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SUB-CHAPTER 15.5 – LEVEL 3 PSA: ASSESSMENT OF OFF-SITE RISK DUE TO POSTULATED ACCIDENTS

1. INTRODUCTION

A probabilistic assessment (a Level 3 PSA) of the UK EPR design has been performed to determine the off-site risk to the public due to postulated accidents. This sub-chapter summarises the process followed to perform this assessment and presents the results in terms of:

- Individual risk to any person off the site, i.e. frequency / consequence (dose band) couplets.
- Societal risk, in terms of the annual frequency of events which could potentially lead to more than 100 immediate or eventual fatalities in the wider population.

2. BACKGROUND

The Level 1 PSA analyses a number of Initiating Events (IEs) together with total and partial failure of associated protection or mitigation measures. The Level 1 PSA failure states consider events that lead to core damage. Other, less onerous, endpoints are modelled in the event trees but are combined into a general ‘success’ state, as they do not result in a designated failure state (i.e. a state beyond the design basis).

The Level 2 PSA takes the failure states, analyses the containment response to such sequences and assigns a release category (RC) to each Containment Event Tree (CET) endpoint, which represents the characteristics of the activity released to the off-site environment.

The Level 3 PSA estimates the likely impact of radiologically significant faults. Potentially there are many possible endpoints to a Level 3 PSA covering a wide range of health and socio-economic impacts. At this stage only two endpoints are considered - individual risk and societal risk.

The approach followed and the results obtained for each of these assessments is described below.

3. PROCESS OF ASSESSMENT

The assessment uses the results from existing analyses of the EPR design elsewhere in the PCSR, as is described in detail below.

3.1. IDENTIFICATION OF INITIATING EVENTS FOR ASSESSMENT

The list of initiating events has been drawn from three sources:

- Those considered in the Design Basis Analysis (see Chapter 14). These events are listed in Sub-chapter 15.5 - Table 1.
- Initiating events modelled in the Level 1 PSA (see Sub-chapter 15.1). These events are listed in Sub-chapter 15.5 - Table 2.
- An expert review [Ref-1] of the design and operating practices to identify additional initiating events whose consequences would be within the design basis. These events are listed in Sub-chapter 15.5 - Table 3.

The main purpose of the expert review was to identify those initiating events not already modelled in the Level 1 PSA which have the potential to result in off-site radiological consequences within the dose band range considered for the assessment of individual risk. These include reactor based faults and non-reactor faults. The panel of experts have knowledge of both the EPR design and safety assessment, with experience of safety cases for facilities licensed in the UK.

The result of the expert review process is a combined list of initiating events not included in the Level 1 PSA. These are listed in Sub-chapter 15.5 - Table 4. Note that not all the events listed in Sub-chapter 15.5 - Tables 1 and 3 appear in Sub-chapter 15.5 - Table 4. This is because some of the design basis events (Sub-chapter 15.5 - Table 1) are considered in the Level 1 PSA, and some of the ‘additional’ events identified in the expert review had also been identified in the Level 1 PSA. (The expert review panel deliberately chose not to look at the Level 1 PSA list, in order to take a more open-ended view).

It should also be noted that, as explained in Sub-chapter 15.0, only some of the initiating events in the ‘hazards’ category have been considered at this stage, as follows:

- Internal flooding,
- Internal fire,
- External hazards leading to Loss of Ultimate Heat Sink,
- External hazards leading to Loss Of Off-site Power,
- Accidental aircraft crash.

3.2. ASSESSMENT OF INDIVIDUAL RISK

The assessment of the individual risk to any person off-site takes input from four sources: the Level 1 PSA, the Level 2 PSA and the Design Basis Assessment (DBA) together with the additional initiating events identified in section 3.1 of this sub-chapter.

The objective of the Level 1 PSA is to identify those sequences that result in a designated failure state and hence, by definition, fall outside the design basis, i.e. those that lead to core damage. The other sequences are classified as success states within the Level 1 PSA, or may not have been included in the PSA from the outset as they could not lead to core damage or are non reactor faults. Therefore, the success states within the Level 1 PSA may not contain sufficient detail to provide all the information required for the present Level 3 risk assessment.

The DBA (Chapter 14) does consider Low Consequence (LC) faults in considerable detail, and covers some of the initiating events excluded from the PSA as they cannot lead to core damage. Thus the majority of the additional plant analysis and plant data required for this Level 3 risk assessment is available from, or bounded by, the DBA.

The Level 2 PSA considers sequences that lead to core damage. As the containment function is considered as part of the Level 2 PSA, the results contain sufficient detail to allow this Level 3 risk assessment to be performed.

The methodology for the individual risk assessment is summarised in Sub-chapter 15.5 – Figure 1.

3.2.1. Release category allocation

Release Categories (RC) are defined in the EDF document [Ref-1] for the probabilistic assessment of individual risk. Knowledge of the associated source terms allows each of these RCs to be assigned to an off-site dose band.

Note:

The dose band classification for radiological exposure is as follows:

DB1 : 0.1-1 mSv ; DB2 : 1-10 mSv ; DB3 : 10-100 mSv ; DB4 : 100-1000 mSv ; DB5 : > 1000 mSv.

The radiological consequence which is considered is the unmitigated effective dose to a child at 500 m downwind from the point of release during the first 7 days following the release, with standard weather condition “DF2” (see Chapters 12, 14 and 16 on parts concerning radiological consequences).

The process is performed as follows:

- Identification of representative radiological release types, in terms of the systems and locations from which the release occurs
- Identification of the key characteristics of each of the representative radiological release types,
- Analysis of available results from the Level 1 and Level 2 PSA and DBA in order to evaluate the above parameters,
- Assignment of release categories, and hence dose bands, for each representative radiological release type.

Five representative radiological release types have been identified, corresponding to specific accidents: steam generator release, LOCA inside the reactor building, LOCA outside the reactor building, tank or waste treatment system breaks and fuel building accidents. For each main release type a number of variants, defined by the parameters shown as follows, were considered:

Release from a Steam Generator (SG release categories)

The main parameters that define variants are as follows:

- Primary coolant activity (value corresponding to either fuel clad failures or no fuel clad failure),
- Leakage between primary and secondary side (either due to operational leakage or to SGTR),
- Type of release from the SG (either steam or liquid),
- Transient at time of fault (either normal operation or combined with an aggravating transient).

