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Pre-Construction Safety Report

Chapter 04

General Safety and Design Principles

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4.1 List of Abbreviations and Acronyms

ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
BSL	Basic Safety Level
BSO	Basic Safety Objective
CGN	China General Nuclear Power Corporation
DBC	Design Basis Condition
DEC	Design Extension Condition
DEC-A	Design Extension Condition A
DEC-B	Design Extension Condition B
DiD	Defence in Depth
DR	Design Reference
DSA	Deterministic Safety Analysis
EMIT	Examination, Maintenance, Inspection and Testing
EUR	European Utility Requirement
GDA	Generic Design Assessment
HAD	Chinese Nuclear Safety Guide
HAF	Chinese Nuclear Safety Regulation
HBSC	Human-Based Safety Claim
HPR1000 (FCG3)	Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
LOOP	Loss of Offsite Power
MSQA	Management of Safety and Quality Assurance
NPP	Nuclear Power Plant

NRC	Nuclear Regulatory Commission (US)
ONR	Office for Nuclear Regulation (UK)
OPEX	Operating Experience
PCSR	Pre-Construction Safety Report
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
RCC-E	Design and Construction Rules for Electrical and I&C Systems and Equipment
RPT	Radiation Protection Target
RG	Regulatory Guide (US)
RGP	Relevant Good Practice
SFC	Single Failure Criterion
SFP	Spent Fuel Pool
SSC	Structures, Systems and Components
TAG	Technical Assessment Guide (UK)
UK HPR1000	UK version of the Hua-long Pressurised Reactor
URD	Utility Requirements Document (US)
WENRA	Western European Nuclear Regulators Association

4.2 Introduction

The main objective of this chapter is to present the General Safety and Design Principles used in the design and assessment of the UK version of the Hua-long Pressurised Reactor (UK HPR1000). These principles are developed based on internationally recommended good practice with consideration of UK context. More details are presented in the *General Safety Requirements*, Reference [1]. The information presented in this document is based on the version 3 of the UK HPR1000 Design Reference (DR3), as described in the *UK HPR1000 Design Reference Report*, Reference [2].

4.2.1 Chapter Route Map

The ***Fundamental Objective*** of the UK HPR1000 is that: *The Generic UK HPR1000 could be constructed, operated, and decommissioned in the UK on a site bounded by the generic site envelope in a way that is safe, secure and that protects people and the*

environment.

To underpin this objective, five high level claims (Level 1 claims) and a number of Level 2 claims are developed and presented in Chapter 1. This chapter supports ***Claim 2.3*** derived from Level 1 ***Claim 2***.

Claim 2*: The UK HPR1000 design will be developed in an evolutionary manner, using a robust design process, building on relevant good international practice, to achieve a strong safety and environmental performance.*

Claim 2.3*: Suitable General Safety and Design Principles are in place to ensure the design meets the nuclear safety objective.*

To support ***Claim 2.3***, this chapter develops several relevant arguments and evidence:

a) ***Argument 2.3.SC04-A1****: The Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)) is designed based on Chinese nuclear regulatory requirements and international good practice. The UK HPR1000 General Safety and Design Principles are developed from HPR1000 (FCG3) principles with the consideration of UK context.*

   1) ***Evidence 2.3.SC04-A1-E1:*** *The principles used in the design and assessment of HPR1000 (FCG3) have been demonstrated to comply with Chinese nuclear regulatory requirements. The HPR1000 (FCG3) principles are also developed based on International Atomic Energy Agency (IAEA) safety standards. In the development of HPR1000 (FCG3), the conceptual design has passed IAEA Generic Reactor Safety Review. Furthermore, other internationally recommended good practice are also considered, such as Western European Nuclear Regulators Association (WENRA) safety reports, Regulatory Guides (RGs) from the Nuclear Regulatory Commission (NRC), European Utility Requirements (EURs) and Utility Requirements Document (URD).*

   2) ***Evidence 2.3.SC04-A1-E2:*** *The UK HPR1000 General Safety and Design Principles are developed based on the China General Nuclear Power Corporation (CGN) HPR1000 (FCG3) principles, which have been revised to reflect and adapt to the UK context, with the consideration of relevant documents such as Health and Safety at Work etc. Act 1974, Reference [3], Safety Assessment Principles for Nuclear Facilities, Reference [4], and a series of Technical Assessment Guides (TAGs).*

      Moreover, Relevant Good Practice (RGP), Operating Experience (OPEX) and experience from other Generic Design Assessment (GDA) projects are also considered.

b) ***Argument 2.3.SC04-A2:*** *Adequate and suitable General Safety and Design Principles are developed with a logical process to be applied to the design to*

achieve the nuclear safety objective.

Evidence 2.3.SC04-A2-E1: *The principles are formulated following a logical process as follows:*

- The nuclear safety objective is applied in the design of the UK HPR1000.

  The nuclear safety objective is based on the fundamental safety objective presented in *Fundamental Safety Principles* of IAEA, Reference [5] and on the expectation of UK regulators. In order to meet the nuclear safety objective, the radiation safety requirements are formulated on the basis of the numerical targets provided in the *Safety Assessment Principles for Nuclear Facilities*, Reference [4], and it is required to reduce risk to a level that is As Low As Reasonably Practicable (ALARP). The nuclear safety objective is presented in Sub-chapter 4.4.1.

- Defence in Depth (DiD) is implemented primarily through the combination of a number of consecutive and independent levels of protection.

  International consensus is that the application of DiD is the appropriate strategy of preventing accidents and mitigating the consequences of accidents in order to achieve the nuclear safety objective. In accordance with *Safety of Nuclear Power Plants: Design*, Reference [6], DiD is implemented through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would still be available. DiD is presented in Sub-chapter 4.4.2.

- Safety Analysis is carried out for the UK HPR1000 to support the demonstration that the level of nuclear safety risk meets the nuclear safety objective.

  Comprehensive safety analysis to assess the integrity and adequacy of levels of DiD, using a range of complementary techniques appropriate to the level(s) in question, is the recognised good practice in demonstrating nuclear safety in design. The safety analysis is used to derive specific requirements for the safety measures. The safety analysis is presented in Sub-chapter 4.4.3.

- The safety functions are identified to ensure a structured and comprehensive implementation of safety requirements in the design.

  A methodology is applied to identify the safety functions and the safety measures (engineering and administrative) that deliver the safety

functions. This is based on recognised good practice for ensuring a structured and comprehensive implementation of safety requirements in design. The identification of safety functions is presented in Sub-chapter 4.4.4.

- Categorisation of safety functions and classification of Structures, Systems and Components (SSC) are provided to establish a link between the importance of a safety function and the level of reliability and quality of SSCs that perform the safety functions.

  The classification of SSCs related to safety functions is based on their role in achieving the function. The categorisation of safety functions and classification of safety functions are presented in Sub-chapter 4.4.5.

- Engineering substantiation of the safety measures is applied to ensure that the safety measures satisfy the requirements identified by safety analysis.

  Engineering substantiation is applied to ensure that the safety measures deliver the requirements (including support to the safety functions), in the conditions identified by the safety analysis, with an appropriate level of reliability according to the classification of the SSCs. Engineering substantiation is presented in Sub-chapter 4.4.6.

- The selection of applicable codes and standards is carried out to ensure that the reliability of SSCs is commensurate with classification of SSCs.

  The principles and the process for selection of codes and standards ensure that SSCs are designed, manufactured, constructed, installed, commissioned, operated, tested, inspected and maintained based on applicable codes and standards to make the reliability of SSCs commensurate with the importance of the safety function delivered by SSCs. The principles and the process for selection of codes and standards are presented in Sub-chapter 4.4.7.

4.2.2 Chapter Structure

This chapter presents the General Safety and Design Principles to apply to the design and assessment of the UK HPR1000. The structure of Chapter 4 is as follows:

a) Sub-chapter 4.1 List of Abbreviations and Acronyms.

   This sub-chapter lists the abbreviations and acronyms used in the chapter.

b) Sub-chapter 4.2 Introduction.

   This sub-chapter presents the purpose and scope of Chapter 4:

   1) Sub-chapter 4.2.1 presents the claims, arguments and evidence for Chapter 4;

2) Sub-chapter 4.2.2 presents the structure of Chapter 4;

3) Sub-chapter 4.2.3 presents interfaces between Chapter 4 and other chapters.

c) Sub-chapter 4.3 Applicable Codes and Standards.

This sub-chapter presents the applicable codes and standards:

1) Sub-chapter 4.3.1 presents the codes and standards applied in the HPR1000 (FCG3);

2) Sub-chapter 4.3.2 presents the codes and standards applied in the UK HPR1000.

d) Sub-chapter 4.4 General Safety and Design Principles.

This sub-chapter presents the General Safety and Design Principles:

1) Sub-chapter 4.4.1 presents the nuclear safety objective, the requirement that risk is reduced to be ALARP and radiation safety requirements in terms of Radiation Protection Targets (RPTs);

2) Sub-chapter 4.4.2 presents DiD as reflected through all safety related activities for the UK HPR1000;

3) Sub-chapter 4.4.3 presents the principles for safety analysis of the UK HPR1000 which supports the demonstration that the level of nuclear safety risk meets the nuclear safety objective;

4) Sub-chapter 4.4.4 presents the principles for identification, decomposition and application of the safety functions;

5) Sub-chapter 4.4.5 presents the principles of safety classification for SSCs and relevant safety requirements;

6) Sub-chapter 4.4.6 presents the engineering substantiation principles to meet the safety requirements, including Single Failure Criterion (SFC), diversity, independence, etc.;

7) Sub-chapter 4.4.7 presents the selection principles of applicable codes and standards for engineering substantiation of SSCs for their reliability.

e) Sub-chapter 4.5 Concluding Remarks.

This sub-chapter presents the concluding remarks for this chapter.

f) Sub-chapter 4.6 References.

This sub-chapter lists all the references cited in this chapter.

