



ONR GUIDE			
RADIOLOGICAL ANALYSIS for FAULT CONDITIONS			
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1. INTRODUCTION

- 1.1 The Office for Nuclear Regulation (ONR) has established its Safety Assessment Principles (SAPs) which apply to the assessment by ONR specialist inspectors of safety cases for nuclear facilities that may be operated by potential licensees, existing licensees, or other duty-holders. The principles presented in the SAPs are supported by a suite of guides to further assist ONR's inspectors in their technical assessment work in support of making regulatory judgements and decisions. This Technical Assessment Guide (TAG) is one of these guides.

2. PURPOSE AND SCOPE

- 2.1 ONR has the responsibility for regulating the safety of nuclear installations in Great Britain (GB). It does this by operating a non-prescriptive, permissioning regime which places the onus on the duty holder to build a safety case to justify permissioned activities. The Safety Assessment Principles for Nuclear Facilities (SAPs) (Ref 1) provide a framework to guide ONR's decision making in the nuclear permissioning process. The SAPs are supported by Technical Assessment Guides (TAGs) that provide guidance on the interpretation of the principles to assist assessors in the exercise of their professional regulatory judgements about the adequacy of safety submissions.
- 2.2 The purpose of this TAG is to illuminate and support the guidance within the SAPs on the radiological analysis of faults, including Targets 4 – 9 which lay down numerical criteria for fault frequencies, radiation doses and risks to employees and others. The background to these Targets and an explanation of the associated Basic Safety Levels (BSLs) and Basic Safety Objectives (BSOs) are given in the Explanatory Note on the Numerical Targets and Legal Limits (Ref 2).

3. THE SITE LICENCE AND OTHER RELEVANT LEGISLATION

- 3.1 The Nuclear Installations Act 1965 (as amended) permits ONR to attach to the site licence conditions as may appear to be necessary or desirable in the interests of safety. Licence Condition 14 (Safety documentation) requires duty holders to make adequate arrangements for the production and assessment of safety case documentation to justify safety during design, construction, manufacture, commissioning, operation and decommissioning phases of the installation. Safety case documentation is also relevant to Licence Conditions 15 (Periodic review), 19 (Construction or installation of new plant), 20 (Modification to design of plant under construction), 21 (Commissioning), 22 (Modification or experiment on existing plant) and 23 (Operating rules). Radiological analysis of the consequences of faults forms an important component of these safety cases.
- 3.2 In the UK, the operators of nuclear installations in England, Scotland and Wales, in common with other employers, must comply with the general provisions of the Health and Safety at Work etc. Act 1974 (HSW Act) (Ref 3). In particular, it is their duty:
- 1) to ensure, so far as is reasonably practicable, the health, safety and welfare at work of all their employees (Section 2 of the HSW Act); and
 - 2) to conduct their undertaking in such a way as to ensure, so far as is reasonably practicable, that persons not in their employment who may be affected are not thereby exposed to risks to their health or safety (Section 3 of the HSW Act).

An assessment of the radiological consequences of faults is an important input into decisions on reasonably practicable measures to ensure the safety of persons on and off the licensed site.

4. RELATIONSHIP TO SAPS, WENRA REFERENCE LEVELS AND IAEA SAFETY STANDARDS

- 4.1 Part of the specification for the update of the Safety Assessment Principles was to consider the Reactor, Decommissioning and Storage Safety Reference Levels published by the Western European Nuclear Regulators' Association (WENRA), and International Atomic Energy Agency (IAEA) Standards, Guidance and Documents. Nevertheless, detailed review of the documents shows that the WENRA reference levels are not directly relevant to an assessment guide for radiological consequences analysis. While there are no IAEA documents dedicated to radiological consequences analysis, documents on the design of facilities in the Safety Standards series can be used to define requirements for radiological consequence analysis.

Application of ONR SAPs to radiological analysis for faults

High level principles for performing the radiological analysis for faults are set out in the Fault Analysis section of the SAPs (Ref 1, pp134-149), specifically in Fault Analysis Principles FA.1 – FA.16, FA.25 and AV.1 – AV.8. In addition, the Fundamental Principles FP.1 – FP.8 (Ref 1, pp16-17) also apply, notably FP.5 and FP.6, which highlight the need for all reasonably practicable measures in order to control radiation risks so that no individual bears an unacceptable risk of harm and also to prevent and mitigate nuclear or radiation accidents.

The numerical targets (Ref 1, pp150-164, paragraphs 721-758) support the SAPs by providing the assessor with more detailed guidance for specific types of radiological analysis. There is no obligation for the duty holder to use these numerical targets, however the assessor should assess against these targets or their equivalents where the duty holder uses alternative approaches.

Numerical radiological targets

- 4.2 Numerical Targets 1-3 apply to normal operational situations, so are not considered in this guidance. The remaining Targets 4-9 have been reproduced below for convenience. Targets 4-9 are not "legal" dose limits; they are akin to reference levels that have been derived from arguments about the acceptable level of risk from licensed nuclear facilities made at public inquiries and UK policy statements on the relative risk from different activities in the UK (Ref 7 and Ref 8). Background on the derivation of the Targets is given in the Appendix attached to the SAPs (Ref 2).

Design basis fault sequences – any person	Target 4
<p>The targets for the effective dose received by any person arising from a design basis fault sequence are:</p> <p>On-site</p> <p>BSL: 20 mSv for initiating fault frequencies exceeding 1×10^{-3} pa 200 mSv for initiating fault frequencies between 1×10^{-3} and 1×10^{-4} pa 500 mSv for initiating fault frequencies less than 1×10^{-4} pa</p> <p>BSO: 0.1 mSv</p> <p>Off-site</p> <p>BSL: 1 mSv for initiating fault frequencies exceeding 1×10^{-3} pa 10 mSv for initiating fault frequencies between 1×10^{-3} and 1×10^{-4} pa 100 mSv for initiating fault frequencies less than 1×10^{-4} pa.</p> <p>BSO: 0.01 mSv</p>	

Individual risk of death from accidents – any person on the site	Target 5
<p>The targets for the individual risk of death to a person on the site, from accidents at the site resulting in exposure to ionising radiation, are:</p> <p>BSL: 1×10^{-4} pa BSO: 1×10^{-6} pa</p>	

Frequency dose targets for any single accident – any person on the site	Target 6	
<p>The targets for the predicted frequency of any single accident in the facility, which could give doses to a person on the site, are:</p>		
Effective dose, mSv	Predicted frequency per annum	
	BSL	BSO
2 - 20	1×10^{-1}	1×10^{-3}
20 - 200	1×10^{-2}	1×10^{-4}
200 - 2000	1×10^{-3}	1×10^{-5}
> 2000	1×10^{-4}	1×10^{-6}

