FUNDAMENTAL NUCLEAR SAFETY PRINCIPLES

Purpose

It is common practice among UK licensees to produce a set of Nuclear Safety Principles (*NSPs*) for application in the assessment of the safety of their installations. While the specific content varies from one licensee to the next, the general intent is the same: to provide a high-level set of principles for reviewing the adequacy of their nuclear safety cases, taking due consideration of available international and regulatory guidance, such as the IAEA Fundamental Safety Principles and the Office for Nuclear Regulation (*ONR*) Safety Assessment Principles (*SAPs*).

The development and application of NSPs to the production of nuclear safety cases represents established good practice in UK safety case development that should be considered for adoption by any future licensees. To aid future licensees in the development of their own NSPs, the sections that follow present a set of candidate principles that can be adopted or developed to fit each individual licensees' requirements. These requirements have been deliberately developed to be as generic as possible so that they may be applied to all types of nuclear site in the UK, including reactors, process plants, fuel manufacture, decommissioning and military applications.

Key Sources of Information

The nuclear safety principles presented in this section have been developed from a review of the existing available guidance and knowledge of current UK licensees arrangements. The key sources of information include:

- IAEA Fundamental Safety Principles.
- IAEA Requirements and Safety Guides.
- ONR Safety Assessment Principles (SAPs) for Nuclear Facilities.

Fundamental Safety Objective

The fundamental objective of all organisations undertaking activities involving nuclear material must be to ensure that those activities do not adversely impact people or the environment.

It is therefore appropriate to specify the following fundamental safety objective:

- FSO 1 - The public, workers and the environment shall be protected from the harmful effects of ionising radiation associated with the activities undertaken on the nuclear site.

This aligns closely with the IAEA fundamental safety objective stated in its Fundamental Safety Principles document. The more detailed NSPs presented in the following sections are intended to outline the means by which this ultimate objective shall be ensured.

Nuclear Safety Principles

The NSPs presented in this section have been developed from a review of the existing available international, national and industry guidance and knowledge of current UK licensee's arrangements. The principles are split into the following topic areas, based on the structure of the IAEA General Safety Requirements documents:

- Leadership and Management.
- Radiological Protection.
- Safety Assessment.
- Radioactive Waste Management.
- Decommissioning.
- Emergency Preparedness.

Leadership and Management Principles

The following Leadership and Management Principles are derived primarily from Part 2 of the IAEA General Safety Requirements – Leadership and Management for Safety.

LM1	**The senior management shall be ultimately responsible for ensuring that the risks to workers and the public associated with the activities undertaken on the site are understood, adequately controlled and reduced to ALARP.**
Basis	*GSR Part 2 Requirements 1, 9 & 12.* *GSR Part 4 Requirements 3, 5 & 12.*
Guidance	*The senior management are ultimately responsible for ensuring the safety of the workers on the site and protecting the public.* *The senior management are responsible for setting the organisation's safety policy and promoting and reinforcing it throughout the organisation.* *The safety policy should:* • *Prioritise the safety of people and protection of the environment above all other business drivers.* • *Consider all aspects of safety, not just radiological.* • *Emphasise personal responsibility for safety.*

LM1	**The senior management shall be ultimately responsible for ensuring that the risks to workers and the public associated with the activities undertaken on the site are understood, adequately controlled and reduced to ALARP.**
	It is the responsibility of senior management to lead by example and instil a positive safety culture within the organisation (see LM2).
Further Guidance	HSE, Health and Safety at Work Act (HSWA), 1974. HSE, The Management of Health and Safety at Work Regulations, 1999. ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition (see Leadership and Management for Safety Principles MS.1). Revision 1 (January 2020)

LM2	**The organisation shall embody a positive culture for safety, starting with the senior management, to ensure that the fundamental safety objective is achieved.** **The senior management shall be responsible for ensuring that suitable corporate arrangements and policies are in place and SQEP resources are available within the organisation to enable this culture for safety to be maintained.**
Basis	*GSR Part 2 Requirements 2 & 10.*
Guidance	*A positive Culture for Safety is fundamental to maintaining nuclear safety. It helps to prevent the erosion of safety margins, enables the identification and remediation of unsafe occurrences, and promotes continuous improvement and personal accountability. There are numerous sources of guidance on what constitutes a positive safety culture, and the following key principles have been distilled form a review of the specific sources of guidance listed below.* • *Leadership for safety – it is essential the management within the organisation demonstrate their commitment to safety through both what they say and what they do. They are responsible for setting and communicating nuclear safety policy and messages.* • *Personal Accountability – everyone is responsible for ensuring nuclear safety is maintained. This should be led by the management and flow down through the entire organisation.* • *Questioning attitude / challenge culture – the organisation should provide an environment in which individuals fell empowered to challenge and question actions, decisions, errors and behaviours where they feel safety may be adversely impacted.*

LM2	**The organisation shall embody a positive culture for safety, starting with the senior management, to ensure that the fundamental safety objective is achieved.** **The senior management shall be responsible for ensuring that suitable corporate arrangements and policies are in place and SQEP resources are available within the organisation to enable this culture for safety to be maintained.**
	• *Trust / Respect – for a strong safety culture it is essential that this permeates throughout the organisation.* • *Safety informed decision making – decisions that may affect nuclear safety are made in a rigorous and systematic way with all personnel within an organisation encouraged and empowered to make decisions that place the plant in a safe condition when faced with uncertain or potentially unsafe conditions.* • *Learning from experience – capturing operating experience through the identification, reporting and analysis of unexpected or unsafe events, near misses and deviations from normal operations is essential to ensuring lessons are learned, repeat events avoided and improvements to safety made.*
Further Guidance	IAEA, Safety Culture in Nuclear Installations: Guidance for Use in the Enhancement of Safety Culture, IAEA-TECDOC-1329, IAEA: Vienna, 2002. INPO, Principles of a Strong Nuclear Safety Culture, November 2004. WANO Principles, Traits of a Healthy Nuclear Safety Culture, PL 2013-1, May 2013. ONR, Challenge Culture, Independent Challenge Capability (including an Internal Regulation function), and the provision of Nuclear Safety Advice, July 2018. IAEA, Key Practical Issues in Strengthening Safety Culture (INSAG-15), IAEA: Vienna, 2002. IAEA, The Operating Organisation for Nuclear Power Plants (NS-G-2.4), IAEA: Vienna, 2001. IAEA, Recruitment, Qualification and Training of Personnel for Nuclear Power Plants (NS-G-2.8), IAEA: Vienna, 2002. ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition (see Leadership and Management for Safety Principles MS.1 to MS.4). Revision 1 (January 2020)

LM3	**The safety management system provides a framework for implementing the nuclear safety principles to identify, assess, control and mitigate radiological risks ensuring the safety of the activities undertaken on the licensed site at all stages in the lifecycle of the plant and safety case.**
Basis	*GSR Part 2 Requirement 3.*
Guidance	*It is the responsibility of the senior management to ensure an appropriate management system is developed and implemented. Senior managers must ensure that the all members of the organisation are provided with the necessary processes, procedures and tools to safely undertake their role.* *Senior management must demonstrate their commitment safety actively endorsing and promoting the management system.* *The maturity of management system arrangements shall be commensurate with the stage in the plant lifecycle. Early in the lifecycle the management arrangements will focus on the demonstration of organisational capability. As the plant design and safety case develops suitable arrangements must be put in place to manage the design and supporting analysis and facilitate the continued demonstration that the other nuclear safety principles are satisfied.*
Further Guidance	IAEA, Safety Standards, Application of the Management System for Facilities and Activities, Safety Guide No. GS-G-3.1, IAEA, Vienna, 2006.

LM4	**The safety management system shall employ a graded approach to managing nuclear safety to ensure that the level of rigour, effort and resource applied to the assessment, management and control of activities is proportionate to the risk.**
Basis	*GSR Part 2 Requirement 7.*
Guidance	*It is recognised that licensees / requesting parties do not have unlimited resources and funding at their disposal with which to manage the nuclear safety of their installations and while commercial concerns shall never be prioritised to the detriment of nuclear safety it is appropriate that a graded approach is adopted to ensure that the available resources are focussed on the areas of greatest potential risk / benefit.* *The proportionate application of the safety management system shall take into account the risk associated with the plant, process, activity or modification as well as other factors including:* • *Novelty*

LM4	**The safety management system shall employ a graded approach to managing nuclear safety to ensure that the level of rigour, effort and resource applied to the assessment, management and control of activities is proportionate to the risk.**
	• *Complexity* • *Experience of previous similar activities (e.g. like for like replacement of consumable items).* • *Risk if ill-conceived or inadequately implemented.* • *Importance of the product or activity to safety, health, environmental, security, quality and economic expectations.* *Application of a graded approach will enable plant and processes of greater and lesser significance to be identified informing the level of controls and checks to be applied including but not limited to quality arrangements, qualification, training, level of verification, inspection, and testing, materials, and records.*
Further Guidance	IAEA, Safety Standards, Application of the Management System for Facilities and Activities, Safety Guide No. GS-G-3.1, IAEA, Vienna, 2006.