4.2.3 Interfaces with Other Chapters

The interfaces between this chapter and other chapters in Pre-Construction Safety Report (PCSR) are presented in T-4.2.3-1.

T-4.2.3-1 Interfaces between Chapter 4 and Other Chapters in PCSR

PCSR Chapter	Interfaces
Chapter 1 Introduction	Chapter 4 provides arguments and evidence to support the Level 2 Claim 2.3 presented in Chapter 1.
Chapter 5 Reactor Core	Chapters 5 to 11 present the design of systems and components which is based on the relevant safety and design principles provided in Chapter 4.
Chapter 6 Reactor Coolant System	
Chapter 7 Safety Systems	
Chapter 8 Instrumentation and Control	
Chapter 9 Electric Power	
Chapter 10 Auxiliary Systems	
Chapter 11 Steam and Power Conversion System	
Chapter 12 Design Basis Condition Analysis	Chapter 12 presents the Design Basis Condition (DBC) analysis which is based on the safety analysis principles provided in Chapter 4.
Chapter 13 Design Extension Conditions and Severe Accident Analysis	Chapter 13 presents the Design Extension Condition (DEC) analysis which is based on the safety analysis principles provided in Chapter 4.
Chapter 14 Probabilistic Safety Assessment	Chapter 14 provides the methodology and results of Probabilistic Safety Assessment (PSA) which are based on the safety analysis principles set out in Chapter 4.
Chapter 15 Human Factors	Chapter 15 presents the substantiation of human factors principles which are provided in Chapter 4.
Chapter 16 Civil Works & Structures	Chapter 16 presents the design of civil works and structures which is based on the relevant principles in Chapter 4.
Chapter 17 Structural Integrity	Chapter 17 demonstrates the structural integrity by applying design requirements based on the relevant principles presented in Chapter 4.
Chapter 18 External Hazards	Chapter 18 provides the methodology for protection of external hazards with the consideration of relevant principles presented in Chapter 4.

PCSR Chapter	Interfaces
Chapter 19 Internal Hazards	Chapter 19 provides the methodology for protection of internal hazards with the consideration of relevant principles presented in Chapter 4
Chapter 20 MSQA and Safety Case Management	Chapter 20 presents codes and standards applied in Management of Safety and Quality Assurance (MSQA) which is based on the selection principles of codes and standards in Chapter 4.
Chapter 21 Reactor Chemistry	Chapter 21 presents codes and guidelines applied in chemistry based on the selection principles of codes and standards provided in Chapter 4.
Chapter 22 Radiological Protection	Chapter 22 presents the design of radiological protection based on the radiation safety requirements provided in Chapter 4.
Chapter 23 Radioactive Waste Management	Chapter 23 and Chapter 28 present the design of systems and components containing radioactive materials based on the safety and design principles provided in Chapter 4.
Chapter 28 Fuel Route and Storage	
Chapter 24 Decommissioning	Chapter 24 presents codes and guidelines applied in decommissioning based on the selection principles of codes and standards provided in Chapter 4.
Chapter 25 Conventional Safety and Fire Safety	Chapter 25 presents applicable codes and standards in conventional safety and fire safety which are compliant with the selection principles of codes and standards provided in Chapter 4.
Chapter 29 Interim Storage of Spent Fuel	Chapter 29 presents applicable codes and standards in interim storage of spent fuel which are based on the selection principles of codes and standards provided in Chapter 4.
Chapter 30 Commissioning	Chapter 30 presents the arrangements and requirements for commissioning, which considers the relevant principles presented in Chapter 4.
Chapter 31 Operational Management	Chapter 31 uses the relevant engineering substantiation principles presented in Chapter 4 to develop the operational management for the UK HPR1000.
Chapter 32 Emergency preparedness	Chapter 32 presents the emergency preparedness required by principles in Chapter 4.
Chapter 33 ALARP Evaluation	Chapter 33 presents the ALARP Evaluation of the UK HPR1000, which based on the compliance with RGP and the RPTs addressed in Chapter 4.

4.3 Applicable Codes and Standards

The codes and standards applied in Chapter 4 are selected and determined based on the *General Principles for Application of Laws, Regulations, Codes and Standards*, Reference [7]. The principles applied in the UK HPR1000 are developed from

HPR1000 (FCG3), with additional the consideration of UK context. The analysis of applicability of the codes and standards of the UK HPR1000 are presented in Sub-chapter 4.3.1 and 4.3.2.

4.3.1 Codes and Standards Applied in the HPR1000 (FCG3)

The principles applied in the design of HPR1000 (FCG3) are mainly developed from Chinese codes and standards and from IAEA safety standards.

The Chinese codes and standards followed by HPR1000 (FCG3) are mainly Chinese Nuclear Safety Regulation (HAF) series such as HAF 101 (Safety Regulation on Site Evaluation for Nuclear Power Plants), HAF 102 (Safety Regulation on Design of Nuclear Power Plants) and their supporting guides the Chinese Nuclear Safety Guide (HAD) series. The HAF and HAD series are generally developed based on the related IAEA safety standards with consideration of Chinese regulatory requirements.

IAEA safety standards were considered during the HPR1000 (FCG3) design process. In the development of HPR1000 (FCG3), the conceptual design has passed IAEA Generic Reactor Safety Review. IAEA safety standards are internationally recognised and designed to ensure a high level of safety in the use of nuclear materials and radiation sources around the world. Many member states of IAEA have adopted the IAEA safety standards for use in their national regulations. The standards are applied by designers around the world to improve the safety level of nuclear power plants.

WENRA is an internationally recognised association of the heads of the Nuclear Regulatory Authorities of the European Union countries with Nuclear Power Plants (NPPs) and Switzerland. WENRA produces safety reports to develop a harmonised approach to nuclear safety, radiation protection and their regulation. These reports contribute to improvement of national nuclear safety requirements. The related reports are also considered in the process of forming safety principles, such as *Safety of New NPP Designs,* Reference [8], *Safety Reference Levels for Existing Reactors,* Reference [9].

Furthermore, EUR and URD are also referred to in the design process. Though they are not codes and standards, their requirements represent the expectation of utilities and are worth considering.

4.3.2 Codes and Standards Applied in the UK HPR1000

The principles applied to the UK HPR1000 are developed from HPR1000 (FCG3) and also reflect UK context.

The IAEA safety standards and WENRA safety reports that are applied to the design of HPR1000 (FCG3) are the sources of RGP.

To suit UK context, *Health and Safety at Work etc. Act 1974*, Reference [3], *Tolerability of Risk from Nuclear Power Stations*, Reference [10], *Reducing Risks,*

Protecting People, Reference [11], *Safety Assessment Principles for Nuclear Facilities*, Reference [4], TAGs and other sources of RGPs, such as *Nuclear Power Plants - Instrumentation and Control Important to Safety - Classification of Instrumentation and Control Functions*, Reference [12], *Design and Construction Rules for Electrical and I&C Systems and Equipment*, Reference [13], are considered in the development of the principles of the UK HPR1000.

The main legislation, codes and standards applied in the chapter are listed in T-4.3.2-1.

T-4.3.2-1 Main Legislation, Codes and Standards Applied in Chapter 4

No.	Codes and Standards
1	IAEA, Fundamental Safety Principles, No.SF-1, November 2006.
2	IAEA, Safety of Nuclear Power Plants: Design, No.SSR-2/1, Revision 1, February 2016.
3	IAEA, Deterministic Safety Analysis for Nuclear Power Plants, No.SSG-2, 2009.
4	IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, No.SSG-30, May 2014.
5	IAEA, Application of the Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA-TECDOC-1787, April 2016.
6	IAEA, Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants, IAEA-TECDOC-1791, May 2016.
7	IAEA, Assessment of Defence in Depth for Nuclear Power Plants, Safety Report Series No. 46, February 2005.
8	IAEA, Storage of Spent Nuclear Fuel, No. SSG-15, February 2012.
9	WENRA, Safety of New NPP Designs, March 2013.
10	WENRA, Safety Reference Levels for Existing Reactors, September 2014.
11	The Stationery Office, Health and Safety at Work etc. Act 1974, 1974
12	Health and Safety Executive, The Tolerability of Risk from Nuclear Power Stations, 1992.
13	Health and Safety Executive, Reducing Risks, Protecting People, 2001.
14	IEC, Nuclear Power Plants - Instrumentation and Control Important to Safety –Classification of Instrumentation and Control Functions, IEC 61226, Edition 4.0, September 2020.
15	IEC/ Institute of Electrical and Electronics Engineers (IEEE), Nuclear Facilities – Electrical Equipment to Safety - Qualification, IEC/ IEEE 60780-323, 2016.
16	French Association for Design, Construction and In-Service Inspection Rules for Nuclear Steam Supply System Components, Design and Construction Rules for Electrical and I&C System and Equipment, RCC-E, December 2016.

4.4 General Safety and Design Principles

4.4.1 Nuclear Safety Objective

The nuclear safety objective in the UK HPR1000 is as follows, which is also defined as Claim 3 in PCSR Chapter 1:

The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is as low as reasonably practicable.

4.4.1.1 Fundamental Safety Objective

In accordance with the *Fundamental Safety Principles* of IAEA, Reference [5], the fundamental safety objective of the UK HPR1000 is to protect people and the environment from harmful effects of ionising radiation. The nuclear safety objective is consistent with the fundamental safety objective of the UK HPR1000 and based on the requirements of the *Health and Safety at Work etc. Act 1974*, Reference [3].

4.4.1.2 As Low as Reasonably Practicable

Based upon the requirements of the *Health and Safety at Work etc. Act 1974*, Reference [3], as expressed in *Reducing Risks, Protecting People*, Reference [11] and the nuclear industry specific application in *Tolerability of Risk from Nuclear Power Stations*, Reference [10], it is necessary to show that the nuclear safety risks to the workers and the public are ALARP. This requires that all reasonable measures are taken in the design, construction and operation of the plant to minimise the radiation dose received by workers and public, unless the implementation of such measures would involve disproportionate effort.

