HORIZON
NUCLEAR POWER

WYLFA NEWYDD

ENERGY WORKING FOR BRITAIN

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1 Introduction

1.1.1 A consideration of effects from accidental releases is included within this appendix to fulfil two requirements.

- The April 2016 Scoping Opinion advises that Horizon should take into account the requirements of article 15 of the new Environmental Impact Assessment directive 2014/52/EU which states that for certain projects (because of their vulnerability to major accidents, and/or natural disasters) "*it is important to consider their vulnerability (exposure and resilience) to major accidents and/or disasters, the risk of those accidents and/or disasters occurring and the implications for the likelihood of significant adverse effects on the environment*".
- The requirements of the Espoo treaty [RD1] whereby when severe accidental impacts, however unlikely, are associated with a project, the parties will engage in consultation.

1.1.2 This appendix presents background information on the safety features of the UK Advanced Boiling Water Reactor (UK ABWR) and an assessment of the environmental effects that might result in the event of accidental release scenarios.

1.1.3 For new nuclear designs, the safety of a generic reactor design is assessed under the Generic Design Assessment (GDA) process, overseen by the Office for Nuclear Regulation (ONR) and the environment agencies. In December 2017, ONR, the Environment Agency and Natural Resources Wales granted Design Acceptance Confirmation and a Statement of Design Acceptability for the UK ABWR reactor design [RD2].

1.1.4 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 [RD3].

1.1.5 The general approach to safety design follows safety assessment, analysis and verification and is outlined in both International Atomic Energy Agency (IAEA) and ONR guidance.

1.1.6 IAEA guidance (see paragraph 3.2.4) can be summarised as follows.

- Safety assessment is an iterative and systematic process that is carried out throughout the design process to ensure that all the relevant safety requirements are met by the design of the plant.
- A safety analysis of the plant design is conducted in which methods of both deterministic and probabilistic analysis are applied. On the basis of this analysis, the design basis for items important to safety would be established and confirmed. It would also be demonstrated that the plant as designed is capable of meeting any prescribed limits for radioactive releases and acceptable limits for potential radiation doses for each category of plant states, and that defence in depth has been effected.

- The independent verification should be carried out under the responsibility of the operating organization by a team of experts who are, as far as possible, independent of the designers and those performing the safety assessment.

1.1.7 This appendix describes the following.

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

1.1.8 Detailed documentation on the various aspects of the UK ABWR safety case are available in the relevant chapters of the GDA reference material [RD4], particularly the generic pre-construction safety report (PCSR), but a summary of the information is presented here. In addition, Horizon has prepared an assessment as required by Article 37 of the Euratom treaty document, which includes consideration of doses to the local population and the nearest member state resulting from routine and accidental releases.

The Control of Major Accident Hazards assessment

1.1.9 An assessment of scenarios relevant to the Control of Major Accident Hazards (COMAH) regulations will be undertaken for the Power Station. At present the potential scenarios and relevant inventories are being compiled. An assessment will be developed as this work progresses. It is expected that the site will be managed so that the COMAH Regulations do not apply during the construction stage, and that at operation, the site will be a Lower Tier COMAH site.

2 Description of the Power Station

2.1.1 The general layout of the Power Station is shown in figure A2-1 in the Introduction to the project and approach to the EIA Figure Booklet (Application Reference Number: 6.1.11) and figure D14-2 in the WNDA Development figure booklet (Application Reference Number: 6.4.101). The buildings, structures and systems within the site are arranged into the following groupings.

- Main plant – those parts of the Power Station that enable generation of power. Two UK ABWR units would operate on a single power island, comprising:
  - Unit 1 buildings, structures and facilities;
  - Unit 2 buildings, structures and facilities; and
  - Cooling Water System (excluding common structures).
- Common plant – those parts of the Power Station that service the generation of power, including:
  - shared service building, Cooling Water System common structures; and
  - combined radioactive waste facilities and buildings for both units.
- Supporting facilities, buildings, structures and features – those parts of the Power Station that are integral to the Power Station, but would not be process related.

2.1.2 The main plant components within the Power Station Site are on the twin unit power island which includes two reactor buildings, two turbine buildings and two control buildings and a common service building.

2.1.3 Some of the Cooling Water System elements are common to both generating units, namely the cooling water intake, pumping plant, seal pit, discharge tunnel and outfall and the reserve ultimate heat sink.

2.2 Overview of UK ABWR

2.2.1 Boiling water reactors (BWR) are one of the most common types of nuclear power generating plants. BWR technology has continued to evolve and the first advanced BWR (ABWR) was built in Japan and commenced commercial operation in 1996. The ABWR design introduced features to enhance safety and reliability as well as making the plant easier to operate and maintain. Another key feature was modular design, which resulted in shorter construction times.

2.2.2 The design reference for the UK ABWR is the first ABWRs (Kashiwazaki-Kariwa units 6 and 7, plus improvements implemented at Shika unit 2, Shimane unit 3 and Ohma unit 1, in addition to incorporation of post-Fukushima enhancements).

2.2.3 The UK ABWR will incorporate further safety enhancements and additional resilience against severe external hazards. These include aircraft impact countermeasures and post-Fukushima countermeasures based on learning from that event. Specifically, the UK ABWR will include enhancements such as:

- enhanced strategies for comprehensive management of accidents;
- provision of diverse connections for mobile equipment;
- systems to provide alternative and flexible water injection for cooling the reactor and the spent fuel storage pool;
- structural enhancements for protection against aircraft impacts to primarily protect the reactor and control buildings;
- mobile equipment to provide alternative power supplies and coolant injection capability, as well as heavy machinery to maintain plant access in the event of wide-spread disruption;
- engineered measures to enable manual operation of specific isolation valves if power supplies are lost; and
- robust instrumentation that will provide credible data in the hostile environment of a severe accident.

2.2.4 In the UK ABWR, ordinary (light) water is used to remove the heat produced inside the reactor core by the thermal nuclear fission process. The water coolant boils in the reactor pressure vessel (RPV) and the resulting steam passes through a steam separator and dryer above the core and then, via the main steam lines, to the turbine generator which generates electricity. This is called the nuclear steam supply system. The steam passes through condensers where it is water-cooled and returns as feedwater to the reactor. The water coolant also acts as a moderator to enable thermal fission.

2.2.5 The reactor building is a reinforced-concrete structure with a steel frame supporting a reinforced-concrete roof. The external structure of the reactor building provides protection against aircraft impact.

2.2.6 The nuclear steam supply system is comprised of the following systems:

- RPV and internal components;
- reactor recirculation system;
- control rod drive system; and
- nuclear boiler system.

2.2.7 The RPV houses the reactor core (nuclear fuel) which is the heat source for steam generation. The vessel contains the heat, produces steam within its boundaries and serves as one of the fission product barriers during normal operation.

2.2.8 The reactor recirculation system has two main functions:

- provision of forced circulation of reactor coolant for energy transfer from the fuel to the coolant and, as a result, generates a larger amount of steam compared to passive circulation; and
- variation of reactor power by changing the recirculation flow by adjusting the reactor internal pumps speed.

2.2.9 The control rod drive system controls changes in core reactivity during power operation by movement and positioning of the neutron absorbing control rods within the core. The drive mechanism for this mode of operation is the fine motion control rod drive which uses two different power sources for shutdown: an electric motor drive for normal operation and a conventional hydraulic drive for rapid insertion, in the event of an emergency shutdown of the reactor (colloquially known as a SCRAM).

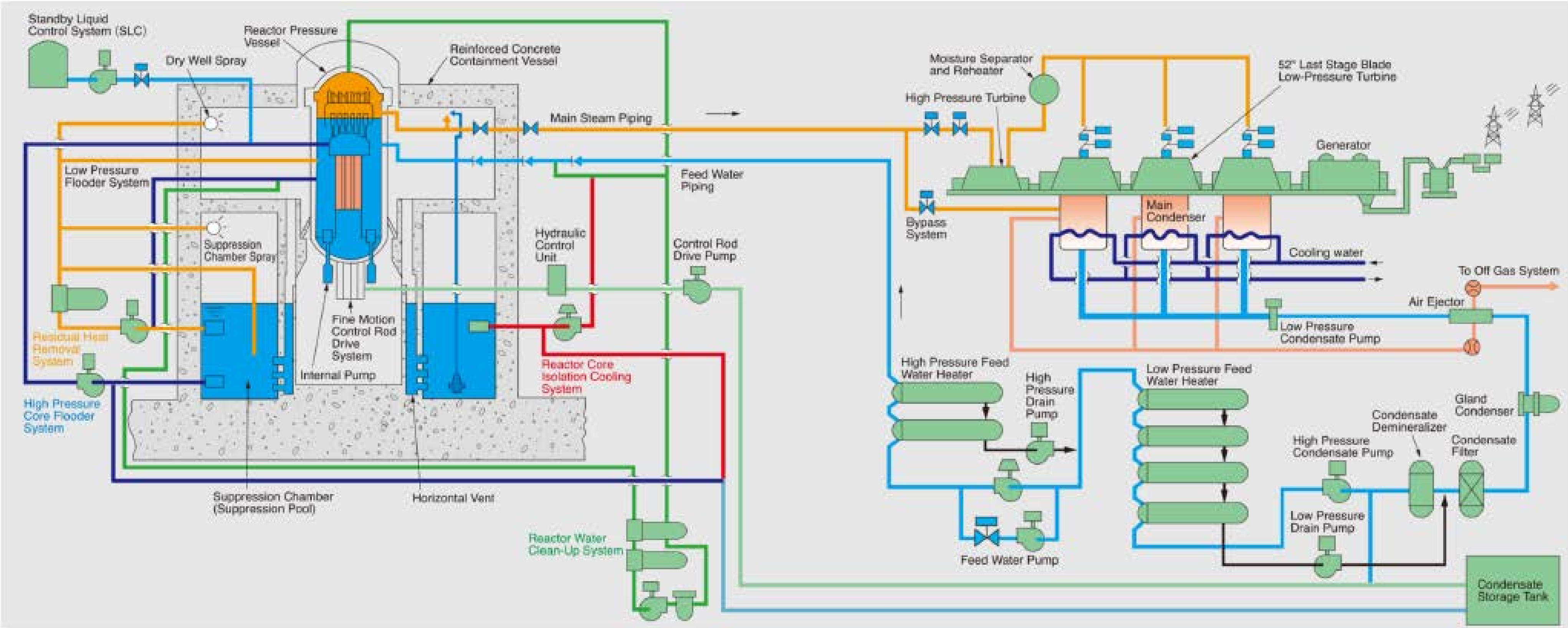
2.2.10 The nuclear boiler system is divided into two subsystems:

- the main steam system which consists of four steam lines to direct the steam flow from the RPV to the main turbine; and
- the feedwater system consists of two lines that transport the feedwater from the condensers in the turbine building back to the RPV.

