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General Nuclear System

# General Nuclear System Ltd.

UK HPR1000 GDA Project

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**Title:**

# Pre-Construction Safety Report

# Chapter 31

# Operational Management

This document has been prepared on behalf of General Nuclear System Limited (GNS) with the support of China General Nuclear Power Corporation (CGN) and Électricité de France S.A. (EDF).

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## 31.1 List of Abbreviations and Acronyms

| | |
|---|---|
| AAD | Startup and Shutdown Feedwater System [SSFS] |
| AFCEN | French Association for Design, Construction and In-Service Inspection Rules for Nuclear Steam Supply System Components |
| ALARP | As Low As Reasonably Practicable |
| APA | Motor Driven Feedwater Pump System [MFPS] |
| ARE | Main Feedwater Flow Control System [MFFCS] |
| CGN | China General Nuclear Power Corporation |
| CRS | Chemistry and Radiochemistry Specification |
| DBC | Design Basis Condition |
| DEC | Design Extension Condition |
| DEC-A | Design Extension Condition A |
| DEC-B | Design Extension Condition B |
| EBA | Containment Sweeping and Blowndown Ventilation System [CSBVS] |
| EMIT | Examination, Maintenance, Inspection and Testing |
| ENIQ | European Network for Inspection Qualification |
| EOP | Emergency Operating Procedure |
| GCT | Turbine Bypass System [TBS] |
| GDA | Generic Design Assessment |
| HFE | Human Factors Engineering |
| HIC | High Integrity Component |
| HMI | Human Machine Interface |
| HPR1000 (FCG3) | Hua-long Pressurised Reactor under Construction at Fangchenggang Nuclear Power Plant Unit 3 |
| IAEA | International Atomic Energy Agency |
| ISI | In-Service Inspection |
| KDS | Diversity Actuation System [DAS] |
| KIC | Plant Computer Information and Control System [PICS] |
| MCR | Main Control Room |
| MCS | Maintenance Cold Shutdown |
| NDT | Non-destructive Testing |

| | |
|---|---|
| NOP | Normal Operating Procedure |
| NS/RIS-RHR | Normal Shutdown with RIS-RHR |
| NS/SG | Normal Shutdown with Steam Generators |
| ONR | Office for Nuclear Regulation |
| OTS | Operating Technical Specification |
| PCSR | Pre-Construction Safety Report |
| PSI | Pre-Service Inspection |
| PT | Periodic Test |
| RCD | Reactor Completely Discharged |
| RCM | Reliability Centred Maintenance |
| RCP | Reactor Coolant System [RCS] |
| RCS | Refuelling Cold Shutdown |
| RCV | Chemical and Volume Control System [CVCS] |
| RGL | Rod Position Indication and Rod Control System [RPICS] |
| RGP | Relevant Good Practice |
| RHR | Residual Heat Removal |
| RIS | Safety Injection System [SIS] |
| RP | Reactor in Power |
| RPN | Nuclear Instrumentation System [NIS] |
| RRI | Component Cooling Water System [CCWS] |
| SAMG | Severe Accident Management Guideline |
| SG | Steam Generator |
| SOA | State Oriented Approach |
| SSC | Structures, Systems and Components |
| TEG | Gaseous Waste Treatment System [GWTS] |
| TEP | Coolant Storage and Treatment System [CSTS] |
| TSC | Technical Support Centre |
| UK HPR1000 | UK version of the Hua-long Pressurised Reactor |
| WENRA | Western European Nuclear Regulators Association |

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Safety Injection System (RIS [SIS]).

## 31.2 Introduction

The main objective of this chapter is to present the approach that is used for operational management and demonstrate the safety aspects of operation and management are substantiated to ensure that the requirements of, and assumptions in the safety case are identified in operational management.

According to *Generic Design Assessment Guidance to Requesting Parties*, in Reference [1], the means of operational management for the UK version of the Hua-long Pressurised Reactor (UK HPR1000) are identified, which include operating procedures; operating limits and conditions; Examination, Maintenance, Inspection and Testing (EMIT) procedures; and ageing and degradation management procedures.

Only the methodology, principles and contents of operational management can be provided during the GDA phase, and the final operational management documents will be developed in the future nuclear site licensing phase.

In Sub-chapter 31.2.1, a number of relevant arguments and their associated evidence are presented to support the claims in Chapter 1. Sub-chapter 31.2.2 presents the general structure of the Chapter. Sub-chapter 31.2.3 lists the interfaces with other Chapters.

### 31.2.1 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 3.3*** derived from high level ***Claim 3***.

***Claim 3****: Nuclear safety*

*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.*

***Claim 3.3****: The design of the processes and systems has been substantiated and the safety aspects of operation and management have been substantiated.*

***Claim 3.3.15****: The safety aspects of operation and management have been substantiated.*

To support Claim 3.3.15, this chapter develops a number of relevant arguments and evidence.

a) ***Argument 1****: Operating procedures are developed to provide a guide for operator response suitably to normal, emergency and severe accident conditions.*

   1) ***Evidence 1.1****: Normal Operating Procedure (NOP) development has been arranged to ensure the plant is operated within the operating limits and conditions (see Sub-chapter 31.4.1);*

   2) ***Evidence 1.2****: Emergency Operating Procedure (EOP) development has been arranged to prevent or mitigate the consequences of accident conditions, to safeguard core integrity (see Sub-chapter 31.4.2);*

   3) ***Evidence 1.3****: Severe Accident Management Guideline (SAMG) development has been arranged to limit the consequences of severe accidents (see Sub-chapter 31.4.3).*

b) ***Argument 2****: The operating limits and conditions are identified to ensure the plant is operated safely at all times.*

1) ***Evidence 2.1****:The arrangement of Operating Technical Specification (OTS) has been made to ensure the plant operation is within the limits derived from safety analysis (see Sub-chapter 31.5.1);*

2) ***Evidence 2.2****: The arrangement of Chemistry and Radiochemistry Specification (CRS) has been made to ensure chemistry-based limits and conditions are identified in the safety case (see Sub-chapter 31.5.2);*

3) ***Evidence 2.3****: The accounting of loading conditions is managed by a specific operating document for operators to respect the initial design substantiation in the demonstration of structural integrity(see Sub-chapter 31.5.3);*

4) ***Evidence 2.4****: The operating limits derived from the core design requirements are managed by a specific operating document to ensure fuel safety during the plant operation (see Sub-chapter 31.5.4).*

c) ***Argument 3****: The EMIT, ageing and degradation procedures are developed to ensure the requirement of operating limits and conditions is effective.*

1) ***Evidence 3.1****: The arrangement of Periodic Test (PT) procedure has been made to ensure that the safety functions of Structures, Systems and Components (SSC) continuously meet the design intent during the plant operation (see Sub-chapter 31.6.1);*

2) ***Evidence 3.2****: The arrangement of In-Service Inspection (ISI) programme has been made to provide forewarning of failure and ensure continued safe operation (see Sub-chapter 31.6.2);*

3) ***Evidence 3.3****: The arrangement of maintenance procedure has been made to ensure that the degradation of SSC related to safety can be detected and mitigated, or the performance of design functions of failed SSC can be restored to an acceptable level during the plant operation (see Sub-chapter 31.6.3);*

4) ***Evidence 3.4****: The ageing and degradation management is arranged to ensure that the ageing and degradation effects of SSC can be detected and the associated reductions in safety margins can be addressed over the plant operating lifetime (see Sub-chapter 31.6.4).*

### 31.2.2 Chapter Structure

It should be noted that Chapter 31 is a new chapter that was not detailed in the preliminary safety report.

The structure of this chapter refers to items (a) (b) and (c) of Section 106, and Section 212 in Reference [2].

The general structure of this chapter is presented as below:

a) Sub-chapter 31.1 List of Abbreviations and Acronyms

This section lists all the abbreviations and acronyms used in this chapter.

b) Sub-chapter 31.2 Introduction

This section introduces the objective, scope and strategy of this chapter.

c) Sub-chapter 31.3 Applicable Codes and Standards

This section discusses the applicable codes and standards which apply to the area of operational management for the UK HPR1000.

d) Sub-chapter 31.4 Operating Procedures

This section presents the types of operating procedures, and illustrates the framework of the NOP, EOP, SAMG, and procedure development. It also presents operating modes, and plant normal start-up and shutdown processes.

e) Sub-chapter 31.5 Operating Limits and Conditions

This section presents the operating limits and conditions, including operating technical specifications, chemistry and radiochemistry specifications, loading conditions and core design limits.

f) Sub-chapter 31.6 EMIT and Ageing Degradation

This section presents the contents of testing, inspection and maintenance, and it explains generic principles and general processes of periodic tests, in-service inspection and maintenance, as well as the management of ageing and degradation.

g) Sub-chapter 31.7 ALARP Assessment

This section presents the As Low As Reasonably Practicable (ALARP) evaluation.

h) Sub-chapter 31.8 Concluding Remarks

This section presents a general conclusion for operational management.

i) Sub-chapter 31.9 References

This section lists all the references of this chapter.

### 31.2.3 Interfaces with other Chapters

The interfaces with other Chapters are listed in the following table.

T- 31.2-1 Interfaces between Chapter 31 and Other Chapters

| PCSR Chapter | Interfaces |
|---|---|
| Chapter 1 Introduction | Chapter 1 provides the high-level safety case route map and the methodology for route map development.<br><br>Chapter 31 provides the description to support the level 3 claim (claim 3.3.15). |
| Chapter 4 General Safety and Design Principles | The general requirements of equipment qualification, EMIT, ageing and degradation are described in Chapter 4.<br><br>Chapter 31 provides the arrangement of the EMIT, and ageing and degradation procedures in compliance with the requirements from Chapter 4. |

| PCSR Chapter | Interfaces |
|---|---|
| Chapter 5 Reactor Core | Reactor core design is discussed in Chapter 5.<br>Chapter 31 presents the arrangement of operating limits and conditions for core design. |
| Chapter 6 Reactor Coolant System | Chapter 6 provides the reactor coolant system design substantiation relevant to EMIT, ageing and degradation.<br>Chapter 31 provides the arrangement of EMIT ageing and degradation procedure. |
| Chapter 7 Safety Systems | Chapter 7 provides safety systems design substantiation relevant to EMIT, ageing and degradation.<br>Chapter 31 provides the arrangement of EMIT, ageing and degradation procedure. |
| Chapter 8 Instrumentation and Control | Chapter 8 provides instrumentation and control systems design information relevant to EMIT, ageing and degradation.<br>Chapter 31 provides the arrangement of EMIT, ageing and degradation procedure. |
| Chapter 9 Electric Power | Chapter 9 provides electrical power system design substantiation relevant to EMIT, ageing and degradation.<br>Chapter 31 provides the arrangement of EMIT, ageing and degradation procedure. |
| Chapter 10 Auxiliary Systems | Chapter 10 provides auxiliary systems design substantiation relevant to EMIT, ageing and degradation.<br>Chapter 31 provides the arrangement of EMIT, ageing and degradation procedure. |
| Chapter 11 Steam and Power Conversion System | Chapter 11 provides steam and power conversion system design substantiation relevant to EMIT, ageing and degradation.<br>Chapter 31 provides the arrangement of EMIT, ageing and degradation procedure. |
| Chapter 12 Design Basis Condition Analysis | Chapter 12 provides the safety analysis of design basis conditions.<br>Chapter 31 provides the arrangement of operating limits and conditions consistent with the safety analysis. |