The UK HPR1000 ALARP methodology and evaluation are provided in Chapter 33.

4.4.1.3 Radiation Safety Requirements

In order to achieve the nuclear safety objective and limit risks of radiological hazards to an ALARP level, radiation safety requirements are defined for the UK HPR1000 through dose and frequency targets for normal operation, fault and accident condition, along with their corresponding legal limits. These are provided in the *Safety Assessment Principles for Nuclear Facilities*, Reference [4].

Each target is set in terms of a Basic Safety Level (BSL) and a Basic Safety Objective (BSO). The BSLs will be met in the first place, especially the mandatory legal limits. The region under BSO represents the broadly acceptable region.

The UK HPR1000 is provided with RPTs corresponding to the Numerical Targets set in *Safety Assessment Principles for Nuclear Facilities*, Reference [4], including radiation protection targets for normal operation and fault and accident conditions.

These targets are used in the assessment to demonstrate that the plant adequately controls radiological hazards and that risks are reduced to an ALARP level.

4.4.1.3.1 Radiation Protection Targets for Normal Operation

a) On-site Radiation Protection Targets for Normal Operation

RPT 1: the targets and a legal limit for effective dose in a calendar year for any person on the site from sources of ionising radiation are:

1) Employees working with ionising radiation:

BSL: 20 mSv;

BSO: 1 mSv.

2) Other employees on the site:

BSL: 2 mSv;

BSO: 0.1 mSv.

RPT 2: the targets for average effective dose in a calendar year to defined groups of employees working with ionising radiation are:

BSL: 10 mSv;

BSO: 0.5 mSv.

b) Off-site Radiation Protection Targets for Normal Operation

RPT 3: the target and a legal limit for effective dose in a calendar year for any person off the site from sources of ionising radiation originating on the site are:

BSL: 1 mSv;

BSO: 0.02 mSv.

4.4.1.3.2 Radiation Protection Targets for Fault and Accident Conditions

a) On-site Radiation Protection for Fault and Accident Conditions

RPT 4 on site: the targets for the effective dose received by any person on the site arising from a design basis fault sequence are:

BSL:

20 mSv for initiating fault frequencies exceeding 1×10^{-3} pa;

200 mSv for initiating fault frequencies between 1×10^{-3} pa and 1×10^{-4} pa;

500 mSv for initiating fault frequencies between 1×10^{-4} pa and 1×10^{-5} pa.

BSO: 0.1 mSv.

RPT 5: the targets for the individual risk of death to a person on the site, from accidents at the site resulting in exposure to ionising radiation, are:

BSL: 1×10^{-4} pa;

BSO: 1×10^{-6} pa.

RPT 6: the targets for the predicted frequency of any single accident in the facility, which could give doses to a person on the site, are as presented in T-4.4.1-1.

T-4.4.1-1 Frequency Dose Targets for Any Single Accident - Any Person on the Site

Effective Dose(mSv)	Predicted Frequency Per Annum	
	BSL	BSO
2~20	1×10^{-1}	1×10^{-3}
20~200	1×10^{-2}	1×10^{-4}
200~2000	1×10^{-3}	1×10^{-5}
> 2000	1×10^{-4}	1×10^{-6}

b) Off-site Radiation Protection Targets for Fault and Accident Conditions

RPT 4 off site: the targets for the effective dose received by any person off the site arising from a design basis fault sequence are:

BSL:

1 mSv for initiating fault frequencies exceeding 1×10^{-3} pa;

10 mSv for initiating fault frequencies between 1×10^{-3} pa and 1×10^{-4} pa;

100 mSv for initiating fault frequencies between 1×10^{-4} pa and 1×10^{-5} pa.

BSO: 0.01 mSv.

RPT 7: the targets for the individual risk of death to a person off the site, from accidents at the site resulting in exposure to ionising radiation, are:

BSL: 1×10^{-4} pa;

BSO: 1×10^{-6} pa.

RPT 8: the targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site, are as presented in T-4.4.1-2.

T-4.4.1-2 Frequency Dose Targets for Accidents on an Individual Facility - Any Person off the Site

Effective Dose(mSv)	Total Predicted Frequency Per Annum	
	BSL	BSO
0.1~1	1	1×10^{-2}

Effective Dose(mSv)	Total Predicted Frequency Per Annum	
	BSL	BSO
1~10	1×10^{-1}	1×10^{-3}
10~100	1×10^{-2}	1×10^{-4}
100~1000	1×10^{-3}	1×10^{-5}
>1000	1×10^{-4}	1×10^{-6}

c) Overall Radiation Protection Target for Fault and Accident Conditions

RPT 9: the targets for the total risk of 100 or more fatalities, either immediate or eventual, from accidents at the site resulting in exposure to ionising radiation, are:

BSL: 1×10^{-5} pa;

BSO: 1×10^{-7} pa.

4.4.2 The Concept of Defence in Depth

The concept of DiD applied in the design of the UK HPR1000 is described as five levels based on *Safety Assessment Principles for Nuclear Facilities*, Reference [4] with consideration of UK context:

a) The purpose of the first level is to prevent abnormal operation and failures by design. The barrier for this purpose is conservative design, construction, maintenance and operation in accordance with appropriate safety margins, engineering practices and quality levels;

b) The purpose of the second level is to prevent and control abnormal operation and detection of failures. This second level of defence necessitates control, indication, alarm systems or other systems and operating procedures to prevent or minimise damage from failures;

c) The purpose of the third level is to control faults within the design basis to protect against escalation to an accident. This level requires engineered safety features, multiple barriers and accident or fault control procedures;

d) The purpose of the fourth level is to control severe plant conditions in which the design basis may be exceeded, including protecting against further fault escalation and mitigation of the consequences of severe accidents. This is achieved by additional measures and procedures to protect against or mitigate fault progression and for accident management;

e) The purpose of the fifth level is to mitigate radiological consequences of significant releases of radioactive material. This requires the provision of emergency control and on-site and off-site emergency response.

To implement the concept of DiD, successive physical barriers are in place to confine radioactive material at specified locations. For example, three barriers against the

escape of radioactivity from the core are as follows:

a) Barrier 1: Fuel cladding;

b) Barrier 2: Reactor coolant pressure boundary;

c) Barrier 3: Containment building.

According to *Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants*, Reference [14], the concept of DiD for the UK HPR1000 is considered not only as the requirement to achieve a number of consecutive barriers and levels of protection, but also as a general principle for the formulation of safety requirements, including quality and reliability of barriers and systems to ensure their capability.

As far as reasonably practicable, independence between different levels of DiD are also required. This means that if one level were to fail, the fault sequence can be terminated by a subsequent level. The independence between levels to be provided in the UK HPR1000 design is as follows:

a) To the extent practicable, SSCs used in Level 1 and Level 2 of DiD are independent from SSCs designed for the protection in Level 3;

b) To the extent practicable, SSCs designed for Level 3 are independent from SSCs for Level 4;

c) To the extent practicable, SSCs specifically designed to mitigate consequences of severe accidents in Level 4 of DiD are independent from other levels of DiD.

The means to achieve independence between SSCs are described in Sub-chapter 4.4.6.

4.4.3 Safety Analysis

To provide assurance that DiD has been appropriately implemented in the design of the plant, safety analysis is carried out using a number of complementary techniques. This supports the demonstration that the level of nuclear safety risk meets the nuclear safety objective.

The safety analysis for the UK HPR1000 is based on the IAEA approach in *Safety of Nuclear Power Plants: Design*, Reference [6], and uses the Deterministic Safety Analysis (DSA) methodology combined with PSA.

4.4.3.1 The Scope of Analysis

In the design of the plant, it is recognised that challenges to all levels of DiD can occur and design measures need to be provided to ensure that the necessary safety functions are accomplished and the safety objectives are met. These challenges stem from the Postulated Initiating Events (PIEs).

The PIEs considered include all foreseeable failures of SSCs of the plant, as well as

operating errors and possible failures arising from internal and external hazards, whether in full power, low power or shutdown states, in the reactor, the spent fuel pool or other systems containing radioactive or fissile material.

Besides the occurrence of PIEs, there may be consequential effects caused by the PIE which are considered as a part of the original PIE. The combinations of randomly occurring individual events are evaluated using appropriate screening criteria to ensure that credible coincident events are considered in the design.

Where PIEs are excluded from the deterministic analysis, and in some cases from the probabilistic analysis (being beyond design basis or not credible), a comprehensive technically supported justification shall be provided.

The identified PIEs are common inputs for both DSA and PSA. Further detail about the methodology of PIE identification is provided in *Methodology of PIE Identification*, Reference [15].

4.4.3.2 Deterministic Safety Analysis

The plant states considered in the design of the UK HPR1000 are shown in T-4.4.3-1. The plant states are identified and grouped into a limited number of categories primarily on the basis of their frequency of occurrence and consequence.

T-4.4.3-1 Plant States

Operational States		Accident Conditions		
Normal operation	Anticipated Operational Occurrences (AOOs)	Design basis accidents	Design Extension Conditions	
			Without significant fuel degradation	With core melting (severe accidents)

The plant states can be subdivided into 4 categories of DBC and 2 categories of DEC. The DSA is carried out with the analysis of DBC and DEC.

a) Design Basis Condition

The DBC is classified into 4 categories according to its frequency and good practice from other projects or standards: DBC-1, DBC-2, DBC-3, and DBC-4.

Based on the UK practice, they can also be classified into frequent faults and infrequent faults.

b) Design Extension Condition

The DEC is classified into 2 categories:

1) Design Extension Condition A (DEC-A): DEC events without significant fuel degradation;

2) Design Extension Condition B (DEC-B): DEC events with core melting.

Details about the plant states for the UK HPR1000 are provided in *The Design Condition List and Acceptance Criteria*, Reference [16].