2.2.11 The power conversion system is designed to produce electricity by passing the steam generated in the RPV through turbines, collect and condense this steam into water and then link back to the nuclear steam supply systems. Steam is fed first through a single high pressure turbine; it is then reheated and fed through three low pressure turbines. The exhaust steam is condensed in a single condenser, which is supplied with cooling water from the sea by three circulating water pumps; heat extracted from the condensate is removed by the cooling water and discharged to the heat sink (the sea). The condensate is then returned to the RPV via the feedwater system.

2.2.12 Figure 2-1 shows a schematic of the configuration and the main systems of the UK ABWR.

Figure 2-1 Schematic diagram showing the configuration and main systems of the UK ABWR



2.3 Safety features of the design

2.3.1 Many plant features have intrinsic safety functions during normal operation, but there are also specific safety systems which are invoked in the event of non-operational scenarios, as described in section 3.

2.3.2 Engineered safety systems comprise the reactor containment systems and the emergency core cooling system (ECCS). These are provided in order to suppress or 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 (SSC) of the reactor installation.

2.3.3 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.

2.3.4 Individual systems which are included in the ECCS and the containment systems are as follows:

- ECCS:
  - high pressure core flooder;
  - reactor core isolation cooling system;
  - automatic depressurisation system;
  - low pressure flooder system;
  - filtered containment venting system; and
  - emergency generators.
- Containment systems:
  - PCV;
  - primary containment isolation system;
  - PCV gas control system;
  - containment heat removal system;
  - drywell cooling system;
  - reactor building; and
  - standby gas treatment system.

Emergency Core Cooling system

2.3.5 The ECCS is provided to maintain cooling to the reactor and prevent fuel temperature limits being exceeded 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.

2.3.6 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.

Containment systems

2.3.7 The reactor containment systems have the function of isolating the radioactive substances generated in the reactor from release to the environment. Leakage rates are kept below a specified low level, thus minimising the amount of radioactive substances discharged into the atmosphere.

Primary containment facility

2.3.8 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.

2.3.9 In the event of a loss of coolant accident (LOCA, see section 3), 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.

Primary containment isolation system

2.3.10 The main role of the primary containment isolation system is to provide protection against the dispersion of radioactive material from the primary containment to the environment during normal operations as well as fault conditions. This is achieved by completely isolating the system pipes penetrating the primary containment and thus forming a barrier to confine the radioactive material within the primary containment boundary.

PCV gas control system

2.3.11 The PCV gas control system consists of the flammability control system and the atmospheric control system with the principal role of maintaining an inert and non-explosive atmosphere within the PCV. The systems are designed to prevent build-up of hydrogen and oxygen which could be generated within the reactor and released into the PCV in a design basis event (see section 3).

- The flammability control system is provided to control the potential build-up of hydrogen from the radiolysis of water.
- The atmospheric control system is provided to establish and maintain an inert atmosphere within the PCV except during shutdown for refuelling outages or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power.

Containment heat removal system

2.3.12 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 (see section 3).

Drywell cooling system

2.3.13 The drywell cooling system is designed for the following purposes.

- To maintain the required thermal environment and humidity so that the components in the drywell operate in a proper manner during plant normal operation and in the event of a LOOP.
- To cool the atmosphere in the drywell so that the working environment temperature during plant inspection and maintenance is acceptable for personnel access.

Secondary containment facility - reactor building

2.3.14 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.

2.3.15 During normal plant operation, the secondary containment areas are kept at a negative pressure with respect to the environment and clean zone by the heating, ventilation and air conditioning (HVAC) system. Radioactive substances (fission products, activation products) which, following an accident, may leak from the primary to secondary containment are processed and captured by the standby gas treatment system, which ensures adequate delay time before discharge to the environment. The HVAC exhaust systems and standby gas treatment system are located within the secondary containment to ensure collection of any leakage.

2.3.16 The building gaseous effluents are monitored for radioactivity. If the level of radioactivity rises above defined levels, the secondary containment discharge can be routed through the standby gas treatment system, which incorporates high efficiency particulate air filters and charcoal beds to remove radioactivity before release.

Severe accident management systems

2.3.17 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 (see section 3).

2.3.18 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 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.

Emergency generators

2.3.19 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.

2.3.20 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.

2.3.21 In order to provide the necessary capacity, resilience and reliability to meet the demands of the Power Station, the following equipment would be installed at the Power Station Site.

Emergency Diesel Generators

2.3.22 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 LOCA (see section 3) 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.

Backup Building Generators

2.3.23 The backup building will provide alternative safety management capacity during an emergency if the main control building and associated safety systems are not operational.

2.3.24 Two backup building generators and associated equipment would service each generating unit, and would be installed in a single backup building. The backup building generators are rated to supply power to the backup building equipment when off-site power is lost.

3 Accident scenarios

3.1.1 A key component of the nuclear site licensing process is the nuclear safety case. This is a set of documents that describe the radiological hazards in terms of a facility or site and modes of operation (including potential undesired modes) and the measures that prevent or mitigate against harm being incurred. The safety case should provide a coherent demonstration that relevant standards have been met and that risks to persons have been reduced to As Low As Reasonably Practicable (ALARP).

3.1.2 Plant safety is assured by several nested layers of protection. The plant is protected by safety-related SSCs. Safety functions and the SSCs that provide them are then classified according to their importance in ensuring plant safety. Deterministic and probabilistic safety assessments demonstrate that the resulting design reduces risks to ALARP.

3.1.3 The UK ABWR design aims to provide four fundamental safety functions:

- control of reactivity;
- fuel cooling;
- long-term heat removal; and
- confinement and containment of radioactive materials.

3.1.4 As part of the safety evaluation, a fault schedule is developed to analyse systems and equipment failure modes. This details the sequence of events leading to specific failures and documents the likely frequency of occurrence of the initiating events. The fault schedule will demonstrate adequate safety measures for each design and beyond design basis event:

- for each event, the schedule identifies the categories of the required UK ABWR safety functions;
- for each UK ABWR safety function, the schedule identifies the claimed SSCs and their safety classifications; and
- for each UK ABWR safety function, the schedule identifies the SSCs that are passive, initiated automatically or initiated manually.

3.1.5 For frequent faults (those with a frequency of occurrence of $>10^{-3}$ per year – see section 3.2), two diverse means of protection are required to deliver the fundamental safety functions. The SSCs which form the principal means of delivering each safety function are termed Class 1 and those providing the secondary means are at least Class 2.

3.1.6 Class 1 safety systems are provided with sufficient redundancy to ensure that even in the event of failure, the function can still be delivered. Safety systems and the systems that support them are physically separated to prevent common cause failure of all systems delivering an essential function due to an internal hazard. The systems providing this function would also be independent of each other to avoid common cause failure due to the loss of a support system.

3.2 Identification of accident scenarios

3.2.1 Based on the UK ABWR design, the fault schedule was developed from the systematic identification of initiating events, which are grouped based on similar fault sequences and demands on safety functions during the event.

3.2.2 The initiating events to be analysed were initially identified by using logic tree analysis. The scope of the logic tree analysis involved all plant operating modes and plant facilities. Abnormal states which may lead to damage to nuclear fuel or the reactor coolant pressure boundary, and potentially lead to the release of radioactive materials from the nuclear facility were identified. Next, the causes of abnormal states were identified for each abnormal state and grouped together. Finally, postulated disturbances were identified for each cause.

3.2.3 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 to establish the list of initiating events to be evaluated by probabilistic safety assessment (PSA), design basis analysis, beyond design basis analysis and severe accident analysis.

3.2.4 Bounding faults identified in the logic tree analysis were compared with IAEA Specific Safety Guides ([RD5], [RD6], [RD7]) and the US-ABWR design control document [RD8], and the completeness of bounding faults was confirmed. As a result, bounding faults for the UK ABWR are almost the same as those in the IAEA Safety Guides and the US-ABWR design control document.

3.2.5 Full details of the fault schedule for the UK ABWR can be found in the GDA PCSR [RD4] and accompanying documentation. Initiating events are grouped according to their plant impact and an indication of their frequency of occurrence is provided. The fault schedule identifies both frequent and infrequent design basis faults, and some beyond design basis faults, but does not identify severe accidents.

Probabilistic Safety Assessment

3.2.6 International best practice recommends that PSA should be used to inform the design process and help ensure the safe operation of the site and facilities [RD9]. PSA is an established technique to numerically quantify risk measures and determines what undesired scenarios can occur, with what likelihood, and what the impact could be.

3.2.7 A full-scope Level 3 PSA, which considers risks to the public from off-site releases, has been developed for the UK ABWR adopting a general approach.

- All sources of radioactivity on the Power Station Site are identified, as are all postulated initiating events which can result in a release of the radioactivity.
- The faults and hazards (both internal and external) which can result in a postulated initiating event are identified for all modes of operation.
  - Events may be consolidated into bounding events.

  - Unlike the design basis analysis (described below), the probabilistic assessment should not exclude events of low dose or low frequency.
- The preliminary PSA analyses the risks to the public and workers arising from all events. This takes into account any passive safety features and claimed safety measures, but at this time does not take benefit from any additional risk reduction measures such as systems that have an appreciable reliance on operator action.
- The annual summated risks are compared with the limits and targets for public risk.

Design Basis Analysis

3.2.8 Design basis analysis is a further method for demonstrating the adequacy of the design of the plant. The approach uses conservative assessment methodologies and/or assumptions to provide high confidence that the design will achieve its design intent.

3.2.9 The purpose of design basis analysis is to assess all the initiating faults/events identified as falling within the design basis. The ONR Safety Assessment Principles (SAPs) [RD9] define the design basis as "*the range of conditions and events that should be explicitly taken into account in the design of the facility, according to established criteria, such that the facility can withstand them without exceeding authorised limits by the planned operation of safety systems*".

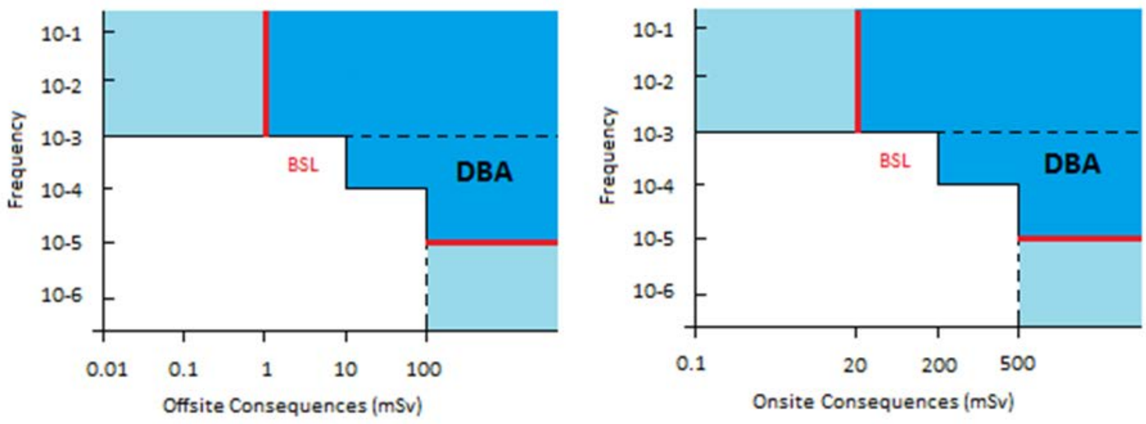
3.2.10 The fault frequency estimates are also used in combination with the unmitigated consequences to determine the further fault analysis that each initiating fault/event should be subject to, and by extension the number of diverse safety measures providing the required safety function (i.e. frequent faults recognise the possibility of common cause failures in the primary protective measures which may defeat provisions of redundancy).