| PCSR Chapter | Interfaces |
|---|---|
| Chapter 13 Design Extension Conditions and Severe Accident Analysis | Chapter 13 provides the safety analysis of design extension conditions and severe accident conditions.<br>Chapter 31 provides arrangement of operating limits and conditions consistent with the safety analysis. |
| Chapter 15 Human Factors | Chapter 15 provides the principles and methodology of human factors design that shall be considered in procedure development.<br>Chapter 31 provides the process of procedure development, which takes human factors into account. |
| Chapter 17 Structural Integrity | Chapter 31 presents the arrangement of plant operational management.<br>Chapter 17 demonstrates the structural integrity of metal SSC by taking into account plant operational management. |
| Chapter 20 MSQA and Safety Case Management | The organisational arrangements and quality assurance arrangements set out in Chapter 20 are implemented in the design process and production of Chapter 31. |
| Chapter 21 Reactor Chemistry | Chapter 21 provides the limit of reactor chemistry.<br>Chapter 31 provides the arrangement of the chemistry specification identified from reactor chemistry. |
| Chapter 22 Radiological Protection | Chapter 22 describes the source terms during normal operation.<br>Chapter 31 needs to consider the source terms in the arrangement of CRS. |
| Chapter 23 Radioactive Waste Management | Chapter 23 provides radioactive waste management systems design substantiation relevant to EMIT, ageing and degradation.<br>Chapter 31provides the principles, content, and methodology of EMIT, ageing and degradation procedures. |
| Chapter 28 Fuel Handling and Storage | Chapter 28 provides fuel handling and storage design substantiation relevant to EMIT, ageing and degradation.<br>Chapter 31provides the principle, content, and methodology of EMIT, ageing and degradation procedures. |
| Chapter 30 Commissioning | Chapter 30 provides the requirements and arrangements for commissioning.<br>Chapter 31 presents the arrangement of operating procedure and operating technical specification which shall be respected during the commissioning activities. |

| PCSR Chapter | Interfaces |
|---|---|
| Chapter 32 Emergency Preparedness | The EOP and SAMG in Chapter 31 are the means of on-site accident management for emergency preparedness in Chapter 32. |
| Chapter 33 ALARP Evaluation | Chapter 31 provides the assessment in which risk associated with operational management has been reduced as low as is reasonably practicable, supporting the ALARP assessment in Chapter 33. |

## 31.3 Applicable Codes and Standards

The following principles are used in the selection of codes and standards, which is consistent with the requirement of Chapter 4:

a) The Relevant Good Practice (RGP) of international organisations or other countries recognised by the UK regulators are taken into account adequately;

b) The selection experience used in previous Generic Design Assessment (GDA) projects is considered.

The codes and standards applied in Chapter 31 are shown in T-31.3-1, which are considered as RGP for operational management. Further identification of RGP for operational management is discussed in Sub-chapter 31.7. All the applicable codes and standards are sufficient to guide the arrangements of operational management.

T-31.3-1 Applicable Codes and Standards

| No. | Codes and Standards |
|---|---|
| 1 | WENRA, Safety Reference Levels for Existing Reactors, September 2014. |
| 2 | IAEA, Safety Requirements: Safety of Nuclear Power Plants: Design, No.SSR-2/1, Rev. 1, February 2016. |
| 3 | IAEA, Safety Requirements: Safety of Nuclear Power Plants: Commissioning and Operation, No.SSR-2/2, Rev. 1, February 2016. |
| 4 | IAEA, Safety Guide: Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, No.NS-G-2.2, November 2000. |
| 5 | IAEA, Safety Guide: Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants, No.NS-G-2.6, October 2002. |
| 6 | IAEA, Safety Guide: Ageing Management for Nuclear Power Plants, No.NS-G-2.12, January 2009. |
| 7 | IAEA, Specific Safety Guide: Chemistry Programme for Water Cooled Nuclear Power Plants, No. SSG-13, January 2011. |
| 8 | IAEA, Safe Reports Series No. 48: Development and Review of Plant Specific Emergency Operating Procedures, 2006. |

| No. | Codes and Standards |
|---|---|
| 9 | AFCEN, In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands, RSE-M, 2010 edition+2012 addendum, 2010, 2012. |
| 10 | JRC, European Methodology for Qualification of Non-Destructive Testing, THIRD ISSUE, ENIQ Report No.31, EUR 22906EN-2007. |
| 11 | US NRC, Human Factors Engineering Program Review Model, NUREG-0711, Rev. 3, November 2012. |

## 31.4 Operating Procedure

Operating procedures are important to plant safety because they support and guide personnel interacting with plant systems and responding to plant-related events under different conditions. These procedures form an essential part of the administrative safety measures.

Operating procedures include NOP, EOP and SAMG. There are several types of procedures contained in each of these broad categories, as described in Sub-chapter 31.4.1, 31.4.2 and 31.4.3 below. The methodology used to develop the above procedures for the UK HPR1000 is discussed in Sub-chapter 31.4.4.

Detailed procedures will be developed during the nuclear site licensing phase.

### 31.4.1 Normal Operating Procedure

NOP is used for normal operation, including plant start-up and shutdown, power operation and some transients within the normal operating conditions. NOP generally includes unit operating procedure, system operating procedure, system alarm sheet, and abnormal operating procedures.

Unit operating procedure is used for changing the state of the plant including start-up, shutdown, preparation of refuelling, etc., and for providing integrated operation of the plant. System operating procedure is used for system charging, energising, starting up, shutting down, changing modes of system operation, and other instructions appropriate for the operation of various systems in the plant. The system alarm sheet is used to determine the remedial action to be taken after an alarm appears. Abnormal operating procedure is used to manage an abnormal situation and bring the plant back to normal operation before the protection systems are initiated.

a) Plant Normal Operating Domain

Plant normal operating domain is the envelope of all normal operating conditions, which cover operating modes from Reactor in Power (RP) to Reactor Completely Discharged (RCD).

The definition of the plant normal operating domain is given in Reference [3].

b) Operating Modes and Standard Operating States

To ensure the safe operation of the plant, it must be operated within the plant normal operating domain under normal conditions. According to similar thermodynamic conditions, reactor physical properties, similar operating conditions, and safety management requirements, the plant normal operating domain is divided into different partitions, called operating modes.

Operating modes are used in operating procedures for operational management of the plant. Six modes are defined in Reference [4]:

1) Reactor in Power (RP);
2) Normal Shutdown with Steam Generators (NS/SG);
3) Normal Shutdown with RIS-RHR (NS/RIS-RHR);
4) Maintenance Cold Shutdown (MCS);
5) Refuelling Cold Shutdown (RCS);
6) Reactor Completely Discharged (RCD).

According to reactor power level, reactivity, average primary temperature, pressure and other parameters, the operating modes are divided into standard operating states for convenience of design and operation.

The detail of standard operating states is described in Reference [4].

c) Normal Shutdown and Start-up Process

The normal shutdown process consists of plant shutdown, primary system draining and opening, and core unloading. The normal start-up process of a plant consists of core reloading, primary system closing and filling, and plant start-up.

1) Plant Shutdown

Plant shutdown begins with a reduction of the turbine load from power operation in RP mode. The turbine-generator can be tripped from any load that is compatible with the plant and supporting system requirements. However, except in emergency conditions, it is desirable to gradually reduce load with a rate as low as possible prior to tripping the turbine with consent from the grid control centre for grid disconnection. When turbine is disconnected, the reactor is brought to a subcritical state through rod insertion and an increase in boron concentration, as required for hot shutdown. At this point, the plant operates in NS/SG mode. Primary pressure is controlled by the pressuriser spray valves and heaters, and primary temperature is controlled by Turbine Bypass System (GCT [TBS]). The Steam Generator (SG) continues to remove heat, and the water levels of the SG are controlled by the Startup and Shutdown Feedwater System (AAD [SSFS]) and Main Feedwater Flow Control System (ARE [MFFCS]). After the reactor is stabilised at the hot shutdown state, relevant inspections and periodic tests are performed.

Before leaving the hot shutdown state, the primary system is borated to the boron concentration required for cold shutdown. Then the primary temperature and pressure are decreased. The accumulators are isolated to avoid injection into primary system during the depressurisation process. When the primary temperature reaches Residual Heat Removal (RHR) connection condition, the Safety Injection System (RIS [SIS]) is connected to primary system in RHR mode. At this point, the plant reaches NS/RIS-RHR mode, and the primary system is cooled by RIS [SIS]. The valves of the GCT [TBS] are closed, and feedwater pumps are stopped.

After the primary system has cooled to approximately {    }, the pressuriser steam bubble collapses. When the pressuriser is in water solid state, the primary pressure control is switched from two-phase mode to single-phase mode, and the primary pressure is automatically controlled by the Chemical and Volume Control System (RCV [CVCS]) high pressure let-down line. The coolant contraction in the Reactor Coolant System (RCP [RCS]) cooling process is compensated by the charging pumps.

When the primary temperature is below { }, hydrogen peroxide is injected for primary system oxidisation. During oxidation, the spray valves are fully opened, and three RCP [RCS] pumps are in operation. When the primary radiochemistry conditions and targeted boron concentration are met, the last RCP [RCS] pump is allowed to be shut down.

After shutdown of the last RCP [RCS] pump, primary pressure control is switched from the high-pressure letdown line to the low pressure letdown line. Then depressurisation is continued and, when the primary system is depressurised to { }, the reactor trip breakers are opened to drop all control rods to the bottom of the reactor core.

2) Primary System Draining and Opening

RCP [RCS] is drained to mid-loop level by the RCV [CVCS] letdown line. Whilst the level of RCP [RCS] is maintained at mid-loop level, the RCP [RCS] is swept by injecting nitrogen, and vented to the Gaseous Waste Treatment System (TEG [GWTS]) by a vacuum pump. The sweeping of RCP [RCS] is switched from nitrogen to air sweeping when the hydrogen limit is satisfied, and vented to the Containment Sweeping and Blowndown Ventilation System (EBA [CSBVS]). When radiochemistry parameters in the RCP [RCS] are satisfied, and the RCP [RCS] is filled to the vessel flange level, opening of the primary system is authorised. When the blind flange in vent line is disconnected, the plant reaches MCS mode.

After the vessel head thermal insulation has been removed, the multi-stud tensioning machine is positioned to carry out vessel head opening operations.

3) Core Unloading

Whilst the vessel head is lifted, the reactor cavity is filled with reactor coolant. Then the control rod drive mechanisms are disconnected and the upper reactor internals are withdrawn. While the reactor cavity is full, the plant reaches RCS mode, and fuel unloading operations can begin. While all fuel assemblies are in the fuel pool, the plant reaches RCD mode.

4) Core Reloading

Before core reloading, the reactor cavity needs to be filled with reactor coolant. Then the transfer tube is opened, and core reloading can begin. The plant is in RCS mode when it is reloaded with one fuel assembly. After core reloading is finished, the transfer tube is closed, and the upper reactor internals are put back in position.