Release following LOCA inside the Reactor Building (RB release categories)

The main parameters that define variants are as follows:

- Activity released into the containment (initial primary coolant activity with value corresponding to either fuel clad failures or no fuel clad failures, with or without subsequent core damage),
- Mode of containment leakage (either intact or not),
- Ventilation / filtration of containment leakage (either operating as expected or not).

Release following LOCA outside Reactor Building (CR release categories)

The main parameters that define variants are as follows:

- Activity of the leakage (value corresponding to either fuel clad failures or no fuel clad failure),
- Type of leakage (either liquid or steam),
- Amount of leakage (leakage isolated or not),
- Ventilation / filtration of building (either operating as expected or not).

Tank or waste treatment system break (T release categories)

The main parameters that define variants are as follows:

- Type and amount of release (release as either liquid or gas, total or partial),
- Ventilation / filtration of building (either operating as expected or not).

Fuel Building accidents (FB release categories)

The main parameters that define variants are as follows:

- Activity of the spent fuel pool (issued from either primary coolant activity value before shutdown or value corresponding to fuel rod failure of one fuel assembly, including radioactive decay),
- Pool temperature (either maximum Technical Specification value or boiling),
- Ventilation / filtration of building (either operating as expected or not).

The RCs that correspond to these release types and variants and their associated dose band allocations are shown in Sub-chapter 15.5 - Table 5.

3.2.2. Assessment of Level 1 PSA non core damage sequences

The aim is to assess the frequency and consequences of off-site releases resulting from non core- damage (success) sequences of the Level 1 PSA. All initiating events considered in the Level 1 PSA that can result in off-site releases are analysed. The analysis is performed for all the reactor states that are considered in the Level 1 PSA (state A to state E).

Methodology

This analysis is performed in two stages:

- End state allocation Level 1 PSA success sequences are evaluated from the standpoint of potential releases. Each success sequence is allocated a consequence end state that characterises the state of the plant with respect to the amount of radioactivity released and the potential pathways for that release.
- Source term and dose band identification: non core damage event trees are developed to produce the dose / frequency couplets resulting from the success sequences analysed in the first stage. These event trees model the mitigation and containment of the radioactive releases from the primary system. A simple source term model [Ref-1] is used to allocate dose bands to the different types of releases.

Level 1 PSA end state allocation

The success sequences are split into the following families:

- Transient sequences. All sequences where the integrity of the RCP [RCS] pressure boundary is maintained. The only potential activity release route is through the normal operational route from the primary side to the secondary side.
- LOCA sequences. Primary side activity is released into the Reactor Building.
- ISLOCA (Interfacing System LOCA) sequences. Primary side activity is released into the safeguard building or the nuclear auxiliary building.
- SGTR sequences. Primary side activity is directly released into the secondary side, and potentially to atmosphere through the affected steam generator.

End states are defined within each family to differentiate sequences on the basis of the amount of fuel cladding failure. The amount of fuel cladding failure has a direct influence on the radioactive inventory of the primary system. The amounts considered are no cladding failure, 1% cladding failure and 10% cladding failure. For SGTR sequences, another criterion used to differentiate between sequences is whether the affected steam generator is isolated or not. For ISLOCA sequences the distinction is made based on the timing of the break isolation. The list of end states used in this analysis is presented in Sub-chapter 15.5 - Table 6.

In general, the end state allocation is based upon the deterministic safety requirements of the Flamanville 3 (FA3) PSAR. In particular, the upper limit of 10% cladding failure corresponds to the maximum amount of clad failure expected for non-core damage events in the FA3 PSAR.

By applying the above process, the following end states are allocated:

- Transients: T or TF depending on whether cladding failure is expected (based on the FA3 FSAR safety analyses). The end state LF is allocated if feed and bleed is used.

- LOCAs inside the reactor building (i.e. excluding ISLOCAs): The end state LF is generally used, except for the following cases, based on expert judgment:

    - For small LOCAs and reactor coolant pump seal LOCAs with all mitigating equipment available, the end state L is used (i.e. no cladding failure is expected).

    - For medium and large LOCAs with all mitigating equipment available, the end state LF1 is used (i.e. 1% cladding failure is expected). For 2A-LOCAs, 10% cladding failure is assumed.

    - For LOCAs during shutdown reactor states, the end state L is used (i.e. no cladding failure is expected based on the Level 1 PSA success criteria for shutdown states).

- ISLOCAs: the end state V1 is used if automatic isolation is successful (i.e. if there is a limited release of coolant). The end state V2 is used if the break is isolated manually (i.e. there is potentially a large release of coolant). Success sequences for ISLOCAs are considered only for shutdown reactor states, therefore no cladding failure is expected.

- SGTRs: The end states PI or PN are used depending on whether the affected steam generator is isolated or not. Based on the Level 1 PSA support studies, no cladding failure is expected for success sequences.

<u>Source term and dose band identification</u>

For each of the end states defined above and shown in Sub-chapter 15.5 - Table 6, the potential fission product pathway is modelled using non core damage event trees. The following barriers to a release are modelled in these event trees:

- Containment isolation,

- Annulus ventilation,

- Auxiliary Building ventilation.

The event trees define sequences based on the availability or unavailability of each barrier. The system models developed in the Level 2 PSA are used to represent the barriers.

When necessary, different event trees are designed for each reactor state to reflect the status of the RCP [RCS] and of the containment during different reactor states.

Based on the success or failure of these barriers, each non core damage event tree sequence is assigned to an RC, as shown in Sub-chapter 15.5 - Table 5.

The radiological consequences of each RC are based on the DBA radiological study (Sub-chapter 14.6). The RCs thus provide the basis for linking non core damage event tree sequences with the appropriate off-site dose band. Sub-chapter 15.5 - Table 5 shows the correspondence between RCs and dose bands.

Cases that were not analysed in the DBA radiological study are shown in italic in Sub-chapter 15.5 - Table 5. For those, the dose band was extrapolated based on comparable cases within the analysis.

Summary of Results

The frequency contribution from each end state to each dose band in the individual risk assessment, and the fraction of the total Level 1 PSA contribution from each end state for that dose band is shown in Sub-chapter 15.5 - Table 7.

The table gives the frequency (per reactor year) that each end state contributes to each dose band together with the total frequency in each dose band. Also, for the main contributing end states, the percentage contribution of each end state to that total is shown.

3.2.3. Assessment of Level 2 PSA core damage sequences

In the Level 2 PSA (Sub-chapter 15.4), individual fault sequences leading to core damage are grouped, into a smaller number of fault classes with similar characteristics. These are termed Core Damage End States (CDES). The containment responses to these CDES are then quantified using the Containment Event Tree (CET). The resulting output of the Level 2 PSA is a further reduced number of accident classes, termed Release Categories, similar in principle to those defined for non core damage sequences as described in section 3.2.1, together with a predicted frequency and associated source term for each RC.

