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NPP Wylfa Newydd (UK)

Expert Statement on the Environmental Impact Assessment

Federal Ministry
Republic of Austria
Sustainability and Tourism

NPP WYLFA NEWYDD (UK)

Expert Statement on the Environmental Impact Assessment

Oda Becker

By Order of the Federal Ministry for Sustainability and Tourism
Directorate I/6 General Coordination of Nuclear Affairs
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CONTENT

Index of Tables and Figures

SUMMARY

Horizon Nuclear Power is proposing to construct and operate a new nuclear power plant (NPP) at the Wylfa Newydd site in Wales at the coast on the Island of Anglesey. The new NPP shall comprise two UK Advanced Boiling Water Reactors (UK ABWR). The expected operation time is 60 years.

The construction and operation of Wylfa Newydd NPP must be authorised by a Development Consent Order (DCO) granted by the relevant Secretary of State. Horizon submitted a DCO application in June 2018. The DCO process requires an Environmental Impact Assessment (EIA), the findings of which must be reported in an Environmental Statement.

The Environmental Impact Assessment (EIA) according to British law (Planning Act 2008, Infrastructure Planning Regulations 2009) and the ESPOO Convention is ongoing. Austria is taking part in this EIA procedure because significant transboundary effects of this project on Austria cannot be excluded.

Description of the project

Regarding the EIA procedure, the British authorities' open and transparent approach to making relevant documents available to the public was appreciated. However, although of particular concern to evaluate the possible risk to Austria site-specific factors that could endanger the safety of the Wylfa Newydd NPP are not discussed appropriately in the Environmental Statement (ES). Site-specific aspects, which are evaluated in the ongoing nuclear site licence (NSL) application should be included in the ES.

In December 2017, the Generic Design Assessment (GDA) for the UK ABWR was completed as the first step of the UK licensing procedure. The UK ABWR reactor design received the Design Acceptance Confirmation (DAC) and the Statement of Design Acceptability (SoDA). However, for the important topics "Severe Accidents" and "Probabilistic Safety Analysis (PSA)", ONR has identified 22 Assessment Findings that are important to safety and still need to be resolved. The Expert Statement describes ONR´s assessments to the extent necessary to evaluate possible severe accidents at Wylfa Newydd NPP which could have significant transboundary effects on Austria.

The GDA documentation prepared by Hitachi-GE sets out the generic safety, environment and security cases for the UK ABWR design. Further development of the design will continue after the GDA, during the site-specific phase. The safety of a site-specific implementation of design modification of nuclear reactor is assessed as part of the review process undertaken prior to granting of the nuclear site licence by the ONR. Horizon submitted its application for a Nuclear Site Licence (NSL) in March 2017.

The Reactor type

The design reference for the UK ABWR will be the standard design of the first ABWR (Kashiwazaki-Kariwa Units 6 & 7 in operation since 1996/97) incorporating further improvements and optimisation from the subsequent ABWR plants and changes made during Generic Design Assessment (GDA).

The ABWR design can be considered as being rather old, the development having started in 1978. In the meantime, the development of the successor model ESBWR (with passive safety features) has been completed. Hitachi-GE has also adapted the outdated ABWR design to the UK safety requirements. To address Fukushima Dai-ichi learning, Hitachi-GE claims that the UK ABWR incorporates a number of enhancements compared to the standard Japanese plant. However, these measures rely more on the use of mobile equipment and other active measures than on the implementation of passive safety systems in the design.

The UK ABWR includes complementary safety features specifically designed to fulfil safety functions required in postulated core melt accidents. Hitachi-GE claimed that challenges to containment integrity are prevented by specifying an appropriate design envelope and by providing severe accident mitigation measures to keep the design envelope from being exceeded.

However, ONR's thorough GDA Step 4 assessment of severe accidents for the UK ABWR revealed that there are several issues which could endanger the containment integrity or lead to a containment bypass. The need for further examination of the capability and the reliability of the severe accident systems and measures was addressed in several assessment findings by ONR. Taking into account all the facts, the safety design and features of the UK ABWR do not guarantee that the radioactive substances will be kept in the containment, neither in the long nor in the short term.

Accident analysis

The consequences of three basis accident scenarios and one severe accident scenario were analysed according to the Environmental Statement.

The approach to calculate the radiological consequences of a possible accident in the Wylfa Newydd NPP is well documented in the Environmental Statement. However, there are no reasons mentioned for the choice of the representative severe accident. This is important because its assumed release for Caesium-137 is relatively small (1.86E+08 Bq). As mentioned above, a core-melt accident with containment failure or by-pass resulting in the release of huge amounts of radioactive material in the environment cannot be excluded for the UK ABWR.

The reference accident scenarios as well as the associated releases are based on probabilistic safety analysis.

In general, probabilistic safety analysis (PSA) results should only be taken as rough indicators of risk. All PSA results are beset with considerable uncertainties, and there are factors contributing to NPP hazards which cannot be included in the PSA. ONR´s review of the PSA for the UK ABWR during the GDA Step 4 came up with a number of shortcomings. Many factors were not included or not addressed appropriately (for example adverse environmental conditions, human failure events (HFEs), specific common cause failures (CCFs), internal and external hazards).

The shortcomings of the outdated design of the UK ABWR are reflected in relatively high values of core damage frequency (CDF) and large release frequency (LRF). To meet the regulation expectations, Hitachi-GE undertook a refinement study of the internal hazard PSA over the course of GDA Step 4, mainly removing conservatisms. In this way, the total large release frequency (LRF) for the UK ABWR was reduced by approximately a factor of four.

However, the PSA results for the UK ABWR showed that the safety assessment principle (SAP) Target 9 risk (= total risk of 100 or more fatalities), summed for all large and large early release categories, is approximately 10^{-6}/year. This value is below basis safety level (BSL), but above the basis safety objective (BSO) of Target 9 (BSL: 10^{-5}/yr; BSO: 10^{-7}/yr).

ONR emphasised that the BSOs are ‘objectives’ and not requirements – the overriding legal requirement for new reactor designs consists in demonstrating that the level of risk is as low as reasonably practicable (ALARP). However, ONR pointed out that Hitachi-GE has not sufficiently demonstrated that the risks for the UK ABWR are ALARP from a PSA point of view. Further work is required after GDA.

The WENRA documents for new reactors are taken into consideration with regard to the safety requirements for new nuclear power plants in the UK. In line with the international guidance, ONR’s safety assessment principles (SAPs) also include an expectation that potential severe accident with large and early releases have been ‘practically eliminated’.

To meet UK and international expectations post-Fukushima, Hitachi-GE has provided a demonstration which argues that the generic UK ABWR design practically eliminates large or early releases. However, Hitachi-GE has neither quantified risks for internal hazard initiators for shutdown and the SFP nor considered the PSA contribution from external hazards when considering practical elimination. The claimed “practical elimination” of a large early release is not sufficiently demonstrated for the UK ABWR to date.

For ensuring compliance with the safety goals of new nuclear power plants consisting in the requirement that accidents leading to early or large releases have to be practically eliminated, a comprehensive Probabilistic Safety Analysis (Extended PSA) would be required, its contents taking into consideration all relevant internal and external events and possible accident causes.

It is important to note that site-specific factors (such as hazards of seismic or tsunamis events, climate change impacts) that could endanger the plant are not discussed appropriately in the Environmental Statement. Loss of the ultimate heat sink (LUHS) due to external hazard (e.g. biological fouling) has the potential of significantly contributing to the UK ABWR overall risk profile. Therefore, it is very important to implement a robust reserve ultimate heat sink (RUHS) for the Wylfa Newydd NPP.

Accidents with third parties involved

Severe third parties’ actions (terrorist attacks and acts of sabotage) can have significant impacts on nuclear facilities, also on the Wylfa Newydd NPP, and cause a severe accident with a major radioactive release.

Although precautions against sabotage and terror attacks cannot be discussed in detail in public in the EIA process for reasons of confidentiality, the EIA documents could have provided more information about the protection against possible terrorist attacks and acts of sabotage. At least the necessary legal requirements should be set out in the EIA documents. Of particular interest is the protection of Wylfa Newydd NPP against a crash of a commercial airplane. Furthermore, the protection of the spent fuel pool against terror attacks is also of particular concern, because the SFP is not situated inside the primary containment.

After assessing the generic Conceptual Security Arrangements (CSA) ONR concluded: from a security view point, the UK ABWR design is suitable for construction in the UK. However, three assessment findings were identified which touch important topics which need to be considered and taken forward in the nuclear site security plan by the future licensee: protection of Vital Areas against sabotage, protection against cyber-attacks and provision of back-up power to the security infrastructure.

The construction of a new NPP cannot be discussed without also taking into consideration a potential terrorist attack on the interim storage for spent fuel. The design of the planned Spent Fuel Storage Facility (SFSF) should meet the state-of-the-art requirements of nuclear security, in particular because its operation time will be 140 years.

Transboundary effects

For the estimation of possible transboundary effects, calculations of the flexRISK project are used. The flexRISK project modelled the geographical distribution of severe accident risk arising from nuclear power plants in Europe. Using source terms and accident frequencies as input, a large-scale dispersion of radionuclides in the atmosphere was simulated. For each reactor, an accident scenario with a large release of nuclear material was selected. For Wylfa-1, a Caesium-137 release of 61.5 PBq is used. This source term is comparable with UK ABWR source terms calculated in its generic Pre-Construction Safety Report (PCSR). According to source terms presented in the PCSR, even much higher releases are possible.

A considerable contamination of the Austrian territory would result from a severe accident at the Wylfa NPP site under weather conditions comparable to those on 25 August 1995. Almost all regions in Austria would receive Caesium-137 ground depositions of more than 1,000 Bq/m², which is beyond the thresholds (650 Bq/m²) that trigger agricultural intervention measures.

The results of the analysis of transboundary effects of a potential severe accident at the Wylfa Newydd site illustrate that, in case of a severe accident at the Wylfa Newydd NPP, an impact on Central European regions (including Austria) cannot be excluded.

Currently, it cannot be proven beyond doubt that a severe accident with major radioactive releases cannot occur at the Wylfa Newydd NPP. Therefore, a conservative worst-case release scenario should have been included in the EIA. A source term for severe accident with containment failure or containment bypass should be analysed as part of the EIA – in particular because of its relevance for significant transboundary effects at greater distances.

ZUSAMMENFASSUNG

Horizon Nuclear Power plant die Errichtung und den Betrieb eines neuen Kernkraftwerks (KKW) am Standort Wylfa Newydd in Wales an der Küste der Insel Anglesey. Das neue KKW soll aus zwei Siedewasserreaktoren vom Typ United Kingdom Advanced Boiling Water Reactor (UK ABWR) bestehen, deren geplante Betriebsdauer 60 Jahre beträgt.

Der Bau und der Betrieb des KKW Wylfa Newydd muss durch die Development Consent Order (DCO) genehmigt werden, die vom zuständigen Staatssekretär erteilt wird. Horizon stellte den Antrag auf die DCO im Juni 2018. Das Verfahren für die DCO erfordert eine Umweltverträglichkeitsprüfung (UVP), deren Ergebnisse in der Umweltverträglichkeitserklärung (UVE) dargestellt werden. Aktuell wird die UVP nach britischem Recht (Planning Act 2008, Infrastructure Planning Regulations 2009) und der ESPOO-Konvention durchgeführt. Die Republik Österreich beteiligt sich an diesem UVP-Verfahren, weil signifikante Auswirkungen des Projekts auf Österreich nicht ausgeschlossen werden können.

Projektbeschreibung

Zur Umweltverträglichkeitsprüfung ist anzumerken, dass das offene und transparente Verfahren der britischen Behörden Dokumente für die Öffentlichkeit verfügbar zu machen, zu begrüßen ist. Allerdings wurden die Standort-spezifischen Faktoren, die die Sicherheit des KKW Wylfa Newydd gefährden könnten, und daher für die Abschätzung möglicher Risiken für Österreich besonders wichtig sind, in der Umweltverträglichkeitserklärung (UVE) nicht ausreichend diskutiert. Standort-spezifische Aspekte, die im derzeitigen Verfahren zur Standortgenehmigung geprüft werden, sollten in der UVE behandelt werden.

Im Dezember 2017 wurde die generische Designbewertung (Generic Design Assessment – GDA) für den UK ABWR als erster Schritt im britischen Genehmigungsverfahren abgeschlossen. Der UK ABWR erhielt als Bestätigung für die Eignung des Designs die „Design Acceptance Confirmation (DAC)“ und das „Statement of Design Acceptability (SoDA)“. Allerdings identifizierte die britische Nuklearaufsichtsbehörde ONR bei der Bewertung der wichtigen Themen „Schwere Unfälle“ und „Probabilistische Sicherheitsanalyse (PSA)“ insgesamt 22 Fragestellungen, die sicherheitsrelevant, aber noch unbeantwortet sind. Diese Fachstellungnahme beschreibt die Bewertung der Aufsichtsbehörde in dem Umfang, der für eine Bewertung möglicher schwerer Unfälle im KKW Wylfa Newydd mit Folgen für Österreich erforderlich ist.

Die von Hitachi-GE vorgelegten GDA-Dokumente beschreiben die generischen Nachweise zur Sicherheit, Umweltverträglichkeit und Sicherung beim Design des UK ABWR. Die Weiterentwicklung des Designs wird nach der GDA in der Standort-spezifischen Phase erfolgen. Die Sicherheit der Standort-spezifischen Designänderungen des Reaktors wird während des Prüfungsverfahrens bewertet, die der Erteilung der Standortgenehmigung durch die Aufsichtsbehörde ONR vorangestellt ist. Horizon hat den Antrag auf Erteilung der Standortgenehmigung im März 2017 gestellt.

Der Reaktortyp

Das Referenzdesign für den UK ABWR wird das Standarddesign des ersten ABWR (Blöcke 6 & 7 des KKW Kashiwazaki-Kariwa, in Betrieb seit 1996/97) sein, bei welchem Verbesserungen und Optimierungen der nachfolgenden ABWR Anlagen sowie Änderungen, die aus der generischen Designbewertung (GDA) resultieren, integriert werden.

Das Design des ABWR ist als relativ alt zu betrachten, da dessen Entwicklung im Jahre 1978 begonnen wurde. Mittlerweile wurde die Entwicklung des Nachfolgemodells ESBWR (mit passiven Sicherheitssystemen) abgeschlossen. Hitachi-GE hat auch das veraltete Design des ABWR an die Sicherheitsanforderungen in Großbritannien angepasst. Hitachi-GE erklärte, dass der UK ABWR im Vergleich zu den japanischen Standardanlagen eine Reihe von Verbesserungen hat, um die Erfahrungen aus Fukushima Dai-ichi zu berücksichtigen. Allerdings basieren diese Maßnahmen mehr auf dem Einsatz von mobilen Geräten und anderen aktiven Maßnahmen als auf der Implementierung von passiven Sicherheitssystemen in das Design.

Der UK ABWR verfügt über zusätzliche Sicherheitseinrichtungen, die speziell zur Erfüllung von Sicherheitsfunktionen entwickelt wurden, die bei potenziellen Kernschmelzunfällen erforderlich sind. Hitachi-GE erklärte, dass die Gefährdung der Containment-Integrität durch die Festlegung geeigneter Auslegungsgrenzen verhindert werde, wie auch durch Maßnahmen für schwere Unfälle, die ein Überschreiten der Auslegungsgrenzen verhindern.

Die sorgfältige Prüfung während Schritt 4 der GDA zu schweren Unfällen des UK ABWR durch die Aufsichtsbehörde ONR zeigte allerdings einige Probleme auf, die die Containment-Integrität gefährden oder zu einem Containment-Bypass führen könnten. Mehrere Bewertungspunkte der ONR verweisen auf die Notwendigkeit die Leistungsfähigkeit und die Zuverlässigkeit der Systeme und Maßnahmen für schwere Unfälle noch genauer zu untersuchen. Insgesamt ist festzustellen, dass die Auslegung und die Sicherheitsvorkehrungen des UK ABWR nicht garantieren können, dass die radioaktiven Stoffe im Containment gehalten werden, weder kurz- noch langfristig.

Unfallanalysen

Die Folgen von drei repräsentativen Auslegungsstörfällen und einem repräsentativen schweren Unfall wurden laut Umweltverträglichkeitserklärung analysiert.

Die Methode zur Berechnung der Strahlenfolgen eines potentiellen Unfalls im KKW Wylfa Newydd ist in der UVE nachvollziehbar dargestellt, allerdings ohne eine Begründung für die Auswahl des repräsentativen schweren Unfalls zu liefern. Das ist von Bedeutung, da die angenommene Freisetzung von Cäsium-137 (1,86E+08 Bq) relativ gering ist. Wie bereits erwähnt, kann ein Kernschmelzunfall mit Containment-Versagen oder Containment-Bypass, der zu einer sehr hohen Freisetzung von radioaktivem Material führen würde, für den UK ABWR nicht ausgeschlossen werden.

Die repräsentativen Unfallszenarien mit den dazugehörigen Freisetzungsmengen beruhen auf probabilistischen Sicherheitsanalysen.

Grundsätzlich sind die Ergebnisse von probabilistischen Sicherheitsanalysen (PSA) nur als grobe Richtwerte für das Risiko zu betrachten. Alle PSA-Ergebnisse sind mit beträchtlichen Unsicherheiten behaftet, da es Faktoren gibt, die zu den Gefährdungen für KKW beitragen, jedoch in einer PSA nicht berücksichtigt werden können. Die Überprüfung der PSA durch die ONR während Schritt 4 der GDA zeigte eine Reihe von Schwachstellen auf. Viele Faktoren wurden nicht oder nicht angemessen behandelt (wie zum Beispiel widrige Umgebungsbedingung, Ereignisse mit menschlichem Versagen, bestimmte Ereignisse mit Versagen aus gemeinsamer Ursache sowie interne und externe Gefahren).

Die Schwachstellen des veralteten UK-ABWR Designs spiegeln sich auch in den relativ hohen Werten für die Kernschmelzhäufigkeit (CDF) und für die Häufigkeit von großen Freisetzungen (LRF) wider. Um den Anforderungen der Nuklearaufsicht zu entsprechen, führte Hitachi-GE während Schritt 4 der GDA eine Präzisierung der PSA zu internen Gefahren durch – dabei wurden vor allem Konservativitäten abgebaut. Dadurch wurde rechnerisch eine Reduktion der Häufigkeit für große Freisetzungen (LRF) des UK ABWR um etwa den Faktor vier erreicht.

Die PSA-Ergebnisse für den UK ABWR zeigen jedoch, dass das Target 9 Risiko (= Gesamtrisiko für 100 und mehr Todesfälle) der Sicherheitsprinzipien als Summe aller großen und frühen Freisetzungskategorien bei ca. 10^{-6}/a liegt. Dieser Wert liegt zwar unter dem Wert für das „Basis Safety Level“ (BSL), jedoch über jenem für das „Basis Safety Objective“ (BSO) von Target 9 (BSL: 10^{-5}/a; BSO: 10^{-7}/a).

ONR betonte, dass es sich bei den Basis Safety Objectives um Ziele und nicht um Anforderungen handelt und die gesetzliche Vorgabe für neue Reaktoren den Nachweis vorsieht, dass das Risikoniveau so gering wie vernünftig machbar („as low as reasonably practicable“ – ALARP) ist. ONR wies jedoch auch darauf hin, dass Hitachi-GE nicht ausreichend nachweisen konnte, dass hinsichtlich der PSA die Risiken des UK ABWR so niedrig wie vernünftig machbar (ALARP) sind. Weitere Nachweise sind nach Abschluss der GDA noch zu erbringen.

Die WENRA-Dokumente für neue Reaktoren werden bei den Sicherheitsanforderungen für neue KKW in Großbritannien berücksichtigt. Gemäß den internationalen Richtlinien sehen auch die Sicherheitsprinzipien der ONR vor, dass potentielle schwere Unfälle mit großen oder frühen Freisetzung „praktisch ausgeschlossen“ sind.

Um die internationalen und die britischen post-Fukushima Anforderungen zu erfüllen, legte Hitachi-GE einen Nachweis darüber vor, dass das generische Design des UK ABWR große oder frühe Freisetzungen ausschließt. Jedoch hat Hitachi-GE dabei weder die Risiken für interne Ereignisse während Stillstandszeiten und für das Lagerbecken für abgebrannte Brennelemente quantifiziert noch die Beiträge aus externen Gefahren in der PSA berücksichtigt. Der behauptete „praktische Ausschluss“ von großen oder frühen Freisetzungen ist zum jetzigen Zeitpunkt somit nicht ausreichend nachgewiesen.

Um die Einhaltung der Sicherheitsziele für neue Reaktoren nachzuweisen (den praktischen Ausschluss von Unfällen mit frühen oder großen Freisetzungen), ist eine umfassende PSA (Extended PSA) erforderlich, die alle relevanten internen und externen Ereignisse und möglichen Unfallabläufe einbezieht.