The plant states including DBCs and DECs are considered within the concept of DiD. T-4.4.3-2 provides a comparison between the IAEA, the UK HPR1000 and UK context terminologies to better understand the relationship between the plant states of the UK HPR1000 and the level of DiD.

T-4.4.3-2 Overview of UK Context, the UK HPR1000 and IAEA Terminology

IAEA terminology			The UK HPR1000 Plant State	Frequency (pa)[1]	UK Context Terminology (based on *Safety Assessment Principles for Nuclear Facilities*, Reference [4])		
DiD Level		Plant State			Plant State	DiD Level (used in the UK HPR1000)	
1	Prevention of abnormal operation and failures	Normal Operation	DBC-1	IFF ≥ 1	Normal operation	Prevention of abnormal operation and failures by design	1
2	Control of abnormal operation and detection of failures	Anticipated Operational Occurrences	DBC-2[2]	1> IFF >10^{-2}		Prevention and control of abnormal operation and detection of failures	2
3a	Control of design basis accidents	Design Basis Accidents	DBC-3[3]	10^{-2}> IFF >10^{-3}	Frequent faults	Control of faults within the design basis to protect against escalation to an accident	3
				10^{-3}> IFF >10^{-4}	Infrequent faults		
			DBC-4	10^{-4}> IFF >10^{-5}			
3b	Control of design extension conditions to prevent core melt	Design Extension Conditions	DEC-A[4]	IFF > 10^{-3} and FSF > 10^{-7}	Frequent fault + failure of primary line of protection		
				IFF < 10^{-5}	Beyond design basis conditions (including severe accident)	Control of severe plant conditions in which the design basis may be exceeded, including protecting against further fault escalation and mitigation of the consequences of severe accidents	4
4	Control of design extension conditions to mitigate the consequences of severe accidents		DEC-B	---			
5	Mitigation of radiological consequences of significant releases of radioactive materials	Accident with releases requiring implementation of emergency countermeasures	N/A	---	Off-site emergency	Mitigation of radiological consequences of significant releases of radioactive material	5

Notes:

1. IFF: Initiating Fault Frequency, FSF: Fault Sequence Frequency;
2. DBC-2 corresponds to abnormal operation before a fault (UK context level 2 of DiD), and to frequent faults where there is a demand on a safety system (UK context Level 3 of DiD);
3. DBC-3 corresponds to faults at the lower end of frequent fault range, and at the higher end of infrequent fault range;
4. Although DEC-A is considered as design extension condition under IAEA terminology, for the UK HPR1000 it includes the scenario of a diverse line of protection against frequent faults and is therefore within the UK context design basis when used for this purpose. DEC-A may also be used to represent conditions beyond the UK context design basis, but before DEC-B (severe accident).

Appropriate assumptions are considered in the DSA to ensure the adequacy of safety measures. Further detailed information about DSA is provided in PCSR Chapters 12 and 13:

a) The analysis of DBC-2, DBC-3, DBC-4 applies a conservative methodology at an appropriately (high) confidence level. The analysis assumptions for DBC-2, DBC-3, DBC-4, based on *Deterministic Safety Analysis for Nuclear Power Plants*, Reference [17], are as follows:

   1) For the purpose of conservative calculations, the initial and boundary conditions are set to values that will lead to conservative results for those safety parameters that are to be compared with the acceptance criteria;

   2) The single failure criterion is applied when determining the availability of systems and components. A failure is assumed in the system or component that would have the largest negative effect on the calculated safety parameter;

   3) All the common cause and consequential failures associated with the PIE are included in the analysis, in addition to the single failure;

   4) Equipment that is not qualified for specific accident conditions is assumed to fail unless its continued operation results in more unfavourable conditions;

   5) The time interval between detection of the abnormal event or accident and the required action is sufficiently long, and adequate procedures (such as administrative, operational and emergency procedures) are specified to ensure the performance of such actions;

   6) Consequential Loss of Offsite Power (LOOP) due to turbine trip shall be considered for DBC-2, DBC-3 and DBC-4 accidents at power if it is conservative;

   7) It is assumed that the components are not available during preventive maintenance. The preventive maintenance activity can be carried out only when the safety function's availability and redundancy requirements in the DBC-2/3/4 analysis are satisfied.

   Further explanation of items 1 to 6 is presented in PCSR Chapter 12. Further explanation of maintenance states and maintenance philosophy versus operating mode is presented in PCSR Chapter 31.

b) The analysis assumptions for DEC-A are as follows:

   1) Key initial parameters, main system parameters and time delay are considered with conservative assumptions;

   2) Single failure is not considered;

3) Equipment that is not qualified for specific accident conditions is assumed to fail unless its continued operation results in more unfavourable conditions;

4) The time interval between detection of the abnormal event or accident and the required action is sufficiently long, and adequate procedures (such as administrative, operational and emergency procedures) are specified to ensure the performance of such actions;

5) LOOP is not postulated if it is not considered in the failure combination defined by the DEC-A sequence;

6) The unavailability of a system or component due to preventive maintenance is not considered.

Further explanation of items 1 to 4 is presented in PCSR Chapter 13. Further explanation of maintenance states, and maintenance philosophy versus operating mode, is presented in PCSR Chapter 31.

c) Safety Criteria

The safety criteria are defined to ensure that the nuclear safety objective is met. The safety criteria are presented in Sub-chapter 4.4.1.2, and the decoupling criteria are presented in Chapter 12 and 13.

4.4.3.3 Probabilistic Safety Assessment

PSA studies are used to demonstrate that a balanced design of the UK HPR1000 has been achieved such that no particular feature or PIE makes a disproportionately large or significantly uncertain contribution to the overall risks, and that, to the extent practicable, the levels of DiD are independent. It provides assurance that small deviations in plant parameters that could give rise to large variations in plant conditions (cliff edge effects) will be prevented, and compares the results of the analysis with the acceptance criteria for risk where these have been specified, *Safety of Nuclear Power Plants: Design*, Reference [6].

The UK HPR1000 PSA covers:

a) All sources of radioactivity present at the facility, including the reactor core, spent fuel, radioactive wastes and new fuel. The PSA for the reactor core is presented in Sub-chapters 14.4, 14.5 and 14.8. The PSA for the Spent Fuel Pool (SFP) is presented in Sub-chapters 14.6 and 14.8. The PSA for radioactive wastes is presented in Sub-chapter 14.7. The analysis for new fuel is included in the SFP PSA (which is presented in Sub-chapter 14.6 and 14.8.);

b) All types of initiating events (consideration of internal events, internal hazards and external hazards). The initiating events for internal events Level 1 PSA are presented in Sub-chapter 14.4.1. Internal hazards and external hazards Level 1 PSA is presented in Sub-chapter 14.4.2;

c) All plant operating states for the reactor core and SFP. The operating states for the reactor core are presented in Sub-chapter 14.4.1, and the operating states for SFP are presented in Sub-chapter 14.6.5;

d) Identification of the sequence of events that can lead to core damage and estimation of the core damage (Level 1 PSA), identification of ways in which radioactive releases from the plant can occur and estimation of their magnitude and frequency (Level 2 PSA), and estimation of public health and other societal risks (Level 3 PSA), more details are presented in Sub-chapters 14.4, 14.5 and 14.8.

PSA is applied to demonstrate that the nuclear safety objective can be met, including RPT 5-9, and to justify that the design of the nuclear power plant is balanced. The PSA provides an important insight into the UK HPR1000 design and supports decision-making during the design of the UK HPR1000. During the GDA process, the development of the PSA enables the identification of the potential design vulnerabilities that contribute most significantly to the risk. PSA is also used to analyse the risk impact associated with potential design improvements, and to understand the main contributors. The detailed PSA approach is provided in PCSR Chapter 14.

4.4.4 Identification of Safety Functions

The three fundamental safety functions which are defined in IAEA No. SSR-2/1 (Rev.1) *Safety of Nuclear Power Plants: Design*, Reference [6], are fulfilled for all plant states in the UK HPR1000. They are top-level safety functions for the UK HPR1000.

To facilitate the development of the UK HPR1000 design, the three fundamental safety functions are decomposed into several levels of functions. These are specific to the design and the assessment of the plant. The decomposition of safety functions follows a systematic and comprehensive methodology and process that ensures the sufficiency and completeness of the defined safety functions.

4.4.4.1 Fundamental Safety Functions and Decomposition of Safety Functions

The fundamental safety functions for the NPP are based on good practice in *Safety of Nuclear Power Plants: Design*, Reference [6] and described as follows:

a) R: Control of reactivity (including prevention of accidental criticality);

b) H: Removal of heat from the reactor and from the fuel store;

c) C: Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.

In the UK HPR1000 design, there is an additional type of function named “Extra

Safety Functions", which consists of supporting functions and hazard prevention, protection and mitigation functions.

These fundamental safety functions are further decomposed into high level safety functions. This is based on *Assessment of Defence in Depth for Nuclear Power Plants*, Reference [18], *Storage of Spent Nuclear Fuel*, Reference [19], and *Application of the Safety Classification of Structures, Systems and Components in Nuclear Power Plants*, Reference [20], and also takes into account the UK HPR1000 design features. The high level safety functions are:

a) Control of reactivity

   1) R1: Maintain core reactivity control;

   2) R2: Shutdown and maintain core sub-criticality;

   3) R3: Prevention of uncontrolled positive reactivity insertion into the core;

   4) R4: Maintain sufficient sub-criticality of fissile material stored outside the reactor coolant system but within the site.

b) Removal of heat

   1) H1: Maintain sufficient Reactor Coolant System water inventory for core cooling;

   2) H2: Remove heat from the core to the reactor coolant;

   3) H3: Transfer heat from the reactor coolant to the ultimate heat sink;

   5) H4: Maintain heat removal from fuel stored outside the RCS but within the site.

c) Confinement

   1) C1: Maintain integrity of the fuel cladding to ensure confinement of radioactive material;

   2) C2: Maintain integrity of the Reactor Coolant Pressure Boundary to ensure confinement of radioactive material;

   3) C3: Maintain integrity of reactor containment to ensure confinement of radioactive material;

   4) C4: Maintain integrity of the fuel stored outside of reactor containment;

   5) C5: Store the radioactive material;

   6) C6: Shield against radiation, control planned radioactive releases, and limit accidental radioactive releases.

d) Extra safety functions

1) E1: Support Type R, H or C safety function;

2) E2: Prevent, protect and mitigate hazards impact.

