New Reactors Programme

GDA Close-out for the AP1000 Reactor

GDA Issue GI-AP1000-RC-01 Revision 0 – Accident Source Terms

Assessment Report: ONR-NR-AR-16-044
Revision 0
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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 opted to pause the regulatory process. At that time, it had achieved an Interim Design Acceptance Confirmation (IDAC), which had 51 GDA Issues attached to it. These GDA 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 issues.

This report presents the assessment conducted as part of the close-out of the Office for Nuclear Regulation (ONR) GDA for the **AP1000** reactor design within the topic of Reactor Chemistry. This report specifically addresses GDA Issue GI-AP1000-RC-01 Revision 0 and associated GDA Issue Action related to accident source terms.

GI-AP1000-RC-01 arose because the safety case provided at the end of Step 4 did not provide an adequate justification for two aspects of the accident analysis. The first of these was the justification for the adoption of standard industry guidance regarding the amounts and timings of radioactivity releases. The second was to provide a justification of the likely chemistry effects that may occur, in order to demonstrate the sensitivity of the results and conclusions to these. These are important for the **AP1000** reactor given the differences in accident sequences and phenomena that might be expected because of the design differences to other reactors.

In response to GI-AP1000-RC-01, Westinghouse provided a single main submission which summarised its work to consider and derive a plant-specific source term. This report was supported by the detailed analysis and a subsequent dose assessment that evaluates the impact of resolving this GDA Issue on the ability of the design to meet the SAPs numerical targets. In addition Westinghouse provided responses to my Regulatory Queries, providing additional clarification and evidence to support the main submission.

As a result of my assessment of these submissions, meetings and discussions with Westinghouse experts, and consultations with ONR colleagues in different technical areas, my conclusions are:

- Westinghouse has undertaken an analysis of the **AP1000** plant source term, using the latest versions of industry standard codes and incorporating its latest understanding of chemical behaviour during such events. This considered both the short and long term phases and the specifics of the plant design.
- This analysis specifically provides evidence for:
  - the overall fractions of released nuclides;
  - the timings of nuclide releases; and
  - the long-term behaviour of released nuclides under the containment conditions.
- The results of this analysis demonstrated that the assumptions used within the UK **AP1000** plant safety case, while representative, are not bounded by current industry representative severe accident source terms, in terms of both the timings and magnitude of releases to containment. This is due to a combination of better understanding, and therefore modelling, of such phenomena as well as the features of the **AP1000** design which mean that all Loss of Coolant Accidents (LOCAs) behave as though they are large hot leg LOCAs.
- Westinghouse has also demonstrated the sensitivity of their analysis to the most important chemistry-related assumption and has shown that it remains representative of any changes that might be reasonably expected.
- As a result, a dose assessment was performed that demonstrated margin for meeting the SAPs Target 8 Basic Safety Objective.
- Westinghouse has also provided sufficient arguments and evidence to resolve those specific points that remained incomplete from the Step 4 assessment.

- In response to this GDA Issue, Westinghouse proposed updates to the Pre-Construction Safety Report. I have reviewed these updates and am content that they reflect the responses to the GDA Issue. However, this has highlighted further matters to be considered regarding the coverage of the chemistry aspects of severe accidents. This will be considered further as part of GI-AP1000-CC-02.

As a result of this assessment, I have identified one Assessment Finding. This relates to the provision of site specific analysis. I am satisfied that this matter does not undermine the conclusions of the generic safety submission provided for GDA and requires both licensee and site specific information to resolve.

Overall, on the basis of my assessment, I am satisfied that GDA Issue GI-AP1000-RC-01 can be closed.

LIST OF ABBREVIATIONS

ADS	Automatic Depressurisation System
ALARP	As Low As Reasonably Practicable
BSL	Basic Safety Level
BSO	Basic Safety Objective
CDF	Core Damage Frequency
DAC	Design Acceptance Confirmation
EDCD	European Design Control Document
FP	Fission Product
FPS	Fire Protection System
GDA	Generic Design Assessment
IAEA	International Atomic Energy Agency
IDAC	Interim Design Acceptance Confirmation
IRWST	In-Containment Refuelling Water Storage Tank
IVR	In-Vessel Retention
LOCA	Loss Of Coolant Accident
MAAP	Modular Accident Analysis Programme
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
ONR	Office for Nuclear Regulation
PCS	Passive Containment Cooling System
PCSR	Pre-Construction Safety Report
PSA	Probabilistic Safety Assessment
PWR	Pressurised Water Reactor
PXS	Passive Core Cooling System
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SAP	Safety Assessment Principle
SGTR	Steam Generator Tube Rupture
SLOCA	Small-break Loss Of Coolant Accident
STCP	Source Term Code Package
TAG	Technical Assessment Guide
TSoP	Tri-sodium Phosphate
TQ	Technical Query
US NRC	United States Nuclear Regulatory Commission
Westinghouse	Westinghouse Electric Company LLC
WENRA	Western European Nuclear Regulators Association

TABLE OF CONTENTS

Tables

Figures

Annexes

1 INTRODUCTION

1.1 Background

1. This report presents the assessment conducted as part of the close-out of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA) for the Westinghouse Electric Company LLC (Westinghouse) **AP1000®** reactor design within the topic of Reactor Chemistry. The report specifically addresses the GDA Issue GI-AP1000-RC-01 Revision 0 and associated GDA Issue Action (Ref. 1) related to accident source terms.

2. GDA follows a stepwise approach in a claims-argument-evidence hierarchy. In Step 2, the claims made by Westinghouse were examined and in Step 3 the arguments that underpin those claims were examined. The Step 4 assessment (Ref. 2) reviewed the safety aspects of the **AP1000** reactor in greater detail, by examining the evidence, supporting the claims and arguments made in the safety documentation. Westinghouse completed Step 4 in 2011 and then opted to pause the regulatory process. At that time it, had achieved an Interim Design Acceptance Confirmation (IDAC), which had 51 GDA Issues attached to it. These GDA Issues require resolution prior to award of a complete Design Acceptance Confirmation (DAC) and before any nuclear safety-related construction of this reactor design can begin. Westinghouse re-entered the GDA process in 2014 to close the 51 GDA Issues.

3. The purpose of this report is therefore to provide the assessment that underpins the judgement made in closing GDA Issue GI-AP1000-RC-01. This assessment is focused on the deliverables identified within the Westinghouse resolution plan (Ref. 3) published in response to the GDA Issue and on further assessment that was undertaken of those deliverables.

4. The related GDA Step 4 report (Ref. 2) is published on the ONR website (www.onr.org.uk/new-reactors/ap1000/reports.htm), and this provides the assessment underpinning GI-AP1000-RC-01. Further information on the GDA process in general is also available on the ONR website (www.onr.org.uk/new-reactors/index.htm).

1.2 Scope

5. The scope of this assessment is detailed in the assessment plan (Ref. 4). Consistent with this plan, the assessment is restricted to considering whether the Westinghouse submissions to ONR for GI-AP1000-RC-01 provide an adequate response sufficient to justify closure of the GDA Issue. Importantly, it is not within the scope of this assessment to re-visit areas already found by ONR to be satisfactory unless, during my assessment, important safety issues emerged that required the expansion of my assessment scope.

6. As such, this report only presents the assessment undertaken as part of the resolution of GI-AP1000-RC-01 and it is recommended that this report be read in conjunction with the Step 4 Reactor Chemistry assessment of the **AP1000** reactor (Ref. 2) in order to appreciate the totality of the assessment undertaken as part of the GDA process.

7. This assessment focused on the justification provided for the source terms assumed in severe accidents states in the **AP1000** design. Under these circumstances the safety case uses information from standard guidance produced by the US Nuclear Regulatory Committee (NRC) regarding the timings and amount of radioactivity released for Pressurised Water Reactors (PWR). However this does not consider the specific accident phenomena that may occur; in particular the mitigation approach for molten core states involves In-Vessel Retention (IVR) of the molten core materials. GI-AP1000-RC-01 therefore required Westinghouse to justify that the use of such generic guidance is appropriate for these events, in addition to providing information on the

sensitivity of the conclusions to the assumptions used. The scope of assessment is therefore to assess the justification provided by Westinghouse and to ensure that this is captured within the safety case, as appropriate.

8. Further details of the scope of assessment can be found in Section 2.1 of my report.

9. Due to this scope, the structure of this report differs from that adopted for previous reports produced within GDA, most notably from the Step 4 Reactor Chemistry assessment (Ref. 2). This is because this report details the assessment of GI-AP1000-RC-01 only, rather than close-out of all GDA Issues associated with Reactor Chemistry. This allows closure of GDA Issues as the work is completed rather than waiting for the resolution of all work in this technical topic.

1.3 Methodology

10. The methodology for the assessment follows HOW2 Guidance on Mechanics of Assessment within the Office for Nuclear Regulation (ONR) (Ref. 5).

11. I have sampled all of the submissions made in response to GI-AP1000-RC-01, to various degrees of breadth and depth. I chose to focus my assessment on those aspects that I judged to have the greatest safety significance, as described more fully in my assessment. My sample has also been influenced by the claims made by Westinghouse, my previous experience of similar safety cases for reactors and other nuclear facilities and the specific gaps in the original submissions made by Westinghouse that led to the GDA Issue.

12. The Safety Assessment Principles (SAPs) (Ref. 6), alongside the relevant Technical Assessment Guides (TAGs) (Ref. 7), have been used as the basis for this assessment.

2 ASSESSMENT STRATEGY

13. The intended assessment strategy for resolution of GI-AP1000-RC-01 is set out in this section. This identifies the scope of the assessment and the standards and criteria that have been applied.