Es ist anzumerken, dass Standort-spezifische Faktoren (u. a. Gefahren durch seismische Ereignisse, Tsunamis und Folgen des Klimawandels), die das Kernkraftwerk gefährden könnten, in der Umweltverträglichkeitserklärung nicht angemessen dargestellt werden. Der Verlust der primären Wärmesenke aufgrund externer Gefährdung (z. B. durch biologische Verunreinigungen) hat das Potential zum Gesamtrisiko des KKW Wylfa Newydd signifikant beizutragen. Daher ist es von großer Bedeutung, eine robuste alternative Wärmesenke für das KKW Wylfa Newydd zu implementieren.

Unfälle mit Einwirkungen Dritter

Schwere Angriffe Dritter (Terrorattacken und Sabotage) können signifikante Auswirkungen auf Nuklearanlagen haben, so auch auf das KKW Wylfa Newydd, und zu schweren Unfällen mit großen radioaktiven Freisetzungen führen.

Wenn auch die Vorkehrungen gegen Sabotage und Terrorangriffe im UVP-Verfahren aus Geheimhaltungsgründen nicht im Detail öffentlich diskutiert werden können, hätten doch die UVP-Unterlagen mehr Informationen zum Schutz vor möglichen Terrorangriffen und Sabotagehandlungen bieten können. Zumindest die vorgeschriebenen rechtlichen Anforderungen sollten in den UVP-Unterlagen genannt werden. Von besonders großem Interesse ist der Schutz des KKW Wylfa Newydd gegen den Absturz von Verkehrsflugzeugen sowie der Schutz der Lagerbecken für abgebrannte Brennelemente, insbesondere weil diese sich nicht innerhalb des Containments befinden.

Nach der Prüfung des generischen Sicherungskonzepts kam die Aufsichtsbehörde ONR zu folgender Schlussfolgerung: Unter dem Aspekt der Sicherung ist das UK ABWR Design für die Errichtung in Großbritannien geeignet. Allerdings wurden drei wesentliche Bereiche identifiziert, die beim Sicherungskonzept für die Nuklearanlage vom künftigen Lizenzhalter berücksichtigt und weiterentwickelt werden müssen: Schutz der sicherheitsrelevanten Bereiche des KKW vor Sabotage, Schutz vor Cyber-Angriffen und Bereitstellung einer Reservestromversorgung für die Sicherungseinrichtungen.

Die Errichtung eines neuen Kernkraftwerks kann nicht diskutiert werden ohne potentielle Terrorangriffe auf das Zwischenlager für abgebrannten Brennelemente zu betrachten. Die Auslegung des geplanten Zwischenlagers sollte hinsichtlich des Schutzes vor möglichen Einwirkungen Dritter auf dem Stand von Wissenschaft und Technik sein, insbesondere da dessen Betriebsdauer 140 Jahren betragen soll.

Grenzüberschreitende Auswirkungen

Zur Abschätzung der möglichen grenzüberschreitenden Auswirkungen wurden die Berechnungen aus dem flexRISK-Projekt verwendet. Das flexRISK-Projekt bildet die geographische Verteilung der Risiken von schweren Unfällen in Kernkraftwerke in Europa ab. Um eine großräumige Verteilung von Radionukliden in der Atmosphäre zu simulieren, wurden Quellterme und Unfallhäufigkeiten als Eingangsdaten verwendet. Dazu wurde für jeden Reaktor ein Unfall mit einer großen Freisetzung von radioaktiven Stoffen ausgewählt. Für Wylfa-1 wurde eine Cs-137 Freisetzung von 61,5 PBq verwendet. Dieser Quellterm ist vergleichbar mit den Quelltermen für den UK ABWR, die im vorläufigen Sicherheitsbericht berechnet wurden. Danach sind noch wesentlich höhere Freisetzungen möglich.

Wetterbedingungen, die vergleichbar mit denen vom 25. August 1995 sind, würden bei einem schweren Unfall im KKW Wylfa Newydd zu einer beträchtlichen Kontamination von österreichischem Staatsgebiet führen. Nahezu alle Regionen in Österreich würden Cs-137 Bodenkontaminationen über 1000 Bq/m^2 aufweisen und somit über dem Eingreifrichtwert (650 Bq/m^2) für landwirtschaftliche Maßnahmen liegen.

Die Ergebnisse der Analyse grenzüberschreitender Auswirkungen von potentiellen schweren Unfällen am KKW Standort Wylfa zeigen, dass bei einem schweren Unfall im KKW Wylfa Newydd Auswirkungen auf Regionen in Mitteleuropa (einschließlich Österreich) nicht ausgeschlossen werden können.

Zum jetzigen Zeitpunkt kann nicht zweifelsfrei nachgewiesen werden, dass kein schwerer Unfall mit großen radioaktiven Freisetzungen im KKW Wylfa Newydd auftreten kann. Daher hätte ein konservatives Worst-case Szenario in das UVP-Verfahren aufgenommen werden sollen. Ein schwerer Unfall mit Containment-Versagen oder Containment-Bypass sollte als Teil der UVP analysiert werden – insbesondere aufgrund seiner Bedeutung für Auswirkungen in großen Entfernungen.

1 INTRODUCTION

Horizon Nuclear Power is proposing to construct and operate a new nuclear power plant (NPP) at the Wylfa Newydd site in Wales at the coast on the Island of Anglesey. The new NPP shall comprise two UK Advanced Boiling Water Reactors (UK ABWR). The site already hosts two old Magnox NPPs (Wylfa-1 and 2) that were shut-down in 2012 and 2015.

A nuclear power station is a Nationally Significant Infrastructure Project (NSIP) under the Planning Act 2008 and its construction and operation must be authorised by a Development Consent Order (DCO) granted by the relevant Secretary of State. The DCO process is managed by the Planning Inspectorate. Horizon Nuclear Power Wylfa Limited submitted a Development Consent Order (DCO) application in June 2018.

For this project, an Environmental Impact Assessment (EIA) according to British law (Planning Act 2008, Infrastructure Planning (EIA) Regulations 2009) and the ESPOO Convention is ongoing. Austria is taking part in this EIA procedure because significant transboundary effects on Austria cannot be excluded.

The Environment Agency Austria (Umweltbundesamt) was commissioned by the Austrian Federal Ministry for Sustainability and Tourism to coordinate this expert statement and assist in organizational matters. The Environmental Agency Austrian has assigned Oda Becker, scientific consultant, to elaborate an expert statement on the documents presented by the UK.

The goal of the expert statement at hand is to assess if the EIA documents allow for making reliable conclusions about the potential effects of transboundary emissions. Therefore, this paper assessed the project's safety features and the accident analysis with a focus on air-borne transboundary emissions and the potential effects on Austria.

2 DESCRIPTION OF THE PROJECT

2.1 Treatment in the EIA documents

Horizon Nuclear Power Wylfa Limited, a UK energy company, is planning to construct and operate a Nuclear Power Plant (NPP) on the island of Anglesey, Wales. It will be located to the west of the village of Cemaes and to the south of the existing Magnox power station. The NPP will consist of two UK ABWRs and generate 2.7 gigawatts of electricity. (HNP 2018c, p. 3)

Horizon Nuclear Power Wylfa Limited will develop the Project using technology purchased from HGNE, a joint venture between Hitachi Limited and General Electric Corporation. (HNP 2018c, p. 2)

Land adjacent to the old Wylfa NPPs is identified by the UK Government in the Overarching National Policy Statement for Energy (EN-1) and National Policy Statement for Nuclear Power Generation (EN-6) as potentially suitable for the construction of a new nuclear power station. Horizon Nuclear Power Wylfa Limited proposes to construct and operate a new nuclear power station, known as Wylfa Newydd, on this land and adjacent land, referred to as the Wylfa Newydd Development Area (WNDA). (HNP 2018f, p. 2)

Principal construction activities will start after the major permissions required to build the NPP have been granted. Once construction of the first reactor has reached an advanced stage, it will be commissioned (expected to last two years) to ensure all systems and processes operate as intended. The first reactor will then become operational. This will be followed by the second reactor approximately sixteen months later. The expected operation time of each reactor is 60 years. (HNP 2018c, p. 3)

The sea provides the ultimate heat sink (UHS) for the Wylfa Newydd NPP. Both UK ABWR units draw their cooling water requirements from a single intake structure located at Porth-y-pistyll and share a common cooling water outfall structure in Porth Wnal. (HNP 2018c, p. 17)

Development Consent Order (DCO)

A nuclear power station is a Nationally Significant Infrastructure Project (NSIP) under the Planning Act 2008 and its construction and operation must be authorised by a Development Consent Order (DCO) granted by the relevant Secretary of State. (HNP 2018c, p. 4) Horizon Nuclear Power Wylfa Limited submitted a Development Consent Order (DCO) application and also an application for a Marine License in June 2018.

The Planning Inspectorate is responsible for examining the application and making a recommendation to the Secretary of State as to whether development consent for the Wylfa Newydd Project should be granted. Following submission, the Planning Inspectorate will determine whether to formally accept the application. If accepted, the application will then enter the pre-examination phase where interested parties will be asked to register their interest in the application and make representations.

At the conclusion of the pre-examination phase, the Examining Authority will hold a preliminary meeting to set the timetable for the examination. The 2008 Act requires the examination of the application to be completed within six months and the Planning Inspectorate then has three months from the end of the examination to provide its report and recommendation to the Secretary of State, who then has to decide within three months time. (HNP 2018d, p. 105)

Environmental Impact Assessment (EIA)

The DCO process requires an Environmental Impact Assessment (EIA) of the NSIP, the findings of which must be reported in an Environmental Statement. (HNP 2018f, p. 2)

The scope of the EIA has been informed by a Scoping Opinion provided by the Planning Inspectorate. The EIA Infrastructure Planning Regulations 2017 came into force in May 2017. However, the EIA Infrastructure Planning Regulations 2009 still require an application for development consent where the Secretary of State has been requested to adopt a Scoping Opinion prior to this date. The Scoping Opinion was requested before May 2017 and the Wylfa Newydd Project is therefore subject to the EIA Infrastructure Regulations 2009. Nevertheless, the EIA has been conducted taking into account the additional provisions of the EIA Infrastructure Regulations 2017. (HNP 2018f, p. 3/4)

Environmental Statement (ES)

The ES provides a description of the likely significant effects on the environment arising from the Wylfa Newydd Project. It explains the processes followed, the assessment methods used and the mitigation measures proposed to prevent, reduce and offset any significant adverse effects.

Climate change has been considered within this ES with regard to design resilience and the effects of climate change on the project; consideration for how the EIA takes account of climate change and 'future baseline' when assessing effects caused by the development. (HNP 2018g, p. 24)

Due to the coastal location of the Wylfa Newydd Power Station, sea-level rise and coastal erosion are key concerns. Coastal erosion is considered in the coastal processes and geomorphology topic in this Environmental Statement, with erosion rates given as up to 0.2m per year, whilst sea level rise projections are taken from UKCP (2009) with a projected sea level rise of 488mm by 2090; this rise is not expected to affect the Wylfa Newydd Project directly during its lifetime. (HNP 2018g, p. 24)

Nuclear Site Licence (NSL)

A Nuclear Site Licence (NSL) will be required under the Nuclear Installations Act 1965, as amended to install and operate the NPP. The NSL places Horizon under ONR regulation where it will oversee the licensee's control of the safety of the NPP. This includes activities related to design, construction, installation, commissioning, operation, maintenance, modifications and decommissioning, including the accumulation or storage of radioactive waste. Horizon submitted its application for the NSL in March 2017. (HNP 2018c, p. 6)

Generic Design Assessment (GDA)

GDA is the process by which the nuclear regulators – the ONR and the Environment Agency (EA) – first assess the safety, security and environmental implications of new nuclear reactor designs (without reference to site-specific issues). (HNP 2018c, p. 5)

The GDA documentation prepared by HGNE sets out the generic safety, environment and security cases for the UK ABWR design. The main submissions are the Generic Pre-Construction Safety Report (PCSR), the Generic Environmental Permit Application (GEP) and the Conceptual Security Arrangements (CSA). These submissions are underpinned by relevant detailed reference documents. The PCSR sets out the demonstration that the design meets UK safety requirements and that the risks associated with the design are As Low As Reasonably Practicable (ALARP).

In December 2017, ONR, the Environment Agency and Natural Resources Wales granted Design Acceptance Confirmation (DAC) and a Statement of Design Acceptability (SoDA) for the UK ABWR reactor design. The safety of a site-specific implementation of that design of nuclear reactor is assessed as part of the review process undertaken prior to granting of the nuclear site licence by the ONR.

EU Legislation

On 23rd June 2016 the United Kingdom (UK) public voted in a referendum to leave the European Union, and the UK Government has since confirmed that it intends to negotiate the UK's exit under Article 50 of the Lisbon Treaty. However, exit is unlikely to take place before early 2019, and the status of the UK's legal framework after the exit remains unclear. Therefore, for the purposes of the application, it has been assumed that the current relevant legislative requirements will remain in force for the foreseeable future. (HNP 2018c, p. 4)

2.2 Discussion

Generic Design Assessment (GDA)

Hitachi-GE commenced GDA in 2013 and completed Step 4 in December 2017. The GDA of the UK ABWR has followed a step-wise approach in a claims-arguments-evidence hierarchy which commenced in 2013. Major technical interactions started in Step 2 with an examination of the main claims made by Hitachi-GE for the UK ABWR. In Step 3, the arguments which underpin those claims were examined. The objective of the Step 4 assessments is to undertake an in-depth assessment of the safety, security and environmental evidence. (ONR 2017a, p. 9)

Findings that were identified during the regulators' GDA assessment are important to safety are referred to as Assessment Findings (AF). After GDA, the Assessment Findings will be subject to appropriate control as part of normal regulatory oversight. Further development of the design will be progressed after the GDA, during the site-specific phase. The Generic Design Assessment (the first step of the UK licensing procedure) for the UK ABWR has been completed in December 2017. Therefore, the reactor type as such is determined as suitable in UK, irrespective of the site.

However, ONR has identified several Assessment Findings that are important to safety and still need to be resolved. For the important topics “Severe Accidents” and “Probabilistic Safety Analysis (PSA)”, ONR has identified 22 Assessment Findings.

The following chapters are devoted to ONR´s assessments including the Assessment Findings (AF) to evaluate the possibility of severe accidents at the Wylfa Newydd NPP which could have significant transboundary effects on Austria.

Scope of provided documents

Regarding the procedure, the British authorities’ open and transparent approach to making relevant documents available to the public was appreciated.

However, site-specific factors (like hazard of seismic or tsunamis events, influence of the climate change) that could endanger the safety of the Wylfa Newydd NPP are not discussed appropriately in the Environmental Statement.

2.3 Conclusions, questions and recommendations

In December 2017, the Generic Design Assessment (GDA) for the UK ABWR was completed as the first step of the UK licensing procedure. The UK ABWR reactor design received the Design Acceptance Confirmation (DAC) and the Statement of Design Acceptability (SoDA). However, for the important topics “Severe Accidents” and “Probabilistic Safety Analysis (PSA)”, ONR has identified 22 Assessment Findings that are important to safety and still need to be resolved. In the next chapters, ONR´s assessments are described to the extent necessary to evaluate possible severe accidents at Wylfa Newydd NPP which could have significant transboundary effects on Austria.

By having passed the GDA, the UK ABWR reactor design was found suitable for the UK. The safety of a site-specific implementation of that design of nuclear reactor is assessed as part of the review process undertaken prior to granting of the nuclear site licence (NSL) for Wylfa Newydd NPP.

Regarding the EIA procedure, the British authorities’ open and transparent approach to making relevant documents available to the public was appreciated. However, the Environmental Statement (ES) did not appropriately discuss site-specific factors, although they endanger the safety of the Wylfa Newydd NPP and are of particular concern when evaluating the possible risks for Austria.

Site-specific aspects, which are being evaluated during the ongoing nuclear site licence (NSL) application should be included in the ES.

Question

- In which way will the solutions to the GDA assessments findings be published?

(Preliminary) Recommendations

- Site-specific aspects, which are being evaluated during the ongoing nuclear site licence (NSL) application, should be included in the EIA documents. Site-specific factors that could endanger the safety of the Wylfa Newydd NPP are of particular concern when evaluating the possible risks for Austria.
- It is recommended to inform about the solutions of assessments findings in an appropriate manner.

3 REACTOR TYPE

3.1 Treatment in the EIA documents

The Power Station will consist of two UK ABWR reactors. Chapter 2.3 of the EPA describes the Development of the Reference Design. (HNP 2018c, p. 20 ff.)

The ABWR was developed primarily in Japan and the USA and was based on an evolution of conventional Boiling Water Reactor (BWR) technology. The development was started in 1978 by Japanese electric utilities and plant manufacturers, including Hitachi Limited in Japan and General Electric Company in the US, in collaboration with various international partners.

Hitachi-GE Nuclear Energy Limited (HGNE) has completed the design and construction scope of four ABWR units which have been operational in Japan. The units are:

- Units 6 and 7 of Kashiwazaki-Kariwa NPP of TEPCO (commenced commercial operation in 1996 and 1997 respectively),
- Unit 5 of Hamaoka NPP of Chubu Electric Power Co (commenced commercial operation in 2005) and
- Unit 2 of Shika NPP of Hokuriku Electric Power Company (commenced commercial operation in 2006).

HGNE is also involved in the on-going construction of the Shimane 3 and Ohma ABWRs in Japan. The UK ABWR derives from the design of the ABWR. The design reference for the UK ABWR will be the standard design of the first ABWR (Kashiwazaki-Kariwa Units 6 & 7) incorporating further improvements and optimisation from the subsequent ABWR plants and changes made during Generic Design Assessment (GDA).

Reference Design for Wylfa Newydd

The principal aspects of the Wylfa Newydd NPP design which differ from the UK ABWR design assessed as part of GDA are:

- The Power Station comprises two reactors where only a single reactor design was assessed at GDA,
- The locations of the cooling water intake and outfall have been established. (HNP 2018c, p. 21)

Safety features of the design

Appendix D14-2 (Analysis of accidental releases) provides a very general description of the safety features of the design. (HNP 2018a)

Engineered safety systems comprise the reactor containment systems and the Emergency Core Cooling System (ECCS). These are provided in order to prevent fuel damage or the potential discharge of large amounts of radioactive substances, in the unlikely event of failure or damage to structures, systems and components (SSCs) of the reactor installation. (HNP 2018a, p. 7)

The engineered safety systems are the principal means of delivering the key safety functions of containment and long-term heat removal. The containment systems are provided in order to:

- minimise the release of radioactive materials to the environment (the primary containment vessel (PCV) and reactor building) and
- ensure the integrity of the primary and secondary containment structures is maintained.

Individual systems which are included in the ECCS and the containment systems are listed. (HNP 2018a, p. 7)

Emergency Core Cooling System

The ECCS is provided to maintain cooling to the reactor and prevent exceeding fuel temperature limits in the event of faults, which could result in fuel damage. The ECCS provides the principal means of core heat removal and long-term cooling in fault scenarios.

The ECCS configuration comprises three redundant divisions provided with high-pressure and low-pressure water injection systems, which are powered from the respective divisions of the redundant emergency diesel generator systems, in the event of loss of off-site power (LOOP). The ECCS injection network is comprised of one reactor core isolation cooling system train and two high-pressure core flooder trains for high-pressure injection, and three low-pressure flooder system trains for low-pressure injection in conjunction with the automatic depressurisation system which assists the injection network under certain conditions. (HNP 2018a, p. 7)

Primary containment vessel (PCV)

The PCV is a reinforced-concrete structure with an internal steel liner. It consists of components such as a cylindrical drywell surrounding the RPV, a cylindrical suppression chamber and a basemat. In the event of a loss of coolant accident, the steam water mixture released into the drywell is fed into the suppression pool water through the vent pipes. The steam is cooled and condensed by this pool water, thus suppressing the pressure rise in the drywell. Any radioactive substances are retained inside the containment vessel. (HNP 2018a, p. 7)

Containment heat removal system

The principal role of the containment heat removal system is to prevent excessive containment temperatures and pressure, thus maintaining containment integrity in the long term following a design basis event or a beyond design basis event including severe accidents. (HNP 2018a, p. 8)

Secondary containment facility/reactor building

The secondary containment boundary completely surrounds the PCV except for the basemat, and together with the clean zone, comprises the reactor building. The secondary containment encloses all penetrations through the PCV and all those systems external to the PCV that may become a potential source of radioactive release after an accident. (HNP 2018a, p. 8)

Severe accident management systems

The severe accident management systems provide backup safety facilities, separate from the engineered safety features, to deliver safety functions in the event of beyond design basis events that potentially lead to multiple losses of safety facilities. The backup safety facilities are designed to deliver the following safety functions.

