New Reactors Programme

GDA close-out for the AP1000 reactor

GDA Issue GI-AP1000-FS-05 – Potential Enhancements to the Diverse Safety Injection System

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EXECUTIVE SUMMARY

Westinghouse Electric Company LLC (Westinghouse) is the reactor design company for the **AP1000**® reactor. Westinghouse completed Generic Design Assessment (GDA) Step 4 in 2011 and paused the regulatory process. It achieved an Interim Design Acceptance Confirmation (IDAC) which had 51 GDA Issues attached to it. These issues require resolution prior to the award of a Design Acceptance Confirmation (DAC) and before any nuclear safety-related construction can begin on site. Westinghouse re-entered GDA in 2014 to close the 51 GDA Issues.

This report is the Office for Nuclear Regulation's (ONR's) assessment of the Westinghouse **AP1000** reactor design in the area of fault studies. Specifically, this report addresses GDA Issue GI-AP1000-FS-05 – Potential Enhancements to the Diverse Safety Injection System.

This GDA Issue arose in GDA Step 4 following a review of a preliminary evaluation performed by Westinghouse of the ability of the **AP1000** reactor to respond to small-break Loss of Coolant Accidents (LOCAs) assuming a common mode failure of Class 1 safety measures. Westinghouse identified a role for the Class 2 Normal Residual Heat Removal System (RNS) in the safety case to protect against such faults. The evaluation showed that the RNS could help to ensure that appropriate safety criteria are met, but ONR challenged Westinghouse to demonstrate if further enhancements could be beneficial to safety while also being reasonably practicable. This demonstration was not provided in GDA Step 4. As a result, ONR raised GI-AP1000-FS-05.

To close this GDA Issue, Westinghouse has submitted a report that:

- defines the size and location of a small-break LOCA;
- identifies a number of fault sequences to analyse, each claiming different combinations of safety systems, to show the capability of the extant **AP1000** design;
- identifies and reviews the merits of several potential enhancements to the design, informed by the analysis results;
- concludes that enhancements to the RNS would be grossly disproportionate when compared against the small safety benefits they could bring; and
- recommends a change to the Diverse Actuation System (DAS) which would have a safety benefit following a small-break LOCA and is reasonably practicable to implement.

Westinghouse has also updated the **AP1000** Pre-Construction Safety Report (PCSR) in response to this GDA Issue to provide a safety case for small-break LOCA faults that is consistent in scope with UK relevant good practice.

My assessment conclusion is that Westinghouse has made supportable judgements on what enhancements to the RNS are reasonably practicable. I also judge the updates to the PCSR to be adequate.

I have reached these conclusions following a detailed review of Westinghouse's main submission and its supporting references. I am satisfied with how Westinghouse has defined a small-break LOCA and it is my judgement that Westinghouse's transient analyses to model the behaviour of the plant following the fault do demonstrate the adequacy of the extant RNS design. By extension, I am content that the transient analyses adequately support its conclusions not to modify the RNS, while also demonstrating the merits of adding extra functionality to the DAS.

In summary, I am satisfied that GDA Issue GI-AP1000-FS-05 can be closed.

LIST OF ABBREVIATIONS

ADS	Automatic Depressurisation System
ALARP	As Low As Reasonably Practicable
ATWS	Anticipated Transient Without Scram
BSL	Basic Safety Level
BSO	Basic Safety Objective
BWR	Boiling Water Reactor
C&I	Control and Instrumentation
CCS	Component Cooling Water System
CI	Containment Isolation
CLP	Cask Loading Pit
CMF	Common Mode Failure
CMT	Core Make-up Tank
CVS	Chemical and Volume Control System
CVCS	Chemical and Volume Control System (Sizewell B nomenclature)
DAC	Design Acceptance Confirmation
DAS	Diverse Actuation System
DCP	Design Change Proposal
DVI	Direct Vessel Injection
EDCD	European Design Control Document
GDA	Generic Design Assessment
GRS	Gesellschaft für Anlagen und Reaktorsicherheit
IAEA	International Atomic Energy Agency
IDAC	Interim Design Acceptance Confirmation
IRWST	In-Containment Refuelling Water Storage Tank
LOCA	Loss of Coolant Accident
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PCS	Passive Containment Cooling System
PCSR	Pre-Construction Safety Report
PLS	Plant Control System
PMS	Protection and Monitoring System
PRHR	Passive Residual Heat Removal

PSA	Probabilistic Safety Analysis
PSV	Pressuriser Safety Valve
PWR	Pressurised Water Reactor
RCCA	Rod Cluster Control Assembly
RCS	Reactor Coolant System
RNS	Normal Residual Heat Removal System
RQ	Regulatory Query
SAP	Safety Assessment Principle
SFW	Startup Feedwater System
SGTR	Steam Generator Tube Rupture
SSC	Structures, Systems and Components
SWS	Service Water System
TAG	Technical Assessment Guide
US NRC	United States Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators Association

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1 INTRODUCTION

1.1 Background

1. Westinghouse Electric Company LLC (Westinghouse) is the reactor design company for the **AP1000**® reactor. Westinghouse completed Generic Design Assessment (GDA) Step 4 in 2011 and paused the regulatory process. It achieved an Interim Design Acceptance Confirmation (IDAC) which had 51 GDA Issues attached to it. These issues require resolution prior to the award of a Design Acceptance Confirmation (DAC) and before any nuclear safety-related construction can begin on site. Westinghouse re-entered GDA in 2014 to close the 51 GDA Issues.

2. This report is the Office for Nuclear Regulation's (ONR's) assessment of the Westinghouse **AP1000** reactor design in the area of fault studies. Specifically, this report addresses GDA Issue GI-**AP1000**-FS-05 – Potential Enhancements to the Diverse Safety Injection System.

3. The related GDA Step 4 report (Ref. 1) is published on our website (**www.onr.org.uk/new-reactors/ap1000/reports.htm**), and this provides the assessment underpinning the GDA Issues. Further information on the GDA process in general is also available on our website (**www.onr.org.uk/new-reactors/index.htm**).

1.2 Overview of GI-AP1000-FS-05

4. The **AP1000** reactor has been designed with a consideration of design basis events, defence-in-depth and utilising insights from Probabilistic Safety Analysis (PSA), all of which are consistent with relevant international good practice. However, early on in the original GDA fault studies interactions with ONR, Westinghouse was challenged to review all design basis initiating events with a frequency of greater than 1×10^{-3} per year and to demonstrate that a diverse safety system, qualified to an appropriate standard, is provided for each nuclear safety key function (for example, reactivity control and fuel cooling). The demonstration of diversity for 'frequent' design basis faults is long-established relevant good practice in the UK and has also been required from all the recent requesting parties submitting reactor designs for GDA determinations.

5. This challenge was captured through the Regulatory Observation RO-AP1000-47. One fault identified by Westinghouse in its response to RO-AP1000-47 (Ref. 2) as meeting the criteria for a diversity demonstration was a small-break Loss of Coolant Accident (LOCA), specifically pipe break on the cold leg up to 4 inches in diameter (10 cm). Westinghouse had already demonstrated in its European Design Control Document (EDCD) (Ref. 3) that the **AP1000** reactor has a 'principal' group of Class 1 safety measures which can ensure that relevant safety criteria are met should such a break occur. To demonstrate diversity, as part of its RO-AP1000-47 response (Ref. 2), Westinghouse supplemented EDCD analysis with new work considering two bounding small-break LOCA fault sequences, each of which took credit for the effective operation of a different combination of safety measures. One of these fault sequences claimed the manual actuation of the Class 2 Normal Residual Heat Removal System (RNS) as a means of delivering the important safety function of low-pressure water injection into the core.

6. The designers of the **AP1000** reactor had always assumed that the RNS would be capable of making a contribution to safety following a small-break LOCA event as a defence-in-depth system credited in the PSA. However, it is primarily a multi-purpose duty system intended to provide cooling and pumping functions in normal (non-fault) operations. Crediting it directly in the design basis safety case as a Class 2 safety system was a new claim for Westinghouse. The analysis of the bounding fault

sequence crediting RNS did show that applicable safety criteria could be met if the operator successfully reconfigured the RNS into a safety injection mode within 30 minutes. ONR acknowledged this in the GDA Step 4 fault studies assessment (Ref. 1) but judged that Westinghouse needed to consider whether further enhancements to diverse safety injection capability of the RNS could reduce risks As Low As Reasonably Practicable (ALARP). As a result, GDA Issue GI-AP1000-FS-05 (Ref. 4) was raised, requiring Westinghouse to identify and review the practicability of potential options to improve the RNS, including:

- automating its actuation;
- segregating its water supply from the In-Containment Refuelling Water Storage Tank (IRWST); and
- increasing its pressure head.

7. Dependent on the outcome of the ALARP reviews undertaken, the GDA Issue also required Westinghouse to incorporate any identified modifications into the UK **AP1000** design and update the Pre-construction Safety Report (PCSR) as appropriate.