3.2.11 The initiating events/faults consequences and frequencies are captured in the fault schedule. The design basis fault frequency is classified as either 'frequent' ($>10^{-3}$ per year) or 'infrequent' ($\leq 10^{-3}$ per year).

3.2.12 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 1milliSievert (mSv) or workers (20mSv) identified within safety assessment principle Target 4 [RD9]. Frequent ($>10^{-3}$ per year) low consequence (<basic safety limit) events are considered to be foreseeable events and are assessed as part of normal operations assessment.

3.2.13 Figure 3-1 shows the relationship between frequency and dose that defines the basic safety level interface with the design basis analysis.

Figure 3-1 Design basis region for off-site and on-site consequences



3.2.14 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, see above).

Beyond Design Basis Analysis

3.2.15 In addition to the assessment of the design basis faults discussed above, the ONR also expects the licensee to consider "*fault sequences initiated by internal and external hazards beyond the design basis should be analysed applying an appropriate combination of engineering, deterministic and probabilistic assessments*".

3.2.16 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.

3.2.17 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.

Severe Accident Analysis

3.2.18 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.

3.2.19 A severe accident is defined in the SAPs [RD9] 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*".

3.2.20 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.

3.2.21 Severe accident analysis looks at the accident progression in the event that the identified design basis safety measures fail. It is primarily concerned with accidents with the potential to result in the loss of coolable core geometry and hence significant release of radioactive fission products from within the fuel cladding boundary.

3.3 Reference accidents identified from the fault analysis

3.3.1 A review of the fault schedule was undertaken and three reference design basis accidents (DBA) were identified from the list of faults identified using the methods described in section 3.2. The DBAs presented were chosen on the basis of their radiological consequences. Their predicted frequency of occurrence is greater than 10^{-5} per year.

3.3.2 In addition to the DBAs, a Severe Accident (SA) was chosen which is considered to be well beyond the design basis in terms of likelihood, and involves making a number of pessimistic assumptions concerning the failure of safety-classified plant. The SA is presented to demonstrate plant safety features to mitigate consequences of a rare event that involves core meltdown (including ex-vessel scenarios) and potential radiological releases.

3.3.3 An SA is an accident that will require the implementation of a severe accident management regime and the activation of key response measures. The SA considered has a very low predicted frequency of occurrence and is presented for illustration of the potential impact of such rare events.

3.3.4 The accidents identified result in atmospheric releases. No accidents involving foreseeable significant liquid effluent releases have been identified (see section 3.5).

3.3.5 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.

3.3.6 The quantities of noble gases and selected radionuclides which are significant from the point of view of health, along with the total activity released, are presented in table 4-3 in section 4.1.

Loss of Coolant Accident

Accident scenario

3.3.7 For the design basis LOCA, coolant loss is assumed to occur through a limiting line (i.e. feedwater line or main steamline) which suffers a double ended guillotine rupture inside the PCV. All radioactivity in the coolant and an 'iodine spike' originating from assumed pre-existing minor pin hole defects in the fuel cladding are immediately released into the containment from the ruptured pipe.

3.3.8 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

3.3.9 It is assumed that during a LOCA of this type, of the iodine available for release, 91% is in elemental form, 4% is in organic form and the remaining 5% is in particulate form. Once released to the containment atmosphere, a number of factors further reduce the amounts available for release to the environment. These include radioactive decay, removal processes and leakage to other plant areas. The 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 steamline isolation valves.

Fuel Handling Accident

Accident scenario

3.3.10 During a refuelling 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, allowing the assembly, fuel grapple mast and head to fall on top of the core impacting a group of four assemblies, in turn causing fuel rods to fail. The accident is assumed to occur at the earliest time after shutdown that fuel handling operations can begin.

3.3.11 When fuel rods in the dropped and impacted assemblies fail, radioactive gases are released into the water of the reactor cavity. These gases pass from the reactor water to the reactor building fuel handling area. In response to the increased radioactivity level resulting from the gas release, high radiation alarms on the reactor building operational floor are activated, alerting plant personnel to the situation and initiating the isolation of the reactor building HVAC system. The standby gas treatment system is automatically initiated.

3.3.12 A maximum of two bundles or 184 fuel rods are assumed to be damaged in the accident, out of a total of 872 bundles.

3.3.13 The activity in the gap and plenum regions of the failed fuel is released to the reactor water. The fragments generated by the impact are much larger than aerosols and do not disperse on the pool surface; therefore, radionuclides other than noble gases, halogens and alkali metals are not included in the release estimate.

3.3.14 Chemical fractions of the iodine released to the environment are taken to be 96% elemental and 4% organic iodine. Pool decontamination factors are not credited for organic and elemental iodine; however, the particulate is retained in the pool.

Release to the environment

3.3.15 It is assumed that the reactor building ventilation system is isolated and that upon the receipt of a high radiation alert on the operational floor that the standby gas treatment system is initiated. This is an automatic response to the detection of increased radioactivity from the surface of the pool. 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.

3.3.16 The high radiation level associated with the FHA would be automatically detected by the reactor building gamma monitors. Any delay in terms of environmental release due to the detection time of the radiation monitoring system is considered negligible when considered in the context of the assumed 24-hour period of release.

Off-Gas system Failure

Accident scenario

3.3.17 A number of failure scenarios are considered for the Off-Gas (OG) system:

- rupture of a line or an equivalent gross system failure;
- failure of a thin wall in a process line;
- corrosion of a process line by turbulent flow; or
- process line fails at a thin wall or junction caused by stress concentration due to flow-induced or external vibration.

3.3.18 Ruptures and failures are assumed to occur in the upstream section of a process line in this scenario as the radioactive concentration of the gas is higher.

3.3.19 A rupture or break in the 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.

3.3.20 Key parameters are assumed to be the same as for a LOCA.

Release to the environment

3.3.21 The release of fission products from the OG system is assumed to be isolated one hour after the detection of high radiation levels due to operator action. The pipe break is assumed to occur before the charcoal beds, so the radioactivity of the gas in the system is not reduced.

3.3.22 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.

3.4 Core melt accidents

3.4.1 Core melt scenarios are considered as part of the severe accident analysis for the UK ABWR. The predicted frequency of such event falls well below that quoted as the threshold for the beyond design basis analysis.

3.4.2 The assessment that follows is included to provide information on the potential consequences of a multiple failure of safety-related measures that result in a damaged core. For the sake of creating the analytical basis of a damaged core, a number of pessimistic assumptions regarding plant and operational staff responses have been necessary.

3.4.3 In the main, initial plant response is based on ‘beyond design basis’ systems in a conservative timeframe, until such time that recovery of ‘design basis’ systems can reasonably be claimed. For this reason, no specific initiating event is identified to cause the interruption of feedwater to an operating reactor.

Severe Accident scenario

3.4.4 For a reactor operating at full power, a loss of feedwater leads to a rapid decrease in reactor water level. The transient leads to an emergency reactor shutdown SCRAM or alternatively the standby liquid control system will initiate in the unlikely event that control rods fail to insert into the core. Core cooling should normally be delivered by the main condenser of the plant system; however, this is assumed to be unavailable in this scenario as it is not a safety-classified system.

3.4.5 At this point the high pressure ECCS is expected to start, but in this instance 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.

3.4.6 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 RPV by relief into the suppression pool within the PCV, so the event progresses at low pressure. This will normally allow for initiation of the low pressure injection feature of the ECCS but it is also assumed to fail in this accident scenario, as it is part of the residual heat removal system which is assumed to be wholly unavailable.

3.4.7 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.

3.4.8 As the water level in the RPV decreases, the fuel is cooled by steam flow which is a poor method of heat removal and fails to halt rising fuel and cladding temperatures once the water level falls below 20% of bottom of active fuel. Fuel cladding failure occurs due to creep, melting or ballooning at elevated temperatures.

3.4.9 Rising fuel temperatures cause fission product gases to migrate from the pellet to the pellet cladding gap which is not accredited as a retention measure at high cladding 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.

3.4.10 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.

3.4.11 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 backup 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.

3.4.12 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.

3.4.13 The weld which holds the control rod penetration tubes to the lower head is weakened by the molten corium in the lower vessel. The analysis predicts that a control rod drive tube is ejected due to the melting of the attachment weld or the stress caused by the pressure difference between inside and outside of the RPV and corium weight on the RPV lower head, leading to failure of the lower head after about seven hours.

3.4.14 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.

3.4.15 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.

3.4.16 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.

3.4.17 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

3.4.18 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.

3.4.19 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).

3.4.20 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. Radioactive material leaking into the reactor building may in turn be subject to removal mechanisms due to plate out and deposition, or be lost from the reactor building via leakage and diffusion.

3.5 Spent fuel storage facility and ILW storage facility

3.5.1 A spent fuel storage facility would be provided on-site, for the storage of spent fuel (and decay storage of High Level Waste (HLW) and Intermediate Level Waste (ILW)) until such time as a Geological Disposal Facility becomes operational. As stated in appendix D14-1 (Radioactive waste) (Application Reference Number: 6.4.97), the store would be required about five to 10 years after the start of generation until up to 140 years after the end of generation. With regard to this facility, a safety case would be drawn up based on the high level claims stated below and reviewed by the regulator prior to construction.

3.5.2 During normal operation and following frequent and infrequent faults and hazards:

- containment of spent fuel would be maintained;
- temperatures of spent fuel would be maintained within specific limits such that fuel cladding does not fail due to overheating;
- spent fuel assemblies would be maintained in a subcritical state;
- shielding and contamination control would be maintained to reduce radiological dose to the public and operators to ALARP; and
- handling and retrieval of spent fuel would be maintained and faults and hazards would be shown to be of acceptably low frequency.

3.5.3 This facility would be housed in its own building on-site. The possibility of interaction with other facilities leading to a fault scenario would be addressed as part of the site-specific safety case in conjunction with measures to prevent such scenarios or mitigate any impacts. This facility would be designed with respect to the ALARP principle and would be subject to fault assessment in a manner that is commensurate with the recognised risks.