Individual risk to people off the site from accidents	Target 7
The targets for the individual risk of death to a person off the site, from accidents at the site resulting in exposure to ionising radiation, are:	
BSL: 1×10^{-4} pa	
BSO: 1×10^{-6} pa	

Frequency dose targets for accidents on an individual facility – any person off the site	Target 8	
The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site, are:		
Effective dose, mSv	Total predicted frequency per annum	
	BSL	BSO
0.1 - 1	1	1×10^{-2}
1 - 10	1×10^{-1}	1×10^{-3}
10 - 100	1×10^{-2}	1×10^{-4}
100 - 1000	1×10^{-3}	1×10^{-5}
> 1000	1×10^{-4}	1×10^{-6}

Total risk of 100 or more fatalities	Target 9
The targets for the total risk of 100 or more fatalities, either immediate or eventual, from accidents at the site resulting in exposure to ionising radiation, are:	
BSL: 1×10^{-5} pa	
BSO: 1×10^{-7} pa	

5. GUIDANCE FOR ASSESSORS

Role of Radiological Consequences Analysis in the Nuclear Safety Case

- 5.1 Nuclear safety cases commonly argue that the risk from accidents at the plant is acceptable by evaluating the risk and comparing it against standards. As radiological consequences are one of the two components that contribute to risk, the other being the frequency of the accident occurring, safety cases contain radiological consequences analysis in support of safety claims
- 5.2 Since the 1950's a framework for evaluating and controlling risk from exposure to ionising radiation has been developed by the International Commission on Radiological Protection (ICRP) (Ref 4, 5 and 6). In the first instance this was based on epidemiological studies carried out on several highly exposed populations, including the victims of the Japanese atomic bombs and miners exposed to high levels of radon gas.
- 5.3 The framework has been widely adopted and is incorporated in regulations for nuclear safety in nearly all countries with nuclear energy programmes. It forms the basis of the radiological consequences analysis typically presented in nuclear safety cases. A key element of the approach is the use of the committed effective dose quantity (unit Sieverts) (Ref 6) to measure the long term (stochastic) consequences of an exposure to ionising radiation. An alternative formulation based on absorbed dose (unit Grays) (Ref 6) is used for exposures occurring at very high dose rates which may result in deterministic health effects, such as bone marrow syndromes or acute radiation effects.
- 5.4 While the ICRP system for quantifying radiological consequences has been a valuable tool for making decisions that have systematically driven down public exposures since its introduction, the inspector should bear in mind that there are steps in the science that are not fully understood, and that the methodology has uncertainties associated with models that may be a relevant consideration for some regulatory decisions.

Recommended Good Practice (RGP) for Radiological Consequences Analysis

- 5.5 Numerical Targets 4 – 9 and the supporting text in paragraphs 721-758 of the SAPs (Ref 1) are concerned with the predicted radiation doses and the associated risks to persons on and off the site during fault conditions at facilities on the licensed site. For the performance of radiological consequences analysis, this TAG will reference specific pieces of Recommended Good Practice (RGP) as they occur in the guidance. In addition, the reader should be aware of the publications that can be accessed through the public portals of the following organisations
- IAEA in its Basic Safety Standards Series (Ref 9).
 - ICRP for detailed advice on methods for performing radiological consequences analysis for classes of individuals and groups, as well as the conceptual framework employed for radiological consequences analysis (Ref 10).
 - PHE and its predecessors have published a range of authoritative documents on radiological assessment topics since the 1970s (Ref 11, 12).
- 5.6 There are several factors in radiological consequences analysis which may affect how much support it provides for a safety case argument: the strongest and most relevant radiological analysis is always bespoke analysis that has been designed to support a specific safety claim for particular plant within a specific regulatory environment.

Nevertheless, the nature of commercial nuclear plant design is such that this condition will rarely be completely met and the assessor must make a judgement as to the level of support a particular piece of analysis provides.

- 5.7 Radiological consequences analysis generally forms part of a broad analytical approach that interfaces with a number of disciplines, including fault and ALARP studies, human factors and reactor chemistry. Guidance on other aspects is provided in other TAGs (Refs. 13, 14, 15, 16, 17,18).
- 5.8 The assessment of the predicted doses/risks aims to establish whether the actual doses/risks are likely to be within levels that should be met and also the extent to which the doses/risks have been shown to be ALARP. It is ONR's policy that a new facility should at least meet the BSLs, but our SAPs make it clear that the BSOs reflect modern safety standards and expectations. Assessors should use their judgement and discretion to ensure that the assessment is proportionate and targeted. The depth and scope to which this guidance is employed should be on a case by case basis.

Fault Analysis

- 5.9 The radiological analysis involves the predictions of doses to persons on the site and off the site from faults affecting the facility and is an input to three types of fault analysis identified by the SAPs, namely:
- Design Basis Analysis (DBA) (Target 4)
 - Probabilistic Safety Analysis (PSA) and Severe Accident Analysis (SAA) (Targets 5-9)
- 5.10 Individual faults lacking the potential to lead to doses > 0.1 mSv to a person on the site or > 0.01 mSv to a person off site are regarded as part of normal operation and excluded from the fault analysis (SAPs para 618). The results of the radiological analyses should be judged against Numerical Targets 4 – 9.

Radiological consequences to persons on the site

- 5.11 The models for estimating doses to persons on the site, usually to the operators in the facility of interest, are generally specific to the nature of the operations being undertaken and the associated fault conditions. It is reasonable to assume that a person working on the site is an adult. The assessor should be aware of the principles and guidance for the protection of persons on-site in the event of a radiological accident (Ref 19).
- 5.12 The assessor should consider the duty holder's methods on a case by case basis. Doses from all pathways should be considered for all faults analysed. Key assumptions are the locations of the worker in relation to the radioactive material, and the occupancy times: both of these factors should be checked for plausibility. The calculation of the dose to a person should be based on radiation levels, airborne contamination levels and exposure times. Attenuation provided by any shielding and the use of personal protective equipment may be taken into account if it is reasonable to expect it to be worn when calculating the mitigated consequences.
- 5.13 The assumed response by the operators should be consistent with, for example, the time taken to recognise the fault condition, the radiation levels, the extent of likely contamination and the defined roles and responsibilities of the operators. Where the assumptions appear to be optimistic, particularly for the most limiting high consequence faults, the assessor should seek evidence of the feasibility of the operator response e.g. evacuation times.