This existing information on the off-site consequences of core damage sequences is used to determine the dose band allocation for core damage RCs, as summarised in Sub-chapter 15.5 - Table 5.

The frequencies of the core damage RCs and their associated dose bands are shown in Sub-chapter 15.5 - Table 8.

3.2.4. Assessment of additional non-core damage sequences

Sub-chapter 15.5 - Table 4 lists the non core damage initiating events not considered in the Level 1 PSA.

To assess the risk contribution from the resulting sequences, the following steps have been followed for each of the events identified:

- Screen out initiating events which are assessed as low frequency (less than 1E-06/ry) and hence are unlikely to affect the outcome of the assessment significantly.

- Group the remaining initiating events into broad consequence bins based on the available design basis analysis. In a process comparable to that used for the Level 1 PSA success sequences, each consequence bin characterises the potential for off-site impact.

- Screen out initiating events which result in a deterministic off-site effective dose of less than 0.1 mSv, as this magnitude of release is sufficiently low to be discounted from this assessment. This may be done directly for DBA initiating events where detailed plant analysis is available. For those initiating events not considered in the DBA, the screening process is on the basis of similarity to DBA events and / or the Low Consequence release categories of Sub-chapter 15.5 - Table 5.

- Allocate the remaining initiating events to off-site dose bands according to the release categories of Sub-chapter 15.5 - Table 5. At this stage some consideration of the off-site consequences and conditional probabilities of partial failure of barriers to release is made.

Two sequences remain after the frequency and consequence screening steps: fuel handling accident in the spent fuel pool and fuel assembly drop occurring in the reactor building.

These sequences have been assessed individually, so they can be assigned to an appropriate dose band. This initial assessment uses the analysis results presented in Section 3.2.2 of this sub-chapter. Hence, particularly in the lowest dose bands, the assigned off-site consequences are likely to be upper bound estimates.

Two further potential accidents in the spent fuel pool, loss of cooling and pool drainage, have been included in the analysis using results available in Sub-chapter 15.3. It should be noted that this latter sub-chapter does not cover fuel handling accidents.

Complementarily to Sub-chapter 15.2, risk contribution from accidental aircraft crash on non fully protected buildings where radioactive containing systems are located (i.e. Nuclear Auxiliary Building and Effluent Treatment Building) has been assessed. Deterministic studies have shown that only transport aircraft crash on those buildings could lead to a non negligible off-site effective dose. Considering dose assessment for the release category T2 ‘Multiple failures of tanks, no filtration in Nuclear Auxiliary Building or Effluent Treatment Building’ in Sub-chapter 15.5 - Table 5 that corresponds to dose less than 0.1 mSv, this event has been assigned conservatively to dose band DB2.

The results, in terms of the summated frequency contribution to the dose bands resulting from these additional sequences, are shown in Sub-chapter 15.5 - Table 9.

3.2.5. Combination of results

The results from the three strands of the assessment (the Level 1 PSA, the Level 2 PSA and the additional non core damage sequences as described in sections 3.2.2 to 3.2.4 of this sub-chapter respectively) are presented in Sub-chapter 15.5 - Table 10. In addition to the contributions from each strand, the total for each dose band is presented.

Sub-chapter 15.5 - Figure 2 presents these results on a graph. The BSL and BSO value of Target 8 of the HSE Safety Assessment Principles are also shown.

It can be seen that the summated frequency of faults predicted to result in an off-site effective dose in excess of 0.1 mSv is 1.4E-03/ry (DB1 to DB5). Around 99% of this frequency is associated with very low off-site consequences in the lowest dose band, < 1mSv, and for this dose band the frequency is nearly one order of magnitude below the BSO. Only 2.3E-07/ry of this frequency is associated with off-site consequences above 100 mSv (DB4 and DB5).

It appears that the frequency associated with the lowest consequence dose band (DB1) is significantly higher than that associated with the higher consequence dose bands (DB2 to DB5). As discussed in section 3.2.4 of this sub-chapter, this is an artefact of using some conservative results of the consequence assessment performed for the DBA. If a less conservative, best estimate, consequence assessment were applied it is likely that the consequences of some of the dominant sequences would be shown to be below dose band 1. Screening out of these sequences as low consequence would then significantly reduce the frequency allocated to the lowest dose band, leading to a more balanced risk profile.

In dose band DB1 (0.1 to 1 mSv), the dominant events are non core damage sequences (86%), mainly due to SGTR (affected SG isolated), a fuel handling accident in the fuel building with 1 fuel assembly partially damaged (all fuel rods along one edge) and filtration available and fuel assembly drop in the reactor building (14%).

In dose band DB2 (1 to 10 mSv), the dominant events are fuel handling accident in the fuel building with 100% clad failure and filtration available (76%) and non core damage sequences, mainly due to SGTR (affected SG not isolated) (21%).

In dose band DB3 (10 to 100 mSv), the dominant events are a fuel handling accident in the fuel building with 100% clad failure and filtration not available (43%), core damage accidents with containment intact (annulus and building ventilation operational) (41%) and non core damage LOCA inside containment with 10% clad failure, containment intact but failure of the ventilation systems (16%).

In dose band DB4 (100 to 1000 mSv), the dominant events are core damage accidents with containment intact (failure of annulus and building ventilation) (~100%). The contribution from non core damage LOCA inside containment with 1% clad failure and containment bypass events is negligible (~0%).

In dose band DB5 (> 1000 mSv), the dominant events are core damage accidents with containment failure (95%) and fuel damage after loss of cooling or rapid drainage of the spent fuel pool (3%). The contribution from non core damage LOCA inside containment with 10% clad failure and containment bypass events is small (2%).

It is emphasised that the identification of the dominant events, as well as any analysis of risk balance, must be considered with care as modelling assumptions, especially where varying degrees of conservatism are introduced, lead to distortions in the risk breakdown and profile.

3.3. ASSESSMENT OF SOCIETAL RISK

One of the goals of the EPR design philosophy is to reduce the frequency of releasing substantial amounts of radioactivity into the off-site environment, compared to the current generation of PWRs. Consideration of the frequency of accidents leading to such large releases is an appropriate way to assess societal risk. Target 9 from the HSE Safety Assessment Principles sets a BSL of 1.0E-05/ry and a BSO of 1.0E-07/ry for the frequency of exceeding 100 immediate or eventual fatalities in the wider population

The Level 3 PSA identifies those accident sequences with the potential to result in environmental source terms sufficient to cause 100 or more fatalities in the wider population, both immediate and eventual, taking appropriate responses into account. The likelihood of early radiation induced fatalities is expected to be relatively low. The majority of the off-site fatalities are expected to be 'statistical deaths' arising from integrating low doses over very large populations over a significant period of time.

