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Chapter Pilot: A. L'HUILLIER		
Name/Initials <i>A. Lhuillier AL</i> Date <i>14-11-12</i>		
Approved for EDF by: A. MARECHAL	Approved for AREVA by: G. CRAIG	
Name/Initials <i>A. Se. Maehal</i> Date <i>15-11-2012</i>	Name/Initials <i>G. Craig</i> Date <i>15-11-2012</i>	

### REVISION HISTORY

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00	First issue for INSA information	18/12/2007
01	Integration of technical and co-applicant review comments	30-04-2008
02	PCSR June 2009 update: – Clarification of text	27-06-2009
03	Issue for INSA review (excluding section 4)	09-10-2009
03 Rev 1	Issue for INSA review	15-10-2009
04	PCSR Step 4 submission: – Sections 3, 4, 5 and 6.1 added covering normal operation procedures, operating limits, periodic tests and in-service inspection – Section 1 updated to reflect changes	28-11-2009
05	Consolidated Step 4 PCSR update: - Significant changes to all sections: - Section 1 amended to reflect change of content of sub-chapter and to describe process for definition of design limits and conditions. - Section 2 modified to describe monophasic start-up. - Section 2.12 added along with associated figure and table - Section 4 modified to describe Design and Operating Limits; Section 4.1 re-written to describe generic principles for Operating Technical Specification production with associated figure. Description of FA3 and OL3 approaches to OTS production removed from Section 4.	31-03-2011

Continued on next page

<b>UK EPR</b>		
	Title: PCSR – Sub-chapter 18.2 – Normal Operation	
	<b>UKEPR-0002-182 Issue 06</b>	Page No.: II / V

<b>REVISION HISTORY</b>		
<b>Issue</b>	<b>Description</b>	<b>Date</b>
05 cont'd	Consolidated Step 4 PCSR update: <ul style="list-style-type: none"> <li>- Sections 4.2, 4.3 and 4.4 added covering Chemical and Radio-chemical Specifications, Loading Condition Accounting and Safety Analysis Bounding Limits and Fuel Design Limits with associated table</li> <li>- Section 5 re-written to describe generic principles for Periodic Testing production. Figure 1 added.</li> <li>- Section 6.1 re-written to provide more details on the development of the scope and detail of the PSA and In-Service Inspection programmes</li> <li>- Sections 6.2.5 and 6.2.6 added to discuss the development of the Maintenance schedule and the interface with future Licensees</li> </ul>	
06	Consolidated PCSR update: <ul style="list-style-type: none"> <li>- References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc</li> <li>- Minor editorial and typographical changes</li> <li>- Addition of cross-references to other PCSR sub-chapters (§1, §2.12.1, §3.1, §3.2, §6.2.4.1)</li> <li>- Text updated concerning the applicability of the OTS to all relevant plant operations (§4.1.2.2, §4.1.2.4)</li> <li>- Section 4.2.5 updated to describe the chemistry philosophy for auxiliary systems including chemical conditioning, chemical specifications and the role of chemistry/radiochemistry during normal operation and transients</li> <li>- Minor updates and addition of cross-references to Sub-chapter 18.1 (§1, §2.12.1, §3.1, §3.2 and §6.2.4.1).</li> <li>- Text updated regarding removal of the reactor cavity cover slabs (§2.2)</li> <li>- Text updated regarding in-service inspection (§6.1.2, §6.1.3) and addition of new §6.1.5 “Main Secondary System In-Service Inspection”.</li> <li>- Text updated regarding the preventive maintenance of DVL/DEL systems and the use of Steam Generator nozzle dams in reactor state D (§6.2.3.3).</li> </ul>	15-11-2012

<b>UK EPR</b>		
	Title: PCSR – Sub-chapter 18.2 – Normal Operation	
	<b>UKEPR-0002-182 Issue 06</b>	Page No.: III / V

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#### **For information address:**



AREVA NP SAS  
Tour AREVA  
92084 Paris La Défense Cedex  
France



EDF  
Division Ingénierie Nucléaire  
Centre National d'Equipement Nucléaire  
165-173, avenue Pierre Brossolette  
BP900  
92542 Montrouge  
France

<b>UK EPR</b>		
	Title: PCSR – Sub-chapter 18.2 – Normal Operation	
	<b>UKEPR-0002-182 Issue 06</b>	Page No.: IV / V

## TABLE OF CONTENTS

- 1. INTRODUCTION**
- 2. PRINCIPLES OF NORMAL OPERATION**
  - 2.1. GENERAL REMARKS**
  - 2.2. REACTOR SHUTDOWN**
  - 2.3. DRAINING AND OPENING THE PRIMARY SYSTEM**
  - 2.4. CORE UNLOADING**
  - 2.5. CORE RELOADING**
  - 2.6. CLOSING AND FILLING THE PRIMARY COOLANT SYSTEM**
  - 2.7. HEATING THE PRIMARY COOLANT**
  - 2.8. FROM HOT SHUTDOWN TO POWER OPERATION**
  - 2.9. POWER OPERATION – LOAD FOLLOWING**
  - 2.10. STRETCH-OUT OPERATION**
  - 2.11. SPECIFIC OPERATIONS**
  - 2.12. OPERATING PROCEDURES, STANDARD REACTOR STATES AND OPERATING LIMITS**
- 3. NORMAL OPERATING PROCEDURES**
  - 3.1. NORMAL OPERATING DOCUMENTS**
  - 3.2. INTERFACES WITH OPERATING PROCEDURES FOR ABNORMAL OPERATION**
- 4. DESIGN AND OPERATING LIMITS AND CONDITIONS**
  - 4.1. OPERATING TECHNICAL SPECIFICATIONS**
  - 4.2. CHEMICAL AND RADIOCHEMICAL SPECIFICATIONS**
  - 4.3. LOADING CONDITIONS ACCOUNTING**
  - 4.4. SAFETY ANALYSIS BOUNDING LIMITS AND FUEL DESIGN LIMITS**

<b>UK EPR</b>		
	Title: PCSR – Sub-chapter 18.2 – Normal Operation	
	<b>UKEPR-0002-182 Issue 06</b>	Page No.: V / V

**5. PERIODIC TESTING**

**5.1. GENERIC PRINCIPLES FOR PT PRODUCTION**

**5.2. GENERIC PROCESS OF PT PRODUCTION**

**5.3. DEFINITION OF RULES AND REQUIREMENTS FOR PT**

**5.4. CORE PHYSICS TESTS (RELOADING TESTING)**

**5.5. GDA/LICENSEE DOCUMENTATION BOUNDARY**

**6. IN SERVICE INSPECTION AND MAINTENANCE**

**6.1. IN-SERVICE INSPECTION**

**6.2. MAINTENANCE PROGRAMME**

## **SUB-CHAPTER 18.2 – NORMAL OPERATION [REF-1]**

### **1. INTRODUCTION**

The PCSR aims to demonstrate that the UK EPR achieves the fundamental objective that the radiological risk to workers and the public is as low as reasonably practicable, which is the basic legal requirement underpinning UK nuclear safety regulations.

The protection of the public and the environment towards radioactive substances is ensured by three successive barriers: the fuel cladding, the reactor coolant pressure boundary and the containment building. Provisions apply to each barrier according to the defence in depth concept to avoid its failure and limit the consequences of its failure, if any. The PCSR presents a detailed description of these provisions. It presents a description of the architecture of the EPR systems, their safety functions, robustness and availability requirements, and an explanation of the design codes and standards that are used in the design. Fault analyses are presented to demonstrate the adequacy between the design and all potential faults threatening the integrity of one of the three barriers. They include the Design Basis Analysis (PCC - see Sub-chapters 14.0 to 14.7), analysis of multiple failure events (RRC-A and RRC-B - see Sub-chapters 16.1 to 16.4), Primary and Secondary Overpressure analysis (see Sub-chapter 3.4, section 1.5), in-containment mass and energy release analysis (see Sub-chapter 6.2 and Appendix 6A) and Probabilistic Safety Analysis (see Chapter 15). The PCSR aims to demonstrate prior to commencement of construction, that sufficient analysis and engineering substantiation has been performed to give high confidence that the declared safety objectives of the UK EPR are met.

The PCSR contains assumptions and requirements necessary to ensure the results claimed in the overall safety case. Operating documents will have to be defined in order to ensure the plant is operated consistently with these assumptions and requirements. This sub-chapter outlines the methods that will provide design limits and conditions for the UK EPR. It details the arrangements for moving the PCSR to an operating regime which will ensure that the requirements and assumptions contained in the PCSR are captured in operating documents.

There are several sources of parameters and values in the PCSR which form the design limits and conditions:

- Regarding systems design, claims typically are made on structural integrity in term of loading conditions systems will have to face (thermal hydraulics conditions in circuits and buildings, nature and number of transients to meet) and chemical provisions in circuits.
- Regarding faults, claims are made for each plant state on thermal-hydraulic conditions in circuits, systems performance, system availability, neutronic parameters, etc.

Depending on the nature of these parameters, they will be captured in different operating documents, as described in this sub-chapter.

The information presented in PCSR Sub-chapter 14.1 details the plant characteristics that apply to accident analyses. These characteristics, which are specific to a particular accident analysis, are specified in the section describing the accident analysis. That information establishes some preliminary limits and conditions applicable at this stage of the EPR design.

Information such as plant initial conditions is reflected in OTS documents and system performance characteristics which are confirmed by periodic testing.

By presenting, in this sub-chapter, the process steps followed in the design of the EPR and the parameter values, such as initial conditions and system performance requirements, a link is made between the assumptions and requirements included in the EPR design and the information provided to the Licensee, who will be responsible for defining and developing operating documents to ensure the plant is operated consistently with these assumptions and requirements. The on-going role of the Architect-Engineer in support of the Licensee is described in Chapter 21. Human Factors considerations relating to operating documents are discussed in Sub-chapter 18.1.

The detailed analysis to capture the safety limits in the operational documentation is not fully complete and will be provided in the NSL phase in a format that allows the required safety limits to be compared with the operational documentation, and presents the rationale for choosing one or the other type of operational document.

One of the key elements of the EPR design approach is the use of safety features. Safety features are a set of Structures, Systems and Components (SSCs) of the same kind (I&C, mechanical, support systems) contributing to a safety function. A safety function is a safety objective which is independent from the SSCs that provide it (see Sub-chapter 3.2). The functional analysis [Ref-1] of the requirements on these safety features provides a basis on which many of the operational documents will be developed. Information on system functional requirements for individual systems is presented in relevant sub-chapters of the PCSR.

During normal operation the plant must be operated:

- to manage normal scheduled operating transients and certain specific operations involving unplanned events,
- in a manner consistent with the assumptions of the safety case, i.e. within the safety limits established for the plant.

The first objective is achieved using the principles and procedures for normal operation described in sections 2 and 3 of this sub-chapter, respectively.

The second objective is achieved by establishing operating rules applicable in normal operation. The operating rules are developed in the following documentation:

- Operating Technical Specifications (OTS)

The general objective of the OTS is to set out the rules which must be observed during normal operation of the nuclear plant in order to keep it within the operating envelope justified by the safety analyses presented in the safety report.

The OTS are described in section 4 below.

- Chemical and Radio-chemical Specifications

Chemical and radiochemical parameters are controlled and monitored in order to ensure that the safety analyses presented in the safety report are complied with. These parameters are principally related to control of coolant activity, material structural integrity (particularly for the pressure boundary), fuel performance and integrity, limitation of out of core radiation fields and any releases as a result of fault conditions.

The Chemical and Radiochemical Specifications are described in section 4 below.

- Accounting of Loading Conditions

The mechanical design of pressurised nuclear equipment is based on the analysis of primary and secondary component integrity and piping against different damage mechanisms. This demonstration of integrity must take into account normal operating conditions, incident operating conditions and accident and post-accident conditions. Accounting of loading conditions is a process for counting of transients in order to verify that the number of loading conditions taken into account for mechanical design of pressurised nuclear equipment is not exceeded.

The accounting of loading conditions is described in section 4 below.

- Safety Analysis Bounding Limits and Fuel Parameters

The bounding values of neutronic parameters defined in the nuclear design section (see Sub-chapter 4.3) are presented in a schedule of the safety analysis bounding limits along with the assessment methodology for these neutronic parameters. Typical key limits captured in these are bounding neutronic parameters taken into account in fault studies. Some key limits vary with fuel load. Key limits associated with a particular fuel load must be verified by the Licensee to ensure that they remain within the safety analysis bounding limits.

The safety analysis bounding limits and fuel design limits are described in section 4 below.

- Periodic Tests (PTs)

The objective of PTs is to verify that, for the safety functions, the safety criteria defined at the design stage are achieved over the whole of the plant operating lifetime. The tests are carried out according to preset frequencies, procedures and plant configurations.

PTs are described in section 5 below.

- In-service inspection and maintenance tests

Maintenance is defined as the actions carried out to enable the systems, equipment and structures to be maintained in (or restored to) a state in which they can perform their safety duty (consistent with the Operating Technical Specifications), while meeting availability and ALARP objectives, and complying with the rules for environmental protection, personnel safety, radiological protection and other regulatory requirements, over the lifetime of the plant.