Radiological consequences to persons off the site

- 5.14 The radiological consequences to a person off the site may arise from several different exposure pathways e.g. direct shine, inhalation and ingestion from radioactivity released off the site. A person off the site may also be exposed to direct shine if there is a significant release of radioactivity within the facility or if there is a significant increase in the radiation levels in the facility e.g. as a result of a criticality accident. In the case of a release of radioactivity within the facility the estimated direct shine dose to a person off the site would include the following factors:
- The distribution of the radioactivity in the facility
 - The attenuation provided by the building structure and other relevant equipment
 - The distance between the person off the site and facility
 - The period of time the person is exposed to the direct radiation
- 5.15 If there is a release of radioactivity outside the facility the determination of the associated doses from direct shine, inhalation, ingestion and from deposited radioactivity normally requires knowledge of how the radioactivity is dispersed off the site.
- 5.16 For assessment of accidents from UK sites, the dominant exposure pathways for members of the public will usually involve an airborne release of radioactivity, however the possibility of liquid releases should not be excluded. The important parameters for an airborne release can generally be broken down into the following components:
- The radioactive source term
 - Quantity of radioactivity released
 - Isotopic composition
 - Chemical composition
 - Particle size distribution
 - The features of the release
 - Duration of release including any phases such as changes in isotopic composition over time
 - Height of release
 - Energy of release (including heat e.g. buoyancy effects)
 - Building wake effects
 - Atmospheric dispersion
 - Weather conditions
 - Wind speed
 - Dry deposition
 - Wet deposition
 - Dose estimation
 - Location – identify the “representative” person
 - Duration of exposure
 - Consideration of radioactive decay
 - Age of receptor – highest doses are often to children
 - Inhalation rate
 - Dose coefficients
 - Exposure pathways
 - Protective actions, also known as countermeasures

The following paragraphs provide advice to assessors on the approach which is expected to be followed by duty holders in their radiological analysis.

The source term

- 5.17 As consequences generally scale directly with source term (or radioactive inventory released), assumptions made with regard to the source term are a key part of the radiological analysis. It is also a means through which the duty holder can take credit for engineering design that limits the spread of radioactivity.
- 5.18 Given the large number of events and competing chemical and physical processes that will be taking place in an accident situation, plus the unpredictability of operator response, the assessor should be aware that this is also an area of the analysis that may contain some large uncertainties. The assessor is cautioned not to place excessive confidence on the detailed, mechanistic modelling of radionuclide release phenomena.
- 5.19 The source term should be characterised by the amount of radioactivity released, isotopic composition, chemical composition and particle size distribution. (Note: the assessment of the source term may involve ONR assessors in different technical disciplines e.g. fault analysis and chemistry.)
- 5.20 In addition, for reactor faults the assessment should look at assumptions for the fractions of material released from the fuel and coolant and the subsequent attenuation of this material as it passes through engineered barriers to reach the locations where workers are exposed or to the environment. The source term should use an appropriate inventory code for determining final inventory for the fuel as a function of the design and operating history at the time of the accident. The effect of decay of the inventory prior to vessel and containment failure should be considered, plus factors that add to and remove volatile radionuclides from the containment atmosphere.

Features of aerial releases

- 5.21 The assessor needs to be satisfied that the duration assumed for the release is appropriate to the fault being considered. While release of noble gases is considered to be quick and will take place more or less instantly, some of the less mobile radionuclides may continue to be released tens and hundreds of hours after the accident: in these cases, early termination of the release duration may result in an underestimate of the total source term. The estimate of releases may be represented by phases to take account of the potential changes in isotopic composition over time.
- 5.22 After a period of several hours or a day, it is reasonable to assume that sheltering will take place. Where credit is taken for sheltering and other protective actions, also known as countermeasures, justification should be provided: the site emergency plan is a key document in this regard.
- 5.23 The assumed height of the release can have an impact on the off-site doses. A conservative approach is to assume that the release is at ground level if the person receiving the dose is located at the site boundary. Where releases occur at height, depending on the degree of mixing, the main body of the plume may take some time to reach the ground, and the location corresponding to the highest off-site dose may not be the site boundary. Justification should be given for the height of releases and care exercised where the safety case employs fixed positions for receptors.

Atmospheric dispersion

- 5.24 The meteorological data should be appropriate to the site of interest and have been collected close to the site over a number of years. High quality weather data obtained from Numerical Weather Prediction (NWP) datasets is also suitable, and may provide better data than a poorly maintained or located local weather station. The data would

typically include wind direction, wind speed, weather conditions, rainfall and mixing layer depth.

- 5.25 Where computer codes are used to model the atmospheric dispersion of the radioactive release and to determine the dose to an individual, the assessor needs to be satisfied that the computer models and codes have been validated and that the methods and assumptions for determining the dispersion of the released radioactivity are relevant and justified for the particular application.
- 5.26 The assessor should note that the UK Atmospheric Dispersion Modelling Liaison Committee (ADMLC) is a group chaired by the Meteorological Office and includes representatives from the nuclear industry, ONR, PHE and HSE. In addition to providing a forum for identifying and procuring research, one of its objectives is to provide guidance to, and to endorse good practice, in the dispersion modelling community, particularly in the nuclear industry. It was formed about 40 years ago and during that period has reviewed a wide range of topics associated with atmospheric dispersion.
- 5.27 The assessor should be aware that for many years to 2013 the National Dose Assessment Working Group (NDAWG) was also important in the development and review of methods for assessing the doses to public from current and future authorised discharges, and has a wider focus than the ADMLC. NDAWG reports (Ref 20) are currently available on a range of topics.
- 5.28 The most widely used dispersion model in general use is the so-called R-91 methodology (Ref 21) recognised by the UK ADMLC (Ref 22), and supported by ONR for short and medium range modelling. This assumes that the plume disperses in a Gaussian manner both vertically and horizontally. As with most dispersion models, there are limitations on the range at which the model is used, since it uses weather data from the release point for the whole release path. R-91 is a “medium” range model, considered reliable between 100 metres to several tens of kilometres from the release point in constant conditions.
- 5.29 R-91 is fit for purpose for analysing radiological consequences at the site boundary or close to nuclear sites. For very short range work where building wake is important, or long range range modelling, where large scale weather systems are important, other dispersion models will be better (Ref 23, 24).
- 5.30 Considerable progress has been made in predictive modelling capability using Advanced Nuclear Codes. Some of these employ “Monte Carlo” techniques, where a large number of alternative “histories” are run. More recently finite element methods that employ “adaptive meshes” have demonstrated the capability to model real time variations in air concentrations associated with turbulent flows. These methods are currently employed in codes that have until recently been considered to be research tools, but are expected to migrate to general usage in time.
- 5.31 The assessor needs to be aware of the limitations of the R-91 model and also of the refinements to the methodology highlighted in reports of the UK ADMLC. For example, the assessor is pointed to a review of the limitations of the R-91 model made in a submission to the ADMLC (Ref 25), which concluded that there is an uncertainty in short-time averaged air concentrations of up to a factor of 10 in the model. For long-term time averaged air concentrations close to the site, an uncertainty of up to a factor of 2 is quoted. As well as the model range, a limitation arises from the treatment of deposition within R91, which overestimates the removal rate of material from the plume once the plume comes in contact with the ground (Ref 26, 27).