2.1 Assessment Scope

14. This report presents only the assessment undertaken for resolution of Reactor Chemistry GDA Issue GI-AP1000-RC-01, related to accident source terms (Ref. 1). This report does not represent the complete assessment of the **AP1000** reactor in the Reactor Chemistry topic area for GDA, or even the complete assessment of all aspects associated with chemistry during accidents. It is recommended that this report be read in conjunction with the Step 3 and Step 4 Reactor Chemistry assessments of the Westinghouse **AP1000** design (Refs 8 and 2) in order to appreciate the totality of the assessment undertaken as part of the GDA process. Further information on the assessment performed on the severe accident aspects of the **AP1000** design can be found in the Step 4 containment and severe accidents assessment report (Ref. 9). Section 3 of this report provides a brief overview of the background to GI-AP1000-RC-01.

15. This assessment does not revisit aspects of the safety case already accepted as being adequate during previous stages of GDA. However, where the assessment of the Westinghouse responses highlight shortfalls not previously identified during Step 4, or cast doubt on previously accepted arguments, these were assessed within this report.

16. The focus for this assessment was on two aspects of the accident analysis for the **AP1000** design. The first was the justification for the adoption of standard US NRC guidance regarding the amounts and timings of radioactivity releases, particularly regarding the accident sequence and phenomenology differences that might be expected in the **AP1000** design given the differences to other PWRs. The second was to provide a justification for the likely chemistry effects that may occur, to understand the sensitivity of the results and conclusions to these. These need to be reflected appropriately within the safety case, including any limits or conditions that result. This scope of assessment is appropriate for GDA because there needs to be a clear demonstration at the design stage that the assumptions used within the safety analysis are appropriate, otherwise it is possible that changes would be necessary either to the safety case or perhaps the plant engineering.

17. This report refers to 'source term' frequently. To avoid ambiguity this should be considered to have the same meaning as that defined within the SAPs (as per the glossary from Ref. 6), namely: 'Data on quantities of radioisotopes released in an accident, the location of the release and other related parameters from the facility needed as inputs to radiological consequence calculations.' Similarly, a 'severe accident' can be defined as (Ref. 6); 'An accident with off-site consequences with the potential to exceed 100 mSv, or to a substantial unintended relocation of radioactive material within the facility that places a demand on the integrity of the remaining physical barriers.'

18. Annex 1 of this report contains the full text of the GDA Issue and Action (Ref. 1). The Westinghouse resolution plan, which details the methods by which the requesting party intended to resolve this GDA Issue via identified timescales and deliverables, is contained in Ref. 3 and discussed further in Section 3.

2.2 Related GDA Issues

19. ONR's GDA Guidance to Requesting Parties (Ref. 10) states that the information required for GDA may be in the form of a Pre-Construction Safety Report (PCSR).

ONR guidance (NS-TAST-GD-051: The purpose, scope and content of nuclear safety cases, Ref. 7) sets out regulatory expectations for a PCSR. The PCSR is the highest level summary of the safety case and provides the links to the detailed arguments and evidence that may reside in a suite of supporting documentation.

20. At the end of Step 4, ONR and the Environment Agency raised GDA Issue GI-AP1000-CC-02 (Ref. 11), requiring Westinghouse to submit a consolidated PCSR and associated references to provide the claims, arguments and evidence to substantiate the adequacy of the **AP1000** design reference point. A separate assessment report has been prepared to consider the adequacy of the PCSR and closure of GDA Issue GI-AP1000-CC-02. Therefore, this report does not discuss the overall adequacy of the Reactor Chemistry aspects of the PCSR. However, this assessment does consider the specific aspects related to GI-AP1000-RC-01 and severe accident source terms.

2.3 Assessment Approach

21. The assessment was undertaken by examining the evidence provided by Westinghouse in response to GI-AP1000-RC-01. This was assessed against the expectations and requirements of the SAPs and other guidance considered appropriate. Forming the basis of the assessment undertaken to prepare this report were:

- submissions made to ONR in accordance with the resolution plan;
- consideration of internal and international standards and guidance, international experience, operational feedback and expertise and assessments performed by other regulators, especially their findings;
- interaction with other relevant technical areas (where appropriate);
- raising and issuing of Regulatory Queries (RQs) as appropriate, followed by assessment of Westinghouse responses; and
- holding technical meetings to progress the identified lines of enquiry.

22. The following subsections provide an overview of the outcome from each of the information exchange mechanisms in further detail.

2.3.1 Regulatory Queries

23. A total of three Regulatory Queries (RQs) were raised with Westinghouse for the assessment of GI-AP1000-RC-01. The responses to the RQs were assessed as part of this assessment. Commentary on the most important and relevant RQ responses is included in the assessment section later in this report as appropriate. The responses provided further evidence to support resolution of the GDA Issue.

2.3.2 Technical Meetings

24. A number of technical meetings with Westinghouse were held during assessment of the GI-AP1000-RC-01 responses. The principal focus of these meetings was to discuss progress and responses, to facilitate technical exchanges and to hold discussions with Westinghouse technical experts on emergent issues.

2.4 Standards and Criteria

25. This assessment has been undertaken in line with the requirements of NS-PER-GD-014 (Ref. 12). The standards and criteria adopted within this assessment are principally the SAPs (Ref. 6), internal TAGs (Ref. 7), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites. Further details are provided below.

2.4.1 Safety Assessment Principles

26. The key SAPs applied within this assessment are included within Table 1.

27. As the SAPs (Ref. 6) constitute the regulatory principles against which duty holders' safety cases are judged, they are therefore the basis for ONR's nuclear safety assessment. It is worth noting that the 2014 Edition (Revision 0) of the SAPs was used when performing the assessment described in this report, whereas the original Step 4 assessment used the 2006 Edition. From a Reactor Chemistry perspective the main change is that the current edition includes specific SAPs relating to chemistry (ECH.1 to 4).

2.4.2 Technical Assessment Guides

28. The TAGs (Ref. 7) that have been used as part of this assessment are set out in Table 2.

2.4.3 National and International Standards and Guidance

29. There are both International Atomic Energy Agency (IAEA) standards (Ref. 13) and Western European Nuclear Regulators Association (WENRA) Reference Levels (Ref. 14) of relevance. It should be noted that the latest version of the SAPs (Ref. 6) has been benchmarked against both IAEA and WENRA guidance.

2.5 Use of Technical Support Contractors

30. No technical support work was undertaken to support the assessment of the submissions made in response to GI-AP1000-RC-01.

31. However, a technical contractor was used during the Step 4 assessment to review severe accident chemistry, which included some of the matters discussed in this assessment. The output of this work (Ref. 15), which is reflective of the state of the Westinghouse safety case at that time, has been considered as part of this assessment.

2.6 Integration with Other Assessment Topics

32. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot therefore be carried out in isolation as there are often safety issues of a multi-topic or cross-cutting nature. To assess the adequacy of the submissions provided by Westinghouse for GI-AP1000-RC-01, I have required only limited input from other technical disciplines and the assessment reported here is consistent with this. As described in Section 2.2, this assessment was integrated with the wider requirements of GI-AP1000-CC-02 (PCSR).

2.7 Out of Scope Items

33. This assessment report for GI-AP1000-RC-01 focuses solely on the accident source terms. No specific items within the remit of this GDA Issue have been identified as out of scope.

3 REQUESTING PARTY'S SAFETY CASE

3.1 Overview of the Westinghouse Safety Case for Accident Source Terms

34. The assessment of risks arising from nuclear facilities needs to consider those arising both from normal operation and accident conditions. Conservative design, good operational practices, and adequate maintenance and testing should minimise the likelihood of accidents. Nuclear facilities are therefore designed to cope with, or are shown to withstand, a wide range of faults without unacceptable consequences by virtue of the facility's inherent characteristics or safety measures. The design of the **AP1000** reactor has many engineered safety systems, some passive and some active, which are claimed to avoid and ultimately mitigate such scenarios. It is important to note that Westinghouse claims that severe accidents (ie those resulting in core damage) are 'virtually excluded' and that the design has been optimised to minimise the risk of accidents. The Core Damage Frequency (CDF) claimed by Westinghouse is of the order of 2E-7 per year, with the large release frequency for these events even lower.

35. A severe accident generally arises because, even when the nuclear reaction is stopped, the core of a PWR still contains sufficient energy to require a period of cooling. Should this cooling fail or be sufficiently impaired the heat release can be high enough to damage the fuel, degrade the reactor core and ultimately lead to releases of radioactive species to the containment.

36. Westinghouse has designed the **AP1000** reactor with a range of passive and active safety systems which operate to prevent and mitigate accidents. I do not discuss all the safety features here; they are described in more detail in Refs 8 and 9. However, of particular relevance to this assessment is the Passive containment Cooling System (PCS) and the concept of IVR, which are described briefly below.

37. The **AP1000** plant containment building structure is described in the PCSR (Ref. 16). The reactor building comprises two concentric shells with the inner being a steel vessel and the outer being a steel and concrete structure.. The shells are separated by a ventilated annular space. The inner containment holds the primary circuit and portions of associated structures, systems and components. In the event of an accident that releases large amounts of steam (and hence heat) inside the containment, Westinghouse has equipped the **AP1000** reactor with a PCS which serves as the means of removing heat. The PCS uses the steel containment shell as a heat transfer surface; air is drawn from the environment via an 'always open' airflow path over the containment vessel and is returned to the environment after removing heat from the containment shell. The containment shell can be wetted by gravity draining of a water storage tank that is incorporated into the shield building structure above the containment (the PCS water storage tank). By keeping the metal containment shell cool, either by air flow alone or enhanced by evaporation of cooling water on the outside of the shell, condensation is induced on the inner surfaces. While this is primarily aimed at controlling the temperature and pressure in containment, it also forms the main system for removing radioactivity from the containment atmosphere. This cooling causes radioactive aerosols to deposit by enhancing the passive mechanisms that would occur anyway. This is shown in Figure 1 below.