Within any Level 3 PSA, there is a large volume of input data that is not directly related to the power plant design and is either specific to the site or to the critical group habits. At this stage of the licensing process it is not possible to define such variables and a more generic approach is required. Therefore, the present assessment of societal risk is a scoping analysis that allows an early indication of the likelihood of a major accident, based on available information from the EPR Level 2 PSA, existing consequence analysis for UK sites and other relevant literature.

The process for assessing the likelihood of a major accident is summarised in Sub-chapter 15.5 - Figure 3.

The assessment relies on a screening of those RCs that have the potential to be a major accident, i.e. those expected to result in a substantial number of eventual fatalities. Other possible Level 3 PSA endpoints are not considered in the current process [Ref-1].

The screening rules are expressed in terms of the release to the environment of above threshold fractions of the core inventory, for specific radionuclide groups. These rules are formulated to be readily applied to the Level 2 PSA output; and it should be noted that they are different from the Large Release Frequency thresholds used as part of the Level 2 PSA.

The results of previous assessments of UK power station sites have been used to give an estimate of the consequences of a given release [Ref-2]. This data has been compared with information on the consequences of events at the Sizewell 'B' plant [Ref-3], output from the Level 2 PSA and criteria for the EPR design [Ref-4] to allow an estimate of the release that would lead to in excess of 100 fatalities.

This process has accounted for the likely variation in consequences across the different UK operating sites [Ref-2] by using the data for the most onerous site. Even though the site data from this reference are not up to date, the choice of the most onerous site allows the end result to be considered as reasonably conservative for a generic UK site.

From these data, a set of major accident screening rules have been produced as described below. A number of assumptions have been made in developing these screening rules. It should also be noted that the screening rules are intended to identify some RCs whose consequences may be below the expectation value of 100 fatalities but all identified RCs should result in expectation values of at least a few tens of eventual fatalities.

<u>Screening Rules:</u>

Any Level 2 PSA RC giving rise to predicted environmental releases that exceed any of the following three criteria is assumed to have the potential to result in more than 100 fatalities in the wider population:

1. > 0.1% of the core inventory in the iodine group,

2. > 0.01% of the core inventory in the caesium group,

3. > 0.001% of the core inventory in the ruthenium group AND either

- > 0.0001% of the core inventory in the lanthanides group, OR
- > 0.0001% of the core inventory in the cerium group.

Results

The results of this screening process are shown in Sub-chapter 15.5 - Table 11. The events which are identified on the basis of these criteria, as resulting in greater than 100 fatalities, are those leading to RCs that lie in dose band 5 of the individual risk assessment. The remaining sequences fall significantly below the screening criteria.

Events within the design basis are covered by the PCC analysis of Chapter 14 and the off-site releases are considered to be insufficient to lead to more than 100 fatalities. Similarly, the off-site releases of the RRC-A analysis presented in Chapter 16 are also insufficient to cause 100 fatalities.

The events considered to result in greater than 100 eventual fatalities are:

- those Release Categories from the Level 2 PSA that fall within dose band 5,
- the Release Categories from the Level 1 PSA non core damage sequences that fall within dose band 5 (section 3.2.2 of this sub-chapter),
- the Release Categories for the ‘additional events’ fuel damage sequences following water drainage in the spent fuel pool (see section 3.2.4 of this sub-chapter).

The summated frequency of the release categories which lead to more than 100 eventual fatalities, for a generic UK site, is 8.0E-08/ry.

The implications of the results for individual and societal risk are discussed more fully in Chapter 17.

SUB-CHAPTER 15.5 - TABLE 1

Initiating events considered in the design basis analysis of Chapter 14

N°	Initiating event	PCC
1	Increase and reduction in RCP [RCS] temperature	1
2	Step load changes	1
3	Ramp load changes	1
4	Load reduction, up to and including the design full load rejection	1
5	Loss of the grid, with the auxiliary system available	1
6	Loss of main feed water system (ARE [MFWS]) with the start-up and shutdown system (AAD [SSS]) available	1
7	Partial reactor trip	1
8	Spurious reactor trip	2
9	Main feed water (ARE [MFWS]) malfunction causing a reduction in feed water temperature	2
10	Main feed water (ARE [MFWS]) malfunction causing an increase in feed water flow	2
11	Excessive increase in secondary steam flow	2
12	Turbine trip	2
13	Loss of condenser vacuum	2
14	Loss of normal feed water flow (loss of all ARE [MFWS] pumps and the start-up and shutdown (AAD [SSS]) pump)	2
15	Partial loss of core coolant flow (loss of one reactor coolant pump)	2
16	Uncontrolled rod cluster control assembly (RCCA) bank withdrawal	2
17	RCCA Misalignment up to rod drop, without limitation function	2
18	Start-up of an inactive reactor coolant loop at an improper temperature	2
19	RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant	2
20	RCV [CVCS] malfunction causing increase or decrease of the reactor coolant inventory	2
21	Primary side pressure transients (spurious pressuriser spraying, spurious pressuriser heating)	2
22	Uncontrolled RCP [RCS] level drop	2
23	Loss of one cooling train of the RIS/RRA [SIS/RHRS] in RHR mode	2
24	Loss of one train of the fuel pool cooling system (PTR [FPCS])	2
25	Small steam or feed water system piping failure including break of connecting lines (not greater than DN 50)	3
26	Inadvertent opening of a pressuriser safety valve	3
27	Inadvertent opening of a steam generator relief train (VDAa [MSRT]) or of a steam generator safety valve (MSSV)	3
28	Small break (not greater than DN 50), including a break occurring on the extra boration system (RBS [EBS]) injection line	3
29	Steam generator tube rupture (SGTR) (1 tube)	3
30	Inadvertent closure of one/all main steam isolation valves (VIV [MSIV])	3
31	Inadvertent loading of a fuel assembly in an improper position	3

