which reduced the predicted forces on the graphite core during a seismic event. The changes to the building model have been assessed by an ONR civil engineering specialist inspector who concluded that the changes adequately justified (Ref. 13 & 46).

80. The revised seismic input data was used to produce predictions of the core distortion during a seismic event; the results of these predictions have been assessed by an ONR graphite specialist inspector along with predictions of core distortion in normal operation. The ONR graphite inspector judged that the conclusions of NGL's predictions of control rod channel distortions – that no control rod would be impeded in normal operation or seismic events – could be supported (Ref. 12 & 45).

81. NGL identified the set of plant faults with the potential to lead to core distortion (Ref. 11) and concluded that none of the faults with the potential to lead to core distortion would lead to significant levels of core distortion until after the control rods had already inserted into the graphite core, as the core distortion was driven by thermal expansion/contraction effects.

82. In my view the conclusions of Reference 11 are supportable, however I challenged NGL that the potential for a depressurisation fault to lead to core distortion prior to the insertion of control rods had not been considered (Ref. 10).

83. In response to my challenge NGL asserted that the only pressure vessel penetrations with the potential to affect core distortion were the top cap penetrations as these are the only penetrations with a direct path to the graphite core (Ref. 10). NGL stated that largest top cap penetration not classed as Incredibility of Failure items are the control rod standpipe with a failure probability $<1 \times 10^{-5}$ pry (per reactor year). NGL stated that no explicit analysis of the consequences for core distortion of a control rod standpipe failure is available, but that work is in progress. NGL presented the arguments and evidence discussed below to justify that shutdown would be secured following a depressurisation fault.

84. NGL argue that the interstitial bricks are not cracked and are sufficiently robust such that they would not crack due to high differential pressures, and as such the control rod channel distortion would remain small. NGL additionally argues that the constraint of the surrounding bricks would prevent the distortion spreading to other control rod channels and as such core shutdown would not be threatened.

85. I have discussed the potential effects of a depressurisation fault with an ONR graphite specialist inspector. It is the view of the graphite specialist inspector that test rig evidence suggests control rod insertion, albeit with some resistance, is still possible at the levels of distortion that might be conservatively expected from such a depressurisation fault (Ref. 33), and that as such NGL's position (Ref. 10) can be supported.

86. NGL provided details of the flow resistances associated with the various gaps and orifices in control rod channels (Ref 36), and argue that the extent of neighbouring channels affected would be limited (Ref. 10). In my opinion, the flow resistances presented in Reference 36 demonstrate that the flow due to the depressurisation would preferentially enter the control rod channel at the bottom of the core or above the core, rather than through the side of the channel (see figure 4). For every additional channels distance from the fault channel the flow resistance laterally through the core would significantly increase due to the restricted flow path through the core, and thus it is likely that the flow would preferentially take the path of least resistance to

the top or bottom of the core and through the low flow resistance routes to the depressurisation. Therefore, in my view NGL have demonstrated that any significant lateral forces on neighbouring channels would only occur over a limited distance from the channel which suffered the fault, and thus the extent of any resulting core distortion would be limited.

87. Based on the evidence, and logical arguments provided by NGL I conclude that it is likely that following a depressurisation fault in an interstitial channel no control rods would have their movement impaired. I also judge it likely that even if some control rods did have impaired movement that only a small number would be affected such that reactor shutdown was not threatened, as reactor operations are managed such that failure of the most onerous two control rods to insert would not threaten reactor shutdown. I note that NGL is currently undertaking technical work to examine the effects of a depressurisation fault, which should be available before the next safety case for HNB R4. In my view the evidence presented is sufficient to justify the position for the next 4 months of operation as it would be grossly disproportionate to withhold permission to restart the reactor until the further technical work is complete. In my view NGL should ensure that the technical work examining the effects of a depressurisation fault on core distortion is completed and available for inclusion in any future safety cases for the operation of HNB Reactor 4, and I have raised a regulatory issue to track the progress of this work (issue 7300).

88. I judge that NGL has met my expectations that there be no impediment to the free movement of control rods as per IAEA SSR-2/1 Rev 1 Requirement 44 (Ref. 6), and that as such Argument 1.3 can be supported. As NGL have demonstrated that there would be no impediment to the free movement of control rods, there is also no change to the capability of the control rods to fulfil the shutdown function, and no change to the level of shutdown margin. I therefore judge that the expectations of SAP ERC.2 have been met.

89. As the effects of graphite brick cracking on core distortion are such that freedom of movement of the control rods is unaffected, the reliability of the Primary Shutdown System (PSD) is also unaffected and there was no need for a fault studies assessment of the effects of a change in the reliability of the shutdown function.

90. NGL has considered potential improvements to existing shutdown systems which could be made in order to improve the reliability of the shutdown function; such as increasing the number of super articulated control rods, or installing a fast acting nitrogen shutdown system in place of the existing nitrogen hold-down system. Reference 1 concludes that there are no reasonably practicable improvements which can be made.

91. In my opinion NGL has adequately demonstrated that there will be no reduction in the reliability of the PSD due to brick cracking or core distortion; modifications to the PSD aimed at mitigating the effects of core distortion, such as increasing the number of super articulated rods, would therefore not improve the PSD reliability and not provide much safety benefit.

92. Whilst modifications to the Nitrogen injection system in order to enhance its capabilities such that it could act as a shutdown system would provide a safety benefit, it would be limited as Reference 1 has demonstrated that the PSD reliability is unaffected following a seismic event. Additionally the modifications required would be extensive and require modifications inside of the pressure vessel, and potentially the reactor core, likely leading to significant operator dose and making them highly impracticable.

93. Having considered the potential improvements which could be made to the reliability of the shutdown function in the presence of brick cracking and core distortion I concur

with NGL's judgement that there are no reasonably practicable improvements which could be made to reduce the risks associated with failure to shutdown.

4.1.2 Fuel movement

94. Graphite brick cracking has the potential to increase the core distortion which may impede the movement of fuel, additionally debris produced due to graphite brick cracking could cause an obstruction and lead to fuel snags or ledges. If fuel were to become stuck during fuel movement then there is an increased probability of the fuel being dropped which would likely lead to fuel damage, and a release of radioactive isotopes into the primary circuit.

95. In the submission (Ref. 1), NGL states that the fuel handling safety cases (covering low power refuelling, pressurised & depressurised offload refuelling and radial shuffling) remain valid over the proposed 4 month operating period. This section of the report records my assessment of the evidence supporting this statement. During the four month period, NGL intends to carry out one low power refuelling campaign.

96. If a snag were to occur, there are design basis measures already in place in the extant safety case to prevent any radiological consequences being realised as a result of a snag. On the charge machine there are two diverse lines of engineered protection to trip the hoist (and reactor if on-load) on detection of an increase or reduction of load beyond pre-set limits; there are also procedures which instruct the operator in recovery of the fuel. The effectiveness of these measures has not changed as a result of this safety case. This is in line with SAP FA.8, linking of initiating faults, fault sequences and safety measures.

97. NGL does not expect an increase in snag frequency over the 4 month JPSO. Based on assessment of NGL's arguments and evidence of core distortion, production of debris and fragments over the four month operating period and the potential effect of these factors on fuel movement, the ONR graphite specialist inspector has concluded that it is unlikely that there will be a significantly increased challenge to fuel movement during the next four months of operation, supporting NGL's position (Refs. 25, 26 & 45). However recognising that there is uncertainty attached to this judgement, NGL has carried out a sensitivity study (Ref. 1, Appendix A), assessing the increase in risk if the snag frequency were to increase by an order of magnitude (from 0.15 snags per year to 1.5 snags per year). The Probabilistic Safety Analysis (PSA) gave the following risk figures:

Dose Band	Fuel Route Upper Tolerable Limit, pry	Current Fuel Route Risk, pry	Fuel Route Risk with 10x Snag Frequency, pry
DB1	5.00E-01	1.49E-02	2.02E-02
DB2	5.00E-02	5.25E-03	7.75E-03
DB3	5.00E-03	1.94E-03	2.23E-03
DB4	5.00E-04	5.10E-05	1.02E-04
DB5	5.00E-05	3.75E-05	4.91E-05

Table 1 – The sensitivity of fuel route risk to fuel snag frequency (Ref. 1)

(DB1: 0.1-1mSv, DB2: 1-10mSv, DB3: 10-100mSv, DB4: 100-1000mSv, DB5: >1000mSv)

98. Here the Fuel Route Upper Tolerable Limit quoted is calculated as 50% of the Upper Tolerable Limit, defined in NGL's Nuclear Safety Principles (the Upper Tolerable Limit in the NSPs is equivalent to the Basic Safety Level (BSL) as defined in the SAPs Numerical Target 8). For single faults the frequency targets given in SAP NT.8 are generally reduced by an order of magnitude in accordance with paragraph 749 of the SAPs (Ref. 3). However, the frequencies given in table 1 cover all fuel route faults and not just a single fault group. Paragraph A12 of the SAPs (Ref. 3) state that the

frequencies given in NT.8 may need to be adjusted to reflect a reduced scope PSA. In my view the use of 50% of the SAP NT.8 targets to cover the entire fuel route is therefore appropriate. I additionally note that the analysis discussed here is a sensitivity study, and simply aims to demonstrate that uncertainty in the snagging frequency can be tolerated, and thus the specific level of the risk target is less significant.

99. The figures presented in Table 1 indicate that even if the judgements above were significantly wrong (to an order of magnitude increase in snag frequency) the overall fuel route risk would increase for each dose band but remain below the basic safety level. I note additionally that NGL states that the fuel route PSA contains a number of conservatisms and therefore the true summated risk is likely to be less than stated in Table 1; however I have not looked at the detail of this. The use of PSA to inform the safety case on the sensitivity of fuel route risk to snag frequency is in line with SAPs FA.14, use of PSA and AV.6, sensitivity studies.

100. Recognising that there could be an increase in risk but likely not by a factor of 10, I judge that the risks are acceptable, providing any reasonably practicable measures have been implemented. This is discussed further below.

101. From a safety cultural perspective, NGL states that “multiple snagging events are not and would not be tolerated”. Additionally, should a snag occur, there would be significant investigation and scrutiny to determine the cause. The case also states that NGL would inform INA and ONR in the event that a snag occurs.

102. In addition to the design basis protection measures, fuel grab load traces are reviewed to monitor condition of the core. NGL is developing procedures to trend load trace data in order to provide forewarning of increased snagging, however these measures will not be in place for this 4 month period of operation. I queried whether this timescale could be moved up (Ref. 10) and NGL stated that the process is still in the early stages of development and is not yet in a position to be implemented, but that it is intended to trial the new procedure over the next period of operation and to implement it fully by the end of 2019. Regulatory issue 6765 had already been raised in order to track developments in fuel grab load trace monitoring/trending. Since there is only one low power refuelling planned for the 4 month JPSO and NGL are planning to pilot the new process during this refuelling, I do not think it would be appropriate to delay return to service to wait for the final process to be in place.

103. NGL identified a number of risk reduction measures but they were deemed not to be reasonably practicable. The most significant of these was the suspension of low power refuelling (LPR) operations, replacing these with offload pressurised refuelling (OPR) operations. I queried the basis for this judgement since there was no supporting substantiation in the case (Ref. 10). In response, NGL carried out a PSA study through which they identified that the resulting reduction in dose band 5 frequency in the fuel route PSA is comparable in size to the resulting increase in dose band 5 frequency in the reactor PSA. The reduction in risk in the fuel route PSA is due to the removal of “failure to trip” and “failure of post trip cooling” fault sequences since during OPR operations the reactor would already be tripped and post trip cooling operable. The increase in risk in the reactor PSA is due to the increased number of planned reactor trips per year, reactor trips carry an inherent risk due to the challenge placed on the reactor protection systems. NGL stated that OPR is the less familiar process to operators since it is carried out less frequently. Considering the above, overall it is not clear that any significant safety benefit would be realised as a result of switching from low power to offload refuelling operations.

104. I am content that NGL has sufficiently addressed the risk of snags during the four month operating period. I base this judgement on the following:

- The fuel snag frequency is unlikely to increase significantly over the four month period of operation; this judgement is supported by the assessment by the ONR graphite specialist inspectors (Refs. 25, 26 & 45).
- If this judgement were misjudged and the snag frequency increased by a factor of 10, the station risk would remain within the basic safety level in Target 8 of the SAPs.
- Design basis measures are already in place to prevent escalation of a fuel snag to a radiological release.
- Further risk reduction measures have been considered with no further measures found to be reasonably practicable within the four month period. Noting that the development of improved fuel grab load trace monitoring is ongoing and will be tracked by ONR.
- Culturally, fuel snags due to the graphite core would not be accepted and should one occur there would be significant scrutiny from both NGL and ONR.

105. I queried with NGL the potential effect of debris on sub-veto height faults. If debris falls onto the support stool it may prevent correct seating of the stringer during fuel charging when the hoist underload protection is vetoed (Ref. 10). NGL stated that if debris were on the support stool, it would likely be crushed due to the mass of the fuel assembly. If it were to cause the assembly to seat incorrectly, this would be detected and if it couldn't be rectified by a lift and reseat manoeuvre the assembly would be removed from the channel for investigation. This is already captured in existing procedures.

106. NGL additionally states that the primary concern with "false bottoming" is the fuel assembly being hung up and then dropping undetected onto the support stool. In this case the maximum drop height (6cm) is insufficient to damage the fuel, i.e. has no radiological consequences. I am therefore content that the risk due to debris on sub-veto height faults is small and there are suitable procedures in place to correct any issues.

107. NGL identified that graphite weight loss could cause increased fast neutron dose to the fuel sleeves resulting in shrinkage to the sleeves, this could potentially increase the hangman's drop distances in a ledge and release fault. To counter this, NGL has written a safety case to reduce the irradiation limit for fuel shuffling. I have confirmed that this safety case has been implemented; incorporating the new operating limit into the appropriate technical specification. I am therefore content that this risk has been satisfactorily addressed.