LM5	**A Review, Learn and Improve approach shall be implemented to monitor, measure and assess the effectiveness of the safety management system, safety case processes and safety culture. This shall include the continuous monitoring and review of national and international operating experience (OPEX).**
Basis	*GSR Part 2 Requirements 13 & 14.*
Guidance	*Learning from experience is essential to driving improvements in safety and is a fundamental aspect of a robust safety management system.* *The licensee / requesting party shall implement suitable processes for monitoring the performance of plant, systems and procedures to identify and capture operating experience. An essential part of this is establishing a culture and environment in which capturing operating experience is encouraged (see LM2 and RP4).* *It is important that learning is gathered both directly from the performance of the organisation and its own processes and also from external organisations, both nuclear and non-nuclear. This should include:* • *Internal – Monitoring of the effectiveness of the safety management system procedures and arrangements to identify where failures or near misses have occurred*

LM5	**A Review, Learn and Improve approach shall be implemented to monitor, measure and assess the effectiveness of the safety management system, safety case processes and safety culture. This shall include the continuous monitoring and review of national and international operating experience (OPEX).**
	or where efficiencies or improvements can be made. This should include the regular collection of data on plant and organisational performance including records of incidents such plant faults, human errors, demands on and the performance of safety systems, radiation doses, and the generation of radioactive waste and effluents. • *External – Reviewing external national and international OPEX and research to identify potential challenges and learning. This should include the review of OPEX collated by organisations by organisations such as the IAEA International Reporting System, WANO and INPO.* *Licence Condition 7 requires licensees to implement adequate arrangements for the notification, recording, investigation and reporting of such incidents occurring on the site.*
Further Guidance	IAEA, A System for Feedback of Experience from Events in Nuclear Installations (NS-G-2.11), IAEA: Vienna, 2006. IAEA, Nuclear Power Plant Operating Experience from the IAEA/NEA International Reporting System for Operating Experience 2012-2014, IAEA: Vienna, 2018. IAEA, Operating Experience with Nuclear Power Stations in Member States, IAEA: Vienna, 2018. Operating Experience and Learning. A Guide to Good Practice, Safety Directors Forum, April 2015. ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition (see Leadership and Management for Safety Principles MS.4). Revision 1 (January 2020)

LM6	**The senior management shall develop and implement an organisational structure with appropriate roles, competence and resources to facilitate the efficient application of the safety management system and ensure that the fundamental safety objective is met.**
Basis	*GSR Part 2 Requirement 9.*
Guidance	*The senior management are responsible for ensuring the organisation is focussed on delivering high standards of safety at all times.* *In accordance with Licence Condition 36, existing and prospective licensees are required to establish a Nuclear Baseline for their organisation to demonstrate that its organisational structure, resources and capability are suitable and sufficient to maintain control and oversight of nuclear safety.* *While the organisation is not expected to retain all of the necessary competency internally, there is a minimum level of technical, managerial and operational capability it is expected to be able demonstrate. This is referred to as core capability and the relevant roles form part of the nuclear baseline. Contractors may be utilised to occupy nuclear baseline roles but these must be shown to be fully embedded secondees.* *The Nuclear Baseline is also fundamental in the demonstration of the organisation's Intelligent Customer Capability, which requires the organisation to have sufficient understanding to intelligently specify, oversee and accept work undertaken on their behalf by contractor organisations.* *The impact of any changes to the nuclear baseline, both changes in roles and responsibilities and the individuals occupying the specified roles, must be assessed with regard to their potential impact on nuclear safety.*
Further Guidance	Safety Directors Forum, Organisational Capability and Resilience, Organisational Capability Working Group, September 2018. ONR, NS-TAST-GD-048, Organisational Change, September 2018. ONR, NS-TAST-GD-065, Function and Content of the Nuclear Baseline, August 2018. ONR, NS-TAST-GD-072, Function and Content of a Safety Management Prospectus, July 2018. ONR, NS-TAST-GD-027, Training and Assuring Personnel Competence, July 2017.

LM6	**The senior management shall develop and implement an organisational structure with appropriate roles, competence and resources to facilitate the efficient application of the safety management system and ensure that the fundamental safety objective is met.**
	ONR, NS-TAST-GD-080, Challenge Culture, Independent Challenge Capability (Including an Internal Regulation Function) and the Provision of Nuclear Safety Advice, July 2018. Safety Directors Forum, Nuclear Industry Good Practice Guide 'Independent Oversight', January 2014. Safety Directors Forum, Development and use of Safety Performance Indicators: A UK Nuclear Industry Good Practice Guide, 2016. Safety Directors Forum, Nuclear Baseline and the Management of Organisational Change: A good practice Guide, Organisational Capability Working Group, March 2017. IAEA, Managing Organizational Change in Nuclear Organizations, IAEA Nuclear Energy series, (NG-T-1.1), IAEA: Vienna, 2014. ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition (see Leadership and Management for Safety Principles MS.2). Revision 1 (January 2020)

Radiological Protection Principles

RP1	**Radiological protection shall optimised to ensure that the risks associated with all activities on the site involving radiological material can be shown to be ALARP.**
Basis	*GSR Part 3 Requirement 1, 9, 10, & 11.*
Guidance	*This principle is fundamental to UK nuclear regulation and should be continually applied by designers and operators alike. Only when there are no further reasonably practicable measures available to reduce the radiological risk can this principle be deemed to be satisfied.* *'Optimise' in the UK regulatory context should be interpreted as being compliant with both the 'As Low As Reasonably Practicable' (ALARP) principle (nuclear safety) and also compliant with 'Best Available Techniques' BAT (radiological aspects of environmental safety).* *Compliance with the ALARP principle is all-encompassing (i.e. it applies to both normal operation and accidents).* *BAT is applicable to the radiological hazard posed by on site radioactive waste arisings and radioactive releases offsite which are a consequence of normal operations (including anticipated operational occurrences) on site. The nuclear industry code of practice on BAT should be followed.*
Further Guidance	ONR, NS-TAST-GD-005, Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable), December 2019. IAEA, Basic Professional Training, Module 2: Radiation Protection in Nuclear Facilities, May 2015. Safety Directors Forum, Best Available Techniques (BAT) for the Management of the Generation and Disposal of Radioactive Wastes Good Practice Guide, December 2010

RP2	**Radiation doses to workers and the public arising from normal operations shall be within statutory limits and discharges shall be within permitted levels.**
Basis	*GSR Part 3 Requirement 12, 20, 21, & 30.*
Guidance	*Dose limits set out in the sub-principles below are whole body effective doses. There are other legal limits on doses for specific groups of people, tissues and parts of the body in the Ionising Radiations Regulations (IRR).*

RP2	**Radiation doses to workers and the public arising from normal operations shall be within statutory limits and discharges shall be within permitted levels.**
	In recognition that the radiological hazard to the public from normal discharges also requires to be limited, the principle also includes the requirement to separately demonstrate compliance with permitted discharge limits in accordance with the Radioactive Substances Regulation (RSR) under the Environmental Permitting Regulations. *The requirements of RP1 (compliance with ALARP/BAT) should always be considered at the same time as assessing compliance with RP2.*
Further Guidance	ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition. Revision 1 (January 2020)

RP2.1	**The following worker dose criteria shall apply to normal operation in a calendar year:** **Site employees or contractors working with ionising radiation – individual dose levels:** BSL: 20 mSv (the legal limit for a classified worker from the IRR) BSO: 1 mSv **Other site employees – individual dose levels:** BSL: 2 mSv BSO: 0.1 mSv **For a defined group of site employees and/or contractors working with ionising radiation, the average effective dose levels are:** BSL: 10 mSv BSO: 0.5 mSv
Basis	*GSR Part 3 Requirement 12.* *Statutory Instruments 2017 No. 1075, Health and Safety, The Ionising Radiations Regulations 2017.* *ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition, Target 1 & Target 2.*

RP2.2	**The following public dose criteria shall apply in a calendar year:** **Any person off the site – individual dose levels:** BSL: 1 mSv (the legal limit from the IRR) BSO: 0.02 mSv
Basis	*ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition, Target 3.*

RP2.3	**Discharges from the plant in normal operation shall be within the permitted levels set by the environmental regulators.**
Basis	*GSR Part 3 Requirements 29 & 31.*
Guidance	*The Radioactive Substances Regulation requires that the licensee minimises discharges to the environment through the plant design and procedures to limit the radiological impact on the general public and the environment.* *The licensee is required to apply to the Environment Agency (EA) / Natural Resources Wales (NRW) / Scottish Environmental Protection Agency (SEPA) for a discharge permit. The permit will specify total activity, radionuclides and form of material than can be discharged from the site via authorised discharge routes and during a specified time period.* *The licensee is required to demonstrate that "Best Available Techniques" have been applied in the management, storage and processing of wastes (see principle RW5).* *The Environment Agency applies the following maximum limits for doses to individuals:* • *0.3 mSv/year from any source from which radioactive discharges are made; or* • *0.5 mSv/year from the discharges from any single site.*
Further Guidance	Environmental Permitting Guidance, Radioactive Substances Regulation For the Environmental Permitting (England and Wales) Regulations 2010, September 2011, Version 2.0. Environmental Permitting Regulations (England and Wales) 2010, Criteria for setting limits on the discharge of radioactive waste from nuclear sites, June 2012.

RP3	**Suitable equipment shall be provided to monitor radiation in normal operational states and design basis fault conditions.**
Basis	*GSR Part 3 Requirement 14 & 32.*
Guidance	*Although the plant design will incorporate features which will limit the expected doses and off site discharges to acceptable levels, suitable and sufficient monitoring and control equipment to confirm that this remains the case in normal operational states. Requirements for monitoring in accident conditions are considered separately in principle EP2.*
Further Guidance	IAEA, Radiation Protection Aspects of Design for Nuclear Power Plants (NS-G-1.13), IAEA: Vienna, 2005. IAEA, Environmental and Source Monitoring for Purposes of Radiation Protection (RS-G-1.8), IAEA: Vienna, 2005. IAEA, Basic Professional Training, Module 2: Radiation Protection in Nuclear Facilities, May 2015. IAEA, Radiation Oncology Physics: A Handbook for Teachers and Students, IAEA: Vienna, 2005. Operational Monitoring Good Practice Guide, The Selection of Alarm Levels for Personnel Exit Monitors, Industry Radiological Protection Coordination Group, Interim Issue, December 2009.

