The Sizewell C Project

SZC Co.'s Response to the Secretary of State's Request for Further Information dated 18 March 2022: Appendix 6 – Extract of Article 37 Submission for Sizewell C – Chapter 6

Revision: 1.0

April 2022

SizewellC
Doing the power of good for Britain
EDF CGN

6 UNPLANNED RELEASES OF RADIOACTIVE EFFLUENTS

6.1 Review of Accidents of Internal and External Origin which could result in Unplanned Releases of Radioactive Substances

6.1.1 UK EPR™ safety principles

630. Nuclear safety includes all the technical provisions and organisational measures relating to the design, construction, commissioning, operation and decommissioning of installations. It involves the use of fissile material, or the transportation, storage and disposal of radioactive waste, intended to control the hazard and prevent accidents and to ensure risks are ALARP i.e.:

- to ensure normal plant operation, while keeping the radiological impact for workers and the public As Low As Reasonably Practicable (ALARP), below the limits prescribed by the Ionising Radiations Regulations 2017; and
- to prevent faults and hazards leading to failures of safety functions (control of reactivity, maintain heat removal, maintain containment) and where this is not practicable, limit the consequences of any possible fault or hazard by taking measures to control radiation risks to ensure that no individual bears an unacceptable risk of harm.

631. In terms of normal operation, the methods implemented for the UK EPR™ (including its spent fuel and ILW interim storage facilities at SZC) and the resulting impacts are described in detail in Chapter 3, Chapter 4 and Chapter 5.

632. Nuclear and radiological risks are controlled using measures which ensure that the plant can be managed, by assessing a comprehensive range of potential faults and hazards and demonstrating the maintaining of safety functions related to:

- Reactivity control;
- Removal of residual thermal power and decay heat; and
- Containment and shielding of radioactive substances.

633. With reference to risk management, the measures put in place during the plant design, construction, commissioning, operation and decommissioning stages, cover:

- risk prevention to reduce the probability of occurrence of initiating events;
- monitoring and detection of operating anomalies; and
- limiting consequences with the aim of making residual risks acceptable with regard to personnel, the public and the environment.

634. In order to guarantee a high level of safety, a large number of independent measures are implemented. This collection of measures results from the application of the ‘defence in depth’ concept, which involves systematically taking plant or human failures into account and providing several levels of protection against potentially significant faults and hazards.

635. The UK EPR™ safety process, which has been implemented at the design stage, is based on defence in depth over five levels, based on the detail in IAEA Safety Requirements SSR2/1 [1]l.

- Level 1 is a combination of design, quality assurance and control margins aimed at preventing the occurrence of abnormal operating conditions or plant failures.

- Level 2 consists of the implementation of protection devices which make it possible to detect and correct the effects of deviations from normal operation or the effects of system failures. This defence level is aimed at ensuring the integrity of fuel cladding and that of the primary cooling system so as to prevent accidents.

- Level 3 consists of safeguard systems, protection devices and operating procedures which make it possible to control the consequences of accidents that may occur, so as to contain radioactive material and prevent the occurrence of severe accidents.

- Level 4 comprises measures aimed at preserving containment integrity and controlling severe accidents.

- Level 5 includes, in the event of the failure of previous levels of defence, all measures for protecting the public against the effects of significant radiological releases.

636. A systematic, comprehensive analysis of all potential faults and hazards is carried out to verify that even in these situations, defined safety objectives are met and consequences for the environment and populations are minimised so far as is reasonably practicable and remain below the thresholds prescribed by national and international authorities.

637. Design Basis Initiating Faults (DBIFs) are identified via identification of Postulated Initiating Events (PIEs) – plant failures or human errors, which if unmitigated could challenge some or all of the main safety functions and hence lead to an unacceptable off-site release. Once identified, DBIFs are recorded in a 'Fault and Protection Schedule' (F&PS), which list faults and the means by which they are protected against / mitigated.

638. The hazards approach is based on ensuring that hazards cannot lead to onerous DBIFs. Internal hazards are those originating inside the licenced site but outside the primary circuit. These include hazards such as fire, flood, etc and if these hazards are not controlled they have the potential to initiate DBIFs and simultaneously damage the Structures, Systems and Components that are included in the design to manage initiating events. Internal hazards are also managed in accordance with the defence in depth concept in paragraph 6. In addition, the divisionalisation / segregation of safety trains plays a significant role in ensuring that sufficient safety trains remain available in order that reactivity control, containment and cooling can be maintained following the occurrence of an internal hazard and any initiated reactor fault.

639. External hazards are those hazards of man-made original (e.g. aircraft crash) or natural origin (e.g. seismic) which are initiated outside the boundary of the Nuclear Licenced site. If external hazards are not adequately catered for in the design, then they could simultaneously initiate reactor faults and cause wide spread damage to the Structures, Systems, and Components that are included in the design to manage DBIFs. The approach to the control of the hazard is to demonstrate robustness to design basis external hazard events (i.e. the magnitudes that occur at return frequencies of 1E-5/year for man-made hazards, and 1E-4/year for natural hazards). In addition, the safety case provides additional demonstration that 'cliff edge' effects leading to the inability to control reactivity, maintain containment and cooling, do not take place even if higher magnitude events that are just beyond the design basis occur.

640. With regard to reducing the potential consequences of DBIFs and hazards, safety performance is improved in the following four main ways:

- By accounting for, eliminating and where this is not possible reducing the frequency of initiating events (which cause transients) liable to occur during the different states which the reactor may encounter during operation (including full power and shutdown states, and states with the core completely unloaded in the spent fuel pool). The defence in depth approach is enhanced by taking internal hazards into account on a deterministic basis in accordance with design principles similar to those used for simple initiating events.

- By taking into account external hazards at high severity levels, whether the hazards are of human origin (aircraft crashes, explosions etc.) or of natural origin (earthquakes, extreme temperatures, flooding etc.). In addition to their direct effects, these hazards are studied from the point of view of the damage they may cause on non-protected structures and equipment, inside or outside the plant.

- By taking severe accidents (such as a core melt accident) into account at the design stage and implementing physical measures to ensure "practical elimination" of events and sequences that could have a significant radiological impact on the environment during the power plant's service life. For events which cannot be prevented by design, the probability of environmental releases is minimised by strengthening the containment, including conditions which could lead to containment bypass.

- By use of Probabilistic Safety Analysis (PSA) at the concept design phase, to confirm the design approach and identify the multiple failure sequences that should be considered in the design basis, so as to prevent core meltdown accidents.

641. The safety assessment performed in support of the GDA of a single UK EPR™ (as described in Section 0.2.2) is based on the well-established deterministic methods, augmented by probabilistic methods using appropriate numerical targets and analysis. The main deterministic method is adoption of a defence in depth approach and the concept of independent physical barriers against the escape of radioactivity. The principal quantitative safety targets are outlined below and are based upon UK practice:

- The collective dose to workers shall be ALARP;

- Doses to workers during normal operation of the plant will not exceed UK statutory limits. The dose limits to be applied, shall be those specified in the Ionising Radiations Regulations 2017.

- Doses to the public during normal operation of the plant will not exceed UK statutory limits. The dose limits to be applied shall be those specified in the Ionising Radiations Regulations 2017.

- An annual whole body effective dose for individual employees and contracted workers involved in the operation of SZC due to normal operation of a reactor unit shall not exceed 10mSv.

- The maximum dose to an individual off-site (member of the public) due to normal operation of SZC shall not exceed $0.3 mSv.y^{-1}$ and the combined dose from discharges from SZC, and the existing reactors at the site (Sizewell A and Sizewell B) shall not exceed $0.5 mSv.y^{-1}$.

- The risk of an individual worker fatality due to exposure to radiation from an on-site accident (all facilities) will be below $1x10^{-6}$ y^{-1} and/or demonstrated as ALARP.

- The risk of fatality of any person off-site (member of the public) due to exposure to radiation from on-site accidents (all facilities) will be below $1x10^{-6}$ y^{-1} and at least demonstrated as ALARP.

- The total predicted frequency of accidents (from all facilities) resulting in more than 100 fatalities (either immediate or delayed) of members of the public will be below $1x10^{-7}$ y^{-1} and/or demonstrated as ALARP.

642. The primary purpose of the GDA was to demonstrate that the generic aspects of a UK EPR™ unit have been conservatively assessed and the reactor technology is acceptable to be built in the UK.

643. A number of additional safety reports will be produced for SZC taking into account any site specific considerations, the twin UK EPR™ units to be built, and additional nuclear facilities such as the ILW ISF and ISFS facility. This will build on the arguments in the GDA and demonstrate that people and the public are protected from the harmful effects of ionising radiation.

6.1.2 Development of the safety reports for SZC

644. The GDA for the UK EPR™ was submitted by AREVA and EDF SA in the form of a Generic Pre-Construction Safety Report and Generic Pre-Construction Environmental Report for a single UK EPR™ reactor unit. This demonstrated that a series of fundamental safety principles are applied to the design, construction, commissioning, operation and eventual decommissioning of a UK EPR™ reactor.