The in-service inspection and maintenance tests are identified in section 6 below.



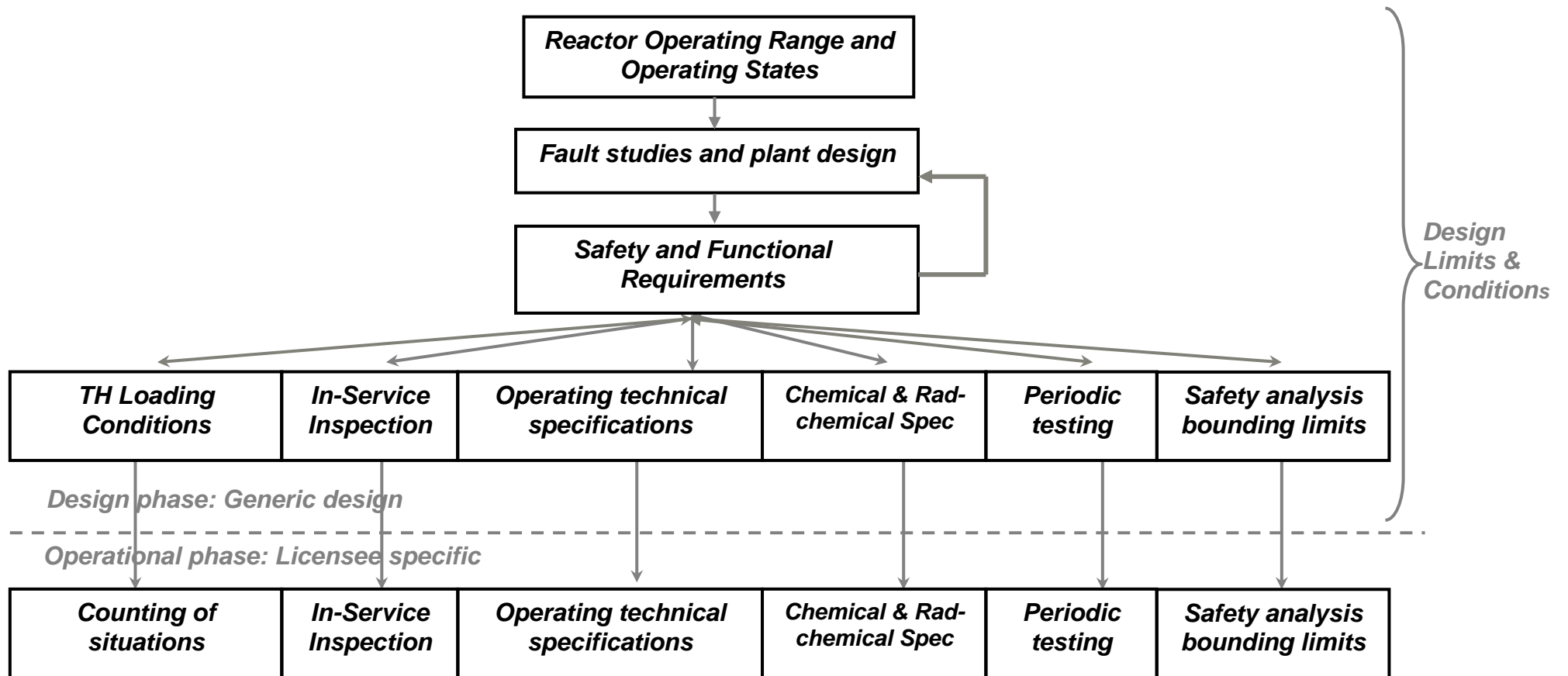
The safety case in the PCSR provides the main input data for establishing the operating documents that will need to be put in place before the plant is commissioned and placed into service. As the operating documents are operator dependent, they will be submitted by the Licensee as part of the Nuclear Site Licensing (NSL) process.

Section 18.2.1 - Figure 1 illustrates the generic design phase sources of limits and conditions and indicates the interface between the design phase and the NSL phase. It does not present the interfaces between operational documentation because of the complexity of such interfaces and dependence on the format chosen by the Licensee.

The Licensee phase documentation detailed in Section 18.2.1 - Figure 1 will form the core of the operating rules that will apply to EPR operation in the UK, supported by operating procedures and the Licensee organisational structure. These operating rules are Licensee specific and will not be discussed in this report.

**SECTION 18.2.1 - FIGURE 1**

EPR Limits and Conditions: Identification of the links between operational documentation



## 2. PRINCIPLES OF NORMAL OPERATION

### 2.1. GENERAL REMARKS

Normal operation comprises:

- power operation and normal scheduled operating transients such as increases in load, reductions in load, plant shutdown or start-up,
- specific operations due to unplanned events, such as house load operations or loss of power sources.

With the exception of refuelling shutdowns, the plant can be shut down for long or short periods for operations involving maintenance or repair, fuel saving or power grid management. The shutdown mode, i.e. hot shutdown or RIS/RRA [SIS/RHRS] shutdown, will depend on the nature of the intervention and on the shutdown duration. During prolonged operation at hot shutdown, the boron concentration in the primary circuit, which ensures the required shutdown margin, is adjusted according to the fuel burnup and the duration of the shutdown.

A switchover to cold shutdown is made to carry out refuelling or to enable maintenance or repair operations which require the unit to be at cold shutdown.

The main operating principles are set out below in chronological order, from reactor shutdown at the end of the fuel cycle through to power operation at the beginning of the following fuel cycle. Operation with an extended cycle is also described.

### 2.2. REACTOR SHUTDOWN

The initial mode considered is the state of the reactor during power operation at the end of a fuel cycle. Unit shutdown begins with a reduction in the turbine load. The power level for disconnection from the grid and turbine shutdown will depend on the turbine generator unit chosen. The control rod system is switched from "average temperature control" mode to "flux level control" mode at 25% nominal power. The turbine can be disconnected from the grid between 25% and 12% nominal power. The load is then automatically transferred to the turbine bypass system (GCT [MSB]).

The control rods are inserted manually to shut down the reactor. The primary coolant temperature is controlled by the turbine bypass system. The steam generators continue to be fed and their water level is controlled by the start-up and shutdown systems (AAD [SSS]) and the feedwater flow control system (ARE [MFWS]). The primary system is borated to maintain the required shutdown margin. Hot shutdown tests and inspections are carried out.

Whilst cooling, the turbine is on its barring gear.

The primary system is then cooled to approximately 120°C by the GCT [MSB], using the four primary pumps to circulate primary fluid through the steam generator tubes. The maximum cooling rate is 50°C/hr. At the same time, the primary pressure is reduced by normal pressuriser spray to approximately 25 bar, whilst maintaining the required sub-cooled margin. An automatic sequence ensures simultaneous cooling and depressurisation of the primary system and boration is carried out in parallel. At 120°C and 25 bar, the RIS [SIS] trains 1 and 4 are connected and started up in RRA [RHRS] mode to continue the RCP [RCS] cooling. The turbine bypass system may be isolated and the feed-water plant stopped and cooled in preparation for maintenance operations. Two primary reactor coolant pumps are shut down (RCP\_1110 and 4110).

The first three Reactor Cavity Cover Slabs are removed when the RCP [RCS] is below 120 °C.

Below 120°C and during the primary circuit cooling, the pressuriser level is increased to reach water solid phase. Primary pressure control is then ensured by the RCV [CVCS] and the pressuriser is cooled.

When the primary temperature is less than 100°C, RIS [SIS] trains 2 and 3 can be connected and started up in RRA [RHRS] mode in order to increase the primary system cooling capacity. The RCV [CVCS] letdown line is connected to the RIS/RRA [SIS/RHRS] system. The final three Reactor Cavity Cover Slabs are removed.

At 90°C, the equipment hatch may be opened to enable tools and equipment to be taken into containment.

The reactor coolant oxygenation is performed by injection of hydrogen peroxide in the RCP [RCS], as soon as the coolant and pressuriser temperatures are respectively below 80°C and 120°C.

The number of primary reactor coolant pumps in operation is adjusted for efficient cooling and to allow coolant purification after the oxygenation. The last primary pump (number 3 so as to keep normal pressuriser spray available) is shut down when the radiochemical criteria are met. The primary system is maintained at about 55°C.

Throughout the primary system cooling process, contraction is compensated by RCV [CVCS] charging pumps and REA [RBWMS] pumps, which draw boric acid and demineralised water from the storage tanks. Boration is continued until the required boron concentration during cold shutdown for fuel reloading is achieved.

After the last reactor coolant pump has been stopped, the primary coolant, in water solid phase, is depressurised to atmospheric pressure.

### 2.3. DRAINING AND OPENING THE PRIMARY SYSTEM

One RIS [SIS] train in RRA [RHRS] mode is stopped prior to primary system draining.

The primary system is drained to  $\frac{3}{4}$  loop level by the RCV [CVCS] letdown line (via the RIS/RCV [SIS/CVCS] connection) and the excess volume of primary coolant is transferred to the TEP [CSTS] storage tanks for recycling. Adjusting the water level to  $\frac{3}{4}$  loop level prevents uncovering the core and ensures safe RIS [SIS] operation in RRA [RHRS] mode.

Before opening the primary system, the RCP [RCS] is swept by injecting nitrogen via the reactor coolant pumps and the vessel head vent, and venting is carried out by the vacuum pump linked to the pressuriser vent. The RCP [RCS] is then air-swept. The RCP [RCS] water level is maintained and controlled automatically at  $\frac{3}{4}$  loop level before opening the RCP [RCS].

The electrical connections of the rod control system mechanisms and core instrumentation are removed. The mechanical seals are removed, thereby opening the primary system. After the thermal insulation of the vessel head has been removed, the multi-stud tensioning machine is positioned to carry out vessel head opening operations.

## 2.4. CORE UNLOADING

Whilst the vessel head is being removed, the compartments of the reactor refuelling cavity are filled with borated water from the IRWST by a LHSI pump. When the cavity is full, the core instrumentation, including the lances of the system for measuring neutron flux using aeroballs, is removed and the rod control system mechanisms are disconnected. When the internals are withdrawn and placed in the reactor internal storage pool, fuel unloading operations can begin (approximately 70 hours or less, after the uncoupling) using the handling devices (loading machine, transfer tube, fuel handling crane). The primary coolant temperature is maintained below 50°C by the RIS [SIS] in RRA [RHRS] mode. Fuel unloading takes approximately 40 hours. When the reactor pool has been filled, an electrical train may be declared unavailable, to enable maintenance.

The decay heat of the unloaded fuel elements adds to that of the fuel elements already stored in the fuel building spent fuel pool. Thus, the second cooling train of the fuel pool cooling (and purification) system (PTR [FPC(P)S]) must be started up to maintain the pool temperature below 50°C (from the beginning of core unloading to the end of reloading).

Once the core is unloaded, two electrical trains may be declared unavailable to enable maintenance. Depending on the scheduled work, the sluice gate between the vessel compartment and the internals storage compartment is placed in position. The RCP [RCS] can be drained, if needed, into the IRWST to the level of the steam generator plena using the PTR [FPC(P)S] purification pumps. Maintenance tasks can be carried out in this 'Fully Unloaded Reactor' state: inspection of steam generator tubes and valve maintenance. In the fuel building, changing of rod cluster control assemblies is carried out.

## 2.5. CORE RELOADING

After closing the primary components (i.e. steam generator manways), the vessel refuelling cavity is filled with borated water from the IRWST using the LHSI pumps. The sluice gates are then removed, the transfer tube opened and the fuel is loaded into the vessel by means of the handling devices (fuel handling crane, transfer tube, refuelling machine). The primary temperature is maintained below 50°C by the RIS [SIS] in RRA [RHRS] mode. Core loading and mapping operations last approximately 45 hours.

Once the loading is completed, the transfer tube is closed. The upper internals are put back in position, the rod cluster control assemblies reconnected, the aeroball lances inserted and the core instrumentation installed.

## 2.6. CLOSING AND FILLING THE PRIMARY COOLANT SYSTEM

The reactor building pool compartments are drained down to the vessel flange, the water being transferred to the IRWST using the purification pumps, demineralisers and filters of the PTR [FPC(P)S].

The bottom of the vessel refuelling cavity and the vessel casing flange are cleaned. The reactor vessel is closed by the multi-stud tensioning machine. The vessel head penetration seals are re-assembled and the vessel head vent closed. The electrical connections of the rod control system mechanisms and core instrumentation are re-installed.

During these operations, the primary coolant temperature is controlled by the RIS [SIS] in RRA [RHRS] mode.

The primary system is then drained down to  $\frac{3}{4}$  loop level by the RCV [CVCS] letdown line (via the RIS/RCV [SIS/CVCS] connection) so that the gas-filled parts of the vessel, pressuriser and steam generator tubes are connected together. The RCP [RCS] level is automatically controlled by the RCV [CVCS] so as to prevent uncovering the core and ensure safe RIS [SIS] operation in RRA [RHRS] mode.