1.3 Scope

8. The scope of this assessment is detailed in the assessment plan (Ref. 5). Consistent with this plan, the assessment is focused on considering whether Westinghouse's submissions to ONR for GI-AP1000-FS-05 provide an adequate response to justify the closure of the GDA Issue. As such, this report only presents the assessment undertaken as part of the resolution of the GDA Issue and it is recommended that this report be read in conjunction with the Step 4 fault studies assessment of the Westinghouse **AP1000** plant (Ref. 1) to appreciate the totality of the assessment of LOCA faults.

9. In the context of considering what improvements could be ALARP for the RNS, this assessment has focused on the means of providing safety injection and fuel cooling following a small-break LOCA that are diverse from the primary Class 1 means originally identified in the EDCD (Ref. 3). The main safety case arguments and evidence that show that there are adequate Class 1 Structures, Systems and Components (SSCs) for small-break LOCA faults have already been assessed in GDA Step 4 and judged to be adequate.

1.4 Method

10. This assessment has been undertaken consistent with internal guidance on the mechanics of assessment within ONR (Ref. 6).

2 ASSESSMENT STRATEGY

2.1 Pre-Construction Safety Report

11. ONR's GDA guidance to requesting parties (Ref. 7) states that the information required for GDA may be in the form of a Pre-Construction Safety Report (PCSR), and the Technical Assessment Guide (TAG) NS-TAST-GD-051 sets out regulatory expectations for a PCSR (Ref. 8).

12. At the end of Step 4, ONR and the Environment Agency raised GDA Issue GI-AP1000-CC-02 (Ref. 9) requiring that Westinghouse submit a consolidated PCSR and associated references to provide the claims, arguments and evidence to substantiate the adequacy of the **AP1000** design reference point.

13. A separate regulatory assessment report is provided to consider the adequacy of the PCSR and closure of GDA Issue GI-AP1000-CC-02, and therefore this report does not attempt to assess the totality of the fault studies safety case presented in the PCSR, including the aspects relevant to LOCAs. However, a key requirement of this GDA Issue is for the primary safety case arguments made for the 'standard' **AP1000** reactor in the EDCD to be supplemented with a UK-specific demonstration of diversity for frequent small-break LOCA faults and a justification of why the design is ALARP. Westinghouse has stated, as part of its strategy for addressing GI-AP1000-CC-02, that it will no longer maintain the EDCD. Instead, the UK **AP1000** safety case will be centred on the consolidated PCSR. This means that, as part of my assessment of GI-AP1000-FS-05, I have needed to consider the adequacy of sections of the PCSR, notably Chapter 9 (Section 9.6 covers decrease in reactor coolant inventory faults) and Chapter 8 (the fault schedule).

2.2 Standards and Criteria

14. The assessment has been undertaken in line with the requirements of the HOW2 BMS document NS-PER-GD-014 (Ref. 10). In addition, the Safety Assessment Principles (SAPs) (Ref. 11) constitute the regulatory principles against which dutyholders' safety cases are judged, and, therefore, are the basis for ONR's nuclear safety assessment. When performing the assessment described in this report, I have used SAPs 2014 Edition (Revision 0); the original GDA Step 4 fault studies assessment used the 2006 Edition.

2.2.1 Safety Assessment Principles and Technical Assessment Guides

15. The assessment plan (Ref. 5) identified the following SAPs (Ref. 11) as being appropriate to judge the adequacy of the arguments in the area of fault studies for the UK **AP1000** reactor:

- Fault Analysis SAPs FA.1 to FA.9
- Severe Accidents SAPs FA.15 and FA.16
- Engineering SAPs EKP.2 to EKP.5, ECS.1, ECS.2, EDR.1 to EDR.4, ESS.2, ESS.4, ESS.6 to ESS.9, ESS.11, ERC.1 to ERC.3, EHT.1 to EHT.4
- Computer Codes and Calculation Methods SAPs AV.1 to AV.8
- Numerical Target for DBA Consequences Target 4

16. It is important to note, however, that the scope of the assessment to close out the GDA Issue is narrowly defined and is less than that of a typical ONR assessment, such as that undertaken in GDA Step 4. Three fault analysis SAPs, FA.6, FA.7 and FA.8, which set expectations for design basis analysis, have been of most relevance to this scope-constrained assessment.

17. My expectations for judging the adequacy of Westinghouse's ALARP considerations have been informed by ONR's TAG NS-TAST-GD-005 which provides guidance to inspectors on the topic (Ref. 30). I have also taken cognisance of paragraphs 698 and 701 of the SAPs, which establish a link between the graded approach to ALARP when considering potential enhancements and the (radiological) consequences predicted by design basis transient analysis.

2.2.2 National and International Standards and Guidance

18. There are both International Atomic Energy Agency (IAEA) standards (Ref. 12) and Western European Nuclear Regulators Association (WENRA) reference levels (Ref. 13) which are relevant to the fault studies assessment of the **AP1000** reactor. The original GDA fault studies assessment undertaken during Steps 3 and 4 took cognisance of the international standards published at the time. The GDA Issues that emerged from that original assessment can generally be characterised as having their origins in the application of the SAPs and UK relevant good practice rather than through the comparison against international guidance. Therefore, the SAPs (and not the international references) are the foremost standards considered. It should be noted that the latest version of the SAPs (Ref. 11) were benchmarked against the extant IAEA and WENRA guidance in 2014.

2.3 Use of Technical Support Contractors

19. No Technical Support Contractors have been used directly in support of this GI-AP1000-FS-05 assessment.

20. As part of the work to close out GI-AP1000-FS-02 (Ref. 14), ONR placed a contract with the German company Gesellschaft für Anlagen und Reaktorscherheit (GRS) to review the applicability of a new generation of design basis transient analyses that Westinghouse has proposed for inclusion in the PCSR. Under this contract, GRS reviewed the impact of modifications made to the design since ONR did its original GDA Step 4 assessment of small-break LOCA faults, as well as the impact of various changes in analysis methods and assumptions (Ref. 15).

21. In the case of small-break LOCA faults, GRS concluded that there had been some non-trivial changes to the analysis methods (notably a change in the assumed limiting single failure). However, it concluded that ONR's GDA Step 4 assessment conclusions on small-break LOCA faults remained valid, and that Westinghouse's new analyses are appropriate for the extant design reference point (Ref. 16).

22. I have therefore undertaken this assessment on the basis that Westinghouse's analyses of small-break LOCA faults have used methods that were judged by ONR in GDA Step 4 to be adequate (Ref. 1) and remain appropriate for the latest design reference point (Refs 14 and 15).

2.4 Integration with Other Assessment Topics

23. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot, therefore, generally be carried out in isolation as there are often safety issues of a multi-topic or cross-cutting nature. As part of this assessment, I have consulted colleagues who are specialists in the areas of PSA and structural integrity.

24. The initiating event frequencies attributed to small-break LOCAs are of relevance to both the design basis and PSA safety cases, and therefore I have checked for consistency in both the sources and application of pipe failure data across the two disciplines.

25. A key objective for the safety features included in the **AP1000** design for managing the consequences of a small-break LOCA is to provide a means to reduce the Reactor Coolant System (RCS) pressure sufficient for low-pressure safety injection to be effective. This is principally achieved by the four-stage Automatic Depressurisation System (ADS). The design intent is that motor-operated valves of the ADS Stages 1 to 3 help to bring the RCS pressure down 1.4 MPa (200 psi), at which point the squib valves of ADS Stage 4 are expected to function. However, as part of its diversity demonstration, Westinghouse has identified sequences where the ADS Stage 4 squib valves are assumed to open at higher RCS pressures. To come to a judgement on the adequacy of Westinghouse's arguments associated with this change in actuation pressure, I obtained advice from colleagues who specialise in the structural integrity topic area.

2.5 Out of Scope Items

26. As I stated in Subsection 2.3, I have undertaken this assessment assuming that Westinghouse's analysis methods for small-break LOCA faults are adequate. I have not attempted to repeat the assessment made in GDA Step 4 on Westinghouse's computer codes (Ref. 1).

3 REQUESTING PARTY'S DELIVERABLES IN RESPONSE TO THE GDA ISSUE

27. Westinghouse's main submission is a single consolidated report covering the full scope of the GDA Issue (Ref. 17). The approach this submission takes, the conclusions it reaches on what enhancements to the RNS are ALARP, and the resulting UK-specific safety case arguments it establishes for small-break LOCA faults are described in Subsection 3.1 below.

28. In addition, Westinghouse has incorporated a summary of the final safety case for small-break LOCA faults into a consolidated version of the PCSR (Ref. 18). An overview of what Ref. 18 provides is given in Subsection 3.2 below.

3.1 Overview of Westinghouse's Response to GI-AP1000-FS-05

29. In Ref. 17, Westinghouse does not immediately start reviewing enhancements to the RNS. Instead, it starts by defining what constitutes a small-break LOCA, and then it identifies a number of sequences to analyse, each claiming different combinations of SSCs. Only once it has characterised the role that the RNS plays in providing a diverse capability for safely managing the consequences of a small-break LOCA does it consider whether it is reasonably practicable to modify it.