N°	Initiating event	PCC
32	Forced decrease of reactor coolant flow (4 pumps)	3
33	Leak in the gaseous waste processing systems	3
34	Uncontrolled rod cluster control assembly (RCCA) bank withdrawal	3
35	Uncontrolled single rod cluster control assembly withdrawal	3
36	Loss of primary coolant outside the containment	3
37	Long term loss of off-site power supplies (> 2 hours), fuel pool cooling aspect	3
38	Loss of one train of the fuel pool cooling system (PTR [FPCS])	3
39	Isolatable piping failure on a system connected to the fuel pool	3
40	Long term loss of off-site power in state C (> 2 hours)	4
41	Feed water system piping break	4
42	Inadvertent opening of SG relief train or safety valve	4
43	Spectrum of RCCA ejection accidents	4
44	Intermediate and Large Break LOCA (up to the surge line break in states A and B)	4
45	Small break LOCA (not greater than DN 50), including a break in the RBS [EBS] injection line	4
46	Reactor coolant pump seizure (locked rotor)	4
47	Primary coolant pump shaft break	4
48	Steam generator tube rupture (2 tubes in 1 SG)	4
49	Fuel handling accident (spent fuel pool)	4
50	Boron dilution due to a non-isolatable rupture of a heat exchanger tube	4
51	Isolatable safety injection system break (≤ DN 250), in residual heat removal mode	4
52	Safety injection system break in residual heat removal mode - fuel pool drainage aspect	4
53	Rupture of radioactivity containing systems	4

SUB-CHAPTER 15.5 - TABLE 2

Initiating events considered in the Level 1 PSA

N°	**Initiating event**
54	Loss of primary coolant accident (LOCA)
55	Interfacing system LOCA (ISLOCA)
	Secondary system breaks:
56	Breaks on secondary side (steam or feed water)
57	Secondary break and SGTR
58	Steam generator tubes rupture (SGTR)
	Secondary system transients:
59	Total loss of main feed water
60	Loss of the start-up and shutdown feed system
61	Loss of condenser
62	Spurious Turbine Trip
	Loss of off-site power (LOOP):
63	Short loop < 2 hours – Plant states A and B
64	Short loop < 2 hours – Plant states C and D
65	Long LOOP < 24 hours – Plant states A and B
66	Long LOOP < 24 hours – Plant states C and D
67	Short consequential LOOP
68	Long consequential LOOP
	Primary system transients:
70	Homogeneous dilutions
71	Total loss of RIS [SIS] cooling in RHR mode
72	Uncontrolled drop of primary coolant level
73	Spurious Reactor Trip
	Loss of cooling water systems:
74	Partial or total loss of cooling system in power states
75	Total loss of the cooling chain in shutdown
76	Transients without automatic reactor shutdown (ATWS)
77	Heterogeneous external dilutions
102	Double ended primary guillotine break (LOCA - 2A)
124	Hazards (including accidental aircraft crash)

SUB-CHAPTER 15.5 - TABLE 3

Sequences added after expert review, including RCC-A, and low consequence events

N°	Initiating events
78	ATWS caused by rods failure - excessive increase of the steam flow rate on the secondary side (opening of the GCT [MSB])
79	ATWS caused by rods failure – loss of normal SG feed water supply
80	ATWS caused by rods failure - loss of the main power grid
81	ATWS caused by rods failure - failure of the RCV [CVCS] that leads to a decrease in the boron concentration of the primary coolant
82	ATWS caused by rods failure – uncontrolled withdrawal of a group of control rods
83	ATWS caused by failure of the protection system signal - excessive increase of the steam flow rate (opening of the GCT [MSB])
84	ATWS caused by failure of the protection system signal – loss of normal SG feed water supply
85	ATWS caused by failure of the protection system signal – loss of main power grid
86	ATWS caused by the failure of the protection signal - failure of the RCV [CVCS] that leads to a decrease of the boron concentration of the primary coolant
87	ATWS caused by failure of the protection system – uncontrolled withdrawal of a group of control rods
88	Total loss of off-site and onsite power supply (Station Blackout)
89	Total loss of feed water supply to the steam generators
90	Total loss of the cooling chain leading to leakage at the seals of the primary coolant pumps
91	LOCA (breach of size smaller than 20 cm²) with failure of the Safety Injection signal
92	LOCA (breach of size smaller than 20 cm²) without MHSI
93	LOCA (breach of size smaller than 20 cm²) without LHSI
94	Uncontrolled drop in the primary level without ISI signal of the protection system
95	Homogeneous dilution not from the RCV [CVCS] with failure by the operator to isolate the dilution source
96	Total loss of cooling chain in state D
97	LOCA outside containment on RIS/RRA [SIS/RHRS] train and failure of the automatic isolation signal or the injection signal
98	Total loss of the ultimate heat sink for 100 hours in states A and C
99	Loss of the two main trains of the fuel pool cooling system (PTR [FPPS/FPCS]) during shutdowns for refuelling/ Station Blackout
100	Common Cause Failure of LH switchboards [CCF-LH]
101	Loss of RCV [CVCS] pumping
103	Multiple steam generator tube rupture (SGTR - 20A)
104	Fuel assembly drop (into reactor building)
105	Fuel assembly mishandling (into reactor building)

N°	**Initiating events**
106	Fuelling machine fault leading to dropping of heavy items into the core leading to clad failure
107	Fuel element stuck in the fuel transfer mechanism
108	Heavy object dropped into fuel pool
109	Faults associated with spent resin transfer
110	Inadvertent discharge of liquid waste
111	Inadvertent discharge of gaseous waste
112	Dropping of solid waste package
113	Dropping of activated filter
114	Fire involving solid waste material
115	Inadvertent operation of the auxiliary feed water system
116	Inadvertent partial cooldown to below 80 bar
118	RCCA Bank misalignment
119	Decay of frequency of off-site power system
120	Faults in sensors common to the control and protection system
121	Loss of spent fuel pool cooling
122	Rapid drainage of the spent fuel pool
123	Boiling of the spent fuel pool