RP4	**Operational experience information from relevant plants, processes and activities shall be used to support the safety assessment.**
Basis	*GSR Part 3 Requirement 14.* *GSR Part 4 Requirement 19.*
Guidance	*The use of operational experience data from other nuclear installations around the world is a significant aid to judging the acceptability and robustness of the plant design and safety case. While this principle is presented as a radiological protection principle, it should be interpreted in its widest sense which includes:* ▪ *Personnel dose assessment data* ▪ *Control of radioactivity in normal operation (e.g. Coolant Chemistry)* ▪ *Equipment performance and reliability (to support evaluations of fault potential and*

RP4	**Operational experience information from relevant plants, processes and activities shall be used to support the safety assessment.**
	also radiological risk in fault conditions) ▪ *Discharges in normal operation*
Further Guidance	IAEA, A System for Feedback of Experience from Events in Nuclear Installations (NS-G-2.11), IAEA: Vienna, 2006. IAEA, Nuclear Power Plant Operating Experience from the IAEA/NEA International Reporting System for Operating Experience 2012-2014, IAEA: Vienna, 2018. IAEA, Operating Experience with Nuclear Power Stations in Member States, IAEA: Vienna, 2018. Operating Experience and Learning. A Guide to Good Practice, Safety Directors Forum, April 2015. ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition (see Leadership and Management for Safety Principles MS.4). Revision 1 (January 2020)

Safety Assessment Principles

SA1	**Systematic fault and hazard identification shall be undertaken to identify all credible conditions and initiating events with potential to lead to radiological harm.**
Basis	*GSR Part 4 Requirements 6 & 8.*
Guidance	*Principle SA1 identifies the requirement to undertake adequate and appropriate hazard identification to ensure that all credible conditions and initiating events are identified which have the potential to result in radiological consequences (dose) to workers or members of the public.* *The scope of the hazard identification exercise shall be comprehensive and include normal operation, AOO, faults, internal and external hazards. The principle requires that all events should be identified even events with very low probability or consequences should be recorded to demonstrate completeness. Events with low probability or consequences may be excluded from subsequent analysis if they comply with the criteria defined under principles SA1.1 and SA1.2.* *Human interactions which can lead to abnormal conditions or faults shall be identified in the hazard identification exercise.* *A hierarchy of control is applied to manage radiological risk in the design and operation of nuclear plant within the UK. While not a formal requirement the hierarchy provides a structured approach for the consideration of risk and generally includes the following stages.* • *Eliminate – where reasonably practicable, radiological risks should be eliminated from plant or processes. This can be achieved in a variety of ways,* • *Reduce – where risks cannot be completely eliminated they should be reduced where reasonably practicable. This could potentially be achieved by reducing the quantity of radiological material involved, or substituting the material for something less hazardous.* • *Isolate – where risks cannot be designed out or reduced to a tolerable level through reduction they can be isolated. Isolation involves the provision of physical barriers such as lead shielding or engineered containment to isolate the radiological hazard.* • *Control – if complete isolation of the radiological hazard is not possible, for example where the operations require man access or access is required for maintenance, control measures may be put in place to control the hazard. For example, engineered interlocks to prevent accidental entry to a hazardous area during period so risk.*

SA1	**Systematic fault and hazard identification shall be undertaken to identify all credible conditions and initiating events with potential to lead to radiological harm.**
	Protection – while the reduction, isolation and control measures are designed to protected people, protection in the context of the hierarchy of control refers to personnel protective equipment as opposed to engineered safety measures, hence its position low down the hierarchy. Protection includes things like Respiratory Protective Equipment (RPE), dosimetry, lead aprons etc. *Both internal and external hazards that may potentially affect the safe operation of the Nuclear Power Plant (NPP) must be identified and their effects considered in safety assessments.* *Internal hazards are those that originate on the licensed site as a result of the activities being undertaken and over which the licensee can therefore be expected to exhibit some control. This includes hazards such as fires associated with flammable inventories on the site, vehicle impacts from on-site movements, floods resulting from the failure of tanks and pipework on the site, toxic gas release, collapses, dropped loads, impacts and explosion/missiles etc.* *External hazards are those that originate from off the site over which the licensee has limited or no control. These can be both man-made, such as loss of grid or aircraft impact, and natural events, such as extreme weather or earthquake.* *The ONR SAPs outline the requirement for hazard identification and characterisation in principle EHA.1 which states:* *An effective process should be applied to identify and characterise all external and internal hazards that could affect the safety of the facility.* *The commentary on this principle explains that:* - Hazards should be identified in terms of their severity and frequency of occurrence and characterised as having either a discrete frequency of occurrence (discrete hazards), or a continuous frequency-severity relation (non-discrete hazards). All hazards should be treated as initiating events in the fault analysis. - Discrete hazards are those that are realised at a single frequency (or set of discrete frequencies) with associated hazard severity / magnitude(s). Most internal hazards such as steam release are discrete hazards. - Non-discrete hazards are those that can occur across a continuous range of frequencies and are defined in terms of a hazard curve (a plot of hazard severity

SA1	**Systematic fault and hazard identification shall be undertaken to identify all credible conditions and initiating events with potential to lead to radiological harm.**
	against the frequency of this severity being exceeded). Seismic hazard is an example of a non-discrete hazard. *The hazard identification itself may utilise a number of different techniques, including but not limited to:* • *Structured What IF Technique (SWIFT)* • *HAZard and OPerability (HAZOP) Studies* • *Failure Modes and Effects Analysis (FMEA)* • *Failure Modes, Effects and Criticality Analysis (FMECA)* • *Task Analysis* • *Review of Standards, RGP and Operating Experience* *The follow on safety assessment of the radiological hazards to people and the environment under normal operating conditions is addressed in principle RP2. The remaining hazards are expected to be compiled onto a fault schedule which will be subject to subsequent safety assessment in accordance with principles SA2 to SA5.*
Further Guidance	ONR, NS-TAST-GD-013, External Hazards, October 2018. ONR, NS-TAST-GD-014, Internal Hazards, November 2019. IAEA, Extreme external events in the design and assessment of nuclear power plants, IAEA-TECDOC-1341, IAEA: Vienna, 2003. ETI. 2018. Enabling Resilient UK Energy Infrastructure: Natural Hazard Characterisation Technical Volumes and Case Studies, Volumes 1 – 12. IMechE, IChemE. IAEA, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants (NS-G-1.7), IAEA: Vienna, 2004. IAEA, Fire Safety in the Operation of Nuclear Power Plants (NS-G-2.1), IAEA: Vienna, 2000. IAEA, Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants (NS-G-1.11), IAEA: Vienna, 2004.

SA1	**Systematic fault and hazard identification shall be undertaken to identify all credible conditions and initiating events with potential to lead to radiological harm.**
	IAEA, Safety of Nuclear Power Plants: Design, Specific Safety Requirements No. SSR-2/1, IAEA: Vienna, 2016. IAEA, Safety of Nuclear Power Plants: Commissioning and Operation, Specific Safety Requirements No. SSR-2/2, IAEA: Vienna, 2016.

SA1.1	**The consequences of each initiating event shall be determined.**
Basis	*GSR Part 4 Requirements 6, 9, 15 & 16.*
Guidance	*In order to assess condition/initiating events the consequences arising from the initiating event must be determined. If the radiological consequences are below 10 µSv (public) or 100 µSv (worker) then no further assessment is required, although, the reason for excluding the condition/fault from the assessment shall be recorded. If the condition/event is above either of these thresholds the fault shall be subject to subsequent safety assessment in accordance with principles SA2 to SA5.* *Initially the consequences may be cast in terms of plant damage, e.g. potential to cause thermal damage to fuel. Later once the plant faults have been grouped then it will be possible to define radiological release source terms for the group of initiating events.* *For GDA dose assessments will be based upon generic assumptions about the site envelope (e.g. nearest habitation) and will be carried out using an approved methodology.* *For deterministic assessments conservative methods (e.g. transient analysis, fault progression and dose assessments) shall be used. For probabilistic assessments best estimate transient analysis, fault progression and dose assessments are preferred so that conservatisms in the analysis do not distort the conclusions of the analysis, however, where best estimate analysis is not practical conservative methods may be used.*
Further Guidance	ONR, NS-TAST-GD-045, Radiological Analysis for Fault Conditions, July 2019. Various publications of the International Council on Radiological Protection (ICRP) and National Radiological Council on Radiation Protection and Measurements (NCRP).

SA1.2	**The probability of each initiating event shall be determined.**
Basis	*GSR Part 4 Requirements 6, 9, 15 & 16.*
Guidance	*Initiating events (single fault sequences) with a frequency below 10^{-5} pa are not included in the deterministic assessment and initiating events with an initiating frequency below 10^{-8} pa are not included in the probabilistic assessment.* *Initiating event frequencies shall be estimated on a best estimate basis.* *The 10^{-8} threshold is based on the assumption that there will be no more than 100 initiating events that could give rise to a dose to a single member of the public exceeding 1,000 mSv. Justification for their exclusion (e.g. low frequency) shall be recorded.* The general lack of adequate reliability data for the disruptive failure of metal components and structures leads to assessments being based primarily on established engineering practice. As a result, although the radiological consequences of the failure of some components or structures may be significant (into the range where there are societal risks), it is not possible to calculate a plausible failure frequency for inclusion in a fault analysis. At best it might be possible to adopt a representative failure rate that would allow the effects of the component or structure failure to be included in a fault analysis in a nominal way or as a sensitivity study. If the safety case is sensitive to the failure frequency, then the estimate will need substantial support from engineering analyses and engineering judgement. At the least, an engineering judgement would be needed to confirm that the component or structure in question has characteristics similar to those in the database used to determine reliability values. If engineered safety systems are provided to cope with the effects of the initiating component or structure failure, the overall case may not demand high confidence in the structural integrity claim. *Failures of SSCs where appropriate specific arguments (e.g. high integrity arguments) have been made can be excluded from the deterministic and probabilistic safety assessment.*
Further Guidance	ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition Revision 1 (January 2020) (see Engineering principles: external and internal hazards EHA.4, Fault analysis: design basis analysis, FA.5).