645. Assessment of the UK EPR™ was completed in July 2011 by the regulators, the Office for Nuclear Regulation (ONR) covering safety and security, and the Environment Agency (EA) covering waste management and environmental protection. The ONR and the EA granted DAC and SoDA for the UK EPR™ Reactor Design in December 2012.

646. A safety case will be produced for SZC encompassing the twin units and additional support facilities on site. This will demonstrate that the nuclear and radiological risks are ALARP taking account of the site specific nature and additional facilities not covered under the GDA. SZC has a phased approach to the development of safety cases which allows the identification, assessment and mitigation of hazards and their associated risks. The phases of the safety case are aligned with the key phases of the design, construction, commissioning and operation of facilities on the site and are summarised as:

- Pre-Construction Safety Report (PCSR). A series of reports that are prepared during detailed design and are submitted prior to the construction of key safety related structures.
- Pre-Commissioning Safety Report (PCmSR). Prepared during construction of the facility and submitted before the start of plant and process commissioning (non-active commissioning).
- Pre-Operation Safety Report (POSR). Prepared during non-active commissioning and submitted before the start of nuclear operations (active commissioning).
- Station Safety Report. Documents the Safety Case throughout the operational phase of the site.

647. The safety case will be produced for SZC in line of the recommendations of the ONR. The ONR is the regulatory body with responsibility for regulating the nuclear power industry in the UK. Each nuclear site licence includes the requirement to produce and manage safety cases. The safety case will be subjected to rigorous and comprehensive internal assessment by the site operator.

648. The ONR will then assess the SZC safety case as part of the licensing process for the project. This independent regulatory scrutiny examines both plant design and the management systems supporting the lifecycle of the station. It provides additional confidence that the claims made within the safety case are valid and that the safety targets referenced above will be met. Once the safety case has been implemented the ONR will undertake a programme of inspection and review. This programme is designed to demonstrate that the plant and systems and operators correctly implementing them are performing as expected and that the claims that are made within the safety case remain valid. Deviations are the subject of corrective action which could include enforcement activities.

649. It should be noted that HPC, for which Sizewell C is a replica, is successfully undergoing construction, with the site specific PCSR approved, and PCmSR under development.

6.1.3 Design scope

650. The object of the design scope is to define the events taken into account in the design basis and to categorise them. The PSA is a confirmatory step which is used to support the robustness of the design.

651. The initiating events considered are grouped and are dealt with differently. In terms of the design process, the overall approach is the same:

- Definition of the design basis list of plant-based faults, internal and external hazards, and events/sequences with consideration of combinations.
- Quantification of the event/sequence impacts, the results being used for the design of systems and structures and/or the demonstration that the safety requirements are met.
- Design verification which completes the safety analysis by providing a further demonstration that the safety requirements are met. It invariably includes the use of PSA and in some cases a deterministic verification is carried out. This step can result in design feedback.

6.1.4 UK EPR^{TM} Reactor Faults – Design Basis Analysis

652. The safety approach applied to the UK EPR^{TM} requires consideration of a number of representative reactor faults and enveloping conditions, which could occur during normal operation and various associated reactor states. The relating initiating events are grouped together in four categories based on their estimated frequency of occurrence and their impact.

653. On this basis, events are grouped into four Plant Condition Categories (PCC's) as follows:

- PCC-1 which includes all normal operating conditions characterised by initiating events whose estimated frequency of occurrence is greater than 1 per year.
- PCC-2 which includes design basis transients, characterised by initiating events with an estimated frequency of occurrence in the range of 10^{-2} to 1 per year.
- PCC-3 which includes all design basis incidents characterised by initiating events with an estimated frequency of occurrence is within the range of 10^{-4} to 10^{-2} per year.
- PCC-4 which includes all design basis accidents characterised by initiating events with a frequency of occurrence is within the range of 10^{-6} to 10^{-4} per year.

654. Identification of these events and their classification by category determines the design of systems intended to control them, preventing unacceptable consequences for the plant or the environment. The transients of categories 2, 3 and 4 for a single UK EPR^{TM} unit, ISFS facility and ILW ISF are listed in the following tables.

655. The PCC design basis transients consider a number of operating conditions or 'states' which are summarised below.

- State A. Power states and hot and intermediate shutdown (P>130bar). In these shutdown states, all the necessary automatic reactor protection functions are available as in the power state.
- State B. Intermediate shutdown above 120°C (P<130bar). State B covers all shutdown states during normal plant operation, where primary heat is removed by the steam generators.

- State C. Intermediate and cold shutdown with safety injection system/residual heat removal.
- State D. Cold shutdown with the reactor cooling system open so that the steam generators cannot be used for decay heat removal.
- State E. Cold shutdown with the reactor cavity flooded for refuelling.
- State F. Cold shutdown with the core fully unloaded.

6.1.4.1 Category 2 events (PCC-2): design basis transients

656. The Category 2 transients studied in the Generic Design Assessment of a single UK EPR^{TM} unit are listed in Table 6-1.

Table 6-1 Category 2 events (PCC-2): design basis transients

Design basis transients with internal causes[1]
Main feed water system malfunction causing a reduction in feed water temperature
Main feed water system malfunction resulting in an increase in the feed water flow rate
Excessive increase in secondary steam flow
Spurious turbine trip
Loss of condenser vacuum
Short-term loss of off-site power (≤ 2 hours)
Loss of normal feed water flow (loss of all main feed water system pumps, start-up and shutdown pump)
Partial loss of core coolant flow (loss of one reactor coolant pump)
Uncontrolled rod cluster control assembly bank withdrawal at power
Uncontrolled rod cluster control assembly bank withdrawal from hot zero power conditions
Rod cluster control assembly misalignment up to rod drop, without control system action
Start-up of an inactive reactor coolant loop at an incorrect temperature
Chemical volume and control system (RCV) malfunction resulting in an uncontrolled increase or decrease in boron concentration in the reactor coolant

[1] Where the status of the reactor is not shown, the event is assumed to be analysed for at power state

Design basis transients with internal causes[1]
RCV malfunction causing increase or decrease in reactor coolant inventory
Primary side pressure transients (spurious operations of pressuriser sprays or heaters)
Uncontrolled reactor cooling system level drop (states C, D)
Loss of one cooling train of the safety injection system/residual heat removal system in residual heat removal mode (states C, D)
Loss of one train of the fuel pool cooling system or of a supporting system
Spurious reactor trip (state A)

6.1.4.2 Category 3 events (PCC-3): benchmark incidents

657. The Category 3 incidents studied in the preliminary safety report are listed in Table 6-2.

Table 6-2 Category 3 events (PCC-3): benchmark incidents

Design basis incidents with internal causes[2]
Small steam or feed water system piping failure (DN<50), including break of connecting lines to a steam generator (DN<50)[3] (states A, B)
Long-term loss of off-site power (>2 hours)
Inadvertent opening of a pressuriser safety valve (state A)
Inadvertent opening of a steam generator relief train or a safety valve (state A)
Small break Loss Of Coolant Accident (LOCA) (≤ DN 50), including a break on the extra boration system injection line (states A, B)
Steam Generator Tube Rupture (1 tube)
Inadvertent closure of one or all main steam isolation valves
Inadvertent loading of a fuel assembly in an incorrect position

[2] Where the status of the reactor is not shown, the event is assumed to be analysed for an initial power state

[3] DN - Nominal diameter in mm

Design basis incidents with internal causes[2]
Forced reduction of reactor coolant flow (4 pumps)
Leak in the gaseous or liquid waste processing systems
Uncontrolled rod cluster control assembly bank withdrawal (states B, C, D)
Uncontrolled single control rod withdrawal
Long term loss off-site power, fuel pool cooling aspects (state A)
Loss of one train of the fuel pool cooling system or supporting system (state F)
Isolatable piping failure on a system connected to the fuel pool

6.1.4.3 Category 4 events (PCC-4): benchmark accidents

658. The Category 4 accidents studied in the PCSR are listed in Table 6-3.

Table 6-3 Category 4 events (PCC-4): benchmark accidents

Design basis accidents with internal causes[4]
Long term loss of off-site power in state C (>2 hours)
Main steam line break
Feed water system pipe break
Inadvertent opening of a steam generator relief train or of a safety valve (state B)
Spectrum rod cluster control assembly ejections
Intermediate or large break LOCA (up to surge line break – states A, B)
Small break LOCA (≤ DN 50), including a break on an extra boration system injection line (states C, D)
Reactor coolant pump seizure (locked rotor)
Reactor coolant pump shaft break

[4] Where the status of the reactor is not shown, the event is assumed to be analysed for an initial power state

Design basis accidents with internal causes[4]
Steam Generator Tube Rupture (2 tubes in 1 steam generator)
Fuel handling accident
Boron dilution due to a non-isolatable rupture of a heat exchanger tube
Rupture of systems containing radioactivity in the nuclear auxiliary building
Isolatable safety injection system break (≤ DN 250) in residual heat removal mode (states C, D)
Non-isolatable small break (≤ DN 50) or isolatable safety injection system break residual heat removal mode (≤DN 250), fuel pond drainage aspects (state E)

6.1.5 Multiple failure accidents under Risk Reduction Category A (RRC-A)

659. In addition to examining the incidental and accidental situations with simple initiating events, the scope of the analysis is extended to situations involving multiple failures based on probabilistic safety evaluation. The purpose of studying operating conditions involving multiple failures is to define specific measures, which may be manual actions intended to limit the risks of core melt associated with these scenarios. The accidental transients associated with RRC-A multiple failure, studied in the GDA are listed in Table 6-4.