3.1.1 Small-Break LOCA Safety Case Background

30. As part of its original safety case (Ref. 3), Westinghouse has always recognised that there is a potential for a spectrum of possible break sizes to occur in the **AP1000** RCS in a fault condition, ranging from failure of small-bore instrument penetration pipework (< 1 inch diameter) up to double-ended guillotine failure of the main coolant pipework (>9 inches diameter). Examples of other potential initiating events for LOCAs on the RCS are valve failures, steam generator tube ruptures and stuck open pressuriser relief valves.

31. In a change from the position it set out in GDA Step 4, Westinghouse claims that the limiting break size for a small-break LOCA is a break in the RCS up to an equivalent pipe diameter of 2 inches. Any break above this size is considered as 'infrequent' and therefore, in accordance with it design basis safety case approach set out in Chapter 8 of the PCSR (Ref. 18), Westinghouse states that only one Class 1 mitigation capability is required for each Category A safety function. For 'frequent' faults (initiating event frequency > 1×10^{-3} per year), Westinghouse states in Chapter 8 that two diverse mitigation capabilities are required for each Category A safety function.[1]

32. By design, the principal protection against a small-break LOCA is provided by the Class 1 Protection and Monitoring System (PMS), which would automatically trip the reactor on low reactor circuit pressure, and would then actuate the Core Make-up Tanks (CMTs) to inject cooling water into the core. Once the CMTs have emptied to a threshold low level, the PMS would initiate a further depressurisation of the reactor circuit by actuating the ADS Stage 1 to 3 valves, allowing extra cooling water injection from the accumulators. CMT injection resumes once the accumulators have emptied. Eventually they reach the low level specified for ADS Stage 4 actuation. ADS Stage 4 actuation allows the reactor to depressurise to atmospheric pressure, and the IRWST is used to gravity-feed additional cooling water into the core. In the longer term, the PMS would open containment recirculation valves to allow gravity-driven containment recirculation. Cooling of the containment is achieved using the passive containment cooling system and actuation of the Containment Isolation (CI) system.

33. Westinghouse claims that the RNS can be used in its current form, together with some passive features, as a diverse means of mitigating small-break LOCA events.

[1] *In GDA Step 4, in its response to RO-AP1000-47 (Ref. 2), Westinghouse assumed a small break LOCA with a frequency consistent with a 'frequent fault' designation could be up to 4 inches.*

However, breaks of this size (up to 2 inches) are insufficient themselves to depressurise the RCS quickly enough to allow the RNS to inject, and there is a need to assist the depressurisation. This depressurisation can be provided by the PMS automatically opening ADS Stages 1 to 3, or by the Passive Residual Heat Removal (PRHR) system which is claimed to provide additional RCS depressurisation by condensing any steam arising in the reactor circuit. Should automatic actuation from the PMS be lost due to Common Mode Failure (CMF), then the Diverse Actuation System (DAS) controls can be used to actuate the ADS Stage 1 to 3 valves (manually) or the PRHR valves (automatically).

34. In the event of a CMF of ADS Stage 1 to 3 valves, Westinghouse claims that the RCS can be depressurised by manual operation of ADS Stage 4, which can be safely and effectively operated at an RCS pressure higher than its operational design pressure of 1.4 MPa (200 psi).

3.1.2 Determination of the Size of a 'Frequent' Small-Break LOCA

35. Westinghouse has used a US Nuclear Regulatory Commission (US NRC) study, 'NUREG-1829' (Ref. 19), to determine the relationship between break size and initiating event frequency. The stated objective of Ref. 19 is to provide a technical basis for a risk-informed definition of design basis LOCA break sizes, informed by an expert elicitation process.

36. Westinghouse states that Ref. 19 is the (US) nuclear industry's preferred source for Pressurised Water Reactor (PWR) and Boiling Water Reactor (BWR) LOCA event frequencies. It characterises NUREG-1829 as the largest collection of available plant operating experience, and observes that it builds on and supersedes earlier industry documents.

37. Westinghouse claims that there are no unique features to the **AP1000** reactor that would preclude usage of NUREG-1829 to calculate small-break LOCA initiating event frequencies, and that the **AP1000** design utilises similar or improved materials in comparison to those used on the operating plants which informed the US NRC study.

38. Ultimately, through the use of NUREG-1829, Westinghouse has determined that an effective break size of 0.8 inches corresponds to an exceedance frequency of $> 1 \times 10^{-3}$ per year (the usual definition of a frequent fault), while 2 inches corresponds to an exceedance frequency of $> 1 \times 10^{-4}$ per year. However, it has chosen to analyse breaks in the RCS up to an equivalent pipe diameter of 2 inches to demonstrate that there are no 'cliff-edge' effects associated with its definition of a small-break LOCA fault.

3.1.3 Identification of Relevant Design Basis Sequences

39. Having defined the size of a frequent small-break LOCA fault, Westinghouse has identified three bounding design basis sequences for further analysis. Each of these sequences utilises different combinations of SSCs to achieve a safe, stable state following the initial event, assuming a CMF of one (or more) Class 1 SSC that is credited in Westinghouse's main safety case for small-break LOCA faults.

40. The relevant sequences are:

- Diversity Case 1 – small-break LOCA with failure of ADS Stages 1 to 3 (and accumulators)
- Diversity Case 2 – small-break LOCA with failure of PRHR (and accumulators)
- Diversity Case 3 – small-break LOCA with failure of PMS (together with consequential failure of automatic ADS actuation and the CMTs)

41. In addition, one extra sequence has been identified in order to demonstrate additional defence-in-depth. This is a variation to Diversity Case 3, in which no stages of the ADS are assumed to operate even with manual actuation:

- Defence-in-Depth Case 4 – sensitivity to Case 3 with complete failure of ADS

42. The RNS is identified as a means of providing low-pressure safety injection in Diversity Case 3 and Defence-in-Depth Case 4.

43. The plant combinations assumed in each small-break LOCA sequence are summarised in Table 1 below.

Diversity Case	PMS	DAS	PRHR	CMTs	ACC	ADS 123	ADS4	RNS	IRWST
1	Y	N	Y	Y	N	N	Y	N	Y
2	Y	N	N	Y	N	Y	Y	N	Y
3	N	Y	Y	N	Y	Y*	N	Y*	N
4	N	Y	Y	N	Y	N	N	Y*	N

*** denotes operator action**
ACC = accumulators

Table 1 – Small-break LOCA Core Cooling Diversity Cases

3.1.4 Small-Break LOCA Transient Analysis

44. Westinghouse has analysed each of the four small-break LOCA sequences using the RELAP-5 computer code to show that appropriate success criteria have been met. Ref. 17 identifies two criteria to demonstrate:

- The peak cladding temperature shall not exceed 1204°C (2200°F).
- The maximum cladding oxidation shall not exceed 17% of the total cladding thickness before oxidation.

45. Ref. 17 summarises the analysis for each sequence, clearly stating which SSCs have been credited and which SSCs have been assumed to be unavailable due to a CMF, listing in a table the timings of key events within each sequence, and providing plots of key parameters during each transient.

46. For Diversity Case 3, Westinghouse has assumed that the operator takes action to align the RNS and actuate ADS Stage 1 one hour into the event following the initial break in the RCS.

47. For Defence-in-Depth Case 4, Westinghouse has analysed three versions of the sequence to inform its ALARP review of the RNS:

- Case 4.1 – the RNS injection is assumed to start when the RCS pressure reaches the RNS pump head of 1.28 MPa (185 psi) with no delay (this is more than one hour after the initial break in the RCS).
- Case 4.2 – the RNS injection is assumed to be delayed until two hours after the initial break in the RCS.
- Case 4.3 – the RNS injection is assumed to be delayed until two hours after the initial break in the RCS and its pump head is assumed to be increased by 10%.

3.1.5 ALARP Options and Ultimate Conclusion

48. Consistent with ONR's expectations set out in the wording of the GDA Issue, Westinghouse has considered the feasibility of increasing the RNS injection pressure, segregating the RNS water supply from the IRWST and automating the actuation of the RNS. In addition, it has considered adding an automatic PRHR actuation function to the DAS on detection of a low pressuriser level.

49. Each option is discussed in turn in Ref. 17, informed as appropriate by the RELAP-5 transient analysis. Westinghouse has concluded that it is not ALARP to adopt the majority of the considered options. However, it does consider adding the automatic PRHR function to the DAS to be an appropriate enhancement to make.

50. It observes that on the standard **AP1000** plant design, the PRHR is actuated on the high hot-leg temperature signal via the DAS and is not actuated on low pressuriser level. As a result, it is not actuated until the fuel uncovers and the clad temperature starts to increase. Westinghouse states that changing the DAS to actuate the PRHR on low pressuriser level would be a relatively simple change that would benefit **AP1000** operation during small-break LOCAs when the Class 1 SSCs have suffered multiple failures, such as a CMF of the PMS or ADS Stage 4. The change could both prevent cladding temperature excursions and increase the operator action time to align the RNS.

51. Having concluded that the change to the DAS is ALARP, Ref. 17 identifies Design Change Proposal (DCP) APP-GW-GEE-5099 (Ref. 20) as the mechanism for introducing the modification into the design and safety case.