3.5.4 Spent fuel that is moved to the spent fuel storage facility would have spent a number of years in the spent fuel storage pool. This decay time means that spent fuel stored in this facility would be significantly less active than fuel recently removed from the reactor.

3.5.5 The ILW storage facility is a heavily shielded, self-contained, multi-storey building. The on-site ILW store is a stand-alone, self-supporting facility. The store is designed to hold all of the packaged ILW generated in the operating lifetime of the Power Station (i.e. 60 years) plus any additional ILW resulting from post operation clean out of the liquid effluent management systems.

3.5.6 During the storage phase, inspection and monitoring of packages are part of the storage procedures. When corrosion of a waste package is detected, early measures are possible, preventing potential spreading of contamination. Thus, release of contamination from a waste package is considered extremely unlikely and it can be expected that only conventional wastes are generated during operation of the interim on-site storage facilities.

3.5.7 Accidents arising from the improper handling of ILW and HLW have not been selected as reference accidents. It was found that handling accidents involving irradiated fuel were more significant in terms of radiological impact, and therefore ILW/HLW handling accidents have been scoped out.

3.6 Release to the aquatic environment

3.6.1 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.

3.6.2 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.

3.6.3 The primary containment of the UK ABWR is a reinforced concrete containment vessel consisting of the reinforced concrete, liner and metal containment components (such as the drywell head). The reinforced concrete containment vessel is integrated with the reactor building. This containment provides an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment (see section 2.3).

3.6.4 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.

3.6.5 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.

3.6.6 A number of other features have been included in the design which reduce the likelihood of a release of liquid radioactive effluent to the aquatic environment.

- Measures would be taken, so far as is reasonably practicable, to minimise leakages from pipework transferring radioactive effluents.
- The liquid effluent management control system includes monitoring of all the main process parameters (pressure, flow, temperature, tank levels etc.) with appropriate alarms provided to the operators in the event of abnormal conditions.
- The liquid effluent management control system includes level control for all tanks including appropriate interlocks to prevent tank overflows. However, all high chemical impurity waste, low chemical impurity waste and controlled active drains system tanks would also have engineered overflow routes to alarmed sumps where appropriate.
- Bunding would be provided in line with UK regulatory requirements and relevant industry good practice, including all tanks and, where appropriate, any other piece of equipment containing liquids.
- Bunding sumps include leak detectors/alarms and pumps to recover any spilt liquids into the liquid effluent management system.
- Bunding is provided at all external doors to liquid effluent management system buildings to prevent the spread of any spilt liquids to the outside of the buildings.

3.6.7 The measures outlined provide sufficient control that accidents resulting in releases to the aquatic environment have been scoped out.

4 Assessment of the radiological impact of the reference accidents

4.1 Assumptions used to calculate the impacts from releases to atmosphere

4.1.1 The assessment considers the radiological consequences of releases to 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).

4.1.2 The local reference group is assumed to be members of the public located at the distance at which maximum exposure levels are experienced, for distances greater than 200m (i.e. beyond the site boundary).

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

4.1.4 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.

4.1.5 It is assumed that the weather conditions remain constant for the duration of the release. For the assessment for the nearest country (Ireland) it is also assumed that the weather conditions are constant during the period of plume travel.

4.1.6 For both the short-range and long-range assessments for local and Ireland reference groups respectively, the radionuclides which make up the reference accident source terms are modelled. Decay and ingrowth of radiologically-significant daughter radionuclides are also modelled during the period of plume travel.

4.1.7 For reasons discussed in section 3.6, an accident scenario resulting in a liquid release to the aquatic environment is not considered.

Release paths and release durations

4.1.8 The release paths and release durations for the reference accidents are summarised in table 4-1. For the LOCA and FHA scenarios, a nominal release period of 24 hours was chosen. For the OGF, a period of one hour was chosen for the release, which is consistent with the description of the accident scenario given in section 3.3. For the SA, the release paths and release durations are consistent with the PSA analysis for internal events at power (leading to a degraded core).

4.1.9 The long-range model used for calculations to the nearest country is based on a nominal release duration of 12 hours. For the OGF and SA scenarios, this minimum release duration of 12 hours has been applied to the calculations for Ireland whereas the release duration for calculations for areas close to the Power Station site is less than 12 hours. It is noted that the main effect of an increased release duration is the broadening of the plume in the cross-wind direction due to wind meander (i.e. small variations in the wind direction over time).

4.1.10 The release height for the reactor building stack is assumed to be 75m. For releases from the building, it is assumed that a release into the building wake occurs as described in section 4.1.16. The release height makes little difference to the assessed concentrations to the Ireland reference group due to the much greater distance involved.

Table 4-1 Release paths and release durations

Accident identifier	Reference accident	Release duration – local (hours)	Release duration – Ireland (hours)	Release paths
LOCA	Loss of Coolant Accident	24	24	88% from the plant stack 12% from the turbine building
FHA	Fuel Handling Accident	24	24	100% from the plant stack
OGF	Off-Gas system Failure	1	12	100% from the turbine building
SA	Containment leakage from Drywell (failed RPV)	4	12	100% from the plant stack

Amounts and physico-chemical forms of radionuclides

4.1.11 Some radionuclides can be released in different physico-chemical forms which may affect their behaviour in the environment and the radiological consequences. For these assessments, the most important of these are the iodine isotopes which can have three main forms: particulate (ionic iodide, most likely caesium iodide), elemental iodine (molecular iodine vapour, I_2), and organic iodine (many forms are possible but usually considered to be methyl iodide, CH_3I). The particulate form will behave like other particulates.

4.1.12 Elemental iodine is reactive and has a low boiling point (184°C), and therefore will deposit more readily than particulate iodine. Organic iodine, on the other hand, is relatively inert chemically and will deposit less readily than particulate iodine.

4.1.13 Each chemical form will behave differently in the body after inhalation and thus have a different inhalation dose coefficient; elemental iodine being the most radiotoxic and organic iodine the least radiotoxic.

4.1.14 Treating the different forms of iodine explicitly is therefore important. It is assumed that the proportions of different iodine species for the accident scenarios are as shown in table 4-2.

Table 4-2 Proportions of iodine species assumed for the assessments

Reference accident scenario	Percentage of iodine species		
	Organic iodine	Elemental iodine	Particulate iodine
LOCA			100%
FHA			100%
OGF			100%
SA	4%	91%	5%

4.1.15 The core melt scenario source terms have been generated using a simple modifying factors approach to incorporate iodine chemistry effects and the effect of filters in the standby gas treatment system. The standby gas treatment system filter array provides a decontamination factor of 1,000 for all iodine types.

Table 4-3 Summary of reference accident source terms

Nuclide	Release (Bq)			
	LOCA	FHA	OGF	SA
H-3	5.60E+09	0.00E+00	1.00E+11	0.00E+00
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
Kr-83m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	2.50E+09	9.70E+13	7.10E+06	4.38E+14
Kr-85m	6.40E+07	2.20E+08	2.20E+10	1.49E+14
Kr-87	1.20E+07	0.00E+00	1.20E+11	5.70E+13
Kr-88	0.00E+00	4.80E+05	7.80E+10	1.45E+14
Xe-131m	3.90E+08	3.60E+12	8.70E+07	0.00E+00
Xe-133	3.90E+10	4.70E+14	6.70E+10	3.13E+16
Xe-133m	4.00E+08	6.50E+12	1.40E+09	0.00E+00
Xe-135	1.00E+09	2.40E+11	9.10E+10	5.24E+15
Xe-135m	6.20E+07	0.00E+00	1.70E+11	0.00E+00
Xe-138	1.4E+06	0.00E+00	2.50E+12	0.00E+00
Other	5.74E+09	0.00E+00	4.22E+12	1.45E+12
Total	4.92E+10	5.77E+14	7.27E+12	3.74E+16

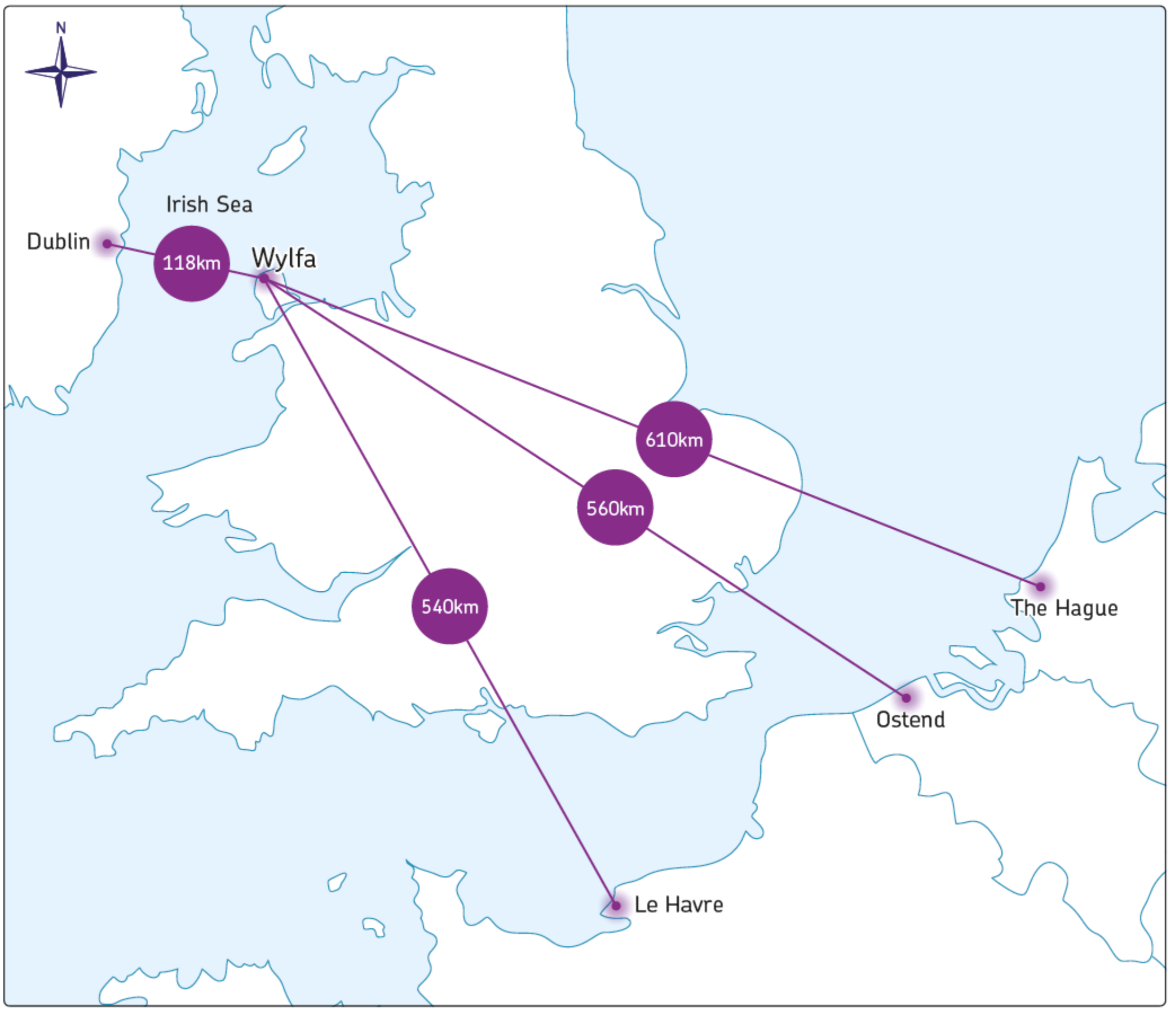
4.1.16 For all of the reference accidents, a large fraction of the total release is made up of noble gases. The quantities of noble gases and selected radionuclides which are significant from the point of view of health, along with the total activity released, are presented in table 4-3. It should be noted that the radionuclides identified are not significant for the OGF scenario as less than half the release is represented by the short list of radionuclides. This is demonstrated as those marked 'Other' make up a larger proportion of the total release.