The vacuum pump is used to create a vacuum in the primary system. The resultant pressure is reduced to approximately 200 mbar to minimise the primary coolant air content.

The primary system is then filled by REA [RBWMS] makeup using the RCV [CVCS] pumps. The primary coolant is degassed at a high flow rate by means of the RCV [CVCS] and related systems. RCP [RCS] filling is stopped when the pressuriser level reaches about 90% of the range.

With regards to the conventional island, operations on the turbine, the generator and the transmission power systems are completed. The turbine generator unit is on its barring gear. Two steam generators are needed and filled to their zero load level. The main atmospheric steam relief line (VDA [MSRT]) is available. The feed-water plant is filled, a vacuum is created in the condenser, and the heating and chemical treatment of the feed-water plant begins.

## 2.7. HEATING THE PRIMARY COOLANT

After the vacuum pump has been shut off, the RCV [CVCS] pressure control on LP is activated, without exceeding a gradient of 4 bar/min, whilst the RIS/RRA [SIS/RHRS] flow rate is increased to its standard value. The residual gaseous phase in the pressuriser is evacuated and the RCP [RCS] goes into water solid mode. The RCP [RCS] pressure is increased by automatic control of the RCV [CVCS] letdown flow through the LP letdown valve. When the RCP [RCS] pressure reaches 25 bar the pressure is stabilised, the RIS/RCV [SIS/CVCS] connection is isolated, the pressure being sufficient to enable normal RCP/RCV [RCS/CVCS] letdown. The Primary reactor coolant pumps and the RIS/RRA (SIS/RHR) pumps are started up at a minimum RCS pressure of 25 bar. The spray valves are opened to homogenise the pressuriser.

After the reactor coolant pumps have started up, the RCP [RCS] is heated at stabilised pressure up to 90°C thanks to the power supplied by the four pumps and the fuel decay heat. The heating rate is limited to 40°C/hr. The LHSI heat exchangers are automatically by-passed. Nevertheless, as a minimum, LHSI trains 1 and 4 are connected to the RCP [RCS] in RIS/RRA [RIS/RHR] mode and can be used to maintain the rate of RCP [RCS] temperature rise below 40°C/h if necessary. At 90°C the RCP [RCS] temperature is stabilised and hydrazine is added to reduce the concentration of oxygen in the primary circuit. The bubble is then formed in the pressuriser. Once the pressuriser level is below 90%, the pressuriser pressure control in

two-phase mode is started. The RCP [RCS] heat-up is then restarted up to 120°C. Only LHSI trains 1 and 4 can be used for the control of the RCP [RCS] temperature above 100°C.

At the same time as this operation is being carried out on the primary circuit, the secondary circuit is made available. The condenser is under vacuum and the GCT [MSB] is available. The feed-water plant is heated and chemically treated. Above 120°C, the four steam generators are required to be available and can be supplied by the AAD [SSS].

When the RCP [RCS] temperature reaches 120°C, the last two RIS [SIS] trains (trains 1 and 4) still connected in RRA [RHRS] mode are isolated. Temperature control is then ensured via the steam generators (GCT [MSB] and AAD [SSS]).

During the heatup process, the excess coolant volume due to primary expansion is drawn off through the RCV [CVCS] letdown line (automatic control of the pressuriser level) to the TEP [CSTS] storage tanks. At the same time, the pressure is progressively and automatically increased until it reaches hot shutdown conditions.

At hot shutdown the pressuriser level is set to the no load setpoint by the RCV [CVCS] letdown. The pressure is controlled using the pressuriser heaters and normal spray and the temperature is controlled by the steam generators. The steam generator levels are maintained by means of ARE [MFWS] very low load valves, and their pressure controlled by GCT [MSB].

The feed-water plant is available and in service. The turbine generator unit is on its barring gear.

## **2.8. FROM HOT SHUTDOWN TO POWER OPERATION**

In hot shutdown, various tests are carried out, such as measurements of rod drop time. The primary coolant temperature is automatically controlled by the GCT [MSB]. The primary coolant is diluted by injection of demineralised water from the REA [RBWMS] using the RCV [CVCS] charging pumps and tests at zero power are carried out. Power is then increased by controlling the flux level. The steam generators are supplied by the start-up and shutdown pump (AAD [SSS]), then by the feed-water pumps (APA [MFWPS]) using the normal feed-water flow control system (ARE [MFWS]).

The turbine is commissioned, the generator is connected to the main grid and power is gradually increased. At 25% nominal power, the rod control system is switched from "flux level control mode" to "Average temperature control mode". At this power level, all RCP [RCS] control loops are in automatic mode (see Sub-chapter 5.1) and power is gradually stepped up to 100%.

## **2.9. POWER OPERATION – LOAD FOLLOWING**

In basic operation, only long-term reactivity effects (fuel burnup, build up of samarium) need to be compensated for by gradually diluting the primary coolant to a boron concentration of approximately 5 to 10 ppm at the end of the fuel cycle.

If required for reasons of grid production and consumption balance, the power plant may have to reduce power and then resume full power production a few hours later (see load following and power variation in Sub-chapter 3.4). As previously discussed, steam generator control is entirely automatic. Rod control cluster assemblies are inserted or extracted by core regulation (temperature and power distribution control) to compensate for rapid changes in reactivity. Slow variations (Xenon changes) are compensated for by modifying the boron concentration or moving the rod cluster control assemblies.

In addition to boration and dilution needs, the primary coolant is chemically treated in order to meet chemical and primary activity criteria. The corresponding fluid volumes may be recycled.

## **2.10. STRETCH-OUT OPERATION**

In power operation, during the operating cycle, available reactivity is compensated for by primary boration. As burnup increases, the boron content is reduced. The end of the cycle is reached when the boron content reaches a value close to zero.

In order to extend power operation beyond the natural end of the cycle, the fall in reactivity due to fuel depletion may be compensated for by reducing the primary temperature.

When the control rods are almost entirely extracted and the turbine inlet valves are fully open, the power level is determined by the core reactivity balance and turbine characteristics.

As there is no available built-in reactivity to ensure a constant average primary coolant temperature, the average primary coolant temperature, reactor power and steam pressure decrease steadily.

The extended cycle operation, based on repeated setpoint adjustments, consumes the remaining built-in reactivity.

Demonstration studies of a cycle extended by a maximum of 70 equivalent full power days (EFPD) and an early shutdown of a typical value of 30 EFPD will be provided in the pre-operational safety report.

## **2.11. SPECIFIC OPERATIONS**

In the circumstance of an event not caused by emergency conditions and when normal operating procedures are not suited to the management of the event (such as house load operations or loss of power sources), specific operating procedures will be applied by the operators to replace or back up normal operating procedures so as to manage the event.

## **2.12. OPERATING PROCEDURES, STANDARD REACTOR STATES AND OPERATING LIMITS [REF-1]**

Before dealing with each of these operating documents, this section provides some key definitions regarding plant standard reactor states, operating procedures and operating limits, as these constitute key input data to the safety case. Then, each of the operating documents listed in section 1 of this sub-chapter is dealt with in turn, in order to provide an understanding of the inputs to each area, the process associated with each, the outputs and where these outputs are then used or specified.



### 2.12.1. Operating Procedures – Identification of Perimeters

To allow good management of a complex, high-hazard installation such as a nuclear power plant unit, it is necessary to produce operating procedures, aligned closely with hiring, training and organisation procedures, that are appropriate to all the conditions encountered by the installation, whether in "normal" conditions (installation undamaged with respect to the Safety criteria), and emergency conditions, whether plausible (degraded safety), or implausible (core damage). These requirements were reinforced in particular by the TMI accident, followed seven years later by the Chernobyl accident.

For an EPR power plant unit, there is a need to propose a clear basis for drafting a precise policy for operational coverage, regardless of the condition encountered by the installation. It should take account of the results of the work done by the EPR Emergency Operating Procedures (EOP) working group and that done under the EOP Heritage Project (created at the end of 2004). In particular, this work was based on extensive operating feedback regarding the efficiency of State Oriented Approach procedures. A discussion of the UK EPR operating procedure concept is provided in Sub-chapter 18.1, including State Oriented procedures. Sub-chapter 18.3 provides a supplementary description of the State Oriented Approach.

Operation of a nuclear power plant unit such as the EPR can be split into several categories:

- Normal operation,
- Incident and accident operation "EOP" (emergency operation),
- Post-EOP operation after successful EOP (in other words transition between the EOP and the normal operating procedures),
- Post-EOP operation after failure of EOP, in other words severe accident management.

Such plant operations should take account of the constraints and specificities such as:

- Conformity with the OTS concerning the normal operating phase
- Conformity with the safety case concerning the emergency operating phase
- Team organisation, which can be specific:
  - in emergency operation (number of operators including field operators and specific roles of the control room team),
  - in normal operation in the case of internal and external hazards (off-site emergency, fire service, etc.)

Consistently with this section, the goal of this sub-chapter is to present:

- How the operational documentation ensures the plant safety under normal operations,
- How emergency operation, post-EOP operation and Severe Accident Management claims are met thanks to operational documentation (for example; periodic tests made under normal conditions of plant operations demonstrate the performance of a safety feature and underpin fault studies requirements that claim the safety feature)

### 2.12.2. Definitions of Standard Reactor States

Several definitions are used to characterise normal operating conditions. These are safety analysis envelope states, operating range and standard reactor states.

#### Safety Analysis Envelope States (often referred to as Reactor States)

For the EPR design, a safety analysis envelope has been defined. This envelope, used for the safety analyses, is the result of dividing the normal reactor operating range into different states (A to F) within this range. These represent the initial conditions upon which the initiating events for incidental and accidental analyses contained in the safety case are postulated. The boundaries between the states of the safety analysis envelope are essentially defined by the protection signals and the safety and safeguard systems likely to be called upon in incidental and accidental conditions.

Events postulated in the safety analysis are assumed to occur during normal operation. The initial conditions assumed in the safety analysis cover all possible standard reactor states from full power operation to cold shutdown. The following six standard reactor states are defined in Sub-chapter 14.0 and are provided below (**Note:** these values are preliminary values for monophasic start-up and may be subject to modification).

#### State A:

Power states and hot and intermediate shutdown ( $P \geq 130$  bar). In these shutdown states, all the necessary automatic reactor protection functions are available as in the power state. In fact, some protection functions may be deactivated at low power, but there are always sufficient automatic protection functions to meet the acceptance criteria if a transient occurs.

#### State B:

Intermediate shutdown,  $T_{\text{prim}} \geq 120^{\circ}\text{C}$  ( $P < 130$  bar). State B covers all shutdown states during normal plant operation, where primary heat is removed by the SG. It extends from 130 bar to 25 bar/ $120^{\circ}\text{C}$  (connection of RIS/RRA [SIS/RHRS]) RCP [RCS] conditions. Above  $120^{\circ}\text{C}$ , the LHSI in RHR-mode (LHSI/RHR) is not connected to the RCP [RCS] in normal operation. Note that the LHSI/RHR can be connected to the RCP [RCS] at  $180^{\circ}\text{C}$ , if necessary, but this is not an initial state corresponding to normal operation and therefore it does not need to be considered as an initial state in the deterministic safety analysis. In state B, some automatic reactor protection functions available in state A may be deactivated.

#### State C:

Intermediate and cold shutdown with LHSI/RHR. The RCP [RCS] is closed or can be rapidly re-closed, e.g. when a vent line is open, so that the SGs can be used for decay heat removal, if necessary. The RCP [RCS] is full of water or at partial loop level e.g. for SG tubes draining and for RCP [RCS] purging. Reactor state C covers the RCP [RCS] temperature range between  $120^{\circ}\text{C}$  and  $55^{\circ}\text{C}$ . Three different sub-states C1, C2 and C3 are defined depending on the different levels of RCP [RCS] water inventory, operating status of reactor coolant pumps and LHSI/RHR pumps and SG availability for heat removal:

#### State C1

- RCP [RCS] pressure around 25 bar abs (range : 25 – 32 bar abs).
- RCP [RCS] temperature between  $120^{\circ}\text{C}$  and  $100^{\circ}\text{C}$ .

- RCP [RCS] water inventory corresponding to the pressuriser level at hot shutdown conditions.
- Two SG participating in heat removal.
- A minimum of two reactor coolant pumps in operation.
- RIS/RRA [SIS/RHRS] operating via two LHSI/RHR trains, the other two trains are on stand-by.

State C2

- RCP [RCS] pressure around 25 bar abs (range: 25– 32 bar abs).
- RCP [RCS] temperature between 100°C and 55°C.
- RCP [RCS] water inventory between pressuriser level at hot shutdown conditions and PZR level > 90% (solid phase)
- Two SG available for heat removal.
- One or two reactor coolant pumps in operation, at least.
- RIS/RRA [SIS/RHRS] operating via three or four LHSI/RHR trains.