3.2 PCSR

52. The **AP1000** safety case for LOCAs is described in Chapter 9 of the PCSR (Ref. 18). The discussion is split as follows:

- Section 9.6.4 describes the safety case for large-break LOCAs. A large-break LOCA is defined as all RCS ruptures with break sizes sufficient to produce a depressurisation of the RCS that allows gravity injection from the IRWST. This corresponds to a break size of 229 mm (9 inches) equivalent diameter or larger, up to the size of a double-ended break of a cold or hot leg.
- Section 9.6.5 describes the safety case for medium- and small-break LOCAs. These LOCAs are defined as all RCS ruptures with break sizes insufficient to depressurise the RCS to the RNS operating pressure without the operation of the ADS, but sufficient to allow the automatic actuation of ADS Stage 4 without operation of ADS Stages 1 to 3.

53. The bulk of the information provided in these two sections is consistent with EDCD (Ref. 3), updated with new analyses in response to GI-AP1000-FS-02 (Ref. 14) to be consistent with the extant design reference point. However, extra discussion has been added to Section 9.6.5 on small-break LOCAs (up to 2 inches in diameter) to include the diversity demonstrations set out in Ref. 17. The conclusions of the ALARP review of potential enhancements to the RNS are also referenced.

54. Chapter 8 of the PCSR includes the fault schedule. LOCAs of various sizes are included as discrete events, with two entries for small-break LOCAs in operating modes 1 and 2:

- Fault ID 1.8.1 shows the protection provided for small-break LOCAs less than medium LOCAs (4 inches / 10.2 cm) but greater than cliff-edge small LOCAs (2 inches / 5.1 cm). This is categorised as an infrequent fault.

- Fault ID 1.8.2 shows the protection provided for small-break LOCAs greater than RCS ‘leaks’ (0.95 cm / 0.375 inches) and less than infrequent fault small LOCAs (5.1 cm / 2 inches). This is categorised as a frequent fault and therefore diversity in protection is demonstrated, consistent with the three main diversity cases identified in Ref. 17. Where applicable, the diversity cases take credit for the automatic PRHR actuation function to the DAS on detection of a low pressuriser level incorporated into the design by APP-GW-GEE-5099 (Ref. 20).

4 ONR ASSESSMENT OF GDA ISSUE GI-AP1000-FS-05

55. I have broken my assessment of Westinghouse's submissions in response to GI-AP1000-FS-05 into five parts, each captured within an individual subsection:

- In Subsection 4.1, I have examined the adequacy of Westinghouse's justification for the size and location of the bounding frequent small-break LOCA.
- In Subsection 4.2, I have examined the adequacy of Westinghouse's identification of design basis fault sequences against the expectations of SAP FA.6.
- In Subsection 4.3, I have reviewed the results of Westinghouse's transient analyses against the expectations of SAP FA.7 with a view to determining if the analysis of the fault consequences is sufficiently conservative and if appropriate safety criteria have been met.
- In Subsection 4.4, I have assessed the adequacy of Westinghouse's arguments on what enhancements to the RNS are ALARP.
- In Subsection 4.5, I have commented on the adequacy with which Westinghouse has summarised the outcome of the work from GI-AP1000-FS-05 in the PCSR.

4.1 Assessment of the Size and Location of the Assumed Small-Break LOCA

56. Westinghouse has identified breaks in the RCS up to an equivalent pipe diameter of 2 inches in the cold leg to demonstrate that there are no cliff-edge effects associated with its definition of a frequent small-break LOCA fault.

57. Westinghouse has stated that assuming the break is at the bottom of the RCS cold leg is a conservative modelling choice. It argues that if the break was on the hot leg, the RCS would depressurise more quickly, allowing cooling water from other sources (accumulators, the RNS and the IRWST) to be injected earlier in the transient to the benefit of fuel cooling. In contrast, a break on the cold leg leads to a greater loss of mass inventory and a lower depressurisation rate. Hence any additional cooling water (from accumulators, the RNS and the IRWST) would be injected into the core at a later stage in the transient when the RCS pressure drops below the injection pressure thresholds for these cooling water sources.

58. I accept these arguments. The objective of Westinghouse's analysis is to inform its optioneering of improvement to the RNS, including enhancements that could be beneficial in the event of a failure of the ADS valves. Assuming the break is in the hot leg helps to offset the impact of the ADS valves not working and get the RCS to the RNS injection head pressure. If there are any benefits from earlier RNS injection, they would be more prominent for the slower depressurising cold-leg scenario.

59. The source of the initiating event frequencies that Westinghouse has attributed to breaks of various sizes, which informs its determination of what constitutes a frequent LOCA, is NUREG-1829 (Ref. 19). I accept Westinghouse's claim that this reference is probably the most comprehensive and widely used source of LOCA initiating event frequencies, and therefore I have no objections to its use. I am content with the justifications given in Ref. 17 as to why this generic light water reactor document is appropriate for the **AP1000** reactor (indeed likely to be conservative), and specifically the Class 1 RCS pipework.

60. It should also be noted that NUREG-1829 has been used in the **AP1000** PSA as the reference for LOCA initiating event frequencies and judged to be appropriate by ONR specialists (Ref. 22).

61. NUREG-1829 relies heavily on an 'expert elicitation' process to predict pipe break frequencies. However, it also includes a significant amount of discussion on Operational Experience (OPEX) and real events on operating US nuclear plants. Table 1.1 of NUREG-1829 summarises the results of several comparative OPEX pipe break frequency studies. The highest small-break LOCA frequency reported is ~3 x 10^{-3} per year but this includes non-nuclear pipework from other industries and therefore will be very conservative (indeed, not strictly appropriate) for the **AP1000** RCS Class 1 pipework. A further study quoted in NUREG-1829, which is based only on US nuclear plant OPEX, suggests a small-break LOCA initiating event frequency of ~4 x 10^{-4} per year. This corresponds (reasonably) closely to the cliff-edge 1 x 10^{-4} per year derived by Westinghouse using the expert elicitation algorithm from the same reference. I am therefore satisfied that the results Westinghouse has arrived at using NUREG-1829 are reasonable when compared with OPEX reported in the same reference.

62. Having reached a conclusion that the frequency arrived at for small-break LOCAs was a sensible analysis assumption, I still wanted further clarity on exactly what type of failures it should be applied to. I asked Westinghouse through a Regulatory Query (RQ) about the physical reality of the cold- leg pipework compared against modelling assumptions. Westinghouse stated in its response to the RQ that the two cold legs are single bent pipes with no 'T'-sections which could be a source of a small-break LOCA (Ref. 21). The branches off the cold legs to the CMTs are welded to nozzles integral to the cold legs. There are some small-bore branch line connections for flow instrumentation but these are restricted to 0.95 cm (0.375 inches). Even if these instrument lines failed through a guillotine break, the RCS losses should be within the leakage make-up capability of the plant. From this, my understanding is that Westinghouse is applying the NUREG-1829 data to partial breaks (up to 2 inches equivalent diameter) in the larger Class 1 pipework of the RCS, rather than the complete guillotine break of specific pipes which are 2 inches or less in diameter.

63. Westinghouse states in Ref. 17 that NUREG-1829 predicts passive system failures. Specifically, the elicitation process has considered failures associated with the following:

- geometry
- loading
- history
- materials
- ageing mechanisms
- mitigation and maintenance practices

64. Significantly, these failure mechanisms exclude internal hazards and spurious valve openings. I am satisfied that the potential for an internal hazard to result in a pipe break while the reactor is at power and the containment is sealed should be low (for example, as a result of a dropped load). In addition, the design basis internal hazards safety case (assessed by ONR outside this report) should demonstrate that the risks from hazards have been reduced to be ALARP. Therefore, I do not expect internal hazards to be a cause of RCS breaks larger than 2 inches occurring more frequently than 1 x 10^{-4} per year.

65. The mechanical failure or spurious actuation of valves connected to the RCS could result in breaks larger than 2 inches. It is my judgement that it is reasonable to assume that a catastrophic mechanical failure of a Class 1 valve connected to the RCS should not be a frequent fault. However, the spurious actuation of a valve merits more discussion. The Pressuriser Safety Valves (PSVs) and ADS Stages 1 to 3 are all connected to one of the RCS hot legs via the pressuriser. If one of these valves opened spuriously, the consequences would be equivalent to a break in the hot leg larger than 2 inches.

66. Westinghouse has identified these valve failures as separate events to the frequent small-break LOCA fault on the fault schedule, claiming that their initiating event frequencies are lower than 1 x 10^{-3} per year. In the case of motor-operated ADS Stage 1 to 3 valves, I am satisfied that the work done to implement a 'blocker' against spurious PMS actuation in response to GI-AP1000-CI-04 (Ref. 24) is sufficient to support this assumed low frequency.[2]

67. The fault schedule states that the initiating event frequency for an inadvertent PSV actuation is 3.9 x 10^{-4} per year. Like the 2-inch cold-leg break scenario, this could perhaps be considered as a cliff-edge scenario in order to demonstrate diversity in protection, especially if there is some uncertainty with this frequency. However, even if this event is treated as a frequent fault, an inadvertent opening of the PSV would assist the process of depressurising the RCS to allow water injection into the core, analogous to the deliberate actuation of ADS Stages 1 to 3. I would therefore expect the 2-inch cold-leg break to still be the limiting scenario to inform considerations of potential enhancements to the RNS.