4.2 Models and parameter values used

Distances to nearest countries

4.2.2 The nearest country to the Power Station is Ireland, which is approximately 118km away. The distances to the nearest major conurbations in the nearest countries, and their populations, are provided in table 4-4. Figure 4-1 details the location of the Power Station in relation to Europe and its nearest major conurbations.

Figure 4-1 Location of the Power Station in relation to Europe and its nearest major conurbations



4.2.3 The potential impact of radioactive discharges from the Power Station to other countries has been assessed by considering the impact to reference groups in Ireland (Dublin), as it is closest to the Power Station. Activity concentrations in air decline rapidly with increasing distance. The impacts of the radioactive discharges from the Power Station to reference groups in Ireland are assumed to be bounding over reference groups in any other country.

Table 4-4 Nearest major international conurbations to the Power Station

Centre	Country	Distance/km	Population
Dublin	Ireland	118	516,255 (estimate)
Le Havre	France	540	238,171
Ostend	Belgium	560	69,980
The Hague	Netherlands	610	495,083

Local reference group assessment

Dispersion and deposition

4.2.4 For the local assessment, time-integrated activity concentrations are calculated using the model described in NRPB-R91 [RD10]. Depletion of the plume by dry and wet deposition is included using the approach described in NRPB-R122 [RD11]. The dry deposition velocities and the washout coefficients used are presented in table 4-5.

Table 4-5 Dry and wet deposition parameters

Species	Dry deposition velocity (m/s)	Washout coefficient (s-1 per mm/hr of rain)
Particulate	1E-03	1E-04
Organic iodine	1E-05	2E-08
Elemental iodine	3E-03	3E-06
Noble gases and CO_2	0E+00	0E+00

4.2.5 For releases from a building, it is assumed that a release into the building wake occurs at a height of one third of the building height from a virtual source upwind of the building as described in NRPB-R157 [RD12]. The effect of the virtual source is to broaden the plume in the vertical and cross-wind directions leading to increased dispersion. This approach assumes that the building wake is well-mixed and the release is effectively from the downwind face of the building.

Meteorological conditions

4.2.6 The meteorological conditions are based on the NRPB-W54 methodology [RD13] for assessing short-term discharges, which states that these conditions are realistically cautious. A surface roughness length of 0.4m has been selected, which is appropriate for the area around the Power Station; consisting of farmland and residential areas. These meteorological parameters are presented in table 4-6. The wind is assumed to blow in the same direction for the duration of the release. A wind meander factor which depends on the duration of the release is applied.

Table 4-6 Meteorological parameters for the local reference group assessment

Pasquill stability category	Mixing layer depth (m)	Windspeed at a height of 10m (m/s)	Rainfall rate in wet conditions (mm/hr)	Surface roughness length (m)
D	800	3	0.1	0.4

Habit data

4.2.7 The Centre for Environment, Fisheries and Aquaculture Science (Cefas) undertakes periodic (normally five-yearly) surveys of the diet and behaviours of members of the public residing in the locality of major nuclear licensed sites in the UK. These surveys collate information on the food consumption rates and occupancy habits of adult, child and infant age groups[1], as well as other relevant occupational and recreational activities that take place close to nuclear licensed sites that could result in the exposure of members of the public to radioactivity in the environment. The output from these surveys provides information on the habits of members of the public which may influence their radiation exposure.

4.2.8 Cefas has performed three habits surveys for the Existing Power Station ([RD14]; [RD15]; [RD16]) which cover the areas that are most likely to be impacted by discharges to the marine environment, discharges to air and from direct radiation emanating from the Existing Power Station. The Power Station is situated immediately adjacent to the Existing Power Station site and so the habits data presented in the Cefas habits survey reports for the Existing Power Station are therefore considered to directly apply to the Power Station without the need to make any modifications.

4.2.9 Ingestion rates used in the assessment are based on the top two approach[2]. For accident scenarios, the food groups varied depending on the accident scenario. For the LOCA, FHA and SA, green vegetables and milk were consumed at 97.5th percentile rates. For the OGF, root vegetables and milk were consumed at 97.5th percentile rates. Other foods were consumed at mean rates, using the median value of the 97.5th percentile and mean values from the three Cefas reports. This approach is expected to result in robust dose assessment outcomes.

4.2.10 The habit data for the local reference group are shown in table 4-7.

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[1] For the purposes of the assessment children are assumed to be 10 years old and infants one year old

[2] A screening dose assessment is carried out using food intake rates set at ‘critical levels’ (97.5th percentile) for all food categories. The two food categories contributing the highest dose are retained at critical levels whilst the remaining categories are set to mean levels (50th percentile rates). This approach avoids undue pessimism in overall food consumption rates and ensures that calorific intakes are within reasonable limits. It can be implemented using either generic or site-specific data

Table 4-7 Consumption rates for the local reference group assessment

Food type	Consumption rate (kg/y)					
	Adults		Children		Infants	
	Mean	97.5th %-ile	Mean	97.5th %-ile	Mean	97.5th %-ile
Cow's milk	140.3	193.1	140.3	193.1	187.0	257.4
Root vegetables	141.3	172.4	91.5	110.5	44.7	54.9
Green vegetables	47.4	69.7	22.2	32.4	10.1	14.9
Fruit	34.2	36.9	21.7	24.0	13.8	16.1
Beef	31.5	31.5	21.0	21.0	7.0	7.0
Sheep meat	12.2	17.0	4.9	6.8	1.5	2.0

Inhalation rates

4.2.11 The adult inhalation rate for the local reference group is based upon the NRPB-W41 [RD17]. The '24 hour total' value for a heavy worker is used, which includes eight hours of sleep, eight hours of heavy work and eight hours of non-occupational activity.

Adult inhalation rate = 27m^3/24 hours = 1.123m^3/hr.

4.2.12 Inhalation rates for the infant and child members of the local reference group are based upon NRPB-W41 [RD17].

Child inhalation rate = 0.64m^3/hr.

Infant inhalation rate = 0.22m^3/hr.

Assessment for the nearest country (Ireland)

Dispersion and deposition

4.2.13 For the nearest country (Ireland), the time-integrated activity concentration, the dry deposition and the wet deposition are calculated as described in NRPB-R124 [RD18], whilst the plume depletion due to deposition is calculated as described in NRPB-R122 [RD11]. The results obtained are for the 90th percentile as recommended in NRPB-R124 [RD18]. The dry deposition velocities and the washout coefficients are the same as those used for the local assessment as presented in table 4-5.

Meteorological conditions

4.2.14 The meteorological conditions are based on the NRPB-R124 methodology [RD18] and are presented in table 4-8.

Table 4-8 Meteorological parameters for the assessment for the nearest country (Ireland)

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

Habit data

4.2.15 The Radiological Protection Institute of Ireland (now part of the Irish Environmental Protection Agency) published an assessment of the potential radiological implications for Ireland from proposed new build of nuclear power plants in the UK [RD19]. This assessment included habit and occupancy data for reference groups impacted by routine discharges to air and sea and for accident scenarios.

4.2.16 The habit data presented in the assessment for releases to atmosphere were used when assessing the doses to the nearest country (Ireland). Mean and 95^{th} percentile consumption rates were presented and the 95^{th} percentile rates have been used to represent individuals with higher than average consumption rates. This approach is expected to result in robust dose assessment outcomes.

4.2.17 The greater Dublin area contains a number of dairy farms and milk processing centres, as well as farms growing vegetables for sale. The reference group is a hypothetical dairy and market gardening farming family living and working in the greater Dublin area that will be expected to eat higher than average rates of locally-grown foods. The family is assumed to live in a rural location and spend an above-average length of time outdoors, for example adults working the land.

4.2.18 The top two approach has been used regarding the use of food ingestion habits data. A local fraction of 1.0 (consumption of 100% locally produced food) has been assumed for the food categories consumed at critical levels and 0.5 (50% locally produced food) for all other groups. Given the scale and reach of modern food distribution systems, these were considered conservative values. For accident scenarios, the food groups varied depending on the accident scenario or age group. For adults the top two foodstuffs were milk products and milk (FHA, OGF and SA), and milk products and beef (LOCA). For children the top two foodstuffs were milk products and milk for all scenarios.

4.2.19 The habit data for the nearest country (Ireland) reference group are presented in table 4-9.

Table 4-9 Consumption rates for the assessment for the nearest country (Ireland)

Food type	Consumption rate (kg/y)					
	Adults		Children		Infants	
	Mean	97.5th %-ile	Mean	97.5th %-ile	Mean	97.5th %-ile
Cow's milk	77.4	336.9	100.7	311.7	105.9	449.2
Cow's milk products	24.5	113.2	27.4	104.8	18.3	84.9
Root vegetables	58.0	196.0	42.7	124.8	14.5	67.8
Green vegetables	26.3	109.9	9.9	45.6	3.8	20.6
Fruit	17.9	71.9	15.0	48.5	8.0	33.6
Beef	25.4	117.2	15.5	60.5	5.1	26.0
Sheep meat	4.9	31.9	3.8	18.9	0.5	3.8

Inhalation rates

4.2.20 Inhalation rates for the nearest country (Ireland) reference group have been aligned with those previously used by the Radiological Protection Institute of Ireland and are based upon NRPB-W41 [RD17] values for all age groups.

Adult inhalation rate = 0.92 m^3/hr.

Child inhalation rate = 0.64 m^3/hr.

Infant inhalation rate = 0.22 m^3/hr.

4.3 Calculation of doses

4.3.1 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.

4.3.2 The activity concentration in soil and terrestrial foods per unit deposit values were obtained using the FARMLAND model within PC CREAM 08 [RD20]. Although FARMLAND was originally formulated to estimate annual individual doses from normal operational discharges of radionuclides to atmosphere, the model can be used satisfactorily to produce an integrated dose to an individual following a single deposit on the ground by a straightforward dimensional conversion of the units in which the results are produced.