5.32 In addition, weather data used for the whole plume path is fixed to that at the source of the release, which will likely become inaccurate once the plume travels more than several tens of kilometres, or inland from a coastal location or vice-versa (Ref 28).

5.33 The assessor should respect the limitations and inherent uncertainties associated with any models used for radiological analyses. This applies also to the results from computer codes which are mostly only indicative of what might be expected from a given accident, hence the need for proper consideration, by the duty holder, of sensitivity and uncertainty.

Dose estimation

5.34 The approach taken to assess the dose to a person off the site should be flexible, and reflect the type of analysis that is being performed and its regulatory purpose. The location and habit data should be chosen to be conservative for an analysis in support of a DBA, and best estimate for input to a PSA or a SAA. Similarly, in estimating the societal risk (Target 9) and in optioneering for ALARP, a best estimate approach should be used.

5.35 The most appropriate person should be identified as the receptor for the analysis. ICRP has provided advice on how such a person should be identified (Ref 29). Usually this is an individual whose physical attributes (e.g. age) and habits give rise to a relatively high exposure (95th percentile) among the local population whilst remaining credible. As the exposure is prospective over the lifetime of a facility, this individual may be hypothetical in nature, and not an actual person living near the facility. Age-specific dosimetry data has been provided by ICRP for foetus, newborn, adult and intermediate ages (Ref 30). In addition, habit surveys are carried for UK licensed nuclear sites, such as that for the Hartlepool site (Ref 31).

5.36 The dose estimates for an individual should also take account of how long the individual is exposed, the rates of inhalation, the dose coefficients, the exposure pathways and the implementation of any justified protective actions.

5.37 The up to date International Commission on Radiological Protection (ICRP) recommended values should be used e.g. for dose coefficients. Where this is not the case the assessor should seek a justification for the use of alternative values, especially if their use results in optimistic dose predictions.

5.38 Similarly, where doses to persons off the site are estimated using models other than those that reflect good practice i.e. those recognised by the UK ADMLC group, the assessor should seek justification and evidence that the models do not significantly under-estimate the radiological consequences when compared to the predictions of the UK ADMLC models.

5.39 An important element of the assessment is to establish whether the analysis has identified the principal exposure pathways for the representative person, and that the correct person has been identified. In order to do this, the analysis should demonstrate that all potentially significant exposure pathways have been considered for candidate representative persons. On the use of habit data, the assessor should note that generic habit data may differ significantly from local site-specific data in terms of the exposure pathways and the types of persons exposed and their habits, and that PHE and its predecessor organisations has produced guidance on the use of habit data for radiological analysis (Ref 32, 33, 34)

5.40 Protective actions may be taken into account subject to the likelihood that they will be implemented. However, protective actions should not be treated as a means of reducing doses to meet relevant target levels.

Risks to individuals

- 5.41 Radiation doses (Sv) can be converted to risk of death from an accident by applying a risk conversion factor and the frequency of the fault. At relatively low doses the most appropriate fatal risk conversion factor is the ICRP recommendation (Ref 6) for late health effects, 5.5% per Sv for the whole population. Since the factors are subject to review from time to time, assessors should be aware of the most up to date authoritative guidance.
- 5.42 For high doses received in a relatively short period of time the most appropriate factors increase, as they will also include deaths resulting from acute radiation effects, which manifest themselves following exposures of above a threshold at around 0.1 Gray (Gy) whole body. At very high dose levels, above several Gray whole body dose, the risks are usually fatal, in which case the risk of death is the frequency of the associated fault.
- 5.43 Where the dose/risk conversion factors used are lower than those recommended by authoritative bodies e.g. ICRP and PHE, the assessor should examine the underlying assumptions and seek an adequate justification.

Treatment of Uncertainties

- 5.44 There are uncertainties associated with each step of the methodology for predicting the risk of death from accidents at nuclear licensed sites. The overall level of uncertainty may be considerable, and may be a relevant regulatory consideration if estimated doses are close to targets. Parameter based studies that have been performed to estimate the uncertainties will generally underestimate the total uncertainty, since they will not capture uncertainties associated with a lack of knowledge: these may be aleatory in nature, associated with evolution of an accident scenario, or misrepresentation of physical/chemical phenomena within simplified models.
- 5.45 Estimates are made in Table 1 below of the typical degree of uncertainty in each step of an analysis that predicts off-site risk from a single radionuclide species from two different types of accident. Eleven steps have been identified for the overall analysis in the table, labelled a) to k). The rationale for assigning uncertainties for these steps is given below. Assessors should consider using these estimates as a basis to challenge duty holders to use significantly different ranges, or to consider the impact of uncertainty. Assessors should also look for uncertainty to be considered in an holistic rather than piecemeal fashion.
- 5.46 *Step a) Radioactive inventory estimation*

The uncertainty associated with estimating the radioactive inventory of the source will depend on the type and history of the facility. For poorly documented legacy wastes, there may be considerable uncertainties, and much will depend on the quality of waste characterisation efforts. The two cases taken here, however, are associated with accidents affecting reactor fuel, which will generally be well documented.

The radionuclide inventory of a reactor core and common activated structural components can be estimated with a high degree of confidence, since processes governing transmutation of radionuclides are well understood and nuclear data for common reactions or decay pathways contains small uncertainties (e.g.10%) (Ref 35). The codes used to estimate radionuclide inventories in the UK, such as ORIGEN (Ref 36), FISPIN (Ref 37) and FISPACT (Ref 38), have been used over many years and have been extensively benchmarked. It is reasonable to estimate uncertainties in predicted inventory for key radionuclides to be less than 20% if the isotopic

compositions of the fuel components are well specified, the power history is known and neutron energy spectra have been well characterised.

5.47 However, there are factors which may increase the uncertainty in the inventory within a specific sample of fuel or other core components. Fuel burn-up will vary across the core, as neutron fluxes vary over the radial profile, and burn-up in a specific sample will be subject to the management of the fuel. Radiologically significant radionuclides may originate from impurities at the ppm level in fuel components, which have not been controlled.

5.48 *Step b) Source term*

Definition of source terms is one of the largest sources of uncertainty for radiological consequences analyses: for example for the Large Break Loss of Coolant Accident (LOCA) causing the iodine-131 release, this includes estimating damage to the fuel matrix by temperature and pressure transients, migration of radionuclides to the coolant, subsequent interactions with plant, transfers between different phases and chemical forms, containments failures and human management of the accident prior to escape of radioactivity to the environment.