SA2	**The design basis shall be established.**
Basis	*GSR Part 4 Requirement 8.*
Guidance	*The IAEA defines the design basis as:* *The range of conditions and events taken explicitly into account in the design of a facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits by the planned operation of safety systems.* *The design basis is separated into different regions based on the frequency and consequence of fault and hazard events. Within these different regions different approaches are adopted to the analysis of faults and hazards and the identification of corresponding safety measures.* *Design Basis Analysis (DBA) is applied to faults and hazards for which it is considered that there is a credible risk that they will occur during the facility lifetime and are therefore considered to be reasonably foreseeable.* *DBA is intended to demonstrate the adequacy of the safety measures and the overall tolerance of the plant to the identified design basis events. It is used to derive the operational parameters and constraints that represent Limiting Conditions for Operation and are ultimately captured as Operating Rules.* *DBA does not quantify risk, it promotes the implementation of robust engineered solutions based on suitably conservative analysis where required to account for any uncertainty. The adequacy of the safety measures is determined by comparison with the deterministic safety criteria. Satisfying these criteria alone is not necessarily sufficient to demonstrate ALARP, however compliance with the deterministic criteria should result in a robust design and safety demonstration, and minimise the amount of additional risk reduction measures required to demonstrate ALARP. The residual risk and the risk profile of the plant are assessed in the Probabilistic Safety Assessment.* *In DBA, any uncertainties in the fault progression and consequence analyses are addressed by the use of appropriate conservatism. In this approach, risk is not quantified, but the adequacy of the design and the suitability and sufficiency of the safety measures are assessed against deterministic rules. However, DBA alone may not be sufficient to demonstrate adequate safety of the facility.*
Further Guidance	IAEA, Deterministic Safety Analysis for Nuclear Power Plants (SSG-2) Rev 1, IAEA: Vienna, 2019. Safety Directors Forum, Design Basis Assessment (DBA) Schemes, April 2018.

SA3	The Safety Functions of the plant shall be defined.
Basis	*GSR Part 4 Requirement 7.*

Guidance	*Safety can be assured by satisfying a small number of fundamental safety functions, equally failure of any one of the fundamental safety functions could potentially read to significant risks. The following fundamental safety functions are identified by* IAEA SSR-2/1*:*
	• *Control of Reactivity;*
	• *Removal of heat from the reactor and from the fuel store;*
	• *Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.*
	These fundamental safety functions muse be demonstrated at all states in the plant lifecycle. To achieve this the fundamental safety functions should be decomposed into plant, system and component level safety functional requirements.
	For example, each fundamental safety function may be broken down into safety functions relating to different plant operating modes:
	• *Control of Reactivity*
	    ○ *Reactivity shall be controlled at all times during normal operations.*
	    ○ *Reactivity shall be controlled at all times during fault conditions.*
	    ○ *Adequate shutdown margin shall be maintained at all times during normal operation and fault conditions.*
	    ○ *Long-term hold-down capability shall be available in all planned shutdown and post trip states.*
	    ○ *Fuel handling criticality outside the reactor:*
	    ○ *etc.*

Further Guidance	IAEA, Safety of Nuclear Power Plants: Design, Specific Safety Requirements No. SSR-2/1, IAEA: Vienna, 2016. IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants (SSG-30), IAEA: Vienna, 2014.

SA3.1	**A categorisation scheme shall be defined for the high level safety functions.**
Basis	*GSR Part 4 Requirement 16.*
Guidance	*To ensure that a graded approach is applied the relative importance of the safety functions needs to be determined.* *There is no prescribed framework for categorisation of safety functions, however, the following example is presented in the ONR SAPs, taking account of the IAEA guidance in* SSG-30 *and has been widely adopted by UK licensees as the basis of their own arrangements.* *The safety categorisation should be determined on the following basis:* • *Category A – any function which forms a **principal** means of ensuring nuclear safety;* • *Category B – any function which makes a **significant** contribution to safety;* • *Category C – any other structure, system or component.* *The ONR does not explicitly define what it considers the terms principal and significant to mean and therefore each licensee has generally made its own interpretation. Generally this is linked to the consequences of failing to deliver the safety function, quantified in terms of unmitigated doses to persons onsite and members of the public associated with failure to deliver the safety function, and the likelihood, the best estimate frequency of a demand on the safety function (i.e. the sum of the frequencies of all initiating faults in which the safety function is required).* *In Figures 3a and 3b of* TAG 094 *the ONR roughly equates **principal** to mean above the BSO. For off-site consequences this results in all functions that the fall above the BSL being Category A, while for on-site consequences, for some of the lower consequence regions immediately above the BSO, Category B is considered sufficient.*
Further Guidance	ONR, NS-TAST-GD-094, Categorisation of Safety Functions and Classification of Structures, Systems and Components, July 2019. IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants (SSG-30), IAEA: Vienna, 2014.

SA4	**Safety analysis shall use deterministic, probabilistic and severe accident approaches in order to identify the systems claimed to provide high level safety functions; to quantify the duty of SSCs claimed and to identify assumptions, limits and conditions.**
Basis	*GSR Part 4 Requirement 15, 16 & 17.*
Guidance	*The purpose of the deterministic, probabilistic and severe accident assessments is to determine:* • *Which SSCs are claimed to satisfy the high level safety functions.* • *The functional capability of the SSCs, e.g. flow rates, reliability requirements, etc.* • *Integrity claims, e.g. systems claimed against fire must be able to withstand the fire they are claimed against.* • *Determine that the assessed doses from faults comply with the limits.* • *Determine whether it is reasonably practicable to implement further risk reduction measures (i.e. demonstrate that risks are As Low as Reasonably Practicable – ALARP).* • *Determine any assumptions, assumptions, limits and conditions from the safety assessment which can be used to develop technical specifications for operating the reactor.*
Further Guidance	IAEA, Deterministic Safety Analysis for Nuclear Power Plants (SSG-2), Revision 1 IAEA: Vienna, ONR, NS-TAST-GD-094, Categorisation of Safety Functions and Classification of Structures, Systems and Components, July 2019. IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants (SSG-30), IAEA: Vienna, 2014.

SA4.1	**General Protection Requirements**
Guidance	*A number of requirements apply to both deterministic and probabilistic assessments: these requirements are considered in principles SA4.1.1 – SA4.1.5.* *Requirements which apply solely to deterministic assessment are considered under principles SA4.2.1 - SA4.2.4* *Requirements which apply solely to probabilistic assessment are considered under principles SA4.3.1 –S A4.2.*

SA4.1.1	**The principles of inherent safety and fault tolerance shall be integral to the design of the nuclear plant.**
Basis	*ONR SAPs – EKP.1 & EKP.2* *SSR-2/1 – Requirement 26*
Guidance	*Rather than relying on safety measures to control the response to hazardous situations, modern nuclear power plants should be designed to be inherently safe where practicable, avoiding hazards i.e. the design of the plant makes it physically impossible for a hazardous situation to occur. For example, the design of fuel storage racks may both limit the amount of fuel in a particular area and also maintain adequate separation to avoid criticality in the event of plant hazards or a failure to follow operating procedures.* *Where inherent safety cannot be achieved, fault tolerance should be strived for, with the plant being capable of riding through fault conditions without safety being challenged or naturally defaulting to a safe state.*
Further Guidance	ONR, NS-TAST-GD-075, Safety of Nuclear Fuel in Power Reactors, September 2017.

SA4.1.2	**Defence in depth shall be applied to ensure that multiple barriers are in place all events with potential to lead to radiological harm.**
Basis	*GSR Part 4 Requirement 13*
Guidance	*The IAEA defines the following layers of defence in depth:* • *Address deviations from normal operation.*

SA4.1.2	**Defence in depth shall be applied to ensure that multiple barriers are in place all events with potential to lead to radiological harm.**
	• *Detect and terminate safety related deviations from normal operation should deviations occur.* • *Control accidents within the limits established for the design.* • *Specify measures to mitigate the consequences of accidents that exceed design limits.* • *Mitigate radiation risks associated with possible releases of radioactive material.* *Defence in depth requires that multiple layers of defence are provided via engineered features and management arrangements for preventing failures, and if prevention fails, limiting the consequences and prevention of evolution of events to more serious conditions. It may be noted that defence in depth applies to fault escalation and does not simply require multiple containment barriers or levels of protection.* *The IAEA layers shall be interpreted as a hierarchy and priority given according to the position in the hierarchy. For example, elimination of deviations from normal operation should first be sought. Only if this is not possible should measures to terminate any possible deviations be considered and so on throughout the hierarchy.*
Further Guidance	IAEA, Defence in Depth in Nuclear Safety (INSAG-10), IAEA: Vienna, 1996. IAEA, Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev. 1 (INSAG-12), IAEA: Vienna, 1999. Implementation of Defence in Depth at Nuclear Power Plants, Lessons Learnt from the Fukushima Daiichi Accident, NEA 7248, OECD, 2016.