5) Primary System Closing and Filling

After the upper reactor internals are put back in position, the draining of the cavity begins, then the plant reaches MCS mode. The reactor cavity is drained to the vessel flange level by gravity. The reactor cavity water is transferred to the in-containment refuelling water storage tank. By using the multi-stud tensioning machine, the reactor vessel is closed. The electrical connections of the control rod drive mechanisms and core instrumentation are re-installed.

The primary system is then drained down to mid-loop level by the RCV [CVCS] letdown line. When the blind flange in the vent line is in place, the plant is in NS/RIS-RHR mode. The primary system is vacuumed using a vacuum pump, and is then filled using charging pumps.

6) Plant Start-up

After the vacuum pump is stopped, the primary system is continuously filled with water until full, and static venting can begin. The primary pressure can be increased by reducing the letdown flowrate. When the primary system is pressurised to { }, the RCV [CVCS] low pressure letdown line can be switched to the high pressure letdown line. The primary pressure is stabilised by the high pressure letdown line. The first RCP [RCS] pump is started before the primary temperature reaches{ }. Then the remaining two RCP [RCS] pumps and all pressuriser heaters are started to heat the reactor coolant. Whilst the reactor coolant is heated, the RIS [SIS] is operated in RHR mode to control the primary temperature.

After the primary temperature increases to roughly { }, hydrazine ($N_2H_4$) is added for deoxidisation. After the primary temperature increases to approximately { }, and water chemistry meets requirements, and the pressuriser steam bubble is built. The primary pressure control is the switched from two-phase mode to single-phase mode, and the primary pressure control is switched from the RCV [CVCS] high pressure letdown line to the pressuriser. The RCV [CVCS] letdown line is used to maintain pressuriser level at a level corresponding to zero power.

When the primary temperature meets the required RIS [SIS] disconnection conditions, the RIS [SIS] in RHR mode is disconnected. The plant operating status subsequently enters into NS/SG mode. The primary temperature is controlled by the SG (steam is released by GCT [TBS], and SG water supply is provided by the AAD [SSFS]).

In parallel, the primary temperature and pressure continuously increase to the hot shutdown state, and excess coolant produced by volumetric expansion is sent to the Coolant Storage and Treatment System (TEP [CSTS]) storage tank through the RCV [CVCS] letdown line (automatic pressuriser level control).

In a hot shutdown state, selected tests are performed, such as the control rod drop test. The primary temperature is controlled by the GCT [TBS], and criticality is reached by dilution or rod withdrawal. At this point, the plant operates in RP mode. After the zero-power physical test is completed, the power is increased through rod withdrawal.

After nuclear power reaches approximately {} nominal power, the SG water supply is switched to the Motor Driven Feedwater Pump System (APA [MFPS]) from AAD [SSFS].

With nuclear power at approximately { } nominal power, the process to run up the turbine and synchronise the generator to the grid is undertaken. While the turbine load increases, the GCT [TBS] is gradually closed. Once fully closed, the control rods are switched to automatic mode, and the nuclear power is automatically adjusted according to the target load of the turbine. At this power level, all controls are in automatic mode and power is gradually increased to 100% nominal power.

7) Interfaces with other procedures

The NOP is implemented during the normal operation of plant. If an accident occurs, the implementation of EOP is required to mitigate the consequences of the accident. It is possible to transfer to normal operation from emergency operation if the exit conditions stated in the EOP are fulfilled.

### 31.4.2 Emergency Operating Procedure

EOP for the UK HPR1000 is an indispensable part of the defence in depth concept. EOP aims to guide the plant operation and define the operator actions needed to bring the plant to a safe and steady state after transients, incidents and accidents.

EOP defines post-accident mitigation strategies and actions to be performed to reach the objective state. The principles of EOP are briefly described below.

a) State Oriented Approach

EOP applies the State Oriented Approach (SOA), which is based on the following fact: while there may be an infinite number of event and failure combinations, the exact number of physical states is known. The physical state is defined and corresponds to a set of physical parameter values which characterises the behaviour of the plant at a given moment. These physical parameters are divided into six state functions in closed states (primary system is closed) which can be monitored using the information provided by specific instrumentation (see table T-31.4-1). In non-closed states, no state functions are defined, and the objective of EOP is to ensure the three basic safety functions (reactivity control, residual heat removal, and radioactive material containment) by available means.

EOP covers Design Basis Condition (DBC) and Design Extension Condition A (DEC-A), Reference [5]. They are essential to supporting the deterministic safety analysis and must address the manual actions claimed in safety analysis.

T-31.4-1 State Functions of SOA

| No. | State Functions | Information |
|---|---|---|
| 1 | Nuclear power (control of sub-criticality) | Ex-core neutron flux measurement |
| 2 | RCP [RCS] water inventory | Core outlet coolant saturation margin or reactor vessel water level measurement |
| 3 | Primary temperature and pressure (residual heat removal) | Core outlet coolant saturation margin |
| 4 | SG integrity | Secondary pressure (per SG) and secondary radioactivity (per SG) |

| No. | State Functions | Information |
|---|---|---|
| 5 | SG water inventory | SG water level |
| 6 | Containment integrity | Containment pressure and containment activity |

b) General Operating Principles

With the application of SOA, a limited set of strategies are developed depending on the physical state of the plant rather than the sequence of events or failures that lead to this state.

Generally speaking, an operating strategy includes three parts:

1) Identifying the physical state of the plant;

2) Determining the operating objectives;

3) Identifying the resources or systems needed to meet these objectives.

When the reactor primary system is in a closed state, a prescribed diagnosis of the six state functions is used to assess the plant status and to define the most appropriate operating strategy for the plant.

EOP addresses the implementation of additional alternative measures whenever possible, and principally normal operational systems are implemented with priority to avoid demands on engineered safety features.

Furthermore, as the actions to control one state function may contradict the actions proposed to control another state function, the state functions to be controlled are prioritised. A non-exhaustive list of primary side state function priorities corresponding to different plant states is shown T-31.4-2.

T-31.4-2 Priorities in Different Plant States

| No. | Plant States | State Function Priorities |
|---|---|---|
| 1 | Loss of sub-cooling margin or RCP[RCS] water inventory degraded | RCP [RCS] water inventory control;<br>Residual heat removal. |
| 2 | High sub-cooling margin | Residual heat removal. |
| 3 | Degradation of margin to criticality | Control of sub-criticality;<br>Residual heat removal;<br>RCP [RCS] water inventory control. |
| 4 | SG integrity degraded | Control of sub-criticality;<br>Residual heat removal;<br>RCP [RCS] water inventory control. |

| No. | Plant States | State Function Priorities |
|---|---|---|
| 5 | SG water inventory degraded | Residual heat removal;<br>RCP [RCS] water inventory control;<br>Control of sub-criticality. |
| 6 | Non degraded state | Control of sub-criticality;<br>Residual heat removal;<br>RCP [RCS] water inventory control. |

In non-closed states, there is no need to introduce typical state functions to define objectives and operating actions. The operating objectives for the reactor coolant system are to be met in the following order of priority:

1) Water inventory control and residual heat removal take priority if they are significantly degraded (orientate to degraded operation if they are highly degraded, and to stabilisation in all other cases);

2) If water inventory and residual heat removal functions are not significantly degraded (no SI signal and at least one RIS [SIS] train in service in RHR mode), the priority is to control nuclear power.

c) Operating Strategies

The set of EOP covers all plant operating modes: RP, NS/SG, NS/RIS-RHR, MCS, RCS and RCD. Depending on the initial (pre-event) configuration of the primary system and the severity of the event, different EOP strategies are elaborated.

As an example, the following table T-31.4-3 displays accident strategies and their objectives in primary closed states.

T-31.4-3 Examples of Accident Strategies and Objectives in Primary Closed States

| No. | Accident Strategies | Objectives |
|---|---|---|
| 1 | Special measures to reduce risks relating to station black out and Fukushima-type events. | Transition to a specific fallback mode using special measures provided in a station black out situation. |
| 2 | Transition to cold shutdown (with safety injection, without safety injection, with steam generator tube rupture). | Rapid transition to cold shutdown, to reach the safe and steady state. |
| 3 | Water inventory restoration. | Recover water inventory by maximising water injection, cool down, and depressurisation. |
| 4 | Feed and bleed. | Establish heat removal by feed and bleed operation and fall back to cold shutdown. |

| No. | Accident Strategies | Objectives |
|---|---|---|
| 5 | Reduction of high saturation margin. | Avoid pressurised thermal shock to reactor vessel and the pressuriser surge line and limit the consequence of overcooling. |

When the operator is confronted with a certain accident, the state of the plant is diagnosed on the basis of the six state functions. This assessment helps the operator to identify an appropriate strategy and corresponding operating actions. If the plant state changes and is diagnosed, a more appropriate strategy needs to be identified and adopted.

The SOA has the advantage of being considerably tolerant to potential human errors thanks to:

1) No need to identify initiating events;

2) Loop structure for each strategy, the operator is likely to keep alert to the previous error;

3) Permanent monitoring of state parameters in order to apply the most appropriate strategy with the evolution of plant status.

d) EOP Structure

EOP deals with incident and accident conditions separately in closed states while there is no distinction made between incident and accident conditions in non-closed states due to the absence of state functions.

1) Incident conditions in closed states, which cover disturbed situations with no degradation of the physical states of the plant, but require a specific operating strategy;

2) Accident conditions in closed states, which cover degradation of the physical states of the plant. It involves the implementation of a limited number of strategies. Theoretically, EOP covers an unlimited number of situations and is optimised depending on the complexity of the situation;

3) Incident/accident conditions in non-closed states, which are specific to situations to be dealt with in non-closed states and those occurring in the fuel building (there is no need to distinguish between incident condition/accident condition for fuel building events. In practice, fuel building events are managed by applying dedicated strategies).

The EOP is presented with operational flowcharts where Human Factors Engineering (HFE) is taken into account during the whole EOP design process so as to limit the risk of human errors as low as reasonably practicable.

Since different Human Machine Interfaces (HMIs) are designed for the UK HPR1000, the EOP set for each interface is a little different from each other:

1) The EOP for the Plant Computer Information and Control System (KIC [PICS]), which covers the DBC-2/3/4 and DEC-A conditions, is partially computerised, and dedicated monitoring displays are deployed to assist the state diagnosis and to identify the most appropriate strategy continuously;

2) The EOP for auxiliary control panel, which covers the DBC-2/3/4 conditions,

is paper based. And it is consistent with the safety analysis;

3) The EOP for Diversity Actuation System (KDS [DAS]), which is dedicated to the anticipated transient without scram management, is paper based.

e) Interfaces with other procedures

The criteria for entry into emergency operation are as follows:

Criterion 1: in response to identified system/equipment alarms, automatic reactor trip, and core protection and safeguard actions; or

Criterion 2: requested by a NOP.

Exit from emergency operation is possible, if either of the following two conditions is fulfilled:

1) Emergency operation is successful, i.e. it has brought the plant to a safe and steady state and lost functions have been restored. It is possible to exit from emergency operation if the exit conditions listed in the EOP are fulfilled;

2) Emergency operation fails, i.e. the core damage is inevitable and core melt is anticipated. In this case, transition from EOP to SAMG is required and the focus is on containment integrity rather than core integrity.