108. Reference 34 presents consideration of the potential for the tie bar guide tube to become blocked due to carbon deposition on the tie bar, and possibly due to the effects of graphite dust. The analysis presented in Reference 34 concludes that if it is assumed that all of the fuel stringers in the core have blocked tie bar guide tubes then the probability of a dropped fuel stringer increases by between 50 and 60 times.

109. Table 1 examines the effects of a 10 fold increase in the snag frequency, which is closely related to the dropped fuel frequency. The largest increase occurs in dose band 4 in which the risk is doubled upon a 10 fold increase in snag frequency. If a further 5 fold increase is assumed then this can be assumed to lead to a further 50% increase in risk, then from the Table 1 it can be seen that the risks are below the Target 8 BSL for all dose bands.

110. In my view it is extremely conservative to assume that all of the stringers in the core have blocked tie bar guide tubes, as no blocked guide tube has been observed, and Reference 34 reports the most severely deposited tie bar observed as having significant margin before the guide tube would be blocked. The bounding analysis in Reference 34 is for other AGR power stations, and the report concludes that the results for Hunsterston B would be much less significant due to the lower refuelling power, and lower levels of carbon deposition.

111. This discussion is only relevant to this report and Reference 1 as there is the potential for graphite cracking to lead to an increase in the amount of graphite dust in the primary circuit which could be available to cause a blockage of the tie bar guide tube. I have discussed the potential increase in graphite dust with an ONR graphite specialist inspector (Ref. 37 & Ref. 45 Section 4.6.4) who has advised that the increase in the amount of dust in the primary circuit would be small compared to the existing dust load.

112. In conclusion I judge that complete blockage of a tie bar guide tube is unlikely and that there is no increase in the likelihood of a blockage due to the increase in graphite brick cracking. As a sensitivity study, if an extremely conservative assumption is made that all of the tie bar guide tubes are blocked, and no account is taken of the likely improved position at HNB, then the risks are of the order of the Target 8 BSL. I judge that the potential for blockage of the tie bar guide tube has no effect on my regarding the validity of the fuel handling safety case over the proposed 4 month operating period.

113. Given the information in the case (Ref. 1) and further evidence provided in response to queries, I conclude that the fuel handling safety cases remain adequate over the proposed 4 month operating period.

4.1.3 Section Summary

114. Argument 1.3 of Reference 1 states that core distortion will not affect the insertion of control rods, and Argument 1.4 of NGL's safety case (Ref. 1) states that the safety functions of the graphite core will be fulfilled in the presence of graphite brick cracking. In this section (4.1 and sub-sections) I have examined the evidence that NGL has presented in support of demonstrating that the safety function of allowing unimpeded movement of the control rods and fuel has been met.

115. In coordination with the ONR graphite specialist inspector I have assessed the evidence presented by NGL to demonstrate that control rod movement is not impeded following faults, and concluded that this argument can be supported. As the ONR graphite specialist inspector concluded that NGL have adequately demonstrated the control rod channel distortion would not impeded control rod movement in normal operation, it was not necessary for me to assess the effects of a degraded shutdown function.

116. The ONR graphite specialist inspector in coordination with the ONR civil engineering inspector concluded that there would be no impairment of control rod movement during or following a seismic event. I therefore conclude that Argument 1.4 of Reference 1 as it relates to the movement of control rods in normal operation, faults, and the seismic hazard, can be supported. I additionally concluded that NGL's position that there were no reasonably practicable improvements which could be made to the shutdown function was acceptable.

117. I examined a sensitivity study presented by NGL which demonstrates that there are large margins to the risk target limits for dropped fuel caused by fuel snagging, and additionally challenged NGL to demonstrate that there were no further reasonably practicable improvements which could be made. I concluded that the arguments and evidence presented by NGL adequately demonstrated that the risks associated with operation over the next Justified Period of Safe Operation (JPSO) have been reduced ALARP, and that Argument 1.4 can therefore be supported.

118. In summary, I judge that NGL has demonstrated that the nuclear safety function of the graphite core to allow unimpeded movement of control rods and fuel, will be adequately fulfilled over the proposed 4 month JPSO, and that Arguments 1.3 & 1.4 can be supported in this regard. I additionally judge that NGL has taken all reasonably

practicable measures to reduce the risk associated with impaired movement of fuel and control rods, and that in this regard Argument 2.2 can be supported.

4.2 Direct gas flows to ensure adequate cooling of the fuel and core

119. Increased cracking of the graphite bricks has the potential to change the gas flow paths within the core; this has the potential to reduce the cooling of core components and fuel. My expectations in this regard are that the safety case should demonstrate that sufficient coolant flow should be maintained to ensure that fuel and core component temperatures remain within their operational limits in accordance with SAP EHT.2, and that in the event of a fault there are sufficient barriers to a radiological release remaining, in accordance with SAP FA.7.

4.2.1 Arrow head to annulus flow

120. The gas flow paths within the graphite core are complex, and a large part of the total gas flow takes a route through the gaps between graphite bricks in order to keep the graphite bricks cool, this is called re-entrant flow. One aspect of the re-entrant flow is that it flows from the area called the arrow-head passage on the outside of the fuel channel bricks (graphite bricks which form the channels through the graphite core into which the fuel stringers are inserted) to the inside of the fuel channel bricks, in to an area between the bore of the fuel channel brick and the outside of the fuel stringer's graphite sleeve, called the annulus (Figure 8).

121. Cracking in the fuel channel bricks has the potential to lead to an increase in the flow from the arrow-head passage to the annulus as it may create new flow paths to pass through the fuel channel brick. I expect that NGL should demonstrate that the fuel clad temperatures are maintained within the operational limit in normal operation and that fuel clad integrity and fuel sleeve integrity are maintained in all fault scenarios in accordance with SAPs FA.7 & EHT.2.

122. If fuel sleeve integrity were not maintained then the coolant gas flow path in the stringer would be disrupted which could potentially lead to overheating of the fuel clad. If fuel clad integrity were to be lost then radioactive contamination would be released into the primary coolant, this would not lead to any off-site radiological consequences if only a small number of fuel pins were to fail.

123. NGL presented Reference 14 in support of Argument 1.4 of the safety case (Ref. 1), which argues that the effects of increased arrow-head to annulus flow are acceptable as the temperatures of the affected fuel and core components are maintained below their respective limits.

124. Reference 14 concludes that the effect of fuel channel brick cracking is to decrease the temperatures of the affected components down-stream of the crack as the flow increases, and to increase the temperatures of the fuel channel bricks and fuel stringer sleeves up stream in the annulus flow from the crack due to reduced coolant flow. The fuel stringer sleeve temperature increases due to brick cracking by 12-24°C in the most onerous position. However the fuel clad temperature was found to decrease by 0.8-2.3°C at the peak location in all cases. The calculation of the fuel clad and sleeve temperatures assumed the reactor was operating at 100% power, whereas the Hunsterston reactors are limited to operating at 80% of full power. In my view this introduces further margin as the temperatures would be likely significantly reduced from those discussed.

125. The extant safety case for HNB R4 does not have an explicit operational temperature limit on the fuel sleeve and therefore I sought additional confidence from NGL that the structural integrity of the sleeve will be maintained should such temperature increases be experienced.

126. In response to my request NGL presented Reference 15 which demonstrates that the temperature gradient across the fuel sleeve reduces in the presence of increased arrow-head to annulus flow, both upstream and downstream of the crack. In my view this acceptably demonstrates that the stresses in the fuel sleeve would not be significantly increased, and likely reduced, and therefore that the risk of fuel sleeve failure is not significantly increased. I note that the analysis presented in Reference 15 has not been recently undertaken specifically for this submission, but I do not consider that any of the parameters significant to the analysis have changed sufficiently to cause me concern over the validity of the results.

127. The analysis performed in support of the conclusions of Reference 14 modelled fuel channel bricks with keyway root cracks 16mm wide. This is larger than any observed crack to date (10mm), and significantly larger than the average brick crack width (3mm) but within the limiting crack discussed within Reference 1 (18mm). However the analysis models 7 layers of cracks orientated in the direction expected to maximise flow redistribution, and as such in my view the core conditions modelled are likely conservative, and meets the expectations of SAP FA.6 that the most onerous initial plant state is used in DBA.

128. I also note that the calculations performed in support of the conclusions of Reference 14 did not model multiply cracked bricks, only singly cracked bricks with large openings. It is my judgment that the change in arrow-head to annulus flow would likely be smaller for several small cracks than for a single large crack; I have not assessed this area in detail, but I am comfortable making this judgement due to the large safety margins predicted in the analysis. I therefore conclude that the analysis presented in Reference 14 is adequately bounding of the current predicted core state that its conclusions remain valid.

129. In my view NGL has adequately demonstrated that the effects of increased arrow-head to annulus flow are acceptable, this opinion is based on the fact that the predicted can temperature changes are small and negative, and that the predicted fuel sleeve temperature gradients are reduced. I therefore conclude that my expectations in this regard have been met.

130. The calculation methods employed in Reference 14 employed the whole core modelling code the PWR and AGR Neutronic and Thermal Hydraulic Evaluation Route (PANTHER) which has been extensively validated against operational plant data for normal operation, and the Finite Element Analysis Tool (FEAT) to perform Computational Fluid Dynamic (CFD) calculations. Reference 14 states that the CFD applications of FEAT have been benchmarked against a variety of cases and the results of FEAT and PANTHER calculations were compared for consistency. I have not chosen to assess the validation argument for this area as the predicted consequences are low, and ONR has previously considered the use and validation arguments for PANTHER and FEAT (Refs. 40 & 47), and no concerns were raised.

4.2.2 Channel distortion

131. As discussed in section 4.1, increased cracking of the graphite core has the potential to increase the freedom of movement of the graphite components within the core and thus potentially lead to greater core distortion. The fuel sits within the graphite core in channels in the graphite bricks and thus distortion of these fuel channels could lead to changes in the shape of the annulus (the gap between the fuel channel bore and the outside of the fuel sleeve), and thus changes to the flow paths around the fuel sleeve.

132. The design intent is that the fuel elements form a rectilinear free-standing column supported from below and sited concentrically within the fuel channel (see Figure 3 for a depiction of the column of elements). Figure 6 shows how elements are held in alignment by their lower sleeve ends fitting into the sleeve of the element below. If the

fuel channel walls distort there would be changes to the shape of the annulus and to flow paths. If the degree of distortion was large, it is possible that that fuel channel bricks could touch an element pushing the column out of the intended alignment. This could open up gaps between fuel elements with the potential to impair cooling of the fuel.

Eccentric Annulus

133. If the annulus was perfectly concentric then the gas flows down the annulus would be approximately symmetric, and thus the cooling of the fuel sleeve would be approximately even around the circumference of the sleeve. The effects of fuel channel distortion could be to make the annulus eccentric such that the size of the flow path on one side of the annulus is larger than the other, and thus the cooling of one side of the fuel sleeve would be reduced, and the other increased. My expectations are that the fuel channel bricks and fuel clad temperatures should be maintained within operational limits, and that the fuel sleeve should maintain its function in directing the gas flow as per SAP EHT.2.

134. Failure of the fuel sleeve would lead to a disruption to the coolant flow paths similar to that for gaps between the fuel sleeves as discussed later, however failure of the fuel sleeve may have a greater effect than sleeve gaps. It is also plausible that failure of the fuel sleeve could produce graphite debris which could cause disruption to the coolant flow, the consequences of which would be similar to those discussed in Section 4.2.3. I therefore judge that NGL should demonstrate that there are large margins to the failure of the fuel sleeve.

135. The most limiting case for annulus eccentricity is if the fuel stringer sleeve were to be touching the fuel channel bore on one side for the entire length of the channel. Reference 14 presents the results of NGL’s analysis of the effects of annulus eccentricity on fuel channel brick temperature, fuel sleeve temperature and fuel clad temperature.

136. The analysis conservatively adds the effects of annulus eccentricity and the effects of flow diversion due to brick cracking and tilting discussed above. This is a conservatism in my view as the two effects are unlikely to be additive as the extreme annulus eccentricity scenario assumes no gas flow to the one side of the fuel channel and thus there would be little further reduction in gas flow at this point due to cracking or tilting.

137. The analysis gives the peak fuel channel brick temperature of 476.5°C if the effects of brick cracking and tilting are included in the temperature rise along with annulus eccentricity; this gives significant margin to the fuel channel brick temperature limit of 550°C. I am therefore satisfied that the fuel channel brick temperatures are maintained below operational limits.

138. The peak fuel sleeve temperature is given as 610°C when adding the effects of both eccentricity and cracking and tilting. There is no specified temperature limit on the fuel sleeve, and thus the material property changes and oxidation rate need to be considered to ensure that the sleeve integrity is maintained at these temperatures. Reference 14 discusses the results of a sensitivity study which considered the impact on sleeve integrity at 700°C, the sensitivity study concluded that sleeve integrity would not be threatened at such high temperatures. The use of sensitivity studies and demonstration that there is no cliff edge in the analysis meets the expectations of SAP AV.6.

139. Due to the conservative approach taken in the analysis and the large margins demonstrated by the sensitivity study, I have not sought additional information on the underpinning methodology. However, from what is presented in the submission and

my interactions with the NGL, I am satisfied that NGL has adequately demonstrated that the fuel sleeve integrity is not threatened by the effects of core distortion and brick cracking.

140. The increase in peak fuel clad temperature due to annulus eccentricity given in Reference 14 is 4.8°C, which is very small in comparison to the ~100°C margin to the clad temperature limit at HNB in normal operation, and as such I judge that the effects of annulus eccentricity on fuel clad temperature are acceptable, and that the expectations of SAP EHT.2 have been met.