SA4.1.3	**No single random failure assumed to occur anywhere within the systems provided to perform a safety function shall be capable of compromising the delivery of that safety function.**
Basis	*GSR Part 4 Requirement 16* *SSR-2/1 Requirement 25*
Guidance	*Consequential failures resulting from the assumed single failure should be considered as an integral part of the single failure.*

SA4.1.3	**No single random failure assumed to occur anywhere within the systems provided to perform a safety function shall be capable of compromising the delivery of that safety function.**
	The single failure criterion is applied at the safety function level and applies during all normally permissible plant configurations i.e. must be demonstrated during all operating modes, including maintenance and testing when plant availability may be deliberately degraded. *The level of redundancy provided is generally linked to the importance of the function being delivered, indicated by the Class, and whether failures would be immediately revealed, such as in continuously operating systems, or unrevealed such as in standby systems.* *Failures of SSCs for which appropriate specific arguments have been presented are exempt from principle SA4.1.2. e.g. safety related design provisions such as civil structures where failure cannot be tolerated must be the subject of specific arguments which demonstrate that such SSCs have adequate integrity.* *Application of the single failure criterion is typically mandatory for all Category A & B safety functions, but may also be applied to Category C safety functions where the deterministic or probabilistic assessment identifies that redundancy would be beneficial.* *Typical levels of redundancy (where N is the number of systems required to deliver the safety function) are as follows:*

Safety Function Categorisation	***Standby Systems***	***Continuously Operating Systems***
A	*N+2*	*N+1*
B	*N+1*	*N+1*
C	*N*	*N*

The above levels of redundancy are indicative only. The level of redundancy selected in any given case must be justified by appropriated deterministic and probabilistic assessment.

Further Guidance	ONR, NS-TAST-GD-036, Redundancy, Diversity, Segregation and Layout of Mechanical Plant, 2017.

SA4.1.4	**Common cause failure shall be addressed explicitly in the safety assessment.**
Basis	*SSR-2/1 – Requirement 24*
Guidance	*Common cause failure (CCF) should be addressed explicitly even where a safety measure employs redundant or diverse components, measurements or actions to provide high reliability.* *A cut-off is generally applied which limits claims on safety related systems to 10^{-5} fpd, however, where good quality data has been accumulated this limit may be relaxed if an acceptable case can be made.*
Further Guidance	IAEA, Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev. 1 (INSAG-12), IAEA: Vienna, 1999.

SA4.1.5	**Human failure shall be addressed explicitly in the safety assessment.**
Basis	*GSR Part 4 Requirement 11* *SSR-2/1 Requirement 32*
Guidance	*Human interactions which are claimed in response to abnormal conditions or faults shall be identified in the safety assessment.* *Identification of human actions which can lead to abnormal conditions / faults is dealt with under SA1.* *Substantiation of human actions claimed in the safety case is dealt with under principle SA5.1.* *Where possible the facility designer should identify training requirements so that this information can be passed to any prospective licensee and captured in its training arrangements.* *The specific manning levels and any associated claims on the sufficiency of staffing levels shall be justified by the licensee. However, information on the minimum operational staffing requirements shall be provided by the designer to be used by any prospective licensee.*
Further Guidance	IAEA, Human Factors Engineering in the Design of Nuclear Power Plants (SSG-51), IAEA: Vienna, 2019. ONR, NS-TAST-GD-058, Human Factors Integration, March 2017.

SA4.2	**A comprehensive deterministic safety assessment shall be undertaken to ensure that safety requirements are identified and satisfied by the at all stages in the plant lifecycle.**
Basis	*GSR Part 4 Requirement 15*
Guidance	*Deterministic assessment is a robust demonstration of the fault tolerance of the facility, and of the effectiveness of its safety measures. Its principal aim is to establish which safety measures are claimed in the safety case and the associated functional capability of the safety measures. Deterministic assessment is the primary means of determining limits to safe operation (technical specifications), so that safety functions can be delivered reliably during all modes of operation and under reasonably foreseeable faults.* *In deterministic assessment, any uncertainties in the fault progression and consequence analyses are addressed by the use of appropriate conservatism.*
Further Guidance	IAEA, Basic Professional Training Course, Module 6 - Deterministic Safety Assessment, IAEA: Vienna, 2015. IAEA, Deterministic Safety Analysis for Nuclear Power Plants (SSG-2), Revision 1 IAEA: Vienna, 2019.

SA4.2.1	**The deterministic acceptance criteria shall be met for doses to workers and the public.**
Basis	*GSR Part 4 Requirements 15 & 16*
Guidance	*Each individual licensee develops deterministic criteria that must be satisfied to ensure that the risk to workers has been adequately mitigated.* *These criteria are generally derived from Target 4 in the ONR SAPs, which sets the Basic Safety Level (BSL) and Basic Safety Objective (BSO) in terms of effective dose and initiating event frequency.* *A number of Design Basis Regions are typically specified in terms of initiating event frequency and consequence and in relation to the BSL and BSO, and these regions are used to determine the minimum number and integrity of safeguards required to protect against faults with that fall within them.* *There is no single design basis scheme that is accepted as best practice across the UK nuclear industry. However, the various existing schemes developed by individual licensees tend to have the following common traits:*

SA4.2.1	**The deterministic acceptance criteria shall be met for doses to workers and the public.**
	• *Three distinct Regions, usually denoted as 0, I and II, corresponding to the different safety function categories and associated SSC classes.* • *A Low Consequence Region is often specified between the BSO and BSL in which additional safety measures may be specified.* • *The region boundaries should maintain the hierarchy so far as practicable i.e. transitions directly from Region II to Region 0 should be avoided.* *Figures 3a and 3b of* TAG 094 *the ONR infer a potential design basis scheme that is typical of those adopted by UK licensees. For off-site consequences this results in all functions that the fall above the BSL being Category A, while for on-site consequences, for some of the lower consequence regions immediately above the BSO, Category B is considered sufficient.* *A number of other options for design basis schemes are presented in the* Design Basis Assessment (DBA) Schemes guidance *note recently developed by the UK Safety Directors Forum.*
Further Guidance	IAEA, Deterministic Safety Analysis for Nuclear Power Plants (SSG-2), IAEA: Vienna, 2010. ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition (see Target 4). IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants (SSG-30), IAEA: Vienna, 2014. ONR, NS-TAST-GD-094, Categorisation of Safety Functions and Classification of Structures, Systems and Components, July 2019. Safety Directors Forum, Design Basis Assessment (DBA) Schemes, April 2018.

SA4.2.2	**A classification scheme shall be defined for the Structures, Systems and Components claimed in the deterministic assessment based on the contribution they make delivery of the safety functions.**
Basis	*GSR Part 4 Requirement 16* *SSR-2/1 Requirement 22*
Guidance	*The classification will inform the reliability, redundancy, diversity and independence of support service systems (e.g. C&I and Electrical) and the provision of features for their isolation and for testing their functional capability.*

SA4.2.2	**A classification scheme shall be defined for the Structures, Systems and Components claimed in the deterministic assessment based on the contribution they make delivery of the safety functions.**
	To ensure that a graded approach is applied the relative importance of the contribution that individual SSCs make to the delivery of associated safety functions needs to be determined. *As per Safety Function Categorisation, there is no prescribed framework for the Classification of SSCs and individual licensees may approach it differently. However, the following example is presented in the ONR SAPs, taking account of the IAEA guidance in* SSG-30 *and has been widely adopted by UK licensees as the basis of their own arrangements.* *The safety classification should be determined on the following basis:* • *Class 1 – any SSC that forms a **principal** means of fulfilling a Category A safety function.* • *Class 2 – any SSC that makes a **significant** contribution to fulfilling a Category A safety function, or forms a principal means of ensuring a Category B safety function.* • *Class 3 – any other SSC that contributes to a categorised safety function.* *For SSCs delivering protective or mitigative functions in response to a fault or accident condition, then the principal / significant / other means normally relates to the position in the hierarchy of defence in depth and the order in which the SSCs are called upon to respond in the fault sequence.* *For SSCs delivering preventative functions as part of the normal operation of the plant then it is likely that these will be in continuous or frequent demand. They should initially be considered as a principal means of delivering the safety function, acknowledging that in practice it may not always be practicable to provide normal operational systems delivering other non-safety related functions to level of integrity associated with higher class SSCs.*
Further Guidance	ONR, NS-TAST-GD-094, Categorisation of Safety Functions and Classification of Structures, Systems and Components, July 2019. IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants (SSG-30), IAEA: Vienna, 2014.

SA4.2.3	**The reliability, redundancy, diversity and independence of support service systems (e.g. C&I and Electrical) and the provision of features for isolation testing of their functional capability shall be commensurate with the most onerous classification requirements of the SSCs that they support.**
Basis	*SSR-2/1 Requirements 21, 23 & 24*
Guidance	*Interference between safety systems or between redundant elements of a system shall be prevented by means such as physical separation, electrical isolation, functional independence and independence of communication (data transfer), as appropriate* *The reliability of items important to safety shall be commensurate with their safety significance.*
Further Guidance	NRC, Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems, NUREG/CR-7007, December 2008.