The transitions, in case of exit from emergency operation, can be characterised by the following three scenarios:

1) Transition to normal operation

   When some certain requirements are satisfied (e.g. primary temperature and pressure, boron concentration, position of control rods, and operation of reactor coolant pumps are compliant with the requirements of normal operating limits and conditions), the operator can use NOP to re-start the plant and reconnect it to the grid after re-configuring the systems in normal operating modes, and thus the OTS can be applied.

2) Transition to repair state

   Following an incident/accident, the plant may be damaged from an accident. If emergency operation successfully guides the plant to a safe and steady state, the operating shift team and the plant director will identify whether a repair state is required and a practicable way to reach it (with the necessary support from Technical Support Centre (TSC)). The repair state is characterised by a safe state in which the nuclear steam supply system is shut down and necessary maintenance can be performed. Depending on the nature of the damage, the repair state might not be a standard state (notably the MCS mode).

3) Transition to Severe Accident Management

   The severe accident entry criteria are characterised by core outlet temperature and/or dose rate in containment as indicative of core melt conditions, or by the reactor cavity water level as indicative of core-uncovered risk. These criteria are continuously monitored while EOP is in progress. When the entry criteria are fulfilled, SAMG is used instead of EOP. The decision to use SAMG instead of EOP is taken by duly authorised personnel.

### 31.4.3 Severe Accident Management Guideline

SAMG and EOP are the different procedures to deal with accidents in nuclear power plants. While EOP focuses on protecting core integrity, SAMG pays attention to ensuring containment integrity and limiting the release of fission products to the environment. SAMG, the general objective of which is to reach a controlled and stable state, should cover the whole of Design Extension Condition B (DEC-B) using realistic assumptions, and the equipment required for severe accident mitigation is qualified for the conditions and the necessary mission time.

The transition from EOP to SAMG is based on core outlet temperature and/or dose rate in containment, (water level and/or dose rate in spent fuel pool). These criteria are continuously monitored while EOP is in progress. When the entry criterion is reached, SAMG will be used instead of EOP.

The SAMG includes the sections used by the Main Control Room (MCR) and TSC. The section used by the MCR includes the initial response guideline and the guideline for transients after the TSC is functional. The section used by the TSC includes severe accident diagnosis and severe accident guidelines at the initial stage, diagnosis and severe challenge guidelines used when the safety barriers are under serious challenge, and long-term monitoring and exit guidelines after severe accidents are mitigated.

In case the entry criterion of SAMG detected by safety engineers is reached, the required action is to start the application of SAMG. Once transition is made from EOP to SAMG, a guideline is provided to the MCR for performing systematic actions known as “immediate actions”. These actions, performed by the operators on entry into or during the use of SAMG, do not have to be evaluated by the TSC. After the end of the “immediate actions”, the TSC staff will suggest some “delayed actions”. The TSC staff will validate mitigation strategies and give instructions to the MCR staff.

The main concern areas in severe accidents consist of: primary system depressurisation, hydrogen control, basement protection, decay heat removal, containment pressure control, and fission product release. In view of the strategies mentioned above, dedicated systems have been designed in the UK HPR1000. SAMG considers the specific systems and other normal operating systems to mitigate the accidents. The roles and responsibilities of the MCR and TSC teams are defined for undertaking SAMG.

SAMG is further discussed in Chapter 13.

### 31.4.4 Procedure Development

This sub-chapter describes the process for NOP, EOP and SAMG development. Procedures are developed by incorporating human factors, along with all other design requirements to make them technically accurate, comprehensive, explicit, easy to use, verified and validated per the requirements of NUREG-0711, Reference [6]. Additional details regarding the verification and validation of procedures are described in Reference [7].

a) Procedure Development Base

Procedure development is an iterative process. Procedures are developed and then tested to see if they meet the HFE requirements. Feedback from the HFE team must be reflected in the procedures and then revised procedures are rechecked to see if the HFE requirements are met. This feedback loop is continued until the procedures are technically accurate, comprehensive, explicit, and easy to use.

The UK HPR1000 procedures are based on the procedures for the Hua-long Pressurised Reactor under Construction at Fangchenggang Nuclear Power Plant Unit 3 (HPR1000 (FCG3)), and then modified to appropriately reflect the UK HPR1000 system design and the relevant UK regulatory requirements and operational experience. The following additional resources of information are also identified as the basis for developing procedures:

1) Plant-design basis;
2) System-based technical requirements and specifications;
3) Results of task analysis;
4) Important human actions;
5) Initial conditions considered in the procedure development;
6) Generic technical documents for operating procedures;
7) HMI design.

b) Procedure Writer's Guide

A procedure writer's guide establishes a standard guide to develop technical procedures that are complete, accurate, consistent and easy to understand and follow. It outlines criteria for ensuring that the content, organisation, and style are consistent across all procedures of a similar type. Additionally, the writer's guide ensures that relevant HFE principles are appropriately incorporated into the procedures, including consistency in form and function between computerised procedures and paper-format procedures. The writer's guide should at least provide instructions regarding the following aspects:

1) Title, revision and date;
2) Applicability and purpose;
3) Prerequisites or entry conditions;
4) Precautions such as warning and cautions;
5) Important human actions;
6) Limitations and actions;
7) Acceptance criteria;
8) Check-off lists;
9) Reference material.

The NOP and EOP will have their own formats due to the unique attributes of these procedure types.

c) NOP Development

The UK HPR1000 NOP is being developed using the following principles and development process:

1) The NOP follows the applicable procedure writer's guide to ensure consistency, accuracy, completeness, readability, and high quality among the various documents in the NOP set.
2) The principles for technical parts are provided in Reference [8].

3) The NOP is developed considering the HMI design, operational analysis and other relevant HFE information. Typically, human factors experts are involved in the design process.

4) The draft NOP is then reviewed by the HFE team to ensure that they reflect and appropriately apply human factors.

5) Feedback from the verification steps is evaluated to determine whether procedure revision is needed to address the issues identified.

6) The NOP is then validated by simulator, table top, or walk-through methods.

7) A plan for maintaining the NOP and controlling updates will be proposed and executed by the future licensee.

d) EOP Development

The UK HPR1000 EOP is being developed using the following principles and development process:

1) The EOP follows the applicable procedure writer’s guide to ensure consistency, accuracy, completeness, readability and high quality among the various documents in the EOP set.

2) The principles for technical parts are provided in Reference [9].

3) The EOP is developed considering the HMI design, operational analysis and other relevant HFE information. Typically, human factors experts will be involved in the design process.

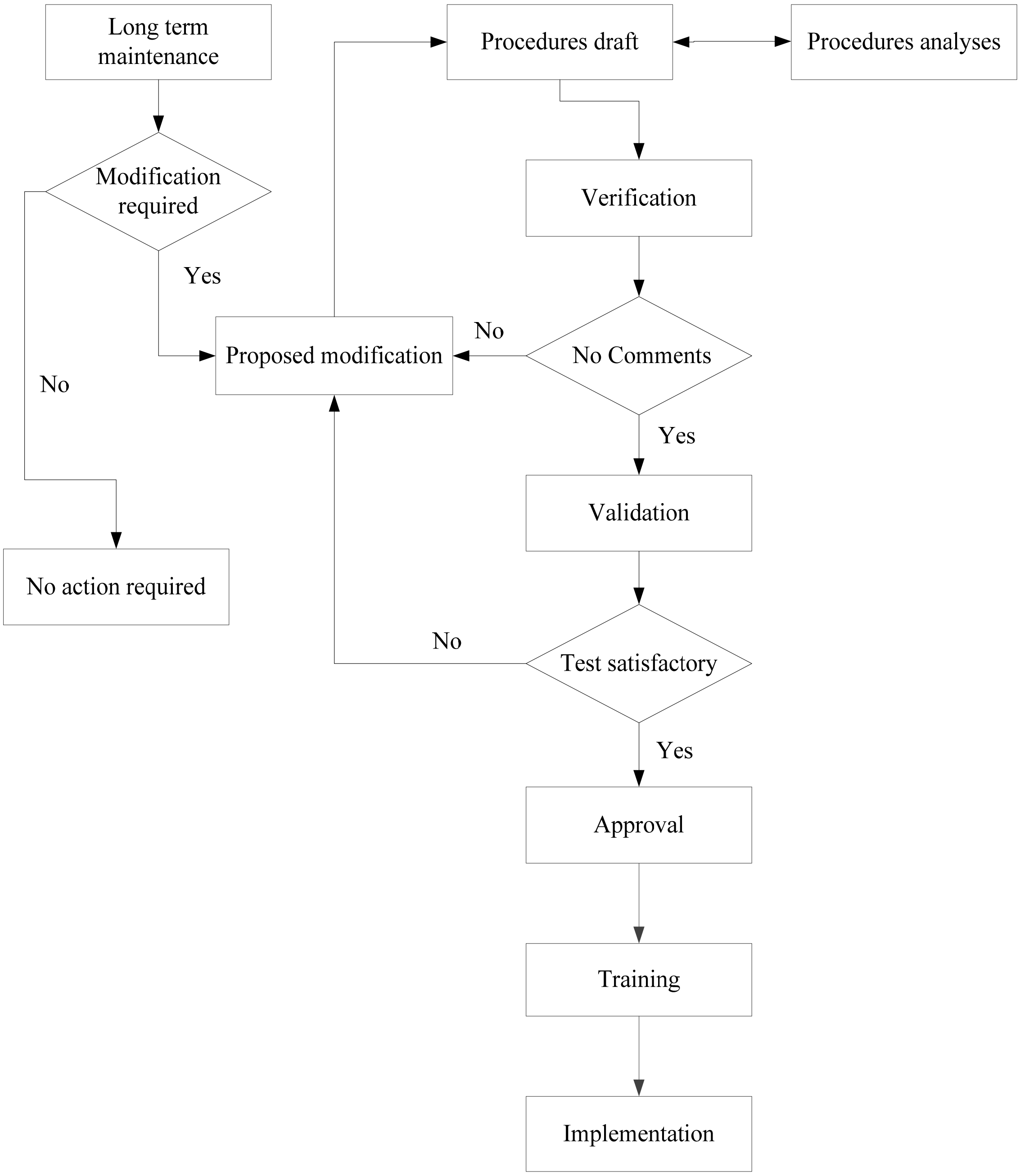
4) The draft EOP will be then reviewed by the HFE team to ensure that they reflect and appropriately apply human factors.

5) Feedback from the verification steps will be evaluated to determine whether procedure revision is needed to address the issues identified.

6) The EOP will be then validated by simulator, table top, or walk-through methods.

7) A plan for maintaining EOP and controlling updates will be proposed and executed by the future licensee.

F-31.4-1 provides a flowchart of activities to be performed in developing and maintaining the procedures, Reference [10].




F-31.4-1Procedures Development and Maintenance Flowchart

e) SAMG Development

According to the design characteristics of the UK HPR1000, the development of SAMG includes main steps listed below:

1) Review the OTS, Level 1 and Level 2 Probabilistic Safety Assessment, and define the characteristics of different standard operating states as the basis of developing SAMG.

2) Calculate dominant accident sequences and initiators to gain important insights into different accidents.