State C3

- RCP [RCS] pressure between 32 and 0.2 bar abs.
- RCP [RCS] temperature between 15 °C and 55°C.
- RCP [RCS] water inventory between solid phase and low level operation (3/4 loop).
- Two SG available for heat removal.
- No reactor coolant pumps in operation.
- RIS/RRA [SIS/RHRS] operating via two or three LHSI/RHR trains, the other train is on standby.

State D:

Cold shutdown with RCP [RCS] open so that the SGs cannot be used for decay heat removal. The RCP [RCS] level can be at partial loop level. In state D with lowered RCP [RCS] level (operation at  $\frac{3}{4}$  loop level), three out of four LHSI/RHR trains are required to be in operation to maintain a RCP [RCS] temperature below 55°C. The fourth LHSI/RHR train is on stand-by.

State E:

Cold shutdown with the reactor cavity flooded for refuelling.

### State F:

Cold shutdown with the core fully unloaded. During this state work is performed on RCP [RCS] components. This state does not need to be analysed for core protection.

### **Operating Range**

In parallel to States A to F, the normal reactor operating range is grouped into distinct operating ranges:

- Reactor at power (RP)
- Normal shutdown with SG (NSD/SG)
- Normal shutdown with RIS/RRA [SIS/RHRS] (NSD/RIS-RRA)
- Cold shutdown for maintenance (SDM)
- Shutdown for refuelling (SDR)
- Core fully unloaded (CFU)

These operating ranges group several standard states (see below) with similar thermal hydraulic and neutronic characteristics or similar operating purposes. The standard states are defined as the stable states of the reactor. The boundaries of these standard states are easily identifiable by the operator.

### **Standard States**

The standard states of the reactor are defined as the stable states of the primary circuit based on their thermo-hydraulic and neutronic characteristics. They are defined by the combination of parameters relative to the water inventory of the primary circuit, the pressure of the primary circuit, the temperature of the primary circuit, the boron concentration, the nuclear power as well as the functional configuration of the different systems and/or components. The boundaries between standard states are also defined by the most significant essential operations.

The operating ranges and standard states must be defined so that they are consistent with the *safety analysis envelope* i.e. each standard state must be within the safety analysis envelope. However, the boundaries between the operating range and standard states are not necessarily identical to the boundaries of the safety analysis envelope.

The safety analysis envelope, operating range and standard states are central to the definition of the key limits and conditions.

The operating range parameters associated with these states are summarised in Section 18.2.2 - Table 1. Section 18.2.2 - Figure 1 shows the operating envelope that defines the reactor operating ranges.

**Note:** These values are preliminary. The limits provided above may be slightly modified, especially at low pressure/temperature.

### 2.12.3. Reactor Operating Limits

Operation of the Reactor Coolant System (RCP [RCS]) within the temperature and pressure ranges defined in Section 18.2.2 - Figure 1 ensures compliance with the safety limits associated with the second barrier.

The pressure and temperature (P, T) limits for plant operations are imposed for either safety or mechanical reasons. These limits are presented and explained below.

Section 18.2.2 - Figure 1 shows the allowed pressure and temperature ranges of the Reactor Coolant System during start-up, shutdown and normal operation so obtained.

The main limits<sup>1</sup> (boundary limits) are:

- The pressure limit **P=155 bar** and the temperature limit **T=303.3°C** which correspond to normal operating conditions at Full Load and Hot Shutdown,
- The limit (**Psat, Tsat -30°C**) ensures sufficient margins to avoid saturation conditions other than in the pressuriser,
- The limit (**Psat, Tsat -180°C**) ensures that the temperature difference between the surge line and the hot leg N°3 is lower than the maximum authorised by the surge line stress analyses. This limit is not applicable when the pressuriser is in a solid state,
- The limit (**Psat +110 bar, Tsat**) prevents the reactor coolant primary-secondary differential pressure from exceeding 110 bar, this being the maximum value used for the design of the tube sheet of steam generators,
- The conditions around the intersection of the limits P=24.5bar abs. and (Psat, Tsat-30°C) imply very low values of Net Positive Suction Head (NPSH)<sub>a</sub>. Between 150°C and 250°C, the NPSH criterion is limiting and a supplementary limit curve is defined called "**RCP NPSH limit**",
- **P=32 bar** is the RHRS maximum connection pressure,
- **P=24.5bar** is the minimum pressure for Reactor Coolant Pump (RCP) operation to cope with the NPSH requirement,
- In normal conditions, during the cool-down phase, RHR trains 1 and 4 are connected at **T=120°C**,
- The RCP [RCS] temperature limit for the start-up of the first reactor coolant pump after an outage with respect to RCP [RCS] homogenisation is T=65°C, however this is bounded by T≤ 55°C, an initial condition used in the accident analyses,
- The minimum temperature in the RCP [RCS] **T=15°C** is induced by the minimum temperature of the IRWST.

<sup>1</sup> When reading the pressure and temperature of the Reactor Operating Domain the references are:

- For pressure above 110 bar abs., the pressure is measured at the pressuriser (abs. Pressure),
- For pressure below 110 bar abs., the pressure is measured in the hot leg (abs. Pressure),
- The temperature corresponds to the RCP [RCS] average temperature for the Full Load and Hot Shutdown reactor states and to the maximum loop temperature for the other limits.

**Note:** It should be noted that these values are preliminary. The limits provided above may be slightly modified, especially at low pressures/temperatures.

#### **2.12.4. Optimised Pressure – Temperature Curve**

A Pressure - Temperature (P, T) limit curve is determined using existing procedures. The consequences of moving the (P, T) limit curve, determined through fast fracture analysis, as a 'rigid body' to higher temperatures on a (P, T) diagram, for example a shift to the right of the (P, T) limit curve of 10°C, 20°C or 30°C, are analysed [Ref-1]. The (P, T) limit curve is used to provide limits for the operating pressure and temperature of the primary circuit during start-up and shutdown. The (P, T) limit curve is applicable to pressure boundary components made from ferritic steel. Such steels have a characteristic change from low to high fracture toughness through a temperature range.

**SECTION 18.2.2 – TABLE 1**

**Standard Reactor States: Preliminary Values**

Reactor State	Operating range	Standard states	Primary coolant inventory	Primary coolant pressure (bar abs.)	Average primary coolant temperature (°C)	Boron Concentration in primary circuit	Nuclear power (% P <sub>n</sub> )	Containment Status (Equipment Hatch)
A	Reactor at power (RP)	Reactor at power	Two-phase Pressuriser	155 ± 2.5 bar	307.7 ≤ T ≤ 311.5 ± 2.5° C	Critical BC	25 ≤ P ≤ 100	
		Reactor at reduced power		155 ± 2.5 bar	303.3 ≤ T ≤ 307.7 ± 4° C		4 ≤ P ≤ 25	
		Hot standby		155 ± 2.5 bar	303.3 ± 4° C		P ≤ 4	
		Criticality search		155 ± 2.5 bar	303.3 ± 4° C	Criticality search BC		
B	Normal shutdown with SG (NSD/SG)	Hot shutdown	Level at setpoint ± 8.5% GM	155 ± 2.5 bar	303.3 ± 4° C	BC ≥ BC <sub>AAC</sub>	0	
		Intermediate shutdown with SG P <sub>primary</sub> ≥ 130bar		130 ≤ P ≤ 155	213 ≤ T ≤ 303.3			
		Intermediate shutdown with SG P <sub>primary</sub> < 130bar		25 ≤ P < 130	120 ≤ T ≤ 303.3			
C	Normal shutdown with RIS/RRA [SIS/RHRS] (NSD/RIS-RRA)	Intermediate shutdown	Two-phase Pressuriser. Level close to setpoint ± 8.5% (The level is higher during transition to monophasic operation or during bubble formation)	25 ≤ P ≤ 32	50 ≤ T ≤ 120	BC ≥ BC <sub>AAF</sub>	0	
		Two-phase with RIS/RRA [RHRS]						
		Single-phase Intermediate shutdown	Full (PZR level > 90%)	1 < P ≤ 32	15 ≤ T ≤ 120	BC ≥ BC <sub>AAF</sub> or BC <sub>RE</sub>		
		Normal cold shutdown (pressurised primary circuit)	¾ Loop ≤ Level ≤ Full	0.2 ≤ P ≤ P1 P1 ~ atmospheric	15 ≤ T ≤ 55	BC ≥ BC <sub>RE</sub>		
D	Cold shutdown for maintenance (SDM)	Normal cold shutdown for maintenance (primary circuit not pressurised)	≥ ¾ Loop	Atmospheric	15 ≤ T ≤ 55	BC ≥ BC <sub>RE</sub>		
E	Shutdown for refuelling (SDR)	Normal cold shutdown for refuelling	Fuel pool level ≥ 18.9 m		15 ≤ T ≤ 55			
F	Core fully unloaded (CFU)	Core fully unloaded	N/A	N/A	N/A	N/A		Opening permitted below 90°C

**SECTION 18.2.2 – TABLE 1 (CONT'D)****Standard Reactor States Glossary**

BC Boron Concentration

BC<sub>AAC</sub> Boron Concentration Arret a Chaud  
Concentration

Hot Shutdown Boron

BC<sub>AAF</sub> Boron Concentration Arret a Froid  
Concentration

Cold Shutdown Boron

BC<sub>RE</sub> Boron Concentration Recharge

Reload Boron Concentration

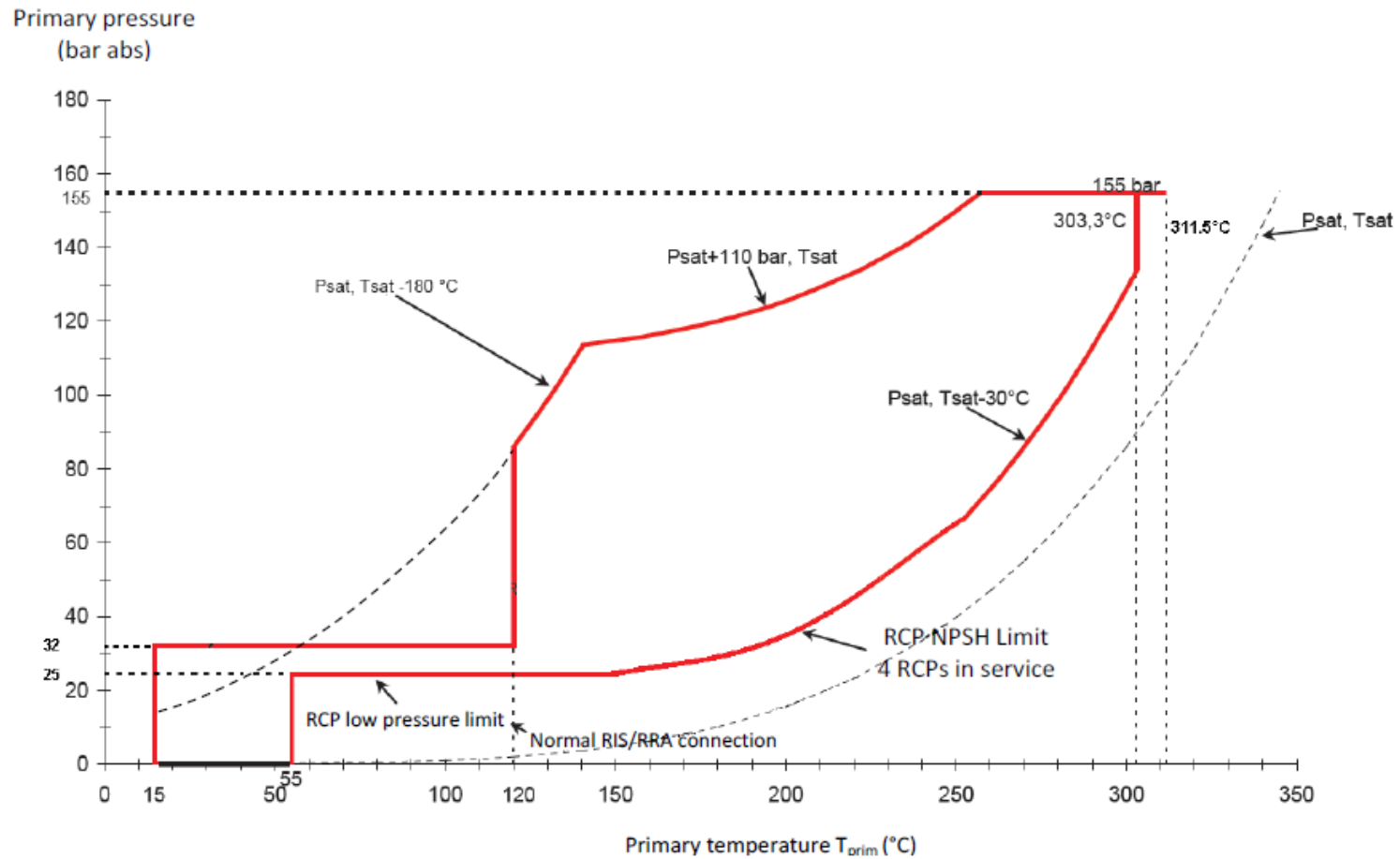
GM Gamme de Mesure

Measurement Range



**SECTION 18.2.2 - FIGURE 1**

**Typical (P, T) Reactor Operating Range**



### **3. NORMAL OPERATING PROCEDURES**

#### **3.1. NORMAL OPERATING DOCUMENTS**

Normal operating procedures consist of two types of documents: “normal operating rules” and “normal operating instructions”.