Table 6-4 Category RRC-A internal accidental transients

Accidental transients with internal causes
Anticipated transient without scram through the mechanical blocking of the rods (state A)
Anticipated transient without scram due to the failure of the protection system (state A)
Total loss of off-site power and failure of the four main diesel generators (state A)
Total loss of the water supply to the steam generators (state A)
Total loss of the cooling chain and failure of the stand still seal system leading to a loss of primary coolant (state A)
LOCA up to 20cm2 and failure of the protection system for the safety injection signal activation (state A)
LOCA up to 20cm2 without medium head safety injection (state A)

Accidental transients with internal causes
LOCA up to 20cm2 without low head safety injection (state A)
Uncontrolled level drop and failure of the protection system for the activation of the safety injection system (state C and D)
Total loss of the cooling chain (state D)
Total loss of the ultimate heat sink for 100 hours (states A to C)
LOCA without medium head safety injection (states C and D)
LOCA outside containment on safety injection system and residual heat removal system train (states C and D)
LOCA outside containment on safety injection system and residual heat removal system train and failure of the automatic isolation signal (states C and D)
Non-isolatable homogenous boron dilution outside the volume control tank and failure of the operator's actions (states C and D)
Loss of the two main trains of the spent fuel pool cooling system during core refueling (state F)

6.1.6 Core melt accidents under category DEC-B

660. The purpose of some specific safety improvements made to the UK EPR^{TM} is to reduce the risk of core melt accidents involving perforation of the reactor vessel, to one tenth of that associated with the existing reactors, for which the risk is already extremely low. The risk of such an event occurring is estimated overall for the UK EPR^{TM} using a probabilistic approach, at:

- 1 in 100,000 per reactor per year when taking into account all the reactor states and all types of event (internal events, internal and external hazards).
- 1 in 1,000,000 per reactor per year when taking into account internal events only, i.e. with internal and external hazards excluded.

661. The practical measures contributing to this reduction in risk are, for example:

- The physical separation of important safety systems into four compartments, to increase the reliability of these systems.
- The installation of a borated water tank inside the reactor building.
- Improvements to protection measures for the main external hazards (aircraft crashes, earthquakes, extreme temperatures, etc.).
- Improved diversification of support systems (diversified main diesel generators, diversified ultimate heat sink, etc.).

- An optimised human-machine interface, based on information from the most recent unit commissioned in France.

662. A core melt accident is still taken into account at the design stage even though the probability is extremely low. Measures are employed to prevent any premature failure of the containment, manage low pressure meltdown scenarios and reduce any associated impacts.

663. The specific aim is to eliminate the need to evacuate any population beyond the immediate vicinity of the power station, even in the event of a severe accident. This aim is achieved by the presence of a molten fuel spreading area, installed below the reactor vessel to enable cooling and a metal lined containment installed to minimise accidental releases to the environment.

664. The low pressure core meltdown scenarios studied in the PCSR cover all water loss events which could lead to exposure of the core and subsequent damage. Since safety injection is unavailable, the progress of an accident is dictated by the size of the break of the reactor coolant system if there is one and the heat removal capacity of the steam generators.

6.1.7 Consideration of additional safety related scenarios

665. The scenarios considered above relate to the presence of a single UK EPR™ unit. SZC will have two UK EPR™ units, ILW ISF and ISFS facility. The presence of these additional nuclear facilities has been considered to determine the degree to which they influence the identification and selection of reference accidents for the purpose of this submission. The current assumption is that, in terms of design basis accidents, any release from an accident in these additional facilities and that any release associated with an interaction between these facilities in an accident scenario, is bounded by the DEC-B reference accident for a single UK EPR™ unit. This assumption has been developed using the arguments presented in Section 6.1.7.1, Section 6.1.7.2 and Section 6.1.7.3.

6.1.7.1 Presence of two UK EPR™ units on the site

666. The site specific PCSR prepared for HPC did not identify any design basis accidents that relate to both units. Noting that the Nuclear Island of SZC is a replica of HPC the same argument therefore stands.

667. The interactions between the two UK EPR™ units have been explored in the probabilistic safety assessments that were undertaken to support the preparation of the HPC PCSR. These assessments show that interactions between the two UK EPR™ units do not make a significant contribution to the overall risk associated with the HPC site and that the numerical targets identified in Section 6.1.1 will be met and ensures risks are ALARP.

668. Interaction between the two units will be subjected to additional safety analysis and risk assessment during the further development of the safety reports described in Section 6.1.2. These developments will provide a comprehensive demonstration that the numerical targets will be met and ensures risks are ALARP.

6.1.7.2 Presence of an interim storage facility for spent fuel

669. The ISFS was excluded from the original GDA of the UK EPR™. Since the GDA the design of the ISFS has been developed.

670. After removal from the reactor, spent fuel will be initially stored in the spent fuel pool in the Fuel Building for an initial cooling period. The spent fuel will then be loaded into a MPC which will be sealed, drained, dried and filled with helium. The MPC provides the confinement barrier and is designed to ensure passive cooling of the spent fuel. The loaded MCP will then be transferred in a shielded transfer container (HI-TRAC) to the ISFS facility where it will be placed into a concrete storage shelter (Hi- STORM) which provides shielding and protects the MPC. The spent fuel will be stored in this way for up to 120 years. The process is reversible should it be

deemed necessary to return an MPC containing spent fuel to the fuel building for inspection. At the EoG, defueling of the reactors and transfer of the fuel from the reactor core to the fuel building pool is planned to be undertaken at the earliest safely practicable opportunity. After a period of cooling in the fuel building pool the final spent fuel assemblies will be transferred from the Fuel Building to the ISFS facility via a series of ISFS operational campaigns. At some point after all spent fuel has been removed from the Fuel Building, but well within the 120-year operation of the ISFS facility, the Fuel Building will be decommissioned. This will foreclose the ability to carry out reverse operations. To offset this risk, space on the SZC Site Plot Plan has been reserved for a Spent Fuel Inspection and Repackaging Facility (SFIRF). Prior to the start of decommissioning of the Fuel Building, the SFIRF will be operational and able to provide capability for the ISFS reverse operations including inspection and repackaging if required. Pending the availability of a Geological Disposal Facility (GDF), the SFIRF will ultimately be converted into a Spent Fuel Encapsulation Facility (SFEF) wherein spent fuel will be transported from the ISFS facility for repackaging prior to offsite disposal to the GDF.

671. A preliminary safety assessment for ISFS has been written into the HPC safety case, and incorporated in the RC2 design adopted into SZC, based on: identifying a suitable concept design; demonstrating the feasibility of implementing that concept within the existing Station design; and gaining confidence that the risks associated with operating the concept design will be tolerable and ALARP. This includes fault and hazard assessments allowing the process of screening faults into those which are reasonably foreseeable and those which are not.

672. The bounding transients identified in these assessments are listed in Table 6-5.

Table 6-5 ISFS bounding accidental transients

Bounding accidental transients associated with the ISFS
Loss of cooling fault during drying of the fuel canister prior to transfer to ISFS
2 m drop and topple of a HI-TRAC containing an MPC during handling in the Fuel Building Extension Building
Turbine disintegration impacting HI-STORMs in the ISFS facility

673. Loss of Cooling during drying of the fuel canister was deemed to be bounded by the loss of cooling in the spent fuel pool in the fuel building which is already covered by the GDA and earlier sections of this report.

674. Whilst in the case of turbine disintegration or 2 m drop and topple the probability of such an event occurring and resulting in a radiological release has been deemed as not reasonably foreseeable (~1 x 10^{-6} per year).

6.1.7.3 Presence of an interim storage facility for ILW

675. The ILW ISF was also excluded from the original GDA of the UK EPR^{TM}, but the design of the facility has since been developed.

676. The site will have a ILW for the safe storage of solid intermediate level radioactive waste generated during the operation and decommissioning of SZC. This will store the ILW until it can be disposed of to the GDF.

677. The design of the ILW ISF is in the form a shielded store, designed to interpose physical barriers and a proportionate number of safety measures between the ILW and the environment. During

the design development a number of bounding faults have been identified which are likely to have the greatest consequences while being credible enough to fall with the design basis.

678. The bounding transients identified are listed in Table 6-6.

Table 6-6 ILW Interim Storage Facility bounding accidental transients

Bounding accidental transients associated with the ILW Interim Storage Facility
Handling fault involving dropped packages within the facility
Internal fire involving multiple packages
Earthquake involving dropped packages within the facility

679. The ongoing development of the design will ensure the risks are such that events leading to a radiological release from operations in the ILW ISF will be eliminated and / or appropriately mitigated.