5.49 The evolution of an accident is unpredictable and not conducive to being modelled by conventional means. In addition, there have been only a small number of serious accidents that have resulted in release of radionuclides to the environment, and these cases have featured different plant and accident scenarios, so there is limited evidence for determining source terms for particular plant on a purely empirical basis.

5.50 The common approach adopted by duty holders is to develop mechanistic models for each of the processes that determine the release of radionuclides and combine them in a modular code. There are significant uncertainties with modelling many of these processes, as well as the management of the plant in accident conditions.

5.51 As an indication of the size of uncertainties, US NRC discuss uncertainties in in-vessel release from fuel in NUREG 1645 (p16, Section 4.4, Ref 39), and concluded that there are ranges of 4 to 6 orders of magnitude in the fractions for release into containment for the non-volatile radionuclides discussed there, with factors of 5 to 20 between the mean and 75th percentile values, hence the relatively large uncertainty associated with the release fraction for plutonium. Note that there are further uncertainties associated with release from the containment to the atmosphere.

5.52 The uncertainty range for iodine is tighter, since the release fraction from the fuel matrix will be close to unity, however there are uncertainties associated with complex physico-chemical interactions in the transfer of the iodine between physical phases and chemical forms, and transfer from the coolant to the atmosphere via containment.

5.53 *Step c) Dispersion modelling*

Traditional, analytical modelling of short and medium range atmospheric dispersion is good for a release that resembles an idealised release in stable weather conditions. However, in practice, releases can depart from the ideal in several ways, which may introduce significant uncertainties into models. The estimated uncertainty associated with the commonly used R91 atmospheric dispersion model is discussed in paragraph 5.31, and has been used as the basis for the numbers in Table 1. In the example of the plutonium fire, the assumption of neutral buoyancy in R91 will be challenged, and contributes to the higher uncertainty associated with the dispersion model relative to the iodine example.

In addition to the uncertainty associated with the predicted concentration in air, there is the uncertainty associated with deposition of the radionuclide, which is sensitive to the

physical characteristics and chemical form of the radionuclide. In the case of iodine, the fraction in organic form will have a lower deposition velocity than inorganic and particulate forms. For the plutonium case, uncertainty in the range of particle sizes associated with a fire will introduce uncertainty in the deposition rate.

5.54 *Steps d–k) Food chain and dosimetry models*

A quantitative estimation of uncertainties related to food chain modelling and the use of ICRP dosimetry models (d - k) was made in the PC COSYMA uncertainty study (Ref 40). This study tested a novel, mathematical algorithm for combining expert elicitations, nevertheless gives a good insight into the uncertainties associated with an accident consequence calculation with the PC COSYMA code. Further, a discussion of uncertainty in the use of ICRP dosimetry models is provided by a number of studies published in the scientific literature, for example the study on the Leggett recycling biokinetic model for systemic plutonium (Ref 41). Key considerations for these models are accumulation of the radionuclides in particular tissues and the quality of the radiation. Thus there are low uncertainties for an intake of iodine since nearly all the iodine will accumulate in the thyroid gland and the principal radiation emissions are penetrating in quality.

- 5.55 Plutonium, on the other hand, accumulates across a range of radiosensitive tissues, including lung, liver, bone marrow and gonads, and the principal radiation emissions are short range alpha particles. Furthermore, experimental data for the distribution of plutonium is limited, and is based upon a small number of studies based on plutonium workers and terminally ill patients. Consequently, the uncertainties associated with the dosimetry model are much higher than for iodine.
- 5.56 The conversion factors (k) for translating dose received from ionising radiation to risk of detriment have typically been derived from studies of populations that have been exposed to very high dose rates and subsequently suffered from a high rate of mortality. There is uncertainty associated with applying these risk factors to the low level exposures that would result in most cases from the nuclear accidents that are analysed in safety cases.
- 5.57 It is important to state that these numbers are illustrative, and are given here because the sizes of these uncertainties are sometimes not appreciated. At the same time, it should also be stated that there are strategies for dealing with uncertainty in safety cases. The assessor should look for due account being taken for uncertainties in analysis. A key message to take away from these numbers is that safety cases should not place too much emphasis on absolute estimates of risk that are not supported by empirical evidence.

Table 1 Examples of typical uncertainties¹ associated with steps within an analysis of off-site radiological consequences from two single radionuclide releases

ANALYSIS STEP	Uncertainty¹ in best estimate of risk from I-131 released from a Large Break LOCA in a PWR	Uncertainty¹ in best estimate of risk from Pu-239 released from a fire at a fuel reprocessing facility
a) Radioactive inventory estimation	<1.2	<1.2
b) Source term modelling	5 - 50	10 - 1000
c) Dispersion model	10	20
d) Food transfer model	2-20	2-20
e) External dosimetry model	<2	<2
f) Inhalation model	<1.2	5
g) Ingestion model	<1.1	10
h) Biokinetic models	<1.1	2-5
i) Tissue doses in reference phantom	<1.2	<1.1
j) Individual variability in physiological factors	1.2	1.5
k) Risk conversion factor (Sv to risk of death PA)	3	2

Validation of software

5.58 A variety of software tools are used for radiological consequences analysis. The duty holder should demonstrate that the analysis has been performed with tools that are assured to industry standards and that the scope of analysis falls within the range of applications for which the tools have been validated. Note should be taken of the SAPs that apply to the assurance of validity of data and models [pp146-149, AV.1-8] and the

¹ Uncertainty defined here as the ratio between the mean and the 5% and 95% percentile in a lognormal distribution.

associated TAG (reference NS-TAST-GD-42, Validation of Computer Codes and Computational Methods).

- 5.59 For important pieces of analysis, it is reasonable to expect the duty holder to provide dedicated documentation (AV.5) that validates the use of the analysis tool for the range of calculations that have been undertaken. Documentation provided for a particular application of the code may be called a “validation statement”. The provision of validation statements should be proportionate to the importance of the analysis in the safety case.

Use of SAPs Numerical targets 4-9 for radiological consequences

- 5.60 The numerical risk targets contained in the SAPs quantify ONR’s risk policy, and assist proportionate decision making and targeting resources. The targets that apply to fault conditions are not legal quantities: they are a snapshot of the acceptability of risk from licensed nuclear sites given the ALARP principal and public appetite for risk, benefits from activities on nuclear sites and the relative risks from other activities (Ref 2). Specific guidance is given below on the interpretation of each numerical target.