The “normal operating rules” define and justify the operating strategies. The “normal operating instructions”, which are used by the Operators in the Main Control Room, are written using the “normal operating rules” and describe the precise actions to be performed by the Operators.

Different normal operating procedures are necessary to manage normal scheduled operating transients and certain specific operations in response to unplanned events. For example:

“DEM1”: from shutdown state to RHRS (Residual Heat Removal System) connected,

“DEM2”: from RHR connected to hot shutdown state,

“DEM3”: from hot shutdown state to full power.

Further discussion of normal operating documents is provided in Sub-chapter 18.1.

#### **3.2. INTERFACES WITH OPERATING PROCEDURES FOR ABNORMAL OPERATION**

Procedures for abnormal operation are described in Sub-chapter 18.3. A discussion of Human Factors aspects is provided in Sub-chapter 18.1.

For each situation for which operating procedures are necessary, criteria have been established to define boundaries between the domains of normal and abnormal operation. For example, criteria requiring a switchover to abnormal operating procedures are: second barrier damaged, protection/safety systems actuated etc.

### **4. DESIGN AND OPERATING LIMITS AND CONDITIONS [REF-1]**

This section presents the generic principles and requirements that will be used as a basis for defining the OTS, chemical and radio-chemical specifications, allowance for occurrences of loading conditions, safety analysis bounding limits and fuel design limits of the EPR.

This section describes the process that will be used to produce these different specifications and demonstrates how the results obtained through the safety analyses are used to establish the parameters that must be observed during normal operation of the plant. In this way, when the requirements contained within operational documentation are met, the safe operation of the plant, within the assumptions contained in the safety case, is guaranteed.

This section also justifies why the operating documents will be adequate to ensure that the safety limits and conditions will be complied with throughout the plant lifetime.

Finally, this section presents the boundary between GDA documentation and Licensee operational documentation.

## **4.1. OPERATING TECHNICAL SPECIFICATIONS**

### **4.1.1. Generic Principles for OTS production**

The OTS form part of the operating documentation that must be developed for the UK EPR. The general objective of the OTS is to set out the rules that must be followed to ensure that during normal operation the reactor remains within the limits justified by the safety case.

For this purpose, the OTS must:

- Specify the normal operating limits on the parameters which will ensure compliance with the parameter values assumed in the safety analyses contained in the safety case,
- Determine the operability requirements for the safety systems, structures and components (SSCs) necessary to mitigate transients, incidental scenarios and accidental scenarios considered in the safety case,
- Define in the event of inoperability of the required safety SSCs or any abnormal change in an operating limit, the recovery actions that are required so that the main safety functions are achieved. Regarding each inoperability condition or event and its associated recovery action, the OTS specify a completion time, during which the plant can be maintained in the degraded condition without compromising plant safety.

This sub-chapter will address inoperability conditions (or events); however, the associated corrective measure and completion time will not be presented as they must be defined in agreement with the Licensee.

#### **4.1.1.1. OTS Scope and Criteria**

The content to be covered by the OTS is defined through specific criteria. Regardless of the approach adopted in developing the OTS, criteria will allow definition of parameters, systems, structures or components required to ensure the safe operation of the plant i.e. to ensure that the assumptions contained in the safety cases are complied with and that the OTS content is consistent with these assumptions.

The criteria must be applied to each plant state.

In addition to the requirements on safety systems, the OTS also deal with parameters, systems, structures or components related to radiation release and monitoring, and chemistry limits applicable to the plant, and also requirements for monitoring the integrity of barriers.

The OTS criteria also apply to the safety of the storage and containment of spent fuel in the fuel building.

Depending upon the approach for the production of the OTS, some specific operating requirements (fire, over-pressurisation protection, RCS brittle fracture or prevention of non-ductile failure), considered in the safety analyses, may either feature in the OTS or in separate operating documents.

The OTS requirements, established on the basis of these criteria, are complied with if the parameters remain within the limits, and SSCs covered by the criteria are operable. The concept of operability reflects the ability of a system, or component of a system, including its necessary auxiliaries, supports and electrical power supplies, to perform its functions and meet the safety objectives.

#### **4.1.1.2. Definition of the Requirements Relative to the Different Operating Ranges and Standard States**

The OTS criteria must be applied to each state of the plant.

The operating ranges and standard states are consistent with the safety analysis envelope defined in section 2.12.2 of this sub-chapter. The consistency between them guarantees completeness of the OTS requirements, which ensures compliance with the assumptions made in the accident studies.

The list of the SSCs for which operability is required in the OTS can vary from one operating state to another. For some SSCs the possible variations follow the standard states (for example on the number of trains). These variations also apply to parameter limits.

Accident analyses are not carried out for all the states in the safety analysis envelope when it is considered that one of the states may be bounding for other states (see Chapter 14). However the requirements cover all of the states where the initiating event remains credible. As a consequence, SSCs whose operability is required to meet the assumptions in the fault studies presented in the safety case, are subject to the requirements in all the states covered by the studies.

#### **4.1.1.3. Non-Compliance with OTS Requirements**

The OTS principles make it necessary to address inoperability in case of non-compliance with a requirement. Any inoperability is addressed by an OTS condition (or event). For each condition (or event), associated recovery action(s) and associated recovery time(s) must be defined. These recovery action(s) and recovery time(s) are both defined in agreement with the Licensee. Recovery action(s) and recovery time(s) will be defined so that the main safety functions are achieved and so that the plant could be maintained in the degraded condition without compromising plant safety. Recovery times can be derived using either deterministic or probabilistic considerations where appropriate, and depending on the OTS production approach adopted.

Note: the OTS do not have to specify the means of executing a recovery action that the operator might adopt.

#### **4.1.1.4. Preventive Maintenance Consideration through the OTS**

To take into account the principle of preventive maintenance during normal operation of the plant that the EPR design permits, the OTS requirements are defined considering the inoperability of a train due to preventive maintenance. As such the number of trains required in the OTS reflects the number of trains required to ensure the safety function when preventive maintenance and the single failure criterion are both applied, where appropriate.

#### **4.1.2. OTS link with other safety operational documentation**

##### **4.1.2.1. Interface between the OTS and Normal Operating Procedures**

The OTS are used during the production of the normal operating procedures so that they comply with the requirements prescribed for each operating state.

##### **4.1.2.2. Interface between the OTS and Emergency Procedures**

Written procedures exist for all operating states of the plant, covering normal and emergency operations. During normal operation, OTS compliance ensures that the plant is consistent with the assumptions of the accident studies described in the PCSR. During emergency operation, certain plant limits and conditions relating to safeguard systems may not conform to OTS requirements and therefore emergency procedures ensure that the plant is managed within the safety envelope. The emergency procedures will, where practicable, aim to limit the time when they are applicable.

The end conditions of an emergency procedure, after an event requiring its application, are defined in the scope of the emergency procedures. When these conditions are satisfied, either the OTS are applied or the plant will enter a long term event recovery phase.

##### **4.1.2.3. Link between the OTS and Periodic Testing**

Limits defined for any parameter or operability of any SSCs required in the OTS are confirmed by performing periodic testing. Failure to achieve a success criterion in a periodic test is a possible means of entry into an OTS condition. Then, the periodic tests related to these parameters or SSCs are defined in a manner consistent with the OTS. The OTS requirement and the equipment functional requirement checked in the periodic test are complementary.

Any deviation from a general OTS prescription necessary to carry out a periodic test is identified at each stage of a test programme.

Periodic testing is discussed in section 5 of this sub-chapter.

##### **4.1.2.4. Link between the OTS and Physics Test**

The execution of the physics tests during start up or during a cycle must not lead to a reduction in safety. The conduct of physics tests does not lead to non-compliance with technical specification requirements. However, some tests can only be run in specific plant conditions. Consideration must be given to these types of tests in developing OTS requirements.

Such test conditions are well identified and are analysed to allow development of specific written procedures that cover the required plant configuration for managing physics tests.

Measures that permit the execution of the physics tests will be written into operational documentation. These measures will ensure that the plant conditions required for testing are analysed in order to ensure that correct constraints are established for the physics tests and to ensure that the safety of the plant is maintained during the performance of these physics tests.

Depending on the approach adopted in developing the OTS, the physics tests and the means to manage the OTS during physics tests may either feature in the OTS or in a separate operating document.

The physics tests are discussed in section 5.4 of this sub-chapter.

#### 4.1.2.5. Link between the OTS and In-Service Inspection Procedures

The OTS are used in the production of the In-Service Inspections (ISI) procedures so that the requirements prescribed are met and remain effective for each operating state.

In-Service Inspection is discussed in section 6.1 of this sub-chapter.

#### 4.1.2.6. Link between the OTS and Radiation and Chemistry Limits

Chemical and radiochemical parameters are control parameters, which are defined as parameters having a direct link with the mitigation and control of safety consequences, including radiation fields, environment, hazards, maintenance and operational issues.

For example, release limits for radioactive materials are parameters that are checked by radiochemical monitoring during normal operation to ensure that defined legal limits are complied with.

Release limits for radioactive materials are based on applicable limits for the permitted dose that may be received by an individual, due to normal operation of a nuclear power plant in any one-year period.

Chemical and radiochemical limits are limits that must be respected, depending on the reactor state, to achieve the following objectives:

- Fuel-cladding and primary pressure boundary integrity in the medium and long term,
- Minimisation of the consequence of the process for radiation fields,
- Minimisation of the consequences of the process for environment.

The chemical and radioactive parameters that need to be monitored during normal operation to comply with the safety objectives of the plant have been identified as being:

- equivalent I-131 activity levels (at power and during a unit outage),
- noble gas activity threshold,
- reactor coolant/secondary cooling system leak flow rate deduced from activity measurements in the secondary coolant system,
- Xe-133 and Xe-133 to Xe-135 activity ratio for radiochemistry parameters,

The boron concentration requirements in the RCP [RCS] and the primary auxiliary system (Fuel Pool, IRWST, RBS [EBS], RIS [SIS] Accumulators) are for reactivity control and are linked to, but do not arise from, a chemistry requirement.

Depending on the approach adopted in developing the OTS, the radiation and chemistry limits may either appear in the OTS and chemical and radiation specification documents or only in chemical specification and radiation specification documents.

Radiation and chemistry are discussed in section 4.2 of this sub-chapter.

#### 4.1.2.7. Link between the OTS and Fuel Reload

Some parameters considered as assumptions in the safety case are cycle and/or burn-up dependent parameters which depend on the fuel reload (for example: boron concentration, thermal hydraulic conditions (pressure/temperature) during stretch-out operation, RCCA insertion limits...). Fuel reload is discussed in section 4.4 of this sub-chapter.

In the OTS, the core operating limits will be specified such that all assumptions of the safety case are met. The OTS will be updated at each fuel reload to take account of the evolution of core operating limits related to the cycle specific parameters.

#### 4.1.2.8. Link between the OTS and Design Features

Design features form an inherent part of the safety case. Any modification of the design features that may affect safety must be controlled. Depending on the approach adopted in developing the OTS, the design features may either be specified in the OTS or in a separate operating document.

Within the UK context, such documents will comply with Licence Condition 22, which requires the Licensee to manage all modifications to the plant that have an impact on the safety case. Any modifications to design features and related OTS requirements will be dealt with through the compliance with this Licence Condition.

#### 4.1.2.9. Link between the OTS and Administrative Controls

A requirement for the operation of nuclear sites is to have in place administrative controls. These controls are encompassed by the requirements of the Nuclear Site Licence Conditions. Depending on the approach adopted in developing the OTS, the administrative controls may either be specified in the OTS or in a separate operating document.

Development of administrative controls are the responsibility of the Licensee, who will decide an appropriate means of complying with their requirements by means of operating documentation regardless of the approach used to develop that documentation.

The OTS and associated administrative control procedures adopted by a UK EPR Licensee will comply with the relevant Licence Conditions.

#### 4.1.3. Generic Process of OTS Production

The methodology described to produce the preliminary OTS will follow a generic framework, reflecting the main input data to be used to produce the OTS. The list of the main input data to be used, and their role in the OTS documents are illustrated in Section 18.2.4 - Figure 1.

**Step 1:** Determine the parameters, systems, structures and components to be presented in the OTS to ensure the safe operation of the plant within each state of the safety analysis envelope in accordance with the OTS criteria.

The parameters, systems, structures and components introduced in the OTS, according to the relevant criteria, are defined using the following input data:

- The initial conditions, which are assumed in the safety case, and the safety limits which ensure that the main safety functions are fulfilled. These inputs are defined in the PCSR.

- Descriptions of the system safety features and the associated components necessary to mitigate the transients according to the safety analyses and as required in the OTS, in accordance with the criteria. A system safety feature is defined by a list of components involved in meeting a given objective. This definition is completed by support system function requirements associated with the system safety feature.
- Section 18.2.2 – Table 1 determines the link between the safety analysis envelope, the operating ranges and standard states. Each system safety feature is analysed to determine the applicability of the feature in each of the safety analysis envelopes.