68. NUREG-1829 also excludes Steam Generator Tube Ruptures (SGTRs). However, while these are frequent small breaks associated with the RCS, their fault sequences proceed differently to other LOCAs. SGTR faults were assessed in the Step 4 GDA report (Ref.1), which raised Assessment Finding AF-AP1000-FS-33 for a future licensee to confirm that following the CMF of any one safety system that fulfils a safety function role for SGTR faults there is either a diverse means of protection or that the radiological consequences are ALARP. As a result, I do not propose to discuss the SGTR fault further in this report.

69. An event which is unique to the AP1000 reactor and is effectively excluded from NUREG-1829 is a PRHR tube rupture. Westinghouse has identified the event in its fault schedule as an infrequent fault with an initiating event frequency of 1.1 x 10^{-4} per year, and discussed it in more detail in Chapter 9 of the PCSR (Ref. 18). The basis of the initiating event frequency is not clearly stated in the documentation I have reviewed, and there is inevitably very limited or no reliability data for the AP1000 PRHR tubes (they are very similar to steam generator tubes but will experience different conditions during their lifetime). However, I do recognise the points made by Westinghouse in the PCSR:

- A rupture of a PRHR tube (inner diameter of 15.7 mm / 0.62 inches) results in LOCA that is smaller than the limiting small-break LOCA considered in Ref. 17.
- If the Chemical and Volume Control System (CVS) or PRHR isolation valves work correctly, the RCS inventory will not be reduced to a significant extent and no demand will be placed on the SSCs identified to protect the reactor from the consequences of a LOCA.
- The Class 1 SSCs identified to protect the AP1000 reactor from a small-break LOCA could all be credited. However, if a conservative assumption is made that the PRHR is unavailable (its performance could be degraded by a tube rupture but would not be lost completely), the design basis transient would proceed differently to the 2-inch small-break LOCA event. The initial steam generator secondary side inventory would be sufficient to remove heat for an extended period of time and bring the RCS pressure down to the steam generator secondary side pressure (8.34 MPa). This would result in ADS Stages 1 to 3 opening at a higher pressure than would otherwise be the case, but within their capability.

[2] *The ADS Stage 4 valves are connected to the RCS hot legs. They have also been enhanced with a blocking device to protect against spurious opening. In addition, the size of the resulting opening will not meet the small break LOCA criteria. The resulting transient, involving a significant depressurisation, would not be informative to reaching judgements about enhancements to the RNS.*

70. During its lifetime, the PRHR heat exchanger should experience significantly less onerous operational conditions and transients compared to the similar steam generator tube bundles and therefore I do not think it is unreasonable to assume a tube rupture failure is less likely than a SGTR event. However, even if it were to be shown that the infrequent designation for a PRHR tube rupture fault cannot be substantiated and therefore a demonstration of diversity is required, not all of the analysis and safety case discussion included in Ref. 17 would be directly applicable. Despite this, I do not anticipate that specific analysis of these (very) small-break LOCA faults would be informative to any response to the main objective of this GDA Issue to identify if any enhancements to the RNS are ALARP. I have therefore not considered this fault further in this report.

71. In conclusion, I am satisfied that:

- assuming a small break at the bottom of the RCS cold leg is an appropriate conservative modelling choice to inform judgements on what enhancements to the RNS are ALARP;
- using NUREG-1829 to determine what size of RCS break should be considered a frequent fault is acceptable, even though it is restricted to passive pipe failures and excludes internal hazards and spurious valve operations; and
- the determination (by application of NUREG-1829) that a 2-inch break is an appropriate cliff-edge size to bound small-break LOCAs is reasonable and is consistent with reported OPEX.

4.2 Assessment of the Identification of Relevant Design Basis Sequences

72. As described in Section 3, Westinghouse has identified three small-break LOCA diversity cases which assume that different combinations of SSCs act in response to the initiating event. It has also identified one extra sequence aimed at showing extra defence-in-depth.

73. SAP FA.6 requires that “For each initiating fault within the design basis, the relevant design basis fault sequences should be identified”. As part of my assessment, I have undertaken my own review of the potential design basis small-break LOCA fault sequences, and compared the results against Westinghouse’s four sequences.

74. The fault schedule (Chapter 8 of Ref. 18) identifies the following Class 1 SSCs as providing the main protection for small-break LOCAs:

- PMS
- automatic Rod Cluster Control Assembly (RCCA) insertion, actuated by the PMS
- automatic PRHR initiation, actuated by the PMS
- automatic CMT initiation, actuated by the PMS
- automatic ADS Stages 1 to 4 initiation, actuated by the PMS
- passive accumulator injection
- automatic IRWST gravity injection and recirculation, actuated by the PMS
- automatic Passive Containment Cooling System (PCS) operation, actuated by the PMS
- automatic CI operation, actuated by the PMS

75. In Table 2, I have systematically assumed a CMF of each of these SSCs and identified the Class 1 SSCs that would still be available. I have also identified any additional Class 2 SSCs that could be needed to deliver necessary safety functions, notably the DAS, the Plant Control System (PLS) and the RNS.[3] I have then mapped the list of

[3] *The PLS is mainly used for normal operational control of the plant. However, there are a number of safety claims placed on it to actuate SSCs which are diverse means of providing the Category A residual heat removal safety function, including RNS, the*

sequences I have identified to the diversity cases identified by Westinghouse in Table 1.

76. Of the eleven sequences I have identified, eight are equivalent to, or are bounded by, Westinghouse's diversity cases. Sequences 2, 10 and 11 are not explicitly covered by Westinghouse's cases, but they are not associated with short- and medium-term core cooling and are therefore not relevant for the considerations of the potential RNS enhancements identified. I also acknowledge that additional discussion is included in the latest revision of the PCSR (Ref. 18) on these three sequences (compared to what was provided in GDA Step 4), and that there are extant assessment findings from GDA Step 4 (Ref.1) for these CMFs. As a result, the associated safety case arguments for the three excluded sequences will be considered further (by both a future licensee and ONR) in site licensing, and I have not considered them further in this report.

77. On this basis, I am satisfied that Westinghouse's three diversity cases listed in Table 1 (and included in the fault schedule, Ref. 18) will bound all relevant fault sequences. I am therefore content that the expectations of FA.6 have been met for small-break LOCA faults.

Startup feedwater system (SFW), the Component Cooling Water System (CCS) and the Service Water System (SWS). As a result, the relevant parts of the PLS responsible for these systems are Class 2 rather than Class 3.

Sequence Number	Assumed Common Failure	Description of Case	Comment
1	PMS	Small-break LOCA with CMF of PMS resulting in consequential failure of automatic ADS 1–4 and IRWST. Note that DAS will activate PRHR, CMTs and PCS. Accumulators work passively. RNS is manually actuated using PLS and one of the ADS 1–4 lines is operated manually using DAS.	Bounded by Diversity Case 3 which also assumes loss of CMTs.
2	RCCA insertion	Small-break LOCA with CMF of RCCAs to insert resulting in an Anticipated Transient Without Scram (ATWS) event.	Separately discussed in the PCSR (Ref. 18). Covered by AF-AP1000-FS-42.
3	PRHR	Small-break LOCA with CMF of PRHR. All other systems will work.	Bounded by Diversity Case 2 which also assumes accumulators fail.
4	CMTs	Small-break LOCA with CMF of CMTs resulting in consequential failure of automatic ADS 1–4 and IRWST. Note that DAS will activate PRHR and PCS. Accumulators work passively. RNS is manually actuated using PLS and one of the ADS 1–4 lines is operated manually using DAS.	Diversity Case 3.
5	ADS 1, 2, 3	Small-break LOCA with CMF of ADS 1–3 resulting in operation of ADS 4 at potentially high operating pressure.	Diversity Case 1. Also covered by AF-AP1000-FS-43.
6	ADS 4	Small-break LOCA with CMF of ADS 4. RNS provides diverse injection.	Bounded by Diversity Case 3.
7	ADS blocker	Small-break LOCA with CMF of ADS 1 – 4.	Bounded by Diversity Case 3
8	Accumulators	Small-break LOCA with CMF of accumulators. All other systems would work.	Bounded by Diversity Case 2 which also assumes PRHR fails.
9	IRWST injection	Small-break LOCA with CMF of IRWST injection. RNS provides diverse injection	Bounded by Diversity Case 3.
10	PCS	Small-break LOCA with CMF of PCS.	Covered by AF-AP1000-FS-27.
11	CI	Small-break LOCA with CMF of CI.	Covered by AF-AP1000-FS-44.

Table 2 – Postulated small-break LOCA fault sequences following CMFs to Class 1 SSCs

4.3 Assessment of Westinghouse's Transient Analysis

78. SAP FA.7 states that "analysis of design basis fault sequences should use appropriate tools and techniques, and be performed on a conservative basis". To come to a view on the adequacy with which Westinghouse has complied with this expectation, I have looked at generic aspects of Westinghouse's analyses that are common to all of its diversity cases, and the specific assumptions and modelling approaches in the individual cases.