3) Define the suitable interfaces with other procedures.

4) Determine a significant diagnosis process.

5) Develop detailed guidelines.

6) Assess the proper computational aids.

## 31.5 Operating Limits and Conditions

Operating limits and conditions are required to ensure the plant is operated safely at all times. It includes six categories (below) and compliance will be maintained during the operating phase.

a) Operating Technical Specification (OTS);

b) Chemical and Radiochemical Specification (CRS);

c) In-Service Inspection (ISI);

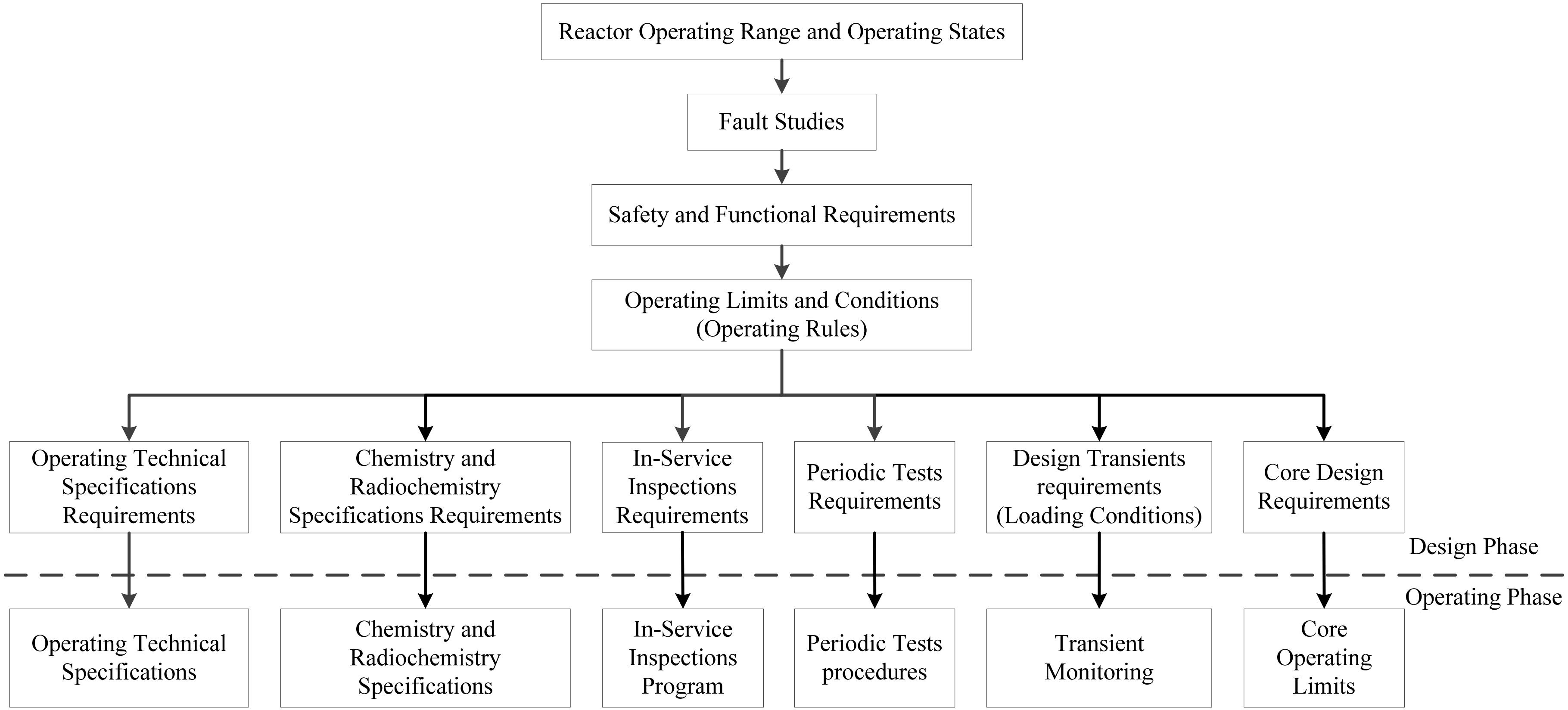
d) Periodic Test (PT);

e) Loading Condition (Design Transient);

f) Core Design Requirement.

In addition, ISI and PT are discussed in Sub-chapter 31.6.

F-31.5-1 presents the design phase sources of operating limits and conditions, and the interface between the design phase and operating phase for operating limits and conditions.




F-31.5-1 Interfaces with Operating Limits and Conditions

### 31.5.1 Operating Technical Specification

a) Generic Principles

The general objective of the OTS is to set out the limits and conditions that must be followed to ensure the plant operation within the limits justified by the safety analysis. The contents of OTS are as follows, Reference [11], [12] and [13]:

1) Safety limits. The aim is to protect the integrity of barriers that provide shielding against the uncontrolled release of radioactive material.

2) Safety system settings. A range of parameters, which aims to ensure automatic actuation of safety systems in accordance with parameters assumed in the safety case.

3) Limits and conditions for normal operation. These contain operability requirements of systems and components important to safety in all operating modes (from RP mode to RCD mode), which ensure that the plant operates in a status compliant with the assumptions of the safety analysis.

4) Required action for deviation from the operating limits and conditions. The required actions include fallback mode, fallback initiation time, and repair time.

OTS defines the normal operating domain limits. In the domain defined by OTS, all operating actions shall be realised using NOP. OTS contains two parts, OTS related to reactor, and OTS related to spent fuel. During fuel building events that require the implementation of spent fuel EOP, OTS related to spent fuel are not applicable, and those related to reactor are applicable. During implementation of other EOP strategies, OTS related to reactor are not applicable.

All cases, which are inconsistent with the requirements related to each operating mode, are called “event” (unavailability of required safeguard function or beyond normal operating limits).

A non-compliance with an OTS requirement generates an OTS event when it significantly challenges the compliance with safety criteria or radiological consequences issued from design basis condition studies.

The plant must comply with the normal operating limits and conditions under each operating mode. Generally, a single OTS event is allowed to occur in plan, only if it meets the “limit conditions” during preventive maintenances, refer to T-31.5-1.

In case of deviation from normal operation limits and conditions, the corresponding measures required in OTS must be implemented.

If there are several OTS events, which are not satisfied with the OTS requirements at the same time, the required actions are accumulated in a conservative way. The cumulative rules will be developed in the nuclear site licensing phase.

b) OTS Design Process

The OTS design process is shown in F-31.5-2.The detailed methodology of OTS is described in Reference [14].

1) Criteria of OTS

The OTS must be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

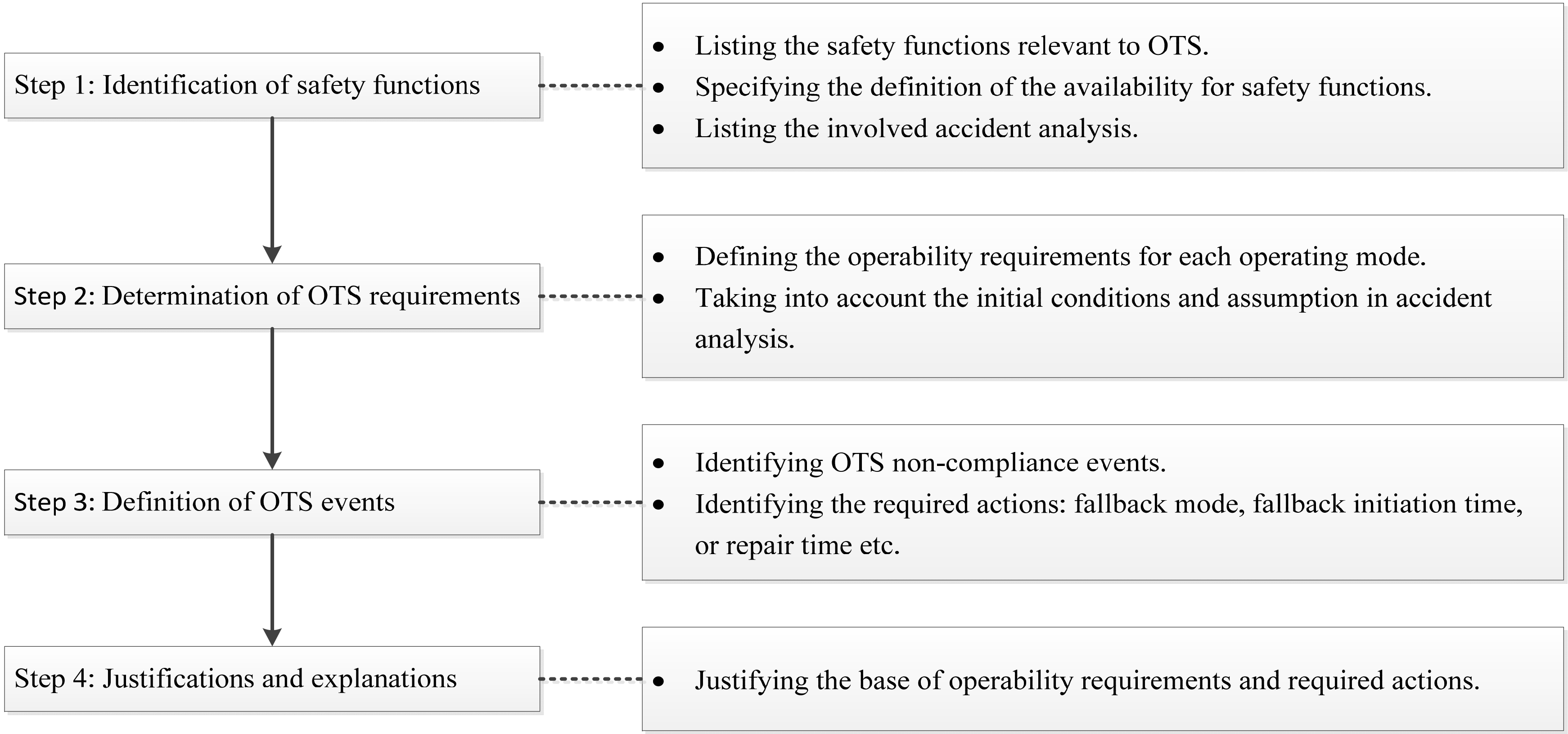
Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

2) Definition of the OTS Required Actions

The OTS shall define the required actions corresponding to events to ensure that the fundamental safety functions are available. The required actions are as follows:

- If the plant is required to fall back, specification of the fallback mode and the fallback initiation time;
- If the plant is not required to fall back, specification of the repair time;
- If necessary, according to the feedback and experience from similar nuclear power plants, specification of the corresponding special actions.




F-31.5-2 OTS Design Process Flowchart

c) Interfaces with Other Operating Documents

The interfaces with other operating documents are listed in the following table.