141. The station compliance route for fuel clad temperature includes an allowance for all of the Systematic errors, Random errors, and Uncertainties (SRU) so that the operational clad temperatures are controlled to a level such that in the event of a fault the fuel clad integrity is not threatened. NGL has included an allowance in the SRU dataset for the effects of annulus eccentricity on fuel clad temperature (Ref. 14). I therefore judge that NGL has taken adequate account of the effects of annulus eccentricity in fault conditions, and that the expectations of SAP FA.4 that the design should be demonstrated to be tolerant to faults have been met. I additionally note that the calculations of the fuel clad and sleeve temperatures discussed assumed the reactor was operating at 100% power, whereas the Hunsterston reactors are limited to operating at 80% of full power. In my view this introduces further margin as the temperatures would be likely significantly reduced from those discussed.

Sleeve Gapping

142. The fuel stringer sleeve directs the coolant gas flow over the fuel pins in order to ensure adequate fuel cooling. The fuel elements consist of 36 fuel pins within a graphite sleeve (Figure 2 **-** The fuel design has changed from this diagram, but the main components are the same for the purposes of this discussion). 8 fuel elements are joined together by a metal bar (tiebar) running through the centre of each element which holds the weight of the fuel stringers when the fuel is moved, but is unloaded when the fuel is in situ.

143. The element sleeves form a pipe from the bottom of the reactor core to the top where it joins to the upper reflector and the rest of the fuel assembly (Figure 3), the gas flow continues up through the assembly to the outlet ports which form the other end of the pipe and release the coolant above the gas baffle dome (Figure 4).

144. The gas flow through the reactor is at the highest pressure at the outlet of the gas circulators, from here between 40% and 60% of the flow goes up the side of the graphite core and in through the gaps and channels in the core, this is the re-entrant flow (Figure 4). The re-entrant flow passes down the arrow head passages and the fuel channel annulus (among other channels) to the bottom of the core. The remainder of the gas flow goes to the bottom of the core directly from the gas circulators where it mixes with the re-entrant flow and goes into the fuel stringers.

145. The pressure is reducing along the gas path due to the flow resistance of the core components, and thus the gas pressure on the outside of the fuel stringer sleeves (the re-entrant flow) is higher than the gas pressure within the fuel sleeve.

146. Core distortion has the potential to lead to fuel stringer distortion; if the fuel channel distortion is large enough then the channel may begin to impinge upon the fuel stringer and distort the fuel stringer. When the fuel stringer begins to become distorted gaps will begin to open between the sections of the fuel stringer sleeve. There is tolerance to some fuel stringer distortion as the interface between fuel element sleeves is lipped to provide some degree of gas seal (Figure 2).

147. If the gaps between fuel element sleeves becomes large enough then gas will flow from the outside of the sleeve into the stringer, this will increase the pressure within the stringer at the point of the sleeve gap. The pressure gradient up the channel will then have changed, with the pressure gradient from the bottom of the channel to the gap being reduced; this will reduce the coolant flow over the fuel between the bottom of the channel and the gap leading to increased temperatures.

148. The pressure gradient between the gap and the top of the stringer will have increased potentially increasing the flow of coolant over the fuel elements above the gap; additionally the gas flowing into the stringer through the gap will be cooler than the gas that has travelled up through the stringer over the lower fuel pins and thus the fuel channel gas outlet temperature (CGOT) will be reduced. There is then the possibility that the operator or the auto-control system could try to increase the power in the channel in order to raise the temperature as indicated by the CGOT thermocouple further increasing the temperatures in the fuel pins below the gap.

149. I expect that NGL should demonstrate that the effects of sleeve gapping on fuel clad temperatures are such that the operating limit on fuel clad temperature is not breached, and that adequate account has been taken of uncertainties in accordance with SAPs SC.5 & AV.2.

<u>Predicted Size of Sleeve Gapping in Normal Operation</u>

150. Reference 16 predicts that the maximum sleeve gap that will occur over the JPSO will be <1mm in normal operation; this report has been assessed by an ONR graphite specialist inspector who judges that the conclusions of the report can be supported (Ref 12 & 45).

151. There are some scenarios in which sleeve gaps in excess of 1mm could arise in normal operation:

- As part of the refuelling strategy at HNB fuel can be moved from one channel to another (shuffled). When a fuel stringer is irradiated it can become bowed due to the effects of irradiation of the fuel sleeve. I asked NGL what size of sleeve gapping would result if a bowed fuel stringer was shuffled into a distorted fuel channel (Ref. 10); NGL stated that the maximum sleeve gapping would be ~2.4mm distributed over 2-3 sleeve interfaces giving individual sleeve gaps <2mm. An ONR graphite specialist inspector has considered NGLs response to this question and agreed that their position can be supported (Ref. 12 & Ref. 45 Section 4.5.6).
- NGL states (Ref. 1) that in the unlikely scenario that a piece of graphite debris becomes lodged in the annulus between the fuel stringer and fuel channel bore it could force the fuel stringer out of shape and cause gapping between fuel sleeves. NGL states that due to the geometry of the fuel channel and size of the stringers the gap could be no larger than 2mm, although this doesn't account for the potential fuel channel distortion. NGL additionally argues that any debris becoming lodged between the fuel sleeve and the fuel channel wall would have low strength and thus be crushed; an ONR graphite specialist inspector has assessed this argument and judged that it can be supported (Ref. 12 & Ref. 45 Section 4.6.2).

In my view - due to the effects discussed – NGL should demonstrate that the effects on can temperature are acceptable for sleeve gaps up to 2mm.

<u>Calculation of the Effects of Fuel Sleeve Gapping</u>

152. I examined the methodology employed by NGL to determine the effects of sleeve gapping on fuel clad temperature (Reference 17) and found that the methodology was

suitable. A modified PANTHER flow network is employed to determine the pressure drop across the sleeve gap and thus the gap flow, and then a bespoke code implemented in VBA is used to calculate the temperatures at each point in the fuel stringer. The route can then be rerun modelling the channel power increase due to the auto-control system responding to the CGOT temperature changes to give a more onerous result.

153. The PANTHER flow network is used in many of the thermal hydraulic applications of PANTHER, and is extensively validated against real plant data for normal operation. The change introduced in order to model sleeve gapping is simply another flow path with an associated flow resistance; as such I judge that it is highly unlikely for this to introduce significant errors into the calculation, and that the methodology for calculating the gap flows is appropriate.

154. The fuel clad temperature calculations are performed using a Visual Basic for Applications (VBA) code implemented in Excel (Microsoft program). The code itself uses well established and simple formulae to calculate the temperatures in the fuel stringer, and performs a best-estimate calculation. In my view the use of a best-estimate calculation is acceptable at this stage due to the treatment of uncertainty that is applied to the results of the calculations which I will discuss later. I challenged NGL on the implementation of the code using a third party software package such as EXCEL as this introduces complications into the verification of the calculations.

155. In order to address my challenge NGL stated that the code was verified to QA2 standard which is in my view an appropriate standard to have applied within NGL's quality assurance processes. NGL additionally provided me with a copy of the verification statement for the fuel clad temperature calculations, and I am satisfied that appropriate measures were taken to verify the correct performance of the code, and that the expectations of SAP AV.4 were met.

156. The calculations of fuel clad temperature were performed for a range of sleeve gap sizes and locations which provides a good sensitivity study of the fault; this meets the expectations of SAP AV.6.

157. The calculations of the gap flows are highly dependent on the flow resistance values used to represent the sleeve gaps, and thus I examined the validation of the flow resistance values used.

158. Reference 18 presents the results of comparisons between the calculation model used to predict sleeve gap flow resistances and rig tests measuring sleeve gap flow resistances. For small wedge shaped sleeve gaps of the sort which may occur in HNB over the JPSO, gaps of 1mm were examined and the predicted gap resistances were shown to be conservative by ~30%. For larger sleeve gaps, full circumferential sleeve separation was examined between 12mm and 72mm, and the results showed the predicted flow resistances to be non-conservative by an order of magnitude. The rig tests performed in Reference 18 used fuel sleeve designs which are different to the fuel sleeves which are currently in the AGRs, and this clearly introduces uncertainty into the results.

159. In my view the validation of sleeve gap flow resistances is inadequate due to the limited range of gap sizes tested, especially in the wedge shaped gap geometry of most interest, and the use of fuel sleeves of a different design to that in use in the reactor. I was not initially satisfied that the flow resistances were supported by the referenced test rig results (Ref 18) as the predicted limiting sleeve gap sizes in normal operation (up to 2mm) were outside of the range of gap sizes tested.

160. The ONR TAG on the validation of computer codes and calculation methods (Ref. 4) states that the selective use of experiments to support validation should be avoided

and that explicit justification would be required for exclusion of relevant experimental results. Therefore, in my view the only sufficiently conservative interpretation of the validation test results would be to assume that the order of magnitude non-conservatism seen for large circumferential gaps applies to all sleeve gaps. In forming this opinion I asked NGL to justify why it would not be appropriate to assume that a larger uncertainty applied (Ref. 10).

161. In response to my question NGL stated (Ref. 10) that there is good agreement with experimental results for gap sizes of 1mm, and that predicted sleeve gap flow resistances are nearly constant between 1mm and 5mm sleeve gaps (Figure 5) and that this makes sense physically as the spigot is still engaged over this range of gap sizes (Figure 6) leading to a roughly constant flow area. As such NGL argues that it is highly likely that uncertainties of an order of magnitude would be bounding for wedge shaped gap sizes <5mm.

162. I accept NGL's argument that it is highly unlikely that uncertainties greater than an order of magnitude to apply to small wedge shaped sleeve gaps, and maintain my opinion that it is appropriate to consider the effect of uncertainties in the sleeve gap flow resistances of up to an order of magnitude. I judge that this provides a robust conservative position on the uncertainties associated with the lack of directly applicable validation.

Consequences of Fuel Sleeve Gapping

163. Reference 1 states that the temperature effects of a 7mm sleeve gap at the most onerous position have been demonstrated to be acceptable, I discuss this in the following paragraphs. Reference 1 argues that a 7mm sleeve gap corresponds to a 30°C increase in fuel clad temperature (assuming 100% reactor power), and that the reduction in fuel clad temperature due to the HNB reactors now operating at only 80% of nominal full load thermal power more than offsets this potential increase.

164. A 30°C predicted clad temperature increase had been set as an informal limit by NGL as the point at which they would review the treatment of uncertainties for sleeve gapping to determine whether the temperature effects should be included in the station clad temperature compliance routes (Ref. 10); however the results of Reference 23 show that if an order of magnitude reduction in the sleeve gap flow resistances is assumed then the flow resistance associated with a ~2mm gap would be reduced such that a ~30°C clad temperature rise would be associated with a ~2mm gap.

165. In response to my queries (Ref. 10 & 24) NGL has stated (Ref. 24) that if sleeve gaps were predicted in excess of 2mm then NGL would review the uncertainties associated with the sleeve gap flow resistance assessment, and are imbedding this process within the sleeve gapping assessment checks.

166. I am satisfied that even with the likely conservative treatment of sleeve gap flow resistances the fuel clad temperature rise due to the maximum predicted sleeve gapping over the proposed JPSO (2mm) is <30°C. I am also satisfied that the clad temperature effects of sleeve gapping do not need to be included in the station fuel clad temperature compliance calculations at present as there is ~100°C margin to the fuel clad temperature limit due to operation at 80% power, which clearly gives significant margin. I am therefore satisfied that NGL's arguments in this regard can be supported. I judge that NGL has demonstrated the robustness of the fault tolerance of the design in accordance with the expectations of SAP FA.4, and that this has included an adequate accounting for uncertainties in accordance with SAP SC.5.

167. In my opinion, there shall be a need for further validation of the flow resistances associated with sleeve gapping should the predicted levels of sleeve gapping exceed 4mm. I judge that up to 4mm the application of an order of magnitude non-

conservatism as discussed within this report is sufficiently bounding of the uncertainties, due to the near-constant flow resistances predicted between 1mm and 5mm, but that >5mm the argument for an order of magnitude uncertainty being limiting is less obvious. I therefore judge that 4mm provides margin to the boundary of this stable region of flow resistance. I shall not make a recommendation in this regard as I judge that the position is robust for the proposed 4 month JPSO, and future safety cases will provide adequate hold-points to review this position, but I have communicated my view to NGL.

Effects of sleeve gapping on tie bar temperatures

168. When the fuel is moved the weight of the fuel elements is taken by the tiebar. In a fuel snagging event the probability that the tiebar will fail leading to a dropped fuel event is related to the temperature of the tiebar.

169. NGL's safety case (Ref. 1) states that the effects on the tiebar temperature of sleeve gapping are an increase of 10-20°C for a gap of 6mm, and that the effects of such a temperature rise have a negligible effect on the probability of tiebar failure during fuel movement. If the order of magnitude uncertainty is applied to the sleeve gap resistance as discussed above then the flows at a 6mm gap then relate to a 2mm gap which is the limiting gap size predicted in the proposed JPSO. I therefore judge that the effects of sleeve gapping should have a negligible effect on tiebar failure over the JPSO.

170. My conclusion on the effects of sleeve gapping are that NGL has adequately demonstrated that the operational limits on fuel clad temperature will not be breached and that the tiebar failure probability will not be significantly altered during the proposed JPSO as per my expectations. I judge that should sleeve gapping >4mm be predicted in future safety cases then an improved sleeve gap flow resistance validation argument will be required.

4.2.3 Debris

171. As the graphite bricks in the reactor core crack, there is the potential for debris to be produced. Once debris is loose in the gas circuit, it is possible that it could make its way into the inside of the fuel sleeve and create an obstruction to the gas flow. Argument 1.4 of NGL's safety case (Ref. 1) states that a blockage of 16.5% of the flow area at the element 1 support grid (Figure 2) would be required in order for the fuel clad temperature to reach the operational limit set in the Technical Specifications of 870°C. Reference 1 states that NGL does not consider it credible for a larger area to become blocked.