79. SAP FA.7 also states that "analysis should demonstrate, so far as is reasonably practicable, that none of the physical barriers to prevent the escape of a significant quantity of radioactivity is breached or, if any are, then at least one barrier remains intact and without a threat to its integrity". Westinghouse's means for demonstrating this for small-break LOCA faults in Ref. 17 is to show by analysis that the fuel will not become uncovered or, failing this, demonstrate that the peak fuel clad temperature does not exceed a temperature limit of 1204°C and the maximum clad oxidation does not exceed 17%.

80. To inform my own judgements on whether improvements to the RNS are ALARP, I have looked particularly closely at those transients which are predicted to result in peak clad temperatures in excess of 600°C. While this is not a safety case limit that Westinghouse has considered, it represents a cladding temperature at which clad ballooning could start (Ref. 23). Reaching 600°C is not a direct challenge to SAP FA.7 (the cladding will not immediately start to fail). However, it would be desirable to avoid subjecting the fuel to such conditions. It is also worth avoiding from an analysis perspective, as Westinghouse's methodology does not model the resulting change in cooling geometry from ballooning or predict the point of failure if high temperatures (>600°C) are sustained for a period of time. I therefore judge there to be merit in considering whether an enhanced RNS could help to limit temperatures to less than this value.

81. For this part of my assessment, I have looked at the transient analyses in Ref. 17. I have also looked at the supporting Westinghouse calculation note which provides some further details on the same analyses (Ref. 25).

4.3.1 Common Aspects

82. Westinghouse has used the RELAP-5 computer code to model the identified small-break LOCA cases. The appropriateness of this code for **AP1000** transient analysis was considered during GDA Step 4 alongside other computer codes in GDA Step 4 (Ref. 1) and I have therefore not attempted repeat an assessment against the AV series of SAPs (Ref. 11). It should be noted that although RELAP-5 is a well-established thermal hydraulic code extensively used for modelling PWR LOCA faults, Westinghouse has not used it to support its **AP1000** safety case submissions in other countries and as a result has not generated the same level of validation evidence for its appropriateness as it has for other 'licensing' codes.

83. The GDA Step 4 report (Ref. 1) raised AF-AP1000-FS-24 for a future licensee to provide the validation evidence to support the use of the RELAP-5 code for application to **AP1000** analysis, or to make use of other codes that have been validated for the **AP1000** reactor. While I am satisfied that Westinghouse's use of the RELAP-5 code is appropriate to support its decision-making on potential improvements to the RNS, the requirement to address AF-AP1000-FS-24 remains.

84. Westinghouse has identified a 2-inch break on a cold leg as the limiting fault to analyse. However, there are two cold legs which are not completely identical. Westinghouse undertook some early sensitivity calculations and determined that the most onerous location for the break is on the bottom of the cold leg associated with the PRHR. In this situation, the effectiveness of the PRHR is slightly compromised as some of the cooling water from the PRHR is lost through the break. As a result, the final set of studies reported in Ref. 17 assumed that the break occurred on the cold leg associated with the PRHR. I am content that this is a conservative assumption that is consistent with the expectations of SAP FA.7.

4.3.2 Diversity Case 1

85. Diversity Case 1 assumes that the PMS is operational. The PRHR and CMTs are actuated on a low pressuriser pressure. A CMF is assumed to fail the ADS Stages 1 to 3 and the accumulators. The RCS pressure is reduced by the inventory loss through the break and the heat transferred to the IRWST by the PRHR heat exchanger. As inventory is lost, the CMTs begin to drain, eventually reaching the set-point for ADS Stage 4 actuation.

86. Once ADS Stage 4 actuates, the RCS pressure is reduced and injection from the IRWST is established. No operator action is assumed for this case as the Class 1

SSCs (PRHR, ADS Stage 4) are sufficient to depressurise the RCS to the IRWST injection pressures.

87. Clad temperature transients for Diversity Case 1 are given in Figure 3.1-5 of Ref. 17. This shows that the maximum clad temperature never challenges the 600°C temperature for the onset of clad ballooning. The maximum clad temperature is at ~350°C at the start of the transient, and remains at the saturation temperature throughout the transient, decreasing monotonically as the transient progresses. Therefore there is a very wide clad temperature margin for Diversity Case 1.

88. A significant factor in judging if Diversity Case 1 can be considered an acceptable design basis sequence is whether the ADS Stage 4 valves can safely and effectively open at a higher pressure than their normal design intent (1.4 MPa / 200 psi). This is not a new concern. The effectiveness of the ADS Stage 4 actuation following a CMF of ADS Stages 1 to 3 is modelled as 'success' in the PSA. In addition, during the course of the GDA Step 4 fault studies assessment (Ref. 1), the need to demonstrate this capability in the small-break LOCA design basis safety case was also anticipated, resulting in ONR writing AF-AP1000-FS-43:

> "The future licensee shall demonstrate the structural integrity of the ADS Stage 4 lines following actuation of ADS Stage 4 valves during a frequent fault SBLOCA [short-break LOCA] in which there is creditable Common Mode Failure of the ADS Stage 1 to 3 valves to open. PRHR operation may be assumed."

89. For the purposes of this GDA Issue, Westinghouse imposed a limit of 4.1 MPa (600 psi) for the safe operation of the ADS Stage 4. The transient analysis results for Diversity Case 1 show that the RCS pressure at the time of ADS Stage 4 actuation is below this limit (~2.5 MPa).

90. To substantiate the basis for this limit, Westinghouse submitted Ref. 26 which details a structural integrity analysis of the hydrodynamic loads on piping following an ADS Stage 4 actuation at 4.1 MPa (600 psi). Its conclusions were as follows:

- The ADS Stage 4 piping is rigidly supported and the increase in hydrodynamic loads has little effect on pipe stress.
- Pipe support loads are affected by the increase in hydrodynamic load; however, there is a high confidence in the structural integrity of the pipe supports.
- Based on these results, there is a high degree of confidence that ADS Stage 4 piping and supports will maintain their structural integrity if actuated at 4.1 MPa (600 psi).

91. As stated in Subsection 2.4, I consulted specialist structural integrity colleagues for advice on the adequacy of this analysis. They expressed no objections to Westinghouse's analysis or its conclusions.

92. Subsequent to the submission of Ref. 26, Westinghouse undertook an expert panel review of the consequences of spurious ADS Stage 4 actuation at full RCS pressure (Ref. 27) – that is, a much higher pressure than 4.1 MPa. This review concluded that:

- RCS pipe deformation would occur but it would not significantly affect the venting performance;
- a change in elevation of the ADS Stage 4 valves due to pipe whip is not anticipated to affect the ADS Stage 4 venting capability; and
- Class 1 SSCs that would be assumed to respond following an inadvertent ADS Stage 4 actuation at full RCS pressure (accumulators, CMTs, ADS Stage 4, IRWST injection and recirculation) are not anticipated to be affected by a jet

impingement and water missiles generated due to the initiating event, with the exception of ADS Stage 4 valves and one of the CMTs, depending on the specific scenarios (spuriously opening only one ADS Stage 4 valve could have a negative consequence on the second valve in the same compartment).

93. Ref. 27 has been considered by mechanical engineering colleagues as part of their work on GI-AP1000-ME-01 (Ref. 28) and assessed in more detail by structural integrity colleagues (Ref. 29). Neither expressed any objections to Westinghouse's conclusions. On this basis, I am satisfied that Diversity Case 1 is a credible design basis sequence and the RELAP-5 assumptions on the **AP1000** plant's behaviour later on in the transient are not compromised by the physical consequences of ADS Stage 4 actuation at a (modestly) higher pressure than the design intent. I do note that Ref. 27 did identify a potential threat to one of the CMTs from spurious ADS Stage 4 actuation at full RCS pressure, and Diversity Case 1 does take credit for the CMTs. However, at less than 4.1 MPa, this threat should not exist, and at the point of (deliberate, not spurious) ADS Stage 4 actuation, the CMTs have effectively completed their main safety function.

4.3.3 Diversity Case 2

94. In Diversity Case 2, the PMS is assumed to remain operational. A CMF is assumed to occur on both the PRHR and the accumulators. The CMTs are actuated on a safety injection signal that is generated by low pressuriser pressure indication. Without the PRHR, the reactor decay heat is removed by the flow out of the break, and boil-off of the steam generator inventory. The RCS pressure remains high, until sufficient inventory is lost through the break to begin CMT drain-down.

95. When the RCS level reaches the ADS Stage 1 set-point, the ADS 1, 2 and 3 valves actuate in sequence to quickly reduce the RCS pressure. The pressure is reduced again when the ADS Stage 4 actuates on low CMT level, allowing for stable gravity injection from the IRWST. As with Diversity Case 1, no operator actions are claimed.

96. Clad temperature transients for Diversity Case 2 are given in Figure 3.2-4 of Ref. 17. It shows that the maximum clad temperature (~350°C) remains at the saturation temperature and never challenges the 600°C temperature relevant for the onset of clad ballooning.