The High Level Safety Functions are developed more specifically into low level safety functions, and then applied in safety system design.

Further information about the safety functions are presented in *Decomposition of Safety Functions*, Reference [21].

4.4.4.2 Application of Safety Functions to Safety Measures

The decomposition of high level safety functions to low level safety functions takes account of the plant design and operating modes. The low level safety functions identified are allocated to the technical means or measures for the purpose of achieving the functional requirements.

Each high level function (R, H, C or E) is decomposed into low level safety functions or a set of functions which are required for the high level function to be achieved in that particular circumstance. The SSCs providing a low level safety function are then identified. It is important to note that:

a) A single system may support more than one low level function, for different faults;

b) A single low level function may be supported by multiple systems;

c) Supporting systems (essential services such as component cooling, power supply, air supply, etc.) that are necessary for the delivery of a function by the claimed systems are identified;

d) Supporting indications that are necessary or complementary for the delivery of a function by the claimed systems are identified, depending on whether the systems are manually or automatically initiated. Means of monitoring the status of the plant shall be provided for ensuring that the required safety functions are fulfilled.

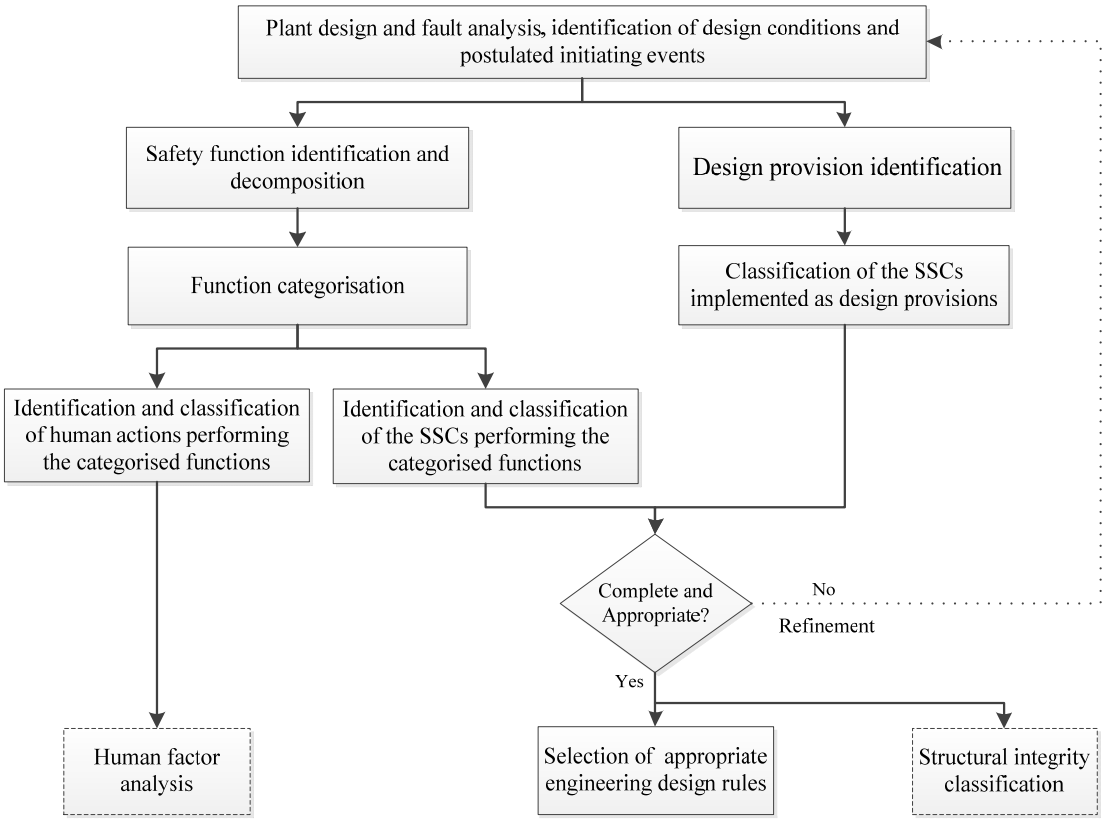
The safety requirements against which a system is to be substantiated (see Sub-chapter 4.4.6) are derived based on the low level safety function(s) to be provided in order to remain within the appropriate DiD levels (normal operating or design basis limit), and on the conditions under which the functions are to be provided, at the appropriate reliability.

4.4.5 Categorisation of Safety Functions and Classification of Structures, Systems and Components

Safety classification of SSCs provides a link between the significance of a safety function and the level of reliability and quality of the SSCs that are needed to perform the safety function.

The UK HPR1000 safety categorisation and classification approach is based on *Safety Classification of Structures, Systems and Components in Nuclear Power Plants*, Reference [22]. The following relevant technical documents: *Application of the Safety Classification of Structures, Systems and Components in Nuclear Power Plants*, Reference [20]; *Categorisation of Safety Functions and Classification of Structures, Systems and Components*, Reference [23]; and *Nuclear Power Plants - Instrumentation and Control Important to Safety - Classification of Instrumentation and Control Functions*, Reference [12], are also taken into account.

Classification is a top-down process that is based on an understanding of the plant design, fault analysis and how to achieve the fundamental safety functions. This process is illustrated in F-4.4.5-1.



F-4.4.5-1 Safety Categorisation and Classification Process Overview

4.4.5.1 Categorisation of Safety Functions

The classification of SSCs is based on their role in achieving a safety function. The categorisation of safety functions considers the following:

a) The consequences of failure to perform the safety function;

b) The frequency of the occurrence of the PIEs for which the function will be called upon;

c) The states of plant that the function contributes to be achieved after an initiating

event.

The potential radiological consequences of failure are graded as High, Medium or Low as an input to the categorisation and classification process. The rating of High, Medium or Low is directly related to safety category. For the UK HPR1000, the High, Medium and Low consequences are defined as follows:

a) For off-site releases:

   1) High consequences are defined as a release of radioactive material that exceeds the BSL for off-site radiation protection targets for fault and accident conditions (corresponding to RPT 4) presented in Sub-chapter 4.4.1.3.2;

   2) Medium consequences are defined as a release of radioactive material that exceeds the limits for normal operation for faults with initiating fault frequency greater than 10^{-4} pa, or exceeding 1mSv for faults with an initiating fault frequency less than 10^{-4} pa;

   3) Low consequences are defined as a release of radioactive material that exceeds the BSO, but are below limits of normal operation for faults with an initiating fault frequency greater than 10^{-4} pa, or below 1 mSv for faults with an initiating fault frequency less than 10^{-4} pa;

b) For on-site releases:

   1) High consequences are defined as a dose to an individual on site exceeding 200mSv for faults with an initiating fault frequency $\geq 10^{-3}$ pa (frequent faults), or exceeding 500 mSv for faults $< 10^{-3}$ pa (infrequent faults);

   2) Medium consequences are defined as a dose to an individual on site exceeding 2 mSv for faults with an initiating fault frequency greater than 10^{-4} pa, or exceeding 20 mSv for faults with an initiating fault frequency less than 10^{-4} pa;

   3) Low consequences are defined as a dose to an individual on site exceeding the BSO but below 2 mSv for faults with an initiating fault frequency greater than 10^{-4} pa, or below 20 mSv for faults with an initiating fault frequency less than 10^{-4} pa.

The contribution that the functions make in reaching the states of plant after an initiating event is considered in the categorisation of safety functions. For DBCs, the states of plant include controlled state and safe state. For DECs, there is no distinction made between the controlled and safe states, hence the designation of final state is used.

The definitions of controlled state and safe state are given in T-4.4.5-1.

T-4.4.5-1 Characteristics of Controlled State and Safe State

	For Reactor Core	For Spent Fuel
Controlled State	• The core is subcritical • Heat removal is assured for a time sufficient to implement provisions to reach a safe state, e.g. via steam generators • Coolant inventory is stable • Radioactive release meets regulatory limits	• Fuel storage is subcritical • Water level is stabilised and the water inventory is adequate for spent fuel cooling and shielding • Radioactive release meets regulatory limits
Safe State	• The core is subcritical • Residual heat removal is assured on a long term basis, e.g. via residual heat removal heat exchanger • Reactor coolant inventory is recovered • Radioactive releases are controlled and acceptable	• Fuel storage is subcritical • Water level is stabilised and the water inventory is adequate for spent fuel cooling and shielding • Spent fuel pool cooling is established on a long term basis • Radioactive releases are controlled and acceptable

Therefore, a controlled state is considered to be a stable and safe state in the analysis of the reactor as the key safety significant parameters identified above will be shown to be stable. A safe state is defined as maintaining a stable, safe state. The controlled state and safe state of DBC is also applied to the analysis of diverse line. For DECs the final state defined as maintaining a stable and safe state.

The initial categorisation of functions under different plant states in the UK HPR1000 nuclear power plant is shown in T-4.4.5-2 for Level 1 DiD, T-4.4.5-3 for Level 2 DiD, and T-4.4.5-4 for Level 3, 4 and 5 DiD. Note that the UK HPR1000 uses the terminology FC1, 2, 3 and NC to indicate Cat1, 2, 3 and Non-Categorised.