680. In addition, in the highly unlikely event that an event was the occur the accidental releases and associated impacts from these facilities will be enveloped by those identified for an UK EPR™ unit noting:

- the design of the waste packages - These packages are designed to remain leak tight should they drop following any handling fault or earthquake. In addition, packages are required to sustain fire loadings while maintaining the containment of radioactive materials;
- the design of the facility - the building is seismically qualified and the fire loadings are strictly limited by design.
- the timescale of operation - the ILW ISF is expected to be emptied and decommissioned promptly after the GDF is available for receiving new build ILW wastes. The radiological inventory is significantly smaller than the inventory contained elsewhere on the Sizewell C site.

6.1.7.4 Interactions between facilities in accident scenarios

681. The methodology adopted for the preparation and evolution of the SZC safety report is not expected to result in the identification of additional design basis accidents related to multiple facilities on the site. Interactions have been explored during preliminary probabilistic safety assessments.

682. Comprehensive safety analysis and risk assessment will be undertaken to demonstrate that the numerical targets identified in Section 6.1.1 will be met so far as to ensure risks are ALARP.

6.2 Reference Accidents Taken in Consideration

683. An assessment of the radiological consequences of the following design basis accidents (PCC-4) is presented:

- Fuel handling accident;
- Steam generator tube rupture; and

- LOCA.

684. These three design basis accidents are considered, which represent the worst design basis accidents, as they present the most significant consequences to the local reference group in terms of radiological impact.

685. In addition, a severe accident scenario (DEC-B), based on a core melt accident is also assessed.

686. Representative operating conditions, from the point of view of radiological consequences, are selected from the design basis accidents studied in the GDA PCSR of the UK EPR™. These set out the initial operating conditions and the limit conditions, such as the discharge type, path, height and operating modes, for SZC.

687. Among the ILW ISF and ISFS facility design basis faults, none are identified as a reference accident comparable with those of the reactors in terms of frequency, inventory released or resultant impact. The design is such that large scale releases are not conceivable in those situations and therefore it is considered appropriate not to undertake an assessment of the ILW ISF and ISFS facility design basis faults.

688. For atmospheric discharges, the different possible locations of leaks inside the plant were considered (containment, safeguard buildings, fuel building, nuclear auxiliary building, main steam and feed water systems, steam generators, etc.) in order to select representative cases.

689. For liquid waste, the precautions taken to ensure total containment in the event of an accident are described in Section 2.5, this includes the ability to recover radioactive effluent by sumps, retention pits and retention tanks. In addition, the core catcher provides the containment function in severe accident situations. On the basis of the above, and taking into account the considerations of ONRs Report on Fukushima [2], accidental releases of radioactive liquid waste into aquatic environments are therefore excluded from any further assessment.

690. The radiological consequences of the design basis accidents studied below, therefore relate only to accidental atmospheric radioactive discharges.

691. The approach to identifying reference accidents has resulted in the selection of three Plant Condition Category (PCC)-4 design basis accidents and one Design Extension Condition (DEC)-B accident as the reference accidents relating to the reactors. The accidents have been selected on the basis of their radiological consequences.

692. The reference accidents selected reflect the design basis that is defined for the UK EPR™ in the UK regulatory context. The DEC-B accident is considered bounding for accidents that inform the design basis for reactors at Sizewell C. While it is recognised that the ISFS facility will continue operation well beyond EoG and defueling of the reactors, the consequence of an event is still bounded by the DEC-B accident.

6.2.1 Category 4 (PCC-4) accidents

6.2.1.1 Main hypotheses

693. The main hypotheses used to evaluate the radiological consequences of accidents are as follows:

6.2.1.1.1 Activities taken into account in the primary circuit and secondary circuit water

<u>Primary circuit water activity</u>

694. The activity selected for primary circuit water is based on the maximum values adopted for the technical specifications for all French nuclear power plants in operation, equal to:

- primary circuit activity in stable operation: 20GBq t^{-1} equivalent to Iodine-131[5]; and
- primary circuit activity after power transient (iodine spiking): 150GBq t^{-1} equivalent to Iodine-131.

Secondary circuit water activity

695. The maximum water activities on the secondary side of the steam generators are calculated based on the following hypotheses:

- Primary circuit water activity in the unit, corresponding to the maximum values specified in the technical operating specifications.
- A primary-secondary leakage rate of 20L h^{-1}.
- A drainage rate from all of the steam generators corresponding to the plant operating at nominal power.
- Drive factors in the steam from the steam generators corresponding to the values detailed below.

6.2.1.1.2 Drive factors in steam from the steam generators

696. The drive factors taken into account for the transfer of activity during the steam generator steam phase are as follows:

- All noble gases present in the steam generator water are assumed to be transferred in the gaseous phase.
- For other radionuclides, 'healthy' steam generators are distinguished from damaged steam generators (due to a steam generator tube rupture, for example).
- For "healthy" steam generators, the drive factor taken into account is 0.25%.
- For damaged steam generators, the drive factor taken into account is 1%.

6.2.1.1.3 Release of activity in the event of cladding failure

697. During some accidents (particularly LOCAs), the fuel cladding is subject to a thermo-hydraulic transient, which can cause it to fail. The fission product activity, accumulated as a result of pellet/cladding gap, can then be released into the primary system.

698. The release rates for the proportion of fission products in the fuel rod inventory assumed to be discharged into the system when the cladding fails are presented here. Information is provided for both uranium oxide and mixed-oxide (MOX) fuel. The use of MOX fuel in the UK EPR™ is not currently planned. It is included here because it is used in assessments to provide a more conservative estimate of impact.

699. The overall release rates selected, based on the considered fuel assemblies, are shown in Table 6-7.

Table 6-7 Activity release rate in the event of cladding failure

Elements	Selected burn-up for the evaluation of discharge – UO_2 fuel	Selected burn-up for the evaluation of discharge – MOX fuel

5 Iodine-131 equivalent = I131 + I132 ÷ 30 + I133 ÷ 4 + I134 ÷ 50 + I135 ÷ 10

	≤47 GWd tU^{-1}	> 47 GWd tU^{-1}	≤ 33 GWd tU^{-1}	> 33 GWd tU^{-1}
Krypton-85	8%	25%	8%	50%
Other noble gases	2%	8%	2%	15%
Bromine, rubidium, iodines, caesium	2%	8%	2%	15%

6.2.1.1.4 Deposition of fission products

700. The physical laws on aerosol and molecular iodine deposits in the containment take into account an exponential decay term, the deposition constants of which are equal to 0.035 h^{-1} and 0.014 h^{-1} respectively.

6.2.1.1.5 Leakage rate from the containment vessel

701. The overall leakage rate through the containment inside the UK EPR™ (which has a metal liner) is 0.3% volume per day at the design pressure (5.5bar).

6.2.1.1.6 Filter performance

702. The retention performances of the extraction filters used to reduce radioactive waste are as follows:

703. High efficiency filters:

- Noble gases 0%
- Aerosols (including particulate iodine) 99.9%
- All other substances 0%

704. High efficiency filters and iodine traps:

- Noble gases 0%
- Iodine in organic form 99%
- Elemental iodine 99.9%
- Aerosols (including particulate iodine) 99.9%

6.2.1.2 Specific hypotheses relating to the accidents under consideration

705. In addition to the general hypotheses described above, particular hypotheses relating to the accidents examined are presented below.

6.2.1.2.1 Large primary circuit break during operation at nominal power

706. The Category 4 LOCA is defined as a rupture in the safety injection line in the nozzle of the primary coolant system's cold leg.

707. During this accident, it is assumed that the reactor core becomes uncovered, with a 10% break in the fuel rod assembly. The discharge considered for this accident is a result, on the one hand, of leaks in the containment vessel and on the other hand, of leaks assumed to occur in the reactor core cooling systems outside the containment vessel, in the ventilated and filtered safeguard auxiliary buildings.

6.2.1.2.2 Rupture of two steam generator pipes

708. The radiological consequences of this accident are the result of the release of activity to atmosphere, via the atmospheric steam dump valves on a faulty steam generator. The activity is due to contamination of the secondary circuit by the primary circuit through a break in the steam generator tubes.

709. The peak of activity in the primary circuit (iodine spiking), caused by the transfer of activity due to pellet/cladding gap in the primary coolant following automatic reactor shutdown, is considered for the assessment of the radiological consequences of this accident. To create a more conservative scenario, it is assumed that the iodine spiking is fully developed when emergency shutdown occurs.

6.2.1.2.3 Fuel handling accident

710. The accident examined involves a fuel assembly with a maximum irradiation, dropped into the spent fuel pit in the fuel building. All fuel rods in the damaged assembly are pessimistically assumed to be broken.

711. The cooling time for the damaged assembly is 60 hours, corresponding to the minimum time required between reactor shutdown and the start of fuel handling.

712. The release rates considered are those provided in Section 6.2.1.1.

713. It is assumed that the radioactive isotopes released from the pit in the hall are distributed immediately and evenly throughout the entire free volume of the hall.