- Provide cooling water to the reactor core in order to prevent reactor core damage and to maintain reactor core cooling in case of station blackout and/or loss of all function of digital control and instrumentation equipment.
- Supply water to the PCV spray header, directly cooling the upper drywell atmosphere and scrubbing airborne fission products.
- Provide water to the lower drywell under the severe accident condition of reactor pressure vessel (RPV) failure to remove decay heat from molten core.
- Provide water to the reactor well to prevent PCV flange failure due to excess temperature.
- Provide makeup water to the spent fuel storage pool to remove decay heat and to maintain the pool water level.
- Provide a filtered vent to prevent damage of the PCV due to overpressure in the event of a severe accident. (HNP 2018a, p. 9/10)

Emergency generators

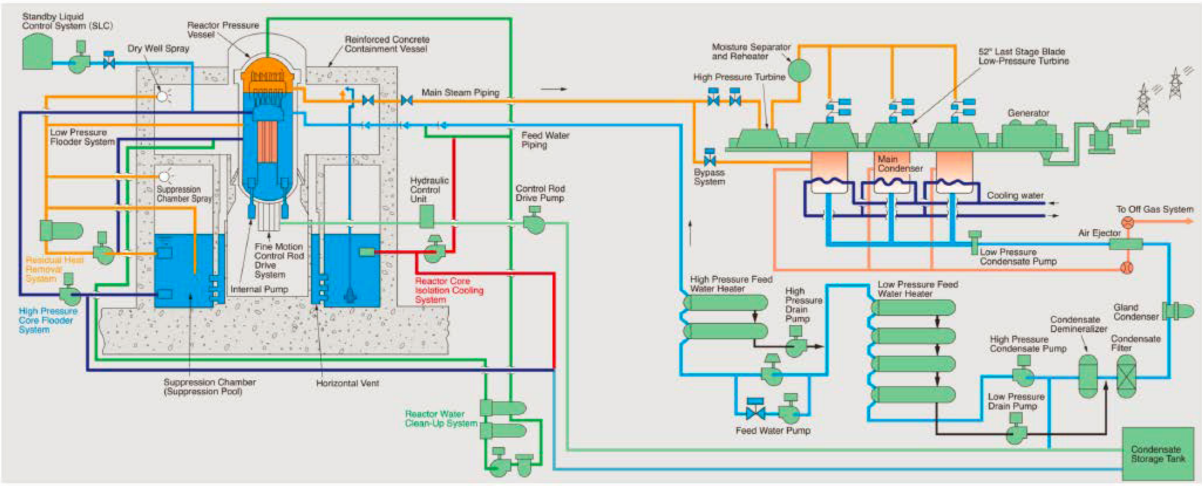
Standby alternating current power generation would provide power to the Power Station safety systems that would be required to shut down and cool the reactor in the event of a LOOP. As a generic design, the UK ABWR is designed to be kept in a stable state by utilising on-site provisions for seven days and DC battery can supply power to site for at least 24 hours.

The role of the emergency diesel generators is to supply the power needed to shut down the reactor safely when off-site power is lost, and to supply power to the electrical systems supporting the delivery of safety functions if a Loss of Coolant Accident (LOCA) occurs simultaneously with a LOOP. The emergency diesel generators are fully independent of each other and are each housed, together with their related ancillary plant, within separate buildings.

The backup building will provide alternative safety management capacity during an emergency if the main control building and associated safety systems are not operational. Two backup building generators and associated equipment would service each generating unit, and would be installed in a single backup building. (HNP 2018a, p. 10)

The following figure shows a schematic diagram of the configuration and the main systems of the UK ABWR. (HNP 2018a, p. 6)

Configuration and main systems of UK ABWR



Quelle: HNP 2018a, p. 6   umweltbundesamt

Figure 1: Schematic diagram of the configuration and the main systems of the UK ABWR

Spent Fuel Storage Facility (SFSF)

Spent fuel will be stored in the spent fuel pools (SFPs) for a period of up to 10 years. After this period, it will be transferred to the SFSF for storage for up to 140 years prior to disposal to the Geological Disposal Facility (GDF). There will be one shared Spent Fuel Storage Facility (SFSF) for the two generating units. The design of the facility is in development and yet to be confirmed. However, it will be designed to accommodate the lifetime arising of spent fuel that will be generated and will be located in the south west corner of the site. (HNP 2018c, p. 19/20)

3.2 Discussion

The Advanced Boiling Water Reactor (ABWR) was one of the first reactors being mentioned as a new-generation reactor. The ABWR is derived from a General Electric (GE) design in collaboration with Toshiba. Two reactors built by Hitachi and two by Toshiba have been in commercial operation in Japan.

In January 1997, the ABWR was among the first reactor designs in the United States to receive the final design certification from the US Nuclear Regulatory Commission (NRC). The initial certification was valid for 15 years, and in 2011 the NRC certified for GE Hitachi an evolved version which allows for aircraft impacts. (WNA 2018) Both Toshiba and GE Hitachi have applied separately to the NRC for design certification renewal. Japan's Toshiba Corporation has withdrawn its application to the US Nuclear Regulatory Commission (NRC) to renew the design certification for its ABWR in 2016. (WNN 2016)

The ABWR has been offered as several slightly different versions by GE Hitachi, Hitachi-GE and Toshiba, so that 'ABWR' is now used as a generic term. It is basically a 1,380 MWe (gross) unit, though GE Hitachi referred to it as 1,350-1,600 MWe net. Toshiba highlights the development of its 1,400 MWe class to a 1,500-1,600 MWe class unit. Tepco was funding the design of a next generation BWR, and the ABWR-II is quoted as 1,717 MWe. Toshiba was promoting its EU-ABWR of 1,600 MWe developed with Westinghouse Sweden. (WNA 2018)

The ABWR reactor model can be considered as rather old, as the development started in 1978. In the meantime, the development of the successor model ESBWR (Economic Simplified Boiling Water Reactor, Generation III+ BWR) has been completed. (SHOLLY 2014) The **ESBWR** (GE Hitachi) is an improved design "evolved from the ABWR" but utilizes passive safety features including natural circulation principles. (WNA 2018)

The experience with operation of the ABWR has been poor: A 6.6 magnitude earthquake at Chuetsu-Oki in 2007 led to a two-year closure of Kashiwazaki-Kariwa 6 and 7, significant upgrades were required before the reactors could be restarted. As a result of the 2007 earthquake, the Hamaoka units were re-assessed and upgraded, the ABWR standing still for over one year. Shika-2 was closed from late 2006 until May 2008 due to a steam turbine failure. (THOMAS 2018)

In Japan, the construction of two more ABWRs had started before the Fukushima accident happened; construction was suspended. The Japanese utility Chugoku announced in February 2018 that it would seek to start up unit Shimane 3. Chugoku is the second Japanese utility to apply to the Nuclear Regulation Authority (NRA) for pre-operation safety inspections for a new NPP since the Fukushima Daiichi accident. The first was Japan Electric Power Development Corp (J-Power), which applied in December 2014 for inspections of unit 1 at its Ohma NPP, also an ABWR. (WNN 2018)

Other proposed ABWRs in Japan have been deferred or suspended. The start-up of two ABWRs (construction start 1999) at Lungmen near Taipei (Taiwan) has been delayed among other reasons also for safety concerns. (BECKER 2013)

3.2.1 Discussion of the safety systems and measures

In this chapter, the safety systems and concept of the UK ABWR are discussed. For this purpose, ONR's review of the GDA "Step 4 Assessment of Severe Accidents for the UK Advanced Boiling Water Reactor" is used. (ONR 2017a)

Emergency core cooling

The emergency core cooling system (ECCS) provides the primary means for fuel cooling for design basis faults. The ECCS consists of three independent divisions, each with functions for high pressure and low-pressure water injection into the RPV in the event of a reactor fault. For both of Divisions II and III of the ECCS high pressure injection is provided by the high-pressure core flooder (HPCF) system. In Division I of the ECCS, the high-pressure water injection function is provided by the reactor core isolation cooling system (RCIC). For each of the three divisions, low pressure injection is provided by the low-pressure flooder system (LPFL). Each division of the ECCS can be powered by one EDG.

Two trains of the ECCS can also be set up for S/P cooling during faults. Coolant for injection is drawn from either the condensate storage tank, external to the R/B, or the S/P.

The RCIC uses a turbine-integrated pump driven by decay heat steam to inject water into the RPV and maintain water level. The RCIC operates automatically to maintain RPV water level and requires only battery power (which lasts for up to 24 hours). Exhaust steam from the RCIC is condensed in the S/P, leading to a rise in PCV pressure and temperature if RHR functions are unavailable for S/P cooling in a beyond design basis event. (ONR 2017a, p. 19/20)

Severe accident in-vessel cooling

A severe accident giving rise to significant core damage usually results from the failure or degradation of the ECCS functions. In this case, an alternative active low-pressure injection system, the flooder system of specific safety facility (FLSS), is provided to prevent and/or mitigate fuel damage. The FLSS consists of two trains of two pumps with a dedicated water source, individual piping and the necessary valves. The FLSS is also designed to provide cooling water to the SFP. The FLSS pumps are located in the B/B and can be operated from either the MCR or the B/B. On-site water storage and fuel supplies are provided for seven days operation without external supplies.

The flooder system of reactor building (FLSR) is a mobile system which replicates the FLSS injection functions. It uses a mobile pump and power truck which would be normally stored on-site and has specific connections to FLSS injection piping. In a severe accident, the FLSR could potentially be deployed in about 8 hours and, if necessary, could provide cooling following termination of the RCIC, or otherwise if the FLSS was unavailable.

In the absence of in-vessel cooling, the UK ABWR does not include any design provision for in-vessel retention (IVR) of core debris. Instead, the strategy for the UK ABWR is to manage core debris in the Lower Drywell (LDW) of the containment. Operators would attempt to manually depressurise the reactor before RPV failure using the Automatic Depressurisation System (ADS) or Remote Depressurisation Control Facility (RDCF). The objective is to avoid high-pressure melt ejection (HPME) and therefore mitigate challenges from direct containment heating (DCH) and rapid steam generation from fuel-coolant interaction (FCI). (ONR 2017a, p. 20/21)

Severe accident ex-vessel cooling

The concrete floor of the Lower Drywell (LDW) is designed as a spreading area for corium released from the RPV. The size of the spreading area should be sufficient to allow the corium to be cooled by overlying water, thereby minimising molten core concrete interactions (MCCI). The LDW floor includes a concrete layer of 1 metre thickness, constructed of basaltic concrete to minimise generation of non-condensable gases formed by MCCI. This is designed to prevent contact of corium with the PCV liner. The severe accident strategy for the UK ABWR is to pre-flood the LDW if RPV failure is considered likely. This is intended to be achieved by manual activation of water injection into the LDW using the FLSS or FLSR. If injection is unavailable, then the Lower Drywell Flooder (LDF), comprising ten fusible (thermally actuated) plug valves, activate passively to flood the LDW with water from the S/P. The design is based on the idea that the containment structures and components should withstand the effects of steam explosions associated with ex-vessel FCI. (ONR 2017a, p. 21)

Containment heat removal

In a severe accident, heat would accumulate in the suppression pool (S/P) by release of steam through the SRVs. Alternatively, if the RPV failed then steam would be generated in the LDW from water overlying the corium; this would also result in heat-up of the S/P due to transfer of steam through the vent pipes. The residual heat removal (RHR) functions of the ECCS include the S/P cooling mode which is able to provide long-term heat removal from the containment in severe accidents. Heat from the RHR is rejected to the closed loop reactor building cooling water (RCW) system which itself rejects heat to the reactor building service water (RSW) system in the heat exchanger building. The RSW takes its water from a water intake pit. A conceptual design for a reserve ultimate heat sink (RUHS) has also been proposed for GDA.

If the RHR function is unavailable, a severe accident will result in an increase in the energy stored in the PCV. In this case, containment heat removal and pressure control is achieved through venting of steam and gases from the PCV to atmosphere through the filtered containment venting system (FCVS). The preferred route is to vent from the wetwell (WW) as this has the benefit of fission product scrubbing by the S/P. The FCVS incorporates filters for the further removal of elemental iodine and particulates prior to discharge to atmosphere through the stack. A hardened, unfiltered venting route is also available if necessary, although venting through the FCVS would be the preferred option.

The FCVS also incorporates a containment overpressure protection system (COPS) designed to ensure that pressure is relieved from the PCV before containment integrity is challenged. The COPS is a passive system which uses bursting disks to release steam and gases from the WW through the FCVS.

Assessment

ONR highlighted that the engineering requirements for severe accident design provisions are insufficient.

The engineering requirements for severe accident design provisions are less well developed than for the design basis. According to ONR, the position is sufficient for GDA, however, to ensure that this is addressed by the future licensee ONR raises the Assessment Finding AF-ABWR-SA-11. (ONR 2017a, p. 81/82)

ONR points out:

- Systems, Structures and Components (SSCs) such as the RDCF, FLSS and FCVS are part of Hitachi-GE’s safety case for design basis, beyond design basis and severe accidents, but the focus of the engineering documentation is principally on design basis requirements.
- Beyond design basis hazards withstand claims for severe accident mechanical systems are generally not considered in the engineering documentation, even though these may be required to operate in a severe accident initiated by a beyond design basis hazard.

- Severe accident withstand claims for some mechanical systems or components are not reflected in the engineering submissions.
- Claims for systems identified in the severe accident safety case as providing defence in depth are generally not reflected in the engineering documentation.
- Severe accident claims for the primary containment function are not clearly identified, although the important elements of a safety case have been provided.

3.2.2 Containment performance

Hitachi-GE's severe accident concept consists in preventing the plant from experiencing primary containment failure. Hitachi-GE claims that challenges to containment integrity are prevented by specifying an appropriate design envelope and by providing severe accident mitigation measures to keep the design envelope from being exceeded. Hitachi-GE further claims that:

- Failures due to over-pressure and over-temperature are prevented by appropriate design of the PCV boundary and provision of measures that are designed to ensure that:
  - conditions in the PCV are maintained below failure criteria by features such as the LDF, COPS and PCV sprays to control temperature and pressure below failure criteria and
  - PCV failures due to DCH and rapid pressurisation due to ex-vessel FCI are prevented by ensuring that the RPV can be depressurised to below 2 MPa before RPV failure.
- Hydrogen concentrations can be maintained below flammable limits and, therefore, there would be no challenges to the containment from hydrogen combustion.
- The pedestal wall will withstand pressure waves should an ex-vessel FCI steam explosion occur, thus preventing containment failure.
- Concrete ablation due to MCCI is limited by flooding of LDW such that collapse of the pedestal wall, leading to gross containment failure, does not occur.
- Corium does not come into contact with the PCV liner due to the layer of concrete in the base of the LDW and the concrete pedestal wall.

Assessment

The review of ONR revealed that there are several issues which are not solved yet and could endanger the containment integrity or lead to a containment bypass.

ONR notes that with the PSA Hitachi-GE has identified accident sequences where the containment could potentially fail, for example due to multiple failures of design basis protection systems or due to failures of severe accident measures. The ONR's assessment of the primary containment vessel (PCV) failure focuses on the success criteria used to show that severe accident measures, when available, are effective in preventing containment failure. (ONR 2017a, p. 39)

Ablation of the pedestal wall: Corium spreading on the LDW floor would initially impinge the inner steel plate of the pedestal wall. After the inner steel plate has failed, the bulk concrete of the pedestal wall is ablated. Hitachi-GE assumes that the load-bearing function of the pedestal wall is maintained by the outer steel plate until ablation has progressed radially through 90% of the thickness of the concrete. At this point, the plate will lose its load-bearing capacity, resulting in collapse of the pedestal wall and failure of the containment due to loss of support to the RPV. According to ONR, Hitachi-GE submissions do not provide evidence to justify why integrity is maintained up to this point. The assumed failure point of the pedestal wall (presumably reached when ablation has progressed through 90% of the wall thickness) has not been adequately justified in GDA. (see Assessment Finding AF-ABWR-SA-01)[1] (ONR 2017a, p. 43)

Suppression pool bypass: An important feature of the UK ABWR PCV is the suppression pool (S/P) which is designed to provide pressure suppression by condensing steam generated during accidents. The S/P also provides scrubbing of steam and gases and acts as an additional barrier for releases from the containment vent. **Pressure suppression and scrubbing would be impaired if a bypass of the S/P occurred and steam passed directly into the WW gas space**. Hitachi-GE's safety case does not include a S/P bypass. The tightness of vacuum breakers (V/Bs) under severe accident conditions is important to prevent a suppression pool bypass. Hitachi-GE has not presented severe accident claims for the V/Bs in its safety case. (see Assessment Finding AF-ABWR-SA-02). (ONR 2017a, p. 45)

RPV re-flooding: Hitachi-GE does not adequately demonstrate that the range of conditions leading to re-criticality (which could endanger the integrity of the containment) has been identified. (see Assessment Finding AF-ABWR-SA-03)

Containment pressure suppression and control

The UK ABWR is provided with containment over-pressure protection (COPS) which is designed to relieve pressure before the ultimate failure point is reached. This is achieved by manually venting the containment before the pressure reaches 2×Pd, or through the COPS passive bursting disks when the WW pressure reaches 2×Pd. In both cases, filtered venting from the WW via the FCVS is the preferred route. There is also the possibility to vent using the unfiltered hardened vent system, but this is not Hitachi-GE's preferred strategy.

Assessment

For slow pressure transients, Hitachi-GE's analysis shows that the vent system, once opened, is effective in limiting the pressure in the WW to below the lower-bound failure criterion of 2×Pd. However, due to differences in pressure between the WW and drywell (DW), pressure in the DW could exceed the COPS setting of 2×Pd before the bursting disks fail. Thus, the pressure in the DW could reach the assumed ultimate failure pressure before venting occurs. This possibility has not been addressed in Hitachi-GE's analysis. (see Assessment Finding AF-ABWR-SA-04)

Furthermore, Hitachi-GE's analysis of venting assumes that the systems will be designed to release an amount of steam corresponding to 1% decay heat power at a PCV pressure of 1×Pd. However, this assumption is not justified yet and thus containment integrity in not assured. (Assessment Finding AF-ABWR-SA-05). (ONR 2017a, p. 52)

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[1] The Assessment Findings are listed in the Annex.

Hydrogen management

During at-power operation, the PCV of the UK ABWR is inerted with nitrogen to limit oxygen concentration to no greater than 4%. Suppressing the level of oxygen limits the potential for combustion of any flammable gases which might accumulate in the PCV in fault or accident conditions. The UK ABWR also includes measures for mitigation of hydrogen in the primary and secondary containments. The overall provision of flammable gas control measures includes:

- Passive autocatalytic recombiners (PARs) are located in the PCV for design basis accidents to prevent the build-up of hydrogen and oxygen which could occur in faults and accidents due to radiolysis of water.
- PCV venting is used to manage hydrogen concentration in the PCV during severe accidents.
- Alternative Nitrogen Injection (ANI), a severe accident provision delivered by mobile equipment, is available post-venting to supporting re-inerting of the PCV.
- PARs and the standby gas treatment system (SGTS) are used for PCV leakage into the R/B during reactor severe accidents.

The R/B blowout panel and door are used for severe accidents involving the shutdown reactor or the SFP.

For the reactor during shutdown with the RPV head removed and for the SFP, the strategy is to provide low-pressure make-up to offset losses due to boil-off or drain-down. Where necessary, any steam and hydrogen discharged into the R/B would be released to the atmosphere to mitigate the risk associated with a hydrogen explosion. A severe accident control and instrumentation (SA C&I) system is provided for the control of severe accident systems and for monitoring of plant conditions using accident qualified equipment. The SA C&I system can be operated from the MCR or the B/B. (ONR 2017a, p.23)

If cooling or make-up cannot be restored, the design basis faults would eventually lead to uncovering the fuel in the shutdown reactor or the SFP, resulting in a severe accident due to fuel damage. Hitachi-GE has identified that a deflagration of hydrogen generated due to steam oxidation of fuel cladding could present a challenge to the R/B structure and important SSCs located within the R/B. Hitachi-GE's strategy for managing hydrogen in the R/B consists of using the blowout panel and a large equipment door at ground level in the R/B to promote natural ventilation, thereby preventing the build-up of flammable concentrations in the R/B. A consequence of the blowout panel being open in a severe accident is that this would allow radioactivity to pass directly from the R/B to the atmosphere.