T-4.4.5-2 Function Categorisation for Level 1 of DiD

DiD Level	Context of Function	Severity of the Consequences if the Function is not Performed		
		High	Medium	Low
Consequences are assessed on the basis that Level 3 of DiD function is performed, i.e. mitigated, protected consequences		DBC-1 Function Category		

DiD Level	Context of Function	Severity of the Consequences if the Function is not Performed		
		High	Medium	Low
Level 1	Function required during DBC-1 to control a duty system, with potential to cause a fault if it fails	1[1]	2	3

Notes:

1. High mitigated, protected consequences implies the associated Level 3 of DiD function is ineffective, absent or bypassed by the fault, all of which are undesirable (e.g. some fuel handling faults). In such a case, it should be shown that functions provided in Levels 1 and 2 of DiD can deliver an equivalent level of safety.

T-4.4.5-3 Function Categorisation for Level 2 of DiD

DiD Level	Context of Function		Function Category
Level 2[1]	Independent function which acts to prevent a fault. Provided to complement the DBC-1 function and allow its Category to be reduced by one level (Category 2 to Category 3 or Category 3 to NC)		3
	Independent function which acts to prevent a fault. Claimed to reduce fault frequency from frequent to infrequent region	Category 2 DBC-1 function or initiating event frequency (assuming failure of Level 1 DiD function) less than 10^{-2}pa	3
		Category 3 DBC-1 function or initiating event frequency (assuming failure of Level 1 DiD function) less than 10^{-1}pa	2

Note:

1 Level 2 DiD functions can either be used to complement Level 1 DiD, allowing balance in design between the integrity level of duty and preventative functions (row 1 in the Table above), or to reduce the demand on Level 3 of DiD, and hence balance the design between prevention and protection (rows 2 and 3 in the Table above). In some circumstances there may be no claimed Level 2 function.

T-4.4.5-4 Function Categorisation for Faults

DiD Level	Functions Credited in the Safety Assessment	Severity of the Consequences if the Function is not Performed		
		High	Medium	Low
Consequences are assessed on the basis that safety functions in later plant states are not performed, and that functions failed earlier in the fault sequence are not available i.e. unmitigated, unprotected consequences		DBC / DEC Function Category		
Level 3	Functions to reach a controlled state under DBC-2, 3 and 4 conditions	1	2	3
	Functions to reach and maintain a safe state under DBC-2, 3 and 4 conditions	2	3	3
	Functions providing a diverse backup to a Category 1 function in a frequent fault	2[1]	---	---
	Functions providing a diverse backup to a Category 2 function in a frequent fault	3	---	---
Level 4	DEC-A functions with initiating fault frequency < 10^{-5}pa	3	---	---
	Functions for the mitigation of severe accidents	3	---	---
Level 5	Any function relating to the monitoring needed to provide off-site emergency services, including monitoring and communication means as part of the emergency response plan	3	---	---

Note:

1. Functions that are used in multiple failure conditions but that are not required in the short term after the onset of the accident may be assigned to Category 3 function.

4.4.5.2 Classification of Systems, Structures, and Components

This sub-chapter defines the classification of SSCs providing safety functions, and SSCs that are treated as a design provision.

4.4.5.2.1 Classification of SSCs Providing a Categorised Function

Once the safety categorisation of the functions is completed, the SSCs performing these functions are assigned a safety class.

All SSCs that fulfil Category 1, 2 and 3 functions must be subject to SSCs classification. The assignment of SSCs classification is shown in T-4.4.5-5.

T-4.4.5-5 Safety Classification of SSCs according to Safety Function Category

Function Category	SSC Class
Category 1	Class 1
Category 2	Class 2
Category 3	Class 3

Note that the UK HPR1000 uses the terminology F-SC1, 2, 3 and NC to indicate Functional Class 1, 2, 3 and Non-Classified. According to *Categorisation of Safety Functions and Classification of Structures, Systems and Components*, Reference [23], the expected reliability of different system classes is shown in T-4.4.5-6.

T-4.4.5-6 Relationship between System Class and Reliability

Function Category	System Class	Failure Frequency / Year (ff) for Continuously-operating SSCs	Probability of Failure on Demand (pfd) for Demand-based Feature
Cat 1	Class 1	$10^{-3} \geq ff \geq 10^{-5}$	$10^{-3} \geq pfd \geq 10^{-5}$
Cat 2	Class 2	$10^{-2} \geq ff > 10^{-3}$	$10^{-2} \geq pfd > 10^{-3}$
Cat 3	Class 3	$10^{-1} \geq ff > 10^{-2}$	$10^{-1} \geq pfd > 10^{-2}$

If an SSC fulfils multiple functions, its classification depends on the function with the highest function categorisation.

Human actions that are required to perform or to support a safety function are classified in a similar way to SSCs.

4.4.5.2.2 Classification of Design Provisions

Categorisation of the functions provided by design provisions is not necessary because the safety significance of the SSC can be directly derived from the consequences of its failure. The consequences of failure are graded as High, Medium or Low, as defined in Sub-chapter 4.4.5.1. The SSCs implemented as design provisions can therefore be directly assigned to a safety class without the need for further analysis of the safety function categories.

Design provisions are classified directly based on their consequences of failure according to T-4.4.5-7 below.

T-4.4.5-7 Classification of Design Provisions

Context of Function	**Severity of the Consequences if the Design Provision Fails**		
	High	**Medium**	**Low**
Consequences of failure of design provisions during normal operation are assessed on the basis that safety functions in Level 3 of DiD are performed, i.e. mitigated, protected consequences			
Design provision whose failure could lead directly to radiological release during normal operation (DPH, DPM, DPL correspond to High, Medium and Low consequences)	Class 1	Class 2	Class 3
Consequences of failure of design provisions during faults are assessed without benefit from later levels of DiD. If the failure of the SSC also affects an active function then the function category are also considered.			
Design provision whose failure could lead to radiological release during a fault (DPA) due to loss of containment of radioactive material.	Class 1	Class 2	Class 3

Note that the UK HPR1000 uses the terminology B-SC1, 2, 3 and NC to indicate Design Provisions Class 1, 2, 3 and Non-Classified.

4.4.5.2.3 Classification of Structural Integrity

Based on safety classification, the structural integrity classification is also factored in to be consistent with the UK context. The structural integrity classification for the UK HPR1000 is presented in PCSR Chapter 17.

4.4.5.3 Design Requirements

Engineering design rules comprise the relevant national or international codes, standards and proven engineering practices that are applied, as appropriate, to the design of SSCs to meet the applicable design requirements.

A complete set of engineering design rules are specified to ensure that the SSCs can be designed, manufactured, constructed, installed, commissioned, operated, tested, inspected and maintained to appropriate quality standards. To achieve this, the design rules identify appropriate levels of capability, reliability (dependability) and robustness.

4.4.5.3.1 General Design Requirements

Practically, capability and reliability of systems performing a categorised function are achieved by meeting the design requirements that are relevant to the safety class of the system. The requirements comprise:

a) Single failure criterion;

b) Physical and electrical separation;

c) Emergency power supply;

d) Periodic test;

e) Environmental qualification;

f) Protection against internal and external hazards (including earthquakes).

The general design requirements for systems are shown in T-4.4.5-8.

T-4.4.5-8 System Design Requirements

System Classification	Single Failure Criterion [1]	Physical and Electrical Separation	Emergency Power Supply	Periodic Test [5]	Environmental Qualification [7]	Protection Against Internal and External Hazards (Including Earthquakes)
F-SC1	Yes[1] System level	Yes	Yes	Yes	Yes	Yes
F-SC2	Yes[2] Function level	Yes [3]	Yes	Yes	Yes	Yes
F-SC3	No	Special requirements [4]	Case by case	Yes [6]	Case by case	Case by case

Notes:

1. Consideration of single failure criterion at the "system level" for F-SC1 systems indicates that these systems must have sufficient redundancy such that they are tolerant to single failure;
2. Consideration of single failure criterion at the "function level" for F-SC2 systems indicates that these systems may not need redundancy. When a system is subject to non-redundant design, another system must fulfil the same function (functional diversity) and single failure evaluation on it must be performed. In this case, physical isolation requirements should be applied on multiple pipelines of different systems fulfilling the same function. An F-SC2 system is not required to meet single failure criterion if it performs a function that provide a backup to a Category 1 function;
3. Physical and electrical separation should be applied to redundant trains of a system or diverse systems that perform the same functions, so that failure of multiple lines due to common cause failure is prevented;
4. F-SC3 systems are required to be separated from F-SC1 and F-SC2 systems, according to analysis results for internal and external hazards;
5. Active safety systems must be designed so that periodic testing can be performed;
6. Except for the case where continuous operation is required;
7. Qualification requirements are determined according to the specific equipment operational environment. In particular, equipment performing Category 3 functions that addresses DECs should meet requirements for environmental radioactivity, pressure, temperature, etc. in the corresponding DECs.

4.4.5.3.2 Relationship between Classes and Design Codes and Standards

The high level principle for system design requirements is that Class 1 and Class 2 systems are designed in accordance with nuclear specific codes and standards where available. Class 3 systems may be designed in accordance with nuclear specific or industrial codes and standards.

The selection of specific codes and standards is carried out in accordance with relevant good practice for the given topic area. The principles relating to selection of applicable codes and standards is presented in Sub-chapter 4.4.7, and the detailed application of codes and standards is presented in relevant PCSR chapters.

4.4.5.3.3 Seismic Requirements

Two seismic categories are defined: Seismic category 1 (SSE1) and Seismic category 2 (SSE2). Non-seismic is designated as NO.

Generally, seismic requirements are divided as follows:

a) Operability (O): Ability of an active component (including all the necessary auxiliary systems, supporting systems and energy supply systems) to fulfil its safety functions and thus meet the safety objective;

b) Functional capacity (F): Ability of all parts of components (active and passive) to withstand the specified loadings such that deformations occurring in these components are limited and their safety function is not impaired;

c) Integrity (I): Ability of active and passive components to withstand the specified loadings at the given frequency of occurrence throughout the service life of the component;

d) Stability (S): Ability of an active or passive component to withstand loads that tend to change its orientation or position (e.g. causing it to sway, fall or slide unacceptably, or causing parts to shear). Component stability includes the necessary resistance and stability of its supports.