**Step 2:** Characterisation of the requirements for each state.

Once the parameters and systems, structures and components (SSC) for each safety feature have been defined, the next step concerns the characterisation of the requirement of the SSC, i.e. for each SSC, give the definition of the operability.

The number of trains required in the OTS is also determined; taking into consideration that preventive maintenance is permitted on some systems during normal operation of the plant. The number of trains required to ensure the safety function after consideration of preventive maintenance considers:

- Number of trains used in the safety analysis transients: For example, for a LOCA the PCSR assumes 2 LHSI trains (of which one is considered lost to the break);
- One train added to the previous number of trains, if the Single Failure Criterion is applicable.

For example, for a LOCA transient, 3 LHSI trains will be required with the fourth train considered to be in preventive maintenance when the accident occurs.

To ensure that all the parameters and SSCs of the safety feature would be operable as required, each parameter or SSC must be associated with a surveillance requirement. So, at this stage of the process, a check is performed to ensure that each required parameter or SSC has an associated surveillance requirement and that they are applied in a manner consistent with the OTS requirements.

**Step 3:** Definition of the OTS conditions.

For each system, once the requirements for a given reactor state have been characterised, it is possible to define all the conditions (or events) of inoperability i.e. each possible way of not meeting that requirement.

The condition (or event) is either a parameter which is not within the required limit, or a single system, structure or component which is inoperable.

**Step 4:** Determine the recovery actions, fallback state and associated recovery time.

Recovery actions will be defined for each condition (or event). Recovery actions will specify the required action which the operator must perform to restore the parameters or SSCs to within OTS limits or to an operable status.

The required recovery actions can be:

- A remedial action;



- An alternative action;
- A fallback action i.e. a fallback state to which the plant should be shutdown and maintained in order to ensure the safe operation of the plant.

A recovery time is associated with a recovery action. It corresponds to the time during which it is considered acceptable for the plant to remain in the considered state. The recovery time will be assessed based on pre-determined criteria and could be based on a deterministic or probabilistic approach.

Depending on the approach adopted in developing the OTS, the Probabilistic Safety Assessment (PSA) (risk informed insight) may be applied to underpin the deterministic approach.

It should be noted that the definition of the required recovery actions and the associated recovery times will be defined in agreement with the future Licensee.

#### **4.1.4. Justification that OTS documents will ensure compliance with safety limits and conditions**

Assurance that the OTS documents are adequate to ensure the safety limits and conditions is demonstrated through the entire process of production of the OTS i.e. in the definition of the OTS principles, the process for producing them and the link between the OTS and other operating documents.

Indeed, the scope and criteria that define the perimeter of the OTS documents ensure that, for any plant state, all the parameters and SSCs necessary to mitigate transients are operable when they are required.

The OTS process ensures that all the assumptions concerned by the OTS scope and criteria for any plant state are captured and presented in the OTS.

Finally, each operating document has a defined purpose and its link with OTS ensures that all the limits and conditions are well captured and that the safety limits and conditions are fully underpinned.

#### **4.1.5. GDA/Licensee documentation boundary**

The GDA and Licensee OTS document boundaries are the following:

- Generic parameters and SSCs to be introduced in the OTS will be presented to the Licensee as part of the transformation of the generic design information into Licensee specific documentation. Some specific requirements will be related to Licensee decisions and can therefore only be introduced in Licensee OTS documents
- The OTS documents provided to the Licensee will address inoperability conditions (or events); however, the associated corrective measures and completion time(s) will be presented only in the Licensee OTS documents as they must be defined in agreement with the Licensee.

- The approach adopted in developing OTS operational documentation will depend on the Licensee. Thus, introduction of some specificities, related to the approach adopted, in the OTS or in other operating documents will be defined by the Licensee. The Licensee will also define and validate the definitive scope of the OTS. The Licensee will put in place adequate arrangements for development, implementation, monitoring, maintaining, updating and modifying OTS documentation that meets Licence Conditions and regulatory requirements.

## 4.2. CHEMICAL AND RADIOCHEMICAL SPECIFICATIONS [REF-1] TO [REF-3]

Further information on chemical and radiochemical specifications is provided in Sub-chapter 5.5.

### 4.2.1. Introduction

For the purpose of the EPR GDA process, reactor chemistry has been defined by the UK Health and Safety Executive (HSE) as:

*The chemistry of the design including the effects of coolant chemistry on reactivity, pressure boundary integrity, fuel and core component integrity, fuel storage in cooling pools, radioactive waste (accumulation, treatment and storage) and radiological doses to workers.*

The UK HSE considers that, for the GDA process, reactor chemistry is principally concerned with five main areas:

- Coolant reactivity control,
- Protection of the structural materials (specifically related to integrity of the pressure boundaries),
- Maintenance of fuel integrity and performance,
- Minimisation of out of core radiation fields,
- Releases during accident conditions.

For the EPR design, boron, required for neutronic reactivity control, dominates primary circuit chemistry considerations (for example it dictates why a number of other chemical additives are included). Boron is also relevant in a number of key faults which have been considered and mitigated in the design [Ref-1] [Ref-2].

Chemistry considerations also apply to the secondary circuit [Ref-3]. The functions and design of all secondary circuit systems must account for the operations required during start up, normal operation (including power changes), hot standby, cold shutdown and during postulated accidents and faults.

Chemical control of the secondary circuit is dominated by the need to preserve the integrity of the system, which may have implications for activity release to the environment or conventional safety hazards.

Chemical and radiochemical control is also performed in the auxiliary systems with the following objectives in order to ensure compliance with the safety functions (mainly reactivity control and iodine mitigation), to avoid internal hazards (hydrogen/oxygen risk) in each of the components (tanks, exchangers, piping etc) and to minimise the build-up of ex-core radiation fields, which could have an impact on the environment and worker dose rates.

The approach chosen to identify the chemical and radiochemical parameters is to focus on site chemical and radiochemical issues. Each issue is associated with an aim, namely:

- Underpinning the safety case
- Radiation fields: control of radioactive source terms (corrosion and fission products) and their consequences
- Environment: control of any discharge to the environment
- Materials and equipment integrity: integrity of barriers and materials over the medium and long term
- Hazards: control of flammability risk
- Operation: control of process performance, integration of Operational Experience

The identification of the chemical and radiochemical parameters lies in the notion of control parameters, which has the following definition: parameters having a direct link with the mitigation and control of the consequences on safety, radiation fields, environment, hazards, maintenance and operational issues. These parameters are associated with a limit value, breach of which is managed by the application of a procedure.

Other chemical or radiochemical parameters are measured: they are complementary to the control parameters and serve exclusively to enable rapid fault finding in case of deviation from the expected value. They do not impose any restriction on plant operation.

The specification methodology [Ref-4] [Ref-5] for identification of various Chemical and Radio-chemical parameters for plant operation is applicable to the methodology employed for Flamanville 3. The methodology presented is expected to be applicable for the UK EPR in order to provide the definitive identification of Chemical and Radio-chemical parameters and values.

#### 4.2.2. Radio-chemistry Specifications

During normal operating conditions the radiochemical control parameters are the following:

##### First barrier control

- The I-131 equivalent (Eq. I-131) is defined as the concentration of I-131, which by itself would produce the same dose as the "isotopic concentration" of the various iodine isotopes present. I-131 is one of the most radiologically toxic isotopes given the consequences of its preferential targeting of the thyroid gland. Therefore the Eq. I-131 activity is a determining parameter for the initial conditions of the PCC radiological consequences studies during steady state operation (state A), during a transient (state A) and during shutdown (applied to states C, D and E: analysis areas that deal with opening of the 3<sup>rd</sup> barrier).

- Total noble gas activity, corresponding to overall activity from the seven main gases in the reactor coolant, is an indicator associated with Eq. I-131 to the extent that, in operational terms, exceeding a total rare gas limit value requires a power transient to verify that peak iodine activity is less than the limiting Eq. I-131 activity value under transient operating conditions. It is applied in state A.
- The Xe-133 activity and the ratio Xe-133/Xe-135 are used as control parameters in order to monitor the first barrier integrity. It is applied to state A.
- I-134 activity in the primary coolant is used as a control parameter in order to determine the release of fissile material into the primary coolant. It is applied in State A (at full power operation).

#### Second barrier control

- The reactor coolant/secondary cooling system leak flow rate is used in the radiological consequences studies on the transients that cause steam generator releases to the environment. This occurs with SGTR transients but also PCC initiating events that are not SGTR.
- The activity of the secondary cooling system is mainly determined from the activity in the RCP [RCS] and the leak flow rate between the reactor coolant and secondary cooling systems. It is applied to states A and B.
- The leak flow rate between the RCP [RCS] and the secondary cooling system is used for monitoring the integrity of the SGs. Monitoring any change in this leak flow rate enables the early stages of an SGTR to be detected and is one of the lines of defence for preventing it developing into a PCC initiator. It applies to states A and B.

#### 4.2.3. Primary Circuit Chemistry Philosophy

Primary coolant chemistry is based on optimised conditioning and on limitation of impurities in order to:

- Minimise coolant Corrosion Product (CP) concentration.
- Optimise CP migration and re-deposition in order to:
  - minimise the deposits on fuel cladding leading to (Axial Offset Anomaly) AOA risk, and corrosion under deposits,
  - limit the accumulation of activated CP on the out-of-core surfaces of the RCS and thus minimise the radiation field build-up.
- Limit the corrosion rate of fuel cladding material (thermal effect and chemistry effects).
- Avoid oxygen and oxidising species formation by radiolysis, by hydrogen addition.
- Prevent localised corrosion (Stress Corrosion Cracking, SCC/pitting) by limiting impurities (chlorides, fluorides, sulphates).

The chemical conditioning of primary coolant consists of the management of lithium, boron, zinc and hydrogen concentrations [Ref-1] to [Ref-3].

The boron-lithium chemistry is defined to reach an optimum  $\text{pH}_{300^\circ\text{C}}$  which is chosen to:

- Minimise material corrosion and release,
- Minimise the transfer deposition of the corrosion products, based on magnetite and nickel ferrite compounds, on the material surfaces of primary system,
- Control the plant radiation field.

The choice of the  $\text{pH}_{300^\circ\text{C}}$  value leads to adoption of a maximum lithium concentration due to the boron concentration related to fuel management scheme. The use of Enriched Boric Acid (EBA) allows the lithium concentration to be limited.

Zinc injection is performed in order to minimise plant contamination, and minimise the dose-rate around the primary circuit. Based on theoretical data, zinc has been chosen to replace the activated cobalt in the oxide layers on the RCP [RCS] surfaces or inhibit the radio-cobalt deposits. This replacement allows limiting Co-58/Co-60 accumulation on the out of core parts compared to operating without zinc injection

The hydrogen concentration is defined in order to limit:

- Water radiolysis (control of the concentration of radiolytically formed oxidation products),
- Corrosion cracking risk of RCP [RCS] materials,
- Plant contamination (transfer and deposition of corrosion products).

Monitoring [Ref-4] and dosing systems are incorporated in the EPR design, which confirm the implementation of optimised conditioning (pH, zinc injection and hydrogen concentrations) thus ensuring that radioactivity levels are reduced as far as reasonably practicable.

It is clear that the design limits and conditions for the EPR include the correct specification of these and other chemical control parameters [Ref-5] to [Ref-9].

#### 4.2.3.1. Normal Operations Primary Circuit Chemistry Requirements [Ref-1] [Ref-2]

During normal operating conditions the primary control parameters are the following:

- Boron linked with neutronic requirements,
- Lithium concentration: the value of the upper limit is relevant to the integrity of the first barrier in the medium and long term, the lower limit is relevant to radiation fields,
- Hydrogen concentration: the limits are relevant to the integrity of the first and second barriers in the medium and long term,
- Impurities such as oxygen, chlorides, fluorides, sulphates, and sodium are relevant to the integrity of the second barrier in the medium and long term.

#### 4.2.3.2. Transient Regimes: Start-Up And Shutdown [Ref-1]

During transient conditions the primary control parameters are the following:

- Boron linked with the neutronic requirement
- Lithium concentration: the value of the upper limit is relevant to the integrity of the first barrier in the medium and long term
- Hydrogen concentration: avoidance of explosive ( $H_2/O_2$ ) mixtures, control of radiation fields and prevention of material degradation
- Impurities for temperatures higher than 120°C such as oxygen, chlorides, fluorides, sulphates are relevant to the integrity of the second barrier in the medium and long term

#### 4.2.4. Secondary Circuit Chemistry Philosophy [Ref-1]

The main objectives of the secondary side chemistry are to avoid:

- Steam generator (SG) tube corrosion
- Flow Accelerated Corrosion (FAC) in secondary side systems, especially in the steam systems with two-phase flow
- General corrosion in secondary side systems
- Heat transfer degradation problems

To select the water chemistry treatment most adapted for the secondary water for the UK EPR reactor, the parameters that have been taken into account are:

- The corrosion phenomena of the secondary system materials,
- The phenomena of SG fouling and Tube Support Plate (TSP) clogging,
- The reducing potential in the SG to avoid local corrosion problems
- The life time of cationic SG blow-down resins (in the case of SG blow-down treatment), and
- The impacts on environment due to the waste releases.