SA4.2.4	**The following deterministic acceptance criteria shall be satisfied for doses to any person (worker, other worker or public)* as a result of activities undertaken on the licensed site.**
Basis	*ONR SAPs Target 4*
Guidance	*The following effective dose criteria shall be applied in deterministic assessment. These are residual dose consequences assuming successful operation of safety measures.* *Any person, from a design basis fault sequence (these limits align with ONR Target 4):*

Fault sequence frequency range	***Effective dose***	
	On-site	***Off-site***
BSL/exceeding 1 x 10^{-3} pa	*20 mSv*	*1 mSv*
BSL/between 1 x 10^{-3} and 1 x 10^{-4} pa	*200 mSv*	*10 mSv*
BSL/between 1 x 10^{-4} and 1 x 10^{-5} pa	*500 mSv*	*100 mSv*
BSO/exceeding 1 x 10^{-5} pa	*0.1 mSv*	*0.01 mSv*

**Workers are classified radiation workers as defined in the (Ionising Radiation Regulations) directly involved with operation and maintenance activities which may result in occupational*

SA4.2.4	**The following deterministic acceptance criteria shall be satisfied for doses to any person (worker, other worker or public)* as a result of activities undertaken on the licensed site.**
	exposure to radiation. Other workers are on site workers who are not routinely expected to be exposed to radiation in the course of their normal duties (e.g. office staff). The public are people external to the site over which the site has no control.
Further Guidance	ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition Revision 1 (January 2020), Annex 2.

SA4.3	**A probabilistic assessment of safety shall be undertaken.**
Basis	*GSR Part 4 Requirement 15* *SSR-2/1 Requirements 10 & 42*
Guidance	*ONRs SAPs state that safety cases for power reactors should include PSA.* *PSA allows the quantification of the overall risk associated with the plant for comparison against the numerical probabilistic risk targets in SA4.3.1 and SA4.3.2.* *PSA assists in achieving a balanced design by enabling the strengths and weaknesses of a particular design to be identified and assessed to ensure that no particular class of accident or feature of the facility makes a disproportionate contribution to the overall risk, e.g. of the order of one tenth or greater.* *PSA also allows complex interactions to be identified and examined in more detail than can be achieved in a deterministic assessment, aiding safety informed decision making.* *There are three levels of PSA:* • *Level 1 – determination of the frequency of events leading to core damage.* • *Level 2 - determination of frequencies and magnitudes of radioactive releases.* • *Level 3 – assessment of societal risk.* *PSA is also used to assess the impact of events that sit outside the design basis owing to them being assessed to be of sufficiently low frequency or in which the provided deterministic safety measures are assumed to fail.*

SA4.3	**A probabilistic assessment of safety shall be undertaken.**
Further Guidance	ONR, NS-TAST-GD-030, Probabilistic Safety Analysis, June 2019. IAEA, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants (SSG-3), IAEA: Vienna, 2010. IAEA, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants (SSG-4), IAEA: Vienna, 2010.

SA4.3.1	**The following probabilistic acceptance criteria shall be satisfied for doses to workers (workers or other workers) resulting from accidents.**
Basis	*ONR SAPs Target 6*
Guidance	*The BSL and BSO for onsite individuals from exposure to ionising radiation from accidents on the facilities across the whole site are as follows (these limits align with ONR Target 5):* *BSL - 10^{-4} per year.* *BSO - 10^{-6} per year.* *For comparison with the BSO/BSL, risk should be derived on a best estimate basis. If best estimate assessments are not practicable, then the assessment of risk should be conservative.* *The targets for the frequencies of any single accidents on individual facilities, which could give doses to a worker/other worker, are (these limits align with ONR Target 6):*

Effective Dose (mSv) - on site	***Predicted Frequency per annum***	
	BSL	***BSO***
2 - 20	*1 x 10^{-1}*	*1 x 10^{-3}*
20 - 200	*1 x 10^{-2}*	*1 x 10^{-4}*
200 - 2000	*1 x 10^{-3}*	*1 x 10^{-5}*
> 2000	*1 x 10^{-4}*	*1 x 10^{-6}*

Further Guidance	ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition Revision 1 (January 2020), Annex 2.

SA4.3.2	**The following probabilistic acceptance criteria shall be met for doses to the public resulting from accidents.**
Basis	*ONR SAPs Target 7*
Guidance	*The BSL and BSO for members of the public from exposure to ionising radiation from accidents on the facilities across the whole site are as follows (these limits align with ONR Target 7):* *BSL - 10^{-4} per year.* *BSO - 10^{-6} per year.* *The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a member of the public, are (these limits align with ONR Target 8):*

Effective Dose (mSv) - off site	***Total Predicted Frequency per annum***	
	BSL	***BSO***
0.1 – 1	*1*	*1 x 10^{-2}*
1 – 10	*1 x 10^{-1}*	*1 x 10^{-3}*
10 – 100	*1 x 10^{-2}*	*1 x 10^{-4}*
100 – 1000	*1 x 10^{-3}*	*1 x 10^{-5}*
> 1000	*1 x 10^{-4}*	*1 x 10^{-6}*

	The total risk of 100 or more fatalities, either immediate or eventual, from accidents arising from all facilities on a site shall be assessed against the following criteria (these limits align with ONR Target 9): *BSL: 10^{-5} pa* *BSO: 10^{-7} pa*
Further Guidance	ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition Revision 1 (January 2020), Annex 2.

SA4.4	**Severe accident analysis shall be undertaken to complement the deterministic and probabilistic assessments and identify any further risk reduction measures required to ensure risks are ALARP.**
Basis	*GSR Part 4 Requirement 15* *SSR-2/1 Requirements 20*
Guidance	*Severe accident analysis (or design extension conditions in IAEA terminology) is aimed at studying the consequences of faults which give rise to an unmitigated public dose in excess of 100 mSv conservatively assessed, or which lead to significant unintended relocation of radioactive material (e.g. degraded core accident).* *Deterministic and probabilistic assessments should ensure that there are no fault sequences which threaten dose limits or lead to significant relocation of radioactive material, however, SAA is required in order to:* • *Demonstrate that risks are ALARP.* • *Assess high consequence events of low frequency beyond the design basis to demonstrate that there is no sudden escalation in consequences.* • *Assess design basis events where the safety provisions are assumed to fail.* • *Inform the emergency plan.* • *Key input to Level 2 PSA.* *Severe accident assessment can be carried out by extending design basis accident sequences so that there is a significant relocation of radioactive material and then determining the consequences of the extended fault sequence. e.g. large Loss Of Coolant Accident (LOCA) could be extended until there is significant core damage.* *Severe accident assessments should use best estimate methods (accepting that a pessimistic view of fault sequence progression is required).*
Further Guidance	ONR, NS-TAST-GD-007, Severe Accident Analysis, September 2017. IAEA, Safety Reports Series No. 56, Approaches and Tools for Severe Accident Analysis for Nuclear Power Plants, IAEA: Vienna, 2008.

SA4.5	**Assumptions, limits and conditions relevant to the safety analysis shall be captured, recorded and monitored to ensure that the plant remains within them.**
Basis	*SSR-2/1 Requirement 28*
Guidance	*One of the key outputs of the safety assessment is a thorough understanding of the assumptions, conditions and limits which have been used in the safety assessment as these assumptions, limits and conditions are key to defining the plant operating envelope and associated technical specifications.* *The purpose of documenting assumptions, limits and conditions relevant to the safety analysis is to provide clarity regarding the boundaries of the safety case. Without such documentation, the safety case may be open to misinterpretation, potentially resulting in the plant being unknowingly operated outside the boundaries of its validity.* *Deterministic analysis is expected to be conservative e.g. pessimisms will be included in the safety analysis which will counteract potential uncertainties in the input data. The overall aim is to build in margins of safety into the analysis. While these are potentially difficult to quantify, the overall analysis is one upon which a high degree of confidence may be placed regarding its use as evidence in supporting safety arguments.* *If assumptions, limits and conditions of the safety case and supporting analysis are not adequately captured and the original design basis and safety margins are not well understood then operational flexibility may be reduced and future plant modifications may be more complicated and costly as the impact of changes will be difficult to assess.*
Further Guidance	IAEA, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants (NS-G-2.2), IAEA: Vienna, 2000.

SA5	**Functional capability and integrity claims on SSCs shall be adequately substantiated.**
Basis	*GSR Part 4 Requirement 10*
Guidance	*In practice each SSC will provide a safety function of varying nuclear safety significance, e.g. the safety significance will depend on the faults being considered. The purpose of the substantiation/qualification is to provide written and auditable justification that the safety measures can meet the safety functional requirements placed up on them, and the level of substantiation should be proportional to the importance of the SSCs to nuclear safety. SSCs ensuring delivery of a post internal/external hazard safety function must be shown to be able to withstand the hazard, i.e. be shown able to deliver the safety function following occurrence of the internal/external hazard. Clearly safety measures whose failure would lead directly to significant radiological consequences and whose failure is claimed to be incredible need*

SA5	**Functional capability and integrity claims on SSCs shall be adequately substantiated.**
	significantly more substantiation than safety measures which are claimed against low risk faults. *Due to the range of safety measures claimed in the safety case it is expected that detailed substantiation criteria will be developed for each class of systems based on advice from IAEA Safety Standards SSR-2/1, Safety of Nuclear Power Plants: Design, ONR SAPs etc.*
Further Guidance	IAEA, Safety of Nuclear Power Plants: Design, Specific Safety Requirements No. SSR-2/1, IAEA: Vienna, 2016. ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition Revision 1 (January 2020).

SA5.1	**Human actions claimed in the safety case shall be demonstrated to be sufficiently robust.**
Basis	*GSR Part 4 Requirement 11* *SSR-2/1 Requirement 32*
Guidance	*Substantiation of human actions should include the following:* • *A clear description of the human action, how it is performed and by whom.* • *A clear description of how the human action contributes to the overall function of the safety measure.* • *A qualitative assessment of the human action. As a minimum, the following factors in should be used as a basis for the assessment, supplemented with a consideration of the adequacy of the procedures and training:* 　○ *Familiarity* 　○ *Time* 　○ *Prompt* 　○ *Ergonomics* • *A justification for any numerical value selected, using references where appropriate.* • *A list of all assumptions and judgements made during the assessment.*

SA5.1	**Human actions claimed in the safety case shall be demonstrated to be sufficiently robust.**
	• *The degree of rigour in preparation and checking of the documentary evidence should be proportionate to the overall importance of the human action required.*
Further Guidance	IAEA, Human Factors Engineering in the Design of Nuclear Power Plants (SSG-51), IAEA: Vienna, 2019. ONR, NS-TAST-GD-058, Human Factors Integration, March 2017.