SUB-CHAPTER 15.5 - TABLE 4

Assessment of additional initiating events not included in Level 1 PSA

N°	Initiating events	Ass[t]
1	Increase and reduction in RCP [RCS] temperature	LowC
2	Step load changes,	LowC
3	Ramp load changes	LowC
4	Load reduction, up to and including the design full load rejection	LowC
5	Loss of the grid, with the auxiliary system available	LowC
7	Partial reactor trip	LowC
9	Main feed water (ARE [MWFS]) malfunction causing a reduction in feed water temperature	LowC
10	Feed water (ARE [MWFS]) malfunction causing an increase in feed water flow	LowC
11	Excessive increase in secondary steam flow	LowC
15	Partial loss of core coolant flow (loss of one RCP)	LowC
16	Uncontrolled rod cluster control assembly (RCCA) bank withdrawal	LowC
17	RCCA misalignment up to rod drop, without limitation function	LowC
18	Start-up of an inactive reactor coolant loop at an improper temperature	LowC
19	RCV [CVCS] malfunction that results in a decrease in boron concentration in the reactor coolant	LowC
20	RCV [CVCS] malfunction causing increase or decrease of the reactor coolant inventory	LowC
21	Primary side pressure transients (spurious pressuriser spraying, spurious pressuriser heating)	LowC
23	Loss of one cooling train of the RIS/RRA [SIS/RHRS] in RHR mode	LowC
24	Loss of one train of the fuel pool cooling system (PTR [FPCS])	LowC
30	Inadvertent closure of one/all main steam isolation valves	LowC
31	Inadvertent loading of a fuel assembly in an improper position	LowC
32	Forced decrease of reactor coolant flow (4 pumps)	LowC
33	Leak in the gaseous waste processing systems	LowC
34	Uncontrolled rod cluster control assembly (RCCA) bank withdrawal	LowC
35	Uncontrolled single rod cluster control assembly withdrawal	LowC
38	Loss of one train of the fuel pool cooling system (PTR [FPCS])	LowC
39	Isolatable piping failure on a system connected to the fuel pool	LowC
46	Reactor coolant pump seizure (locked rotor)	LowC
47	Primary coolant pump shaft break	LowC
49	Fuel handling accident (spent fuel pool)	TBC
53	Rupture of radioactivity containing tank/pipe in radwaste systems	LowC
100	Common Cause Failure of LH switchboards [CCF-LH]	LowF
101	Loss of RCV [CVCS] pumping.	LowF
103	Multiple steam generator tube rupture (SGTR - 20A)	LowF
104	Fuel assembly drop (into reactor building)	TBC
105	Fuel assembly mishandling (in reactor building)	LowC
106	Fuelling machine fault leading to dropping of heavy items into the core leading	LowF

N°	**Initiating events**	**Ass[t]**
	to clad failure	
107	Fuel element stuck in the fuel transfer mechanism	LowC
108	Heavy object dropped into fuel pool	LowF
109	Faults associated with spent resin transfer	LowC
110	Inadvertent discharge of liquid waste	LowC
111	Inadvertent discharge of gaseous waste	LowC
112	Dropping of solid waste package	LowC
113	Dropping of activated filter	LowC
115	Inadvertent operation of the auxiliary feed water system	LowC
116	Inadvertent partial cooldown to below 80 bar	LowC
118	RCCA Bank misalignment	LowC
119	Decay of frequency of off-site power system	LowC
120	Faults in sensors common to the control and protection system	LowF
121	Loss of spent fuel pool cooling	TBC
122	Spent fuel pool drainage	TBC
123	Boiling of the spent fuel pool	LowC

Assessment:

LowC: event screened out on Low Consequence
LowF: event screened out on Low Frequency
TBC: event to be considered in off-site consequence assessment

SUB-CHAPTER 15.5 - TABLE 5

{CCI Removed}

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a

SUB-CHAPTER 15.5 - TABLE 6

End States Definition for PSA Level 1 Success Sequences

End state ID	Description
T	Transient sequence, with no cladding failure
TF	Transient sequence, with 10% cladding failure
L	LOCA inside containment, with no cladding failure
LF1	LOCA inside containment with 1% cladding failure
LF	LOCA inside containment, with 10% cladding failure
PI	SGTR with no cladding failure, where the affected SG is isolated
PN	SGTR with no cladding failure, where the affected SG is not isolated
V1	Interfacing system LOCA isolated automatically with no cladding rupture
V2	Interfacing LOCA isolated manually with no cladding rupture

SUB-CHAPTER 15.5 - TABLE 7

Non core damage End states from Level 1 PSA - Frequencies/Dose Bands

This table presents the frequency (per year) that each end state contributes to each dose band together with the total frequency in each dose band and, for the main contributing end states, the percentage contribution to that total

LCHF End State Description	End state ID	DB0 < 0.1 mSv	DB1 0.1 – 1mSv	DB2 1 – 10mSv	DB3 10 – 100mSv	DB4 100 – 1000mSv	DB5 > 1000mSv	Total
LOCA inside containment without fuel clad failure in plant states A and B	**L**	2.87E-03 (0.2%)		1.07E-07 (3.9%)				2.87E-03
LOCA inside containment without fuel clad failure in plant states C to E	**L-CA,CB D,E**	6.41E-05	8.33E-06 (0.7%)	4.34E-08 (1.6%)				7.24E-05
LOCA inside containment with 1% fuel clad failure	**LF1**	1.73E-05		6.12E-09 (0.2%)		6.44E-10 (100%)		1.73E-05
LOCA inside containment with 10% fuel clad failure	**LF**		5.92E-06 (0.5%)		1.82E-07 (100%)		1.50E-09 (100%)	6.10E-06
SGTR with no fuel clad failure, affected SG isolated	**PI**		1.18E-03 (97.5%)					1.18E-03
SGTR with no fuel clad failure, affected SG not isolated	**PN**			2.61E-06 (94.3%)				2.61E-06
Transient sequence without fuel clad failure in reactor states A to C	**T**	1.84E+00 (99.2%)						1.84E+00
Transients without fuel clad failure in reactor states D	**T-D**	1.05E-03 (0.1%)	6.98E-07 (0.1%)					1.05E-03
Transient sequence with 10% fuel clad failure	**TF**		1.48E-05 (1.2%)					1.48E-05
Small Interfacing LOCA	**V1**	1.03E-02 (0.6%)						1.03E-02
Large Interfacing LOCA	**V2**	1.34E-07	1.36E-08					1.48E-07
	TOTALS	1.85E+00	1.21E-03	2.77E-06	1.82E-07	6.44E-10	1.50E-09	