Design Basis Analysis (Target 4)

- 5.61 Design Basis Analysis aims to demonstrate that the plant is intrinsically safe against against foreseeable faults. It aims to do this through a simple, deterministic analysis of radiological consequences. Where there are uncertainties in the analysis, a conservative or “bounding” approach is taken to demonstrate that plant performance meets the radiological targets.
- 5.62 The effort expended in the consequences analysis should be proportionate to risk: in the UK this is commonly achieved by carrying out two levels of analysis. It is appropriate to conduct a simple, initial calculation to determine whether radiological consequences are large enough to benefit from ALARP considerations. While the BSL for SAPs Target 4 is quoted in the SAPs (Para 628) as a suitable threshold for inclusion of a fault in the Fault and Protection schedule, this is considered too limited an ambition for modern facilities which should aim at the BSO. This simple calculation is therefore essentially a screening calculation to determine whether the fault should be included in the Fault and Protection schedule, and is conducted with “unmitigated” consequences, without taking credit for protective systems and actions.
- 5.63 If the unmitigated calculation shows that the consequences are not significant, then the consequences of the fault are in the “broadly acceptable” region and further analysis of consequences is inappropriate. However, the decision to screen out the fault from further analysis should be appropriately documented. Should the result of the simple, unmitigated analysis challenge SAPs Target 4, the fault should be included in a fault schedule as a design basis fault. The fault should then be reanalysed taking into account the safety mitigation provided by protection systems. The analysis should include sufficient conservatism to provide confidence that the requirements of Target 4 are demonstrably met.
- 5.64 A different set of targets is applied on and off site, reflecting that workers derive benefit directly from the site activities. The SAPs for design basis analysis are given in paragraphs 626 – 643 and 726-729 including Target 4. FA.7 and paragraphs 635 - 639 are particularly relevant for the radiological analysis.
- 5.65 The analysis should demonstrate that for most DBAs, sufficient physical barriers will be maintained to reduce so far as is reasonably practicable, the doses to persons on and off the site. In the most severe design basis accident, no person on the site should receive a dose in excess of 500 mSv and no person off the site should receive a dose greater than 100 mSv. Generally, the larger the potential consequences of an

accident, the smaller should be its frequency. This is illustrated in Target 4 which identifies the relevant BSL and BSO values for ranges of fault frequencies and radiological consequences.

- 5.66 Principle FA.7 states that the ‘analysis of design basis fault sequences should use appropriate tools and techniques, and be performed on a conservative basis to demonstrate that consequences are ALARP’. The assessor should ensure that the radiological analysis predicts the maximum effective dose to a person on the site and to a person off the site directly downwind of the release. It should be assumed for off site releases that ‘the person remains at the point of greatest dose or the maximum duration, although for extended faults a more realistic occupancy may be assumed after a suitable interval.’ (SAP para 729(a)).

On site radiological consequences (Target 4)

- 5.67 The person most at risk on the site is likely to be an operator in the facility. The assessor should ensure that the assumptions made for the operator’s response to each fault take account of the operational conditions likely to prevail in the event of the fault and are conservative but not overly pessimistic.

Off-site methodology (Target 4)

- 5.68 The radiological analysis should be conservative and assume
- The person remains at the point of greatest dose for the duration of the release, although for extended faults a more realistic occupancy may be assumed after a suitable interval.
 - The weather category and conditions have characteristics which produce the highest dose to that person.
 - No off-site emergency protective actions are implemented, other than those whose implementation can be shown to be very highly likely to avert a dose via a particular pathway e.g. food bans.
- 5.69 The maximum effective dose to the range of possible receptors should be considered and the most limiting value used for assessment against Target 4. This range should account for age and sex in the exposed population. Where several exposure pathways exist, evidence should be presented that the most exposed person has been used for the analysis.
- Since rainfall can enhance deposition and uptake of radionuclides via the foodchain, as well as increasing external gamma dose, the likelihood and effect of rainfall during a release should be considered. If excluded, arguments should be presented to show that it is reasonable to do so.
- 5.70 Where a methodology other than that recognised by the UK ADMLC is used, the assessor should seek to establish that the consequences are not unduly lower than those predicted by the UK ADMLC methods.

Worker and Off-Site Analysis (Targets 5 – 8)

- 5.71 Care should be taken that the use of radiological consequence targets should not bias PSA towards conservative assumptions. The PSA predictions of the doses to persons on the site and persons off the site should be based on a best estimate approach. If there are large uncertainties associated with key parameters, and conservative values are adopted, then the impact of these assumptions on the usefulness of the PSA as a tool for optimising design should be considered.

- 5.72 In these circumstances, the first step should be to identify uncertainties and conservatisms and understand their impact on applications of the PSA. Where possible, the conservatisms and uncertainties in the PSA should be reduced or removed.
- 5.73 It should be noted that Targets 5 and 7 differ from Targets 6 and 8 in a fundamental way: 5 and 7 are targets on risk of fatality to an individual from all accidents, while 6 and 8 are targets on the frequency of accidents within a dose band. 5 and 7 apply to all accidents on all facilities on the site, while 6 applies to a single accident and 8 to a single facility.
- 5.74 Targets 5 and 7 therefore require a more comprehensive approach than Targets 6 and 8, and are likely to be invoked where a completely new plant is being built. Furthermore, as Targets 6 and 8 are more likely to be used to justify limited modifications to plant, it is argued that a more conservative approach is appropriate than for Targets 5 and 7. Thus the SAPs (para 751) specify that the receptor is assumed to be downwind of the release for Target 8, whereas a less conservative, probabilistic approach is suitable for Target 7.

On site radiological consequences (Targets 5 and 6)

- 5.75 The assessor should ensure that the assumptions made for the operator's response to the fault takes account of the operational conditions likely to prevail in the event of the fault and are best estimates. The operator should be assumed to be situated at the point of maximum potential exposure and realistic occupancy factors should be used to determine the exposure time. Evacuation pathways should be reviewed against emergency response plans for the site and have realistic expectations for operator actions in real accident situations.
- 5.76 The contribution to the operator risk from recovery actions leading to a safe plant state should be included in the analysis. This contribution should cease once a safe state and control is achieved following the accident.

Consideration should be given to risk from twin reactor units. Justification should be provided where onsite exposures do not scale directly with the number of units. This may arise, for example, where exposures are local in nature and manning arrangements may preclude the use of the same workers for multiple units.