SA5.2	**The functional capability of SSCs to deliver the safety functions that they are claimed to deliver shall be demonstrated by appropriate means and with sufficient confidence.**
Basis	*SSR-2/1 Requirement 29*
Guidance	*The functional capability claimed of any claimed SSCs should be justified by appropriate means.* *The system duty should be identified from fault sequence analysis, e.g., transient analysis, hazard analysis (e.g. fire hazard analysis, hot gas release consequence analysis, seismic analysis), engineering analysis, structural integrity analysis and radiological analysis.* *Suitable methods of analysis to study the physical and chemical processes taking place during fault sequences should be applied.* *Specific methods for demonstrating functional capability should be documented for each class of equipment, e.g. civil structures, pressure systems, metal components and structures, safety systems, C&I systems etc.*
Further Guidance	ONR, NS-TAST-GD-094, Categorisation of Safety Functions and Classification of Structures, Systems and Components, July 2019. IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants (SSG-30), IAEA: Vienna, 2014.

SA5.3	**The required functional integrity and reliability of SSCs shall be ensured through the application of appropriate examination, maintenance, inspection and testing arrangements.**
Basis	*SSR-2/1 Requirement 29*

Guidance	*The integrity claimed of any safety system which comprises a line of protection or any structure important to safety should be justified by appropriate means as follows:*
	• *The integrity and reliability claimed for safety measures should be supported by:*
	  ○ *Suitable maintenance and inspection activities.*
	  ○ *Periodic testing.*
	  ○ *Availability of the plant.*
	• *The integrity of safety measures should be supported by design, construction, inspection and testing to technical standards commensurate with the importance of the safety functions to be performed.*
	• *It should be shown that no unacceptable ageing of any safety measure has taken place. There should be shown to be an adequate margin against plant ageing effects between the intended operational life and the predicted safe working life of all safety measures.*

Further Guidance	IAEA, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants (NS-G-2.6), IAEA: Vienna, 2002. IAEA, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants (SSG-48), IAEA: Vienna, 2018. IAEA, Maintenance Optimization Programme for Nuclear Power Plants (NP-T-3.8), IAEA: Vienna, 2018. IAEA, Handbook on Ageing Management for Nuclear Power Plants (NP-T-3.24), IAEA: Vienna, 2017. ONR, NS-INSP-GD-028, LC28 Examination, Inspection, Maintenance and Testing, September 2019. ONR, NS-TAST-GD-009, Examination, Inspection, Maintenance and Testing of items Important to Safety, May 2019.

SA5.4	**SSCs shall be appropriately qualified to ensure that they remain capable of delivering the required safety functions in all necessary environmental conditions.**
Basis	*SSR-2/1 Requirement 30*
Guidance	*SSCs claimed to deliver plant Safety Functions shall be shown to be capable of withstanding all environmental conditions in which they are claimed to operate without a detrimental impact on their ability to perform the required function(s).* *The conditions in which SSCs are required to function shall be derived for the hazard and fault assessment. This should take account of the expected environmental conditions prior to, during and following the fault or hazard event. For example, equipment claimed to remain available following a steam release will require to be qualified against the elevated temperatures and moisture ingress.* *When deriving qualification requirements for SSCs the potential interactions with other SSCs in the local environment must also be considered. For example, in a seismic event any SSCs claimed to remain available must be qualified to withstand the direct impact of the seismic motion. In addition, their ability to deliver their function must not be impaired by the failure of adjacent equipment that is not required to remain functional and available. This can result in additional qualification requirements on systems not to fail and impair the ability of other SSCs to deliver their safety functions.*
Further Guidance	IAEA, Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing, Safety Reports Series No.3, IAEA: Vienna, 1998.

Radioactive Waste Management Principles

RW1	**The plant shall be designed to minimise waste arisings during all phases of the plant lifecycle.**
Basis	*GSR Part 5 Requirement 8*
Guidance	*Compliance with this principle is expected to show that the following waste hierarchy has been considered in the design and planned operation of the plant:* • *Eliminate* • *Reduce* • *Reuse* • *Recycle* • *Discharge, Storage & Disposal* *Implicit in the principle is the presentation of a radioactive waste strategy which addresses waste arisings throughout the complete lifecycle. The principle of minimisation applies to both waste activity and waste volume, and in general the philosophy of “concentrate and contain” rather than “dilute and disperse” should be followed when designing waste management arrangements.* *The minimisation of waste is considered to be equivalent to minimising the harm to the public and the environment. Since waste arisings are a result of normal plant operation, it is necessary to show that the radioactive waste plant design is compliant with the requirements of BAT.* *Accumulation of radioactive waste on site should be minimised (recognising cases where storage is required for decay purposes), and the waste storage systems should be designed to:* • *Minimise release to the environment in normal operation, fault and accident conditions;* • *Limit the spread of contamination from leakage.*
Further Guidance	ONR / NRW / SEPA / EA, Basic Principles of Radioactive Waste Management, An introduction to the management of higher activity radioactive waste on nuclear licensed sites, February 2015.

RW1	**The plant shall be designed to minimise waste arisings during all phases of the plant lifecycle.**
	HSE, EA, SEPA, The management of higher activity radioactive waste on nuclear licensed sites Part 3a Waste minimisation, characterisation and segregation, 2010.

RW2	**Adequate arrangements shall be put in place to ensure that an inventory of radioactive waste arisings generated by activities undertaken on the site is managed and maintained throughout the plant lifecycle.**
Basis	*GSR Part 5 Requirement 9 & 21*
Guidance	*This principle is complementary to principle RW1 regarding waste minimisation. Whilst waste minimisation is a key objective, the design of the radioactive waste facilities will have to demonstrate that they are fit for purpose in supporting operations throughout the plant lifecycle.* *To this end it is necessary to provide an inventory of the expected waste arisings throughout the plant lifecycle. This is part of the wider material nuclear material accountancy arrangements that monitor and control the location, type, form and quantity of nuclear material on the site at any point in time. Assumptions and conservatisms applied to the data shall be clearly stated. Demonstration of waste minimisation and compliance with BAT will need to take due account of conservatisms in the inventory data since unbalanced conservatism is unlikely to be compatible with demonstrating compliance with principle RW1.*
Further Guidance	IAEA, Classification of Radioactive Waste (GSG-1), IAEA: Vienna, 2009. IAEA, Storage of Radioactive Waste (WS-G-6.1), IAEA: Vienna, 2006. IAEA, Storage of Spent Nuclear Fuel (SSG-15), IAEA: Vienna, 2012. IAEA, The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste, (GSG-3), IAEA: Vienna, 2013. ONR, NS-TAST-GD-024, Management of Radioactive Material and Radioactive Waste on Nuclear Licensed Sites, September 2019. HSE, EA, SEPA, Fundamentals of the management of radioactive waste An introduction to the management of higher-level radioactive waste on nuclear licensed sites, December 2007.

RW2	**Adequate arrangements shall be put in place to ensure that an inventory of radioactive waste arisings generated by activities undertaken on the site is managed and maintained throughout the plant lifecycle.**
	ONR / NRW / SEPA / EA, Basic Principles of Radioactive Waste Management, An introduction to the management of higher activity radioactive waste on nuclear licensed sites, February 2015. NRW, SEPA, NIEA, EA, Hazardous Waste: Interpretation of the definition and classification of hazardous waste, Technical Guidance WM2, 2013.

RW3	**Radioactive wastes generated by activities undertaken on the site will be appropriately characterised, segregated and stabilised to ensure that the risks associated with their processing, storage and ultimate disposal are ALARP.**
Basis	*GSR Part 5 Requirements 9 & 10*
Guidance	*In addition to the radioactivity, the physical chemical and biological properties of nuclear wastes can have a significant impact on the ease with which it can be safely handled, processed, stored, and ultimately disposed of.* *Volatile and mobile waste forms such as active liquors are to be avoided where practicable due to the difficulties associated with storing and controlling them and the risks posed by loss of containment.* *Characterisation of the waste stream should include identification of the radionuclides, their quantities / concentration, the form of the waste, the mass / volume of waste, and any other non-radiological components of the waste stream.* *Waste streams from nuclear power plants are generally considered to be stable and well understood.* *Radioactive waste should be processed into a passively safe state as soon as is reasonably practicable and the design of the radioactive waste processing plant should demonstrate that optimum segregation of waste is delivered. Segregation will need to consider the activity of the waste, radioactive decay times, chemical composition, state (solid/liquid) etc.*
Further Guidance	IAEA, Classification of Radioactive Waste (GSG-1), IAEA: Vienna, 2009. IAEA, Strategy and Methodology for Waster Characterisation, IAEA-TECDOC-1537, IAEA: Vienna, March 2007.

RW3	**Radioactive wastes generated by activities undertaken on the site will be appropriately characterised, segregated and stabilised to ensure that the risks associated with their processing, storage and ultimate disposal are ALARP.**
	ONR / NRW / SEPA / EA, Basic Principles of Radioactive Waste Management, An introduction to the management of higher activity radioactive waste on nuclear licensed sites, February 2015. ONR, EA, SEPA, The management of higher activity radioactive waste on nuclear licensed sites Part 3b Conditioning and disposability, 2011.