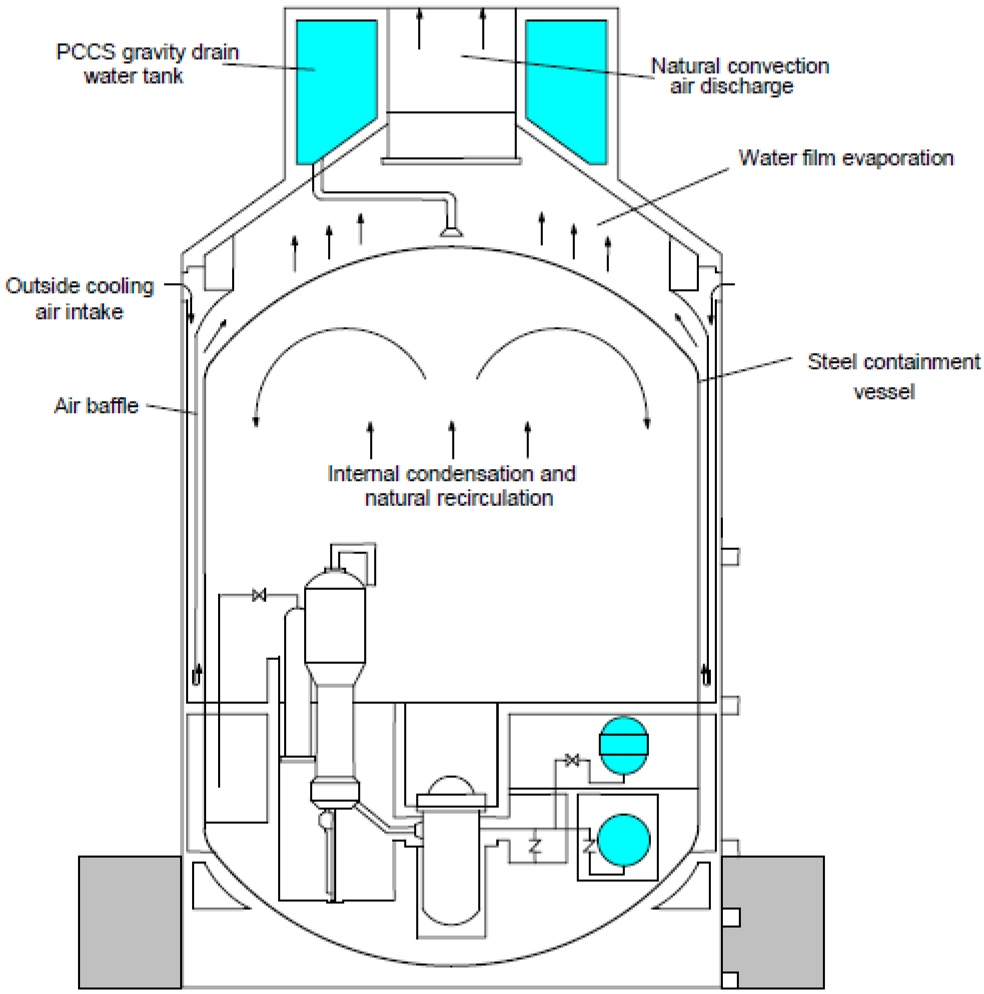

Figure 1: AP1000 reactor Passive Containment Cooling System (PCS)

38. The condensed steam from the inner containment surfaces, containing trapped radioactivity, is collected via a series of gutters and drains and returned to the In-Containment Refuelling Water Storage Tank (IRWST). In accident scenarios where draining of the IRWST occurs, the **AP1000** design includes a means to buffer the sump fluid by dissolving granulated Tri-Sodium Phosphate (TSoP) in the water which collects in the sumps, thereby minimising the evolution of volatile nuclides dissolved in this water.

39. In the most extreme severe accident cases, if the operator was unable to maintain sufficient cooling of the core, there could be a loss of cladding material and melting of the control rods, followed by degradation of the fuel itself. Within several hours the core would eventually degrade to a molten mixture of uranium dioxide, zirconium cladding, waste products and various structural materials such as steel; a mixture often called ‘corium’. The approach taken by the **AP1000** design differs from many other PWRs. By flooding the area surrounding the reactor pressure vessel, the reactor cavity, Westinghouse claim that the corium could be retained, even if it did melt. This strategy is known as IVR. This concept was considered during Step 4 in both Refs 8 and 9. Overall, Westinghouse claim that IVR is sufficiently reliable to ensure that the **AP1000** reactor exceeds UK regulatory expectations and pessimistically assumes that if IVR fails, then containment has also failed. Releases of volatile radionuclides from this hot molten corium over the period it remains within the reactor pressure vessel may be important to the overall source term.

40. The effectiveness of these systems depends on the accident sequence. This changes the extent and timing of the releases into containment and the chemical and physical

forms of the radioactive materials. These factors are highly dependent on the design of the plant and on the accident sequence itself. Therefore it is common practice for the analysis of such events to use standard guidance values for the source terms that may arise. For the **AP1000** design Westinghouse uses the values defined in NUREG-1465 (Ref. 17), produced by the US NRC. This contains information on the amounts, timings and chemical speciation of the release (for iodine). Westinghouse claim these are an appropriate set of source term values for use in the subsequent analysis.

3.2 Assessment during GDA Step 4

41. The assessment of the chemistry aspects of the Westinghouse safety case for accident source terms began during GDA Step 4 (Ref. 2). Since no single report described all the phenomena, Action 6 of RO-AP1000-55 (responded to in Ref. 18) was raised requiring Westinghouse to provide further justification and evidence for the design, and TQ-AP1000-877 (Ref. 19) to obtain further documentation (note that during Step 4 RQs were known as TQs (Technical Queries), but otherwise were the same). On the basis of the assessment of these, TQ-AP1000-1047, 1048, 1049, 1052, 1053 and 1054 (Ref. 19) were also raised. These covered various aspects of fission-product control. However, some of the responses to these TQs were provided too late in GDA to form part of the assessment at that time.

42. Despite this, at the end of GDA Westinghouse had provided much improved clarity on the radioactivity control mechanisms claimed in the **AP1000** design, which are based upon passive processes and backed by an ‘optional’ spray system for use during severe accidents. Many of the questions in Step 4 related to the expected performance of the mitigation processes. For design-basis accidents, ONR concluded that the natural deposition rate combined with a robust containment shell provided adequate protection for controlling iodine release from containment and that the use of the in-containment spray system was not justified (but could still be used if necessary). There were several questions outstanding on assumptions made for physical aerosol behaviour, production rates for several key species, justification for assumptions made in the LOCA (Loss of Coolant Accident) dose analysis and further details on calculations for pH. These are contained in TQ-AP1000-1047 to 1049 and 1052 to 1054. Assessment of the responses to these TQs therefore became part of GI-AP1000-RC-01.

43. Overall, while it was judged that an adequate safety case could be made for the **AP1000** reactor, at the end of Step 4 it was concluded that Westinghouse had not yet presented a consistent and structured safety case containing sufficient evidence, specifically:

- evidence to support the overall fractions of released nuclides;
- evidence to support the timings of nuclide releases; and
- evidence to support the long-term behaviour of released nuclides under the containment conditions.

44. As it is important that these aspects are justified as part of the generic safety case, GDA Issue GI-AP1000-RC-01 was raised.

3.3 Summary of the GDA Issue and Actions

45. The full text of GI-AP1000-RC-01 (Ref. 1) and the associated one Action is in Annex 1.

46. The overall requirement in the GDA Issue was to provide a justification that the source terms used within the **AP1000** plant safety case for accident analysis are reasonable. An important part of demonstrating this was to consider the sensitivity of the conclusions of this analysis to the chemistry assumptions, both in the short and long-term phases of the accidents.

47. As noted above, consideration of a number of TQ responses received late in Step 4 forms part of the assessment, albeit the consideration is not explicitly noted in the text of the Action.

3.4 Westinghouse Deliverables in Response to the GDA Issue and Actions

48. The Westinghouse resolution plan for this GDA Issue is given in Ref. 3. This provides details of the deliverables Westinghouse intended to provide to respond to the Action. The following section contains a brief description of the submitted deliverables that formed the basis of the assessment.

49. According to Ref. 3, to resolve GI-AP1000-RC-01, Westinghouse intended to demonstrate that the source term released into the containment during an accident was appropriate for the **AP1000** plant design and safety features. To do this, Westinghouse would create a UK specific report – **AP1000** Plant Accident Source Term Evaluation and Target 8 Compliance, UKP-GW-GL-098 (Ref. 20). This report would:

- define the accident sequences and boundary conditions as related to the Target 8 of the SAPs;
- perform a quantitative evaluation of these identified accident sequences;
- justify the application of NUREG-1465 (Ref. 17) to the **AP1000** plant including verification of key assumptions; and
- perform an impact assessment of results on other aspects of the plant design.

50. The resolution plan indicates that an important input into this report would be the description of 'best estimate' chemistry of the source term in the **AP1000** plant, including for the long-term time frame. This chemistry description provides the expectations for the post-accident chemistry within the reactor containment for comparison with the results provided in the report. The original intention was to provide this as a separate document; however during the work it was decided to incorporate this into Ref. 20. I accepted this as reasonable, given that the scope of work remained the same.

51. Ref. 3 also indicates that the quantitative evolution would be undertaken with the MAAP (Modular Accident Analysis Program) 4.07 code. This same code was used for the Step 4 submissions, for which details were available. However, since the end of Step 4 in 2011, version 5.03 of this code has been made available. Westinghouse decided to update their analysis to this latest version of the code, and hence also submitted the report containing this analysis, MAAP 5.03 Analysis of the **AP1000**® Plant Severe Accident Fission Product Source Terms to the Containment, UKP-SSAR-GSC-030 – (Ref. 21).

52. Finally, as part of the impact analysis noted above, Westinghouse determined it was necessary to produce a revised dose assessment – UK Severe Accident Dose Analysis for Target 8, UKP-SSAR-GSC-020 (Ref. 22). This was not explicitly noted in Ref. 3, but is implicit in completing the impact assessment referred to above.