Assessment

Hitachi-GE has stated that the PAR units will be selected at the detailed design stage and that the locations and performance characteristics will be confirmed as being adequate at that point. Thus, the effectiveness of hydrogen management measures in the primary containment and reactor building has is not assured yet. (see Assessment Finding: (AF-ABWR-SA-06). (ONR 2017a, p. 57)

In addition to the blowout panel, opening of the R/B equipment door is the last step of Hitachi-GE's strategy for managing hydrogen. Details of how the equipment door could be opened have not been provided in GDA. It is not clear how the door would be opened, for example whether power sources would be required to operate any door mechanisms, or how long it would take. It is also unclear how the door will be opened (for example remotely) and whether workers would need to be protected. There also needs to be consideration of how operators in the MCR or B/B would know actions had been performed correctly and that the measure was effective. (see Assessment Finding AF-ABWR-SA-07) (ONR 2017a, p. 60)

Lessons learnt from the Fukushima Dai-ichi accident

To address Fukushima Dai-ichi learning, Hitachi-GE claims that the UK ABWR incorporates a number of enhancements compared to the standard Japanese plant. These are summarized below:

- For external hazards the plant is designed so that there are no ‘cliff-edge’ effects just beyond the design basis.
- Backup DC power supplies have been enhanced. C&I for Class 1 systems is supported by eight hours DC supplies and the battery backup for the steam-driven RCIC has been extended to 24 hours.
- The FLSS has been introduced as an alternative means for providing fuel cooling; this is a fixed system, independent and diverse from the Class 1 ECCS. This is supported by an independent means for reactor depressurization using the RDCF. The FLSS is able to deliver all low-pressure injection and flooding demands and is self-sufficient in fuel and water for seven days.
- There is provision for the use of mobile equipment with dedicated connection points on the outside of the R/B. This includes the FLSR which replicates the low-pressure injection and flooding functions of the FLSS. The alternative heat exchange facility (AHEF) is available to support re-instatement of containment heat removal in the event of LUHS. These systems are supported by mobile power trucks.
- Inclusion of the B/B which is remote from the R/B and is designed to withstand hazards. The B/B houses severe accident systems including the FLSS and is powered by redundant air-cooled diesel generators, diverse from the Class 1 EDGs.
- Key severe accident systems such as the FLSS, RDCF and FCVS can be operated remotely from either the MCR or the B/B.
- As a further means for depressurising the reactor, there is provision for operation of SRVs by local manual operation using nitrogen cylinders.
- A dedicated severe accident C&I system, independent of the Class 1 system and qualified for severe accident conditions, is provided in the B/B. This can be used for the remote monitoring and control of the plant in a severe accident.
- Improvements have been made to PCV seals to enhance resilience of the primary containment to severe accident loads. The PCV head flange seal can also be protected against high temperatures by emergency flooding of the reactor well.
- The design includes enhanced measures for management of hydrogen in the primary and secondary containments.

Assessment

However, Hitachi-GE has the opinion that a number of the specific recommendations and learning points do not relate to the generic design and cannot be addressed by the Requesting Party as part of GDA. Hitachi-GE considers that such matters will be for the future licensee to address. (see Assessment Finding AF-ABWR-SA-08) (ONR 2017a, p. 68).

ONR points out that IAEA learning from the accident Fukushima Dai-ichi identifies the need for instrumentation and control systems that are necessary during beyond design basis accidents to remain operable. A failure of backup building power sources is a potential way for a fault condition to escalate to a severe accident scenario, resulting in the loss of severe accident control and instrumentation functions. Thus, the licensee shall consider whether it is ALARP to provide a capability for mobile power supply sources to ensure that control and monitoring of severe accident systems can be maintained in circumstances where the fixed backup building power sources have failed. (see Assessment Finding AF-ABWR-SA-09) (ONR 2017a, p. 70)

3.3 Conclusions, questions and recommendations

The ABWR design can be considered as being rather old, the development having started in 1978. In the meantime, the development of the successor model ESBWR (with passive safety features) has been completed. Hitachi-GE has also adapted the outdated ABWR design to the UK market and safety requirements. To address Fukushima Dai-ichi learning, Hitachi-GE claims that the UK ABWR incorporates a number of enhancements compared to the standard Japanese plant. However, these measures rely more on the use of mobile equipment and other active measures than on the implementation of passive safety systems in the design.

The UK ABWR includes complementary safety features specifically designed to fulfil safety functions required in postulated core melt accidents. Hitachi-GE claimed that challenges to containment integrity are prevented by specifying an appropriate design envelope and by providing severe accident mitigation measures to keep the design envelope from being exceeded.

However, ONR's thorough GDA Step 4 assessment of severe accidents for the UK ABWR revealed that there are several issues which could endanger the containment integrity or lead to a containment bypass. The need for further examination of the capability and the reliability of the severe accidents measures was addressed in several assessment findings by ONR.

Taking into account all the facts, the safety design and features of the UK ABWR do not guarantee that the radioactive substances will be kept in the containment, neither in the long nor in the short term.

Question

- Which of the 11 assessments findings of the ONR´s GDA step 4 assessment of Severe Accidents for the UK ABWR have already been solved? How were they solved and if not, when will a solution be found for those?

4 ACCIDENT ANALYSIS

4.1 Treatment in the EIA documents

The analysis of radioactive releases from accidents is included in appendix D14-2 of the Environment Statement (ES). This appendix describes the following issues (HNP 2018a, p. 2):

- The main features of the UK ABWR reactor, including safety provisions and design and system features which designed to contain the radioactive substances.
- Aspects of the development of the nuclear safety case relevant to nuclear accident releases and the identification of reference accidents.
- Assumptions and methods used to calculate doses resulting from releases, and the results of the dose assessment.
- Mitigation with emergency planning.
- An impact assessment based on the likely required countermeasures for the reference accidents.

Identification of accident scenarios

Based on the UK ABWR design, the fault schedule was developed from the systematic identification of initiating events, which are grouped according to similar fault sequences and demands on safety functions during the event. The initiating events to be analysed were initially identified by using logic tree analysis. In addition, a bounding fault is identified for each fault group in terms of severity of consequence among the fault group. These are then used for further analysis and for establishing the list of initiating events as input for the probabilistic safety assessment (PSA), design basis analysis, beyond design basis analysis and severe accident analysis. Initiating events are grouped according to their impact on the plant; an indication of their frequency of occurrence is provided. The fault schedule identifies some beyond design basis faults, but does not identify severe accidents. (HNP 2018a, p. 12)

Design Basis Analysis

The purpose of design basis analysis is to assess all the initiating faults/events identified as falling within the design basis. The lower consequence threshold of the design basis region is the basic safety limit, which is the legal limit for annual doses to members of the public of 1 mSv. By this approach, the bounding fault sequences in the design basis analysis should have core damage frequencies below 10^{-7} per year, thus representing a plant design that is of low overall risk (as confirmed by the complementary PSA).

Beyond Design Basis Analysis

In addition to the assessment of the design basis faults, the ONR expects the licensee to analyse fault sequences initiated by internal and external hazards beyond the design basis applying an appropriate combination of engineering, deterministic and probabilistic assessments. The purpose of beyond design basis analysis is to:

- confirm that no cliff-edge effects exist (i.e. there is no potential for sudden and significant consequences associated with events located just outside the design basis boundary (e.g. 9 x 10^{-6} per year)),
- provide an input into the severe accident analysis and
- provide inputs into the PSA to assess whether the overall risk targets are met and confirm that no single fault type dominates the risk profile.

The beyond design basis analysis considers fault and hazard initiating events that have been excluded from the design basis analysis on the basis of low frequency (<10^{-5} per year) but whose frequency is not sufficiently low (>10^{-7} per year) for them to be discounted completely. (HNP 2018a, p. 14)

Severe Accident Analysis

While the combination of design basis analysis, beyond design basis analysis and PSA should ensure that all credible fault scenarios are identified, and suitable and sufficient safety measures are incorporated into the design to prevent/ protect/mitigate against the consequences and ensure that the residual risk is ALARP, the ONR also expects that licensees undertake severe accident analysis. A severe accident is defined as "*an accident with offsite consequences with the potential to exceed 100mSv, or [lead] to a substantial unintended relocation of radioactive material within the facility that places a demand on the integrity of the remaining physical barriers*".

The main purpose of severe accident analysis is to demonstrate the plant safety features included in the design to mitigate the consequences of rare events that involve severe core damage and/or core relocation. The rare events are derived from highly pessimistic assumptions, such as multiple failures of safety systems provided to fulfil fundamental safety functions. (HNP 2018a, p. 14/15)

Reference accidents identified from the fault analysis

A review of the fault schedule was undertaken and three reference design basis accidents (DBA) were identified from the list of faults. The DBAs presented were chosen on the basis of their radiological consequences. **Their assumed frequency of occurrence is over 10^{-5} per year.**

In addition to the DBAs, one Severe Accident (SA) was chosen which is considered to be well beyond the design basis in terms of likelihood. The SA is presented to demonstrate plant safety features to mitigate consequences of a rare event that involves core meltdown and potential radiological releases. Those identified accidents caused atmospheric releases. No accidents involving foreseeable significant liquid effluent releases have been identified. (HNP 2018a, p. 15)

The selected reference accidents are:

- Reference DBAs:
  - Loss of Coolant Accident (LOCA),
  - Fuel Handling Accident (FHA) and
  - Off-Gas system Failure (OGF).
- Severe Accident scenario (SA):
  - Core melt scenario.

Loss of Coolant Accident

Accident scenario: For the design basis LOCA, coolant loss is assumed to occur through a limiting line (i.e. feed water line or main steam-line) which suffers a double ended guillotine rupture inside the Primary Containment Vessel (PCV). Any leakage from the PCV to the reactor building is released from the plant stack via the standby gas treatment system and is considered as a pathway to the environment for radioactive material. The design leakage rate of the primary containment is 0.4% containment volume/day at design pressure and atmospheric temperature. When due account is taken of the primary containment pressure/ temperature rise associated with the LOCA transient the leak rate is calculated to be 0.6% containment volume/day for the first 10 hours of the event.

Release to the environment: Once released to the containment atmosphere, several factors reduce the amounts of materials which could be released into the environment. Two credible pathways for the release of fission products to the environment are leakage from the PCV into the reactor building and via the main steam-line isolation valves. (HNP 2018a, p.16)

Fuel Handling Accident

Accident scenario: During a re-fuelling operation, a fuel assembly is moved over the top of the core. An equipment failure is assumed to occur while the fuel assembly is raised over the core. A maximum of two bundles or 184 fuel rods are assumed to be damaged in the accident, out of a total of 872 bundles.

Release to the environment: As the reactor building has been isolated, the only pathway to the environment is through the standby gas treatment system which releases via the stack. Radioactive decay over the time taken to draw the radioactive air from the reactor building, combined with 99.9% filter efficiency of the standby gas treatment system for all iodine species, reduces the discharge to the environment. (HNP 2018a, p. 16/17)

Off-Gas system Failure

Accident scenario: A rupture or break in the Off-Gas (OG) system is assumed to be discovered by a high radiation level signal in the turbine hall. The automatic isolation valve for the system normally closes within 10 minutes in response to this signal. However, it is conservatively assumed in this scenario that a manual isolation of this system is undertaken by the plant operator which takes one hour following detection of the high radiation level.

Release to the environment: Radioactivity is instantaneously released into the turbine building in this scenario. The release to the environment is assumed to be at ground level and operations that divert the release to the Reactor Building stack are not credited. (HNP 2018a, p. 17/18)

Severe Accident scenario

Accident scenario: When the reactor operates at full power, a loss of feed water leads to a rapid decrease in reactor water level. The transient leads to an emergency reactor shutdown. Core cooling by the main condenser is assumed to be unavailable in this scenario as it is not a safety classified system. At this point, the high-pressure Emergency Core Cooling System (ECCS) is expected to start, but it is assumed to fail. The water inventory in the core is not replenished and continues to be reduced by boiling due to decay heat. When the water level falls below 20% of the bottom of active fuel, two safety release valves are opened manually in order to depressurise the reactor pressure vessel RPV by relief into the suppression pool within the PCV, so the event progresses at low pressure.

In the absence of any core cooling or water injection, the decay heat boils off the remaining core coolant inventory and the fuel becomes exposed. Steam generated during this process continues to pass to the suppression pool via safety release valves. Fuel cladding failure occurs due to creep, melting or ballooning at elevated temperatures. Water-metal reactions can lead to hydrogen gas production; however, hydrogen burning within the primary containment of the UK ABWR is considered implausible as there is a nitrogen injection system in place to maintain an inert atmosphere.

Damaged fuel melts and slumps to the bottom of the core due to gravity. The melted fuel-containing material (corium) perforates the core support plate and the molten debris drains through the failure opening into the lower drywell as a debris jet. The debris jet disintegrates as it enters the water pooled in the lower plenum and settles into segregated entities of a molten pool, corium oxidic crusts, an overlying metallic layer and a particulate bed.

Operators inject water into the drywell in anticipation of RPV failure, using the flooder systems. This is a severe accident response system located in the back-up building. The lower drywell is filled with water to a depth of 2m, which mitigates the possibility of molten core/concrete interaction and breaks up the corium to leave it with a geometry that can be more readily cooled. In addition to the active flooding of the lower drywell, there is a separate dedicated lower drywell flooder system. This provides the passive means to flood the lower drywell by using the water inventory of the suppression pool.

Corium falls through the perforated RPV into the PCV drywell. The flow rate may increase as the opening in the RPV is expanded by the ablating effect of mobile corium. Sprays into the drywell are provided by the flooder systems. This controls the PCV pressure increase and removes fission products from the containment atmosphere. Additional cooling of the corium debris is provided by the core injection function of the flooder systems. Water injected into the core falls onto the molten core in the drywell via the breach in the RPV.

Drywell sprays are continued until the water level within the PCV rises to within 1m of the vacuum breaker. It is assumed that operators successfully recover the residual heat removal system, approximately 17 hours after the accident begins. Restoration of the residual heat removal system by the operator is considered credible at this time. This system facilitates sprays into the drywell, provides debris cooling and removes heat to the ultimate heat sink via suppression pool cooling. Successful residual heat removal system initiation allows for long-term heat removal to be maintained and PCV pressure can be effectively controlled without venting.

Release to the environment: After the initial transient, some fission products are removed from the reactor to the wetwell via the safety release valves. When the RPV fails, more fission products are released into the drywell, some of which are transported to the wetwell through the vacuum breaker between the chambers of the drywell and wetwell. Leakage of fission products from the drywell to the reactor building is expected at the containment design pressure leakage rate (0.4% of containment volume per day at less than design pressure, 1.3% per day at higher pressures). Radioactive contamination released to the reactor building is removed via the standby gas treatment system and discharged to the environment via the reactor building stack. (HNP 2018a, p. 18ff)

Assessment of the radiological impact of the reference accidents

The assessment considers the radiological consequences of releases to the atmosphere for two reference groups comprising members of the public:

- a local reference group close to the Power Station Site and
- a reference group in the nearest country (Ireland).

The nearest country (Ireland) reference group is assumed to be located at a distance of 118km and a bearing of 266° from north. (HNP 2018a, p. 23)

For both groups, the results presented are based on a Gaussian plume model and correspond to the plume centreline and therefore the maximum concentrations for the distance considered. It is assumed that the weather conditions remain constant for the duration of the release and also during the period of plume travel. The release paths and release durations for the reference accidents are summarised in table 4-1 of the ES. (HNP 2018a, p. 24)

Table 1: Release paths and release durations for the reference accidents (source: HNP 2018a, p. 24)

	Reference accident	Release duration local (h)	Release duration Ireland (h)	Release path
LOCA	Loss of Coolant Accident	24	24	88% from plant stack. 12% from turbine build
FHA	Fuel Handling Accident	24	24	100% from plant stack
OGF	Off-Gas system Failure	1	12	100% from turbine build.
SA	Containment leakage from Drywell (failed RPV)	4	12	100% from plant stack

Table 4-3 of ES presents a summary of reference accident source terms. The relevant nuclides I-131, I-133, Cs-134 and Cs-137 are shown in the following table (HNP 2018a, p. 25):

	Release (Bq)			
Nuclide	**LOCA**	**FHA**	**OGF**	**SA**
I-131	1.40E+06	7.40E+05	1.60E+09	2.50E+09
I-133	1.10E+05	4.90E+04	2.00E+09	2.91E+09
Cs-134	1.80E+05	2.10E+06	6.90E+05	3.18E+08
Cs-137	9.70E+04	1.90E+08	5.70E+05	1.86E+08

Table 2: Reference accident source terms for I-131, I-133, Cs-134 and Cs-137 (Souce: HNP 2018a, p. 25)

Models and parameter values used

For the local assessment, time-integrated activity concentrations are calculated. The dry deposition velocities and the washout coefficients used are presented. For the assessment of the possible consequences of the local reference group, deposition parameters, meteorological conditions, habit data and inhalation rate are described. (HNP 2018a, p. 26ff)

For the nearest country (Ireland), the time-integrated activity concentration, the dry deposition and the wet deposition are calculated as described in (JONES 1981b), whilst the plume depletion due to deposition is calculated as described in (JONES 1981a). The results obtained are for the 90th percentile. The dry deposition velocities and the washout coefficients are the same as those used for the local assessment. The meteorological conditions are based on the (JONES 1981b) methodology and are presented in table 4-8 of the ES (HNP 2018a, p. 28)

Table 3: Meteorological parameters for the assessment for the nearest country (Ireland) (source: HNP 2018a, p. 28)

Mixing layer depth (m)	Wind speed (m/s)	Rainfall rate in wet conditions (mm/hr)
1000	8	0.1

The following exposure pathways are considered in the calculation of doses:

- cloud gamma from the plume,
- ground gamma due to deposited radionuclides,
- inhalation from the plume,
- inhalation as a result of resuspension of deposited radionuclides and
- ingestion of contaminated food.

The activity concentration in soil and terrestrial foods per unit deposit values were obtained using the FARMLAND model within PC CREAM 08 (SMITH 2009). The effective dose coefficients for inhalation and ingestion are taken from International Commission on Radiological Protection (ICRP) data. (HNP 2018a, p. 31/32)

Maximum time integrated concentrations and surface contamination levels

Table 4-10 of the ES presents the maximum time integrated activity concentrations for the two reference groups. For FHA, OGF and SA, the difference between the time integrated activity concentrations for dry weather conditions and wet weather conditions is insignificant and so a single value is presented. (HNP 2018a, p. 33)

Table 4: Maximum time integrated activity concentration (source: HNP 2018a, p. 33)

Reference accident scenario	Time integrated activity concentration (Bqs/m^3)	
	Local reference group	Ireland reference group
LOCA	7.57E+04 (dry) 2.13E+05 (wet)	1.41E+02 (dry) 1.39E+02 (wet)
FHA	7.37E+08	1.67E+06
OGF	2.88E+08	1.37E+03
SA	1.11E+11	1.51E+08

The maximum surface contamination levels for the two reference groups are presented in table 4-11 of the ES. Results for dry weather conditions and wet weather conditions are provided. As expected, wet weather conditions result in higher surface contamination levels than dry weather conditions as a result of washout. Wet weather conditions were conservatively used to assess the doses for the three design basis faults. However, more realistic dry weather conditions were more appropriate for the representative SA assessed given the low frequency of such events. (HNP 2018a, p. 33)

Table 5: Maximum surface contamination levels (source: HNP 2018a, p. 33)

Reference accident scenario	Surface contamination (Bq/m²)			
	Local reference group		Ireland reference group	
	Dry weather	Wet weather	Dry weather	Wet weather
LOCA	8.82E+00	1.01E+02	1.63E-02	1.55E-01
FHA	2.46E-01	7.32E-01	5.52E-04	5.24E-03
OGF	1.04E+04	1.41E+04	4.35E-01	4.13E+00
SA	1.26E+04	1.29E+04	1.49E+01	1.66E+01

Calculated effective doses

Tables 4-13 to 4-15 and tables 4-19 to 4-21 of the ES present the effective dose to an adult, a 10-year-old child and a one-year old infant for the local reference group and the Ireland reference group respectively. (HNP 2018a, p. 34 ff.)

All three DBAs result in low off-site releases and resulting doses are below 1mSv and judged as being of negligible impact and negligible significance. The SA also has an assessed impact of below 1mSv. Based on this, the SA is also judged as being of negligible impact and negligible significance.

Doses in the nearest country (Ireland) are two to three orders of magnitude lower than this. The resulting impact is also assessed as negligible. Assuming an inverse power relationship between air concentration, ground deposition and dose with distance from the Power Station, impacts at greater distances will also be much lower than this. (HNP 2018a, p. 47)

Release to the aquatic environment

There are two potential routes for liquid radioactive wastes to enter the environment from the UK ABWR as a result of a fault or accident:

- release from the reactor building – the reactor building houses structures containing radioactive liquids, namely the reactor coolant and
- release from the radioactive waste building – this houses the liquid effluent management system and, therefore, radioactive liquids.

The confinement of radioactive material offered by the primary and secondary containment structures of the UK ABWR is considered sufficiently robust to negate the risk of a significant release of liquid radioactive effluent to the aquatic environment.