172. NGL argue that the route that graphite debris would have to take to reach the element 1 support grid is very convoluted which places restrictions on the size of the debris which could traverse the various routes. NGL has considered the maximum size of graphite debris which could traverse these routes and concluded that at least 3 pieces of graphite debris of the limiting size (~30mm x ~50mm) would be required to cause a blockage of >16.5% (Reference 20).

173. The potential for the creation of graphite debris and the probability of it migrating to an element 1 support grid and causing a blockage, and the credibility of a blockage occurring >16.5%, has been considered by an ONR graphite specialist inspector (Ref. 32 & Ref. 45). The conclusion reached was that such a blockage cannot be totally discounted as a design basis event. As a result, within this assessment report I have explored the potential consequences of such blockages and made comparisons against the numerical targets in the SAPs to frame my judgements on whether risks have been reduced ALARP for these scenarios.

174. It is plausible that cracking of the fuel sleeve could produce graphite debris, but I have concluded in Section 4.2.2 that NGL have adequately demonstrated that the sleeve integrity would not be threatened by the increased graphite brick cracking over the proposed 4 month JPSO which Reference 1 seeks to justify; I have therefore focussed my assessment on the consequences of graphite debris produced by graphite brick cracking directly.

175. The fuel element support grids and fuel braces (Figure 2) hold the fuel pins in place, and the structure required to do this restricts the space through the grids and braces. The fuel element support grids have smaller spaces in them than the fuel braces, and thus if debris were to pass through the support grid then it is unlikely to become stuck in a fuel brace. The element 1 support grid is considered here as it is the first support grid or fuel brace in the flow path up the fuel stringer and thus larger items of debris are likely to become stuck there. If debris is small enough to pass through the element 1 support grid then the most onerous location that it could become stuck is then an element 6 support grid or fuel brace; this is considered later in this section.

176. My expectations for a design basis faults are that the fuel clad temperature limits are not exceeded as a consequence of debris in normal operation or faults, such that in accordance with SAP FA.7 the integrity of the fuel clad is maintained. Failing this, I would expect the predicted radiological consequences of any fuel failure, taking credit for any design basis measures, to be less than the numerical targets established by the SAPs Target 4 BSL, and the risks to be consistent with SAP Target 8. In addition, I need to be satisfied that all reasonably practicable measures to reduce risk have been taken.

177. For blockages <50% of the flow area NGL states that the bulk gas flow up the channel is not significantly affected, and as such the effect on the CGOT is negligible; this means that the temperature effects of a blockage ~16.5% would not be detectable. However blockages <50% do result in an increase in fuel clad temperature as they create a region of stagnant flow immediately downstream which can significantly impair the heat transfer of the coolant.

178. Reference 19 reports the results of rig tests examining the effects of blockages on the heat transfer for fuel pins in the stagnant flow region immediately downstream of a blockage. The report looks at blockages up to 12.5% of the flow area and reports the results in terms of the ratio of the pre and post blockage Stanton number. The Stanton number is a dimensionless number defined as the ratio of the heat transfer into a fluid to the heat capacity of the fluid, and can be thought of as the efficiency of the coolant.

179. The results presented in Reference 19 demonstrate that a stagnant region of flow occurs immediately downstream of the blockage but only continues for a small distance (~20cm) before the coolant flow conditions are recovered, with the most significant temperature effects occurring directly behind the blockage. The most significant effect occurs at the largest blockage tested (12.5%) directly behind the central region of the blockage, with the Stanton number being 0.35 of the unblocked Stanton number.

180. Reference 20 uses the results of the rig tests presented in Reference 19 to extrapolate the limiting blockage at which the fuel clad temperature operational limit would be reached for HNB normal operating conditions. Reference 20 concludes that the fuel clad temperature operational limit (870°C) would be reached at a blockage of 16.4% as reported in the safety case (Ref. 1), and additionally concludes that the fuel clad melt temperature (1350°C) would be reached at a blockage of 17.6%.

181. The conclusions of Reference 20 imply that there is a cliff edge effect where a small increase in blockage size increases the consequences significantly. Given the uncertainty in the extent of any blockage, the closeness of the assumed amount of

debris to the cliff-edge, and the dangers of making predictions from extrapolating beyond the limits of a dataset, my initial assessment conclusion was to treat these predictions with caution.

182. NGL stated that they consider the results of Reference 20 to be highly conservative as it takes no account of the effects of conduction or thermal radiation on the heat removal, and that these effects would not be impaired by the blockage and would increase in magnitude with increasing temperatures. NGL subsequently produced Reference 21 which attempts to quantify the effects that conduction and thermal radiation would have on the fuel clad temperature.

183. Reference 21 concludes that the effects of conduction are small, but that the effects of thermal radiation would significantly reduce the fuel clad temperatures from those predicted in Reference 20.

184. I accept the conclusions of Reference 21 that the effects of radiation would reduce the fuel clad temperatures significantly, and I judge that the calculations performed in support of these conclusions are conservative in their estimation of the magnitude of the effect. I make this judgment as the calculations in support of Reference 21 have assumed that the material surrounding the fuel pin is a solid of the same thermal emissivity as the fuel pin and with relatively similar temperature to the fuel pin. In reality the surrounding material would be gas of a relatively low thermal emissivity (Reference 22) with a relatively high temperature, and then outside of this, solids of similar emissivity but with relatively low temperatures; as a result the incident thermal radiation would likely be reduced from the situation modelled in Reference 21 and thus the fuel clad temperature would be lower. The magnitude of the conservatism this represents is difficult to quantify without fairly complex calculation, however in my opinion it does represent conservatism in the calculation of the effects of thermal radiation.

185. I judge that there are some significant uncertainties associated with the application of the rig test results presented in Reference 19 in determining fuel clad temperatures. Some of the significant sources of uncertainty are:

- The extrapolation of the results of Reference 19 to blockages >12.5%.
- The use of different fuel geometry and inlet flow conditions to HNB fuel.
- The rig had no thermocouple in the very centre of the blockage where the most onerous flow conditions might be expected (although very close).

186. In my opinion it is reasonable to conclude that the value of 17.6% blockage presented in Reference 20 is a conservative central estimate of the point at which fuel clad melt would occur, but that the uncertainties associated with that central estimate are large in relation to the conservatism.

187. The production of graphite debris is likely to increase as the core ages, and thus the issue of flow obstruction due to graphite debris is likely to become more significant in future safety cases. In my opinion NGL should perform further analysis of the effects of blockage at the element 1 support grid in order to quantify and reduce the uncertainties, and to determine whether or not a cliff edge effect does occur and if so, where. In my view the analysis should consist of further modelling as well as modern rig tests. I have raised a regulatory issue in this regard (issue 7291).

Recommendation 1: For inclusion in future safety cases justifying the operation of the Hunterston B Reactor 4 graphite core, NGL should perform further analysis of the effects of a blockage at the element 1 support grid in order to establish the point at which fuel clad melt temperatures would be reached.

188. Reference 21 attempts to determine the probability of graphite debris reaching the element 1 support grid and causing a blockage >15% and derives a value of $1x10^{-5}$ pry. As discussed this value has been assessed by an ONR graphite inspector who has concluded that a value of $1x10^{-4}$ pry can be supported (Ref 32 & 45), but not 10^{-5} pry as claimed by NGL. SAP FA.5 expects that design basis analysis be applied to all faults with an initiating event frequency greater than 10^{-5} pa; as such the potential for fuel clad melt has to be conceded within the design basis. In my view, only one channel affected by graphite debris at the element 1 support grid need be considered as the coincident fault in more than one channel would be of such low frequency as to be beyond the design basis.

Consequences of an element 1 blockage

189. SAP FA.7 sets an expectation that, as far is as is reasonably practicable, the correct performance of a safety systems should ensure for a design basis fault that none of the physical barriers to prevent the escape or relocation of a significant quantity of radioactive material are breached. As there is no way to detect the presence of a blockage at the element 1 support grid prior to the failure of the fuel clad in the affected element 1 pin(s), there is no way to prevent the failure of the fuel clad in the affected pin(s). Attention therefore turns to the subsequent expectations of SAP FA.7 that at least one barrier remains intact, there is no release of radioactivity, and no person receives a significant dose of radiation. If these expectations cannot be demonstrated, the resulting radiological consequences should compare favourably with numerical target 4 to allow ALARP judgements to be made.

190. I requested that NGL present analysis of the consequences of fuel clad melt due to an element 1 blockage (Ref. 10), so that comparison can be made with the expectations of SAP numerical targets 4 & 8.

191. The radiological consequences of the failure of 1 or 2 element 1 pins fuel clad would be equivalent to or less than 1 or 2 more typical fuel failures. The generic failed fuel safety case (Ref. 29) sets limits on the activity in the coolant such that there are insignificant radiological consequences to operators or the public from fuel failures. NGL states that the failure of 1 or 2 element 1 pins fuel clad would not lead to coolant activity approaching the limits set in the generic failed fuel safety case (Ref. 10). I therefore judge that there would be insignificant radiological consequences from the failure of 1 or 2 element 1 fuel pins unless there was some escalation of the fault.

192. NGL stated that in the event that the fuel clad melt temperature is reached for a fuel pin at the bottom of element 1 it would be a very localised hot spot due to the recovery of the flow short distances downstream of the blockage, and that the results of the rig tests (Ref. 19) show that the reduction in heat transfer was significantly reduced by 7.5 cm down the pin. NGL therefore argues (Ref. 21) that as there are anti-stacking grooves in the fuel pins at pellet numbers 1 and 5 (the bottom and 5th from bottom pellets) which hold the pellets in the fuel clad, that if the fuel clad melt temperature was reached then it would only be those pellets below the anti-stacking groove at pellet 5 which could potentially become mobile.

193. In my view the results of the rig tests presented in Reference 19 support NGL's argument that the hot spot would be localised, and I agree that the presence of the anti-stacking grooves would likely mean that only a small number of fuel pellets could become mobile. I therefore requested that NGL justify what the consequences would be of a small number of fuel pellets becoming mobile in the primary circuit (Ref. 10).

194. NGL presented Reference 27 which states that the only safety implications of fuel debris being loose in the primary circuit is if it were to become lodged at the grids or braces within the fuel stringer and cause a further blockage rather than getting swept out of the channel, as the loose fuel debris would not present a threat to core

components. NGL has implicitly argued that the pressure vessel liner would only be threatened if molten fuel were to be produced, which would require a very significant fuel channel blockage.

195. NGL states (Ref. 10) that the most onerous scenario following a small number of fuel pellets becoming mobile in the primary circuit is if they were to become lodged near to the peak rated fuel pin in the channel (typically in element 6) causing a further blockage. Calculations performed in Reference 27 determine that there is no short term threat to core components if a large amount of fuel debris were to become lodged at the peak rated location in a stringer; the calculations make conservative assumptions on the amount of fuel debris, the power of the peak rated pins, and the coolant conditions, and conclude that the temperature of the grid or brace against which the debris was lodged would not exceed 1050°C at 100% reactor power. At this temperature failure due to oxidation of the material would not be expected for several weeks, and thus the situation would be temporarily stable. NGL argues that the integrity of the grids and braces would prevent a significant fuel channel blockage of the sort required to produce molten fuel.

196. The calculations in Reference 27 do suggest that additional fuel clad melt could occur in the fuel pins near the blockage, but notes that the most onerous assumption is that any fuel debris produced becomes part of the initial blockage, raising the temperature of the grid or brace. NGL states that the initial calculations demonstrated margin in the quantity of fuel debris to further escalation (failure of the grid/brace). Reference 27 concludes that there would be no short-term escalation in the event that 180g of fuel debris (9 pellets) became lodged at the most onerous location in the stringer.

197. There is no validation argument associated with the calculations presented in Reference 27, but the code used was independently verified. Due to the nature of the phenomenon being analysed in Reference 27 I judge that there would be no reasonably practicable means of validating the calculations, and thus the shortfall against the expectations of SAP AV.2 are acceptable. I additionally judge that the conservative assumptions applied in the analysis are adequate such that the conclusions of the calculations can be considered valid.

198. In my view it is possible that the failure of fuel pins near the secondary blockage caused by the loose element 1 pellets could lead to further fuel debris becoming loose in the primary circuit, and that this debris could then lead to another separate blockage again leading to further fuel pins failing. Whilst this escalation could lead to significantly more fuel debris being produced in the channel, the analysis presented in Reference 27 determines that the grids and braces supporting these blockages would not fail for a significant length of time (weeks). The totality of the debris would not therefore be able to come together and cause a significant channel obstruction of the sort which would be required to lead to fuel melt. I therefore judge that it is likely that any significant core component damage would be avoided if action were taken within several days to reduce the temperatures in the affected fuel stringer of the grids and braces.

199. NGL presented Reference 28 which examines the response of the Burst Cartridge Detection (BCD) system, and concludes that if a small number of loose fuel pellets were present in the primary circuit and if they formed a blockage at the peak rated location in the stringer then the BCD signal would reach Action Level 1 as specified in the Generic Failed Fuel Safety Case (Ref. 29). Action Level 1 requires operator action to reduce the reactor power until the BCD signal is below the action level, however even much lower BCD signals would initiate an investigation into the cause of the activity.

200. SAPs ESS.8, EHF.5 & FA.6 are generally interpreted as expecting that in DBA no benefit from operator action should be claimed until 30 minutes after the initiating

event, unless the operator action has been specifically justified on shorter timescales. Applying that expectation to the fault escalation described in Reference 27, it is likely that the BCD signal would be greater than Action Level 1 at 30 minutes and thus the operator would reduce reactor power accordingly. It is possible that the failure of further pins near the secondary blockage would have increased the BCD signal to Action Level 2 by 30 minutes which would require the operator to shutdown the reactor.