53. In addition to the submissions detailed above, responses to the various RQs I raised also informed my assessment. These are referenced throughout Section 4.

54. Finally, Westinghouse provided an update to the PCSR to identify how the resolution of this GDA Issue would be reflected in the overall **AP1000** reactor safety case. This is discussed further in Section 4.

4 ONR ASSESSMENT OF GDA ISSUE GI-AP1000-RC-01

55. This assessment has been carried out in accordance with Purpose and Scope of Permissioning, NS-PER-GD-014 (Ref. 12).

4.1 Scope of Assessment Undertaken

56. The scope of my assessment is described in Section 2.1, alongside the description of the submissions which formed the basis for that assessment in Section 3.4.

4.2 Assessment

57. This section describes my assessment of the Westinghouse responses to GI-AP1000-RC-01.

58. I have structured my assessment around the main report provided to address the GDA Issue (Ref. 20).

4.2.1 Derivation of an AP1000 Plant Specific Source Term

59. In this part of my assessment I considered the approach, justification and conclusions regarding the source terms used for the **AP1000** design.

4.2.1.1 Source Terms from GDA Step 4

60. The text of GI-AP1000-RC-01 refers to 'the source term released to containment during accidents', which therefore covers faults ranging from within the design basis to severe accident conditions. The expectations for how to treat conservatism, uncertainty and chemistry effects within these different analyses differ, as explained more fully within the SAPs (Ref. 6). This is explained further below.

<u>Design Basis Faults</u>

61. The design basis safety case at the end of Step 4, and hence the ONR assessment (Ref. 23), was based largely on analysis presented within the European Design Control Document (EDCD) (Ref. 24). This presented a set of radiological consequences analyses, calculated following typical US methods and meeting dose limits prescribed by US NRC. In particular this used the approach described in regulatory guide 1.183 (Ref. 25). No attempt was made to compare these against Target 4 of the SAPs or to compare the adopted methodology against the expectations for the UK. While site-specific calculations are out of scope of GDA, confidence was needed that acceptable site-specific calculations would be possible in the future for the **AP1000** design. RO-AP1000-48 was therefore raised by the fault studies inspectors to provide this evidence.

62. Westinghouse's response to this RO is summarised in Ref. 26, and assessed by ONR in Ref. 23. In summary, these calculations demonstrated that it is possible to meet the Basic Safety Levels (BSL) defined in Target 4 of the SAPs (Ref. 6). However, to do so, a number of changes were necessary to the assumptions and methods used within the EDCD analysis. Ref. 23 states that 'while the assumptions made in the new radiological consequences analysis generally appear sensible and conservative (and therefore appropriate for GDA), some of them are rather arbitrary, inconsistent, or yet to be justified'. A number of these related to the chemistry aspects of the source terms. AF-AP1000-FS-46 was therefore raised requiring a future licensee to perform design basis site-specific radiological consequence analysis taking due cognisance of UK methodology assumptions and explicitly comparing against Target 4 of the SAPs.

63. Ref. 26 was not assessed by the chemistry inspectors during Step 4, aside from the chemistry effects during Steam Generator Tube Rupture (SGTR) events. The

approach for SGTR events was accepted as sufficient for GDA, but a number of aspects were identified where development was necessary by the licensee. Therefore, to capture these, three Assessment Findings were raised, AF-AP1000-RC-56, 57 and 58. Importantly, other DBA sequences, such as LOCAs, which were also re-calculated as part of the response to RO-AP1000-48 were not sampled. The chemistry during these faults is very different to that during an SGTR, in particular the radioactivity release during a LOCA is within containment and therefore the effects caused by the PCS are not considered. In Ref. 21, Westinghouse noted that the offsite doses of scenarios with frequencies greater than 1E-6 per year (ie US threshold for design basis faults) are bounded by the results of the dose analysis as reported for large LOCAs in Ref. 27. This analysis included the release of fission product in the reactor coolant and fuel clad gap to the containment atmosphere.

64. While these specific accident sequences were not assessed, many of the technical justifications for the adequacy of the mitigation measures induced by the PCS were (Ref. 2, Section 4.6.4). As described in Section 3.2 a number of outstanding TQs remained open, but the main conclusion of Ref. 2 was that an adequate case could be made, provided the source term was demonstrated to be adequate.

65. Given that the response to RO-AP1000-48 showed the analysis performed in support of GDA to be demonstrably conservative and AF-AP1000-FS-46 has already been raised to perform site-specific analysis (and this will include relevant chemistry effects), I am content not to revisit the design basis analysis as part of GI-AP1000-RC-01. However, I will specifically reconsider if this is a reasonable position after assessing the responses to the GDA Issue (ie does this highlight shortfalls not previously identified during Step 4, or cast doubt on previously accepted arguments).

Severe Accidents

66. The activity release model applied to severe accidents that involve core degradation is based on NUREG-1465 (Ref. 17). For example, the original large break LOCA with core melt analysis (Ref. 28) considers the gap release phase and the early in-vessel (partial core melt) phase. Since it is assumed by Westinghouse that IVR is successful, and therefore the reactor vessel does not fail, no ex-vessel or late in-vessel releases were considered. The source term defined within NUREG-1465 is given in Table 3 below.

FP Group	Percentage Release				
	Gap release	Early In-vessel	Ex-vessel	Late In-vessel	Total
Noble gases (Xe, Kr)	5	95	0	0	100
Halogens (I, Br)	5	35	25	10	75
Alkali metals (Cs, Rb)	5	25	35	10	75
Tellurium group (Te, Sb, Se)	0	5	25	0.5	30.5
Barium, strontium (Ba, Sr)	0	2	10	0	12
Noble metals (Ru, Rh, Pd, Mo, Tc, Co)	0	0.25	0.25	0	0.5
Cerium group (Ce, Pu, Np)	0	0.05	0.5	0	0.55
Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	0	0.02	0.5	0	0.52
Duration / hrs	0.5	1.3	2	10	13.8

Table 3: NUREG-1465 Source Terms

67. Westinghouse has chosen to apply some of the NUREG-1465 recommended release fractions, with differing timings, by assuming that all releases from the core stop after 1.8 - 2.0 hours when 40% of iodine and 30% of the alkali metals have been released to the containment. However, the NUREG guide also recommends a further 35% and 45% release for the halogens and alkalis respectively after this period, including a 10% release fraction from the 'late In-vessel' phase, which was not used by Westinghouse. In other words, there is no further release during or following the IVR phase.

68. The release model also considers the different chemical forms of iodine. It is assumed that 5% of the iodine releases is in a volatile form (either elemental or organic). It is assumed that 3% of this elemental iodine reacts with organics to form organic iodine. The remaining 95% of the iodine released is assumed to be in a particulate (ie non-volatile) form. Thus, the iodine chemical fractions modelled are 0.95 particulate, 0.0485 elemental, and 0.0015 organic. All other nuclides, except for noble gases, are modelled as particulates.

69. Importantly it is noted in Ref. 17 that 'Source terms for future reactors may differ from those presented in this report which are based upon insights derived from current generation light-water reactors [pre-1995]. An applicant may propose changes in source term parameters (timing, release magnitude, and chemical form) from those contained in this report, based upon and justified by design specific features.'

70. The key question that needs to be resolved for GI-AP1000-RC-01 was therefore whether these assumptions are appropriate given the design of the **AP1000** plant. The consideration of this during severe accident states is bounding of the demand placed on the mitigation features (eg PCS) during design basis events that involve radioactivity release inside containment. Therefore the approach taken by Westinghouse is to justify it for these most penalising severe accident sequences.

4.2.1.2 Derivation of the Releases to Containment in Severe Accidents

71. The approach taken by Westinghouse to derive the source term for AP1000 considered two stages. The first is the release from the primary circuit to the containment atmosphere and the second is the subsequent interactions of any activity once inside the containment. The first of these used the MAAP code and is discussed further below.

72. NUREG-1465 (Ref. 17) was developed to provide a more realistic source term for a core melt scenario than previous models which assumed an instant release of 100% of core noble gases and 50% of core iodine. In summary, the source term presented therein was developed from a spectrum of accident scenarios and PWR plants using the Methods for Estimation of Leakages and Consequences of Releases (MELCOR) and Source Term Code Package (STCP) computer codes to determine a representative generic source term that could be used for dose analyses. In effect, it provides the average release fractions (or percentages) for all the release phases associated with a complete core melt. Ref. 17 notes that the only accident phase considered bounded is the first, gap release phase which was chosen to be conservative; I note that this further supports the bounding nature of the design basis analysis, Ref. 27, which includes only this phase.

73. The approach taken by Westinghouse in Ref. 21 (and summarised in Ref. 20) was similar to that of NUREG-1465, but instead used the latest MAAP 5.03 code and considers specific **AP1000** plant accident sequences.

Scenario Selection

74. The first stage involved the selection of an appropriate range of accident scenarios for subsequent analysis. To do so Westinghouse made use of the latest Level 2 at-power Probabilistic Safety Analysis (PSA) results for the **AP1000** reactor.

75. This demonstrated that those sequences that involve core melting occur at a frequency lower than the Basic Safety Objective (BSO) in Target 8 of the SAPs (Ref. 6) (the CDF is 2E-7 per year). This target is therefore met. However, a severe accident source term was developed to demonstrate that the offsite doses do not exceed the maximum dose (1000 mSv) for the more likely severe accident scenarios and to demonstrate the performance of the containment mitigation features. This is an important part of demonstrating that resultant risks from severe accidents have been reduced As Low As Reasonably Practicable (ALARP).