In the event of a LOCA, all lines from the drywell sumps are automatically isolated to preclude uncontrolled release of primary coolant outside the primary containment. In the event of a fault condition which results in excessive inflow rates of radioactive liquid waste into the drywell sump, an alarm is actuated. A release of liquid radioactive effluent from the radioactive waste building resulting from an operator error is not considered likely due to the design of the facility and passive mitigation measures in place. In the event of a release of liquid radioactive effluent, the radioactive waste building is equipped with floor drain sump pumps which upon receipt of a high water level alarm automatically remove the spilled liquid to the contained storage tank. The measures outlined provide sufficient control that accidents resulting in releases to the aquatic environment have been scoped out. (HNP 2018a, p. 20/21)

4.2 Discussion

The approach to calculate the radiological consequences of a possible accident in the Wylfa Newydd NPP is well documented in the Environmental Statement. However, there are no reasons mentioned for the choice of the representative severe accident. This is important because its assumed release for Caesium-137 is relatively small (1.86E+08 Bq). As mentioned above, a core-melt accident with containment failure or by-pass, resulting in the release of huge amounts of radioactive material in the environment, cannot be excluded for the UK ABWR.

In the following, the probabilistic safety analysis (PSA) for the UK ABWR is evaluated. For this purpose, ONR's review of the GDA "Step 4 Assessment of probabilistic safety analysis for the UK Advanced Boiling Water Reactor" is used. (ONR 2017b)

Probabilistic safety analysis

PSA results are of considerable value to provide guidance to NPP designers and regulators (for example, to identify weak points in a reactor design). On the other hand, the inherent limitations of PSA should not be forgotten – such analyses are beset with considerable uncertainties, and some risk factors are difficult to include in a PSA, or cannot be included at all:

- Unexpected plant defects or unforeseen physical or chemical processes cannot be included in the PSA.
- Ageing phenomena can only be incorporated in PSAs in retrospect.
- Complex forms of human error are extremely difficult to model.
- Due to the complexity of an NPP, some accident initiators or sequences are simply bound to be overlooked or omitted.

The PSA for the UK ABWR is described in Chapter 25 of the Pre-Construction Safety Report (PCSR). The PSA has been carried out at Level 1, 2 and 3.

The following paragraphs describe the specific limitations of the UK ABWR PSA

Based upon the submissions made by Hitachi-GE during Steps 2 and 3 of the GDA for the UK ABWR, ONR judged that there were serious regulatory shortfalls associated with the development of a modern standards full-scope PSA for the UK ABWR. These had the potential to prevent provision of a Design Acceptance Confirmation (DAC). This was considered to be a serious regulatory shortfall and escalated to a regulatory issue (RI) in July 2015 (RI-ABWR-0002). In response to RI-ABWR-0002, Hitachi-GE extended its PSA capability, and submitted a revised UK ABWR PSA. Following ONR assessment of Hitachi-GE submissions, RI-ABWR-0002 was closed during Step 4. (ONR 2017b, p. 3)

According to ONR, the overall scope of the UK ABWR PSA is sufficient to support the UK ABWR 'generic' PCSR and to reflect the design reference. The scope and content of the PSA is adequate for GDA. However, the PSA needs to be revised beyond GDA to reflect the final detailed design, address shortfalls identified by the GDA review and include site-specific characteristics and operational matters. (*see Assessment Finding AF-UKABWR-PSA-001 Part 1-2)*[2] (ONR 2017b, p. 27)

Assumptions in the PSA not justified

The review of the different technical areas of the PSA has identified shortfalls related to the use of assumptions. Some assumptions have primarily been made either to supplement a lack of design or procedural information, or due to simplifications in the analysis. The PSA assumptions will need to be reviewed beyond GDA when further information becomes available. (see *Assessment Finding AF-UKABWR-PSA-001, Part 3***)**

Some examples are provided below to illustrate the type of shortfalls identified:

- The internal fire PSA and the internal fire PSA refinement, due to a lack of information available during GDA, rely on many assumptions such as cable routing and back-up building barriers. These assumptions have resulted in risk reduction and therefore, it is important that the assumed design features are substantiated and reflected in the detailed design.
- The sensitivity to the PSA assumption of failure of emergency core cooling system (ECCS) due to containment failure shows that the large release frequency (LRF) could be significantly reduced if ECCS survivability can be justified. Analysis of the survivability and/or operating limits versus the expected conditions inside the reactor building is needed.
- The use of basaltic concrete is assumed in the analysis of containment response to molten core concrete interaction (MCCI). Confirmation of this key assumption will be needed beyond GDA.
- Low-pressure injection valves are assumed to close against full reactor coolant system (RCS) pressure. This assumption is currently justified on the basis of the purchase specification to the valve vendor for the Japanese ABWR (J-ABWR) and is included in the UK ABWR assumption list for future resolution. (ONR 2017b, p. 27/28)

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[2] The Assessment Findings are listed in the Annex.

List of Initiating Events (IEs) not complete

The Step 3 review concluded that a significant number of initiating events (IEs) were missing or not explicitly considered in the PSA. ONR review in Step 4 has concluded that the IEs are still missing from the PSA. For example, the PSA does not consider a loss of ultimate heat sink (LUHS) that could lead to the loss of all external water sources such as, for example, blockage of the intake. (ONR 2017b, p. 32 ff.)

Level 1 PSA: Adverse Environmental Conditions not sufficiently analysed

Hitachi-GE considered that **adverse environmental** conditions would be bounded by the consideration of loss of room cooling. However, ONR has found the following conditions which have not been analysed in detail and could be more severe than loss of room cooling:

- Environmental conditions after containment failure or high energy line breaks outside containment may compromise equipment availability.
- There is no consideration that debris, either internal or external to the system or plant, could block screens or filters (with the exception of suppression pool suction strainers being explicitly modelled). (ONR 2017b, p. 50)

Human Reliability Analysis (HRA) not substantiated

The review of the level 1 PSA has raised some concerns regarding the approach used for the inclusion of post-accident human failure events into the system models and the treatment of dependencies in the accident sequences. (ONR 2017b, p. 54)

ONR's review of the level 2 PSA has identified that only a limited set of human failure events (HFEs) are included; some examples of missing HFEs are (ONR 2017b, p. 56)

- Errors of commission (EOCs).
- The potential for adverse effects of severe accident management actions.
- Drywell venting (filtered or unfiltered). This may lead to a different type of release.
- Failure to reclose the containment vent after containment venting, which may lead to the inerting of containment being lost when the reduction in decay heat leads to a reduction in steam generation (which could lead to accumulation of hydrogen).
- Coordination of external water injection and containment water level control.
- Drywell spray for radionuclide release mitigation for temperature and temperature control.

Unavailabilities due to Testing and Maintenance not considered

The review has also identified that outage, maintenance and test unavailabilities were not considered for standby components where an unavailability time is not currently defined. Once the technical specifications are available for these systems their maintenance unavailabilities should be incorporated into the PSA. (Assessment Finding AF-UKABWR-PSA-002). (ONR 2017b, p. 59/60)

Common Cause Failures (CCF) not appropriately considered

The approach selected for modelling CCFs doesn't address intersystem events. The following issues were raised:

- It is considered that several credible CCF combinations may be found in MCS with low failure frequencies below the cut-off used.
- The consideration of diversity between the identified intersystem CCF candidates has not been performed at a sufficient level of detail to provide confidence that the components were sufficiently diverse to exclude an intersystem CCF from being modelled.
- Specific concerns were raised regarding the potential for intersystem CCFs between the EDGs and BBGs.

These shortfalls have resulted in Assessment Finding AF-UKABWR-PSA-003 (ONR 2017b, p. 60/61)

Scope of the internal and external hazards PSA limited

The prioritisation of internal hazards for PSA relies upon the information available at this stage. Thus, a revised systematic prioritisation of all internal hazards, including combined internal hazards consistent with the internal hazards deterministic safety case is missing. The demonstration that the risk associated with all the screened out internal hazards would be insignificant compared to the ABWR total risk is also missing. (*see Assessment Finding AF-UKABWR-PSA-004)* (ONR 2017b, p. 62/63)

The analysis performed for GDA is generic and defers consideration of a number of hazards to the site-specific phase. A significant number of external hazards have been excluded from Hitachi-GE analysis due to lack of site-specific information to be able to evaluate the impact of the hazard. Examples of hazards that have not been assessed by Hitachi-GE due to reliance on unavailable site-specific information are external fire, external explosion and external transport impacts. *(see Assessment Finding AF-UKABWR-PSA-005*) (ONR 2017b, p. 81)

Hitachi-GE performed a PSA sensitivity study to examine the impact on the risk of a loss of the ultimate heat sink (LUHS) caused by biological fouling and the impact on risk from external flooding. This sensitivity study shows that a biological fouling event could represent a significant proportion of the CDF. LUHS due to external hazard has the potential to be a significant contributor to the UK ABWR overall risk profile and requires further analysis in the site-specific phase. It should be noted that the fault schedule considers a reserve ultimate heat sink (RUHS) to provide protection against LUHS events. Design of the RUHS is considered by Hitachi-GE out of the scope of GDA and availability of a RUHS is not considered in the PSA sensitivity study. (*see Assessment Finding AF-UKABWR-PSA-006*) (ONR 2017b, p. 81/82)

Reduces the Internal Fire Risk and Flooding Risks only on paper

Prior to the end of Step 4, Hitachi-GE undertook further refinement of the internal hazard PSAs, removing conservatisms and taking credit for additional mitigating and protective measures.

The output of the internal hazards PSA refinement reduces the CDF and LRF of the fire PSA results by a factor of 3.8 and 6.2, respectively. The CDF from internal fires CDF has been reduced from one third of the total to 12% of the total CDF. According to ONR, the internal fire aspects of the internal hazards PSA refinement has also identified new assumptions about the design which are not always clearly identified. (ONR 2017a, p. 71)

The LRF for the internal flooding at power PSA was reduced by a factor of 4.5 by an internal hazards PSA refinement study, which removed selected conservatisms using newly available design information. However, the internal flooding at power PSA needs further development during the site-specific phase. (ONR 2017b, p. 78)

Seismic PSA (SPSA) not appropriate yet

Based on the outcome of this assessment, ONR has concluded that the SPSA developed by Hitachi-GE is sufficient to support the UK ABWR 'generic' PCSR. However, it is important to note that the SPSA has identified that the risk associated to seismic events for the UK ABWR can be significant (in comparison with the risk from internal events), but this is dependent on specific characteristics of each site. To support future stages of development of the NPP, the SPSA and seismic fragility analysis need to be revised to take site-specific characteristics and plant-specific design into consideration as realistically as possible.

Sensitivity analyses conservatively assumed higher stress factors, which resulted in significant increase of 25% for the LRF associated with the SFP. This is likely due to the higher reliance on operator actions in response to faults affecting the SFP. (ONR 2017b, p. 86ff)

Level 1 PSA: Shutdown Modes and Spent Fuel Pool only simplified

Simplified internal fire and flood analyses have been undertaken for shutdown states and for the spent fuel pool. The analysis should be extended as required to be consistent with the at power internal fire and flood PSAs and reflect the site-specific design, operation and maintenance of the UK ABWR. (see Assessment Finding AF-UKABWR-PSA-007) (ONR 2017b, p. 94)

Uncertainty Analyses not sufficient

Step 3 of GDA revealed that the sensitivity analyses performed by Hitachi-GE were insufficient to demonstrate that the modelling assumptions and uncertainties had minimal impact on the PSA conclusions. In addition, parametric uncertainty propagation analyses for the UK ABWR level 1 and level 2 PSA had not been undertaken. The PSA database identified some assumptions that had a significant impact on the UK ABWR PSA results. However, it was not clear how Hitachi-GE proposed to reduce these uncertainties. Further investigation regarding the differences between the mean and the point estimate is needed, and the PSA model and documentation to be updated, as appropriate, to allow for the uncertainty analysis to be taken into account in any decisions made on the basis of PSA results, and provide confidence that the overall conclusions obtained from the PSA are valid. (see Assessment Finding AF-UKABWR-PSA-008). (ONR 2017b, p. 100 ff.)

The process should also include the review of international guidance to extend the list of potential uncertainties that need consideration, such as ECCS strainer reliability data and severe accident phenomenology. In addition, ONR identified a number of specific shortfalls related to Hitachi-GE's sensitivity analysis. These concerns point to a completeness issue in the identification of modelling assumptions and uncertainties. (ONR 2017b, p. 102)

Limitation of the PSA 2

On the basis of the assessment of the level 2 PSA, ONR concluded that Hitachi-GE's level 2 PSA is sufficient for the 'generic' PCSR. However, improvements to support further stages of the NPP development are required to extend the consideration of severe accident phenomena, reduce uncertainty and conservatisms, reflect the UK ABWR detailed design and SAMGs when available, and to reflect the results of the containment performance analysis. (ONR 2017b, p. 115)

ONR´s review has identified some severe accident phenomena for which there is a lack of clarity and justification regarding their consideration or omission in the PSA. For example:

- Bypass of the suppression pool (S/P) due to vacuum breakers (V/Bs) failed open or other structural failures of the wetwell to drywell interface have not been considered in the PSA. (ONR 2017b, p. 108)

In general, the accident progression analyses have been performed on a 'best estimate' basis. However, ONR's review has identified some areas where there may be excess conservatism or optimism in the accident representations, in particular:

- The PSA assumes that FLSS injection at any point prior to core plate failure is sufficient for achieving in-vessel melt coolability and will not result in RPV vessel breach. There is a lack of justification provided for this assumption, specifically for injection just prior to core plate failure. (ONR 2017b, p. 108/109)

The review identified that some SSCs and failure modes are omitted from the PSA without justification, which could affect the accident progression. The review has identified limitations to how the level 2 PSA considered system operation under degraded conditions with respect to:

- adverse environment,
- system limitations, interlocks or trips,
- operator manipulation success when high radiation may be present.

In particular, environmental conditions related to core damage progression, containment leakage or containment failure are not always included in the PSA. In addition, there is a lack of substantiation regarding the assumption that the SRVs will remain open during the core degradation phase, given the high heat loads expected, and that the SRV tailpipes remain intact. (ONR 2017b, p. 112/113)

Results of the PSA for UK ABWR

Table 2 of ONR (2017a, p. 21/22) presents a summary of the PSA results as reported in the Hitachi-GE's PSA summary report of July 2017.

Table 6: UK ABWR PSA results (source: ONR 2017a, p. 21/22)

	UK ABWR PSA result (yr^{-1})		
Item	**Core damage Frequency (CDF)**	**Large Release Frequency (LRF)**	**Frequency of 100 fatalities***
Internal events at power	2.3 x E-07	4.6 x E-08	6.5 x E-08
Internal events during shutdown POS	8.7 x E-08	6.9 x E-08	7.0 x E-08
Internal events for SPF	4.2 x E-07	4.8 x E-08	4.8 x E-08
Internal fire events at power	Initial: 1.9 x E-06 Refined: 5.0 x E-07	Initial: 1.6 x E-06 Refined: 2.7 x E-07	Initial: 2.4 x E-06 Refined: 3.1 x E-07
Internal flood events at power	Initial: 1.8 x E-06 Refined: 1.8 x E-06	Initial: 7.8 x E-07 Refined: 1.8 x E-07	Initial: 7.9 x E-07 Refined: 5.8 x E-07
Seismic events at power	7.3 x E-07	6.1 x E-07	6.5 x E-07
Seismic events for spent fuel pool	4.5 x E-7	3.9 x E-7	3.9 x E-7
Seismic events during shutdown POS	4.2 x E-08	Not calculated	Not calculated
Tornado missile events	5.2 x E-10	2.4 x E-10	Not calculated
Turbine missile events	7.1 x E-10	8.1 x E-11	Not calculated
Accidental aircraft impact	7.9 x E-10	4.6 x E-10	Not calculated
Total (including refined internal hazards)	**4.3 x E-06**	**1.6 x E-06**	**2.1 x E-06**

* *Frequency of 100 fatalities is related to SAP Target 9. The difference in LRF and the frequency of 100 fatalities is mainly due to some release categories which result in greater than 100 fatalities not being categorised as a large release by Hitachi-GE.*

4.2.1 Safety Standards

According to ONR, standards issued by the International Atomic Energy Agency (IAEA) and guidance from the Western European Nuclear Regulators Association (WENRA) have to be applied for the severe accidents assessment of the UK ABWR.

The latest version of the Safety Assessment Principles (SAPs) (ONR 2014) was benchmarked against the extant IAEA and WENRA guidance in 2014. The general approach adopted in the GDA Step 4 assessment of severe accident has been to assess Hitachi-GE's submissions against the SAPs, and as a result it can be inferred that international guidance is met.

There are specific provisions in the WENRA guidance that ONR refers to in the assessment. For new reactors, WENRA Objective O3 on 'Accidents with Core Melt' is particularly relevant to severe accidents. It sets the expectation that 'large or early' releases are practically eliminated. WENRA has provided further guidance on this Objective, in particular:

- Position 4: Provisions to mitigate core melt and radiological consequences
- Position 5: Practical elimination

In line with the international guidance, ONR's SAPs also include an expectation that potential severe accident states have been 'practically eliminated'. To demonstrate practical elimination, the safety case should show either that it is physically impossible for the accident state to occur or that design provisions mean that the state can be considered to be extremely unlikely with a high degree of confidence. (ONR 2017a, p.12/13)

4.2.2 Practical elimination of large or early releases

There is an international expectation that large or early releases be practically eliminated for new reactors. It is also a requirement of Hitachi-GE's own internal Nuclear Safety and Environmental Design Principles, specifically Principle 8.11.1, that "significant radioactive releases are practically eliminated". Hitachi-GE has interpreted 'significant radioactive releases' as being equivalent to large or early release.[3] Hitachi-GE claims that large or early releases have been practically eliminated for the UK ABWR by:

- identifying the provisions which are designed to prevent or mitigate an accident,
- identifying conditions which could lead to large or early releases and
- demonstrating that large or early releases are of 'extremely low likelihood' with a high degree of confidence.

WENRA guidance states that accident sequences with a large or early release can be considered to have been practically eliminated if it is physically impossible for the accident sequence to occur; or the accident sequence can be considered with a high degree of confidence to be extremely unlikely to arise. Hitachi-GE has not claimed that any specific sequence is 'physically impossible' and instead has identified all relevant severe accident phenomena and analysed these as part of the Level 2 PSA.

The available guidance states clearly that an accident state should not be considered to have been practically eliminated simply on the basis of meeting probabilistic criteria. Hitachi-GE takes PSA results into account, supported by sensitivity analyses, to inform its conclusions on practical elimination. In the opinion of ONR it is appropriate for Hitachi-GE to use results from the UK ABWR PSA to support judgements as one part of its wider case.

In the context of practical elimination of large or early releases, there is no common position in the international guidance on use of numerical targets to define what is 'extremely unlikely'. ONR does not set explicit targets for measures such as large release frequency. However, ONR does equate such measures with Target 9 in the Safety Assessment Principle(s) SAPs. According to the SAPs, safety cases should be assessed against the SAPs numerical targets[4].

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[3] Note: WENRA (2010) defines early releases as situations that would require off-site emergency measures but with insufficient time to implement them. There is no WENRA definition of the time that should be assumed for implementation of emergency measures. Hitachi-GE refers to its PSA definition of an early release; an early release is one where containment failure occurs within four hours of RPV breach, or occurs before RPV breach, but within 10 hours of the initiating event. Furthermore, release category with CsI release fraction greater than 10 percent is regarded as large release

[4] Target 7: Individual risk to people off the site from accidents: BSL 10^{-4}/yr, BSO 10^{-6}/yr

Target 8: Frequency dose targets for accidents for any person off the site; 1) 0.1-1 mSv: BSL 1/yr, BSO 10^{-2}/yr; 2) 1-10 mSv: BSL 10^{-1}/yr, BSO 10^{-3}/yr; 3) 10-100 mSv: BSL 10^{-2}/yr, BSO 10^{-4}/yr; 4) 100-1000 mSv: BSL10^{-3}/yr, BSO 10^{-5}/yr; 5)>1000 mSv: BSL 10^{-4}/yr; BSO 10^{-6}/yr

Target 9: Total risk of 100 or more fatalities: BSL 10^{-5}/yr, BSO 10^{-7}/yr

ONR holds the opinion that the BSOs are relevant measures for new reactor designs proposed for the UK. These are used by ONR as benchmarks that reflect modern standards and expectations, thus ONR refers to these objectives to assess whether analyses are demonstrating adequate results for new reactors.