201. SAP FA.6 expects that fault protection be tolerant to the most onerous single failure, and thus I asked NGL to demonstrate that the single failure of the BCD system would not significantly increase the consequences of the fault (Ref 10). NGL stated that in the event of the unavailability of the BCD system the Gaseous Activity Monitor (GAM) can provide monitoring of the activity in the primary circuit, which would enable the operator to detect a fuel failure.

202. NGL stated that failure of the pressure vessel would only occur following significant channel blockage and fuel melt. I judge that the provision of diverse alarms and indications, as well as sufficient time to allow operator action means that NGL has adequately demonstrated a functional line of protection to protect against significant channel blockage and fuel melt.

203. Whilst I accept that fuel pellets loose in the primary circuit are unlikely to present a significant threat to core components or the pressure vessel liner, due to the significant uncertainties associated with this assumption, on a conservative basis I judge that further consideration of the failure of the pressure vessel liner and the potential consequences is appropriate. This too addresses the expectations of SAP FA.6 regarding the single failure tolerance of the fault protection. I also consider it appropriate to consider further the potential consequences of a large number of fuel clad failures due to the uncertainties associated with the timescales of the fault progression. I discuss these two points in the sections below.

<u>Consequences of a large number of pin failures due to fuel clad melt</u>

204. In order to justify the radiological consequences of fuel debris in the primary circuit NGL presented References 30 & 31.

205. Both Reference 30 & Reference 31 conclude that in the absence of an unfiltered depressurisation route (no liner penetration) the radiological consequences due to normal pressure vessel leakage would be <100mSv (Dose Band 3 or lower) even with large scale fuel clad melt far in excess of that considered likely following a blockage at the element 1 spacer grid.

<u>Consequences of a consequential failure of the pressure vessel liner</u>

206. Reference 30 presents the radiological consequences of a gag shaft failure fault, this is a separate fault to that discussed in this report and is unrelated to the effect of graphite brick cracking; however the gag shaft failure fault has limited potential to cause fuel clad melt at the very extreme of the design basis, and thus the potential consequences have been analysed, and this is relevant to the potential consequences of blockages due to graphite debris.

207. The analysis presented in Reference 30 determines the radiological consequences of fuel clad melt leading to significant fuel channel blockages and fuel melt in the affected channel; which is a much more onerous failure than that predicted for the blockages under consideration in this report, and the sort of fault sequence which I concluded above has been demonstrated to have an adequate line of protection.

208. Reference 30 determines that the radiological consequences could only be >100mSv (Dose Band 4) in the event that molten fuel reached the pressure vessel liner and melted through the liner in the location of a Pressure Vessel Cooling Water (PVCW) pipe and created an unfiltered depressurisation route, and even then the unfiltered depressurisation would have to occur promptly after the release of the activity into the primary circuit in order for radiological consequences >100mSv to occur following the (much more onerous) gag shaft failure fault. This is because the boilers act as a highly efficient filter for the activity in the primary circuit such that after a few minutes there would only be a small proportion of the activity left in the primary circuit.

209. In my view some very conservative assumptions have been made in the above fault progression, and thus I judge that the consequences of a blockage at element 1 with a consequential failure of the pressure vessel should not be considered likely to exceed 100mSv (Dose Band 3) within the design basis.

<u>Comparison of risk to SAPs Targets 4 & 8</u>

210. SAP FA.7 expects that design basis analysis should demonstrate that there is no radiological release and that at least one barrier to a radiological release remains intact. In my view the pressure vessel liner integrity provides a line of protection against a radiological release, however there is a potential threat to this final barrier due to fuel debris loose in the primary circuit, and the barrier cannot be considered 100% effective due to normal leakage from the vessel. In the event that the expectations of SAP FA.7 are not completely met SAP numerical target 4 provides a target for the residual mitigated risk from the fault (SAP para 637 & A34, ONR transient analysis TAG (Ref. 4)).

211. I conclude that significant blockages at the element 1 position should be considered as having the potential to lead to fuel clad melt, and that the potential consequences of this could conservatively be up to Dose Band 3 (10-100mSv). The SAP Target 4 Basic Safety Level (BSL) expects that the resultant mitigated radiological consequences of faults with an initiating event between 10^{-4} pa and 10^{-5} pa should be <100mSv. An ONR graphite specialist inspector has concluded that the probability of a significant blockage at element 1 can be supported at 10^{-4} pry (Ref 32 & 45), and thus I conclude that the expectations of the Target 4 BSL are met (Note that 1 pry is less than 1 pa as a reactor does not typically operate for 100% of a year).

212. Numerical Target 8 of the SAPs provides guidance on the tolerable level of risk from a nuclear facility assessed on a best estimate basis. Paragraphs 749 & A49 of the SAPs state that the targets specified in Target 8 should be reduced by an order of magnitude to give guidance on the tolerable risk from an individual fault.

213. Reference 30 presents analysis and probabilities of the potential fault escalation routes following fuel clad melt in gag failure faults. If the results of Reference 30 are applied directly to a blockage at element 1 with an initiating event frequency of 10^{-4} pry then the risks would be below the single fault interpretation of the Target 8 BSL in all dose bands. In my opinion the probability of a radiological release of greater than DB3 is significantly reduced in the element 1 blockage fault relative to the gag failure fault as the initial quantity of fuel debris produced is lower, and the bulk flow conditions in the channel are not affected in the element 1 blockage fault. I therefore judge that it is likely that the risks due to an element 1 blockage would be close to the single fault interpretation of the Target 8 Basic Safety Objectives (BSO).

214. Comparison of the assessed risks presented by blockages at element 1 against SAP Target 8 demonstrates that the fault presents a minor contribution to the total station risk, and I note that risk of DB3 consequences at 10^{-4} pry is more than an order of magnitude lower than the assessed risk from the fuel route safety case discussed in section 4.1.2.

215. Comparison of the risks associated with blockages at the element 1 support grid against the SAP numerical targets 4 & 8 demonstrate that the risks are tolerable provided that all reasonably practicable measures have been taken by NGL to reduce the risk. As such, I asked NGL to demonstrate that all reasonably practicable measures to reduce risk have been taken and that adequate consideration of what measure could be taken had been made (Ref. 10).

ALARP

216. NGL presented a review of the potential options to reduce the risk associated with a blockage at element 1 (Refs. 48 & 49). This included consideration of the benefits and dis-benefits of each option, as well as the likely sacrifice involved in the implementation of each option in terms of time, trouble, and cost, in accordance with the ALARP principle. This meets my expectation that adequate consideration should be made of the potential options to reduce risk.

217. In my view the ALARP arguments are clear for the majority of the options considered by NGL to reduce the risk associated with element 1 blockages. In these cases I agree with NGL's position that the sacrifice or dis-benefits of the option is grossly disproportionate to the potential risk benefit; however there is one potential option which warrants further discussion.

218. Following my query (Refs. 48 & 49), NGL considered the potential benefits and dis-benefits of altering the action levels and/or responses to high BCD signals. The BCD action levels specified in the generic failed fuel safety case (Ref. 29) require that the operators reduce the reactor power upon reaching action level 1 to reduce the BCD signal to below the action level (3x background), and that the operators shutdown the reactor upon reaching action level 2 (6x background).

219. NGL considered 3 options relating to the BCD action levels;

- maintain the generic failed fuel safety case BCD levels and actions,
- introduce more conservative action levels,
- or change the action to reactor shutdown following detection of failed fuel.

220. NGL concluded that the potential sacrifices of either change are grossly disproportionate to the safety benefit, and thus that maintaining the generic failed fuel safety case (Ref. 29) BCD action levels and actions is ALARP.

221. Consideration of the potential benefits of changes to the BCD action levels or responses requires consideration of the potential fault escalation routes following a small number of fuel pellets being released into the primary circuit. I shall consider the potential benefits of reducing the BCD action levels separately:

Reducing the point at which reactor shutdown is required (BCD action level 2)

222. The most likely scenario following the release of fuel pellets into the primary circuit is that they be swept out of the fuel channel and come to rest benignly somewhere within the pressure vessel. In this scenario NGL states (Refs. 48 & 49) that a high BCD signal would be received and the BCD alarm (1.5x background) is likely to have triggered which would initiate investigation into the source of the alarm. The operators are likely to consider pre-emptively reducing reactor power at this stage if the signal is rising in order to maintain the BCD signal below that of action level 1. NGL states that reducing the BCD action levels to levels low enough such that they mandate reactor shutdown upon receipt of the signal levels likely at this stage would result in spurious reactor shutdowns. NGL presented operating experience of spurious BCD signals and more typical fuel failures which were of the region of the BCD action level 1 signal, to support their argument (Refs. 48 & 49).

223. In my view there would be no benefit to shutting down the reactor in this scenario as it would not avert any risk of radiological release. There is the potential that one of the loose fuel pellets could melt through the pressure vessel liner, however in my view shutting down the reactor would, if anything, make this more likely due to the reduced cooling flows around the vessel, and because the heat generation of the loose pellets would be unaffected by the reactor shutdown as they have already left the active core.

224. The most onerous scenario following the release of a few fuel pellets into the primary circuit is where the fuel pellets become lodged in the peak rated element (element 6). Reference 28 determines that in this scenario BCD action level 1 is likely to be reached (3x background), and the BCD alarm (1.5x background) will certainly have been triggered. From this position it is likely that fuel clad melt would occur in some element 6 pins. NGL states (Refs. 48 & 49) that the BCD signal would very likely reach BCD action level 2 following the clad melt on the element 6 pins which would direct the operator to shut down the reactor.

225. From NGL's responses to my queries (Refs. 48 & 49), I understand that following a secondary blockage at element 6 there are 3 different potential timelines of events which would change the operator response, and determine whether there was any safety benefit to reducing BCD action level 2:

226. In the first scenario, if the operator became aware of a high BCD signal or was alerted by the BCD alarm (1.5x background) following fuel clad melt at element 1 they would seek specialist advice from the Nuclear Safety Group at the station. The Nuclear Safety Group and the operators would closely monitor the situation and if the BCD signal were to escalate then pre-emptive action would be taken to either reduce reactor power or shutdown the reactor ahead of reaching the BCD action levels (Ref. 10). The Nuclear Safety Group would check the GAM readings and with support from NGL's fuel specialists would likely be able to determine that there was significant exposed fuel, rather than a more typical fuel failure. In my view significant fault escalation would be unlikely to occur in this scenario due to the heightened attention that the fault would receive, and thus a change to the BCD action levels would provide little safety benefit. In my opinion this is the most likely scenario.

227. In the second scenario, if the assumption typically applied in DBA of 30 minutes for operator action is assumed, or if the fault is assumed to escalate rapidly and fuel clad melt had occurred in some element 6 pins by the time the operator responded, then NGL states that BCD action level 2 would already have been reached, and thus the operator would be directed to shut down the reactor (Refs. 48 & 49). In this scenario there is clearly no safety benefit to changing the BCD action levels as reactor shutdown is already required.

228. The final scenario is where the operator responds to the BCD alarm at a point prior to BCD action level 2 being reached, but they have not been aware of the issue and monitoring it for escalation, and are therefore likely not in a position to act pre-emptively as would be expected in the first scenario. If there was a small window for the operator to act in which there is the potential that shutting down the reactor could avert fuel clad melt in element 6, but reducing reactor power would not be enough, then in this scenario there would be a safety benefit to reducing BCD action level 2. In my view this is a very specific scenario which represents a small proportion of the possible progressions, and it is likely that the fuel cladding at the element 6 pins would reach melting temperature quickly following a blockage in element 6 leaving a very small window for such a scenario to occur. I therefore judge that the overall safety benefit provided by a change to the BCD action level is small.

229. NGL presented details of past events in which BCD signals of around or approaching BCD action level 1 were seen (Refs. 48 & 49). In many of these events pre-emptive action was taken by the operators to reduce load or shutdown the reactor to prevent

further escalation. as described in the first scenario above. NGL also presented details of spurious BCD readings approaching BCD action level 1, demonstrating that a reduced threshold for reactor shutdown would lead to an increase in spurious shutdowns, and an associated risk dis-benefit.

230. NGL's position is that if BCD action level 2 were reduced then there would be reduced flexibility for the operators and technical support staff to manage the situation, such that the ability to locate and characterise the fuel failure was reduced or eliminated making recovery from the situation difficult or impossible. NGL additionally states that it is much more likely that high BCD signals would be due to more typical fuel failures (expected 10^{-1} pry) than due to clad melt following an element 1 blockage (10^{-4} pry). NGL state that in the event that a reactor shutdown was mandated following a more typical fuel failure, the failed fuel would not be locatable as the BCD can only detect short-lived fission products, and thus there would be a high probability that the fuel would contaminate the fuel route when eventually discharge from the reactor as it would not be properly controlled, leading to increased operator doses.

231. In my view there is a very specific scenario in which reducing BCD action level 2 such that reactor trip is required earlier provides any safety benefit (the final scenario above), but that this scenario is very unlikely to occur. There would be a potential risk dis-benefit to reducing the point at which reactor trip is required due to an increase in the number of reactor shutdowns which would occur due to more typical fuel failures and spurious BCD signals, and an associated increase in operator doses in the fuel route. I therefore conclude that NGL have adequately demonstrated that it is not ALARP to reduce BCD action level 2. ONR's TAG on ALARP (Ref. 4) supports the evaluation of the overall risk profile of a proposed change such as done here.

<u>Reducing the point at which reduction in reactor power is required (BCD action level 1)</u>

232. If a secondary blockage is created at element 6 following a blockage at element 1 then the generic failed fuel safety case (Ref. 29) BCD action levels would require a reduction in reactor power prior to the point at which clad melt would be expected in element 6 pins. As described in the first scenario above the operators would likely act pre-emptively to reduce reactor power prior to BCD action level 1 being reached, and thus a reduction in BCD action level 1 would not provide any benefit with respect to clad melt in element 6.