According to international operating experience feedback, the following choices were made for UK SG EPR reactor design:

- Alloy 690 TT for the SG tube materials to limit the secondary corrosion risks such as Inter-Granular Attack or Stress Corrosion Cracking (IGA/SCC).
- Stainless steel for the Tube Support Plate (TSP) materials to protect them from corrosion.

- Optimised tube to TSP design with “high permeability” trefoil-shaped broached TSP in order to:
  - limit the risks of TSP clogging in confined area,
  - limit hide-out phenomena.

pH plays an important role on generalised corrosion and Flow Accelerated Corrosion (FAC) of carbon steel and/or low alloy steel of the secondary circuit.

To mitigate, as much as possible, FAC in the secondary systems and so the transport of corrosion products in the SGs, an optimal pH value has to be defined. The secondary water chemistry also plays a role in the phenomena of fouling/clogging of the SGs. The optimal pH value must take into account these phenomena.

The risks of corrosion (type IGA/SCC) of SG tubes are especially linked to the impurity presence that can concentrate in SG confined areas. In the absence of pollution, the secondary water pH has a low impact on these types of corrosion.

Localised corrosion such as stress corrosion and/or pitting corrosion depend on impurity concentrations in the secondary water. These types of corrosion can be limited by controlling the concentration of impurities such as oxygen, alkaline products, sulphurs or oxidizing agents. Impurity control is a pre-requisite to ensuring an adequate optimised chemistry.

#### **4.2.4.1. Normal Operations Secondary Circuit Chemistry Requirements [Ref-1]**

During normal operating conditions the secondary control parameters are:

- The pH of the feed-water line relevant to the integrity of the second barrier in the medium term,
- The control of impurities: sodium and cation conductivity at the SG blow-down which are relevant to the integrity of the second barrier in the medium and long term

#### **4.2.4.2. Secondary Coolant Chemistry Specifications During Transient**

During transient conditions the secondary control parameters are:

- The pH of the feed-water line relevant to the integrity of the second barrier in the medium term,
- The control of impurities: sodium and cation conductivity at the SG blow-down which are relevant to the integrity of the second barrier in the medium and long term

#### **4.2.5. Auxiliary Systems Chemistry Philosophy**

The chemistry/radiochemistry strategy in the auxiliary systems is established via three main actions [Ref-1].

- 1) The implementation of optimal chemistry conditioning in each system in order to guarantee compliance with the safety functions attributed to specific safety systems and primary auxiliary systems:

- Reactivity control via the boron concentration control in the Spent Fuel Pool, IRWST, REA [RBWMS] boron, TEP [CSTS], RBS [EBS] and RIS [SIS] accumulators.
- Iodine mitigation by means of sodium hydroxide (NaOH) storage and injection by the EVU [CHRS] system and EDE [AVS] filtration.

Maintaining alkaline conditions in the IRWST limits the production of volatile iodine compounds in the containment during a LOCA or severe accident and is therefore a means of limiting radiological consequences during PCC-4 and RCC-B events.

Achieving the minimum pH level required in the IRWST is performed with a sodium hydroxide injection system using two EVU [CHRS] tanks.

- Fission product activity control during shutdown in order to prevent PCC radiological consequences during a unit outage.

2) The injection of additives to contribute to the conditioning of auxiliary systems or primary circuit:

- Hydrazine (N<sub>2</sub>H<sub>4</sub>) injection once the RIS-RRA [SIS-RHRS] trains are connected during shutdown or during start-up to eliminate the oxygen ingress,
- Trisodium phosphate injection to limit the carbon steel corrosion of the RRI [CCWS] (Cooling Water System for the nuclear auxiliary systems).

3) The application of relevant chemistry specifications that ensure suitable monitoring in order to:

- Detect the eventual ingress of impurities having an adverse effect on material corrosion, component degradation, source term build up, ex-core radiation, waste production and effluent discharges.
- Detect the potential activity releases from the auxiliary systems.
- Identify the degradation of components and/or their functional anomalies (e.g. the resin saturation, primary/secondary leaks, fission product degassing).

#### 4.2.5.1. Normal Operations Auxiliary Systems Chemistry Requirements

During normal operation the auxiliary systems contribute to the reactivity control via the periodic monitoring of several parameters associated with boron control. According to the EPR boron strategy and the technical devices available [Ref-1], the following controls are required:

- Boron-10 concentration (mg/kg) is directly provided by the REN [NSS] boron meters. The RCV [CVCS] boron meter also measures the B-10 content, which can be multiplied by the enrichment correction factor in order to provide the total boron concentration. The measurements are conducted in the RCP [RCS] (and directly connected systems RCV [CVCS], RIS-RRA [SIS-RHRS]).



- Enrichment (%at) is provided by ICP/MS laboratory measurements and it is monitored in the RCP [RCS] and connected systems (RCV [CVCS], RIS/RRA [SIS/RHRS]), REA [RBWMS] boron, IRWST, Spent Fuel Pool, TEP [CSTS], RBS [EBS] and RIS [SIS] accumulator. These measurements ensure the control of boron depletion and contribute to identification of possible events leading to isotopic dilution.
- Total Boron Concentration (mg/kg): this parameter is monitored by the RCV [CVCS] boron meter to allow protection against the risk of dilution in order to ensure the pH control of the primary circuit and control of the re-crystallisation risk in the auxiliary systems.

According to the design and operation of the auxiliary systems, the potential impurities transferred between the different systems can be limited by a suitable specific chemistry surveillance programme. Based on the evidence and the arguments described in Sub-chapter 5.5, the parameters concerned are:

- Oxygen limit is mainly established to avoid the risk of stress corrosion, pitting corrosion and general corrosion.
- Fluoride/chloride limit concentrations are established mainly to avoid the pitting corrosion risk.
- Sodium concentration is limited in order to reduce the caustic corrosion.
- Trisodium phosphate is injected into the systems in contact with air in order to reduce the risks of material corrosion. Its concentration is limited to reduce the environmental impacts.
- Silica, aluminium, magnesium and calcium concentrations are limited in order to anticipate the primary coolant specifications concerning the zeolite concentrations due to their impact on fuel cladding.
- Sulphate concentration is limited due to its potential affect on SCC corrosion.
- Total conductivity provides the total ion content in a solution. It is a good indicator of the purity of the demineralised water from the SED and the TEP [CSTS] distillate water.
- Solid particle concentrations are monitored in RRI [CCWS] and the SED system in order to avoid the ingress of particles to the cooling and make-up water.
- pH is controlled in the RRI [CCWS] system to ensure the integrity of third barrier .

The detection of potential activity releases from auxiliary systems is performed by means of:

- Gamma activity monitoring via the on-line KRT [PRMS] chain.
- Radiochemistry monitoring in RIS/RRA [SIS/RHRS] trains, RRI [CWCS], purification trains of PTR [FPPS], TEP[CPS], RCV [CVCS], REA [RBWMS], TEP [CSTS] concentrates.

#### 4.2.5.2. Chemistry/Radiochemistry Auxiliary Systems during Transient Periods

During shutdown and start-up, the main radiochemistry controls are performed in the RIS/RRA [SIS/SRHS] in order to limit the environmental impact and the radiological consequences following the opening of the primary circuit. The criteria are related to [Ref-1]:

- Xe-133 activity to determine the end of the TEG [GWPS] nitrogen sweeping,
- Xe-133 activity before RPV opening and the end of air sweeping,
- Xe-133 activity before stopping the last main Reactor Coolant Pump,
- Eq I-131 activity at the end of TEG [GWPS] sweeping before EBA [CSVS] service,
- I-131 activity at the end of TEG [GWPS] sweeping before EBA [CSVS] service,
- I-131 activity before RPV opening,
- I-131 activity criteria before stopping the last main Reactor Coolant Pump.
- Total Gamma and Cobalt-58 in RCP [RCS] before Reactor Building pool filling.
- Tritium controlled in RCP [RCS] before Reactor Building pool filling in order to respect the tritium target value defined for IRWST and pools.

Concerning the chemical parameters, the following controls are performed during outages:

- The boron concentration has to be monitored as a requirement of the reactivity control:
  - In the spent fuel pool, IRWST, RBS [EBS] from states B to E.
  - In the RIS/RRA [SIS/RHRS] during state B.
  - In the REA [RBWMS] boron
- The sodium hydroxide concentration in the EVU [CHRS] tanks from states B to D (maintenance outage) has to be controlled as a requirement of the safety case, taking into account that the probability of a severe accident in state E is very low.
- The hydrogen gas concentration in the different biphasic tanks.
- The oxygen concentration during start-up.

#### 4.2.6. GDA/Licensee Documentation Boundary

The GDA/Licensee chemical specification document boundaries are:

- Chemical control parameters having a direct link to the mitigation and control of the consequences on safety, radiation fields, the environment, hazards, maintenance and operational issues.

These parameters are associated with a limit value, breach of which is covered by the application of a procedure, analogously to the parameters defined in the OTS. The Licensee will be required to put in place a process that manages both the control and monitoring of these key chemical parameters and which defines the procedures in cases where a chemical control parameter is breached.

The approach adopted in developing chemical specification operational documentation will depend on the Licensee.

### **4.3. LOADING CONDITIONS ACCOUNTING [REF-1] TO [REF-2]**

Further information is provided in Sub-chapter 3.4.

The mechanical design of pressurised nuclear equipment is based on the analysis of the overall and local integrity of the primary and secondary components and piping against the five following damage mechanisms:

- Excessive deformation,
- Plastic instability,
- Progressive deformation,
- Fatigue,
- Fast fracture risk.

This demonstration of integrity must consider the following conditions:

- Normal operating conditions (provisional approach considering the operating modes of the plant and operations required by the operator),
- Incident operating conditions (based on feedback experience and probabilistic analysis),
- Accident and post accident conditions (based on safety requirements and probabilistic analysis).

The demonstration of integrity is addressed initially in the stress report for each component: it must be verified for the lifetime of the components. Overall integrity is ensured by confirming that the loading conditions taken into account in the initial design substantiation in the stress report are bounding with respect to the actual loadings and transient situations experienced by the components during their lifetime. The occurrence of each situation on each plant is thus recorded and if the number of permitted occurrences is exceeded, the continued integrity of the component(s) must be justified by calculation.

Section 4.3.1 of this sub-chapter specifies the nature of the loads to be considered for pressurised equipment. Section 4.3.2 of this sub-chapter explains how the accounting of loading conditions is achieved.

#### **4.3.1. Definition of Loading Conditions (see Sub-chapter 3.4)**

The conditions under which pressurised nuclear equipment might operate derive from the operating conditions for the particular system, and also, possibly, from situations specific to the equipment, such as the wrenching torques caused by tightening the vessel closure on the vessel cover. The load conditions experienced by an equipment item are characterised by a set of parameters that define the loads to which it is subjected, which include pressure, temperature, internal and external forces, etc.

The operating conditions for the main primary and secondary circuits (RCP [RCS] and MSS) are defined to include transients anticipated in normal reactor operation and accident and emergency conditions. The set of these conditions is called the "conditions list" and includes events considered in fault studies.

A CPP [RCPB]/CSP [SSPB] condition comprises:

- An initiating event (a normal operating condition, anticipated transient, incident or accident),
- A description of the status of the systems included in the definition of the thermo-hydraulic CPP [RCPB]/CSP [SSPB] transient (e.g. for regulation, limitation, protection etc.),
- The resulting thermo-hydraulic consequences, quantified as variations in temperature, pressure and flow rate,
- The number of occurrences.

The operating conditions for an auxiliary system are defined firstly to be consistent with the condition list for the CPP [RCPB]/CSP [SSPB] (for conditions where the auxiliary system either is, or could be, affected), and secondly to include operational transients affecting the auxiliary system, based on their required performance in both normal and accident conditions.

Under RCC-M principles, plant conditions are classified under six categories:

- Normal operating conditions,
- Upset conditions,
- Emergency conditions,
- Fault conditions,
- Test conditions,
- Hydraulic testing conditions.

Each PCC or RRC condition is covered by at least one CPP [RCPB]/CSP [SSPB] operating condition for which the thermal-hydraulic transient bounds the post-accident transient with regard to its mechanical consequences.

The mechanical design of components relies more specifically on the following classification of operating conditions:

- Category 1, for design conditions,
- Category 2, for normal and upset conditions,
- Category 3, for emergency conditions,
- Category 4, for faulted conditions (which include those resulting from multiple event sequences),
- Test conditions and hydraulic testing.

The link between loading conditions categories and RCC-M categories is summarised in Section 18.2.4 - Table 1.