With the help of the PSA, Hitachi-GE has identified accident sequences which could result in containment failure. For these sequences, Hitachi-GE has supported its arguments for its claim of practical elimination with reference to the Level 3 PSA for the reactor at power. **The results show that the Target 9 risk for reactor accidents, summed for all large and large early release categories, is approximately 10^{-6}/year. This value is above ONR's BSO,** but below the BSL (Target 9: BSL 10^{-5}/yr BSO 10^{-7}/yr)**.** The risks are dominated by accidents initiated by internal hazard events. Hitachi-GE has presented arguments that further improvements to the PSA model are planned post-GDA and are will reduce the calculated risk for hazards. (ONR 2017a, p. 70 ff.)

To meet UK and international expectations post-Fukushima, Hitachi-GE has provided a demonstration which argues that the generic UK ABWR design practically eliminates large or early releases. The extent to which GDA can take into account hazards is limited; thus also the completeness of practical elimination claims. **ONR notes that Hitachi-GE has not quantified risks for internal hazard initiators for shutdown and the SFP. Furthermore, Hitachi-GE has not considered the PSA contribution from external hazards when considering practical elimination**. In particular, external hazards will present an additional contribution to the site-specific risk profile. For the specific site ONR expects an update of the arguments on practical elimination. (ONR 2017a, p. 74/75)

Demonstration that risks are ALARP

ONR emphasised that the BSOs are 'objectives' and not requirements – the overriding legal requirement for new reactor designs consists of the level of risk which is demonstrated to be as low as reasonably practicable (ALARP) when the facility starts operation and over its lifetime. To meet the UK requirements, it is necessary to show that the radiation doses to workers and the general public due to the operation of a nuclear facility, taking into account the possibility of accidents, will be ALARP.

Demonstration that risks are ALARP is a fundamental requirement of UK law that a future licensee would have to comply with. ONR considers that Hitachi-GE has identified the most significant PSA insights that need to be considered as part of the ALARP demonstration. (see Assessment Finding AF-ABWR-SA-10). However, ONR points out that the work submitted by Hitachi-GE does not fully demonstrate that the risks for the UK ABWR are ALARP from a PSA point of view. (ONR 2017b, p.126)

The results produced by Hitachi-GE meet the BSOs for Target 7 and dose bands 1 to 4 of Target 8. But Target 9 and Target 8 dose band 5 results are above the BSOs. Further work is required following GDA to demonstrate that the risks are ALARP (ONR 2017a, p. 152).

Over the course of Step 4, the total large release frequency (LRF) for the UK ABWR was reduced by approximately a factor of four, significantly increasing the margin to the BSL of ONR SAPs Target 9. The main cause of this reduction was the refinement of the internal fire and internal flooding PSAs, which removed selected conservatisms and took ALARP options identified as part of the review of PSA insights into account. However, as mentioned above, the refinement relies on a number of assumptions about the design, which require substantiation following GDA. (ONR 2017b, p.124)

It is ONR's policy that new reactors meet the BSLs and strive to meet the BSOs. Comparison of the results of the UK ABWR PSA against SAP Target 9 shows that the estimated risk is well below the BSL. However, the risk remains above the BSO for SAPs Target 9, and for Target 8 for doses above 1 sievert. (ONR 2017b, p.127)

4.2.3 External hazards

Site-specific factors (like hazard of seismic or tsunamis events, influence of the climate change) that could endanger the plant are not discussed appropriately in the Environmental Statement.

Flooding can be catastrophic to a nuclear power plant because it can damage its electrical systems, disabling its cooling mechanisms and leading to overheating and possible meltdown and a dangerous release of radioactivity. The Fukushima accident highlighted the hazard of flooding events for nuclear power plants. One of the main questions after the Fukushima accident was the predictability of the wave height of the tsunami.

In 2012 the ENSREG peer review team concluded that the currently available design basis flood (DBF) assessments in the UK did not take into account recent **tsunami** research work. It was noted that ONR believes that these studies are unlikely to significantly affect previous understanding of maximum credible tsunami heights. (ENSREG 2012)

The ES referred to an outed scientific report on the impact of climate change. According to media, a number of scientific papers published in 2018 suggested that climate change will impact coastal nuclear plants earlier and harder than industry, governments or regulatory bodies have expected, and that safety standards set by national nuclear regulators and the International Atomic Energy agency (IAEA) are outdated and do not sufficiently take into account the effects of climate change on nuclear power.

IAEA's current global safety standards were published in 2011. These state that operators should only "take into account" the 18- to 59-centimeter sea-level rise projected by 2100 in the Intergovernmental Panel on Climate Change (IPCC)'s fourth assessment report, published in 2007. But those safety standards do not factor in the most recent assessment of the IPCC, published in 2013-14. This scientific consensus report has seas rising 26 centimetres to 1 meter by 2100, depending on how high temperatures will continue to rise and the speed of the polar ice caps' melting.

According to scientists it is necessary to consider not only the sea-level rise, but also the added impact of flooding from storm surges. The results of the Global Extreme Sea Level Analysis project showed that the magnitude and frequency of **extreme sea levels** (ESLs, a factor of mean sea level, tide and storm-induced increases), which can cause catastrophic flooding, have increased throughout the world since 1970. New satellite studies by the U.S. government's National

Oceanic and Atmospheric Administration (NOAA), NASA, and other leading scientific institutions all show mean sea level rising and magnifying the frequency and severity of ESLs. (ENSIA 2018)

The seismic hazard for the Wylfa Newydd is not discussed in the ES although it is of particular interest to Austria.

In July 2007, all seven reactors at the Kashiwazaki-Kariwa site were struck by the 6.8 magnitude Niigata Chuetsu offshore (NCO) earthquake in Japan. The plant design was not laid out to withstand the location and magnitude of such an earthquake. Some 63 incidents were confirmed, including a release of radioactive iodine through the main stack at unit 7 (ABWR). Unit 7 was restarted after almost 22 months of checks and repairs. The safety margin of the plant's SSCs prevented a severe accident after the hit of the earthquake; however, failures of some non-safety SSCs caused unexpected damages to the plant. One important lesson learnt from the NCO earthquake is that a well-protected nuclear power plant should have substantial seismic margin. (BECKER 2013).

4.2.4 Prevention of liquid radioactive releases

During and after an accident, the liquid radioactive release can only be prevented if the waste system and release routes are guaranteed to be safe. Otherwise, the radioactive liquid will first flow into the reactor building sump and then overflow. In the worst case, the liquid submerges into floors of the building. Then it continues outside the building into rain water sewers or sinks into the bottom layer sea water draining tunnel. The release will flow into the sea via the cooling sea water outlets.

ENSREG points out that conceptual solutions for post-accident fixing of contamination and the treatment of potentially large volumes of contaminated water should be addressed (ENSREG 2012b). This important issue highlighted by the Fukushima accident is not addressed in the EIA documents.

4.3 Conclusions, questions and recommendations

The approach to calculate the radiological consequences of a possible accident in the Wylfa Newydd NPP is well documented in the Environmental Statement. However, there are no reasons mentioned for the choice of the representative severe accident. This is important because the assumed release is relatively small. As discussed in the previous chapter (reactor type), a core-melt accident with containment failure or by-pass, resulting in the release of huge amounts of radioactive material in the environment cannot be excluded for the UK ABWR.

The reference accident scenarios as well as the associated releases are based on the Probabilistic Safety Analysis (PSA). In general, PSA results should only be taken as rough indicators of risk. All PSA results are beset with considerable uncertainties, and there are factors contributing to NPP hazards which cannot be included in the PSA. ONR´s review of the PSA for the UK ABWR during the GDA Step 4 came up with a number of shortcomings. Many factors were not included or not addressed appropriately (for example adverse environmental conditions, human failure events (HFEs), specific common cause failures (CCFs), internal and external hazards).

To meet the regulation expectations, Hitachi-GE undertook a refinement study of the internal hazard PSA over the course of GDA Step 4, mainly removing conservatisms. In this way, the total large release frequency (LRF) for the UK ABWR was reduced by approximately a factor of four.

However, the PSA results for the UK ABWR showed that the SAP Target 9 risk (= total risk of 100 or more fatalities), summed for all large and large early release categories, is approximately 10^{-6}/year. This value is below basis safety level (BSL), but above the basis safety objective (BSO) (Target 9: BSL 10^{-5}/yr BSO 10^{-7}/yr).

ONR emphasised that the BSOs are ‘objectives' and not requirements – the overriding legal requirement for new reactor designs consists in demonstrating that the level of risk is as low as reasonably practicable (ALARP). However, ONR pointed out that Hitachi-GE, has not sufficiently demonstrated that the risks for the UK ABWR are ALARP from a PSA point of view. Further work is required after GDA.

According to Hitachi-GE severe accidents leading to early and large releases will be practically eliminated for the UK ABWR. However, Hitachi-GE has neither quantified risks for internal hazard initiators for shutdown and the SFP nor considered the PSA contribution from external hazards when considering practical elimination. Thus, all in all the practical elimination of accident sequences leading to early or large releases is not proven.

Currently, it cannot be demonstrated beyond doubt that a severe accident with major radioactive releases could not occur at the Wylfa Newydd NPP.

Therefore, a conservative worst-case release scenario should have been included in the EIA. As mentioned above, a source term, for example for an early containment failure or containment bypass scenario, should have been analysed as part of the EIA – in particular because of its relevance for impacts at greater distances.

It is important to note that site-specific factors (such as hazards of seismic or tsunamis events, climate change impacts) that could endanger the plant are not discussed appropriately in the Environmental Statement. Loss of the ultimate heat sink (LUHS) due to external hazard (e.g. biological fouling) has the potential of significantly contributing to the UK ABWR overall risk profile. Therefore, it is very important to implement a robust reserve ultimate heat sink (RUHS) for the Wylfa Newydd site.

Questions

- What will be the response to the fact that the UK ABWR design does not meet the SAP BSO of target 9? Is there any progress regarding this issue in the ongoing nuclear site licence (NSL) procedure? What could be the consequences for Wylfa Newydd NPP if Horizon fails to meet this safety objective?
- What will be the consequences of the fact that the UK ABWR design does not meet the UK legal requirements for new reactor designs by demonstrating that the level of risk is as low as reasonably practicable (ALARP)? Is there any progress regarding this issue in the ongoing nuclear site licence (NSL) procedure? What could be the consequences for Wylfa Newydd NPP if Horizon fails to meet this legal requirement?

- What will be the consequences of the fact that the UK ABWR design does not meet the safety goal of practical elimination of accident sequences leading to large or early releases of radioactive substances? Is there any progress regarding this issue in the ongoing nuclear site licence (NSL) procedure? What could be the consequences for Wylfa Newydd NPP if Horizon fails to meet this important safety objective for European NPPs?
- Which of the 11 assessments findings of the ONR´s GDA step 4 assessment of Probabilistic Safety Analysis for the UK ABWR are solved already? How were they solved and, if no solution has been found yet, when should they be solved? Which recent national and international studies concerning external hazards (seismic hazard, tsunami and climate change) have to be applied to determine design basis requirements?
- Which margins against external hazards have to be implemented for the Wylfa Newydd NPP? What are the lessons learnt from the NSO earthquake for the UK ABWR design?

Recommendations

- It is recommended to re-**assess external hazards** at the Wylfa Newydd site before the detailed design process for the NPP starts. The re-assessment should be based on the latest state-of-the-art methods and take into account most current data.
- It is recommended to require the implementation of appropriate margins to external hazards in the design of the Wylfa Newydd NPP that are based on current scientific studies and data.
- Because a loss of the ultimate heat sink (LUHS) due to external hazard (e.g. biological fouling) has the potential of being a significant contributor to the UK ABWR overall risk profile, a robust reserve ultimate heat sink (RUHS) for the Wylfa Newydd should be implemented.
- It is recommended to apply the **concept of practical elimination** consistently in the safety requirements for the Wylfa Newydd NPP. Practical elimination of accident sequences has to be demonstrated with state-of-the-art probabilistic and deterministic methods, fully taking into account the corresponding publications of WENRA.
- To achieve the safety goal of new nuclear power plants consisting in the requirement that accidents leading to early or large releases have to be practically eliminated, it is necessary to also consider hazard events with frequencies below $<<10^{-4}$ if their impacts reach beyond the design basis. For ensuring compliance with the safety goals, a comprehensive Probabilistic Safety Analysis (Extended PSA) is necessary, taking into consideration all relevant internal and external events and possible accident causes.
- It is recommended to provide information in a transparent manner about the upcoming demonstration proving that the level of risk of the Wylfa Newydd NPP is as low as reasonably practicable (ALARP).
- It is recommended to include a conservative worst-case release scenario which should have been part of the EIA. A severe accident with a source term for e.g. containment failure or bypass scenario should be analysed as part of the EIA – in particular because of its relevance for impacts at greater distances.

5 ACCIDENTS WITH THIRD PARTIES INVOLVED

5.1 Treatment in the EIA Documents

The GDA documentation prepared by HGNE sets out the generic safety, environment and security cases for the UK ABWR design. The main submissions are the Generic Pre-Construction Safety Report (PCSR), the Generic Environmental Permit Application (GEP) and the Conceptual Security Arrangements (CSA). The CSA describes how the design meets the UK security requirements. (HNP 2018c, p. 5)

Land within and surrounding the Wylfa Newydd Development Area (WNDA) is predominantly agricultural, used for grazing sheep or cattle and crossed by a network of roads, rural lanes, watercourses and overhead electricity infrastructure. The local coastline is used for various recreational activities including walking, bird-watching, water sports and other beach activities. A number of public rights of way, including the Wales Coast Path and the Copper Trail (national cycle route) cross the WNDA. However, these will ultimately need to be diverted for reasons of security and safety. (HNP 2018c, p. 24)

The Environmental Statement Volume D includes a series of figures which illustrate the statutory and non-statutory sites or features of nature conservation and the historic environment. Floor plans for buildings at the Power Station Site are not included in the detailed drawings where appropriate in the interest of the operational security of the Power Station. (HNP 2018e, p. 7)

During operation, site security and a secure fenced boundary have been incorporated into the site design to ensure safe and secure operation of the facility as well as acting to deter crime. (HNP 2018d, p. 63)

5.2 Discussion

The terror threat to nuclear power plants has received considerable public attention in the last seventeen years. This attention has – for obvious reasons – focused on the hazard of the deliberate crash of a large airliner. But already before September 11, 2001, numerous acts of terrorism have taken place. However, the terrorist threat appears to be particularly grave in the early 21st century.[5]

There are numerous potential targets for terrorist attacks. Industrial plants, train stations or full sports stadiums can appear “attractive” for a terrorist group planning to kill as many people as possible in a single attack. Conducting an attack on a nuclear power plant on the other hand could be attractive for a terrorist group because of its immediate effect on power generation, its symbolic character, its double civilian/military character and the global attention it would receive. A successful attack on a nuclear power plant in one country is at the same time an at-

[5] The overall situation, which is determined by economic, military, ideological and political factors, cannot be evaluated here. But is important to note: although general attention is focused on the threat from the direction of Islamic fundamentalism right now, there are, worldwide, many different ideological positions and organizations from which potential terrorists could be recruited.

tack on all NPPs around the world. Countries with a high dependency on nuclear power could face a real dilemma.

In recent years, the rise of well-funded terrorist groups combined with the spread of civil nuclear power has placed nuclear security[6] high on the political agenda.

Nuclear power plants are vulnerable to a broad spectrum of possible pathways of attack, including attack from the ground, the air, water ways, and by insiders; as well as to a broad spectrum of possible means of attack, including bombs, aircraft, shelling, missiles, and application of explosives.

New possible means to support attacks emerge: unmanned flying objects, drones, can – such as in military application – be used for the preparation or support of terror attacks. Attention also needs to be devoted to newly emerged attack scenarios such as cyber-attacks.

The identification of terrorist threats against reactors and spent fuel pools is a necessary part of security planning at all nuclear power plants. There is also a pressing need to more systematically identify potential cyber, insider, and asymmetric[7] security threats. More formalized processes for identifying and analyzing threats – for example probabilistic risk assessment (PRA) – could help to improve security at nuclear power plants. (NAS 2016)

Terror attacks against Wylfa Newydd NPP

Terrorist attacks or acts of sabotage on the Wylfa Newydd may have significant impacts. However, in the Environmental Statement malicious acts of third parties against Wylfa Newydd and their possible effects are not discussed. In comparable EIA procedures such events were addressed to some extent. (UMWELTBUNDESAMT 2018)

It is general consensus that the topic of terror attacks should not be treated publicly in a manner which would provide "useful" information to terrorists and saboteurs and/or provide them with new ideas for attack scenarios. It must be emphasized that this topic can be discussed, if this is done in an appropriately general manner. Since the consequences of a terror attack are potentially very high, and many people can be affected, people have a right to be informed about these risks. To help deciding to which extent the topic can be discussed in public, the "Criterion of the Technically Competent Attacker Group" can be applied (HIRSCH 2005): it does not appear problematic to openly discuss information which any group of attackers which is sufficiently competent to be able to plan and execute an attack with some likelihood of "success" possesses anyway, or can acquire with minimal research effort. Indeed, it would serve no purpose whatsoever to attempt to keep such information secret.

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[6] Nuclear security' refers to the prevention of malicious acts involving nuclear or other radioactive materials and their associated facilities. It is typically used in the context of preventing terrorist groups from perpetrating hostile acts. Nuclear security is distinct from non-proliferation (preventing the spread of nuclear weapons to more countries).

[7] The term *asymmetry* refers to dissimilarities in the capabilities, strategies, and/or tactics between an adversary and a defending force, for example, a terrorist cell intent on attacking a nuclear plant and that plant's security forces.

Information was provided for example that the UK ABWR will be designed to withstand a commercial airplane crash, but without mentioning the relevant airplane category.

It should be noted that, through an effective structural protection, which usually can also be shown publicly, a higher level of protection is achieved as by a non-disclosure of the technical, administrative and personnel protection measures.

A terror attack against the spent fuel of the Wylfa Newydd NPP is of particular concern. If a severe accident occurs in the spent fuel pool, radioactivity would be released directly to the reactor building and from there to the environment. As a result, the effects of the release could be significant, although the environmental consequences would be less severe than for a beyond design basis or severe reactor accident due to the longer decay period of the fuel.

In connection with the construction of the Wylfa Newydd NPP also a potential terrorist attack on the interim storage facility for spent fuel must be considered. For the selection of the technological storage variant the protection against possible terrorist attacks should be considered.

Conceptual Security Arrangements (CSA)

ONR review of the CSA provides some information about the security issues of the UK ABWR.

Hitachi-GE has submitted its CSA as the principal document outlining its claims, arguments and evidence for the security of the UK ABWR to operate within Great Britain. The CSA presents the overarching security position for the closeout of the GDA process.

ONR stated to be satisfied with the claims, arguments and evidence laid down within the CSA. From a security view point, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future development and approval of site-specific security arrangements.

Three assessment findings were identified; these are for the future licensee to consider and take forward in th eir nuclear site security plan. According to ONR, these findings do not undermine the generic security submission but will require licensee input/decision. (ONR 2017c, p. 23)

However, the following three assessment findings touch upon important topics:

- protection of Vital Areas against sabotage (see AF-ABWR-SEC-01),
- protection against cyber-attacks (see AF-ABWR-SEC-02),
- provision of back-up power to the security infrastructure (see AF-ABWR-SEC-03).

5.3 Conclusions, questions and recommendations

Terrorist attacks and acts of sabotage can have significant impacts on nuclear facilities and cause severe accidents – also on the planned Wylfa Newydd NPP. Although precautions against sabotage and terror attacks cannot be discussed in detail in public in the EIA process for reasons of confidentiality, the necessary legal requirements should be set out in the EIA documents. Information was provided for example that the UK ABWR will be designed to withstand a commercial airplane crash, but without mentioning the relevant airplane category.

Information regarding the issue of terror attacks would be of great interest to the Austrian side, considering the large consequences of potential attacks.

Questions

- What are the requirements with respect to the planned NPP design against the deliberate crash of a commercial aircraft?
- Does the UK ABWR fulfil those requirements based on the present state of knowledge (not only relying on the data of the supplier but on the assessment of ONR)?
- Against what potential terrorist attacks must the new interim storage for spent fuel be designed to fulfil the legal requirements?

(Preliminary) recommendation

- Concerning the **protection** of the Wylfa Newydd NPP **against aircraft crash** it is recommended that the NPP should be designed in a way that vital safety functions can be fulfilled despite of the thermal and mechanical impacts corresponding to the assumed crash of passenger aircrafts of the largest class (Airbus A-380) and fast military jets.

6 TRANSBOUNDARY EFFECTS

6.1 Treatment in the EIA documents

Appendix B1-1 of the Environmental Statement (ES) deals with the possible transboundary effects of the Wylfa Newydd project. It provides an overview of the requirements relating to the assessment of the transboundary environmental effects of the Wylfa Newydd Development Consent Order (DCO) Project with respect to Environmental Impact Assessment (EIA) and Habitats Regulations Assessment (HRA). The appendix outlines the legislative context of the transboundary EIA and Horizon's approach to the assessments as well as the conclusions of those assessments.