4.4.5.3.3.1 Seismic Category 1 (SSE1)

The seismic category of Class 1 and Class 2 items is SSE1. SSCs that are determined as Class 3 are not usually categorised as SSE1, unless specifically required by the safety analysis of the seismic event. Important examples of Class 3 items that are categorised as SSE1 include:

a) Class 3 systems required to allow the reactor or spent fuel to reach a safe state under DEC-A conditions;

b) Class 3 systems claimed for severe accident management (DEC-B);

c) Isolation, detection and fire-fighting systems in the buildings containing

mechanical, electrical or Instrumentation and Control (I&C) equipment that fulfil FC1 and FC2 functions;

d) Some systems with additional functions (FC3) that are essential to maintain a safe state but which may be required in the period between 24 and 72 hours;

e) Parts of the process information and control system.

For most SSE1 electrical and I&C components, "operability" during and after an earthquake is required. For SSE1 mechanical components, "functional capacity" or "operability" after an earthquake is required.

For any equipment actuated (such as certain valves, reactor automatic trip and shutdown) or in operation (such as F-SC1 system pumps) during an earthquake, "operability" during the earthquake is required.

4.4.5.3.3.2 Seismic category 2 (SSE2)

The seismic category of non-SSE1 SSCs is SSE2 if any internal hazards caused by failure of the SSCs in a seismic event resulting from an earthquake could cause an unacceptable effect on SSE1 SSCs. This could include the following consequential hazards.

a) Toppling or falling on to SSE1 equipment;

b) Generation of missiles;

c) Effects resulting from failure of high-energy equipment;

d) Flooding resulting from failure of fluid systems;

e) Explosion;

f) Fire.

If SSCs are claimed to limit the effect of a hazard caused by a seismic event, they must also be suitably categorised as SSE2 at least. Consideration is given to the possibility of multiple failures resulting from the effect of an earthquake in the analysis of earthquake induced accident consequences. Secondary failure resulting from failure of electrical and I&C equipment must be prevented by electrically isolating protected and unprotected systems.

More detailed information on safety categorisation and classification is presented in *Methodology of Safety Categorisation and Classification*, Reference [24].

4.4.6 Engineering Substantiation

The nuclear safety principles in Sub-chapter 4.4.1 to Sub-chapter 4.4.5 ensure that the fault analysis could identify safety measures and corresponding safety requirements suitable to meet the overall nuclear safety objective.

The safety requirements on SSCs are substantiated to satisfy the appropriate reliability in accordance with their classification requirements. The envelope defined by the aggregate safety requirements on a system represents the system design basis, aligned with the definition in *Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants*, Reference [14].

This sub-chapter provides the generic principles to be applied in substantiating that safety measures are able to meet the safety requirements placed on them (fulfilling the design basis), and to identify any operational and maintenance activities that will be required to underpin the engineering substantiation.

4.4.6.1 Design for Reliability

4.4.6.1.1 Application of the Hierarchy of Risk Reduction to the Design

The design of the plant and of individual safety measure reflects the preferences identified in the hierarchy of risk reduction, as far as reasonably practicable:

a) Inherent safety: The design avoids radiological hazards rather than controlling them. The potential for harm is eliminated by design;

b) Fault tolerance: A PIE produces no safety significant effects or produces only a change towards safe plant conditions by means of inherent characteristics of the plant;

c) Passive safety: Following a PIE, the plant should be rendered safe by means of passive safety features or by the action of systems that are operating continuously in a state necessary to control the PIE;

d) Engineered safety systems: Following a PIE the plant should be rendered safe by the actuation of safety systems that need to be brought into operation in response to the PIE. In general, automatically acting systems are preferred to those that require manual initiation;

e) Administrative procedures: Following a PIE, the plant should be rendered safe by specified procedures.

f) Mitigation safety measures: Filtration, scrubbing, etc.

4.4.6.1.2 Single Failure Criterion and Redundancy

The reliability of items important to safety is commensurate with their safety significance. According to *Safety of Nuclear Power Plants: Design*, Reference [6], the single failure criterion is a criterion (or requirement) applied to a system such that it must be capable of performing its task in the presence of any single failure.

The single failure includes active and passive failures:

a) An active single failure is defined as a failure which is sufficient to invalidate the

relevant safety function of a component, including the malfunction of a mechanical or electrical component which relies on mechanical movement to complete its intended function upon demand, and the malfunction of an I&C component;

b) A passive single failure is defined as a failure which could occur in a component that does not change its state while realising its function.

The single failure criterion is applied to each safety system considered in fault analysis. A single failure of an active component within systems that deliver FC1 or FC2 safety functions is required to be tolerated at or after the PIE, at the point when their action is demanded. A single failure of passive components within systems that deliver FC1 or FC2 safety functions needs to be assessed at the start of a transient in an appropriate means.

The following are considered in relation to the single failure criterion:

a) FC1 function, F-SC1 system: Provide redundancy within the system to ensure tolerance to a single failure;

b) FC2 function, F-SC2 system: Provide redundancy within the system, or at function level through multiple systems that are capable of providing the same function, to ensure tolerance to single failure;

c) FC3 function, F-SC3 system: Redundancy not required, although it may be necessary to achieve the reliability under the fault sequence in question.

It should be noted that the design of a F-SC2 system that belongs to a diverse protection line is not required to consider the single failure criterion.

In some cases, if one train of a safety system fails due to an initiating fault and the second train fails taking single failure criterion into account, then a third redundant train is needed to ensure that the required safety function is reliably performed.

The design of the UK HPR1000 will ensure reliability during maintenance by providing redundancy in the design and through management procedures. For preventive maintenance, the activities can be managed through adequate planning and compensatory measures in maintenance procedures. These activities are not undertaken when the function is needed in DBC fault sequence analyses. For corrective maintenance, it is managed by providing compensatory measures and recovery time and plant fall-back modes according to operating limits and conditions. If the duration is inadequate to recover the function, it is performed when the plant is under a condition where the function is not required.

4.4.6.1.3 Independence

In addition to the high level principle of independence between levels of DiD, the following principles for independence are applied in the design to improve system

reliability and to have tolerance to faults:

a) Independence between trains of redundant system components is maintained as far as reasonably practicable (avoidance of common cause failure);

b) Independence between components of different safety classes is maintained as far as reasonably practicable (avoidance of impact on the component of higher safety class from an item of lower safety class);

c) Independence between components designed to mitigate a potential initiating event and the effects of this potential initiating event is maintained as far as reasonably practicable;

d) Independence between SSCs important to safety and those not important to safety is maintained as far as reasonably practicable.

Independence is accomplished in the design of systems by using functional isolation and/or physical separation. Functional isolation is used to reduce adverse effects between elements of connected systems or systems redundantly designed. These adverse effects may be caused by the normal operation, abnormal operation or failure of any part of these systems.

Interference between protection systems and control systems are prevented by avoiding interconnection or by appropriate isolation measures. If signals are shared by a protection system or any control system, isolation measures will be ensured and it will be demonstrated that the design can still meet all the safety requirements of protection systems.

Physical separation is applied in the layout of systems as far as reasonably practicable, to reduce the potential of common cause failure due to a localised initiating event. The choice of isolation measures (compartmentalisation, distance, orientation etc.) considers the nature of the initiating events.

4.4.6.1.4 Diversity

The design of equipment considers the potential for common cause failures of items important to safety, to determine how the concepts of diversity have to be applied to achieve the necessary reliability.

Diversity is achieved by incorporating different attributes into redundant systems or components. Such attributes can be different operating principles, different physical variables, different operating conditions, different manufacturers, etc.

The concept of diversity is considered for safety functions which relate to the loss of the primary protection line. In the UK HPR1000 design, a diverse protection line is designed to protect or mitigate frequent faults.

The analysis rules applied for the diverse protection line are as follows:

a) Key initial parameters, main system parameters and time delay are considered with conservative assumptions;

b) Single failure criterion is not taken into account;

c) The unavailability of a system or component due to preventive maintenance is not considered;

d) LOOP is not postulated if not considered as PIE;

e) Equipment that is not qualified for specific accident conditions is assumed to fail unless its continued operation results in more unfavourable conditions;

f) The time interval between detection of the abnormal event or accident and the required action is sufficiently long, and adequate procedures (such as administrative, operational and emergency procedures) are specified to ensure the performance of such actions;

The Categorisation requirements for the diverse protection line of the UK HPR1000 are presented in T-4.4.6-1.

T-4.4.6-1 Categorisation of Diverse Protection Line

Primary Protection Line functions	Diverse Protection Line functions
FC1	FC2
FC2	FC3
FC3	---

4.4.6.1.5 Fail-safe Design

The fail-safe design is considered and incorporated, as appropriate, into the design of systems and components important to safety of the UK HPR1000, so that their failure or the failure of a support feature will not invalidate the performance of the intended safety function.

Spurious operation and unsafe failure modes of systems and components are considered in the design. Priority is given to predictable and revealed failure modes during equipment selection.

4.4.6.1.6 Human Factors

A systematic approach to human factor integration is established and applied throughout the entire lifecycle of the UK HPR1000, especially at the design stage.

Human factor integration covers the plant locations where operations and maintenance activities take place. The following elements are met:

a) The design allocates functions properly to minimise the dependence of safety

functions on human actions;

b) Human actions that could impact safety during normal operation, fault and accident conditions are identified systematically. These human actions important for safety are known as Human-Based Safety Claims (HBSCs);

c) Appropriate human factor analysis, including task analysis and human reliability analysis, is performed on the HBSCs to identify potential improvements to systems, procedures or training;

d) All HBSCs are classified either based on their risk significance or on the significance of the safety system affected;

e) The design supports personnel in the fulfilment of their responsibilities and in the performance of tasks by providing suitable and sufficient user interfaces and workspace.

More detailed information about human factors is presented in Chapter 15.