97. On this basis, I am satisfied that Diversity Case 2 is an acceptable design basis fault sequence, for which Westinghouse has demonstrated that the mitigated consequences are consistent with the expectations of SAP FA.7.

98. It should be noted that the GDA Step 4 fault studies assessment report (Ref. 1) identified a concern associated with the consequences of a passive single failure of the normally open PRHR isolation valve following a small-break LOCA fault. It stated that it expected the work to address GI-AP1000-FS-05 would consider this fault sequence. I am satisfied that the analysis of Diversity Case 2 does this.

4.3.4 Diversity Case 3

99. Diversity Case 3 covers the CMF of both the PMS and the CMTs. As a result of the PMS CMF, automatic actuation of the ADS valves is assumed not to be possible. It assumes that ADS Stages 1 to 3 are actuated manually but ADS Stage 4 is not credited.

100. In the transient analysis, Westinghouse has assumed that the PRHR is actuated on a DAS signal responding to a low pressuriser pressure level. This is the design change identified by Westinghouse as desirable and implemented by APP-GW-GEE-5099 (Ref. 20).

101. The inventory loss out of the break, together with the operation of the PRHR heat exchanger, reduces the RCS pressure. When the RCS pressure reaches 4.83 MPa, the accumulators begin to inject.

102. When the RCS pressure approaches the RNS cut-in pressure, the operator aligns the RNS, and manually actuates ADS Stage 1. ADS Stage 2 and 3 valves open on timers after ADS Stage 1 actuation, allowing the RNS to inject. The reduction in pressure resulting from ADS Stages 1 to 3 empties the accumulators, releasing nitrogen into the RCS. The heat removal from the PRHR heat exchanger is significantly reduced following ADS actuation but there is no re-pressurisation and disruption in RNS flow. Stable RNS injection marks the successful conclusion of the transient.

103. Clad temperature transients for Diversity Case 3 are given in Figure 3.3-5 of Ref. 17, which again shows that the maximum clad temperature (~350°C) remains at the saturation temperature of the coolant and never challenges the 600°C clad ballooning temperature. I am therefore satisfied that the requirements of SAP FA.7 have been met for this sequence.

104. The key feature of this transient is that it demonstrates that the PRHR is very effective at condensing steam and reducing the RCS pressure. Coupled with the loss of inventory through the break, it is able to reduce the RCS pressure down to the RNS cut-in pressure without the need for any additional ADS venting. Uncertainty about the effectiveness of the PRHR was one of the reasons for ONR writing GI-AP1000-FS-05 at the end of GDA Step 4. In my opinion, Diversity Case 3 is showing that the PRHR is already capable of bringing the RCS pressure down to the RNS injection pressure, and therefore there is very limited benefit in increasing the RNS's capability to inject at a higher pressure.

105. On reflection, the effectiveness of the **AP1000** PRHR heat exchanger is not surprising. On a conventional PWR, the emergency operating procedures following a small-break LOCA require the operator to blowdown the steam generators to low pressure to promote increased heat removal and protect against the fault. The **AP1000** design is effectively doing the same thing (but with much less reliance on the operator to take the correct actions).

4.3.5 Defence-in-Depth Case 4

106. Defence-in-Depth Case 4 is effectively a sensitivity study on Diversity Case 3 in which it is assumed that a CMF prevents both the automatic and manual operation of all four stages of the ADS, in addition to the failures of the PMS and CMTs. The PRHR is actuated by the DAS in response to low pressuriser pressure. The loss out of the break and the PRHR removal of reactor decay heat reduce the RCS pressure. When the RCS pressure reaches ~4.8 MPa the accumulators begin to inject.

107. As stated in Subsection 3.1.4, Westinghouse has analysed three variations of the fault sequence:

- Case 4.1 – the RNS injection is assumed to start when the RCS pressure reaches the RNS pump head of 1.28 MPa (185 psi) with no delay (this is more than one hour after the initial break in the RCS).
- Case 4.2 – the RNS injection is assumed to be delayed until two hours after the initial break in the RCS.
- Case 4.3 – the RNS injection is assumed to be delayed until two hours after the initial break in the RCS and its pump head is assumed to be increased by 10%.

108. For Case 4.1, Figure 3.4-6 of Ref. 17 shows that the fuel is briefly uncovered with fuel clad temperatures reaching ~350°C at about 10,000 seconds into the transient. This

spike is coincident with the injection of nitrogen into the RCS from the accumulators. The peak temperature is still below the 600°C clad ballooning temperature.

109. The interaction between the RNS, the accumulators and the PRHR requires discussion. In Case 4.1, it is highly likely that the accumulators will empty and release nitrogen into the primary circuit. As a non-condensable gas, there is a risk that the nitrogen could inhibit the natural circulation flow through the PRHR as it rises up to the highest point in the circuit, which is the PRHR.

110. During some of the early interactions between ONR and Westinghouse of GI-AP1000-FS-05, what became Defence-in-Depth Case 4 was being considered to have a more prominent role in the **AP1000** safety case demonstration of diversity for small-break LOCA faults. As a result, I challenged Westinghouse through an RQ to provide further substantiation on the continuing effectiveness of the PRHR in such conditions. In its response (Ref. 21), Westinghouse argues that thermal hydraulic tests performed on the SPES-2 and APEX facilities have demonstrated that for other fault sequences (even after the accumulators have emptied) the PRHR is able to remove a significant amount of heat even though its performance is slightly degraded. While these tests are not totally prototypic for this particular fault sequence, they do give me some confidence that there is no cliff-edge effect on PRHR performance. I also note that the RELAP-5 code does have a capability to model non-condensable gases in the RCS (unlike Westinghouse's 'normal' small-break LOCA licensing code, NOTRUMP) and the behaviour of nitrogen is factored into the analysis.

111. I am satisfied with this level of substantiation for the PRHR's effectiveness, cognisant that Westinghouse has not made strong claims on the results of Defence-in-Depth Case 4 in its small-break LOCA design basis safety case and its RNS enhancement ALARP considerations.

112. Case 4.2 assumes a two-hour delay in the operator actuating the RNS. The delay in the operator actuating the RNS gives two spikes in clad peak temperatures (Figure 3.4-13 of Ref. 17): one at ~7,500s (to ~630°C) and a second, larger clad temperature spike (to ~780°C) at ~15,000s into the transient. The second temperature spike is over a period of ~10 minutes as the top of the core is uncovered. The peak clad temperatures in this sensitivity study case are obviously higher than sensitivity study Case 4.1 above and challenges the ~600°C clad ballooning threshold temperature I have considered. However, it is a very low-frequency event, including a CMF of all stages of the ADS and assuming very conservative operator response times. I believe it would be grossly disproportionate to insist on enhancements to the RNS to mitigate the consequences of this extreme fault sequence.

113. I have compared this sequence with a comparable fault sequence in the Sizewell B safety case covering small-break LOCAs with the failure of the safety injection system. Sizewell B's approach requires the operator to cool down and depressurise the RCS and use the Chemical and Volume Control System (CVCS) as the make-up system. My judgement is that the complexity of these actions is comparable to what is required from the **AP1000** operators in Defence-in-Depth Case 4. The Sizewell B safety case typically assumes 30 minutes for an operator action required for design basis events. Therefore, the assumption of a one-hour delay for operator action (Case 4.1) is likely to be a more than adequate analysis assumption, and as a result the higher temperatures predicted by Case 4.2 assuming a two-hour delay are not a concern.

114. Case 4.3 is identical to Case 4.2 but with a 10% increase in the RNS injection pressure. This is a sensitivity study on the effect of increasing the RNS pressure undertaken in response to an RQ (Ref. 21) and subsequently captured in the final revision of Ref. 17. Figure 3.4-20 of Ref. 17 shows a decrease (~50°C) in the peak clad temperature spiking discussed above for Case 4.2 which could be beneficial with respect to clad failure due to ballooning. However, there is no obvious reduction in the

nitrogen injection from the accumulators by increasing the RNS injection pressure. Again, I believe it would be grossly disproportionate to insist on enhancements to the RNS for this limited benefit to an extreme fault sequence.

4.4 Assessment of Westinghouse's ALARP Review

115. As stated in Section 3, Westinghouse identified four potential design enhancements:

- increasing the RNS injection pressure
- segregating the RNS water supply from the IRWST
- automating the actuation of the RNS
- adding an automatic PRHR actuation function to the DAS on detection of a low pressuriser level

116. It concluded that only the last option is a reasonably practicable enhancement to implement.

117. I am satisfied that this in an appropriate set of options to consider. My judgements on Westinghouse's conclusions on each option are set out in the following subsections.

118. It should be noted that Westinghouse has demonstrated through its transient analysis results that the SSCs included in the extant **AP1000** design can successfully mitigate the consequences of a small-break LOCA fault such that there is very limited consequential fuel damage. The transient analysis actually predicts that all applicable acceptance criteria are met, but Westinghouse does not claim there will be no fuel damage for small-break LOCAs in the PCSR (see Section 9.6.5.3.4.4 of Ref. 18). As a result, Westinghouse is not claiming to be beneath the Target 4 Basic Safety Objective (BSO) defined in the SAPs (Ref. 11), but it does claim to be significantly beneath the Target 4 Basic Safety Level (BSL) for these low-frequency design basis events (ie a frequent LOCA fault with a CMF of a Class 1 SSC). My opinions on whether further enhancements should be considered reasonably practicable are informed by this starting position.