714. The automatic closure command for the main air extraction system’s cut-off devices enables the activity to be contained by switching to reduced ventilation and iodine traps. This command is automatically activated by means of a high activity measurement signal, which is located on the operating floor for the irradiated fuel pit.

6.2.2 DEC-B conditions - core melt accident

715. This extreme type of accident, which was not considered in the design of the existing nuclear reactors at the site, has been taken into account in the design measures specific to the UK EPR^{TM} for SZC. The associated radiological consequences are analysed to ensure compliance with objectives in terms of population protection.

716. The design of the UK EPR^{TM} is such that the risk of a core meltdown is extremely low. As part of the implementation of an improved defence in depth philosophy, low pressure core melt accidents constituting DEC-B conditions, have been addressed by means of specific design features that aim to ensure that the integrity of the containment is maintained and the release of radioactive products outside the plant remains within prescribed limits.

717. An examination of DEC-B conditions shows that, given the design features adopted, the following radiological objectives associated with these situations are met:

- Limited requirement for sheltering;
- No requirement for emergency evacuation outside the immediate vicinity of the plant;
- No requirement for permanent relocation; and
- No long term restrictions on the consumption of foodstuffs.

718. The Emergency Reference Levels (ERLs) of averted dose associated with sheltering, evacuation and stable iodine are presented in Chapter 7, together with a discussion on the effectives of longer-term protective measures.

719. Any restrictions concerning consumption of foodstuffs produced in the vicinity of the plant are governed by relevant European marketing regulations applicable in the event of a nuclear accident or other radiological emergency, such as by regulations from the Council of the European Communities which specific intervention levels for radioactive contamination in marketed foods and animal feeds. These are known as the Council Food Intervention Levels (CFILs).

720. Exposures beyond the local population are further discussed in Section 6.3.1.6.

6.2.2.1 Source term

721. A benchmark source term has been defined, based on reasonably conservative disconnection hypotheses which are independent of the accident scenario. The main hypotheses are as follows:

- A 100% core melt is assumed.
- The radionuclide release rates, in terms of the radiological consequences on populations (noble gases, iodine and caesium) have been maximised (100% release into the containment vessel).
- The quantity of suspended aerosols in the containment falls due to natural deposition. The effectiveness of the containment spray systems has not been taken into consideration.
- Iodine is mainly released into the containment in the form of aerosols. A fraction of suspended organic iodine equal to 0.15% is taken into account from the beginning of the accident. This value has been used to bound the quantity measured in the long term phase of the accident at Three Mile Island.

722. During a DEC-B type accident, the integrity of the UK EPR™ containment is guaranteed by specific provisions. This special design justifies the use of hypotheses developed for PCC accidents to assess discharge into the environment:

- An internal containment leakage rate of 0.3% volume per day (maximum internal containment leakage rate at its absolute pressure and design temperature).
- Filtration, downstream from the ventilation, which allows 99.9% of the aerosols and elementary iodine and 99% of the organic iodine to be retained. Noble gases are not removed by filtration.

723. This source term conservatively covers DEC-B type accident sequences.

6.2.2.2 Fraction discharged

724. The UK EPR™ source term, calculated using these hypotheses, is shown in Table 6-8. It is expressed as a percentage of the fractions discharged, in comparison with the initial core activity for a certain number of radionuclides (total activity discharged, taking radioactive decay into account).

Table 6-8 Fraction of radionuclides discharged

Radionuclides	Source Term (% initial core inventory)

Xenon-133	1.50E+00
Iodine-131	6.10E-01
Caesium-137	7.00E-06
Tellurium-132	5.10E-06
Strontium-90	1.30E-06
Ruthenium-106	2.60E-07
Cerium-141	2.60E-07
Plutonium-241	4.60E-08

6.3 Evaluation of the Radiological Consequences of the Reference Accident(s)

6.3.1 Release to atmosphere

725. The assessment considers the impacts from releases to atmosphere to reference groups in the vicinity of the facility and in other Member States. The local reference group is assumed to be located 2km from the facility. The Member State reference groups and their locations have been selected on the same basis as for the assessment of airborne releases in normal conditions (Section 3.4). These reference groups are located in France, Belgium, the Netherlands and Ireland. Distances and bearings from Sizewell C to the reference groups are provided in Table 3-7.

6.3.1.1 Methods and parameters used to calculate the releases to atmosphere

726. The assessment of impacts to the local representative group in the vicinity of the facility uses the approach adopted for the GDA. The GDA defines a set of site characteristics appropriate for the development of an UK EPR™ in the UK. The parameters are chosen as they represent the 'typical data' of potential sites where a new UK EPR™ reactor could be located.

727. The long range assessment takes a simplified approach, based on NRPB-R124 [3] and assessing only key radionuclides. These key radionuclides include the activity contributions of the other radionuclides but they are modelled as a single radionuclide. This presentation of only key radionuclides represents a conservative approach whilst addressing security concerns expressed by the UK nuclear regulator on presenting a full radionuclide inventory. Table 6-9 identifies the key radionuclides used to represent radionuclide groups. The key radionuclides have been chosen as those with the highest radiotoxicity from the members of the radionuclide group. The long range assessment also makes the assumption of a uniform release rate over the duration of the release.

Table 6-9 Key radionuclides and other radionuclides considered

Assessed radionuclide	Other radionuclides included in group
Krypton-85	Other isotopes of Krypton (e.g. Krypton-85m, Krypton-87, Krypton-88)
Xenon-133	Including Xenon-133m
Xenon-135	Including Xenon-135m and Xenon-138
Iodine-131	Other isotopes of tellurium, Iodine (e.g. Iodine-132, Iodine-134, Iodine-135)
Iodine-133	-
Caesium-137	Other beta gamma radionuclides (e.g. Strontium-90, Caesium-138, Barium-140, Lanthanum-140, Cerium-141, Cerium-143, Praseodymium-143, Cerium-144, Neptunium-239)
Alpha	Other actinides (e.g. Plutonium-238, Plutonium-239, Curium-242, Curium-244)

6.3.1.2 Release duration

728. The design basis accidents have different release durations from one hour up to seven days. The simple long range model conservatively assumes that the weather conditions remain constant for the duration of the release and for the time the long range reference group is affected by the plume. For events that last more than a few hours it is unlikely that the weather patterns will remain constant and in reality changes will result in enhanced dispersion that would result in lower impacts than those presented.

729. The DEC-B could result in a longer duration release, of the order of approximately one month, although the majority of particulate activity is released in the first 48 hours. Therefore, the modelling for the severe accident is based on a release duration of 48 hours. This is more realistic in terms of the underlying assumption of constant weather conditions for the duration of the release.

6.3.1.3 Amounts and physico-chemical forms of those radionuclides which are significant from the point of view of health

730. It is likely that, in the event of an accident, the physico-chemical forms of those radionuclides released can only be determined after the event. Dose per unit intake values have therefore been chosen based on the most conservative form of the radionuclide. For iodine, this corresponds to isotopes of iodine being inhaled in vapour form. The accident scenario, presented in the GDA PCSR, assumes a mixture of elemental and aerosol of isotopes of iodine. Table 6-10 presents the lung classes[6] used in the long range assessment for all accidents, which correspond to the highest dose per unit intake values for these isotopes.

Table 6-10 Lung class used in assessment for key radionuclides

Radionuclide	Lung class
Iodine-131	V
Iodine-133	V
Caesium-137	F
Plutonium-239#	M

#Alpha represented as Plutonium-239

731. Table 6-11 describes the quantities of the assessed key radionuclides (Table 6-9) released in each of the accidents assessed.

Table 6-11 Amounts of key radionuclides assessed in each scenario

Assessed Radionuclide	Source Term Released to the Environment (Bq)			
	Fuel handling accident	Steam generator tube rupture	LOCA	DEC-B
Krypton-85	1.6E+14	2.2E+13	1.1E+13	5.7E+16
Xenon-133	2.1E+16	1.1E+14	3.5E+14	1.5E+17
Xenon-135	6.1E+14	4.0E+13	1.9E+13	1.1E+15
Iodine-131	3.2E+10	6.1E+11	4.8E+10	4.5E+12
Iodine-133	3.0E+09	3.6E+11	2.1E+10	1.5E+12
Caesium-137	3.8E+09	2.7E+11	1.9E+11	1.2E+12
Alpha (Plutonium-239)	0.0E+00	0.0E+00	1.2E+09	5.3E+08
TOTAL	2.2E+16	1.7E+14	3.8E+14	2.1E+17

6 Lung classes, developed by ICRP, are indicative of the rate of clearance of inhaled activity from the pulmonary region of the lung, very fast or vapour (V), fast (F), medium (M) and slow (S) are used to represent the clearance rate.

6.3.1.4 Models and parameter values used

732. This section outlines the models and parameters used in the assessment of consequences from the defined accident scenarios. In order to consider the effects of dispersion of activity over long distances, a different model was adopted for the long range assessment to that used in the local assessment.

733. Data for the local assessment is taken from information already presented to the UK regulators as part of the GDA process.

6.3.1.4.1 Short range model used for local reference group

734. The short range model is based on that used in the PCSR developed in support of the GDA process. The most exposed members of the public to gaseous discharges are assumed to be a farming family living 2km from the discharge point.