4.4.6.2 Design to Ensure Functionality

4.4.6.2.1 Equipment Qualification

The equipment qualification for UK HPR1000 includes environmental qualification, seismic qualification, and performance testing or analysis. The general requirements related to environmental qualification and seismic qualification are introduced in this section.

The purpose of equipment qualification is to ensure that the equipment is able to perform its intended safety functions under normal operation, accident conditions and seismic conditions. Equipment qualification simulates various environmental and seismic conditions that may occur during the lifetime of a nuclear power plant to verify the capability of equipment to perform its safety functions under these conditions.

The following types of equipment are considered in equipment qualification: mechanical equipment, electrical equipment, I&C equipment, Heating, Ventilation and Air Conditioning (HVAC) equipment, etc. Considering the results of fault analysis and the safety classifications, the specific equipment to be qualified is listed as following:

a) Equipment required for environmental qualification

   1) Equipment that perform FC1 or FC2 functions;

   2) Equipment that perform FC3 functions, which are required:

      - To maintain a safe state;

      - To protect against DEC.

All the normal operation, fault and accident conditions are considered in the equipment qualification. Normal operation conditions consider the lifetime of the equipment and the environment of the normal conditions in the building where the equipment is located. The variation of environmental conditions arising from the fault and accident conditions are also considered.

b) Equipment required for seismic qualification

The equipment that performs the following functions is required for seismic qualification: operability (O), functionality (F), integrity (I) or stability (S). The detailed explanation for seismic qualification is presented in Sub-chapter 4.4.5.3.3.

The parameters which are related to the environmental conditions and their impact on equipment are presented below:

a) Temperature

Temperature can indirectly change the performance of the equipment by gradual chemical and physical processes, which is also called thermal ageing.

b) Pressure

Pressure and its rapid changes can affect the performance of equipment by exerting additional forces on the equipment. High increase of external or internal pressure may cause structural failure of fully sealed equipment. The rapid increase of pressure may cause structural failure of the imperfectly sealed equipment.

c) Radiation

Radiation could induce changes in the atomic and molecular structure of matter through excitation, oxidation, crosslinking, degradation and shearing process, resulting in the change of equipment performance. Some changes improve the performance of the equipment, but most of the changes cause a decline in the performance.

Four main types of radiation (α, β, γ and neutron) occur within a nuclear power plant. The γ radiation possesses a very strong capacity for penetration. The penetration capacity of β radiation is a lot less, and 1mm steel or 10 mm water layer can shield most of the β radiation. The penetration capacity of α radiation is lower still. Neutron radiation is considered for equipment near the reactor pit.

d) Humidity

Humidity (high humidity) can directly lead to equipment performance degradation and can make other environmental conditions worse. For example, moisture could lead to corrosion and current effects at the interfaces of different

metals. Moisture could directly reduce the performance of organic materials, degrading their physical, mechanical and electrical performance and deforming them. Moisture on the surface of an insulator can significantly reduce its insulation resistance and breakdown voltage.

Methods of equipment qualification are presented below. They are determined accordance with *Design and Construction Rules for Electrical and I&C System and Equipment*, Reference [13], and *Nuclear Facilities - Electrical Equipment to Safety - Qualification*, Reference [25]:

a) Type test under representative conditions, in accordance with an appropriate test standard;

b) Qualification by analysis:

   1) Calculation (design analysis), usually structural load analysis and mechanical analysis in accordance with an appropriate design code;

   2) Operating experience;

   3) Analogy, by comparison with similar qualified equipment.

Considering the specific characteristics of the equipment to be qualified, the methods listed above can be used individually or in combination.

More detailed information on equipment qualification is presented in *Equipment Qualification Methodology*, Reference [26].

4.4.6.2.2 Ageing and Degradation

The design life of items important to safety is evaluated and defined. The power plant of the UK HPR1000 is designed for 60 years. The reactor vessel and all major structures or equipment which are not replaceable, such as the containment, have a design lifetime of 60 years. Some devices are designed for 60 years, though they are replaceable. For SSCs which are not designed for 60 years, the replaceability of these SSCs is considered in the design.

The issues caused by obsolescence of SSCs, especially systems and components, are also considered. Appropriate margins are provided in the design to take due account of relevant mechanisms of ageing, neutron embrittlement, wear out, and of the potential for age related degradation. This ensures that items important to safety are capable of performing their necessary safety functions throughout their design life.

Periodic measurement of material properties and parameters that could change with time is made to assess ageing mechanisms predicted at the design stage and to help to identify unanticipated behaviour of the plant or degradation that might occur in service.

Detailed information relating to the operational management of ageing and

degradation is addressed in Sub-chapter 31.6 and the activities of ageing and degradation are presented in relevant system chapters.

4.4.6.2.3 Examination, Maintenance, Inspection and Testing

An effective Examination, Maintenance, Inspection and Testing (EMIT) is essential for the safe operation of the plant. They ensure that the levels of reliability and availability of all SSCs important to safety remain in accordance with the design over the lifetime of the plant.

The design of SSCs is such to facilitate EMIT activities with the objective to maintain the capability of SSCs important to safety, and therefore to satisfy their reliability requirement. The SSCs are designed to ensure that they are maintainable, inspectable, and testable. The key aspects of EMIT to be considered for the UK HPR1000 are as follows:

a) EMIT activities should be identified according to design intent, regulatory requirements and potential degradation mechanisms and industry practices.

b) The frequency of EMIT activities, which are carried out at predetermined intervals, should be determined to ensure the probability of failure or the degradation of SSCs are appropriately controlled depending on their safety function.

c) Type testing and factory acceptance testing should be considered as reference data to establish a baseline for EMIT.

d) EMIT activities should not unacceptably degrade the continuing validity of equipment qualification of SSCs.

e) Procedures for EMIT should be established that include all of the required maintenance activities, commissioning tests, periodic tests, and inspection activities.

f) Engineered measures should be provided to ensure the design is maintainable, inspectable, and testable. The key engineered measures are as follows:

    1) The plant layout where items important to safety are installed should have adequate space to prevent crowding and be designed for accessibility during EMIT activities. Where the replacement of a component may be needed, the plant layout design should consider the transportation and storage of special tools and of the new component.

    2) Means of failure detection and isolation of SSCs should be provided to enable EMIT activities to be performed.

    3) Means of radiation protection should be provided in the design to ensure that the risk associated with EMIT has been reduced to be ALARP.

4) Environmental conditions are very important to EMIT activities during the service life. Therefore, means for surveillance of environment conditions should be provided in the design phase.

5) When the above engineered measures cannot be fully achieved, alternative provisions should be incorporated into the SSCs design or it should be demonstrated that long-term performance is achieved without additional measures being provided.

g) EMIT should be managed to avoid the unavailability of the safety functions fulfilled by the system being examined, maintained, inspected, or tested, except when the safety function is not required in the specific condition. In special circumstances where this cannot be met (e.g. the safety function is required in all operating modes), the risks/impacts on potential to lose/degrade the safety function should be assessed and justified to identify the EMIT implementation window for the safety function. In addition, EMIT of an SSC important to safety shall not initiate a fault sequence. Generally, functional testing of SSCs should be carried out under conditions similar to those of their intended safety functions.

   Practically, periodic testing should be defined in a condition where the function is required as far as possible. However, it is not always practical for the tests to be done in a condition where the function is required. Where potential risks (e.g. trigger a transient or damage some SSCs) during periodic testing are identified by analysis, a transposition test should be considered in design phase. In such cases, the full test may be carried out in another normal operating mode (e.g. shutdown states), with extrapolation of the test results to relevant operation conditions being calculated analytically. These transpositions values should have to be justified.

h) Following a regular preventive maintenance (i.e.as per the maintenance schedule) or following an Operating Technical Specification (OTS) event requiring maintenance or reparation of an equipment, a test should be performed to revalidate the ability of the equipment to perform safety functions. Indeed, where the structural integrity of the SSCs may have been challenged, it's crucial for nuclear safety to practically check the availability of the equipment.

EMIT operational management is addressed in Sub-chapter 31.6 and the EMIT activities on each SSC important to safety are presented in relevant system chapters.

4.4.7 Codes and Standards

To reflect the functional reliability requirements, SSCs are designed, manufactured, constructed, installed, commissioned, operated, tested, inspected and maintained based on a complete set of applicable codes and standards. The design codes and standards are selected in accordance with the safety classification of SSCs.

The principles for selection of applicable codes and standards consider all necessary

factors including the design characteristics, UK regulatory expectations, requirements of guidance documents, and engineering practice. The main selection principles are as follows:

a) Codes and standards are commensurate with the categorisation of safety functions and classification of the SSCs (detailed information is presented in 4.4.5);

b) Codes and standards meet relevant good practice recognised by regulators;

c) The latest version of codes and standards is applied.

The selection process is applied which includes collection, screening, assessment, justification and analysis. The applicability, adequacy and sufficiency of codes and standards are identified and evaluated, and other relevant factors are also considered, such as scope of application, degree of familiarity, application in practical engineering and relationship with the reference plant.

Details of the selection principles and the process are presented in the *General Principles for Application of Laws, Regulations, Codes and Standards*, Reference [7]. The justification of suitability of codes and standards and the list of main codes and standards for specific areas are provided in Sub-chapter X.3 of all chapters (except for Chapter 1, 2, 3).

4.5 Concluding Remarks

General Safety and Design Principles in the UK HPR1000 are developed from HPR1000 (FCG3) with additional consideration of UK context.

To achieve the nuclear safety objective, the principles are developed with a logical process as follows:

a) Nuclear safety objective;

b) The concept of DiD;

c) Safety analysis;

d) Categorisation of safety functions and classification of SSCs;

e) Engineering substantiation;

f) Codes and standards.

The General Safety and Design Principles of the UK HPR1000 as presented above are adequate, suitable and reflect the UK context. These principles are applied in the design to support the reduction of the nuclear safety risks to an ALARP level.

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