76. Westinghouse selected sequences that resulted in an intact, isolated containment (known as release category INTACT in the PSA). These sequences also include a probability that the containment leaks excessively beyond the design leak rate (of 0.1% per day), which further broadens the coverage of sequences considered. All the sequences which result in containment failure were excluded (as they are already known to result in doses > 1000 mSv, and therefore exceed the BSO). The accident sequences were grouped into the relevant plant damage states and a representative case selected for analysis. This results in the selection of the following four representative cases:

- Reactor Pressure Vessel (RPV) rupture;
- Small Loss of Coolant Accident (SLOCA) along with a Passive Core Cooling System (PXS) injection failure;
- SLOCA along with a PXS recirculation failure; and
- SLOCA along with an Automatic Depressurisation System (ADS) stage 4 failure.

77. Collectively these scenarios represent nearly 85% of the CDF for the **AP1000** design. I queried what other cases were not considered, and why, in RQ-AP1000-1618 (Ref. 29). The response clarified that a number of other, lower frequency cases were screened out of the analysis (on low frequency); these account for around a further 5% of the CDF. The remaining 10% of CDF comprises cases that result in containment failure. The selected scenarios are detailed in Table 4 below.

Group	Group Frequency / per year	% CDF
RPV Rupture	3.0E-8	17.6
LOCA fail PXS Injection	7.3E-8	43.1
LOCA fail PXS Recirculation	1.2E-8	7.2
LOCA and fail ADS-4	2.9E-8	16.5
Total	1.4E-7	84.4

Table 4: Representative Accident Sequences Selected for Analysis

78. I am therefore satisfied that this is a reasonable scenario selection on which to derive the **AP1000** plant source term.

MAAP Analysis

79. As noted earlier Westinghouse performed the analysis for the selected scenarios using the MAAP 5.03 code. MAAP is an industry standard code used for these purposes and is categorised as an 'integral severe accident analysis tool' which means that it integrates a large number of phenomena into a single plant simulation (ie primary circuit, containment and auxiliary building). This same code is used throughout the safety case. In response to GI-AP1000-RC-01, Westinghouse chose to upgrade to the latest version.

80. During the early phases of this work it was noted by Westinghouse that the analysis was not providing the results that were expected (importantly, notably larger releases). In using this new version, Westinghouse recognised that there are options regarding how to set up the Fission Product (FP) models. Ref. 21 describes this in more detail, but they relate to diffusion parameters and the speciation of Cs assumed (to favour Cs_2MoO_4 instead of the default CsI and CsOH). These are based on learning from more recent modelling (but with the different MELCOR code) of experimental results, such as the PHEBUS and VERCORS tests, undertaken by Sandia National Laboratories in the US (Ref. 30). Another factor relates to the design of the **AP1000** plant. The design includes an ADS which acts to depressurise the primary circuit (via connections to the hot legs) via four stages. This means that all LOCAs, no matter where they occur, behave like a large hot-leg break LOCA for accident sequences that lead to actuation of the last, and largest, fourth ADS stage.

81. To understand the significance of these differences a suite of sensitivity analyses was undertaken and benchmarked against a 'standard' PWR response (taken from Ref. 31). To do this the **AP1000** model was made to behave like a conventional PWR by disabling all the passive safety systems. The results demonstrated that:

- the revised FP models give better agreement with the benchmark; and
- FP deposition in the primary circuit from a cold-leg break can be significantly more than the equivalent hot-leg break. For the reasons above (the ADS system) significant retention is not expected to occur for the most likely accident scenarios in the **AP1000** design.

82. Westinghouse confirmed in response to RQ-AP1000-1618 (Ref. 29) that the ADS system is the most significant factor in the differences observed in the analysis. The results in Ref. 31 supported this conclusion. I am therefore satisfied that this sensitivity analysis demonstrates that the change of code has been appropriately considered by Westinghouse (and importantly that the differences in results are not simply due to updating the code).

83. Ref. 21 contains full details of the analysis conducted on the four scenarios identified in Table 4. A range of outputs are presented that show the transient plant behaviour, timings and outputs. Each analysis was run until the in-vessel phase showed no further releases of radioactivity. Due to the findings above regarding the impact of the ADS on the retention of FP within the primary circuit, the scenarios were analysed for both a hot-leg and cold-leg break (except for the RPV rupture case). Therefore seven specific analyses were performed as input to the development of the **AP1000** severe accident source term.

Results

84. Based on the results of the analysis, Westinghouse derived the percentage of the FP that are released from the primary circuit to the containment atmosphere. Inside containment they would be subsequently available for leakage and subject to the effects caused by any mechanisms that occur inside the containment atmosphere. The effects that occur inside containment were considered outside of the MAAP code, and

are discussed later in my assessment (Section 4.2.1.3). The percentages were calculated for each FP group as the frequency-weighted average of the average scenario source term. In the same way, the timings are derived from the frequency-weighted average of the analyses.

85. The MAAP analysis provided the total iodine release, which is a percentage release of [redacted]%. This value was relatively insensitive to the method used by Westinghouse, for example assuming only cold-leg or hot-leg breaks varies the figure by [redacted] to [redacted]%. It was assumed that this iodine is released in accordance with the NUREG-1465 speciation assumptions (namely, 95% particulate, 4.85% elemental and 0.15% organic). However, in the ADS-4 failure cases the release passes through the IRWST water. The IRWST acts as a suppression pool for the ADS stages 1, 2, and 3 blowdown and scrubs aerosols for releases that occur while the ADS sparger is submerged. The benefit is limited, however because the IRWST does drain in severe accidents and the degree of scrubbing is reduced as the water level is reduced. The water drains to the containment, which is pH controlled to minimise re-evolution of volatile iodine. MAAP can derive values for the retention within the IRWST water, which varies during the accident but is at least reduced by a factor of 2. Westinghouse conservatively assumed this reduction. This alters the iodine split fractions to [redacted]% particulate, [redacted]% elemental and [redacted]% organic.

86. These results are summarised in Table 5 below, considering the 'gap release' and 'early in-vessel' phases as defined in NUREG-1465. Also included for comparison are the comparable NUREG-1465 and results generated by Sandia National Laboratories for high burn-up fuels using the MELCOR code. Note also that the FP groupings differ slightly from Table 3, due to the different assumptions in MAAP 5.03.

FP Group	Percentage Release		
	NUREG-1465 (Ref. 17)	SAND2011-0128 (Ref. 32)	Westinghouse Analysis (Ref. 21)
Xe, Kr	100	95.7	[redacted]
I, Br	40	37.4	[redacted]
Cs, Rb	30	23.3	[redacted]
Ba, Sr	2	0.5	[redacted]
Te, Sb, Se	5	30.4	[redacted]
Mo	0.3	8.0	[redacted]
Ru	0.3	0.6	[redacted]
La	0	0	[redacted]
Ce, Pu	0.1	0	[redacted]

Table 5: Comparison of the **AP1000** reactor and Other Source Term Percentage Releases

87. There are clearly some differences between the various analyses. While both the Sandia and Westinghouse results indicate increased releases for Mo and the Te group, the most significant difference is that the **AP1000** plant analysis, unlike Sandia, also shows higher releases of iodine. This is of particularly importance in radiological consequence assessments, given the biological significance of this nuclide. Westinghouse attributed most of these differences to advances in the understanding and modelling of such severe accidents since the development of NUREG-1465. The rationale provided for these were:

- two opposing effects are impacting the Cs release. It is decreased slightly because of the change in form to Cs_2MoO_4, rather than CsOH. However, it is also increased because the amount of iodine, and hence CsI, is increased;
- the Mo release fraction is increased significantly to match the Cs speciation; and
- the Te release fraction is increased because it is not expected to be in a form that is retained in the primary circuit.

88. The biggest impact, of increased iodine releases, seems to be a function of the specific **AP1000** design and in particular the ADS system. I accept these justifications as reasonable.

89. As noted earlier one of the specific queries raised about the application of NUREG-1465 to the **AP1000** plant was why the late in-vessel phase was not considered. During this phase further volatile nuclides may be released. These are the nuclides which deposit in the primary circuit during the early phase of the accident. Ref. 21 provided an explanation for Westinghouse's belief that volatilisation, resuspension and release of the retained FP will not occur. Namely:

- during the IVR phase the RPV and primary circuit piping, where the deposited FP reside, are externally cooled with water (and likely refilled with water);
- the ADS system turns the most likely events into large LOCAs, the amount of deposition is lowered so there is a smaller amount to release later; and
- the cooling water is pH controlled to remain alkali to retain any entrained iodine.

90. These seem reasonable arguments, which are supported by the analysis presented in Ref. 21. Further supporting arguments were also provided in the response to TQ-AP1000-1049 (Ref. 19). I therefore accept these arguments.

4.2.1.3 Consideration of In-containment Chemistry Effects

91. The second part of the Westinghouse methodology considered those effects that occur inside the containment atmosphere.

92. Detail of the consideration given to the in-containment chemistry effects were provided in Ref. 21 and summarised in Ref. 20. I have also considered the information in Ref. 18 as part of this assessment, as this contained additional evidence regarding FP behaviour in the **AP1000** design. A number of the unassessed TQ responses received late in Step 4 were also of relevance here.

93. Most of the FP released into containment do not require any detailed consideration of chemistry effects. Xe and Kr are non-reactive noble gases, whereas organic iodine species are conservatively assumed to behave in the same manner. With the exception of elemental iodine, all other FP species (including metal iodides), are assumed to be aerosols and are therefore subject to agglomeration, settling and phoretic deposition. If these aerosols are sparged through a water pool, such as the IRWST, they can also be scrubbed into the water. The aspects that require more consideration are therefore related to the behaviour of iodine, which is a reactive species and can undergo a number of reactions and interactions once inside the containment.