233. If fuel clad melt at element 1 proceeds reasonably slowly then a reduction in reactor power may be possible prior to significant clad melt at element 1, which may prevent or delay further clad melt. NGL argues that requiring a reduction in reactor power at relatively low BCD signals would be highly likely to make any fuel failures undetectable (Ref. 38) leading to failed fuel contaminating the fuel route upon discharge, and the associated operator doses, as discussed previously. NGL states (Ref. 10) that consideration of a precautionary reduction in reactor power would be part of the management of failed fuel (Ref. 39) which would occur following detection of a high BCD signal, but that the risk would be balanced against the risk of not finding the failed fuel.

234. In my view, the details of past events in which BCD signals occurred of around or approaching BCD action level 1 (Refs. 48 & 49) demonstrate that NGL's processes for the management of failed fuel (Ref. 39) lead to a precautionary approach in which reductions in reactor power and reactor shutdowns are taken in advance of required actions. I therefore conclude that making changes to the BCD action levels to require such a precautionary approach would lead to very little real safety benefit in situations where a precautionary approach is appropriate. I also judge that in some situations a precautionary approach may lead to safety dis-benefits by mandating an unnecessary reactor shutdown or de-load. I therefore conclude that the reduction in flexibility for the

management of failed fuel would lead to little safety benefit, and potential safety dis-benefit, and that the reduction of BCD action level 1 is not ALARP.

235. In considering whether a position is ALARP it is useful to consider relevant good practice. The ONR TAG on the safety of nuclear fuel in power reactors (Ref. 4) expects that adequate measures should be in place to mitigate the consequences of fuel failures, that the release of activity into the coolant should be detected, and that procedures should be followed to ensure that the dispersal of nuclear material is minimised. In my view NGL has demonstrated that the release of activity into the coolant would be detected and that the management of failed fuel process would act to minimise the dispersal of nuclear material. I therefore judge that NGL has met this relevant good practice.

236. NGL states that at present it is not clear how to distinguish between a more typical minor fuel failure at a high power location, and fuel clad melt at the element 1 position, as both could lead to BCD signals of similar magnitudes. NGL has specified a study to determine if fuel clad melt at element 1 would be discernible from more typical minor fuel failures in higher power elements. I support this initiative as if fuel clad melt at element 1 could be readily identified then appropriate action could be taken as part of NGL's management of failed fuel procedures which may provide a reduction in risk without the associated risk dis-benefits; in my view this is an ALARP measure by NGL. Although NGL have initiated this work on their own I have raised a regulatory issue (issue 7292) to track the progress of the work, and ensure that the potential benefits of the work are realised. In my opinion it would be grossly disproportionate to withhold consent to the restart of HNB reactor 4 until the conclusion of this work, but I expect that it should be available for consideration for the next HNB R4 safety case at the end of the proposed 4 month JPSO.

237. The BCD action levels discussed here are from the generic failed fuel safety case (Ref. 29), which is a fleet wide safety case which is intended for implementation at all AGR stations. The changes to Technical Specification action levels as specified in the generic failed fuel safety case (Ref. 29) have not yet been implemented at HNB, and thus I recommend that these be implemented prior to restart of HNB R4, noting that the proposed changes represent a conservative position relative to the existing BCD action levels, and thus should reduce risk.

Recommendation 2: The changes to Technical Specification 8.1.3 proposed in NP/SC 7653 should be implemented at Hunterston B prior to restart of Reactor 4.

<u>Element 6 blockage</u>

238. The discussion in this section of the report has thus far focussed on the potential for graphite debris to block the element 1 support grid. In the following discussion I shall consider the potential effects of graphite debris causing blockages at the peak rated (highest power) location in the fuel stringer. The concern here is that a blockage at the element 6 position has the potential to lead to fuel clad melt, which would result in a release of activity into the coolant, the potential consequences and escalation paths of this would then be comparable to those discussed previously for a blockage at the element 1 position.

239. Reference 35 presents the results of analysis based on rig tests of the effects of debris at the peak rated location in a fuel stringer - the element 6 top brace (see figure 2). Reference 35 examines the effects of a blockage of an entire section of the top brace (figure 7) and determines that the fuel clad temperature remains below the Technical Specification Limit (870°C) at the peak rated location, accounting for the uncertainties in the analysis.

240. The blockage of an entire section of the top brace would require a piece of debris too large to fit through any brace. In order to reach the top brace of element 6 a piece of debris would have to pass through 17 braces and support grids, and thus a piece of debris such as that analysed in Reference 35 could not make its way to the element 6 top brace. Reference 35 additionally analysed the effects of a piece of debris which would be small enough to fit through a brace, and concluded that this would result in a 6°C increase in the fuel clad temperature if it were to become lodged at the element 6 top brace, giving large margins to the Technical Specification limit.

241. The fuel clad temperature increases calculated in Reference 35 do not account for the reduction in reactor power to 80% at Hunterston B, and thus the margins would be even greater than presented. I additionally note that the probability of a piece of debris becoming lodged at the element 6 top brace is likely to be very small. If the debris is prone to becoming lodged at a brace then it is likely to become lodged at one of the 17 previous braces and grids that it would have to pass through, at which the effects on fuel clad temperature would be smaller than those discussed. If the debris is not prone to becoming lodged at a brace then it is not likely to become lodged in the element 6 top brace, and is likely to pass completely through the fuel stringer.

242. In my view the results presented in Reference 35 are clearly conservative and adequately demonstrate that the Technical Specification limit on the fuel clad temperature would not be threatened by debris at the peak rated location.

4.2.4 Section Summary

243. Argument 1.4 of NGL's safety case (Ref. 1) states that the safety functions of the graphite core will be fulfilled in the presence of graphite brick cracking. In this section (4.2 and sub-sections) I have examined the evidence that NGL has presented in support of demonstrating that the safety function of directing gas flows to ensure adequate cooling of the fuel and core has been met.

244. I examined the evidence presented by NGL to demonstrate that the increased coolant flow from the arrow head passageways to the fuel channel annulus due to graphite brick cracking is acceptable. I judged that NGL has satisfactorily demonstrated that the effect on fuel temperatures is small and negative, and that NGL has adequately demonstrated that the thermally induced stresses in the fuel sleeve would likely be reduced. I therefore conclude that Argument 1.4 of Reference 1 can be supported as it relates to arrow head to annulus flows.

245. In assessing the effects of core distortion, I concluded that the evidence presented by NGL adequately demonstrates that the effects of eccentricity of the fuel channel annulus on fuel temperatures are acceptable, and that there is no threat to the integrity of the fuel sleeve.

246. In coordination with the ONR graphite specialist inspector I examined the evidence presented by NGL to demonstrate that the effects of fuel sleeve gapping due to core distortion are acceptable. I concluded that NGL has adequately demonstrated that the operational limits on fuel clad temperature will not be breached and that the tiebar failure probability will not be significantly altered during the proposed JPSO.

247. From my considerations of the effects of core distortion on the capability of the core to fulfil its safety function of directing gas flows to ensure cooling of the fuel and core, I conclude that Argument 1.4 of Reference 1 can be supported.

248. In coordination with the ONR graphite specialist inspector I pressed NGL to produce evidence to demonstrate that the effects of blockages in the fuel stringer due to graphite debris were acceptable. I examined the evidence provided by NGL and concluded that although NGL did not demonstrate that all barriers to a radiological

release were preserved, there was sufficient evidence to determine that the resultant mitigated risk was acceptable as assessed against ONR's risk targets (SAP NT.4 & NT.8).

249. I judged that there is significant uncertainty associated with the point at which fuel clad melt would occur following a blockage at the element 1 support grid. The production of graphite debris is likely to increase as the core ages, and thus the issue of flow obstruction due to graphite debris is likely to become more significant in future safety cases. I therefore judge that NGL should reduce the uncertainties associated with the consequences of a blockage and made a recommendation in this regard:

Recommendation 1: For inclusion in future safety cases justifying the operation of the Hunterston B Reactor 4 graphite core, NGL should perform further analysis of the effects of a blockage at the element 1 support grid in order to establish the point at which fuel clad melt temperatures would be reached.

250. I considered NGL's analysis of whether there were any reasonably practicable improvements which could be made to reduce the risk associated with graphite debris, and concluded that NGL has considered all potential measures to reduce risk, and demonstrated that there were no reasonably practicable measures which could be taken. I made a recommendation that NGL should implement the technical specification changes proposed by the generic failed fuel safety case prior to the restart of Reactor 4:

Recommendation 2: The changes to Technical Specification 8.1.3 proposed in NP/SC 7653 should be implemented at Hunterston B prior to restart of Reactor 4.

251. In summary, I judge that NGL have demonstrated that the nuclear safety function of the graphite core to direct gas flows to ensure adequate cooling of the fuel and core will be adequately fulfilled over the proposed 4 month JPSO, and that Argument 1.4 in this regard can be supported. I additionally judge that NGL have taken all reasonably practicable measures to reduce the risk associated with disrupted coolant flows, and that in this regard, Argument 2.2 can be supported.

4.3 Provide neutron moderation and thermal inertia

252. There is no plausible effect on the thermal inertia of the graphite core due to graphite brick cracking. Graphite weight loss does affect the mass of the graphite core, and thus its thermal inertia, however the effects of graphite weight loss are outside of the scope of this report, and there is no change to the limit on graphite weight loss proposed in NGL's safety case (Ref. 1).

253. The graphite bricks provide moderation of the neutrons, and I considered that it was plausible that the presence of cracks in the graphite bricks could lead to a radial asymmetry in the thermal neutron flux in a fuel channel. I asked NGL whether the potential for this effect had been considered, and whether it would have any effect on the assumptions of reactivity faults in which the symmetry of channel power is a factor (Ref. 10).

254. In response NGL stated that the small scale moderator variation would have a negligible effect on the thermal flux distribution as the neutron slowing down length in an AGR is much larger than the length of graphite brick cracks, and thus the potential effect could be discounted. I judge that this is a logical argument, and conclude that the presence of graphite brick cracks would not have any significant effect on the neutron moderation provided by the graphite core.

255. In summary I judge that NGL have adequately demonstrated that the safety function of the graphite core to provide neutron moderation and thermal inertia is unaffected by

the presence of graphite brick cracking, and thus Argument 1.4 of NGL's safety case (Ref. 1) can be supported.

4.4 Matters from Previous Assessment

256. ONR's fault studies assessment (Ref. 44) of NGL's previous graphite safety case for HNB (Ref. 9) raised two matters which it recommended should be addressed.

257. The first matter raised by Reference 44 was that:

EDF NGL should provide further information at what potential level debris generation could present a safety concern and identify what the predicted level of debris generation could be within the limits of the safety case.

258. NGL's safety case (Ref. 1) did not explicitly state the level of debris generation that would lead to a safety concern. It is clear that the level of debris currently being generated leads to potential safety consequences within the design basis, and this is then addressed in Reference 21, as discussed in section 4.2.3. The level of debris generation that could be tolerated within the safety case is not clear; this has been considered by the graphite inspector, who concluded that the level of debris generation over the proposed 4 month operating period had been adequately determined by NGL such that the initiating fault frequency could be bounded (Ref. 32 & 45). I therefore conclude that the matter raised by Reference 44 was adequately addressed.

259. The second matter raised by Reference 44 was that:

EDF NGL should improve the overall structure of future graphite safety cases to ensure that all safety arguments are clearly articulated and linked to the safety functions. This should include but not be limited to explicit consideration of post trip cooling and cooling post a design basis seismic event.

260. NGL's safety case (Ref. 1) was structured around ensuring that the safety functions of the graphite core were fulfilled. However the consideration of cooling following a seismic event was limited to consideration of the integrity of the fuel sleeve, and did not consider the effects on fuel sleeve gapping. In my opinion due to the low control rod channel distortions predicted following a seismic event, and the evidence presented by NGL to demonstrate tolerance to sleeve gapping at power, the lack of discussion of sleeve gapping during post-trip cooling following a seismic event does not present an issue for permissioning of this safety case. However I judge that it should be explicitly addressed in future graphite safety cases.

Recommendation 3: NGL should include consideration of fuel channel distortions following a seismic event and its effect on fuel sleeve gapping in future graphite safety cases.

4.5 ONR Assessment Rating

261. In my opinion significant regulatory intervention and guidance was needed through the assessment of NGL's safety case (Ref. 1), and many technical issues were raised which required regulatory follow-up. As such I judge that NGL's safety case (Ref. 1) should be given a rating of Amber according to ONR's assessment rating guide (Ref. 8).

4.6 EDF NGL's Due Process

262. NGL's safety case (Ref. 1) has been through the company's process for the production and review of safety cases. The verification statement included within Reference 1 makes clear that the safety case was reviewed and verified by an independent team of suitably qualified and experienced people. The verification team concluded that the

safety case (Ref. 1) justified operation of HNB reactor 4 for the limited 4 month operating period.

263. I note that the temperature effects of a flow obstruction due to graphite debris was not considered in the verification statement, and that this is likely due to the lack of attention that the topic was given within the safety case. NGL produced a subsequent report (Ref. 21) which examined this area in detail. In my view the verification statement for this report (Ref. 21) demonstrated that significant thought had been put into the verification, and that the verification team had clearly maintained independence from the production team. The verification statement for Reference 21 was supportive of the conclusions of the paper.

264. Reference 1 was reviewed by NGL's Independent Nuclear Safety Assessment team (Ref. 42). The independent assessment of NGL's safety case was performed by a group of independent Nuclear Safety Engineers who considered that the safety case was acceptable. The independent assessment concluded that the further safety justification presented in Reference 21 could be supported.

265. NGL's safety case (Ref. 1) was reviewed by the NGL Hunterston B Nuclear Safety Committee, which provides independent expert advice nad includes some members who are external to NGL. The Nuclear Safety Committee was supportive of the safety case (Ref. 43). I note that the Nuclear Safety Committee was presented with version 7 of the safety case, whereas the safety case was subsequently updated to version 11. The later revision of the safety case (Ref. 1) included reference to Reference 21 of this report, but did not change the high level principles of the safety case which are the main points of consideration of the NSC, and therefore I accept that there would have been little value in returning to the NSC.