The information of the appendix is based on chapters C2 (waste and materials management), C6 (traffic and transport) and chapters D3 to D15 (socio-economics, public access and recreation, air quality, noise and vibration, soils and geology, surface water and groundwater, terrestrial and freshwater ecology, landscape and visual, cultural heritage, coastal processes and geomorphology, the marine environment, radiological effects and shipping and navigation) of the ES. (HNP 2018b, p. 1)

Regarding radiological effects, the appendix refers to an impact assessment of radioactive releases caused by accidents in section 6 of appendix D14-2 (Analysis of accidental releases). The results are described as follows: For loss of coolant accidents, fuel handling accidents, and off-gas system failures, the radiation doses for the local population were assessed as being negligible. Severe accident impacts were assessed as being of low significance for local populations. Doses in the nearest Member State are two to three orders of magnitude lower than this, with the resulting impact and significance assessed as being negligible. Assuming an inverse square relationship between air concentration, dose and distance from the Power Station, impacts at greater distance would also be even lower. Radioactive releases from accidents will therefore have no significant transboundary effects. (HNP 2018b, p. 14)

The Environmental Statement concluded that no significant transboundary effects have been identified. (HNP 2018b, p. 18)

6.2 Discussion

Severe accidents at the Wylfa Newydd with considerable Caesium-137 releases cannot be excluded, although their calculated probability is below 1E-7/a. There is no reason why such accidents should not be addressed in the Environmental Statement (ES). Quite to the contrary, it would appear rather evident that they should be included in the assessment since their effects can be widespread and long-lasting and Austria can be affected. Concerning safety and accident analysis, Austria should assess a possible future impact on its territory caused by radioactive releases from accidents at the Wylfa Newydd NPP and develop a catalogue of countermeasures.

In the Environmental Statement, a severe accident with a release of Caesium-137 of 1.86E+08 Becquerel (Bq) was analysed).

Such a release of Cs-137 is very low compared to the releases other EIA procedures mentioned for severe accidents: In the EIA for the planned Dukovany NPP (Czech Republic), the assumption of the maximal release of Cs-137 for a severe accident was 3.0E+13 (30 TBq). (UMWELTBUNDESAMT 2018) The EIA procedure for the Hanhikivi NPP (Finland) calculated possible transboundary effects of Cs-137 release of 1.0E+14 TBq. (UMWELTBUNDESAMT 2014)

As discussed in chapter 4, the choice of the representative severe accident is not justified. A core-melt accident with containment failure or by-pass, resulting in the release of huge amounts of radioactive material in the environment, cannot be excluded. Thus, the analysis of the possible transboundary effects is presented in the following chapter.

Possible source terms

Data on possible UK ABWR inventories are not available. However, based on the thermal power of the nuclear power plants, the ESBWR core inventory can be used and scaled down. The core inventory of the ESBWR was based on 4,590 MWt power. The UK ABWR has a thermal power level of 3,926 MWt and thus the ESBWR inventory was multiplied by a factor of about 0.86. (SHOLLY 2014) The Cs-137 inventory of the UK ABWR can therefore be assessed as 504 PBq (5.04E+17 Bq).

Twenty-three release categories are defined considering the combination of plant damage states (PDS) groups and end states in Level 2 PSA in Generic PCSR of the UK ABWR by HITACHI-GE (2017). The following table lists the calculated Cs-137 releases for these accident sequences.

Despite the calculated frequency being very low, large radioactive releases are possible. Note: As described in the previous chapter, the calculated frequencies are not fully confirmed yet.

Table 7: Calculated Cs-137 releases for a severe accident of the UK ABWR (source: based on HITACHI-GE 2017)

	Release Category description	Source term Cs-137 (Bq)	Frequency (/y)
1	Containment Leakage	4,69E+08	3,84E-08
2	Containment Venting	2,22E+10	1,05E-09
3	Filtered Containment Venting	2,22E+10	1,31E-07
4	Early Containment Failure	3,17E+17	1,54E-08
5-1	Late Containment Failure	3,88E+14	4,39E-09
5-2	Late Containment Failure	6,05E+16	6,33E-11
5-3	Late Containment Failure	3,38E+17	3,47E-10
5-4	Late Containment Failure	1,21E+16	3,93E-09
6	Late Containment Failure with PCV spray	2,97E+14	4,21E-09
7-1	In-vessel Fuel-Coolant Interaction	3,17E+16	5,41E-12
7-2	In-vessel Fuel-Coolant Interaction	1,76E+17	4,22E-13
8-1	Ex-vessel Fuel-Coolant Interaction	2,27E+15	2,51E-10
8-2	Ex-vessel Fuel-Coolant Interaction	7,56E+16	1,50E-11

9	Direct Containment Heating	1,06E+17	2,41E-11
10-1	PCV Isolation Failure	1,16E+15	2,28E-10
10-2	PCV Isolation Failure	1,86E+17	8,06E-11
11-1	Molten Core Concrete Interaction	7,05E+15	1,91E-09
11-2	Molten Core Concrete Interaction	9,57E+16	1,77E-11
12	RPV Rupture	1,91E+17	1,00E-08
13	Containment Bypass	4,53E+17	1,85E-08
14	S/P Bypass	9,07E+16	1,75E-10
15	Direct Debris Interaction	4,41E+15	2,70E-09
16	Long Term SBO	2,47E+17	1,58E-09

6.2.1 Analysis of Transboundary Effects

For the assessment of possible impacts of transboundary emissions of Wylfa Newydd, flexRISK project calculations are used (FLEXRISK 2012). The flexRISK project modelled the geographical distribution of severe accident risks arising from nuclear facilities, in particular nuclear power plants in Europe. Using source terms and accident frequencies as input, the large-scale dispersion of radionuclides in the atmosphere was simulated for about 1,000 meteorological situations.

For each reactor, an accident scenario with a large release of nuclear material was selected. To determine the possible radioactive release for the chosen accident scenarios, the specific known characteristics of each NPP were taken into consideration. The accident scenarios for the dispersion calculation are core melt accidents and containment bypass or containment failure; the release rates are in the range of 20% to 65% of the core inventory of Caesium. The dispersion of radioactive clouds as a consequence of serious accidents in nuclear facilities in Europe and neighbouring countries is calculated for selected accidents with varying weather conditions.

Using the Lagrangian particle dispersion model FLEXPART, both radionuclide concentrations in the air and their deposition on the ground were calculated and visualised in graphs. The total Caesium-137 deposition per square-meter is used as the contamination indicator. For a severe accident at the Wylfa NPP site, a Caesium-137 release of 61.5 PBq is assumed. This source term is comparable with source terms of the UK ABWR calculated in the PSA 2.

For each NPP including the old Wylfa NPPs a release scenario has been evaluated for 88 weather situations in 1995. An evaluation of these results shows that a radioactive release of about one third of these 88 weather scenarios could result in a contamination of Austrian territory.

Figure 2 illustrates the calculated Caesium-137 depositions after a possible severe accident at the Wylfa NPP site.

Caesium-137 depositions after a severe accident

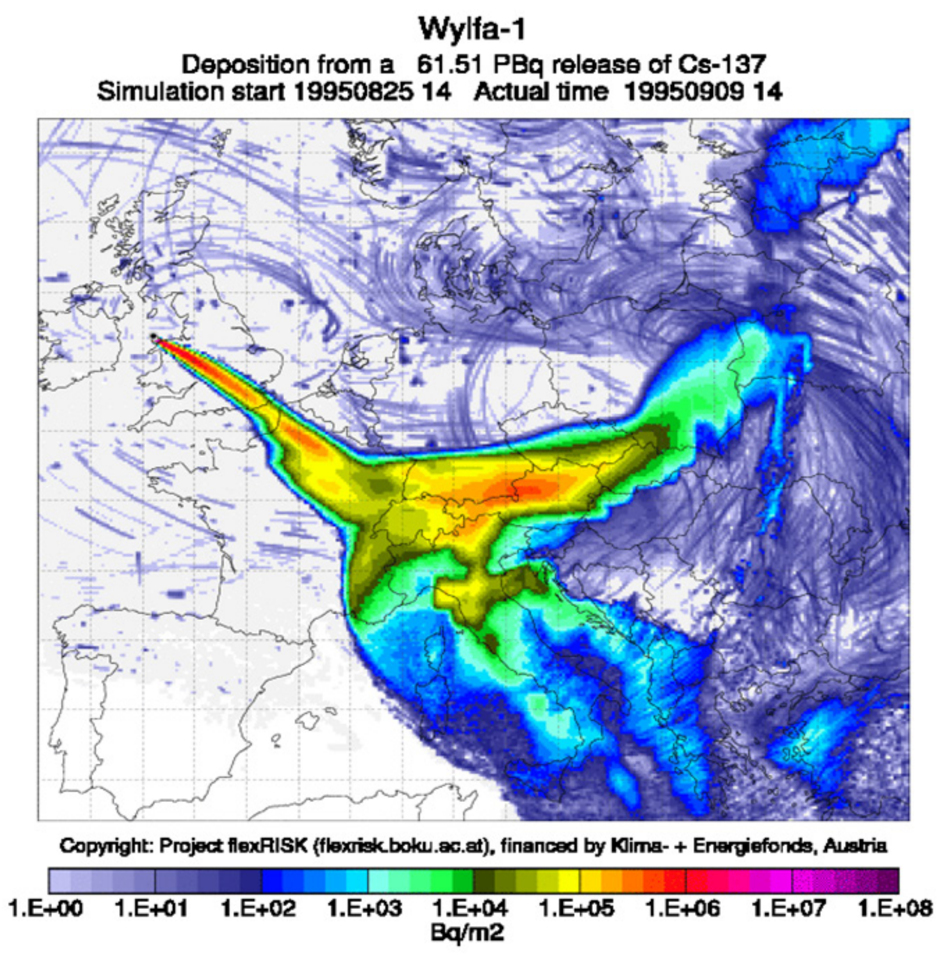

Figure 2: Caesium-137 depositions after a severe accident at the Wylfa NPP site

Quelle: FLEXRISK 2012

A considerable contamination of the Austrian territory would result from a potential Caesium-137 release of 61.5 PBq (6.15E+16 Bq) at the Wylfa NPP site under conditions comparable to those on 25 August 1995. Almost all regions in Austria would receive depositions of more than 1,000 Bq/m² (1E+03 Bq/m²). In large areas the values are above 1E+05 Bq/m², even up to 6E+05 Bq/m².

If the contamination of ground (and air) beyond certain thresholds can be expected, a set of agricultural intervention measures is triggered. These measures include earlier harvesting, closing of greenhouses and covering of plants, putting livestock in stables etc. For these measures, Austrian authorities defined a threshold for Caesium-137 ground deposition of 650 Bq/m² (BMLFUW 2014). Preparing those agricultural measures is quite complex and takes time. Responses are particularly difficult if there is only very limited time between the onset of an accident and the arrival of the first radioactive clouds. For the calculated scenario, ground depositions of all areas are higher than this threshold, i.e. Austria would be severely affected.

It is important, however, to keep in mind that accidents with much higher releases cannot be excluded. Other accident scenarios can lead to releases of more than 50% of the Caesium core inventory.

According to the PSA 2 results of the UK ABWR, a possible severe accident core melt accident with a containment bypass could result in a release of about 450 PBq (4.5E+17 Bq), 7 times more than the release assumed for the old Wylfa NPP.

Figure 2 shows that Austria and many other countries (including France, Germany and Switzerland) could be affected by a severe accident occurring at the Wylfa NPP site.

6.3 Conclusions, questions and recommendations

The results of the analysis of transboundary effects of a potential severe accident at the Wylfa Newydd site indicate that an impact on Central Europe (including Austria) cannot be excluded. The results also indicate the need for intervention measures in Austria.

Moreover, the results emphasise the importance of a serious evaluation and discussion of the severe accident scenarios for the Wylfa Newydd in the framework of the transboundary EIA.

The information the EIA procedure provided so far does not permit a meaningful assessment of the effects that conceivable accidents at the Wylfa Newydd NPP could have on Austrian territory. The analysis of a severe accident scenario would close this gap and allow for a discussion of the possible impacts on Austria. This should be taken into consideration before granting further permissions.

(Preliminary) recommendation

- Because the source term used in the accident analysis of the ES does not reflect a severe accident, it is recommended to calculate the consequences of a **severe accident with a large release** since the effects of severe accidents can be wide-spread and long-lasting and even countries in Central Europe, like Austria, can be affected.

7 QUESTIONS AND RECOMMENDATIONS

7.1 Description of the Project

Question

- In which way will the solutions to the GDA assessments findings be published?

(Preliminary) Recommendations

- Site-specific aspects, which are being evaluated during the ongoing nuclear site licence (NSL) application, should be included in the EIA documents. Site-specific factors that could endanger the safety of the Wylfa Newydd NPP are of particular concern when evaluating the possible risks for Austria.
- It is recommended to inform about the solutions of assessments findings in an appropriate manner.

7.2 Reactor Type

Question

- Which of the 11 assessments findings of the ONR´s GDA step 4 assessment of Severe Accidents for the UK ABWR have already been solved? How were they solved and if not, when will a solution be found for those?

7.3 Accident analysis

Questions

- What will be the response to the fact that the UK ABWR design does not meet the SAP BSO of target 9? Is there any progress regarding this issue in the ongoing nuclear site licence (NSL) procedure? What could be the consequences for Wylfa Newydd NPP if Horizon fails to meet this safety objective?
- What will be the consequences of the fact that the UK ABWR design does not meet the UK legal requirements for new reactor designs by demonstrating that the level of risk is as low as reasonably practicable (ALARP)? Is there any progress regarding this issue in the ongoing nuclear site licence (NSL) procedure? What could be the consequences for Wylfa Newydd NPP if Horizon fails to meet this legal requirement?
- What will be the consequences of the fact that the UK ABWR design does not meet the safety goal of practical elimination of accident sequences leading to large or early releases of radioactive substances? Is there any progress regarding this issue in the ongoing nuclear site licence (NSL) procedure? What could be the consequences for Wylfa Newydd NPP if Horizon fails to meet this important safety objective for European NPPs?

- Which of the 11 assessments findings of the ONR´s GDA step 4 assessment of Probabilistic Safety Analysis for the UK ABWR are solved already? How were they solved and, if no solution has been found yet, when should they be solved?
- Which recent national and international studies concerning external hazards (seismic hazard, tsunami and climate change) have to be applied to determine design basis requirements?
- Which margins against external hazards have to be implemented for the Wylfa Newydd NPP? What are the lessons learnt from the NSO earthquake for the UK ABWR design?

Recommendations

- It is recommended to re-**assess external hazards** at the Wylfa Newydd site before the design process for the NPP starts. The re-assessment should be based on the latest state-of-the-art methods and take into account most current data.
- It is recommended to require the implementation of appropriate margins to external hazards in the design of the Wylfa Newydd NPP that are based on current scientific studies and data.
- Because a loss of the ultimate heat sink (LUHS) due to external hazard (e.g. biological fouling) has the potential of being a significant contributor to the UK ABWR overall risk profile, a robust reserve ultimate heat sink (RUHS) for the Wylfa Newydd should be implemented.
- It is recommended to apply the **concept of practical elimination** consistently in the safety requirements for the Wylfa Newydd NPP. Practical elimination of accident sequences has to be demonstrated with state-of-the-art probabilistic and deterministic methods, fully taking into account the corresponding publications of WENRA.
- To achieve the safety goal of new nuclear power plants consisting in the requirement that accidents leading to early or large releases have to be practically eliminated, it is necessary to also consider hazard events with frequencies below $<<10^{-4}$ if their impacts reach beyond the design basis. For ensuring compliance with the safety goals, a comprehensive Probabilistic Safety Analysis (Extended PSA) is necessary, taking into consideration all relevant internal and external events and possible accident causes.
- It is recommended to provide information in a transparent manner about the upcoming demonstration proving that the level of risk of the Wylfa Newydd NPP is as low as reasonably practicable (ALARP).
- It is recommended to include a conservative worst-case release scenario which should have been part of the EIA. A severe accident with a source term for e.g. containment failure or bypass scenario should be analysed as part of the EIA – in particular because of its relevance for impacts at greater distances.

7.4 Accidents with Third Parties involved

Questions

The following questions on possible terrorist attacks and acts of sabotage should be addressed in the EIA:

- What are the requirements with respect to the planned NPP design against the deliberate crash of a commercial aircraft?
- Does the UK ABWR fulfil those requirements based on the present state of knowledge (not only relying on the data of the supplier but on the assessment of ONR)?
- Against what potential terrorist attacks must the new interim storage for spent fuel be designed to fulfil the legal requirements?

(Preliminary) recommendation

- Concerning the **protection** of the Wylfa Newydd NPP **against aircraft crash** it is recommended that the NPP should be designed in a way that vital safety functions can be fulfilled despite of the thermal and mechanical impacts corresponding to the assumed crash of passenger aircrafts of the largest class (Airbus A-380) and fast military jets.

7.5 Transboundary Effects

(Preliminary) recommendation

- Because the source term used in the accident analysis of the ES does not reflect a severe accident, it is recommended to calculate the consequences of a **severe accident with a large release** since the effects of severe accidents can be wide-spread and long-lasting and even countries in Central Europe, like Austria, can be affected.

8 REFERENCES

BECKER, O. (2015): Critical review on the stress tests of the Lungmen nuclear power plant (NPP4); Study commissioned by Greenpeace East Asia; Oda Becker with Patricia Lorenz; Hannover, August 2013.

BMLFUW – Bundesministerium für Land- und Forstwirtschaft, Umwelt und Wasserwirtschaft (2014): Maßnahmenkatalog für radiologische Notstandssituationen. Arbeitsunterlage für das behördliche Notfallmanagement auf Bundesebene gemäß Interventionsverordnung, Wien, Juli 2014.

ENSIA (2018): What are coastal Nuclear Power Plants doing to address climate threat? August 8, 2018; https://ensia.com/features/coastal-nuclear/.

ENSREG – European Nuclear Safety Regulators Group (2012): United Kingdom, Peer review country report, Stress tests performed on European nuclear power plants.

ENSREG – European Nuclear Safety Regulators Group (2012b): Compilation of recommendations and suggestions; Peer review of stress tests performed on European nuclear power plants; 27 July 2012; http://www.ensreg.eu/sites/default/files/Compilation%20of%20Recommendationsl_0.pdf.

FLEXRISK (2012): The Project „flexRISK“: Flexible Tools for Assessment of Nuclear Risk in Europe; http://flexrisk.boku.ac.at/en/projekt.html.

HIRSCH, H.; Becker, O. et al. (2005): Nuclear Reactor Hazards, Ongoing Dangers of Operating Nuclear Technology in the 21th Century; Report prepared for Greenpeace International; April 2005.

HITACHI-GE (2017): UK ABWR Generic Design Assessment, Generic PCSR Chapter 25: Probabilistic Safety Assessment, Revision C; 2017.

HNP – Horizon Nuclear Power (2018a): Wylfa Newydd Project. 6.4.98 ES Volume D – WNDA Development App D14-2 – Analysis of accidental releases, June 2018.

HNP – Horizon Nuclear Power (2018b): Wylfa Newydd Project 6.2.17 ES Volume B – Introduction to the environmental assessments App B1-1 -Transboundary Effects Assessment, June 2018.

HNP – Horizon Nuclear Power (2018c): Wylfa Newydd Project – Radioactive Substances Regulation – Environmental Permit Application; 2018.

HNP – Horizon Nuclear Power (2018d): Wylfa Newydd Project – 6.11 Environmental Statement, Non-Technical Summary; June 2018.

HNP – Horizon Nuclear Power (2018e): Wylfa Newydd Project – 1.3 Guide to the Application; June 2018.

HNP – Horizon Nuclear Power (2018f): 6.1.1 ES Volume A – Introduction to the project and approach to the EIA A1 – Introduction; June 2018.

HNP – Horizon Nuclear Power (2018g): 6.2.1 ES Volume B – Introduction to the environmental assessments B1 – Introduction to the assessment process; June 2018.

JONES, J.A. (1981a): A procedure to include deposition in the model for short and medium term atmospheric dispersion of radionuclides. The Second Report of a Working Group on Atmospheric Dispersion. NRPB-R122. National Radiological Protection Board: Chilton.

JONES, J.A. (1981b): Model for Long Range Atmospheric Dispersion of Radionuclides Released Over a Short Period, The Fourth Report of a Working Group on Atmospheric Dispersion. NRPB-R124. National Radiological Protection Board: Chilton.

NAS - National Academy of Sciences (2016): Lessons Learned from the Fukushima Nuclear Accident for Improving safety and Security of U.S. Nuclear Plants, Phase 2; Committee on Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Plants; 2016. https://www.nap.edu/catalog/21874/lessons-learned-from-the-fukushima-nuclear-accident-for-improving-safety-and-security-of-us-nuclear-plants.