Local meteorological conditions

735. Depending on the prevailing weather conditions, the dispersion and the deposition of the released radionuclides is subject to a wide variability. Therefore, a probabilistic approach was used using meteorological data. Wind direction is included in the probabilistic assessment. Using this approach, the value which covers 95% of the cases is judged to be adequately conservative.

736. Atmospheric dispersion is calculated using a Gaussian model. The models CORRA and ASTRAL, developed by Institut de Radioprotection et de Sûrete Nucléaire (IRSN) and used by EDF in their assessments, were used to assess the initial and longer term impacts respectively.

Deposition data

737. Deposition and washout factors in the model are used to calculate the amount of radioactive substances that would be deposited during dry weather or during precipitation respectively. The fallout and washout factors are not only a function of atmospheric parameters, e.g. wind velocity or precipitation rate, they also depend on the physico-chemical form of the radionuclides. Deposition velocities and the washout coefficients used in the local assessment are presented in Table 6-12.

Table 6-12 Deposition parameters used in the local assessment model

Substance group	Deposition velocity (m s^{-1})	Washout coefficient (s^{-1})
Iodine isotopes	1.0E-02	7.0E-05
Others	1.0E-04	7.0E-07
Noble gases	1.5E-03	7.0E-05

Exposure pathways

738. The following exposure pathways are considered in the dose calculation for the design basis accidents.

- Gamma radiation from the passing plume.
- Inhalation of radioactive substances by persons affected by the plume for the time during which the plume passes.
- Gamma radiation from radioactive substances deposited onto the ground surface.

- Ingestion of foodstuffs contaminated by radionuclides.

739. The exposure period for the radioactive substances deposited on the ground surfaces, as well as the ingestion of foodstuffs, is assumed to be the whole life of the individual. This is 50 years for adults. Due to the changing dietary habits of children, ingestion doses are not calculated beyond the first year. Given that most of the dose is delivered in the first year this is seen as being a reasonable approach. The committed effective dose from the intake of radionuclides related to internal radiation exposure, due to inhalation and ingestion, is assumed to be 70 years for infants and 50 years for adults.

740. It is assumed that local food consumption restrictions are in place 24 hours after the beginning of the accident, within a radius of 2km from the release point. It is assumed that the food produced in this area is not used for the first year after the accident. Outside this area no mitigation measures are assumed. Therefore, for the ingestion pathway the dose at 2km distance may be greater than the dose at 1km distance.

741. The assessment of food doses is based on the ASTRAL code. The ASTRAL spreadsheets provide dose per unit deposition for each radionuclide, for adult and infants summed across the assessed food groups. The ASTRAL foodchain factors are presented in Table 6-13.

Table 6-13 Dose per unit deposition via food ingestion from ASTRAL (mSv Bq^{-1} m^{-2}) (adapted from Table 6.11, HPC Article 37 Submission [4])

Radionuclide	Adult	Infant
Iodine-131	6.80E-09	1.80E-08
Iodine-133	6.80E-09	1.80E-08
Cobalt-58	4.70E-10	1.30E-09
Cobalt-60	3.90E-09	1.30E-09
Caesium-137	3.30E-08	1.10E-08
Alpha (Plutonium-239)	1.10E-07	9.20E-08

742. In addition, the dose from exposure of the thyroid by the inhalation of radioiodine is assessed.

Dose coefficients

743. The age-dependent dose coefficients for ingestion and inhalation of radionuclides used in the GDA assessment are taken from the International Commission on Radiological Protection (ICRP) publication 72 (compiled in [5]).

6.3.1.4.2 Long range model used in the assessment of impacts to reference groups in other Member States

744. The process for identifying the reference groups for other Member States is described in Chapter 3.4. In summary this was based on:

- proximity to the Sizewell C site;
- wind directions from the location of Sizewell C towards other Member States;
- intake rates of terrestrial foods and habit data;
- information on exports from the area local to Sizewell C.

745. The Member State reference groups identified are located in France, Belgium and the Netherlands. Ireland has also been assessed as a reference group.

746. The long range model used to estimate the consequences to the reference groups for France, Belgium, the Netherlands and Ireland, was based on NRPB-R124 [3]. This makes some simple but conservative assumptions regarding the travel of a short term release over longer distances, where traditional Gaussian plume models are not appropriate. The model used is a deterministic one, which assumes the wind is blowing towards the assessed critical groups for the duration of the release. In addition, it assumes a constant boundary layer height and wind speed. As a result of the same wind speed being applied to each long range reference group, there is no distinction between the two reference groups identified for France considered in the assessment of aerial discharges under normal conditions. Therefore, a single reference group for France has been included in the assessment of unplanned releases occurring over the short term. Parameters used in the assessment are detailed in Table 6-14, taken from NRPB-R124 [3].

Table 6-14 Long range atmospheric dispersion modelling parameters

Parameter/description	Value
Mixing layer depth	1000m
Wind speed	8 m s^{-1}

747. Depletion of the plume by radioactive decay and deposition (wet and dry) were included, based on the approach presented in NRPB-R122 [6]. The parameter values are the same as those used in the long range routine release model described in Section 3.4.1.5.

748. The output of the model is time integrated air concentration (in Bq s m^{-3}) and for depositing radionuclides, surface deposition density on the ground (in Bq m^{-2}).

Exposure pathways

749. The exposure pathways for the reference groups for other Member States are the same as those listed for the local reference group in Section 6.3.1.4.1.

750. A simplified approach was taken in determining ingestion doses in the nearest Member States. The dose from the ingestion of contaminated foods was determined by taking the ratio of the surface concentration value in the Member State, to the surface concentration value in the vicinity of the facility, multiplied by the local ingestion dose. Surface concentration values are presented below in Table 6-16 and the local ingestion doses per unit deposition are presented above in Table 6-13.

Occupancy data

751. The long range model uses the same Member State reference groups as defined in Section 3.4. The same food intake rates, breathing rates and fraction of time indoors have been applied to these reference groups. However, 100% occupancy in the affected area has been pessimistically assessed.

Dose coefficients

752. The inhalation and ingestion dose coefficients are taken from ICRP publication 119 [5]. Since it is uncertain as to the physical and chemical form of the radionuclides released under each scenario, the dose coefficients have been chosen by selecting the most conservative values. External dose per unit exposure factors are derived from US EPA Federal Guidance Report 12 [7], as described in Section 3.4.1.6. The gamma dose from deposited radionuclides are taken from NRPB-W19 [8] as the values after 1 year.

6.3.1.5 Expected levels of radioactive contamination of foodstuffs which might be exported to other affected Member States

753. The assessment of food doses is based on the GDA approach for the local reference group. Calculations for the local reference group have conservatively assumed that all the food consumed by the reference group is sourced locally. Data presented in Chapter 3 indicates that the majority of foodstuffs are distributed locally and that there are no foodstuffs produced in the vicinity of Sizewell C that are exclusively for export to other Member States and not consumed by the local reference group.

6.3.1.6 Maximum time integrated air concentrations and total surface concentrations

754. The maximum time integrated air concentrations and the total surface concentrations resulting from the four scenarios across five locations are presented in Table 6-15 and Table 6-16 respectively.

Table 6-15 Maximum time integrated air concentrations

Location	Maximum time integrated air concentrations (Bq s m^{-3})			
	Fuel handling accident	Steam generator tube rupture	LOCA	DEC-B core melt
Local area (at 2km)	3.2E+11	7.2E+09	5.6E+09	1.8E+12
France	6.9E+06	6.8E+05	1.2E+05	2.3E+08
Belgium	6.5E+06	6.4E+05	1.1E+05	2.2E+08
Netherlands	5.7E+06	5.6E+05	1.0E+05	1.9E+08
Ireland	1.3E+06	1.2E+05	2.2E+04	4.4E+07

Table 6-16 Total surface concentrations

Location	Total surface concentrations (Bq m^{-2})			
	Fuel handling accident	Steam generator tube rupture	LOCA	DEC-B core melt
Local area (at 2km)	1.8E+03	1.5E+05	1.1E+04	9.7E+04
France	1.8E+00	7.1E+02	1.1E+01	1.1E+03
Belgium	1.7E+00	6.8E+02	1.1E+01	1.1E+03
Netherlands	1.5E+00	6.1E+02	9.8E+00	9.5E+02
Ireland	5.0E-01	2.0E+02	3.4E+00	3.1E+02

6.3.1.7 Corresponding maximum doses

755. The corresponding maximum committed effective doses in microsieverts, to the local reference group and reference groups in other Member States, resulting from the four scenarios, are presented below in Table 6-17 to Table 6-21[7].

756. The results provide a breakdown in terms of pathway (inhalation including inhalation of re-suspended materials, external irradiation from immersion in the plume, external irradiation from material deposited on the ground and ingestion of contaminated foodstuffs).