<u>Iodine Speciation</u>

94. As described previously Westinghouse assumed that the iodine released into the containment is released according to the NUREG-1465 speciation. This is based upon preceding analysis by US NRC, notably NUREG/CR-5732, NUREG/CR-5950, NUREG/CR-4327 and WASH-1233 as discussed in Ref. 17, from the 1970's and 80's. At the time of preparation, this speciation was deemed to be conservative. In Ref. 21

Westinghouse states that it still consider this to be conservative, but acknowledge that this remains an area of uncertainty. Ref. 32 was cited as a more recent example of analysis that concludes that this speciation remains appropriate. Conversely, many papers query aspects of this, for example Ref. 33. Given the significance of this assumption I requested evidence of the sensitivity of the analysis to this in RQ-AP1000-1727 (Ref. 29). In response Westinghouse recalculated the final dose analysis (discussed later, Section 4.2.2) assuming a lower organic iodine fraction of 1% (or 0.05% overall; therefore the remainder is 95% particulate and 4.95% elemental). This lowered the resultant offsite dose by around [redacted]%. This demonstrated the relative significance of organic iodine, which is difficult to remove, but did not consider other speciation changes. Further information on this was provided in response to my questions regarding the condensate water volume, discussed in Para. 101 below. Based on these results it is clear that a larger change in the speciation (by many percent) would be necessary to significantly change the final dose analysis results.

95. As part of TQ-AP1000-1049 (Ref. 19) Westinghouse was requested to justify the proportion of organic iodine. Similarly TQ-AP1000-1053 (Ref. 19) queried the fate of deposited iodine and the fraction converted to organic form by contact with paints and other reactions. The response to these queries supplemented the arguments and evidence provided in Ref. 18. Attempts were made to relate the behaviour to more recent experiments, such as PHEBUS. It is notable that some of the arguments have since changed, for example the assumption on Cs form discussed earlier. Others remain valid but are (conservatively) not considered in the analysis, such as the reactions of silver and iodine. Overall, I am content that Westinghouse has considered these appropriately as part of GI-AP1000-RC-01, although CP-AF-AP1000-RC-01 raised later in my assessment is relevant.

<u>Iodine Volatility</u>

96. The main claim made by Westinghouse regarding the control of iodine within the containment is that it is rapidly trapped within water and transferred (via condensation on the inner surfaces caused by the PCS) to the sump which is pH controlled to minimise subsequent volatility.

97. Firstly it is worthwhile considering the pH control aspects. During Step 4 Westinghouse provided additional analysis to demonstrate the adequacy of this via calculations in Ref. 34 and discussed in Ref. 18. This considered the resultant pH possible within the containment water volumes when account is taken of the dissolved TSoP and acidic species that may accumulate from within containment (such as boric acid and acidic radiolysis products). In the **AP1000** design there are three water volumes that need to be considered; the containment sump, the IRWST, and water films (on the containment shell). This latter is particularly important given the PCS.

98. Ref. 34 was assessed during Step 4 (at Revision 0), and appeared to provide a pessimistic assessment of the resultant sump pH to demonstrate that the Westinghouse claim of a minimum pH of 7.2 is achieved and maintained within several hours of the start of an accident. Conservatisms included maximising the boric acid content, assuming conservative dissolution rates for the TSoP and adding margin over the calculated minimum mass of TSoP required. The response to TQ-AP1000-1048 (Ref. 19), which referred to Ref. 34, is also relevant. I am satisfied that the analysis conducted as part of GI-AP1000-RC-01 does not undermine these conclusions.

99. The resultant IRWST pH differs from the sump. The IRWST does not itself have pH control. During an accident IRWST water is drained to the sump where it is used to provide boration and IVR cooling and is thereby evaporated as steam. This is condensed inside containment and returned back to the sump, via the IRWST. Over the course of the accident the IRWST therefore collects the FP retained within the condensate, some of which are further transferred to the sump dependent upon the

rate of water turn over. The approach adopted by Westinghouse in Ref. 34 was to maximise the amount of FP deposited into the IRWST volume (via spurious ADS-1 opening with failure of ADS-4 valves) and acidic species produced. The net effect was to maximise the volatile iodine evolution. In Ref. 34, the calculated pH of the IRWST drops to around 5 which results in around 20% of the iodine being in the elemental form. The volatilisation of this to the gaseous phase depends on the partition coefficient, which using the assumed coefficient resulted in around [redacted] grams of iodine being evolved from the IRWST. However, one of the factors affecting the pH is the amount of CsOH. As described earlier (Para. 80), Ref. 21, reduces the amount of CsOH assumed by changing the main Cs species to Cs_2MoO_4. To examine the effect of this sensitivity calculations are performed assuming no CsOH enters the IRWST. With all other assumptions the same, this increased the amount of iodine released from the IRWST to [redacted] grams.

100. The other difference indicated by Ref. 21 is the increased release of iodine. The analysis in APP-PXS-M3C-036 (Ref. 34), which is based on MAAP 4.04, peaks at a CsI concentration in the IRWST of around [redacted] kg. The same values derived from the Ref. 21 calculations showed varying amounts of dissolved CsI, which ranged from around [redacted] kg in the most frequent scenario to [redacted] kg in the ADS4 failure case. Applying the same frequency averaging as is used to determine the release percentages gave a value of [redacted] kg. The effect of increasing the iodine concentration would be to make a greater proportion into volatile forms, see Figure 2 below (from NUREG/CR-5950, Ref. 35). However, the assumptions in Ref. 34 are very pessimistic for the amounts of acidic radiolysis gases produced and therefore underestimate the resultant pH; more realistic assumptions would result in a higher pH and therefore lower volatile fraction. In any case, the resultant pH and CsI concentration achieved (even assuming a higher concentration and lower pH) do not enter the region where iodine has any significant volatility. Ref. 34 is therefore conservative for the GI-AP1000-RC-01 responses.

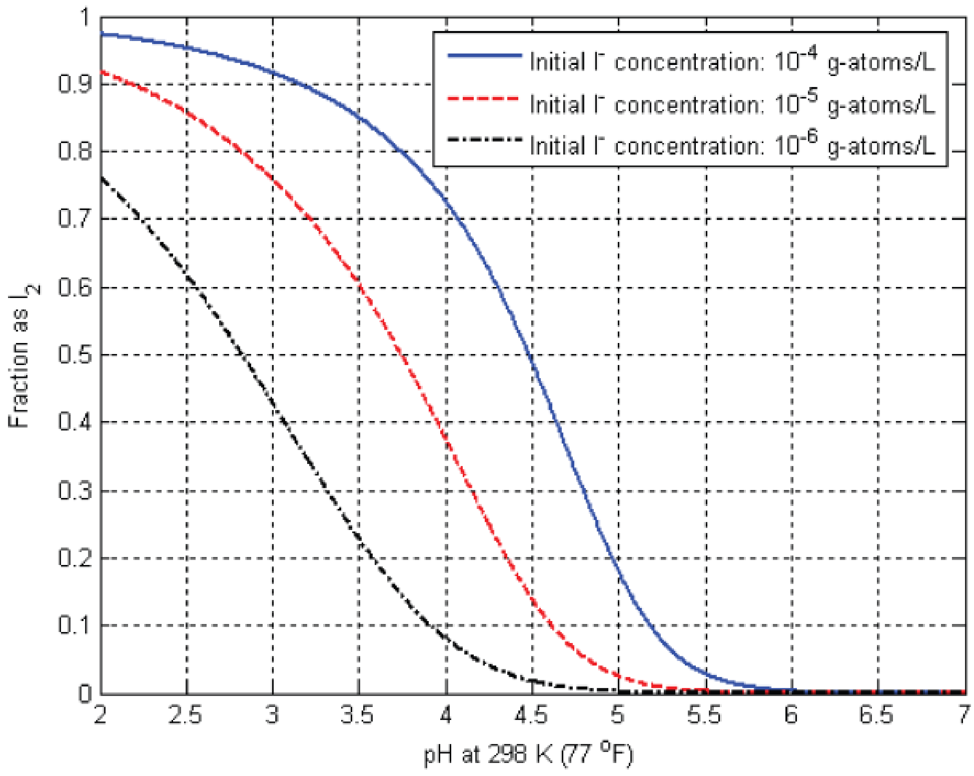

Figure 2: Fraction of Iodide Conversion to Iodine as a Function of pH and Concentration (Ref. 35)

101. The final water source within containment is the condensate films formed on internal surfaces by the action of the PCS on condensing steam. Like the IRWST these films will not be pH controlled and are therefore subject to changes brought about by the various substances that dissolve in them, thus affecting their ability to retain iodine. The response to Action 6 of RO-AP1000-55 (Ref. 18) does not discuss this effect. This was queried in TQ-AP1000-1053 (Ref. 19). The arguments put forward by Westinghouse in response are related to the efficiency of the condensate removal, buffering of the condensate by CsOH and differences between the respective timings for the release of FP and acid generation from radiolysis. These are generally reasonable arguments (albeit that the second regarding CsOH may no longer be valid, given the assumption used in Ref. 21 regarding the chemical form of Cs). I was expecting this to be explicitly reconsidered as part of the responses to GI-AP1000-RC-01 (Refs 20 and 21), but it was not. I therefore raised RQ-AP1000-1618, later followed by RQ-AP1000-1727 (Ref. 29), requesting evidence of the sensitivity of the results to this.

102. In response to RQ-AP1000-1727 (Ref. 29) Westinghouse provided a sensitivity analysis of the 'LOCA fail PXS Injection' case from Ref. 21. This used the acid production rates from Ref. 34, but with less conservative assumptions (on dose rate and the amount of affected cabling). The analysis assumed only CsI, CsOH and HCl are collected within the condensate films. Using a similar approach to Ref. 34 regarding pH and partitioning of iodine resulted in an iodine release from the condensate film of [redacted] grams. This is comparable to that calculated to be released from the IRWST. However, Westinghouse further argue that it is not possible to account for both of these effects simultaneously (ie it is not [redacted] plus [redacted] grams) because applying the same assumptions for the acid generation rates to the IRWST calculation would reduce release here by an order of magnitude. The overall effect (of both condensate films and IRWST) is therefore around the order of [redacted] grams. Further sensitivity studies showed the effect of parameter variations, although in all cases the amount calculated to be released remained well below that necessary to exceed the Target 8 in the SAPs (Ref. 6).