266. In my view NGL have satisfactorily followed the company due process, and I note that several independent reviews (verification, INSA, NSC) concluded that the safety case could be supported. I note that the safety case as originally submitted did not adequately consider the effects of graphite debris on fuel clad temperatures, and that this is reflected in the lack of discussion in the original reviews. Following the issue of Reference 21 further verification and Independent Nuclear Safety Assessment reviews considered the effects of graphite debris, however it could be considered a failing of the original reviews that the issue was not raised.

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

267. This report presents the findings of the fault studies assessment of the Hunterston B Power Station (HNB) return to service safety case for reactor 4 (R4) following core inspection results in 2018 (Ref. 1) and supporting documentation provided by EDF NGL.

268. I have focussed my assessment on determining whether EDF NGL have adequately demonstrated that the safety functions of the graphite core will be fulfilled over the proposed 4 month Justified Period of Safe Operation (JPSO). My assessment has therefore concentrated on the arguments and evidence set out by EDF NGL in Argument 1.3, 1.4 & 2.2 of their safety case (Ref. 1) in support of demonstrating this.

269. The main potential faults/hazards considered in this report were:

- failure to shut down the reactor due to core distortion
- increased snagging of fuel stringers during fuel movement
- disruption in coolant flow paths due to graphite brick cracking and core distortion
- impaired cooling due to graphite debris

270. Of the faults considered, the failure to shut down the reactor has the greatest potential consequences (DB5) but I conclude that there are sufficiently large margins remaining in the analysis. The other faults considered have potential consequences which are unlikely to exceed DB3. I conclude that the risks associated with an increase in the frequency of snagging of fuel are tolerable with large margins to the SAPs numerical targets, and that there are large margins in the safety analysis of the effects of disrupted coolant flow paths in the core. I conclude that the effects of graphite debris are the least well controlled, but that all reasonably practicable measure to reduce risk have been taken.

271. I concluded that the validation of the effects of fuel sleeve gapping was adequate up to ~5mm, but that should sleeve gaps >4mm be predicted in future safety cases then further validation work would be required.

272. From my assessment of the potential effects of graphite debris forming a partial blockage within a fuel stringer I concluded that there is significant uncertainty associated with the point at which fuel clad melt would occur following a blockage at the element 1 support grid and made Recommendation 1 in this regard.

273. From my assessment of the potential measures which could be employed by EDF NGL to reduce the risk associated with graphite debris forming partial blockages within the fuel stringer, I concluded that the EDF NGL should implement the changes to operating limits proposed by the Generic Failed Fuel Safety Case (Ref. 29) prior to the restart of reactor 4. I made Recommendation 2 in this regard.

274. I conclude that NGL has demonstrated that the nuclear safety functions of the graphite core to:

- allow unimpeded movement of control rods and fuel,
- to direct gas flows to ensure adequate cooling of the fuel and core,
- to provide neutron moderation and thermal inertia,

will be adequately fulfilled over the proposed 4 month JPSO, and that Arguments 1.3 & 1.4 can be supported in this regard. I also conclude that EDF NGL has taken all reasonably practicable measures to reduce the risk associated cracking of the graphite core, and thus that Argument 2.2 can be supported.

275. Following my assessment of EDF NGL's safety case for the return to service of Hunterston reactor 4 (Ref. 1), I conclude that, from a fault studies perspective, ONR should agree to the modifications to the safety case described in NP/SC 7785 (Ref. 1) once my Recommendation 2 has been addressed.

5.2 Recommendations

276. My recommendations are as follows.

- **Recommendation 1**: For inclusion in future safety cases justifying the operation of the Hunterston B Reactor 4 graphite core, NGL should perform further analysis of the effects of a blockage at the element 1 support grid in order to establish the point at which fuel clad melt temperatures would be reached.
- **Recommendation 2**: The changes to Technical Specification 8.1.3 proposed in NP/SC 7653 should be implemented at Hunterston B prior to restart of Reactor 4.
- **Recommendation 3**: NGL should include consideration of fuel channel distortions following a seismic event and its effect on fuel sleeve gapping in future graphite safety cases.

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Table 2

Relevant Safety Assessment Principles Considered During the Assessment

SAP No	SAP Title	Description
FA.4	Fault tolerance	DBA should be carried out to provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety measures.
FA.5	Initiating faults	The safety case should list all initiating faults that are included within the design basis analysis of the facility.
FA.6	Fault sequences	For each initiating fault within the design basis, the relevant design basis fault sequences should be identified.
FA.7	Consequences	Analysis of design basis fault sequences should use appropriate tools and techniques, and be performed on a conservative basis to demonstrate that consequences are ALARP.
FA.8	Linking of initiating faults, fault sequences and safety measures	DBA should provide a clear and auditable linking of initiating faults, fault sequences and safety measures.
FA.14	Use of PSA	PSA should be used to inform the design process and help ensure the safe operation of the site and its facilities.
AV.1	Theoretical models	Theoretical models should adequately represent the facility and site.
AV.2	Calculation methods	Calculation methods used for the analyses should adequately represent the physical and chemical processes taking place.
AV.3	Use of data	The data used in the analysis of aspects of plant performance with safety significance should be shown to be valid for the circumstances by reference to established physical data, experiment or other appropriate means.
AV.4	Computer models	Computer models and datasets used in support of the safety analysis should be developed, maintained and applied in accordance with quality management procedures.
AV.6	Sensitivity studies	Data should be collected throughout the operating life of the facility to check or update the safety analysis.
ERC.1	Design and operation of reactors	The design and operation of the reactor should ensure the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor.

ERC.2	Shutdown systems	At least two diverse systems should be provided for shutting down a civil reactor.
ERC.3	Stability in normal operation	The core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their permitted range.
EHT.2	Coolant inventory and flow	Sufficient coolant inventory and flow should be provided to maintain cooling within the limits (operating rules) derived for normal operational and design basis fault conditions.
SC.5	Optimism, uncertainty and conservatism	Safety cases should identify areas of optimism and uncertainty, together with their significance, in addition to strengths and any claimed conservatism.
ESS.8	Automatic initiation	For all fast acting faults (typically less than 30 minutes) safety systems should be initiated automatically and no human intervention should then be necessary to deliver the safety function(s).
EHF.5	Task analysis	Proportionate analysis should be carried out of all tasks important to safety and used to justify the effective delivery of the safety functions to which they contribute.

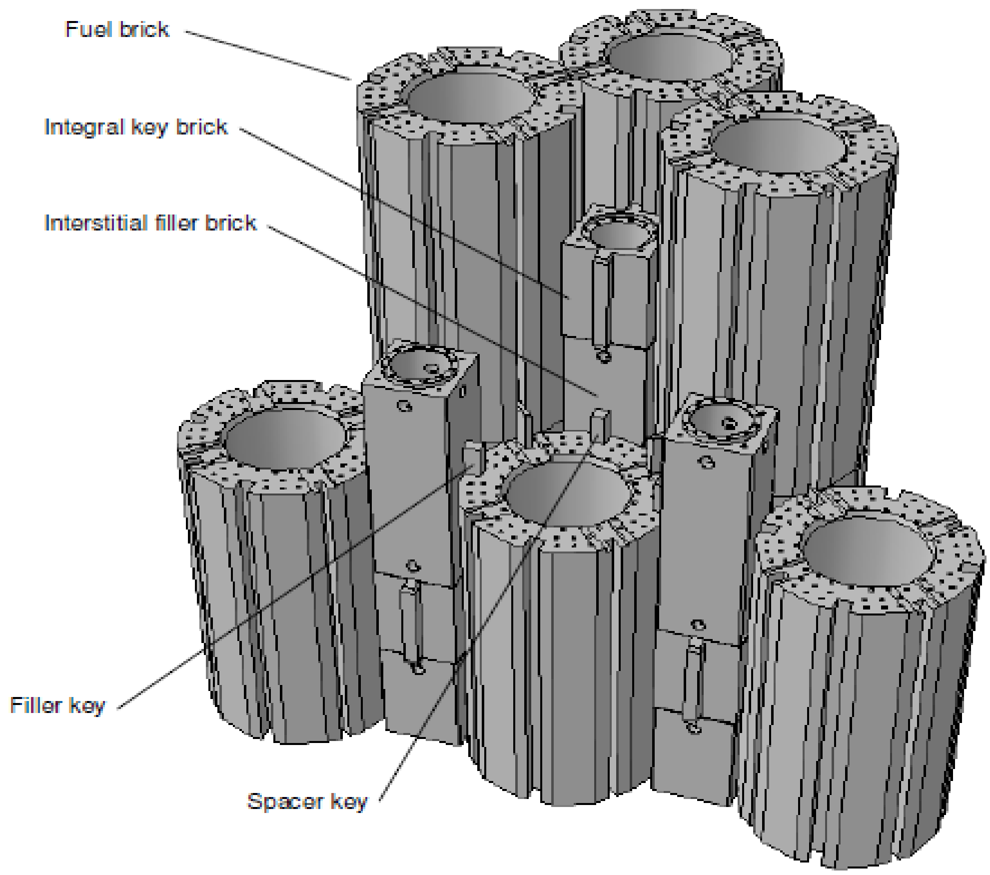

Figure 1 – Illustration of the fuel and interstitial bricks and the keying system at Hinkley Point B and Hunterston B