SUB-CHAPTER 15.5 - TABLE 8

Core Damage accidents (Release Categories) covered in Level 2 PSA

Doseband (**mSv**)	**RC no.**	**Containment failure mode**	**Sprays**	**Frequency (/y)**	**Total Frequency (/y)**
DB5 > 1000	RC 200	Isolation failure – in-vessel recovery	Yes	9.02E-10	7.56E-08 (11%)
	RC 201	Isolation failure – in-vessel recovery	No	3.01E-10	
	RC 202	Isolation failure	Yes	2.60E-12	
	RC 203	Isolation failure	No	3.04E-13	
	RC 204	Isolation failure	Yes	1.95E-09	
	RC 205	Isolation failure	No	4.51E-10	
	RC 206	All small isolation failures (< 2 inch)	No	4.61E-09	
	RC 301	Early failure	Yes	8.06E-12	
	RC 302	Early failure	no	5.84E-12	
	RC 303	Early failure	yes	1.02E-08	
	RC 304	Early failure	no	6.98E-09	
	RC 401	Intermediate failure	yes	2.67E-11	
	RC 402	Intermediate failure	no	8.37E-12	
	RC 403	Intermediate failure	yes	1.23E-09	
	RC 404	Intermediate failure	no	1.09E-09	
	RC 501	Late failure	yes	6.51E-13	
	RC 502	Late failure	no	3.96E-11	
	RC 503	Late failure	yes	1.27E-09	
	RC 504	Late failure	no	3.29E-08	
	RC 602	Basemat failure	no	6.57E-10	
	RC 701	SGTR (scrubbed)	no	4.14E-09	
	RC 702	SGTR (unscrubbed)	no	5.01E-09	
	RC 802	Large ISLOCA unscrubbed, deposition in building	no	3.83E-09	
DB4 100 - 1000	RC 101	Containment intact Deposition in annulus and building	no	1.49E-07	1.49E-07 (21%)
DB3 10 - 100	RC 102	Containment intact Annulus and building ventilation operational	no	4.84E-07	4.84E-07 (68%)
DB2 1 - 10	n/a		-	-	-
DB1 0.1 - 1	n/a		-	-	-
	TOTAL				**7.08E-07**

SUB-CHAPTER 15.5 - TABLE 9

Results from assessment of additional sequences not modelled in the Level 1 PSA

Dose Band (mSv)	Frequency (/ry)			
	IE no.	**Description**	**Frequency**	**Total Frequency**
DB5 > 1000	*See 15.3*	Spent fuel pool drainage	2.3E-09	2.6E-09
	See 15.3	Loss of spent fuel pool cooling	2.5E-10	
DB4 100 - 1000	n/a			
DB3 10 - 100	49	Fuel handling accident (spent fuel pool) 100% clad failure, no filtration	5.0E-07 (1)	5.0E-07
DB2 1 - 10	49	Fuel handling accident (spent fuel pool) 100% clad failure, filtration ok	1.0E-05 (1)	1.0E-05
	See 15.2	Accidental transport aircraft crash	{CCI removed}[a]	
DB1 0.1 - 1	49	Fuel handling accident (spent fuel pool) 6% clad failure, filtration ok	1.0E-04 (1)	2.0E-04
	104	Fuel assembly drop (in reactor building)	1.0E-04	

(1) A small frequency contribution to this dose band has been derived from the identified fuel handling faults in DB1. The part of the initiating event frequency allocated to DB1 represents the initiating event and successful mitigation. A small fraction of this frequency is allocated to higher dose bands, associated with the conditional probability of additional failure of mitigation systems (e.g. 5E-02 for the probability of failure of the fuel building ventilation and filtration systems).

(2) Accidental transport aircraft crash rate for Nuclear Auxiliary Building and Effluent Treatment Building

SUB-CHAPTER 15.5 - TABLE 10

Results for Assessment of Individual Risk

Off-site effective dose band (mSv)	Frequency (/ry)			
	Non core damage sequences from Level 1 PSA	**Additional non core damage sequences**	**Core damage sequences from Level 2 PSA**	**Total**
DB5 > 1000	1.5E-09 (2%)	2.6E-09 (3%)	7.6E-08 (95%)	**8.0E-08**
DB4 100 - 1000	6.4E-10 (0%)	0	1.5E-07 (100%)	**1.5E-07**
DB3 10 - 100	1.8E-07 (16%)	5.0E-07 (43%) (1)	4.8E-07 (41%)	**1.2E-06**
DB2 1 – 10	2.8E-06 (21%)	1.0E-05 (79%) (1)	0	**1.3E-05**
DB1 0.1 - 1	1.2E-03 (86%)	2.0E-04 (14%)	0	**1.4E-03**

(1) The frequency contribution to this dose band is derived from fuel handling faults allocated to DB1. This frequency is associated with the conditional probability of additional failure of mitigation systems.

SUB-CHAPTER 15.5 - TABLE 11

Results for Assessment of Societal risk

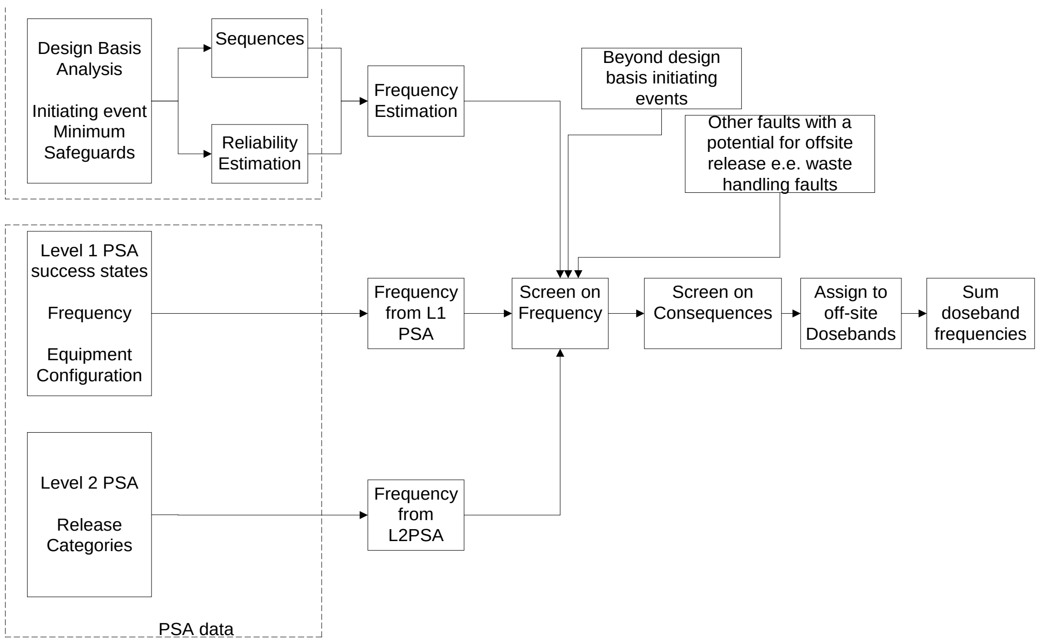
Contributing RCs (numbers for those from Level 2 PSA)	**Frequency (/y)**
504	3.29E-08
303	1.02E-08
304	6.98E-09
702	5.01E-09
206	4.61E-09
701	4.14E-09
802	3.83E-09
Drainage of Spent fuel pool	2.30E-09
204	1.95E-09
Non core damage sequences, containment bypass	1.50E-09
503 (1)	1.27E-09
403	1.23E-09
404	1.09E-09
200	9.02E-10
602	6.57E-10
205	4.51E-10
201	3.01E-10
Loss of Spent Fuel Pool cooling	2.54E-10
502	3.96E-11
401	2.67E-11
402	8.37E-12
301	8.06E-12
302	5.84E-12
202	2.60E-12
501 (1)	6.51E-13
203	3.04E-13
All (2)	**8.0E-08**

(1) RCs 501 and 503 have been identified as borderline using the major accident screening criteria.