T-31.5-1 Interfaces between OTS and Operating Documents

| **No.** | **Other Operating Documents** | **Interfaces** |
|---|---|---|
| 1 | Normal Operating Procedure | The NOP which is prescribed in each operating mode complies with the requirements from OTS. |
| 2 | Emergency Operating Procedure | If the required actions of the OTS event cannot be completed through NOP, then an EOP is applied. Some limits and conditions in OTS no longer apply when an EOP is in progress. The exit conditions of an EOP are defined in the scope of the EOP. When these conditions are satisfied, either OTS is applied or a long-term event recovery phase is entered. |
| 3 | Chemistry and Radiochemistry Specification | Prime parameters of the chemistry and radiochemistry, which play an important role in material integrity in the short term or fuel integrity, will be regulated in the OTS, but the relevant supervision (including the sampling frequency, expected values, limiting values, non-compliance events, fallback modes and fallback time) is detailed in CRS. |
| 4 | Loading Condition | The operating limits derived from loading conditions are managed in OTS, such as the pressure/temperature limits during normal operation. |
| 5 | Core Design Requirement | The OTS interfaces with the core design requirements to maintain alignment between the nuclear and thermal-hydraulic design of the core for each fuel cycle with the operating limits and controls applied in the OTS. |

| No. | Other Operating Documents | Interfaces |
|---|---|---|
| 6 | Periodic Test | The PT meets the requirements of OTS. Failure to achieve a success criterion in a PT is a possible means of entry into an OTS non-compliance event.<br><br>The conduct of some tests may result in a non-compliance with OTS requirements. In this case, required actions and risk-prevented actions are identified. |
| 7 | Maintenance | In a given operating mode, functions required in OTS should not be unavailable due to the preventive maintenance. The implementation window of preventive maintenance must meet the OTS requirements for plant operation. If unavoidable, the window is given in the form of “limit conditions” in OTS, including corresponding preventive or alternative measures, predetermined operating requirements, limited durations, etc. |

d) Interface between GDA and nuclear site licensing

The OTS methodology is defined during GDA, including the criteria, involved definitions, and rules for required actions. However, the development of OTS will be finished in the nuclear site licensing phase, and the implementation of OTS as described herein will be at the discretion of the Licensee and required to satisfy relevant Licence Conditions for the applicable plant.

### 31.5.2 Chemistry and Radiochemistry Specification

a) Generic Principle

CRS defines the technical regulations of chemistry and radiochemistry that shall be respected during normal operation of the plant.

The limits of CRS are identified to ensure the integrity of the first barrier, the second barrier, and the systems and equipment, to ensure the validity of minimisation of source terms, to reduce radiation doses to personnel and to reduce the activity of discharges to the environment.

According to reactor chemistry and operating modes, CRS specifies expected values and/or limiting values, sampling and analysis frequency for chemistry and radiochemistry parameters.

Chemistry and radiochemistry parameters are divided into two categories as defined below, Reference [15]:

1) Control parameters: the parameters which play an important role in material integrity in the short term or fuel integrity, those parameters require strict control;

2) Diagnostic parameters: the parameters, which may likely affect radiation field build-up, corrosion of system materials in the long term, or fuel performance, or the parameters assisting the chemistry staff in interpreting primary coolant chemistry variations.

When any deviation from the requirements on the control parameters occurs, required actions are taken according to the CRS. The required action includes corrective actions, fallback mode, and fallback initiation time etc. Prime control parameters also are mentioned in the OTS.

The scope and principle parameters of the CRS are presented as follows.

b) Chemistry Specification

1) Primary Water Chemistry

The primary water chemistry parameters are limited to minimise coolant corrosion product concentration, optimise corrosion product migration and re-deposition, and prevent localised corrosion during the power operation, start-up, and shutdown conditions.

The concentrations of hydrogen, lithium and boron, etc. are managed in the chemistry conditioning of the primary coolant system.

The main purpose of defining hydrogen concentration is to limit water radiolysis and the corrosion cracking risk of RCP [RCS] materials.

The CRS defines the boron-lithium coordination to reach an optimum $pH_{300°C}$, chosen to minimise the material corrosion and the deposition of the corrosion products, which may become activated.

2) Secondary Water Chemistry

The secondary water chemistry parameters are limited to protect SG tubes during power operation, transients and SG layup conditions.

To select the secondary water chemistry parameters, the main considerations are given to the following key aspects: SG local corrosion problems; SG fouling and tube support plate clogging phenomena; the secondary system materials corrosion phenomena.

The secondary water chemistry plays an important role in SG fouling phenomena. An optimal pH value can efficiently mitigate flow-accelerated corrosion and transport of corrosion products.

In the secondary water, impurity concentrations play an important role in localised corrosion such as stress corrosion and/or pitting corrosion. Controlling the concentration of impurities such as oxygen, alkaline products or sulphur can limit these types of corrosion.

3) Auxiliary Systems Water Chemistry

The water chemistry parameters of auxiliary systems are limited to guarantee the primary and secondary chemistry parameters remain within the expected range by monitoring and controlling the chemistry condition of these systems.

Chemistry control of the auxiliary systems is established via two main actions:

- In order to guarantee compliance with the safety functions attributed to specific safety systems, the implementation of optimal chemistry conditioning of auxiliary systems is necessary, such as reactivity control via the boron concentration control in the spent fuel pool, in-containment refuelling water storage tank, boric acid storage tank, emergency boric acid tank etc.;
- The injection of additives contributing to chemistry conditioning into the auxiliary or primary systems is necessary. For example, during operating states with RIS trains connected in RHR mode, the injection of hydrazine can eliminate oxygen ingress, and the injection of trisodium phosphate into Component Cooling Water System (RRI [CCWS]) can prevent carbon steel corrosion.

c) Radiochemistry Specification

The radiochemistry parameters based on the barrier control and radiation protection are limited to ensure the validity of minimisation of source terms, and to protect the health of staff and public.

1) First barrier control

Iodine isotopes are released to the primary coolant in the presence of a defect when water circulates inside the pellet-cladding gap. Iodine isotopes are expressed as equivalent I-131 ($^{131}I_{eq}$) which represents the activity released by all iodine isotopes. The control of the $^{131}I_{eq}$ is required during normal power operation and during power transients;

Total noble gas activity is an indicator associated with equivalent I-131. The noble gas is not trapped inside the rod and it is able to reach the defect without difficulty.

2) Second barrier control

The measurements of total gamma activity can monitor the leak flowrate between the primary system and the secondary cooling system. Monitoring any change in this leak flowrate enables the early stage of an accident to be detected.

3) The radiation protection

The radioactive waste systems may release radioactive gas or liquid effluent during normal operation of the power plant. In parallel, the reactor coolant and secondary cooling systems during certain transients also cause radioactive material release. The radiological dose is limited to protect the staff working onsite and the public offsite.

d) Interfaces with Other Operating Documents

The interfaces with other operating documents are listed in the following table.

T-31.5-2 Interfaces between CRS and Operating Documents

| No. | Other Operating Documents | Interfaces |
|---|---|---|
| 1 | Normal Operating Procedure | The chemistry and radiochemistry parameters are satisfied before certain specific operations under NOP. For example, opening of primary system, shutdown of the last reactor coolant pump, etc. |
| 2 | Operating Technical Specification | Prime parameters of chemistry and radiochemistry which play an important role in material integrity in the short term or fuel integrity will be regulated in the OTS, but the relevant supervision (including the sampling frequency, expected values, limiting values, non-compliance events, fallback modes and fallback initiation time) is detailed in CRS. |

| No. | Other Operating Documents | Interfaces |
|---|---|---|
| 3 | Loading Condition | Loading conditions may result in a change in chemistry. This change can be realised in the CRS in the form of specific requirements to, or increased monitoring of, critical chemistry parameters. |

e) Interface between GDA and nuclear site licensing

Key parameters which link directly to the mitigation and control of the consequences on safety and the environment are defined and established in the CRS during GDA. The control, monitoring and required actions of key parameters, which are associated with limiting values or expected values, will be managed in the nuclear site licensing phase.

### 31.5.3 Loading Condition

The structural integrity of pressurised nuclear equipment is ensured based on the analysis of the overall and local study of the primary and secondary components against damage mechanisms. These damage mechanisms include excessive deformation and plastic instability, elastic or elastoplastic instability (buckling), progressive deformation, fatigue, and fast fracture risk.

In plant design, the structural integrity of pressurised nuclear equipment is verified against design loading conditions (also called design transients see more detailed information in Sub-chapter 17.7), which are derived from the operating conditions that might occur to the equipment.

The operating conditions for the main primary and secondary circuits are defined to include transients anticipated in normal operation and emergency and accident conditions.

a) Accounting of Loading Conditions

The mechanical integrity of important components is verified in the initial stress report against all these design loading conditions defined in Reference [16]. This process of verification is included in Chapter 17 Structural Integrity.

In plant operation, the demonstration of integrity is ensured by verifying that these design conditions cover the actual transients experienced. Thus all actual transients happened to plant need to be recorded and counted, this process is called “on-site accounting of transients”, which is based on continuous monitoring of the relevant parameter. Once the accounting number of actual transient approaches the anticipated occurrence number, the mechanical integrity of important plant equipment needs to be revaluated to ensure the plant safety.

b) Interfaces with Other Operating Documents

The interfaces with other operating documents are listed in the following table.

T-31.5-3 Interfaces between Loading Condition and Operating Documents

| No. | Other Operating Documents | Interfaces |
|---|---|---|
| 1 | Operating Technical Specification | The operating limits derived from loading conditions are managed in OTS, such as the pressure/temperature limits during normal operation. |
| 2 | Chemistry and Radiochemistry Specification | Loading conditions may result in a change in chemistry. This change can be realised in CRS in the form of specific requirements to, or increased monitoring of, critical chemistry parameters. |
| 3 | Core Design Requirement | The loading condition is an input for the reactor core design compliant with the fault and accident analysis. |
| 4 | In-Service Inspection | The loading condition is considered as an input in the development of ISI and maintenance. |
| 5 | Maintenance | |

c) Interface between GDA and nuclear site licensing

A specific operating document for the operator will be developed in the nuclear site licensing phase, which instructs to account for and record the number of actual transients during plant operation.

### 31.5.4 Core Design Requirement

a) Core Design Requirement

In this section, discussion is limited to the identification of the operating limits and conditions from the fuel system design, nuclear design, and thermal and hydraulic design described in Chapter 5.

The fuel rod is designed to accommodate the in-pile conditions such as exceedingly high internal fission gas pressure, fuel and cladding temperatures, and cladding stresses. As power ramp rate plays a key role in maintaining fuel integrity during DBC-1 and DBC-2, the change rate of linear power has been restricted during plant operation to maintain the integrity of fuel rods.

The key aspects of the operating limits and conditions for core design are given as follows:

1) Shutdown margin;

2) Linear power density;

3) Core reactivity;

4) Moderator temperature coefficient;

5) Control bank insertion limits;

6) Heat flux hot channel factor;

7) Nuclear enthalpy rise hot channel factor;

8) Axial offset etc.

The operating limits and conditions for core design are part of the OTS, and will be managed as part of the OTS.

b) Interfaces with Other Operating Documents

The interfaces with other operating documents are listed in the following table.

T-31.5-4 Interfaces between Core Design Requirement and Operating Documents

| No. | Other Operating Documents | Interfaces |
|---|---|---|
| 1 | Operating Technical Specification | The OTS interfaces with the Core Design Requirements to maintain alignment between the nuclear and thermal-hydraulic design of the core for each fuel cycle within the operating limits and controls applied in the OTS. |
| 2 | Loading Condition | The loading condition is an input for the reactor core design compliant with the fault and accident analysis. |
| 3 | Periodic Test | PT activities including fuel loading verification and core physics tests need to meet core design requirements. |

c) Interface between GDA and nuclear site licensing

Key safety parameters, which reflect the results of the safety analysis for core loadings and cover the bounding values of nuclear design, will be defined and established in nuclear site licensing phase.