Off-site radiological consequences (Targets 7 and 8)

- 5.77 The determination of risk to individuals is based on estimations of the radiological consequence and frequency for each of the identified fault sequence groups. The radiological analysis will clarify the dose band to which each fault sequence should be allocated. The frequency associated with the fault sequence should be below the respective BSL and be ALARP.
- 5.78 The assessor should ensure that the off-site radiological consequences for identified fault sequences have been determined in an appropriate manner. Co-operation is likely to be required between the fault studies specialists and the radiological protection specialists since the former assess the validity of the frequencies.
- 5.79 Best-estimate methods and data should preferably be used for the radiological analysis in the PSA and should
- assume a hypothetical person is located at the at the point of greatest dose considering the habits of the local population and the physical constraints of the site,

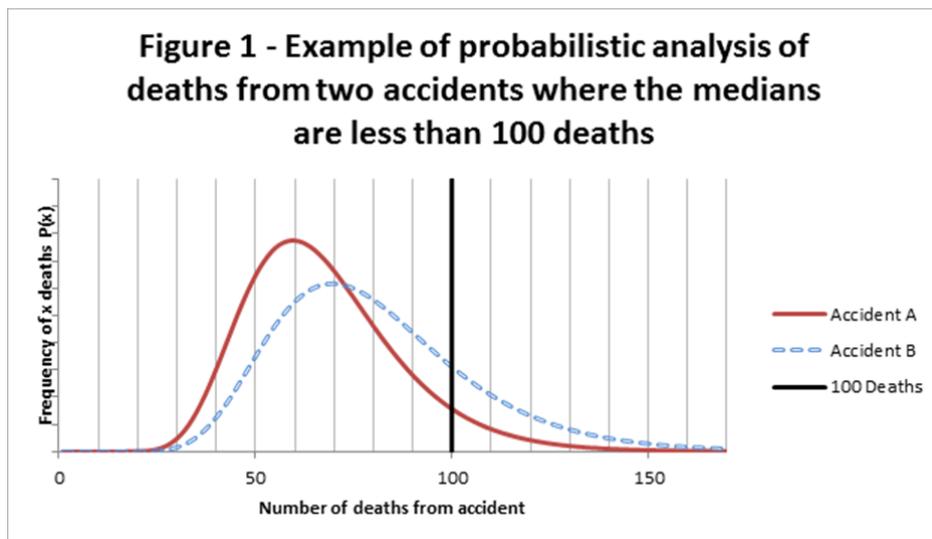
- assume for Target 8 that the hypothetical person is directly downwind of the release for the duration of the release. For extended faults realistic occupancy factors may be assumed after a time interval based on the assumed identity and location of the hypothetical person and an interpretation of the site emergency plan for what a reasonable response time might be,
- take account of the likelihood of the different weather conditions and wind directions for Target 7, including wind rose data, that are most relevant for the site,
- take account of only those protective actions that are highly likely to be implemented. If account is taken of other protective actions, the probability of these actions being implemented should be considered.

- 5.80 Where the duty holder's approach differs significantly from the above, the assessor should seek justification to ensure that the dose predictions are not unduly optimistic.
- 5.81 The PSA will provide estimates of the radiological consequence and frequency for each of the fault sequences that have been identified. The radiological analysis will determine to which of the Target 8 dose bands the fault sequence is allocated. The frequency associated with the fault sequence should be below the respective BSL and must be ALARP.
- 5.82 The assessor will need to form a judgement on the depth and breadth of the assessment of the duty holder's case in order to be satisfied that consequences have been derived in an appropriate way. In some cases, the assessor may wish to perform independent calculations. Ideally there should be enough information in the documentation provided by the duty holder to reproduce the calculations in detail: if not, the assessor may consider making a supplementary request for further information.

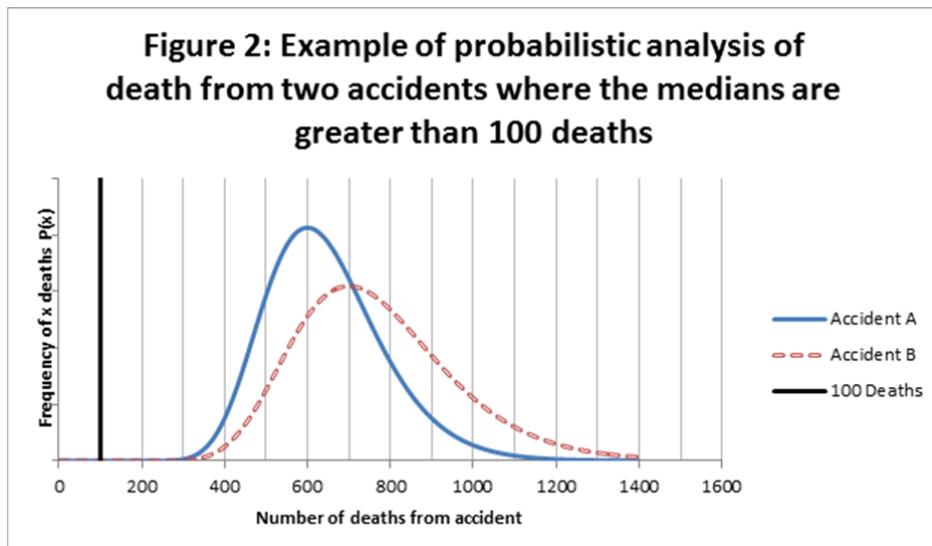
Target for Societal Risk (SAPs Target 9)

- 5.83 It is considered prudent to have a Target that considers the frequency and collective consequences of severe accidents for a population if there is the possibility of such accidents occurring at a particular site, recognising that consequences analysis in DBA and PSA are focused on consequences to individuals, which may prevent them identifying events that have a wide impact. In addition, it is helpful for the response to an accident at a site to know the potential severity of such an event, even if it is very rare.
- 5.84 Target 9 provides a BSL and a BSO for the total risk of 100 or more fatalities from accidents at the site: in effect the target can be interpreted as a means to assess the quality of the design in terms of its ability to limit the likelihood of large releases (i.e. those capable of causing a 100 or more fatalities).
- 5.85 When applying the 100 death threshold the assessor should adopt a liberal attitude to the use of a threshold based on number of deaths, and ensure that the analysis does not overlook events that may narrowly miss the threshold e.g. one in which there are 99 deaths. The severity of a large release is also a consideration: clearly an accident that causes 1000 deaths is less acceptable than one that 100 deaths if both occur with the same frequency.
- 5.86 Given that weather data would typically be sampled for a Target 9 analysis, the output from such a calculation will be a probability density function relating the number of deaths to their probability of occurring. In this case only part of the distribution for a particular release may result in over 100 deaths and a question arises over whether the whole contributes to the Target 9 frequency.

- 5.87 One possible solution is to define a threshold level for counting the contribution towards Target 9: for example, if 50% or more of the distribution results in greater than 100 deaths then it is assumed that the whole frequency of the release contributes to Target 9.
- 5.88 A more precise alternative approach is to count only the part of the distribution that results in over 100 deaths. However, it should be noted that this approach can be unreliable if the accident profile is dominated by releases that result in less than 100 deaths, as only the tails of the distributions are above 100 deaths, as illustrated in Figure 1 below. In the example given there are two accidents, one with a peak value at 60 deaths, another at 70 deaths. Neither of these two accidents would be counted towards Target 9 if a 50% threshold were adopted. One is only marginally worse than the other measured by the total number of deaths, however if Target 9 is applied to the “tails” of each curve that cross the 100 death limit, there is a difference of more than an order of magnitude in the frequency of 100 deaths being exceeded.