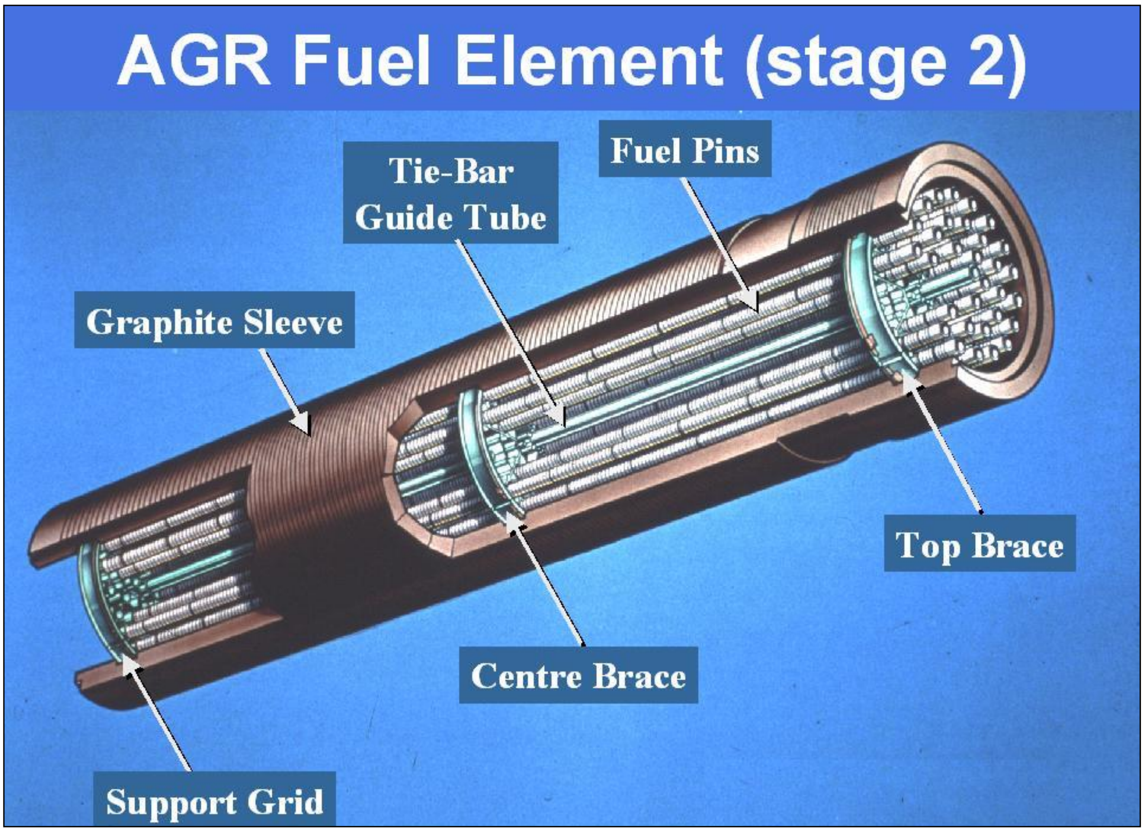

Figure 2 – A Stage 2 Fuel Element

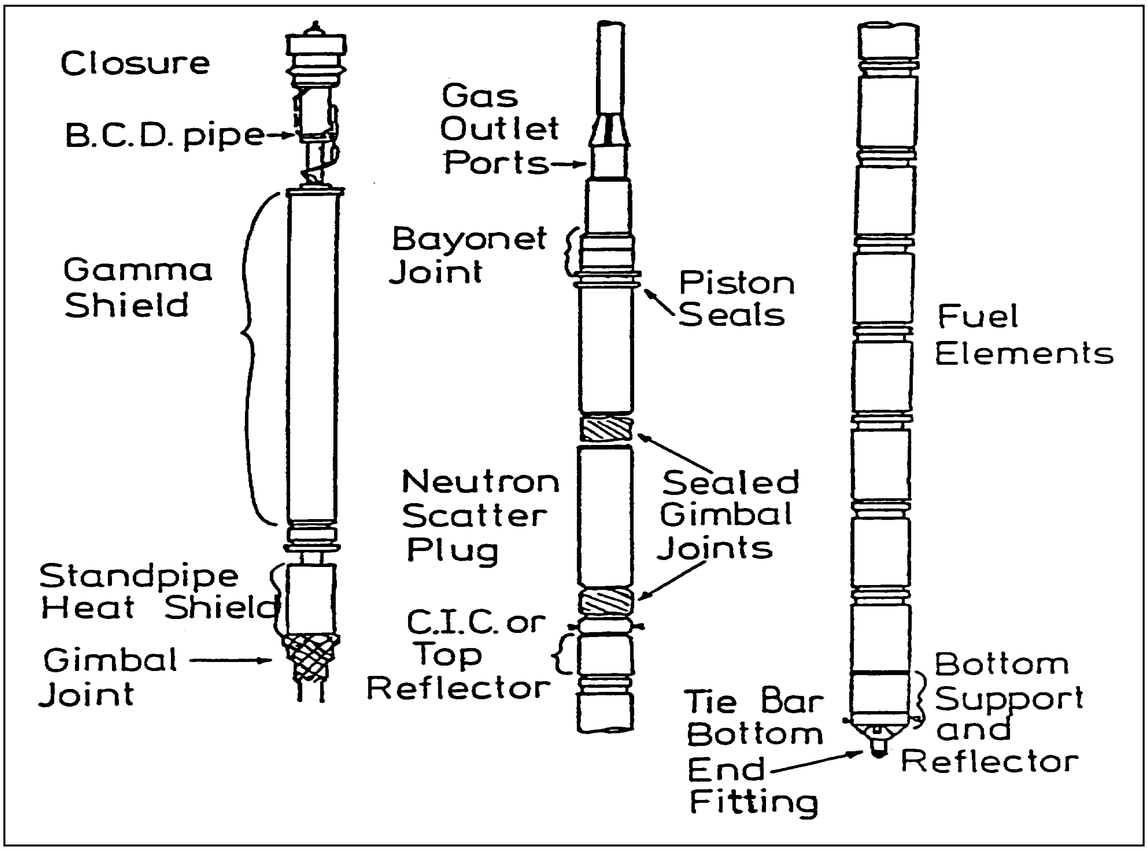

Figure 3 – Fuel Stringer

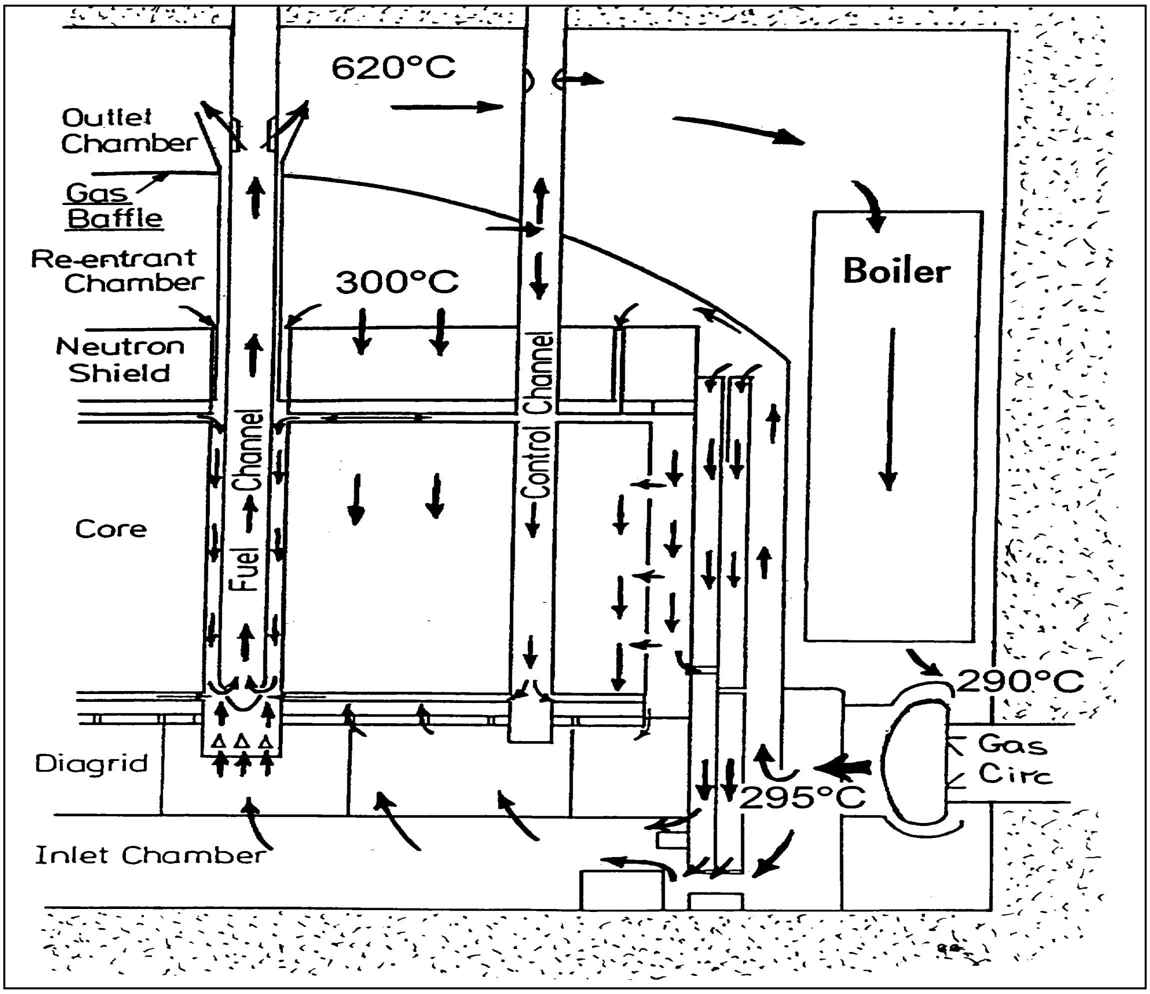

Figure 4 - Typical AGR Reactor Gas Coolant Flow (whole core)

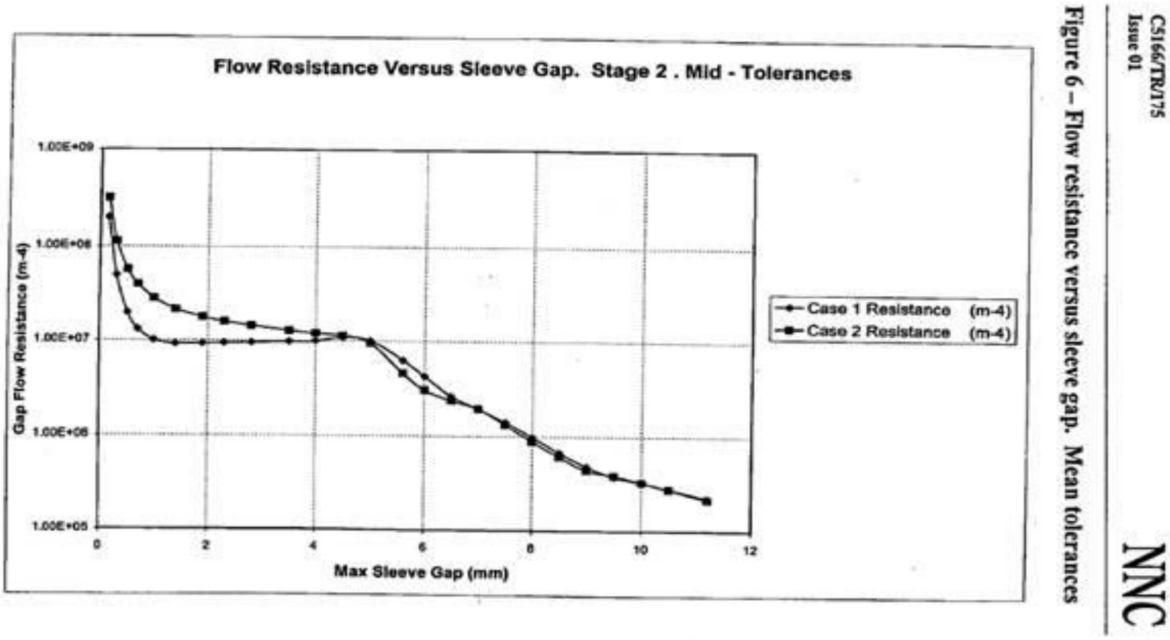

Figure 5 – Flow resistance as a function of sleeve gap size

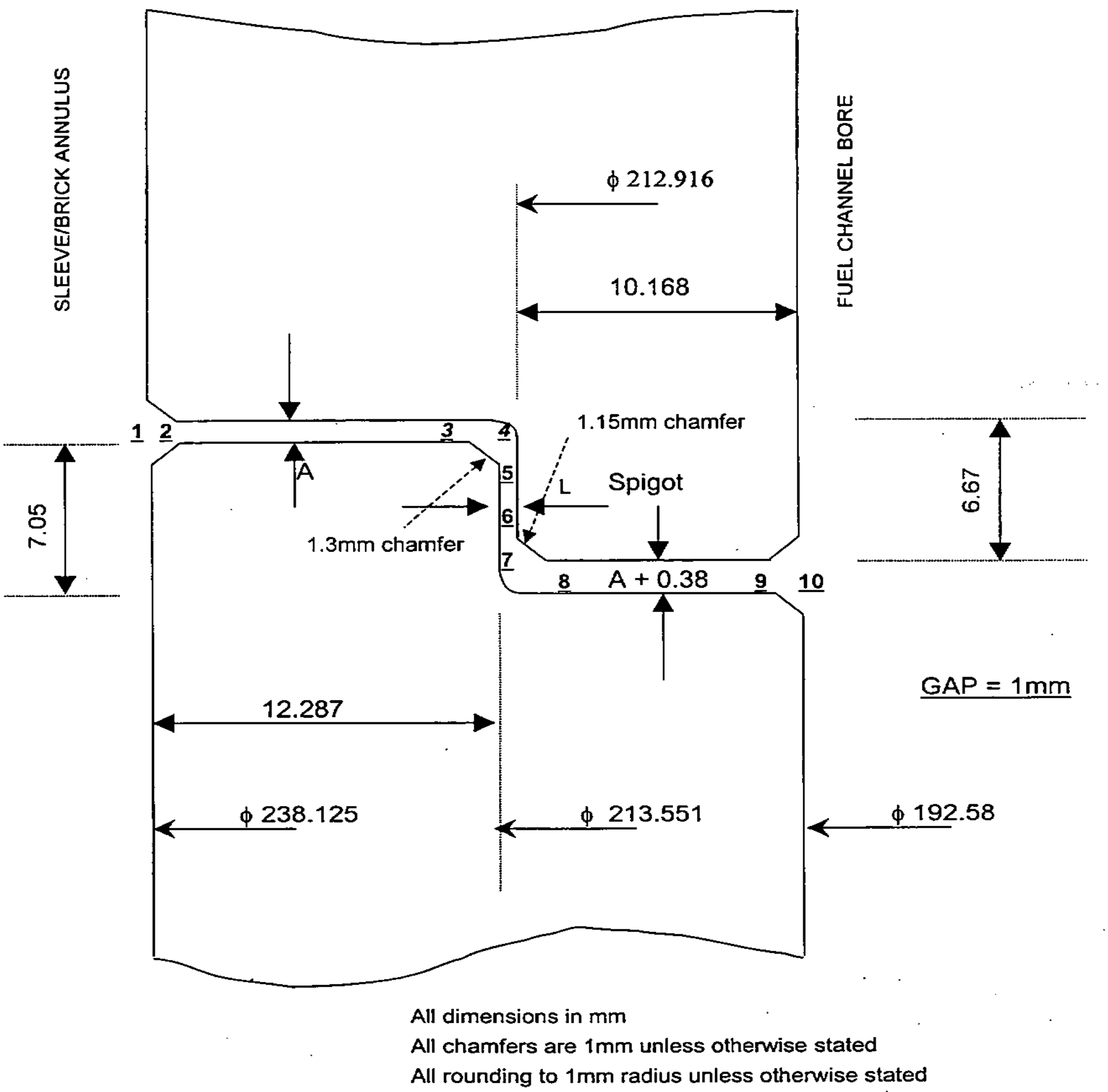

Figure 6 – Fuel sleeve end geometry

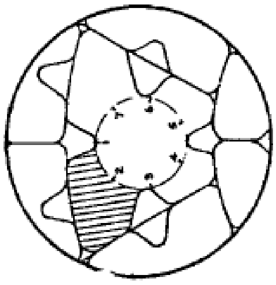

Figure 7 – Blockage of a fuel brace cell

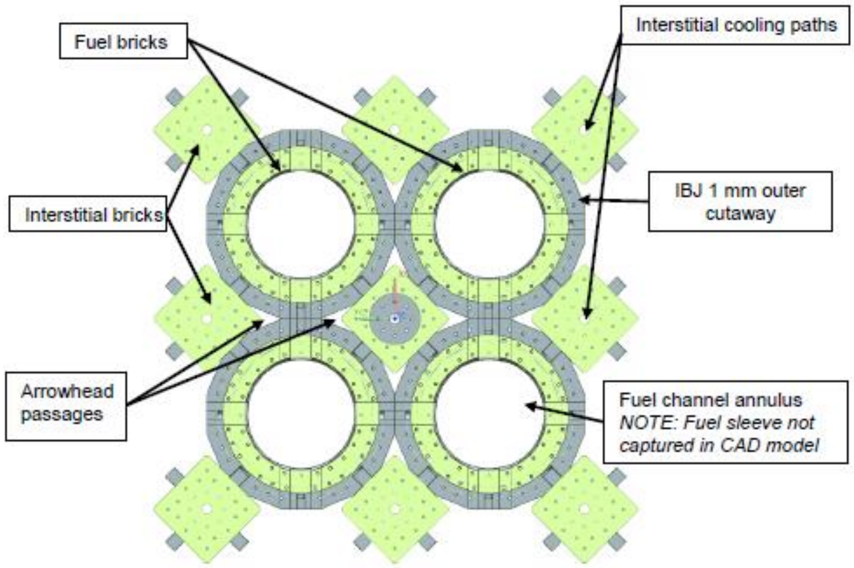

Figure 8 – Plan view of graphite core showing arrowhead passages, and annulus with no fuel stringer in-situ