ONR – Office for Nuclear Regulation (2014): Safety Assessment Principles for Nuclear Facilities. 2014 Edition Revision 0., November 2014. http://www.onr.org.uk/saps/saps2014.pdf.

ONR – Office for Nuclear Regulation (2017a): Step 4 Assessment of Severe Accidents for the UK Advanced Boiling Water Reactor.

ONR – Office for Nuclear Regulation (2017b): Step 4 Assessment of Probabilistic Safety Analysis for the UK Advanced Boiling Water Reactor.

ONR – Office for Nuclear Regulation (2017c): Step 4 Assessment of Conceptual Security Arrangements for the UK Advanced Boiling Water Reactor; Assessment Report: ONR-NR-AR-17-026; Revision 0, December 2017.

SHOLLY, S.; MÜLLNER, N.; ARNOLD, N. & GUFLER, K. (2014): Source terms for potential NPPs at the Lubiatowo site, Poland. Report prepared for Greenpeace Germany, Institut für Sicherheits- und Risikowissenschaften, BOKU Wien.

SMITH, J.G., SIMMONDS, J.R. (2009): The Methodology for Assessing the Radiological Consequences of Routine Releases of Radionuclides to the Environment Used in PC-CREAM 08. HPA-RPD-058.; 2009.

THOMAS, S. (2018): The failings of the Advanced Boiling Water Reactor (ABWR) proposed for Wylfa Nuclear Power Station. Greenpeace, May 2018.

UKCP – CLIMATE PROJECTIONS SCIENCE REPORT: CLIMATE CHANGE PROJECTIONS (2009). Murphy, J.M., Sexton, D.M.H., Jenkins, G.J., Booth, B., Brown, C.C., Clark, R.T., Collins, M., Harris, G.R., Kendon, E.J., Betts, R.A., Brown, S.J., Humphrey, K.A., McCarthy, M.P., McDonald, R.E., Stephens, A., Wallace, R., Wilby, R. and Wood, R.A. 2009. Exeter: Met Office Hadley Centre.

UMWELTBUNDESAMT (2014b): Becker, O.; Hirsch, H.; Indradiningrat, A.Y.; Wallner, A.: NPP Fennovoima (HANHIKIVI 1); Experts Statement to the Environmental Impact Assessment Report; Reports, Bd. REP-0479, Umweltbundesamt, Vienna.

UMWELTBUNDESAMT (2018): Becker, O.; Brettner, M.; Decker, K.; Indradiningrat, A.Y & Mraz, G.: Neues KKW am Standort Dukovany. Fachstellungnahme zur Umweltverträglichkeitsprüfung. Erstellt im Auftrag des Bundesministeriums für Nachhaltigkeit und Tourismus, Abteilung I/6 Allgemeine Koordination von Nuklearangelegenheiten, BMLFUW.1.1.2/0010-I/6/2015. REP-0639, Wien 2018. www.umweltbundesamt.at/fileadmin/site/publikationen/REP0639.pdf.

WENRA – WESTERN EUROPEAN NUCLEAR REGULATORS' ASSOCIATION (2010): Statement on Safety Objectives for New Nuclear Power Plants, November 2010, http://www.wenra.org/media/filer_public/2012/11/05/wenra_statementonsafetyobjectivesfornewnuclearpowerplants_nov2010.pdf.

WNA – WORLD NUCLEAR ASSOCIATION (2018): Advanced Nuclear Power Reactors, (updated August 2018); http://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/advanced-nuclear-power-reactors.aspx.

WNN – WORLD NUCLEAR NEWS (2016): Toshiba withdraws ABWR certification application; 01 JULY 2016; http://www.world-nuclear-news.org/Articles/Toshiba-withdraws-ABWR-certification-application.

WNN – WORLD NUCLEAR NEWS (2018): Safety review sought for new Japanese reactor, 10 August 2018, http://www.world-nuclear-news.org/Articles/Safety-review-sought-for-new-Japanese-reactor.

9 ABBREVIATIONS

ABWR .... Advanced Boiling Water Reactor
ADS...................... Automatic Depressurisation System
AHEF.................... Alternative Heat Exchange Facility
ALARP .... As Low As Reasonably Practicable
B/B .... Backup Building
BBG...................... Backup Building Generator
BDBA .... Beyond Design Basis Analysis
Bq......................... Becquerel
BSL .... Basic Safety Level
BSO...................... Basic Safety Objective
BWR..................... Boiling Water Reactor
C&I .... Control & Instrumentation
CCF...................... Common Cause Failures
COPS .... Containment Overpressure Protection System
Cs-134, Cs-137 .... Caesium-134, Caesium-137
CSA...................... Conceptual Security Arrangements
CST .... Condensate Storage Tank
DAC...................... Design Acceptance Confirmation
DBA...................... Design Basis Accidents
DBF .... Design Basis Flood
DCH .... Direct Containment Heating
DCO .... Development Consent Order
DW .... Drywell
EA .... Environment Agency
ECCS .... Emergency Core Cooling System
EDG .... Emergency Diesel Generator
EIA .... Environmental Impact Assessment
ENSREG .... European Nuclear Safety Regulators Group
ES .... Environmental Statement
ESBWR .... Economic Simplified Boiling Water Reactor
ESL .... Extreme Sea Levels
FCI .... Fuel-Coolant Interaction
FCVS.................... Filtered Containment Venting System
FHA...................... Fuel Handling Accident
FLSR.................... Flooder System of Reactor Building
FLSS .... Flooder System of Specific Safety Facility
GDA .... Generic Design Assessment
GDF...................... Geological Disposal Facility
GEP...................... Generic Environmental Permit Application
HFE...................... Human Failure Events

HGNE .... Hitachi-GE Nuclear Energy Limited
HPME .... High Pressure Melt Ejection
HPCF .... High Pressure Core Flooder
HRA .... Habitats Regulations Assessment
HRA .... Human Reliability Analysis
HVAC .... Heating Ventilation and Air Conditioning
I-131, I-133 .... Iodine-131, Iodine-133
IAEA .... International Atomic Energy Agency
IE .... Initiating Event
IPCC .... Intergovernmental Panel on Climate Change
IVR .... In-Vessel Retention
LCO .... Limiting Conditions of Operation
LDF .... Lower Drywell Flooder
LDW .... Lower Drywell
LOCA .... Loss of Coolant Accident
LOOP .... Loss of Off-site Power
LPFL .... Low Pressure Core Flooder
L(E)RF .... Large (Early) Release Frequency
LUHS .... Loss (of) Ultimate Heat Sink
MAAP .... Modular Accident Analysis Program
MCCI .... Molten Core Concrete Interactions
MCR .... Main Control Room
mSv .... Milli-Sievert
NCO .... Niigata Chuetsu offshore
NOAA .... National Oceanic and Atmospheric Administration
NPP .... Nuclear Power Plant
NRW .... Natural Resources Wales
NSIP .... Nationally Significant Infrastructure Project
OECD-NEA .... Organisation for Economic Co-operation and Development Nuclear
OGF .... Off-gas system failure
ONR .... Office for Nuclear Regulation
PAR .... Passive Autocatalytic Recombiner
PCSR .... Pre-construction Safety Report
PCV .... Primary Containment Vessel
Pd .... Design Pressure
POS .... Plant Operating State
PSA .... Probabilistic Safety Assessment
R/B .... Reactor Building
RCCV .... Reinforced Concrete Containment Vessel
RCIC .... Reactor Core Isolation Cooling
RCW .... Reactor Building Cooling Water System
RDCF .... Remote Depressurisation Control Facility

RHR .... Residual Heat Removal
RI.......................... Regulatory Issue
RO........................ Regulatory Observation
RPV...................... Reactor Pressure Vessel
RQ........................ Regulatory Query
RSW..................... Reactor Building Service Water
RUHS .... Reserve Ultimate Heat Sink
SA C&I .... Severe Accident Control & Instrumentation
SA .... Severe Accident
SAMG................... Severe Accident Management Guideline
SAPs .... Safety Assessment Principles
SBO...................... Station Blackout
SFSF .... Spent Fuel Storage Facility
SFP .... Spent Fuel Pool
SGTS .... Standby Gas Treatment System
SoDA.................... Statement of Design Acceptability
S/P .... Suppression Pool
SPSA.................... Seismic PSA
SRV...................... Safety Relief Valve
SSC...................... System, Structure (and) Component
TSC .... Technical Support Contractor
UDW..................... Upper Drywell
UHS...................... Ultimate Heat Sink
UK ABWR............. United Kingdom Advanced Boiling Water Reactor
UK .... United Kingdom
US NRC................ United States (of America) Nuclear Regulatory Commission
V/B .... Vacuum Breaker
WENRA................ Western European Nuclear Regulators' Association
WNDA .... Wylfa Newydd Development Area
WW .... Wetwell

10 ANNEX

Step 4 Assessment of Probabilistic Safety Analysis for the UK ABWR

Summary of the Assessment Findings

AF-UKABWR-PSA-001	The licensee shall: 1. Develop processes and procedures to ensure that the PSA is kept living and is aligned with the design reference. Implementation of this process should ensure that differences between the PSA and the final GDA design reference are adequately addressed. 2. Develop an overall programme which ensures that the shortfalls and future PSA development needs presented in this assessment report (summarised in Annex 7) are included in the plans for the site-specific PSA, such that risk insights are able to be identified and utilised to inform associated design and operational decision making. 3. Develop processes and procedures to ensure the PSA assumptions are captured in future design, construction and procedure development. This process should also ensure that the PSA model and documentation is updated to reflect any changes to assumptions as more detailed information becomes available.
AF-UKABWR-PSA-002	The licensee shall ensure that the basis for the modelling and assumptions concerning outage, maintenance and test unavailabilities of systems and components (including standby) used in the PSA, is justified and aligned with the technical specifications and maintenance programmes, or alternative values/strategies justified.
AF-UKABWR-PSA-003	The licensee shall use the PSA to identify intersystem common cause failure effects for the UK ABWR following on from the work in GDA. The results shall be used to inform the incorporation of appropriate defences and, where appropriate, intersystem common cause failures should be included explicitly in the model.
AF-UKABWR-PSA-004	The licensee shall provide a revised systematic prioritisation of all internal hazards, including combined internal hazards, for all sources of radioactivity on-site that is representative of the site-specific design and layout and consistent with the internal hazards deterministic safety case. The prioritisation shall include demonstration that the risk associated with all the screened out internal hazards would be insignificant compared to the ABWR total risk.
AF-UKABWR-PSA-005	The licensee shall provide a revised systematic prioritisation of external hazards. The prioritisation shall consider all sources of radioactivity on-site and the specific characteristics of the site. The analysis should address external hazards that could be correlated. The licensee shall provide a demonstration that the risk associated with all the external hazards screened out would be insignificant compared to the total risk. The licensee shall then provide a revised PSA for external hazards on the basis of the prioritisation performed.
AF-UKABWR-PSA-006	The licensee shall consider loss of ultimate heat sink initiating events (including biological fouling) and external flooding initiating events within the site-specific PSA, or adequately justify their exclusion. The analysis shall take site-specific heat sink design and expected operator actions into account. The licensee shall use the analysis to identify any relevant PSA insights to aid improvement of the design or operation of the UK ABWR.

AF-UKABWR-PSA-007	The licensee shall provide revised internal fire and internal flood PSAs for shutdown and spent fuel pool operations which are consistent in detail and scope to the at power analysis. The revised PSAs shall reflect the site-specific design, operation and maintenance of the UK ABWR and take any relevant shortfalls identified by the GDA review into account.
AF-UKABWR-PSA-008	The licensee shall review the uncertainty analysis for core damage frequency and large release frequency, to identify the cause for the significant difference in the monte carlo generate mean and the point estimate results and, if appropriate, the licensee shall put in place measures to resolve the cause of the significant difference.
AF-UKABWR-PSA-009	Because of the site-specific nature of the level 3 PSA and the shortfalls identified in the GDA review, the licensee shall provide a revised level 3 PSA model and documentation, as part of the development of the site-specific PSA, which takes into consideration the following: ● Justification for the decontamination factors applied to the barriers to fission product release. This shall including those for the standby gas treatment system. ● Updating the population data to reflect the most recent census, when reasonably practical to do so. This is needed to provide a more realistic assessment of dose uptake. ● Consideration and justification for the expected increase in notional fatalities projected to the end of station life. The use of the most recent census data will assist this. ● Model multiple release phases to more realistically model spent fuel pool fault sequences, or use and justify an alternate method for comparison against SAPs Target 7. ● Revise the method for comparison to SAPs Target 9 to release frequency multiplied by conditional probability of exceeding 100 fatalities.
AF-UKABWR-PSA-010	Because of the ongoing regulatory expectation to demonstrate that the risks are being managed ALARP, the licensee shall develop and implement processes and procedures to ensure that PSA insights are systematically identified, prioritised and considered as part of design development. This shall take into account the shortfalls identified by the GDA review in Section 4.2.20 of the assessment report. These describe risk reduction options identified but intended for implementation beyond GDA, and shortfalls that when resolved may alter the identification and sentencing of ALARP options. The process shall ensure that: ● The ALARP options identified in GDA submissions for implementation or consideration beyond GDA have been adequately considered and sentenced by the licensee. This shall be done at the appropriate time to ensure the PSA insights from these options are available to risk inform the appropriate aspects of the detailed design. ● The PSA is sufficiently technically developed to support this process, with any relevant shortfalls and insights identified by ONR during GDA being considered and implemented, as appropriate. These shortfalls are identified in Section 4 of the assessment report.

AF-UKABWR-PSA-011	Because of the importance and regulatory expectation of using the PSA to risk inform design and operation of the UK ABWR, the licensee shall provide a programme to revise the PSA model ensuring that the planned development of the PSA is adequate to support the intended PSA applications at the appropriate time, including: ● Development of the detailed design, ● Demonstration of ALARP, ● Development of operating rules and technical specifications, ● Development of arrangements for examination, maintenance, inspection and testing, ● Plant configuration control, ● Development of operating and emergency procedures and severe accident management guidelines. To achieve this, the licensee is expected to programme resolution of the following PSA modelling shortfalls. These are the asymmetric modelling of systems which contain symmetrically redundant trains of equipment, the inclusion of conservatisms to simplify the modelling and various omissions in the PSA identified by the GDA review. The programme shall ensure that the developments are completed and risk insights available prior to the associated design and operational decisions being taken.

Step 4 Assessment of Severe Accidents for the UK ABWR

Summary of the Assessment Findings

AF-ABWR-SA-01	Failure of the pedestal wall has been identified by Hitachi-GE as a potential challenge to the containment in a severe accident. In GDA, Hitachi-GE has not presented detailed design calculations to justify the failure criterion for the pedestal wall when subject to molten core-concrete interaction. The licensee shall substantiate the failure criterion for the pedestal wall in severe accidents, including specific consideration of challenges to the pedestal wall structure from molten core material which may break though into the pedestal wall vent pipes.
AF-ABWR-SA-02	Hitachi-GE has assumed that the vacuum breakers would be robust against severe accident conditions, thus preventing suppression pool bypass. However, specific safety case claims or performance requirements for the vacuum breakers in severe accident conditions have not been identified in the GDA safety case documentation. The licensee shall identify the requirements placed on the vacuum breakers by the severe accident safety case and demonstrate that these can be met by the final design.
AF-ABWR-SA-03	Hitachi-GE has identified the theoretical possibility of re-criticality in a severe accident during re-flooding of the reactor pressure vessel, resulting in potential challenges to the primary containment. Hitachi-GE has presented limited analysis of the conditions which could give rise to re-criticality. To inform site-specific accident management guidelines, the licensee shall perform sufficient additional analysis to identify the range of conditions that could lead to a possible re-criticality. For the conditions which could potentially result in re-criticality, the licensee shall consider the requirements for any design provisions which could reduce the risk of re-criticality so far as is reasonably practicable.
AF-ABWR-SA-04	Ensuring the continuing integrity of the primary containment by protecting it from over-pressurisation is a vital objective for severe accident measures and management strategies. Hitachi-GE's severe accident analysis has shown that the assumed set-point for the containment overpressure protection system would not always ensure that pressure in the drywell remains below the containment ultimate failure pressure. For accident sequences where venting is claimed as an effective severe accident measure, the licensee shall optimise the containment over-pressure protection system opening set-point to ensure that containment pressures remain below the ultimate failure pressure so far as is reasonably practicable. This shall take into account containment conditions in severe accidents, including consideration of potential static and dynamic pressure differences between the drywell and wetwell.
AF-ABWR-SA-05	In the absence of detailed design information during GDA, Hitachi-GE has made assumptions about achievable flow rates in its demonstrations of the effectiveness of primary containment vessel venting in severe accidents. The licensee shall demonstrate that the final design of the filtered containment vent system can meet the safety case claims placed on it by those severe accident sequences which credit venting.

AF-ABWR-SA-06	Hitachi-GE's GDA analysis of the effectiveness of hydrogen management measures in the primary containment and reactor building has been based on provisional design information for passive autocatalytic recombiners. The analysis supports Hitachi-GE's hydrogen management strategy for design basis loss of coolant accidents and reactor severe accidents. The licensee shall update the hydrogen management safety case to reflect the design and performance characteristics of the recombiners selected in the final design, and reconfirm that the hydrogen management objectives are met.
AF-ABWR-SA-07	Hitachi-GE has identified in GDA a need to open the large equipment door in some severe accident conditions as part of the hydrogen management strategy. However the practicalities of how this will be done have not been determined due to limitations in GDA scope. The licensee shall determine the arrangements for opening of the reactor building large equipment door in accident conditions, taking appropriate steps to ensure that the risks to both the public (from a major event escalation caused by not opening the door) and workers performing crucial tasks are considered and reduced to ALARP.
AF-ABWR-SA-08	Hitachi-GE has identified several lessons and learning points from the Fukushima Dai-ichi accident that are site-specific or matters for the licensee to consider, which cannot be fully addressed in GDA. The licensee shall review relevant lessons and learning points identified as being out of GDA scope in Hitachi-GE document AE-GD-0505 Rev.2 and demonstrate that these have been addressed in the design and proposed operation of the site-specific plant.
AF-ABWR-SA-09	For the reactor building, Hitachi-GE has included the provision to connect mobile power units to support Class 1 systems. However, the Severe Accident Control & Instrumentation system is powered by the backup building electrical power system. A failure of backup building power sources is a potential way for a fault condition to escalate to a severe accident scenario, resulting in the loss of severe accident control and instrumentation functions. As part of its work to develop a final design for the backup building, the licensee shall consider whether it is ALARP to provide a capability for mobile power supply sources to be connected to the Severe Accident Control & Instrumentation system, to ensure that control and monitoring of severe accident systems can be maintained in circumstances where the fixed backup building power sources have failed.
AF-ABWR-SA-10	To meet UK and international expectations post-Fukushima, Hitachi-GE has provided a demonstration which argues that the generic UK ABWR design practically eliminates large or early releases. The extent to which hazards, and therefore the completeness of any practical elimination claim, can be considered in GDA is limited. In particular, external hazards will present an additional contribution to the site-specific risk profile. The licensee shall review and update as appropriate the deterministic and probabilistic arguments that support the claim that large or early releases have been practically limited on a site-specific basis, notably to consider the risks associated with the site-specific beyond design basis hazard profile.
AF-ABWR-SA-11	Hitachi-GE's GDA safety case documentation provides limited and variable levels of detail on the claims and performance requirements placed on structures, systems and components (SSCs) in severe accident conditions, unless the SSC's role is specifically for severe accidents. The licensee shall identify so far as is reasonably practicable the expected requirements on SSCs in severe accidents to inform detail design work and equipment qualification work, as appropriate.

Step 4 Assessment of Conceptual Security Arrangements for the

UK ABWR, Summary of the Assessment Findings

AF-ABWR-SEC-01	Modifications to plant design will require a re-evaluation of the VA status. Late design changes to some areas of the plant have been taken into account by Hitachi-GE and a conservative re-evaluation undertaken which has identified potential VAs. In addition, some VAs were identified using generic data, and conservative assumptions made. These VAs should be re-evaluated using site-specific data to confirm or otherwise VA status. The identified anomalies relating to the VAs in the CSA appendices should be reviewed and corrected.
AF-ABWR-SEC-02	The cyber analysis undertaken by Hitachi-GE used a combination of deterministic and probabilistic analyses based on the most capable of threat actors, which was considered adequate for GDA as it supports the evidence related to the overall architecture of the safety systems. A broader risk assessment covering the full range of threat actor capability will need to be adopted by the licensee once site-specific technology has been chosen and when developing the site security plan.
AF-ABWR-SEC-03	The licensee shall identify the requirement for, and provision of power to the site security systems in order to minimise the risk of power failure.

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