4.4.1 RNS Injection Pressure Increase

119. Westinghouse's reason for not implementing this option is that increasing the RNS injection pressure would involve additional costs for a new pump design, increased electrical loading and a redesign of the auxiliary building to accommodate the bigger multi-stage pumps needed to provide the increased pressure head.

120. Given the acceptable consequences predicted by the transient analysis for Diversity Case 3 and Defence-in-Depth Case 4 (assuming the operator initiates injection within one hour), I agree with Westinghouse that the disadvantages of implementing an increase in the RNS pump head far outweigh the benefits of an increase in RNS injection pressure, and therefore this is not an ALARP option.

4.4.2 Segregating the RNS Water Supply from the IRWST

121. The RNS water supply for the diverse small-break LOCA safety case comes from either the Cask Loading Pit (CLP) or the IRWST. ONR's original concern with this arrangement was associated with (larger) Direct Vessel Injection (DVI) line breaks rather than the 2-inch cold-leg break fault considered for this GDA Issue. The Step 4 GDA fault studies assessment (Ref. 1) noted that following a DVI line break, the operator would be expected to deduce the location of the break by studying the differences in the water levels in the two CMTs in order to determine the optimum RNS alignment for safety injection. This was to avoid a potentially unfavourable interaction with the IRWST system. ONR postulated that a separate RNS suction tank could

simplify the diagnostic and reconfiguration tasks the operator would need to perform following a LOCA, and therefore increase the efficiency of the response.

122. Westinghouse's key arguments for not having a segregated RNS water supply are as follows:

- Even if the new suction tank is made significantly larger than the CLP (comparable in size to the IRWST), there would still be the requirement to realign to the IRWST when the new tank became empty. Such a large tank could not be accommodated inside the current auxiliary building and so there would be significant cost increases.
- A new separate RNS suction tank would result in new surveillance requirements to ensure the availability of the tank, and increase maintenance requirements.
- A new tank would also require monitoring of the amount of water, as it is another source of water that could affect internal (to the containment) flooding levels.
- Independent of this ALARP review, Westinghouse has already implemented a design change (APP-GW-GEE-4507) to simplify the operator actions in response to a small-break LOCA fault. This is included within the UK **AP1000** design reference point (Ref. 16).

123. Given the potential dis-benefits of redesigning the **AP1000** plant to accommodate a new RNS suction tank, my judgement is that Westinghouse's arguments above are reasonable and redesigning the plant would not be an ALARP option.

4.4.3 Automatic RNS Actuation

124. Westinghouse's main argument for not automating the RNS actuation is based on the claim that the reliability of the operator to realign the RNS and commence cooling water injection is comparable or superior to the reliability that would be obtained from automatic RNS realignment and actuation. Westinghouse bases this claim on the time available to the operator (a minimum of one hour) to realign and actuate the RNS after a small-break LOCA begins. Westinghouse also argues that, given the large number of potential operating configurations the RNS could be in prior to the fault, the logic for an automatic system would be highly complex and could potentially compromise the effectiveness of the Class 1 passive systems.

125. I do not fully accept all of Westinghouse's arguments about the relative reliabilities of manual versus automatic actions, and the difficulties of designing a control system to manage all the potential configurations of the RNS. However, I judge that it would be disproportionate to insist on RNS automation given the results of the transient analysis for the four diversity sequences, and therefore I agree with Westinghouse's ultimate conclusion that the enhancement would not be ALARP.

4.4.4 PRHR Actuation on Low Pressuriser Level

126. As explained in Section 3, Westinghouse has stated that changing the DAS to actuate the PRHR on low pressuriser level would be a relatively simple change that would benefit the **AP1000** operation during small-break LOCAs when the Class 1 SSCs have suffered multiple failures, such as a CMF of the PMS or ADS Stage 4.

127. The value of this design change is illustrated by the transient analysis for Diversity Case 3 in Ref.17 (which already takes credit for it). The earlier actuation of the PRHR (at about ~80 seconds into the transient) prevents an early spike in fuel temperature.

128. Therefore, given that Ref. 17 states that this is a feasible modification with only a small cost associated with it, I agree with Westinghouse's conclusion that this is an ALARP improvement and I welcome its incorporation into the UK **AP1000** design.

4.5 Adequacy of the PCSR

129. I have reviewed the UK-specific diversity demonstrations for small-break LOCAs that have been added to the broader safety case for LOCAs included in Chapter 9 of the PCSR (Ref.18). I am satisfied that it provides an appropriate summary of the analysis and conclusions of Ref. 17, and these have been brought together in support of adequate design basis safety case arguments for the fault.

130. I am also satisfied that the fault schedule entries in Chapter 8 of the PCSR (Ref. 18) appropriately identify the small-break LOCA fault, attribute an initiating event frequency to it, and clearly identify the SSCs (including their safety classification) that protect against the event. This is consistent with the expectations of SAP FA.8 for a design basis safety case to include clear and auditable linking of initiating faults, fault sequences and safety measures.

131. It should be noted that the general adequacy of the fault schedule has been assessed as part of the work to close GI-AP1000-FS-08 (Ref. 31), and applicability of Westinghouse's transient analyses reported in PCSR Chapter 9 to the UK **AP1000** design reference point have been assessed as part of the work to close GI-AP1000-FS-08 (Ref. 30). Small-break LOCAs have been considered as part of these assessments.

4.6 Assessment Findings

132. Assessment findings are matters that do not undermine the generic safety submission and are primarily concerned with the provision of site-specific safety case evidence, which will usually become available as the project progresses through the detailed design, construction and commissioning stages.

133. Residual matters are recorded as assessment findings if one or more of the following apply:

- Site-specific information is required to resolve this matter.
- The way to resolve this matter depends on licensee design choices.
- The matter raised is related to operator-specific features, aspects or choices.
- The resolution of this matter requires licensee choices on organisational matters.
- To resolve this matter the plant needs to be at some stage of construction or commissioning.

134. In my assessment I did not find any examples of matters which meet these criteria.

135. Several assessment findings raised during GDA Step 4 (Ref. 1) had relevance to this GDA Issue. For completeness, they are summarised below:

- AF-AP1000-FS-24 was raised requiring a future licensee to provide validation evidence on the applicability of the RELAP-5 computer code for performing safety analysis of the **AP1000** design. Alternatively, verified computer codes used in the analysis of **AP1000** design basis events should be used to replace the RELAP-5 safety analysis.

- AF-AP1000-FS-27 was raised requiring a future licensee to demonstrate that the RNS, CCS and SWS cooling chain systems are adequately sized to provide a diverse heat sink function.

- AF-AP1000-FS-42 was raised requiring a future licensee to perform transient analysis to demonstrate adequate protection against the frequent small-break LOCA fault with failure of the RCCAs to insert.

- AF-AP1000-FS-43 was raised requiring a future licensee to demonstrate the structural integrity of the ADS Stage 4 lines following actuation of ADS Stage 4 valves during a 'frequent fault' small-break LOCA in which there is creditable CMF of the ADS Stage 1to 3 valves to open.
- AF-AP1000-FS-44 was raised requiring a future licensee to demonstrate adequate diverse protection against a small-break LOCA with failure of CI, and to demonstrate that radiological releases are ALARP and meet the requirements of Target 4 of the SAPs.

136. Some of these have continuing or even reinforced applicability following the work to address GI-AP1000-FS-05. Some have effectively been addressed by this GDA Issue or parallel work for other fault studies GDA Issues. I am aware of some which may have been partially or completely addressed by general improvements to the PCSR since GDA Step 4. However, the review and sentencing of these assessment findings is beyond the scope of this assessment report and the wider GDA project.

5 CONCLUSIONS

137. This report presents the findings of the assessment of GDA Issue GI-AP1000-FS-05 relating to the **AP1000** GDA closure phase.

138. I am satisfied that Westinghouse has adequately demonstrated in Ref. 17 that it has:

- appropriately defined the size and character of a limiting frequent small-break LOCA fault;
- identified appropriate fault sequences, including CMFs to Class 1 SSCs, to analyse;
- analysed the identified fault sequences to demonstrate the diversity provided by the extant **AP1000** design; and
- used the results of the fault sequence transient analyses to inform ALARP judgements on what improvements to the RNS are ALARP.

139. Westinghouse has concluded that it is not ALARP to make further enhancements to the RNS. I agree with this decision.

140. Westinghouse has identified a change to the DAS to facilitate earlier PRHR actuation. The benefit of this change is demonstrated by Westinghouse's transient analyses and I welcome its inclusion.

141. I am satisfied with how Westinghouse has incorporated the outcome of its work for GDA Issue GI-AP1000-FS-05 into the PCSR (Ref. 18).

142. On this basis, I recommend that GDA Issue GI-AP1000-FS-05 is closed.

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