4.3.3 The values obtained using the FARMLAND model are effectively used as transfer factors to determine the time integrated activity concentrations in various foods using the NRPB-R91 [RD10] and NRPB-R124 [RD18] results for deposition (and activity concentration in the case of H-3 and C-14).

4.3.4 To obtain the activity concentrations for cow's milk products, the same method as implemented in PC CREAM is followed. A scaling factor is used to generate the activity concentration in cow's milk products from the activity concentration in cow's milk for each radionuclide. These scaling factors can be found in appendix E of the 'PC CREAM Help' available in the PC CREAM program itself.

4.3.5 The results for the local reference group are calculated for the distance at which maximum exposure levels are experienced, for distances greater than 200m. For building releases (OGF) this is 200m, whilst for stack releases (LOCA, FHA and SA) this is 1,060m. For LOCA, wet weather conditions were found to reduce the distance of the maximum concentrations and to result in significantly higher concentrations than dry weather conditions. The results for the Ireland reference group are calculated at a distance of 118km.

4.3.6 The effective dose coefficients for inhalation and ingestion are taken from International Commission on Radiological Protection (ICRP) data [RD21]. For most radionuclides, it is necessary to select the appropriate absorption type which corresponds to how readily material is absorbed into the blood from the respiratory tract. Where possible, the absorption type was selected using the guidance given in ICRP Publication 71 [RD22]. In other cases, the absorption type was chosen so that the highest value was selected in the case of two available absorption types or the middle value was selected in the case of three available absorption types.

4.4 Results

Maximum time integrated concentrations and surface contamination levels

4.4.2 Table 4-10 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. For LOCA, the difference between the time integrated activity concentrations for the dry weather conditions and wet weather conditions is significant so two values (wet and dry) are presented.

Table 4-10 Maximum time integrated activity concentration

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

4.4.3 The greatest difference between the time integrated activity concentrations is observed for the OGF, with the value for the local reference group being over 100,000 times greater than the value for the Ireland reference group. This difference is much greater than for the other accident scenarios and is most likely due to the OGF release being from a building, with the local results presented at a distance of only 200m. For all reference accident scenarios, the value for the local reference group is over 400 times greater than the value for the Ireland reference group.

4.4.4 The maximum surface contamination levels for the two reference groups are presented in table 4-11. Results for dry weather conditions and wet weather conditions are provided.

Table 4-11 Maximum surface contamination levels

Reference accident scenario	Surface contamination (Bq/m^2)			
	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

4.4.5 As expected, wet weather conditions result in greater 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.

Expected levels of radioactive contamination of foodstuffs

4.4.6 The integrated activity concentrations in foods are presented in table 4-12 and table 4-22 for the local reference group and the Ireland reference group respectively. Cow's milk products are predicted to have the greatest activity concentrations for both reference groups. However, cow's milk products do not contribute to the ingestion dose for the local reference group since it is assumed that there is no consumption of cow's milk products produced from local cow's milk.

Table 4-12 Expected levels of radioactive contamination of foodstuffs for the local reference group

Reference accident scenario	Activity concentrations in foods, integrated to 50 years (Bqy/kg)					
	Cow's milk	Beef	Sheep meat	Green veg	Root veg	Fruit
LOCA	9.04E-02	6.93E-02	7.00E-02	8.49E-02	7.72E-02	7.88E-02
FHA	4.14E-03	2.11E-02	4.41E-02	3.47E-03	3.07E-03	8.41E-03
OGF	3.39E+01	2.62E+01	2.62E+01	3.00E+01	2.97E+01	2.98E+01
SA	1.26E+01	5.30E+00	6.72E+00	9.52E+00	1.79E+00	6.13E+00

Local reference group doses

4.4.7 Table 4-13, table 4-14 and table 4-15 present the effective dose to an adult, a 10 year old child and a one year old infant for the local reference group for each pathway.

Table 4-13 Maximum effective dose to adults for the local reference group

Reference accident scenario	Effective dose to adults (Sv)				
	Inhalation	Cloud gamma	Ground gamma	Ingestion	Total
LOCA	5.79E-10	1.74E-10	5.72E-08	5.30E-09	6.33E-08
FHA	3.50E-10	4.74E-07	5.13E-08	4.05E-08	5.66E-07
OGF	5.67E-07	7.88E-06	1.37E-07	2.81E-06	1.14E-05
SA	1.58E-05	1.97E-04	1.07E-06	8.02E-05	2.94E-04

Table 4-14 Maximum effective dose to children for the local reference group

Reference accident scenario	Effective dose to children (Sv)				
	Inhalation	Cloud gamma	Ground gamma	Ingestion	Total
LOCA	4.90E-10	1.04E-10	2.91E-08	7.41E-09	3.71E-08
FHA	1.63E-10	2.84E-07	2.61E-08	2.01E-08	3.31E-07
OGF	7.61E-07	4.73E-06	6.96E-08	5.26E-06	1.08E-05
SA	2.17E-05	1.18E-04	5.47E-07	1.59E-04	2.99E-04

Table 4-15 Maximum effective dose to infants for the local reference group

Reference accident scenario	Effective dose to infants (Sv)				
	Inhalation	Cloud gamma	Ground gamma	Ingestion	Total
LOCA	3.96E-10	8.12E-11	1.98E-08	2.02E-08	4.05E-08
FHA	8.42E-11	2.21E-07	1.77E-08	1.95E-08	2.58E-07
OGF	1.00E-06	3.68E-06	4.72E-08	2.01E-05	2.48E-05
SA	2.58E-05	9.21E-05	3.71E-07	6.23E-04	7.41E-04

4.4.8 Since ingestion is the dominant pathway in the majority of cases, a breakdown by food type is presented in table 4-16, table 4-17 and table 4-18 for adults, children and infants respectively.

Table 4-16 Maximum effective dose due to ingestion for adults (local) – breakdown by food type

Reference accident scenario	Effective dose (Sv)					
	Cow's milk	Beef	Sheep meat	Green veg	Root veg	Fruit
LOCA	3.20E-09	2.47E-10	1.23E-10	1.05E-09	4.38E-10	2.39E-10
FHA	1.04E-08	8.69E-09	9.78E-09	2.15E-09	5.67E-09	3.76E-09
OGF	1.74E-06	1.30E-07	6.30E-08	4.68E-07	2.55E-07	1.62E-07
SA	5.14E-05	3.57E-06	1.77E-06	1.33E-05	5.53E-06	4.55E-06

Table 4-17 Maximum effective dose due to ingestion for children (local) – breakdown by food type

Reference accident scenario	Effective dose (Sv)					
	Cow's milk	Beef	Sheep meat	Green veg	Root veg	Fruit
LOCA	5.27E-09	2.34E-10	6.42E-11	1.13E-09	4.38E-10	2.76E-10
FHA	8.06E-09	4.45E-09	2.17E-09	7.76E-10	2.82E-09	1.83E-09
OGF	3.96E-06	1.90E-07	5.59E-08	4.93E-07	3.32E-07	2.29E-07
SA	1.22E-04	5.60E-06	1.67E-06	1.46E-05	8.45E-06	6.81E-06

Table 4-18 Maximum effective dose due to ingestion for infants (local) – breakdown by food type

Reference accident scenario	Effective dose (Sv)					
	Cow's milk	Beef	Sheep meat	Green veg	Root veg	Fruit
LOCA	1.78E-08	1.82E-10	4.35E-11	1.28E-09	5.01E-10	4.26E-10
FHA	1.30E-08	1.78E-09	7.96E-10	4.28E-10	2.04E-09	1.40E-09
OGF	1.81E-05	2.13E-07	5.75E-08	7.74E-07	5.14E-07	4.90E-07
SA	5.62E-04	6.47E-06	1.77E-06	2.34E-05	1.43E-05	1.50E-05

Nearest country (Ireland) reference group doses

4.4.9 Table 4-19, table 4-20 and table 4-21 present the effective dose to an adult, a 10 year old child and a one year old infant for the Ireland reference group respectively. As for the local reference group, in the majority of cases ingestion is the dominant pathway.

Table 4-19 Maximum effective dose to adults for the nearest country (Ireland) reference group

Reference accident scenario	Effective dose to adults (Sv)				
	Inhalation	Cloud gamma	Ground gamma	Ingestion	Total
LOCA	2.72E-13	1.09E-13	8.84E-11	3.70E-11	1.26E-10
FHA	5.63E-13	1.08E-09	3.71E-10	7.57E-10	2.21E-09
OGF	1.67E-11	1.64E-11	3.61E-11	3.85E-09	3.92E-09
SA	1.67E-08	2.23E-07	1.39E-09	5.16E-07	7.56E-07

Table 4-20 Maximum effective dose to children for the nearest country (Ireland) reference group

Reference accident scenario	Effective dose to children (Sv)				
	Inhalation	Cloud gamma	Ground gamma	Ingestion	Total
LOCA	2.81E-13	6.53E-14	4.50E-11	5.71E-11	1.02E-10
FHA	3.19E-13	6.50E-10	1.89E-10	4.65E-10	1.30E-09
OGF	2.73E-11	9.82E-12	1.84E-11	8.34E-09	8.39E-09
SA	2.79E-08	1.34E-07	7.07E-10	1.12E-06	1.28E-06

Table 4-21 Maximum effective dose to infants for the nearest country (Ireland) reference group

Reference accident scenario	Effective dose to infants (Sv)				
	Inhalation	Cloud gamma	Ground gamma	Ingestion	Total
LOCA	2.27E-13	5.08E-14	3.06E-11	1.38E-10	1.68E-10
FHA	1.65E-13	5.06E-10	1.28E-10	5.11E-10	1.15E-09
OGF	3.58E-11	7.64E-12	1.25E-11	2.74E-08	2.75E-08
SA	3.31E-08	1.04E-07	4.80E-10	3.68E-06	3.28E-06

4.4.10 Since ingestion is the dominant pathway in the majority of cases, a breakdown by food type is presented in table 4-23, table 4-24 and table 4-25 for adults, children and infants respectively. It is noted that the inclusion of cow's milk products has a significant effect on the dose, with this being the greatest contributor to the total despite the consumption rates being higher for other food types.