Other Iodine Reactions

103. In Ref. 21 Westinghouse also considered a number of other potential iodine reactions inside the containment, including decomposition of CsI to iodine following combustion, iodine oxide reactions and radiolytic destruction of organic iodine. The effects of these can be both beneficial or detrimental to the release of iodine species from the containment. Arguments were provided as to why these effects are not specifically considered as part of the analysis in Ref. 21. These mainly relate to uncertainties in their application due to shortfalls in understanding of precise post-accident conditions that may be expected, their importance to the overall behaviour, or non-applicability due to the specific conditions expected in the **AP1000** design. I am satisfied with these arguments in the context of GI-AP1000-RC-01, as I am content that the most significant iodine related phenomena are captured above and a degree of conservatism remains within the analysis overall.

104. However, I consider that it would be beneficial for the future licensee to consider these matters further as part of its site specific analysis. **I consider this to be an Assessment Finding**:

CP- AF-AP1000-RC-01: The licensee shall provide site specific analysis for the radiological consequence of accidents involving core melting. This should include consideration of the uncertainties in the reactions of iodine and other in-containment phenomena which could affect the releases to the environment. This should include evidence which demonstrates that the results and conclusions have been appropriately reflected into the affected safety case documents.

Aerosol Behaviour

105. The removal of aerosols in the **AP1000** design is enhanced by the operation of the PCS, which creates a driving force to increase the rates of the natural removal mechanisms, gravity (sedimentation) and heat transfer driven processes (diffusiophoresis and thermophoresis). Sedimentation accounts for around 30% of the overall aerosol removal, with the rest accounted for by the phoretic processes, which vary in their contribution as the accident progresses but are somewhat complementary such that the effect of one increases as the other decreases. The basis, evidence and application of these within the Westinghouse analysis was assessed in Step 4 (Ref. 2).

106. The conclusion at that time was that, overall, the physical processes that govern the removal of aerosols in the **AP1000** plant are treated reasonably. A number of detailed queries were raised in TQ-AP1000-1047 (Ref. 29) and Westinghouse responded to these as part of Ref. 20. These clarified a number of assumptions and their relative importance within the approach used by Westinghouse in calculation of the containment lambda (effectively the measure of the removal rate for aerosols). These responses are sufficient to resolve the questions. I do note that the containment aerosol removal rate is notably quicker than in similar cases I have seen, due to the PCS.

107. An implicit assumption in the **AP1000** approach is that the aerosol deposited on surfaces is washed off by the condensate film. Evidence for this was requested in TQ-AP1000-1053 (Ref. 29) during Step 4. The response provided a number of arguments to support this including the relative timings for film development versus FP release, the rate of steam condensation, the solubility of the various potential species in that condensate and tendency of the process to 'self-compensate' for dried areas (ie dry areas lead to cooler surfaces which enhance condensation). The efficiency of the water return to the IRWST was also the subject of GI-AP1000-FS-06 (Ref. 35). I therefore accept these arguments as reasonable, and consider it unlikely that this process would falter to an extent large enough to be of concern for aerosol removal.

108. I am therefore content that the chemistry related aspects of aerosol behaviour have been given adequate consideration by Westinghouse.

Containment Spray Operation

109. Like many other PWRs the **AP1000** design features a containment spray function, which is a sub-system of the Fire Protection System (FPS). Unlike other plants this is only designed for fire protection, and not for heat and pressure removal from the containment (instead the PCS serves this function). An additional benefit of sprays is in the removal of aerosols from the atmosphere. Westinghouse claim that the containment spray sub-system can be used, if necessary, for this purpose. On the basis of the Step 4 assessment (Ref. 2), AF-AP1000-RC-64 was raised which requires the licensee to 'review severe accident management guidelines for the provision of spray water for fission product control and consider whether any improvements to the containment spray sub-system design or performance would be reasonably practicable'. On the basis of the responses to this GDA Issue I remain content that this Assessment Finding is valid and is an appropriate way to resolve these questions. This would include matters such as spray timings, durations, capacity and efficiency.

110. The unresolved question from Step 4 was whether a recirculating spray system would further reduce risks to ALARP. Westinghouse's arguments and evidence for this are presented in Ref. 18. At that time the conclusion of ONR's assessment was that the additional benefit of a recirculating spray would be small, provided the amount of iodine evolved late in the accident was also small. I am content that the analysis provided in response to Ref. 21 does not change this conclusion, with the majority of FP releases occurring early in the accident and later iodine releases are low. It was

further confirmed in the resultant dose analysis (Ref. 22) that the later iodine releases contribute less than 5% of the overall dose. I am therefore satisfied that AF-AP1000-RC-64 is an appropriate way to take this aspect of the design forward.

Results

111. Based on this consideration of in-containment chemistry effects two further changes to the derived source terms were made. The first of these is to further modify the iodine speciation to reflect the releases from non-pH buffered water in the IRWST or condensate films. The second is to explicitly account for the timings and amount of this additional release to the containment. This is given in Table 6 below, which again includes comparison to other source term guidance.

FP Group	Percentage Release		
	NUREG-1465 (Ref. 17)	SAND2011-0128 (Ref. 32)	Westinghouse Analysis (Ref. 21)
Xe, Kr	100	95.7	[redacted]
I, Br *[Particulate I]* *[Elemental I]* *[Organic I]*	40 *[95 %]* *[4.85 %]* *[0.15 %]*	37.4 *[95 %]* *[4.85 %]* *[0.15 %]*	[redacted]
Cs, Rb	30	23.3	[redacted]
Ba, Sr	2	0.5	[redacted]
Te, Sb, Se	5	30.4	[redacted]
Mo	0.3	8.0	[redacted]
Ru	0.3	0.6	[redacted]
La	0	0	[redacted]
Ce, Pu	0.1	0	[redacted]
Start	0.17 hr	0.16 hr	[redacted]
End	1.97 hr	8.16 hr	[redacted]
Duration	1.8 hr	8 hr	[redacted]
Long-term Iodine release *[Particulate I]* *[Elemental I]* *[Organic I]*	-	-	[redacted]
Duration	-	-	[redacted]

Table 6: Comparison of the **AP1000** reactor and Other Source Terms

112. While the calculated **AP1000** severe accident source term is representative of the results provided within NUREG-1465 (Ref. 17) and SAND2011-0128 (Ref. 32), it is confirmed that it is not bounded by these. The most significant differences relate to the increased fraction of iodine releases, and the change in speciation of this to more volatile forms, and the increased Mo releases caused by consideration of Cs as Cs_2MoO_4.

4.2.2 Impact Assessment

113. As the results in Ref. 21 demonstrated that the source term assumed for the UK **AP1000** reactor is representative, but not bounded by the assumptions in NUREG-1465, WEC undertook a dose assessment using the Ref. 21 source term. This is reported in UKP-SSAR-GSC-020 (Ref. 22). The purpose of this calculation was to demonstrate that Target 8 within the SAPs (Ref. 6) can still be met.

114. In Ref. 22 the RADTRAD code was used to model activity transport, removal, and decay to determine the activity releases to the environment and the resulting offsite doses. This is a standard methodology applied elsewhere in the safety case by Westinghouse (for example in Ref. 27). The analysis specifically considered containment leakage from radioactivity released due to both core degradation and releases from the IRWST in the longer term. Several sensitivity cases were considered, with the base case using conservative dispersion factors, and other cases presented to show more realistic estimates.

115. It is not in the scope of this assessment to assess Ref. 22 in detail; but I have considered the results. These show that the (pessimistic) base case gives a total offsite dose to the most sensitive group of [redacted] mSv. The sensitivity cases vary between [redacted] and [redacted] mSv. In all cases the results are less than the SAPs Target 8 BSO. Westinghouse note that those accidents that would exceed this have a frequency of 1E-8 per year or lower. While not considered in Ref. 22, it is also possible to use the information provided in response to GI-AP1000-RC-01 to understand the sensitivity of these results to chemistry assumptions. Some of the relevant conclusions that can be drawn from this are:

- Most of the dose is derived from particulate iodine releases. This highlights the significance of the PCS, and the effects it has on removing aerosols, on resultant doses;
- For the Target 8 BSO (1000 mSv) to be exceeded would require around [redacted] g of additional iodine to be released from the IRWST or condensate. This is around five times that estimated by Westinghouse; and
- Changing the iodine speciation, to increase the proportion of organic iodine, has a proportionately large effect. However, this would need to be increased to higher levels than could be reasonably expected to exceed the BSO.

116. I am content that this analysis therefore represents a ‘best estimate’ approach, consistent with the expectations of the SAPs, in employing an appropriate approach within the constraints of the models employed (Ref. 6).

117. In RQ-AP1000-1727 (Ref. 29) I asked Westinghouse whether the results and conclusions from the analysis in response to this GDA Issue needed to be reflected in any other aspects of the safety case, in particular relating to the ADS effects on accident progression. Westinghouse responded by arguing that the actuation of ADS is a benefit in other fault studies citing the examples of LOCAs for determining peak fuel cladding temperatures and containment pressures. However, I was not convinced that this response considered all matters of relevance, in particular relating to the increased source terms inside containment. Westinghouse themselves noted in Ref. 21 under ‘open items’ that ’further evaluation is required to determine the impact for the UK design such as the equipment qualification impact‘. This is because use of the postulated accident source term is not confined to matters such as the designs of engineered safety systems, but may also impact on the post-accident environment for qualification of safety-related equipment, post-accident control room habitability requirements, and post-accident sampling systems and accessibility. I am satisfied that these matters should not undermine the generic design, and are best considered following production of the site-specific analysis. I therefore consider this part of CP-AF-AP1000-RC-01.

4.2.3 Summary

118. In response to GI-AP1000-RC-01 Westinghouse has undertaken an analysis of the **AP1000** plant source term, using the latest versions of industry standard codes and incorporating their latest understanding of chemical behaviour during such events. This considered both the short and long-term phases and the specifics of the **AP1000** design.