## 31.6 EMIT and Ageing Degradation

The procedures that include all of maintenance, periodic testing, inspection, and ageing and degradation activities shall be established. These procedures are conducted to maintain the availability of structures, systems and components during the service life by controlling degradation and preventing failures. In the event that failures do occur, maintenance activities shall be conducted to restore the capability of failed structures, systems and components to function within acceptance criteria.

The management of EMIT and ageing degradation satisfies the design requirements of system chapters mentioned in Sub-chapter 31.2. In addition, the management takes into account the environmental protection in EMIT activities.

### 31.6.1 Periodic Test

The goal of periodic tests design is to define a comprehensive list of the periodic tests that are to be performed on a given system. Each periodic test includes content and scope, frequency, the applicable operating mode, criteria to be met, etc.

a) Definition of PT and Requirements

PT is defined at the design stage. Availability of the related equipment must be verified before the carrying out of PT.

The duration of a test must be sufficient so that representative operation of the feature can be demonstrated. The required time must be defined or estimated at the design stage and validated during commissioning.

PTs are completed within the tolerance of their frequencies.

b) Methodology of PT Completeness Analysis

In order to define the periodic test elements specified above, an analysis must be made based on the list of safety functions of the system. The methodology for performing the PT completeness analysis has four steps as follows, Reference [17].

Step 1: List and Describe the Safety Functions of the System

This section gives elements defining the list of the safety functions that are subject to periodic tests.

For all the safety functions of a given system, periodic tests aim to verify that the safety criteria defined as design basis are still applicable during the entire plant lifetime. The periodic tests must be performed in required configurations, according to a predetermined frequency and method.

Step 2: Define the Criteria and Values Associated with the Safety Functions

For each safety function subjected to periodic testing, the related component must be listed, and the safety criteria associated to the safety function must be defined.

Step 3: Define the Periodic Test Criteria

In this step, the overarching system operating conditions are defined and justified, in order to check the criteria of several safety functions at the same time as far as practicable.

Several safety functions of a system can lead to similar criteria, one of which is the envelope. Should this occur, a synthesis must be done to obtain the most relevant envelope criterion.

Step 4: Define the Periodic Tests
In this step, the uncertainty calculation, initial condition, and frequency are defined for the periodic test.

1) Uncertainty Calculation

   All instruments used in the tests have uncertainty. The uncertainty must be determined to satisfy the test requirement based on the safety criteria.

2) Initial Condition

   A suitable operating mode is determined for each periodic test to be performed (consistent with the OTS).

3) Frequency

   The frequency is divided into three types, including calendar-based frequency, effective full power day, and event-based frequency.

c) Core Physics Tests

The main purpose of the core physics tests after core loading is to check consistency of the core against design studies.

For each core loading, nuclear design calculations have guaranteed the core physics parameters do not to exceed the safety values. Core physics tests can ensure the core is operated as designed can be ensured by checking whether the core physics parameters are consistent with design predictions or not.

The acceptance criteria of the core physics parameter measurements are defined in the Core Surveillance Requirements and will be developed in the nuclear site licensing phase.

After each refuelling outage, core physics tests are carried out at the first criticality and at various power levels to:

1) Check core physics parameters in accordance with design and validation of core safety analysis;

2) Check the correct operation of the ex-core detectors of the Nuclear Instrumentation System (RPN [NIS]);

3) Define calibration coefficients for Power Range Channels of the RPN [NIS];

4) Define the calibration curve for Rod Position Indication and Rod Control System (RGL [RPICS]): position setpoint for the turbine load.

Every test must be conducted in compliance with test procedures and standard guidelines, which specify test method, initial conditions, precautions and instructions. The test procedures and standard guidelines will be developed in the nuclear site licensing phase.

d) Interface between GDA and nuclear site licensing

The completeness analysis and core physics tests as described above form the interface between GDA and the nuclear site licensing phase.

A series of integrated analysis documents will be developed to cover all system safety functions. The documents will form the basis of the recommended PT procedures for the UK HPR1000, which will be developed in the nuclear site licensing phase.

### 31.6.2 In-Service Inspection

a) In-Service Inspection Requirements

ISI is a preventive maintenance process involving the use of Non-destructive Testing (NDT) for nuclear pressure mechanical components at scheduled intervals during operation. The ISI will be used to detect the anticipated degradation in good time before it compromises the structural integrity, and confirm the absence of unanticipated degradation that could lead to failure.

The ISI requirements will be determined based on the characteristics of mechanical components, the RSE-M code in Reference [18], the UK requirements and operating experience. The NDT requirements or inspection intervals for ISI may be improved subsequent to the process for characterisation, repair, mitigation and justification of defects identified by the NDT results. An effective ISI ensures:

1) The plant is constructed with good design and manufacture to assure compliance with the safety cases of nuclear pressure mechanical components.

2) The degradation can be identified to guarantee that any unacceptable defect will be detected before it grows to affect the safety of plant throughout the operating lifetime of the plant.

3) The effectiveness of electricity generation and operation of the plant can be guaranteed, through reductions in inspector exposure, inspection cost and inspection time, etc.

Pre-Service Inspection (PSI) will be carried out prior to reactor start-up, and performed with the same NDT techniques and equipment used for ISI. PSI provides the baseline for future ISI, including complete ISI and partial ISI performed during outages.

Additional inspection requirements for High Integrity Component (HIC) should be considered, which are based on the content of the Structural Integrity Assessment in Chapter 17. HIC is assigned to the components and structures whose failure is intolerable and for which no protection is provided or protection provision is not reasonably practicable. All HIC components will be included in the PSI/ISI programme and techniques for HIC components, and will be qualified according to the European Network for Inspection Qualification (ENIQ) methodology in Reference [19].

b) Effective ISI is assured through ‘design for inspectability’

During the design and construction stages, access and inspectability for the implementation of PSI/ISI will be provided. For effective ISI, the mechanical components should be designed, manufactured and assembled so that all of the welds can be accessed and inspected.

c) Interface between GDA and nuclear site licensing

The outline of the PSI/ISI requirements will be produced during GDA along with a review of how the design and manufacturing requirements of HIC components facilitates effective NDT, which is described in the Structural Integrity Assessment in Chapter 17.

The detailed PSI/ISI programme will be defined in the nuclear site licensing phase.

### 31.6.3 Maintenance

The maintenance procedure for the UK HPR1000 covers all preventive and corrective measures, both administrative and technical, that are necessary to detect and mitigate degradation of functioning SSC or to restore a failed SSC to an acceptable level of performance.

The purpose of maintenance activity is also to enhance the reliability of equipment. The range of maintenance activities includes servicing, overhaul, repair and replacement of parts, and often, as appropriate, tests, calibration and inspection.

a) Maintenance Types

The relevant activities in a nuclear power plant can be divided into preventive and corrective maintenance. A considerable part of all maintenance activities is performed while the plant is shut down; however, maintenance may be planned and executed under power operation in which adequate defence in depth is maintained.

1) Preventive maintenance

Maintenance involves all technical, administrative and management actions during the service life cycle of an item of equipment, in order to maintain it in, or restore it to, a state in which it can carry out the function that it is required to perform. Preventive maintenance involves all actions carried out on an item of equipment to reduce the probability of its operational failure.

The aim of preventive maintenance is to ensure that, throughout the service life of the plant, the objectives of safety, availability and cost are achieved, subject to the requirements of ALARP, while complying with applicable rules for the environmental protection, staff safety, radiological protection and other regulations.

Preventive maintenance includes periodic, predictive and planned maintenance activities performed prior to failure of an SSC so as to maintain its service life by controlling degradation or preventing its failure.

The feasibility of preventive maintenance is considered in the design stage.

2) Corrective maintenance

Corrective maintenance includes actions that, by means of repair, overhaul or replacement, restore the capability of a failed SSC to perform its defined function within the acceptance criteria.

b) Maintenance Requirements

The UK HPR1000 maintenance safety requirements include the following considerations:

1) The conditions for carrying out maintenance must take into account the radiological protection for the staff and the public. The process of maintenance activities also takes environmental protection into consideration.

2) Maintenance activities must take human factors into consideration and technical and administrative controls shall be established to avoid human errors as far as possible.

3) Maintenance activities are guided by procedures, and the risks throughout the maintenance activities are clarified in the procedures.

c) Maintenance Strategy

The availabilities of SSC important to safety are identified, on which appropriate maintenance strategies are applied.

A safety function is normally considered as inoperable when the related equipment is taken out of service or isolated for maintenance. However, if the nature of the maintenance on equipment is such that the equipment may be made operational within a suitable timeframe such that the function can be performed if demanded, the safety function may be considered as operable.

In the design process of frequency and extent for preventive maintenance, the following aspects are considered, Reference [20]:

1) The importance of SSC to safety;

2) Recommendations made by the designers and vendors;

3) Relevant experience available;

4) Results of condition monitoring;

5) The probability of failure to function properly;

6) On-line maintenance;

7) The necessity of maintaining radiological doses as low as reasonably practicable.

Maintenance procedures are performed and optimised according to the Reliability Centred Maintenance (RCM) approach. The reliability data for the approach are from manufacturers or operating experience feedback. The achievement and cost of maintenance are balanced, and over-maintenance is avoided according to the requirements of ALARP.

In conclusion, the maintenance strategy contributes towards the achievement of reliability objectives for the plant by maintaining the required safety level. The requirements of frequency and activities for SSC are addressed in the maintenance procedures. The detailed maintenance procedures will be developed during the nuclear site licensing phase.

d) Maintenance Windows

Maintenance activities may affect safety functions required by the safety analysis. Therefore, the safety function's availability and redundancy required in the safety analysis are considered when a maintenance window is defined.

The general principles to define maintenance windows are given as below:

1) Unavailability caused by preventive maintenance, has been considered in the safety analysis for design basic condition (DBC)-2/3/4. Generally, the preventive maintenance activity cannot be carried out unless the safety function's availability and redundancy requirements in the DBC-2/3/4 safety analysis are satisfied.

2) The Design Extension Condition (DEC) analysis does not consider the single failure criteria and loss of offsite power. Therefore, preventive maintenance on redundant trains could be carried out. If SSC including one or more redundant train are designed only for DEC, preventive maintenance on one is allowed in any operating mode;

3) Because the function of a supported system is affected by the supporting system (power supply, cooling water, ventilation system, etc.), preventive maintenance for both the supported and supporting systems, is carried out simultaneously;

4) For SSC unrelated to safety functions, preventive maintenance could be carried out in any condition when the maintenance does not affect plant normal operation.

e) Interface between GDA and nuclear site licensing

The licensee is responsible for putting in place adequate and appropriate arrangements for the development, implementation, management and update of the full EMIT procedure, which will be developed in the nuclear site licensing phase.

### 31.6.4 Ageing and Degradation

An important part of maintaining plant safety is the detection of ageing effects on SSC, which enables associated reductions in safety margins to be addressed and corrective actions to be implemented before loss of integrity or functional capability occurs, Reference [21].