757. The results include the impacts from the initial passage of the plume and accrued during the first pass of the plume, which includes the inhalation and immersion pathways. The longer term dose

[7] The figures presented in Table 6-17 to Table 6-21 have been rounded to 2 significant figures.

is the summation of re-suspension, external and ingestion pathways. The assessment also includes an assessment of thyroid dose from the inhalation of radioiodine.

Table 6-17 Maximum committed effective doses to an adult member of the local reference group by pathway

Location	Maximum committed effective dose (µSv)			
	Fuel handling accident	Steam generator tube rupture	LOCA	DEC-B core melt
Inhalation	1.1E+00	2.4E+01	1.2E+01	1.3E+02
Immersion	5.8E+02	1.0E+02	1.4E+01	2.3E+03
External	1.5E+01	1.3E+03	6.7E+01	2.9E+02
Ingestion	1.8E+01	7.2E+02	4.4E+01	3.5E+02
Total	6.1E+02	2.1E+03	1.4E+02	3.1E+03
First Pass	5.8E+02	1.2E+02	2.7E+01	2.4E+03
Longer Term	3.2E+01	2.0E+03	1.1E+02	6.4E+02
Short term thyroid dose	2.0E+01	3.7E+02	1.9E+01	2.4E+03

Table 6-18 Maximum committed effective doses to an adult member of the reference group for France by pathway

Location	Maximum committed effective dose (µSv)			
	Fuel handling accident	Steam generator tube rupture	LOCA	DEC-B core melt
Inhalation	6.9E-05	2.7E-02	1.3E-02	6.2E-02
Immersion	6.3E-03	1.2E-03	1.2E-04	1.5E-01
External	1.2E-03	1.0E+00	5.1E-02	1.2E+00
Ingestion	1.7E-02	3.4E+00	4.6E-02	4.0E+00
Total	2.5E-02	4.5E+00	1.1E-01	5.4E+00
First Pass	6.3E-03	2.7E-02	1.3E-02	2.1E-01
Longer Term	1.8E-02	4.5E+00	9.8E-02	5.2E+00
Short term thyroid dose	1.1E-03	2.8E-01	1.7E-03	5.3E-01

Table 6-19 Maximum committed effective doses to an adult member of the reference group for Belgium by pathway

Location	Maximum committed effective dose (µSv)			
	Fuel handling accident	Steam generator tube rupture	LOCA	DEC-B core melt
Inhalation	6.5E-05	2.6E-02	1.3E-02	5.8E-02
Immersion	5.9E-03	1.1E-03	1.1E-04	1.4E-01
External	1.2E-03	9.7E-01	4.9E-02	1.2E+00
Ingestion	1.6E-02	3.3E+00	4.4E-02	3.8E+00
Total	2.3E-02	4.3E+00	1.1E-01	5.2E+00
First Pass	6.0E-03	2.6E-02	1.2E-02	1.9E-01

Longer Term	1.8E-02	4.3E+00	9.4E-02	5.0E+00
Short term thyroid dose	9.9E-04	2.7E-01	1.6E-03	5.0E-01

Table 6-20 Maximum committed effective doses to an adult member of the reference group for the Netherlands by pathway

Location	Maximum committed effective dose (µSv)			
	Fuel handling accident	Steam generator tube rupture	LOCA	DEC-B core melt
Inhalation	5.7E-05	2.3E-02	1.1E-02	5.1E-02
Immersion	5.1E-03	9.8E-04	9.8E-05	1.2E-01
External	1.1E-03	8.8E-01	4.4E-02	1.0E+00
Ingestion	1.5E-02	3.0E+00	4.0E-02	3.4E+00
Total	2.1E-02	3.9E+00	9.5E-02	4.7E+00
First Pass	5.2E-03	2.3E-02	1.1E-02	1.7E-01
Longer Term	1.6E-02	3.8E+00	8.4E-02	4.5E+00
Short term thyroid dose	8.7E-04	2.3E-01	1.4E-03	4.4E-01

Table 6-21 Maximum committed effective doses to an adult member of the reference group for Ireland by pathway

Location	Maximum committed effective dose (µSv)			
	Fuel handling accident	Steam generator tube rupture	LOCA	DEC-B core melt
Inhalation	1.4E-05	5.6E-03	2.8E-03	1.2E-02
Immersion	1.1E-03	1.5E-04	1.9E-05	2.7E-02
External	3.7E-04	3.2E-01	1.6E-02	3.7E-01
Ingestion	4.8E-03	9.6E-01	1.4E-02	1.1E+00
Total	6.3E-03	1.3E+00	3.2E-02	1.5E+00
First Pass	1.1E-03	5.4E-03	2.6E-03	3.9E-02
Longer Term	5.2E-03	1.3E+00	3.0E-02	1.5E+00
Short term thyroid dose	2.0E-04	5.5E-02	3.2E-04	1.0E-01

758. Figure 6-1 to Figure 6-4 show a summary of the doses for each age group and accident scenario for the reference groups in France, Belgium, the Netherlands and Ireland.

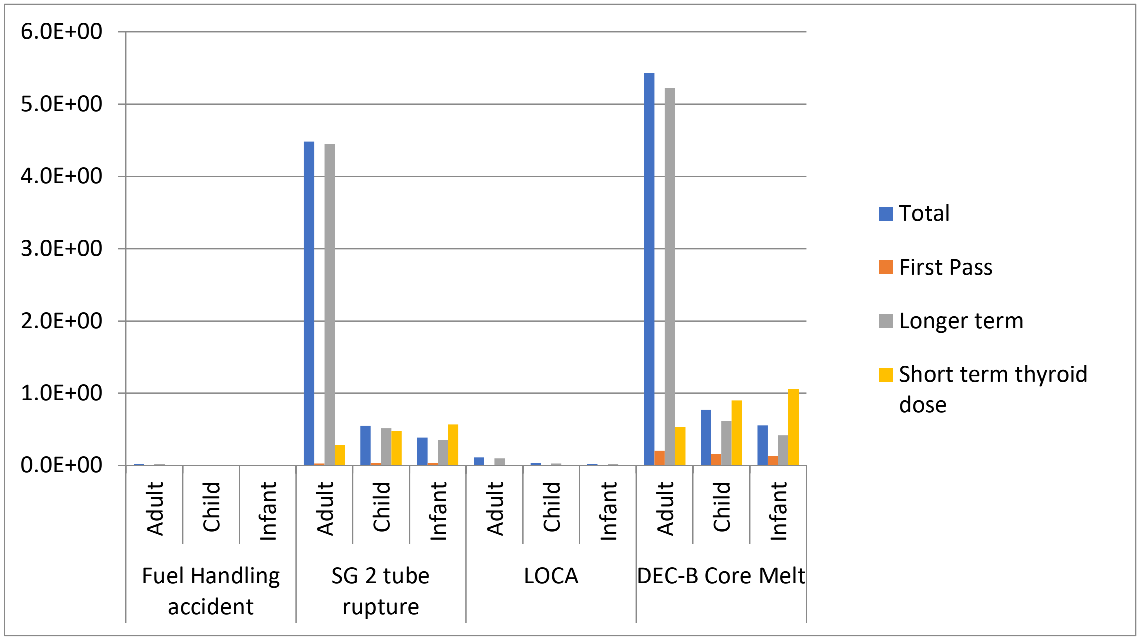

Figure 6-1 Summary of doses by age group and accident scenario for reference group (France)

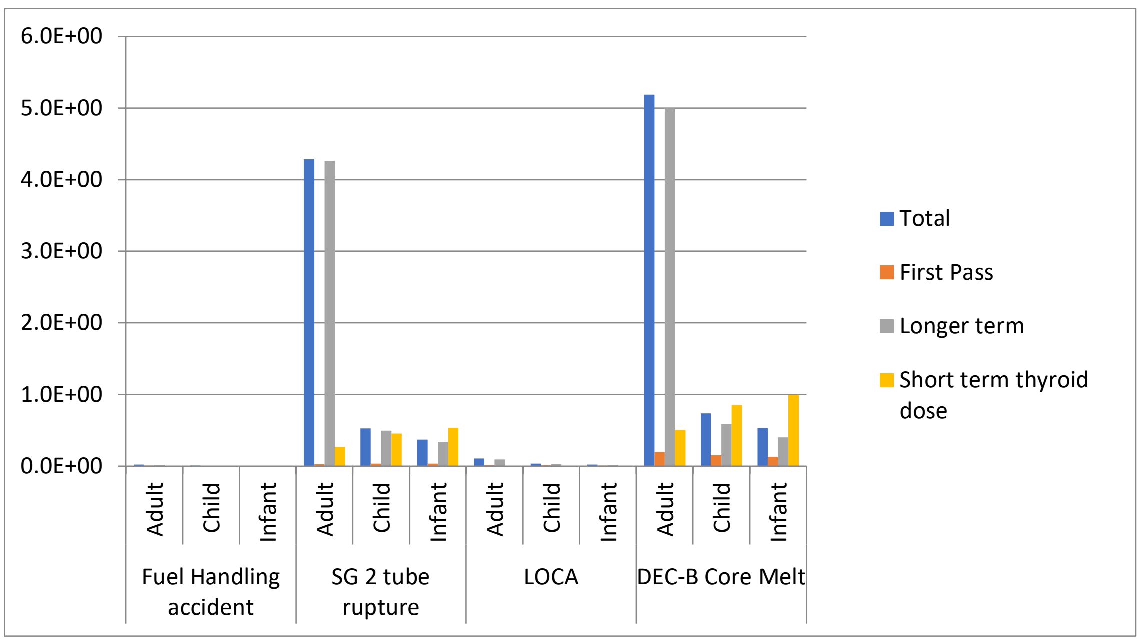

Figure 6-2 Summary of doses by age group and accident scenario for reference group (Belgium)