119. The analysis documented within Ref. 20 demonstrated that the **AP1000** plant specific severe accident source term is representative, but not bounded by industry representative source terms, in terms of both the timings and magnitude of releases to containment. This is due to a combination of better understanding, and therefore modelling, of such phenomena as well as the features of the **AP1000** design meaning that all LOCAs behave as large hot leg LOCAs. As a result, a dose assessment was performed within Ref. 22, which demonstrated margin for meeting the SAPs Target 8 BSO (Ref. 6).

120. Finally, as discussed in Para. 65, I revisit the question raised earlier regarding the applicability of this work to design basis faults. I am satisfied that, given that the response to RO-AP1000-48 (Ref. 26) shows the analysis performed in support of GDA to be demonstrably conservative and that AF-AP1000-FS-46 has already been raised to perform site-specific analysis (and this will include relevant chemistry effects), this Assessment Finding remains the appropriate way to resolve this matter. There is some similarity in the assumptions used in the bounding design basis analysis for large LOCA, Ref. 27, and those described above. I am content that the responses to GI-AP1000-RC-01 do not highlight shortfalls not previously identified during Step 4, or cast doubt on previously accepted arguments. In effect, I consider it likely that any impact from the lessons learnt from resolving this GDA Issue is bounded by the conservatisms in the existing design basis analysis. Resolution of AF-AP1000-FS-46 should provide the evidence for this.

4.2.4 PCSR Update

121. As noted in Section 2.2, GI-AP1000-CC-02 (Ref. 11) required Westinghouse to submit a consolidated PCSR and associated references to provide the claims, arguments and evidence to substantiate the adequacy of the **AP1000** design reference point. This would therefore include resolution of all 51 GDA Issues. This assessment does not consider the entirety of chemistry within the PCSR, but does judge whether the proposed changes as a result of resolving GI-AP1000-RC-01 are adequate.

122. The modifications to the chemistry chapter of the PCSR (Chapter 21, Ref. 37) were relatively minor, simply adding a brief description and reference to Ref. 20. There are a number of other very minor changes in various other chapters, but these are mainly simple clarifications. Chapter 9, Appendix 9F (Ref. 38) was identified as being impacted; however, this appendix was subsequently removed as part of the PCSR development. It is notable that Chapter 10.12 (Ref. 39), regarding severe accident analysis, does not discuss resolution of this GDA Issue. I also confirmed that these changes were applied in the final consolidated PCSR (Ref. 40).

123. Purely in the context of resolving this GDA Issue, I am therefore content that these changes are reasonable (albeit based on the assumption that the existing information was adequate). However, this does bring about other, wider questions about the coverage of the chemistry aspects of severe accidents within the PCSR. I will consider this as part of GI-AP1000-CC-02 (Ref. 11).

4.3 Comparison with Standards, Guidance and Relevant Good Practice

124. The standards considered as part of my assessment are defined in Section 2.4, and included in Tables 1 and 2.

125. The foremost standards considered for this assessment were the relevant SAPs (Ref. 6). I have considered these throughout my assessment. However, a summary against these is provided below:

- SC.4 and SC.5 relate to the production of an adequate safety case, in particular regarding how this demonstrates that it meets its intended purposes and is clear about its own strengths and weaknesses. I am content that Westinghouse has met the intent of these as part of the submissions provided to resolve this GDA Issue.
- ECH.1 relates specifically to the chemistry aspects of safety cases. In the context of GI-AP1000-RC-01 this relates to the consideration of the chemistry effects that influence the timings and amount of release to the containment. I am content that Westinghouse has considered these matters and have identified the most important in the context of this work. A number of these would benefits from further study by the licensee, but I judge that the approach adopted is sufficient for GDA.
- FA.15 expects appropriate consideration to be given to severe accidents. This is therefore wider than just this GDA Issue, but the source terms is clearly an important part of that in helping to judge if further risk reduction measures are necessary. Importantly, I am content that Westinghouse has used a best estimate approach from a chemistry perspective.
- AV.2 and AV.6 relate to ensuring adequate representation of chemical processes in the analysis and understanding the sensitivity to these phenomena. There is therefore some overlap with ECH.1, and I accept that the most important phenomena have been considered and the sensitivity to these demonstrated.
- NT.1 gives the expectation that safety cases will be compared against the relevant numerical targets. Westinghouse has done this as part of this GDA Issue.

4.4 Assessment Findings

126. In line with the ONR guidance (Ref. 41), during my assessment I have identified one item for a future licensee to take forward in its site-specific safety submissions. Annex 2 contains details of this.

127. This matter does not undermine the generic safety submission and is 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. I have raised this item as an Assessment Finding.

4.5 Minor Shortfalls

128. In line with the ONR guidance (Ref. 41), I have not identified any Minor Shortfalls.

4.6 ONR Assessment Rating

129. Not applicable.

5 CONCLUSIONS

130. This report presents the findings of the assessment of GDA Issue GI-AP1000-RC-01 relating to accident source terms for the **AP1000** reactor.

131. The purpose of this report is to document the assessment of the submissions provided by Westinghouse, in order to come to a judgement as to whether sufficient evidence has been provided to meet the intent of the GDA Issue, such that closure can be recommended.

132. In response to GI-AP1000-RC-01 Westinghouse supplied a single main submission (Ref. 20) which provided a summary of their work to derive and consider a plant-specific source term. This report was supported by the detailed analysis and a subsequent dose assessment that evaluated the impact of this work on the ability of the design to meet the SAPs numerical targets. In addition Westinghouse provided responses to my Regulatory Queries, providing additional clarification and evidence to support the main submission.

133. As a result of my assessment of these submissions, meetings and discussions with Westinghouse, and consultations with ONR colleagues in different technical areas, my conclusions are:

- Westinghouse has undertaken an analysis of the **AP1000** plant source term, using the latest versions of industry standard codes and incorporating its latest understanding of chemical behaviour during such events. This considered both the short and long term phases and the specifics of the AP1000 design.
- The results obtained from this analysis demonstrated that, while the **AP1000** plant specific severe accident source term is representative of, it is not bounded by current industry representative source terms. This is in terms of both the timings and magnitude of releases to containment. This is due to a combination of better understanding, and therefore modelling, of such phenomena as well as the design features of the **AP1000** plant meaning that all LOCAs behave as large hot leg LOCAs.
- Westinghouse has also demonstrated the sensitivity of this analysis to the most important chemistry related assumption and has shown that it is representative of any changes that might be reasonably expected.
- As a result, a dose assessment was performed that demonstrated margin for meeting the SAPs Target 8 BSO.
- In response to this GDA Issue, Westinghouse has proposed updates to the PCSR. I have reviewed these updates and am content that they reflect the responses to the GDA Issue. However, this has highlighted further matters to be considered about the chemistry aspects of accidents covered within the PCSR. This will be considered further as part of GI-AP1000-CC-02.

134. As a consequence of my assessment, I have identified one Assessment Findings for a future licensee to consider and take forward in its site-specific safety submissions. This matter does not undermine the generic safety submission and requires licensee input and/or decisions to resolve.

135. Overall, on the basis of my assessment, I am satisfied that GDA Issue GI-AP1000-RC-01 can be closed.

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Table 1: Relevant Safety Assessment Principles considered during the assessment

SAP No	SAP Title	Description
SC.4	Safety case characteristics	A safety case should be accurate, objective and demonstrably complete for its intended purpose.
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.
ECH.1	Safety cases	Safety cases should, by applying a systematic process, address all chemistry effects important to safety.
FA.15	Scope of severe accident analysis	Fault states, scenarios and sequences beyond the design basis that have the potential to lead to a severe accident should be analysed.
AV.2	Calculation methods	Calculation methods used for the analyses should adequately represent the physical and chemical processes taking place.
AV.6	Sensitivity studies	Studies should be carried out to determine the sensitivity of the analysis (and the conclusions drawn from it) to the assumptions made, the data used and the methods of calculation.
NT.1	Assessment against targets	Safety cases should be assessed against the SAPs numerical targets for normal operational, design basis fault and radiological accident risks to people on and off the site.

Table 2: Relevant Technical Assessment Guides considered during the assessment

Reference	Revision	Title
NS-TAST-GD-005	7	Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable)
NS-TAST-GD-042	3	Validation of Computer Codes and Calculation Methods
NS-TAST-GD-051	4	The Purpose, Scope and Content of Nuclear Safety Cases

Annex 1: GDA Issue, GI-AP1000-RC-02 Revision 0 – Reactor Chemistry – **AP1000®**

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

ACCIDENT SOURCE TERMS

GI-AP1000-RC-01 REVISION 0

Technical Area	**REACTOR CHEMISTRY**		
Related Technical Areas	Fault Studies Probabilistic Safety Assessment Radiation Protection		
GDA Issue Reference	**GI-AP1000-RC-01**	**GDA Issue Action Reference**	**GI-AP1000-RC-01.A1**
GDA Issue	Westinghouse should provide justification to demonstrate that the source term released into the containment during accidents is appropriate for AP1000.		
GDA Issue Action	Westinghouse to demonstrate that the accident source term is applicable to the passive design features of AP1000, and justify the sensitivity of the analyses to more realistic radionuclide behaviour in the containment. Westinghouse should provide analyses, or alternative means agreed by the regulator, to justify the duration and quantity of the release including both the short and long term behaviour. With agreement from the Regulator this action may be completed by alternative means.		

Annex 2: Assessment Findings to be addressed during the Forward Programme – Reactor Chemistry

Assessment Finding Number	Assessment Finding	Report Section Reference
CP-AF-AP1000-RC-01	The licensee shall provide site-specific analysis for the radiological consequence of accidents involving core melting. This should include consideration of the uncertainties in the reactions of iodine and other in-containment phenomena which could affect the releases to the environment. This should include evidence which demonstrates that the results and conclusions have been appropriately reflected into the affected safety case documents.	Para. 104