Table 4-22 Expected levels of radioactive contamination of foodstuffs for the nearest country (Ireland) reference group

Reference accident scenario	Activity concentrations in foods, integrated to 50 years (Bqy/kg)						
	Cow's milk	Beef	Sheep meat	Green veg	Root veg	Fruit	Cow's milk products
LOCA	5.48E-05	4.13E-05	4.23E-05	5.59E-05	4.41E-05	4.66E-05	8.35E-05
FHA	2.96E-05	1.51E-04	3.15E-04	2.48E-05	2.20E-05	6.01E-05	3.25E-04
OGF	1.41E-03	1.05E-03	1.06E-03	1.25E-05	1.15E-03	1.20E-03	1.79E-03
SA	1.62E-02	6.84E-03	8.68E-03	1.21E-02	2.31E-03	7.91E-03	1.60E-01

Table 4-23 Maximum effective dose due to ingestion for adults nearest country (Ireland) – breakdown by food type

Reference accident scenario	Effective dose (Sv)						
	Cow's milk	Beef	Sheep meat	Green veg	Root veg	Fruit	Cow's milk products
LOCA	8.01E-12	1.37E-13	3.48E-14	2.84E-13	9.87E-14	8.34E-14	2.83E-11
FHA	1.50E-11	2.31E-10	1.01E-11	4.26E-12	8.32E-12	7.03E-12	4.81E-10
OGF	8.72E-10	1.43E-11	3.53E-12	2.50E-11	1.16E-11	1.17E-11	2.91E-09
SA	1.16E-07	1.86E-09	4.61E-10	3.24E-09	1.47E-09	1.54E-09	3.91E-07

Table 4-24 Maximum effective dose due to ingestion for children nearest country (Ireland) – breakdown by food type

Reference accident scenario	Effective dose (Sv)						
	Cow's milk	Beef	Sheep meat	Green veg	Root veg	Fruit	Cow's milk products
LOCA	1.24E-11	1.20E-13	3.52E-14	2.55E-13	1.20E-13	1.33E-13	4.40E-11
FHA	9.30E-11	1.18E-11	6.01E-12	1.24E-12	4.71E-12	4.53E-12	3.44E-10
OGF	1.89E-09	2.01E-11	6.30E-12	2.21E-11	1.95E-11	2.28E-11	6.35E-09
SA	2.53E-07	2.67E-09	8.39E-10	2.88E-09	2.55E-09	3.04E-09	8.56E-07

Table 4-25 Maximum effective dose due to ingestion for infants nearest country (Ireland) – breakdown by food type

Reference accident scenario	Effective dose (Sv)						
	Cow's milk	Beef	Sheep meat	Green veg	Root veg	Fruit	Cow's milk products
LOCA	4.56E-11	9.30E-14	1.03E-14	2.43E-13	9.83E-14	1.74E-13	9.14E-11
FHA	1.163E-10	4.64E-12	9.48E-13	5.75E-13	1.92E-12	2.91E-12	3.38E-10
OGF	9.46E-09	2.28E-11	2.85E-12	2.94E-11	2.26E-11	4.19E-11	1.78E-08
SA	1.27E-06	3.04E-09	3.82E-10	3.85E-09	3.00E-09	5.62E-09	2.40E-06

5 Emergency planning and countermeasures

5.1.1 Mitigation of the environmental impacts of accidental releases is achieved through the implementation of emergency arrangements, and the utilisation of appropriate countermeasures.

5.2 UK emergency planning arrangements

5.2.1 UK emergency arrangements have been formulated over many years taking into consideration learning and recommendations from both nuclear and non-nuclear events. National doctrine has been established to provide a framework for all civil defence arrangements under the Civil Contingencies Act 2004. In Wales, a dedicated government team supports multi-agency co-operation and engagement with the UK government on issues relating to civil protection and emergency preparedness.

5.2.2 The Radiation (Emergency Preparedness and Public Information) Regulations 2001 (REPPIR) is the main set of UK regulations regulating the emergency arrangements at UK nuclear power plants.

5.2.3 REPPIR establishes a framework of emergency preparedness measures to ensure that members of the public are:

- properly informed and prepared, in advance, about what to do in the unlikely event of a radiation emergency occurring; and
- provided with information if a radiation emergency actually occurs.

5.2.4 REPPIR does not prescribe the actions that a nuclear power plant operator must take in an emergency but it does require adequate on-site and off-site emergency plans to be written to deal with reasonably foreseeable radiation emergencies. Response plans for radiation emergencies are expected to fit within the broader resilience plans at national and local levels to give a robust command and control structure for all potential emergencies.

5.2.5 Under REPPIR the ONR is responsible for the determination of an off-site emergency planning area (being the area within which, in ONR's opinion, any member of the public is likely to be affected in the event of a radiation emergency). REPPIR also provides for prior information to be distributed to the public within such off-site emergency planning areas.

5.2.6 The local authority is required to prepare an off-site emergency plan according to ONR's determination for the defined emergency planning areas with the purpose of protecting the public and minimising any potential radiation exposures. This plan will identify the appropriate protection measures which can be taken, such as sheltering, the taking of stable iodine tablets and evacuation, in order to reduce radiation exposures to members of the public within all or parts of this area.

5.2.7 The local resilience plans would be developed via close coordination between the operator of the Power Station, local authority, local emergency services and other agencies, as required.

5.3 Intervention levels established for different types of countermeasures

5.3.1 In the event of a radiation emergency, mitigation is provided by considering the potential radiological protection benefits, the practical implications and the potential harm of any countermeasures that might be advised.

5.3.2 Public Health England has recommended emergency reference levels of doses for the justification of countermeasures to protect the public [RD23]. These are used to identify which actions will be most suitable in specific circumstances. For each countermeasure, there is a lower and upper reference level of dose averted by the countermeasure. Below the lower level, the countermeasure is unlikely to be worthwhile; above the upper level, it is likely to be worthwhile.

5.3.3 During a radiation emergency, health countermeasures need to be implemented promptly in order to maximise the level of protection provided to members of the public. When considering early countermeasures for an off-site radiation emergency, the primary ways to protect the public are to take one or more of the following actions:

- to shelter;
- to evacuate; and/or
- to administer stable iodine (for operating reactor sites).

5.3.4 The emergency reference levels for these countermeasures are shown in table 5-1.

Sheltering

5.3.5 Sheltering refers to staying inside, with doors and windows closed and ventilation systems turned off. Sheltering provides a degree of shielding against external radiation exposure (depending on the material of the building) and reduces the exposure to inhaled particles.

5.3.6 There are four main situations for which sheltering will be the optimum countermeasure:

- a release consisting mainly of radioisotopes of noble gases (to reduce the external dose);
- a release which will result in relatively low doses;
- a release which will result in very large short-term doses, for which evacuation could not be carried out in advance of the release; and
- circumstances in which evacuation either is not possible or will entail considerable risk to evacuees [RD23].

Evacuation

5.3.7 Where the risk to public health posed by an off-site release of radioactive contamination has been identified or is predicted through radiation monitoring/modelling it may be decided to evacuate the affected areas. The primary purpose of evacuation is to protect the population against the internal and external exposure to radionuclides in the air or deposited on the ground.

5.3.8 Evacuation is the only countermeasure which has the potential to prevent virtually all exposure to a release. However, this is only achieved if the evacuation is carried out before the release occurs. While people are in transit their protection against external irradiation and inhalation is likely to be much less than the protection they will receive from remaining inside typical UK residential dwellings [RD23].

Administration of stable iodine tablets

5.3.9 The administration of stable iodine reduces the exposure from radioactive iodine. Stable iodine can significantly reduce the exposure to radioactive iodine because once the thyroid is 'flooded' with non-radioactive iodine, thyroid-uptake of radioactive iodine will be blocked and radioactive iodine will be expelled more quickly. This countermeasure is only relevant to facilities where radioactive iodine is a potential hazard.

5.3.10 The administration of stable iodine is likely to be in conjunction with sheltering or evacuation to ensure the most effective countermeasure strategy.

Table 5-1 Recommended emergency reference levels for early countermeasures

Countermeasure	Organ	Dose averted (mSv)	
		Lower	Upper
Sheltering	Whole body	3	30
Evacuation	Whole body	30	300
Stable iodine	Thyroid (organ dose)	30	300

Food safety countermeasures

5.3.11 In the event of a radiation emergency, precautionary food safety advice and, if necessary, implementation of food restriction orders will be provided by the Food Standards Agency (FSA). Advice from the FSA may cover different geographical areas and different time periods from other countermeasures. The criteria for intervention for food safety issues (at least initially) will be the Council Food Intervention Levels laid down by the European Union. These set maximum permitted levels of radioactivity in foodstuffs and animal feeding stuffs. The FSA can impose statutory restriction orders, made under the Food and Environment Protection Act 1985.

5.3.12 Council Food Intervention Levels for milk, baby foods and other foodstuffs are listed in table 5-2. Note that these are radioactivity concentration-based rather than dose-based.

Table 5-2 Council Food Intervention Levels for food countermeasures

Radionuclide	Dairy produce and liquid foods (Bq/l)	Other foods (Bq/kg)	Minor foods (Bq/kg)	Baby foods (Bq/kg)
Alpha emitting isotopes	20	80	800	1
Strontium Isotope	125	750	7,500	75
Iodine Isotopes	500	2,000	20,000	150
All other Nuclides of Half-life > 10 days e.g. Cs-137	1,000	1,250	1,250	400

Recovery countermeasures

5.3.13 Recovery countermeasures refer to the countermeasures that would be implemented to protect both individuals and the wider public from longer-term, chronic risks. There are two main approaches to dose reduction (other than countermeasures relating to food) that can be employed in the recovery phase; decontamination and restriction of access [RD24].

5.3.14 Decontamination techniques reduce exposure by treating contaminated areas directly and include such techniques as removing contaminated materials from the area and redistributing or fixing radionuclides so that they are less available to contribute to exposure [RD24].

5.3.15 Restricted access measures reduce exposure by removing people from areas of contamination, or by controlling the time spent in such areas [RD24].

5.4 UK emergency response arrangements

5.4.1 During a radiation emergency there are broadly four discrete but interconnected tiers, which work together to ensure an effective response. These tiers are as follows:

- site level – on which the radiological release has occurred or is expected;
- local strategic level – from which the local level multi-agency strategic response will be coordinated;
- national level – which includes central government engagement through Cabinet Office Briefing Room as well as, where appropriate, support from national agencies and individual departments; and
- international level – which includes organisations such as the IAEA as well as foreign governments with whom the UK Government will liaise to notify of the emergency and potentially request assistance from.

5.4.2 In the event of a radiation emergency in Wales, the UK Government is responsible for overall policy and strategy, and command and control mechanisms will remain through the Strategic Coordination Centre and local mechanisms, including set-up and establishment of the Emergency Control Centre Wales. Responsibilities of the Welsh Government will include:

- keeping the Minister and other members of the Welsh Cabinet fully informed of all aspects of the management of the emergency;
- helping keep local and other authorities, and the public informed;
- advising central government on any adjustments to priorities or redeployment of resources necessary to meet Welsh needs; and
- acting as a central reporting point for local agencies where this can assist central government.

Exchange of information with other countries

5.4.3 In the event of a radiation emergency in the UK, the Department for Business, Energy and Industrial Strategy (BEIS) as the designated Competent Authority, will lead on engagement with multi-national organisations such as the IAEA or EU.