(2) The summated frequency is rounded up.

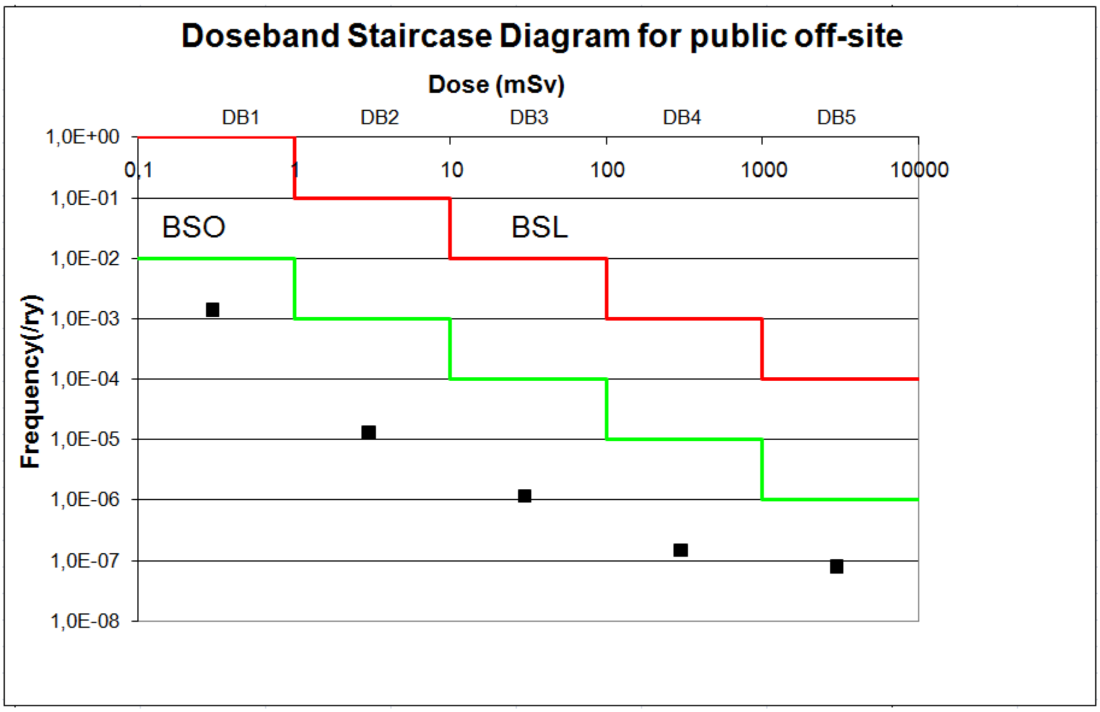
SUB-CHAPTER 15.5 - FIGURE 1

Methodology for Assessment of Individual Risk



SUB-CHAPTER 15.5 - FIGURE 2

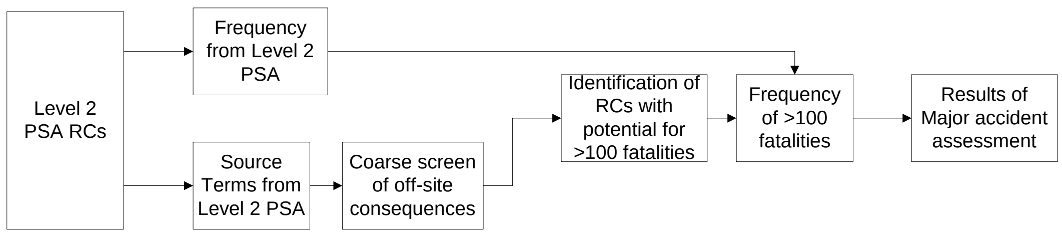
Comparison of the individual risk assessment results to Target 8 of the HSE SAPs



Black squares represent the frequency / dose couplets for the five dose bands.

SUB-CHAPTER 15.5 - FIGURE 3

Methodology for Assessment of Societal Risk



SUB-CHAPTER 15.5 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

3. PROCESS OF ASSESSMENT

3.1. IDENTIFICATION OF INITIATING EVENTS FOR ASSESSMENT

[Ref-1] S C Bubb, C Niculae. Screening of Initiating Events to Support the Dose Band Assessment for the Step 3 Submission.
AMEC report 14782/TR/0001 Issue 01. June 2008. (E)

3.2. ASSESSMENT OF INDIVIDUAL RISK

3.2.1. Release category allocation

[Ref-1] Release categories for the LCHF / Level 3 PSA (supporting study for PCSR chapter 15).
EDF/SEPTEN ENTEAG080122 Revision A. June 2008. (E)

3.2.2. Assessment of Level 1 PSA non core damage sequences

Methodology

[Ref-1] Release categories for the LCHF / Level 3 PSA (supporting study for PCSR chapter 15).
EDF/SEPTEN ENTEAG080122 Revision A. June 2008. (E)

3.3. ASSESSMENT OF SOCIETAL RISK

[Ref-1] E Grindon. Support Study for the Societal Risk Assessment to Support the Step 3 Submission.
AMEC report 14782/TR/0002 Issue 1. June 2008. (E)

[Ref-2] Safety Review Guidebook for the Gas Cooled Reactors.
Nuclear Electric Report. September 1993. (E)

[Ref-3] G N Kelly and R H Clarke. An Assessment of the Radiological Consequences of Releases from Degraded Core Accidents for the Sizewell PWR.
NRPB-R137. July 1982. (E)

[Ref-4] European Utility Requirements for LWR Nuclear Power Plants, Volume 2: Generic Nuclear Island Requirements. Revision C. EUR Document. April 2001. (E)