RW4	**Adequate facilities and arrangements shall be in place to ensure that radioactive wastes generated on the site can be processed, stored and transferred safely during all phases of the plant lifecycle.**
Basis	*GSR Part 5 Requirements 10 & 11*
Guidance	*The onsite wastes processing and storage facilities must be capable of safely dealing with the planned waste streams and be of sufficient capacity to accommodate the highest anticipated rate of waste generation without a detrimental impact on plant operations.* *Potential bottlenecks in the waste processing and storage processes must be understood and contingency arrangements in place to mitigate any potential impacts on nuclear safety.*
Further Guidance	IAEA, Storage of Radioactive Waste (WS-G-6.1), IAEA: Vienna, 2006. IAEA, Storage of Spent Nuclear Fuel (SSG-15), IAEA: Vienna, 2012. ONR, EA, SEPA, The management of higher activity radioactive waste on nuclear licensed sites Part 3c Storage of Radioactive Waste, 2011. NDA, Geological Disposal, Generic Waste Package Specification, NDA/RWMD/067, 2012.

RW5	**The radioactive waste processing and storage facilities shall be shown to be incorporate relevant good practice and shall be shown to satisfy the requirements of BAT.**
Basis	*GSR Part 5 Requirement 14*
Guidance	*Best Available Techniques (BAT) was defined by the Oslo Paris (OSPAR) Convention as follows:* *BAT means the latest stage of development of processes, facilities or methods of operation which indicate the practical suitability of a particular measure for limiting waste arising and disposal. In determining what constitutes BAT consideration shall be given to:* *1. comparable processes, facilities or methods which have been tried out successfully;* *2. technological advances and changes in scientific knowledge and understanding;* *3. the economic feasibility of such techniques;* *4. time limits for installation in both new and existing plants;* *5. nature and volume of the disposals concerned.* *It follows that BAT will change with time in the light of technological advances, economic and social factors, and changes in scientific understanding.* *The application of BAT requires that current best practice is identified and applied to the facilities and processes for the storage and handling of radioactive wastes.*
Further Guidance	IAEA, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors (SSG-40), IAEA: Vienna, 2016. ONR, EA, SEPA, The management of higher activity radioactive waste on nuclear licensed sites Part 3c Storage of Radioactive Waste, 2011. Safety Directors Forum, Best Available Techniques (BAT) for the Management of the Generation and Disposal of Radioactive Wastes Good Practice Guide, December 2010.

RW6	**Suitable and sufficient sampling systems and monitoring equipment shall be provided (e.g. level, volume, radioactivity) to enable the operator to account for the nature and location of radioactive waste at all times during storage and disposal.**
Basis	*GSR Part 5 Requirement 21*
Guidance	*There are two aspects to this principle. Firstly it is necessary for the operator of the waste facilities to be able to demonstrate that the wastes are properly accounted for at all locations within the waste processing and storage facilities. The design needs to provide the operator with appropriate information to demonstrate control.* *The second aspect is instrumentation needs to be provided to alert the operator to faults including leaks.*
Further Guidance	IAEA, Environmental and Source Monitoring for Purposes of Radiation Protection (RS-G-1.8), IAEA: Vienna, 2005. IAEA, Nuclear Material Accounting Handbook, Services Series 15, IAEA: Vienna, 2008. HSE, EA, SEPA, The management of higher activity radioactive waste on nuclear licensed sites Part 3d Managing information and records relating to radioactive waste in the United Kingdom, 2010.

Decommissioning Principles

D1	**The design shall include appropriate features and materials, as far as is reasonably practicable, to minimise wastes and facilitate safe and efficient decommissioning.**
Basis	*GSR Part 6 Requirements 1, 3, 6, 8*
Guidance	*The requesting party / licensee is required to demonstrate that, based on current understanding and existing technologies and techniques, the plant can be decommissioned safely.* *The following shall be considered in the design of the plant:* • *Ease of disassembly* • *Minimisation of contaminated inventory* • *Use of easy-to-clean surfaces and coatings* • *Avoidance of unnecessary corners, joints and dead spaces where radiological material may be accumulated.*
Further Guidance	IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities (SSG-47), IAEA: Vienna, 2018. ONR, NS-TAST-GD-026, Decommissioning, September 2019.

D2	**The safety case shall demonstrate, as far as is reasonably practicable, that the plant may be decommissioned safely using current best practice with regard to technologies, processes and techniques.**
Basis	*GSR Part 6 Requirement 3*
Guidance	*From the earliest stages in the lifecycle the safety case is required to provide sufficient confidence that the plant, process or facility can be safely decommissioned.* *The level of detail required is commensurate with the stage in the lifecycle. For example at the concept design stage demonstration that current best practice regarding decommissioning has been incorporated into the design is sufficient, however, the operational safety case will require to be supported by a high level decommissioning plan outlining the programme and activities involved at a high level. This will be developed throughout the plant lifecycle to account for advances in techniques and technology and*

D2	**The safety case shall demonstrate, as far as is reasonably practicable, that the plant may be decommissioned safely using current best practice with regard to technologies, processes and techniques.**
	ensure that any changes to the plant design or configuration are incorporated, including the effects of ageing and degradation.
Further Guidance	IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities (SSG-47), IAEA: Vienna, 2018. ONR, NS-TAST-GD-026, Decommissioning, September 2019.

Emergency Preparedness Principles

EP1	**Robust arrangements shall be in place for emergency response in severe accident conditions or in the event that the design basis safety measures have failed.**
Basis	*GSR Part 7 Requirements 1, 2 & 21*
Guidance	While the design of the plant and the safety case aim to demonstrate how safety of the plant will be ensured in all identified credible hazard and fault conditions, licensees are also required to develop robust arrangements for dealing with emergencies where the safety measures have either failed or an unforeseeable or extreme event has occurred that is outside the design basis of the plant. Licence Condition 11 specifies the requirement for licensees to *make and implement adequate arrangements for dealing with any accident or emergency arising on the site and their effects.* This requires the licensee to implement arrangements to manage the response to a radiological emergency occurring on the site, including the provision of onsite emergency response capability and any reliance on offsite emergency response actions. In addition to the licence condition, the requirement to ensure that the public around the site are kept adequately informed is detailed in the Radiation Emergency Preparedness and Public Information Regulations (REPPIR). REPPIR places a legal responsibility on licensees to develop emergency response plans that: • Establish an emergency planning area around the site, identifying the population that may be directly impacted by a radiation emergency; • Detail arrangements for ensuring the public are properly informed and prepared of what to do in the unlikely event of radiation emergency; • Detail arrangements for ensuring the public are kept adequately informed following a radiation emergency. REPPIR do not replace existing nuclear site licence conditions but operators of licensed sites who comply with those conditions will satisfy equivalent provisions in REPPIR.
Further Guidance	IAEA, Accident Management Programmes for Nuclear Power Plants (SSG-54), IAEA: Vienna, 2019. Leadership, Human Performance and Internal Communication in Nuclear Emergencies (NG-T-1.5), IAEA: Vienna, 2018.

EP1	**Robust arrangements shall be in place for emergency response in severe accident conditions or in the event that the design basis safety measures have failed.**
	HSE, Radiation (Emergency Preparedness and Public Information) Regulations (REPPIR), 2019. HSE, A Guide To The Radiation (Emergency Preparedness And Public Information) Regulations (Guidance On Regulations, L126), 2001. ONR / HSE, Work with ionising Radiation, The Radiation (Emergency Preparedness and Public Information) Regulations 2019, Approved Code of Practice and Guidance, 10th April 2019.

EP2	**Adequate provisions shall be made for personnel evacuation and shelter, emergency equipment / facilities (including access control points), emergency control centre, radiation monitoring and on-site communications systems for potential use in a nuclear or radiological emergency.**
Basis	*GSR Part 7 Requirements 1, 2 & 21* *ONR Licence Condition 11*
Guidance	*A licensee's duty of care to its employees requires that it establishes and maintains robust emergency response arrangements. These generally include but are not limited to:* • *Providing shelters / safe havens / muster points to which staff evacuate to in the event of a radiation emergency;* • *Establishing safe evacuation routes to enable staff to reach the shelters / safe havens / muster points. This can include the provision of shielded or fire proof escape routes and emergency HVAC systems;* • *Establishment of onsite and offsite emergency control centres to coordinate the emergency response activities;* • *Provision of adequate emergency communications;* • *Onsite emergency services;* • *Radiation monitoring facilities.* *The list of provisions specified in the principle is considered to represent a minimum set and it may be appropriate for further provisions to be made.*

EP2	**Adequate provisions shall be made for personnel evacuation and shelter, emergency equipment / facilities (including access control points), emergency control centre, radiation monitoring and on-site communications systems for potential use in a nuclear or radiological emergency.**
Further Guidance	ONR, NS-INSP-GD-011, LC 11 – Emergency Arrangements, July 2017. IAEA, Accident Management Programmes for Nuclear Power Plants (SSG-54), IAEA: Vienna, 2019. ONR, NS-TAST-GD-007, Severe Accident Analysis, September 2017.

EP3	**The designer shall establish a conservative derivation of the source term(s) for use in assessing the radiological consequences of severe accident scenarios.**
Basis	*GSR Part 7 Requirements Part 3, 4 & 5*
Guidance	*Implicit in the provision of appropriate source term(s) is the expectation that the designer should provide evaluations of the radiological consequences of severe accident scenarios. It may be appropriate to consider a range of accident scenarios, particularly if bounding on-site consequences do not result from a single severe accident scenario. The purpose of such assessments is to evaluate the resilience of the severe accident provisions provided under EP1.*
Further Guidance	IAEA, Accident Management Programmes for Nuclear Power Plants (SSG-54), IAEA: Vienna, 2019. ONR, NS-TAST-GD-007, Severe Accident Analysis, September 2017.