5.4.4 The UK is party to the IAEA Convention on Early Notification of a Nuclear Accident (18 November 1986 INFCIRC/335); this obliges the UK to target a two-hour deadline for informing the IAEA of a ‘general emergency’ at a UK plant with a potential for transboundary consequences. The UK is a signatory to Council Decision 87/600/EURATOM on Community arrangements for the early exchange of information in the event of a radiation emergency; this ensures that the Commission and Member States are promptly informed in the event of a radiation emergency, and sets out the information which is to be provided. The EC Urgent Radiological Information Exchange system provides an exchange platform for managing the graded response between Member States.

5.4.5 BEIS will also undertake initial notification of the emergency to countries with which the UK has relevant bi-lateral agreements, such as Belgium, Denmark, Netherlands, France, Ireland, Norway and Russia.

5.4.6 In the event of a radiation emergency the Radioactive Incident Monitoring Network will be used to collect, collate and communicate data about off-site radiation and contamination levels.

5.4.7 As part of the IAEA Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, BEIS will coordinate UK international assistance arrangements both in the case of a domestic or overseas event, such as through the Response and Assistance Network managed by the IAEA.

5.5 Environmental monitoring in the event of a radiation emergency

5.5.1 Specialist resources and equipment are available to undertake environmental and personal radiation monitoring following a radiation emergency in the UK, or in response to an overseas radiation incident. Within the UK, responsibilities for radiation monitoring in the event of a radiation emergency lie with a number of organisations. Public Health England Centre for Radiation, Chemical and Environmental Hazards is responsible for the overall co-ordination of the activities of organisations undertaking radiation monitoring.

5.5.2 As part of nuclear site licensing requirements and REPPIR, the operator of the Power Station would have arrangements in place that would detail the environmental monitoring that would be carried out in the event of an emergency situation. These arrangements would document what off-site surveillance would be undertaken, including dose rate monitoring and sampling of water, soil and identified items within the food chain. The off-site survey would detail the extent of initial environmental monitoring, with the ability to extend the monitoring out to a further distance depending on the extent of the emergency situation.

5.5.3 The principal responsibilities of other organisations are identified under these key monitoring functions.

- People monitoring – the health services locally are responsible for activating local facilities for monitoring in relation to people, specifically to provide reassurance to members of the public.
- Environmental monitoring – the Environment Agency and Natural Resources Wales in England and Wales have contractors who carry out environmental monitoring programmes in support of their regulatory responsibilities.
- Food monitoring – the FSA is responsible for arrangements for monitoring and food sampling and assessing the results to define any area to be subject to food advice and controls.
- Water monitoring – utility companies and authorities are responsible for ensuring the potability of drinking water supplied to their customers, including its radioactivity content, and identifying potentially contaminated water supplies.

6 Impact assessment of accidental releases

6.1.1 The impact assessment methodology for authorised discharges of radioactivity described in chapter B14 (Radiological effects) (Application Reference Number: 6.2.14) is not appropriate for accidental releases because the latter have a very low probability of occurrence ($<10^{-5}$/yr). The environmental impact of accidental releases can be described by the potential doses incurred and the potential for the release to require the implementation of countermeasures.

6.2 Impact assessment methodology

Value of receptors

6.2.2 One type of receptor has been identified for this assessment, namely members of the public exposed to radiation, who are judged to be of high value/sensitivity.

Assessment of magnitude and significance

6.2.3 For the purpose of environmental impact assessment, the doses for levels of countermeasures that may be required to mitigate off-site impacts have been selected as the basis of the impact and significance criteria (see table 6-1). The most severe of these refers to the requirement to implement evacuation, the most disruptive countermeasure response. Lower dose level bands can also be set to correspond to levels at which sheltering should be implemented and sheltering should be considered. The UK public dose limit is set as the dose equivalent to a negligible impact.

6.2.4 This assessment scale is considered appropriate when considering the magnitude of the UK public dose limit for all sources (1mSv/yr) and the typical background radiation exposure of the UK population (2.8mSv/yr – see chapter D14, Application Reference Number: 6.4.14).

6.2.5 For the impact assessment, the magnitude of change is a measure of the scale or extent of the change, irrespective of the value of the receptor(s) affected. The criteria used to determine the magnitude of change are set out in table 6-1 and are derived from the countermeasure levels described in table 5-1.

6.2.6 The degree of significance is influenced by the value of a receptor and the magnitude of the predicted impact. As all receptors for which impacts are assessed are judged to be of high sensitivity, then the significance can also be assessed using the ranges shown in table 6-1.

Table 6-1 Criteria for impact and significance of effect assessment

Magnitude of impact and significance effect	Dose resulting from accident (mSv)	Basis of level
Large magnitude Major significance	300	Dose at which evacuation should be implemented
Medium magnitude Moderate significance	30	Dose at which evacuation should be considered and sheltering should be implemented
Small magnitude Minor significance	3	Dose at which sheltering should be considered
Negligible magnitude Negligible significance	1	UK public dose limit

6.3 Assessment

6.3.1 From the doses assessed for the reference accidents presented in section 4.4, the following impact assessment is developed for the most exposed members of the local population. This is summarised in table 6-2.

6.3.2 The three DBAs all result in low off-site releases and resulting doses are below 1mSv and are judged as being of negligible impact and negligible significance.

6.3.3 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.

6.3.4 Doses in the nearest country (Ireland) are two to three orders of magnitude lower than this (see section 4.4). The resulting impact and significance 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.

6.3.5 The countermeasure levels used as the basis for the impact assessment also correspond to the doses that would be averted if that countermeasure was implemented. Because of this, the results described in table 6-2 also describe the residual impact.

Table 6-2 Impact and significance assessment of accidental releases

Accident	Local reference group maximum total dose (mSv)	Impact	Significance
Loss of Coolant Accident	6.3E-5	Negligible	Negligible
Fuel Handling Accident	5.7E-4	Negligible	Negligible
Off-gas system Failure	2.5E-2	Negligible	Negligible
Severe Accident (containment leakage from Drywell (failed RPV))	7.4E-1	Negligible	Negligible

6.3.6 The DBAs and SA identified for the UK ABWR are all judged to result in negligible impact and be of negligible significance to the most exposed members of the local population (should they occur). As the resulting radioactivity concentrations, and hence doses, will be much lower at the relevant distances, the impacts to members of the public in other countries will also be judged be of negligible impact and significance.

7 References

Table 7-1 Schedule of references

ID	Reference
RD1	United Nations Economic Commission for Europe. 1991. *The Convention on Environmental Impact Assessment in a Transboundary Context*. [Online]. [Accessed: 3 July 2017] Available from: https://www.unece.org/fileadmin/DAM/env/eia/documents/legaltexts/Espoo_Convention_authentic_ENG.pdf.
RD2	Office for Nuclear Regulation. Generic design assessment of the UK ABWR. [Online]. [Accessed: 15 January 2018] Available from: http://www.onr.org.uk/new-reactors/uk-abwr/index.htm
RD3	Office for Nuclear Regulation. Nuclear site licensing. [Online]. [Accessed: 3 July 2017] Available from: http://www.onr.org.uk/civil-nuclear-reactors/licensing.htm .
RD4	Hitachi-GE Nuclear Energy, Ltd. UK Advanced Boiling Water Reactor. [Online]. [Accessed: 3 January 2018] Available from: http://www.hitachi-hgne-uk-abwr.co.uk/gda_library.html.
RD5	International Atomic Energy Agency (IAEA). 2010. *Deterministic Safety Analysis for Nuclear Power Plants. Specific Safety Guide*. IAEA Safety Standards Series No. SSG-2. IAEA: Vienna.
RD6	International Atomic Energy Agency (IAEA). 2010. *Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants*. Specific Safety Guide. IAEA Safety Standards Series No. SSG-3. IAEA: Vienna.
RD7	International Atomic Energy Agency (IAEA). 2010. *Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants*. Specific Safety Guide. IAEA Safety Standards Series No. SSG-4. IAEA: Vienna.
RD8	US Nuclear Regulatory Commission. Issued Design Certification - Advanced Boiling-Water Reactor (ABWR). [Online]. [Accessed: 3 July 2017]. Available from: https://www.nrc.gov/reactors/new-reactors/design-cert/abwr.html.
RD9	Office for Nuclear Regulation (ONR). 2014. *Safety Assessment Principles for Nuclear Facilities*. London: The Stationery Office.
RD10	Clarke, R.H. 1979. *The first report of a working group on atmospheric dispersion. A model for short and medium range dispersion of radionuclides released to the atmosphere*. NRPB-R91. National Radiological Protection Board: Chilton.
RD11	Jones, J.A. 1981. *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.

ID	Reference
RD12	Jones, J.A. 1983. *The fifth report of a Working Group on Atmospheric Dispersion: Models to Allow for the Effects of Coastal Sites, Plume Rise and Buildings on Dispersion of Radionuclides and Guidance on the Value of Deposition Velocity and Washout Coefficients*. NRPB-R157. National Radiological Protection Board: Chilton.
RD13	Smith, J.G., Bedwell, P., Walsh, C. and Haywood, S.M. 2004. *A Methodology for Assessing Doses from Short-Term Planned Discharges to Atmosphere.* NRPB-W54. National Radiological Protection Board: Chilton.
RD14	The Centre for Environment, Fisheries and Aquaculture Science. 2005. *Radiological Habits Survey: Wylfa 2004*. Environment Report RL 02/05.
RD15	The Centre for Environment, Fisheries and Aquaculture Science. 2010. *Radiological Habits Survey: Wylfa 2009*. Environment Report RL 03/10.
RD16	The Centre for Environment, Fisheries and Aquaculture Science. 2014. *Radiological Habits Survey: Wylfa 2013*. Environment Report RL 03/14.
RD17	Smith, K.R and Jones, A.L. 2003. *Generalised habit data for radiological assessments*. NRPB-W41. National Radiological Protection Board; Chilton.
RD18	Jones, J.A. 1981. *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.
RD19	Radiological Protection Institute of Ireland. 2013. *Proposed nuclear power station in the UK – Potential radiological implications for Ireland*. RPII 13/01.
RD20	Smith, J.G. and 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.
RD21	International Commission on Radiological Protection (ICRP). 2012. *Compendium of Dose Coefficients based on ICRP Publication 60.* ICRP Publication 119. Ann. ICRP 41 (Suppl).
RD22	International Commission on Radiological Protection (ICRP). 1995. *Age-dependent Doses to Members of the Public from Intake of Radionuclides - Part 4 Inhalation Dose Coefficients*. ICRP Publication 71. Ann. ICRP 25 (3-4).
RD23	Documents of the NRPB. 1990. *Statement on Emergency Reference Levels*. Volume 1, No 4. National Radiological Protection Board: Chilton.
RD24	Documents of the NRPB. 1997. *Intervention for Recovery after Accidents*. Volume 8, No 1. National Radiological Protection Board: Chilton.