- 5.89 In contrast Figure 2 show a similar analysis performed for two accidents that have median values at 600 and 700 deaths respectively. Again one might be considered marginally worse than the other in terms of the total number of deaths, however in this case the ratio of the frequency of 100 deaths being exceeded is close to 1.



- 5.90 These examples illustrate that Target 9 should be interpreted with caution when dealing with probabilistic calculations for which a substantial part of the probability density function for a relatively frequent accident is just below the 100 deaths threshold. A problem may not arise if there are more frequent accidents with peak deaths well above 100 in the analysis, since these will dominate the contribution to frequency of >100 deaths. In addition, the assessor may wish to also consider the peak and total number of deaths associated with accidents at the plant, which will reveal the presence of anomalous “tail” effects.
- 5.91 The radiological analysis should generally be carried out using best estimate assumptions, methods and data. The assessor should establish where and when deaths are predicted to occur and assess whether the computational tools are appropriate to predict these with confidence, noting that R91 type dispersion models have been developed for short and medium ranges (from 100m to a few tens of kilometres). The distinction should also be made between early (deterministic) and late (stochastic) deaths, as different analytical methods are employed.
- 5.92 Advice should be taken from SAPs and other RGP, notably from PHE and ICRP, on cut-off levels for low individual doses and contributions from future generations. The SAPs currently state that contributions should be restricted to the whole UK population and integrated over a 100 year period, however advice on cut-off levels and integration time is evolving. The likelihood of protective actions being applied should be considered for long term exposures. Assumptions made with respect to the distribution and consumption of contaminated food should be examined, as this may be the dominant source of long-term exposure.

Summary of guidance on assumptions made with respect to protective actions (countermeasures)

- 5.93 The reader will have noted that assumptions made with respect to whether credit is taken for protective actions, also known as countermeasures, depends on the type of the analysis that is undertaken. For clarity, the guidance given above is summarised in the 5 points below.
1. In all cases it is agreed that protective actions should only be claimed when these are justified by the duty holder or Generic Design Assessment requesting party in their safety case.
 2. For DBA, consideration should only be given to protective actions that are highly likely to be implemented, and are justified as such.
 3. Probabilistic analysis can deal with protective actions explicitly by modelling the likelihood of protective actions failing or succeeding.
 4. For exposures to individuals off site, namely in relation to targets 7 and 8, only long term protective actions can be considered, such as food bans and measures to decontaminate land and buildings.
 5. For exposures to society as a whole, namely in relation to target 9, both short term and long term protective actions can be considered. These need to be justified, as in all cases.

Radiological Consequences in ALARP Studies

- 5.94 Fault analysis performed against the radiological targets may indicate that there is scope for reducing exposures on ALARP grounds. This may drive ALARP studies that

employ radiological consequences analyses within quantitative ALARP arguments. Sometimes the cost (monetary or otherwise) of different options is evaluated against benefits (dose reduction, reduced economic or psychological impact). In such studies the emphasis should be on demonstrating the likely benefits of options, which implies that there should not be unrealistic conservatism or optimism in the calculations. A common error is to use conservative radiological consequence analysis to support optioneering: the result is that these calculations will only evaluate a small subset of the range of possible outcomes and consequently provide weak support for decision making. Cost-benefit analyses such as these also need to be part of an overall case that considers design and operational relevant good practice rather than be seen in isolation. NS –TAST-GD 005 (Ref 16) gives more guidance on the role and use of CBA.

- 5.95 The assessors should ensure that the duty holder aims to reduce the predicted doses to ALARP levels by considering the hierarchy of control measures, which is highlighted in Reg. 9(2) of the Ionising Radiations Regulations 2017 (Ref 42) and in paragraph 155 of the SAPs, together with recognised good practices rather than by the inappropriate refinement of the parameters used in the radiological analyses.

Radiological Consequences Analysis Supporting New Build

- 5.96 In recent times ONR has assessed the generic designs of several reactor designs within nuclear new build programmes prior to formal license application. There is ongoing activity in this area, as several GB sites are being developed for new nuclear plants. While the assessment principles are no different to those that apply to operating nuclear plant, there are particular challenges associated with performing assessments for new generation plant that has not been built in GB.
- 5.97 A feature of the Generic Design Assessments done to date has been that the plant designs have been developed within regulatory regimes that can differ significantly from that which applies to GB licensed nuclear sites: features that are commonplace in a UK nuclear safety case may be absent or treated differently. These safety cases contain analysis that has been developed with different regulatory targets in mind, and may employ plant assumptions that will not eventually apply to the specific plant that is built in GB. The principle of ALARP, which is the cornerstone of the GB regulatory approach, may not be part of the regulatory regimes in which the reactor was designed. Advice on assessing the ALARP aspects of New Build is given in Annexe 2 of NS –TAST-GD 005 (Ref 16).
- 5.98 The assessor should take a non-prescriptive approach to the assessment, accepting that ownership of the safety case rests with the requesting party/potential licensee. In the first instance, the assessor should assess against SAPs and the SAPs numerical targets. Where the requesting party/potential licensee has used an approach that departs significantly, and this is difficult or impossible to do, an equivalent set of standards should be sought.
- 5.99 Care should be taken to examine assumptions relating to the identity, habits and locations of “representative” receptors, and the transfer of radionuclides to food, since these factors can vary significantly between countries. For example, exposure via consumption of fresh milk may be a key exposure pathway in the United Kingdom, but considered insignificant in countries such as Japan or France. Such assumptions may be embedded within modelling tools, and may not be immediately obvious without conducting a full audit of safety case evidence.
- 5.100 If the safety case passes Generic Design Assessment and moves to a license application, it will be appropriate to include site-specific factors in the radiological consequences analysis. This should include the use of site-specific weather, habit and population data.

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7. GLOSSARY AND ABBREVIATIONS

ADMLC	Atmospheric Dispersion Modelling Liaison Committee
ALARP	As low as reasonably practicable
BSL	Basic Safety Level
BSO	Basic Safety Objective
CBA	Cost Benefit Analysis
DBA	Design Basis Analysis
HSE	Health and Safety Executive
HSW	The Health and Safety at Work etc. Act 1974
IAEA	International Atomic Energy Agency
LOCA	Loss of Coolant Accident
NDAWG	National Dose Assessment Working Group
NRPB	National Radiological Protection Board (now PHE)
ONR	Office for Nuclear Regulation
PHE	Public Health England
PSA	Probabilistic Safety Analysis
PWR	Pressurised Water Reactor
RGP	Recommended Good Practise
SAA	Severe Accident Assessment
SAP	Safety Assessment Principle(s)
TAG	Technical Assessment Guide(s)