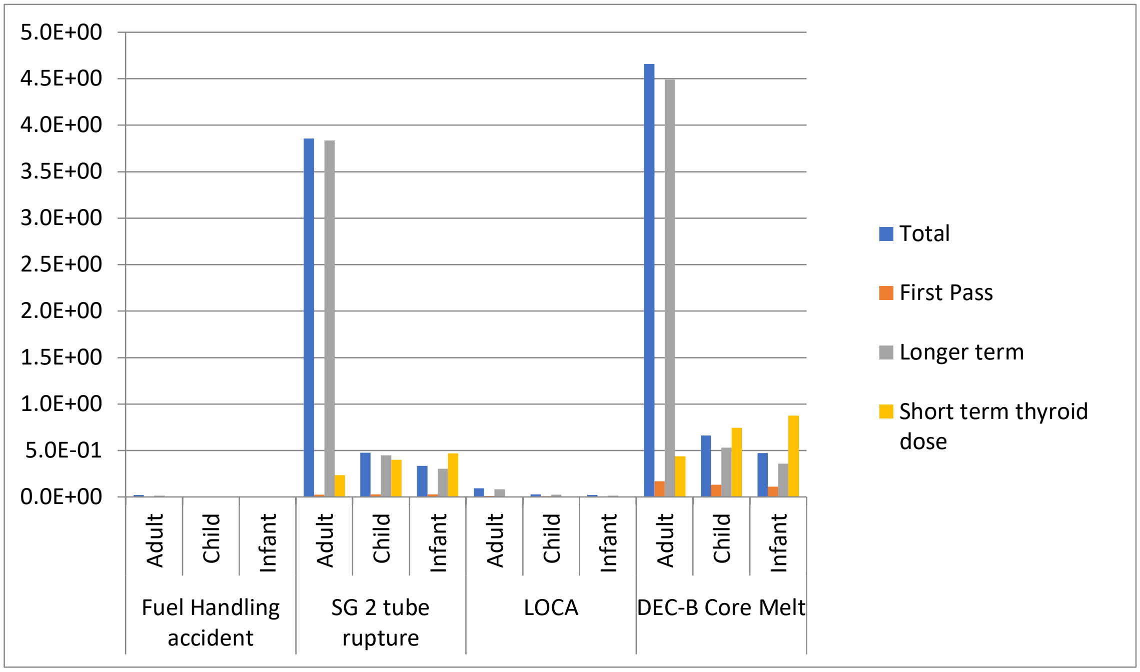

Figure 6-3 Summary of doses by age group and accident scenario for reference (Netherlands)

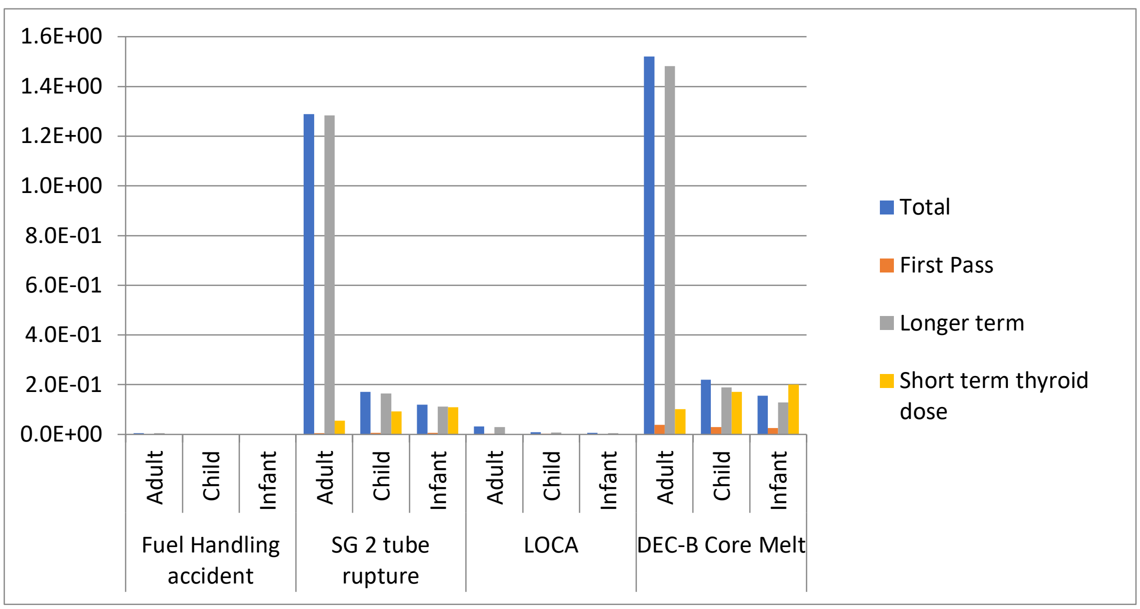

Figure 6-4 Summary of doses by age group and accident scenario for reference (Ireland)

6.3.1.7.1 Results

759. The maximum effective doses arising from a single initiating event from a design basis accident, arises from the rupture of steam generator tube accident scenario. This results in total effective doses of 4.5 μSv, 4.3 μSv, 3.9 μSv and 1.3 μSv for adult members of the reference groups in France, Belgium, the Netherlands and Ireland, respectively. The same scenario results in the highest thyroid doses of 5.7E-01 μSv, 5.4E-01 μSv, 4.7E-01 μSv and 1.1E-01 μSv to an infant member of the reference groups in France, Belgium, the Netherlands and Ireland, respectively.

760. The total effective doses from the severe accident scenario (DEC-B) are 5.4 μSv, 5.2 μSv, 4.7 μSv and 1.5 μSv for adult members of the reference groups in France, Belgium, the Netherlands and Ireland, respectively. The highest thyroid doses for this scenario are 1.1 μSv, 1.0 μSv, 8.7E-01 μSv and 2.0E-01 μSv to an infant member of the reference groups in France, Belgium, the Netherlands and Ireland, respectively.

761. The maximum effective doses arising in the period after the reactors have ceased operation is bounded by the maximum effective doses associated with the reactor operational phase.

6.3.2 Release into an aquatic environment

762. Given the precautions taken to ensure total containment in the event of an accident, as described in Section 2.5, no design basis accident has been identified that would cause the continuous discharge of radioactive material into the aquatic environment.

763. The base mats of the lower part of the nuclear buildings provide a barrier to protect the environment from contamination due to radioactive liquid spills or leaks. In the event of liquid leaks in the nuclear island buildings, the radioactive effluent is recovered by sumps, retention pits and retention tanks. With regard to the reactor building the base mats are sealed by specific measures, which are a membrane below the base mat, a coating on the lowest floors, a sealed lining in the fuel pool and the sealing of the molten core spreading area. All of these measures protect the containment function of the base mat, even in a severe accident situation. Molten core cooling is also monitored to follow up the impacts on the containment.

764. Finally, the reactor building annulus is kept at a lower pressure to collect any leaks from inside the containment, the leaks are discharged to the vent stack after HEPA and iodine filtering.

765. On the basis of the above, and taking into account the considerations of ONRs Report on Fukushima, accidental releases of radioactive liquid waste into aquatic environments are therefore excluded from any further assessment.

6.4 References

[1] International Atomic Energy Agency, Safety Requirements SSR2/1 Safety of Nuclear Power Plants: Design. ISBN 978-92-0-121510-9, Vienna: IAEA, 2007.

[2] Office for Nuclear Regulation, Japanese earthquake and tsunami: Implementing the lessons for the UK's nuclear industry, London: Crown, 2012.

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

[4] EDF Energy, Submission of General Data as applicable under Article 37 of the Euratom Treaty, Bristol: Department of Business, Energy & Industrial Strategy, 2011.

[5] International Commission on Radiological Protection, “Compendium of Dose Coefficients based on ICRP Publication 60. ICRP Publication 119. Ann. ICRP 41(Suppl.),” Elsevier, 2012.

[6] J. Jones, “The Second Report of a Working Group on Atmospheric Dispersion: A Procedure to Include Deposition in the Model for Short and Medium Range Atmospheric Dispersion of Radionuclides (NRPB-R122),” National Radiological Protection Board, Chilton, 1981.

[7] U.S. Environmental Protection Agency, External Exposure to Radionuclides in Air, Water, and Soil. Federal Guidance Report No. 12, EPA-402-R-93-081, Washington: US EPA, 1993.

[8] N. McColl and S. Prosser, “Emergency Data Handbook. (NRPB-W19),” National Radiological Protection Board, Chilton, 2002.