The general design requirements of ageing and degradation have been provided in Chapter 4. The processes for system design also consider the ageing and degradation of SSC. The monitoring and management of SSC degradation is required to ensure that the safety functions of those SSC can be ensured as part of nuclear power plant operational management.

The material properties and parameters of SSC are measured within the stipulated time. ISI and periodic tests help to identify degradation before failure.

The ageing and degradation of SSC important to safety are managed by the ageing management procedure.

The licensee is responsible for the development, implementation, management and update of the ageing management procedure, which will be developed in the nuclear site licensing phase.

## 31.7 ALARP Assessment

As part of the ALARP strategy presented in PCSR Chapter 33, a holistic review of the UK HPR1000 design against applicable operational management RGP is required to identify potential improvements.

### 31.7.1 Identification of Relevant Good Practice

The following codes and standards are the sources of RGP for operational management:

a) IAEA Safety Standards

1) IAEA, Safety Requirements: Safety of Nuclear Power Plants: Design, No.SSR-2/1, Rev. 1, February 2016;

2) IAEA, Safety Requirements: Safety of Nuclear Power Plants: Commissioning and Operation, No.SSR-2/2, Rev. 1, February 2016;

3) IAEA, Safety Guide: Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, No.NS-G-2.2, November 2000;

4) IAEA, Safety Guide: Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants, No.NS-G-2.6, October 2002;

5) IAEA, Safety Guide: Ageing Management for Nuclear Power Plants, No.NS-G-2.12, January 2009;

6) IAEA, Specific Safety Guide: Chemistry Programme for Water Cooled Nuclear Power Plants, No. SSG-13, January 2011;

7) IAEA, Safe Reports Series No. 48: Development and Review of Plant Specific Emergency Operating Procedures, 2006.

b) Recognised Design Codes and Standards

1) AFCEN, In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands, RSE-M, 2010 edition+2012 addendum, 2010, 2012;

2) JRC, European Methodology for Qualification of Non-Destructive Testing, THIRD ISSUE, ENIQ Report No.31, EUR 22906EN-2007.

c) WENRA, Safety Reference Levels for Existing Reactors, September 2014.

d) US NRC, Human Factors Engineering Program Review Model, NUREG-0711, Rev. 3, November 2012.

### 31.7.2 Holistic ALARP Review

The main aspects of the above RGP applied to the operational management design for the UK HPR1000 are shown in T-31.7-1.

T-31.7-1 Holistic ALARP Review against the UK HPR1000

| No. | RGP | The UK HPR1000 design | Gap Analysis |
|---|---|---|---|
| **General to Operational Management** | | | |
| 1 | − IAEA, SSR-2/1, Rev.1, 2016, requirement 28/29/31;<br>− IAEA, SSR-2/2, Rev.1, 2016, requirement 6/14/26/31;<br>− WENRA, Safety Reference Levels for Existing Reactors, 2014, issue H/I/K/L/M.<br>Operational management contains operating procedures, operating limits and conditions, EMIT and ageing degradation. | In the UK HPR1000, the aspects of operational management include operating procedures, operating limits and conditions, EMIT and ageing degradation. | No gap.<br>The scope of operational management refers to Sub-chapter 31.2. |
| **Operating Procedures** | | | |
| 1 | − IAEA, SSR-2/2, Rev.1, 2016, Chapter 7;<br>− IAEA, NS-G-2.2, 2000, Chapter 8;<br>The scope and requirements of NOP. | In the UK HPR1000, NOP is drafted to ensure that the plant can be operated within the operating limits and conditions, including Unit Operating Procedure, System Operating Procedure, System Alarm Sheet, and Abnormal Operating Procedures. | No gap.<br>The development of NOP refers to Sub-chapter 31.4.1 and 31.4.4. |

| No. | RGP | The UK HPR1000 design | Gap Analysis |
|---|---|---|---|
| 2 | − IAEA, Safe Reports Series No. 48, 2006, Chapter 2/3;<br>− WENRA, Safety Reference Levels for Existing Reactors, 2014, issue L/M.<br>The scope, design process and requirements of EOP. | In the UK HPR1000, there is an integrated design process of EOP. | No gap.<br>The development of EOP refers to Sub-chapter 31.4.2 and 31.4.4. |
| 3 | − US NRC, NUREG-0711, 2012, Section 9.<br>The development of UOP, EOP and SAMG. | Procedure development for the UK HPR1000 fulfils the requirements of NUREG-0711, Section 9. | No gap.<br>The development of operating procedures refers to Sub-chapter 31.4.4. |

| No. | RGP | The UK HPR1000 design | Gap Analysis |
|---|---|---|---|
| **Operating Limits and Conditions** | | | |
| 1 | − IAEA, SSR-2/1, Rev.1, 2016, Chapter 5;<br>− IAEA, SSR-2/2, Rev.1, 2016, Chapter 4;<br>− IAEA, NS-G-2.2, 2000, Chapter 6.<br>− IAEA, SSG-13, 2011, Chapter 3 and 4.<br>The scope and requirements of operating limits and conditions. | In the UK HPR1000, the design of operating limits and conditions (operating rules) complies with the requirements in SSR-2/1 (Rev.1), SSR-2/2 (Rev.1), NS-G-2.2 and SSG-13. | No gap.<br>The development of operating limits and conditions refers to Sub-chapter 31.5. |
| 2 | − WENRA, Safety Reference Levels for Existing Reactors, 2014, issue H;<br>− IAEA, NS-G-2.2, 2000, chapter 6.<br>Normal operating limits and conditions and measures to be taken for non-compliance. | Normal operating limits and conditions as well as the measures to be taken for non-compliance are specified in the OTS of the UK HPR1000. | No gap.<br>The development of normal operating limits and conditions refers to Sub-chapter 31.5.1. |

| No. | RGP | The UK HPR1000 design | Gap Analysis |
|---|---|---|---|
| **EMIT and Ageing** | | | |
| 1 | − IAEA, SSR-2/1, Rev.1, 2016, Chapter 5;<br>− IAEA, SSR-2/2, Rev.1, 2016, Chapter 8;<br>− IAEA, NS-G-2.6, 2002;<br>− WENRA, Safety Reference Levels for Existing Reactors, issue K.<br>The design and implementation requirements of EMIT. | In the UK HPR1000, the design and implementation process complies with the requirements for EMIT in SSR-2/1 (Rev.1), SSR-2/2 (Rev.1), NS-G-2.6 and Safety Reference Levels for Existing Reactors. There are different EMIT programmes for the UK HPR1000. | No gap.<br>The development of EMIT refers to Sub-chapter 31.6. |
| 2 | − IAEA, SSR-2/1, Rev.1, 2016, chapter 5;<br>− IAEA, SSR-2/2, Rev.1, 2016, chapter 4;<br>− IAEA, NS-G-2.12, 2009, chapter 4;<br>− WENRA, Safety Reference Levels for Existing Reactors, 2014, issue I.<br>Ageing management in operation. | In the UK HPR1000, ageing management complies with the requirements in SSR-2/1 (Rev.1), SSR-2/2 (Rev.1), NS-G-2.12 and Safety Reference Levels for Existing Reactors. | No gap.<br>The management of ageing refers to Sub-chapter 31.6.4. |

| No. | RGP | The UK HPR1000 design | Gap Analysis |
|---|---|---|---|
| 3 | − Recognised Design Codes and Standards AFCEN, 2010, 2012, RSE-M. | In the UK HPR1000, the recognised design codes and standards RSE-M are used. | The version of RSE-M in the UK HPR1000 is not the latest version, which is discussed in Chapter 17. |
| 4 | − Previous GDA experience: ENIQ, 2007. methodology for inspection qualification | The inspection qualification for PSI/ISI of HIC components in the UK HPR1000 are referred to ENIQ methodology. | No gap. |

### 31.7.3 Conclusions from ALARP Review

This section complies with the UK and internationally recognised regulatory codes and standards, incorporates experience feedback and good practice from the Chinese Pressurised Reactor and other similar plants under construction and operation. It is considered that the approach adopted complies with international and UK RGP in this chapter. No gap has been identified at this stage.

## 31.8 Concluding Remarks

The development of operational management documents has been outlined in this chapter, and the detailed information will be developed in the nuclear site licensing phase. The UK HPR1000 operational management design will also consider good practice from the nuclear power plants which are in operation or under construction in China. In summary, the considerations of operational management are taken in the design phase to ensure that plant operations are compliant with safety requirements. Therefore, the safety aspects of operation and management including operating procedures, operating limits and conditions, EMIT, ageing and degradation have been substantiated through the arrangement of these aspects which the principle, content, and methodology are discussed in this chapter.

## 31.9 References

[1] ONR, Generic Design Assessment Guidance to Requesting Parties, ONR-GDA-GD-001, Rev. 2, June 2016.

[2] ONR, Safety Assessment Principles for Nuclear Facilities, Rev. 0, November 2014.

[3] CGN, Definition of Plant Normal Operating Domain, GHX11100001DOZJ45GN, Rev. B, 2018.

[4] CGN, Definition of Normal Operating Modes and Corresponding Parameters, GHX71100001DOYX03GN, Rev. B, 2018.

[5] WENRA, Safety Reference Levels for Existing Reactors, September 2014.

[6] US NRC, Human Factors Engineering Program Review Model, NUREG-0711, Rev. 3, November 2012.

[7] CGN, Procedure Development Implementation Plan, GHX70000001DOYX03GN, Rev. A, 2017.

[8] CGN, Normal Operating Procedure Principle and Content, NE15BWXYX0000000013, Rev. B, 2018.

[9] CGN, Emergency Operating Procedure Principle and Content, NE15BWXYX0000000020, Rev. B, 2018.

[10] IAEA, Safe Reports Series No. 48: Development and Review of Plant Specific Emergency Operating Procedures, 2006.

[11] IAEA, Safety Guide: Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, No.NS-G-2.2, November 2000.

[12] IAEA, Safety Requirements: Safety of Nuclear Power Plants: Design, No.SSR-2/1, Rev. 1, February 2016.

[13] IAEA, Safety Requirements: Safety of Nuclear Power Plants: Commissioning and Operation, No.SSR-2/2, Rev. 1, February 2016.

[14] CGN, Operating Technical Specifications Methodology, NE15BWXYX0000000012, Rev. B, 2018.

[15] IAEA, Specific Safety Guide: Chemistry Programme for Water Cooled Nuclear Power Plants, No. SSG-13, January 2011.

[16] CGN, Design Transient Specification, GHX11300001DOZJ45GN, Rev. C, 2018.

[17] CGN, Periodic Test Design Methodology, NE15BWXYX0000000021, Rev. B, 2018.

[18] AFCEN, In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands, RSE-M, 2010 edition+2012 addendum, 2010, 2012.

[19] JRC, European Methodology for Qualification of Non-Destructive Testing, THIRD ISSUE, ENIQ Report No.31, EUR 22906EN-2007.

[20] IAEA, Safety Guide: Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants, No.NS-G-2.6, October 2002.

[21] IAEA, Safety Guide: Ageing Management for Nuclear Power Plants, No.NS-G-2.12, January 2009.