The objectives to define these loading conditions are the following:

- Maximise fatigue-induced stresses, i.e. systematically apply the most onerous temperatures and pressure changes,
- Maximise rupture-inducing stresses, i.e. systematically apply peak pressure and temperature changes.

#### 4.3.1.1. Category 1 Loading Conditions

The set of loading conditions representative of this category is defined as the conventional design conditions. These are the conditions in which the equipment would be subjected to constant actions at least equal to or above the most severe actions in category 2 loading conditions.

The pressure and temperatures used to define Category 1 loading conditions are called the design pressure and temperatures.

#### 4.3.1.2. Category 2 Loading Conditions: Normal Conditions

Normal conditions are those to which components may be subjected in the course of normal operation, including steady-state operating conditions and transients corresponding to start-up and shutdown.

#### 4.3.1.3. Category 2 Loading Conditions: Upset Conditions

Upset conditions are the conditions to which components may be subjected during transients resulting from normal operational incidents such as a reactor trip, feed-water or reactor coolant pump trip, loss of offsite power, loss of condenser vacuum and failure of a control system component.

##### Specific requirements for Category 2

According to the European Directive 97/23/CE, for Category 2 conditions (situations where the equipment is intended for the systems required for normal operation) it is required that:

- When justified by its frequency of use, a fatigue analysis in compliance with the RCC-M [Ref-1] will be performed for all equipment (see Sub-chapter 3.8),

- The pressure in the equipment is restricted to the maximum allowed pressure. This is the equipment design pressure, although it may be exceeded for short periods,
- Compliance with essential safety requirements in these conditions is established by specific compliance testing.

For the fatigue assessment, the transient conditions selected are based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients that may occur during plant operation

#### 4.3.1.4. Category 3 Loading Conditions: Emergency Conditions

Emergency conditions are the conditions to which components may be subjected in case of infrequent incidents which have a low probability of occurrence but which must nonetheless be considered. Such conditions may result from the failure of one or more independent functions of the reactor and its control system.

The total number of emergency conditions that an item of pressurised nuclear equipment may encounter during its lifetime must not exceed 25. Conditions specified below this threshold are not taken into account in fatigue analysis.

#### 4.3.1.5. Category 4 Loading Conditions: Fault Conditions

Category 4 conditions are conditions that are highly improbable but postulated for equipment and system design.

Condition categories defined in the RCC-M are related only to single initiating events (PCC postulated initiating events) and exclude multiple failure sequences. For category 4 fault conditions, some specific RRC-A events are added to the list of conditions when the mechanical consequences of an RRC-A transient are not covered by an existing condition.

This category includes Pressurised Thermal Shock loadings leading to potentially high thermal stresses. Fast fracture analyses are performed on sensitive zones that could be subject to these high thermal loadings.

#### 4.3.1.6. Test Conditions

Test conditions relate to planned component testing during normal operation, except for hydraulic testing. Test conditions are classified as upset conditions. These testing conditions include hydraulic tests on a single component.

### 4.3.2. Accounting of Loading Conditions

Integrity of components is verified in the initial stress report against all potential damage and all design loading conditions; the demonstration of integrity is ensured during plant operation by verifying that these design loadings cover the actual transients experienced and associated assumptions. This process is called "on-site accounting of transients" and is based on continuous monitoring of the relevant parameters considered in the design.

A specific document for operator application should define the process through which accounting of loading conditions is managed. This ensures that the operator avoids entering into a condition through the application of operating rule requirements.

This process should also precisely define the rules and conditions to demonstrate, considering acquired transients monitoring parameters, that a transient experienced by the plant is correctly covered by a corresponding design transient, for example:

- How to separate one monitored transient from another one,
- How to consider a global transient and included sub transients,
- What are the transients to be analysed for accounting (consider fatigue thresholds),
- How to manage parameter variations compared to design transient parameter variations (gradients).

#### **Summary of frequency of loading conditions**

The information provided in Section 18.2.4 - Tables 2 to 4 is indicative of the typical number of occurrences of loading conditions assumed for the mechanical design analysis. The actual number of occurrences for a specific UK project will be defined by the Licensee in the equipment specification contract. The values in Section 18.2.4 - Tables 2 to 4 are the same as those presented in PCSR Sub-chapter 3.4 and are not intended to replace them.

#### **4.3.3. GDA /Licensee documentation boundary**

A schedule of thermo-hydraulic transient loading conditions containing the associated number of occurrences, taken into account in the mechanical design of pressurised nuclear equipment, will be provided to a future Licensee as an interface document. This loading conditions schedule will enable the Licensee to develop processes to record the actual transients experienced by the plant over its lifetime and thus to manage the number of occurrences of these conditions, to ensure compliance with design assumptions, and ensure that the number of occurrences is not exceeded. The schedule of thermo-hydraulic transient loading conditions will be developed outside GDA as part of Nuclear Site Licensing.

### **4.4. SAFETY ANALYSIS BOUNDING LIMITS AND FUEL DESIGN LIMITS**

#### **4.4.1. Safety Analysis Bounding Limits**

The FA3 safety demonstration is based on design safety studies and also on a specific safety study performed for each reload. This approach has been adopted for the UK EPR.

The input data, the methodologies and the results in comparison with the safety criteria of the design safety studies are included in the Final Safety Analysis Report (FSAR). The analysis is performed with bounding hypotheses for a type of plant where the fuel management is spread out.

In order to simplify the calculations performed for each reload, the defined approach is based on key parameters. The key parameters are determined with simple models (static calculations). If the values obtained on reload are below the limiting value (defined in the FSAR), the safety of the reload is ensured.

Key parameters are defined as representative of the physics of the core (general key parameter) and as representative of an accident (specific key parameters). Several key parameters can be defined for one fault and their evaluation in the reload safety demonstration is necessary.

The link between the design studies and the reload studies is presented in the "DGES" (General Document for Safety Evaluation). This document contains the key safety parameters, their values and their calculation conditions, which are determined by the Design Safety analysis. It also defines all the hypotheses assumed in the safety studies, which must also be respected for the plant (operating parameters and operating conditions, minimum flow rate value...).

A specific document is established for each reload referred to as the Specific Safety Document (DSS). The DSS contains the results of all the calculations performed with the specific conditions of the reload (fuel loading pattern in particular).

#### **4.4.2. Design Safety Studies**

PCSR Sub-chapter 4.3 describes nuclear design of the EPR reactor on the basis of the proposed core loadings. The bounding values of nuclear design encompass:

- Fuel depletion of the proposed core loadings (first core and equilibrium cycles),
- All core states within limiting conditions of operation (LCO),
- All core states within limiting fault conditions,
- Calculation uncertainties.

Transient analyses described in Chapter 14 are performed with limiting hypotheses related to the evolution of core power and reactivity under various design basis conditions. These hypotheses are chosen conservatively to maximise the consequences of each postulated fault.

It must be noted that most of the transient analyses presented in Chapter 14 are illustrative of the EPR reactor and not necessary related to the nuclear data detailed in Sub-chapter 4.3.

Typical input data are defined for the transient analyses. Point-kinetics analyses are performed using hypotheses related to the evolution of core power and reactivity under various design basis conditions in terms of:

- Fuel cycle depletion
- Power level
- RCCA configuration
- Boron concentration
- Thermal hydraulic conditions (pressure / temperature)

The hypotheses used for the bounding nuclear data calculations are chosen conservatively to maximise the consequences of each postulated fault and to bound the design basis conditions studied.



Typical input data for nuclear design boundaries and transient analysis are:

- Doppler power coefficients
- Doppler temperature coefficients
- Moderator coefficient
- Boron coefficient
- Kinetic parameters (delayed neutron fractions)
- Hot channel factor
- Rod worth

The analysis which establishes the safety analysis bounding limit requirements has been performed [Ref-1]. This analysis derives the bounding values of neutronic parameters defined in the nuclear design section (Sub-chapter 4.3) and the assessment methodology applied for their definition.

#### **4.4.3. Reload Safety Studies**

As explained in the introduction to this chapter, not all the parameters are included in operating documents: a few key parameters are calculated to verify the safety of the reload.

Work is in progress to provide details of the input data used in the safety analysis, the output results and the parameters necessary for the operating documentation.

#### **4.4.4. Fuel Parameters and Design Limits**

The technical and safety case analyses to establish requirements related to the fuel and the fuel design are given in the following:

- Sub-chapter 4.2 details Fuel System Design, the design evaluation of fuel material and performance and reactivity control assemblies.
- Sub-chapter 4.3 presents the EPR nuclear design in terms of the design basis, power distribution, reactivity coefficients, core control, control rod patterns and reactivity worth, residual heat curves, vessel irradiation and methods and tools for core calculations.
- Sub-chapter 14.1 details some key parameters for core and fuel design covering main core geometry, reactivity coefficients and fuel management boron concentration, residual decay heat and RCCA characteristics.
- Sub-chapters 4.2, 4.3 and 14.1 are supplemented by the analysis referred to in section 4.4.2 of this sub-chapter for bounding neutronic parameters

- The correct fuel rod behaviour in-core and in long term storage is ensured through the compliance with design limits and acceptance criteria [Ref-1] taking into account:
  - the plant design bases (plant condition categories, fuel rod functional design and safety requirements)
  - the general limits or acceptance criteria for each PCC and for events associated with the fuel storage pool
  - the assessment methods which are related to fuel temperature, cladding thermal criteria, corrosion and hydriding of the cladding, rod internal pressure, cladding mechanical criteria, strains of the cladding, Pellet Cladding Interaction and Stress Corrosion Cracking risk / SCC, cladding freestanding at beginning of life, cladding long term collapse, pellet stack stability, cladding fretting wear, cladding fatigue damage, fuel rod axial growth and the thermal-hydraulic criteria,
  - the safety design criteria related to LOCA, to RIA and other PCC events,
  - the acceptance criteria for fuel storage pool.
  - the technical requirements assumed for the EPR design and fuel management [Ref-2] [Ref-3].

#### **4.4.5. GDA/Licensee Documentation Boundary**

Key safety parameters will be defined and established in the General Document for Safety Evaluation, which will reflect the results of the Design Safety analysis for core loadings and will cover the bounding values of nuclear design. This document will be produced in the Nuclear Site Licensing phase.

The Licensee will be responsible for ensuring that each fuel load is compliant with the requirements of the safety analysis bounding limits and the other assumptions made in the safety analysis.

**SECTION 18.2.4 - TABLE 1**

**Loading Condition Categories**

DESIGN CONDITION CATEGORIES	<i>Loading condition categories</i>					
	NORMAL	UPSET	TEST	EMERGENCY	FAULT	HYDROTESTS
2	X	X	X			
3				X		
4					X	
Hydro-tests						X

**SECTION 18.2.4 - TABLE 2**

**List of Normal Conditions**

N°	Event - Normal Conditions	N° of occurrences
1	<b>Complete plant start-up from cold shutdown to full load</b>	
	Reloading	120
	Repair	120
2	<b>Complete plant shutdown from full load to cold shutdown</b>	
	Reloading	120
	Repair	85
3	<b>Partial plant start-up and shutdown between cold shutdown and 120°C in SGs</b>	60
4	<b>Partial plant shutdown and start-up between full load and 120°C</b>	60
5	<b>Load ramps from 100% to 0% of full load with 5%/min and back</b>	
	5.1)Normal operation:	
	a) 100-0% FP	1200
	b) 0-100% FP	1200
	5.2)Stretch-out operation:	
	a) 100-0% FP	300
	b) 0-100% FP	300
6	<b>Daily load follow</b>	
	6.1)100-60% FP and back (5%/min)	36000
	6.2)100-25% FP and back (5%/min between 100% and 60% FP, 2.5%/min between 60% and 25% FP)	6000
7	<b>Remote control / frequency control</b>	
	Normal operation	
	a) Load steps ± 2.5% full load	8 10 <sup>5</sup>
	b) Load ramps ± 12.5% full load with 1% per min	5 10 <sup>5</sup>
	Stretch-out operation	
	c) Load steps ± 2.5% full load	2 10 <sup>5</sup>
8	<b>Unscheduled / emergency power variations</b>	
	a) Up to 95% FP with+10% step & + 5%/min ramp	1500
	b) Down to tech. min at -20%/min	1500
	c) Step load changes +/-10% FP	750
	d) 25 to 100% FP at +5%/min	1500
9	<b>Unscheduled/spurious fluctuations at hot shutdown</b>	4000

N°	Event - Normal Conditions	N° of occurrences
10	<p><b>Partial reactor power reduction to 25% of full load</b></p> <p>10.1)Normal operation</p> <p>a) with subsequent start-up to full load</p> <p>b) with subsequent start-up to full load (transfer to house load)</p> <p>c) with subsequent hot shutdown and start-up to full load</p> <p>d) with subsequent cold shutdown</p> <p>10.2)Stretch-out operation</p> <p>a) with subsequent start-up to full load</p>	<p>250</p> <p>170</p> <p>30</p> <p>20</p> <p>90</p>
11	<p><b>Return to hot shutdown after stretch-out operation</b></p>	<p>60</p>

**SECTION 18.2.4 - TABLE 3**

**List of Upset Conditions**

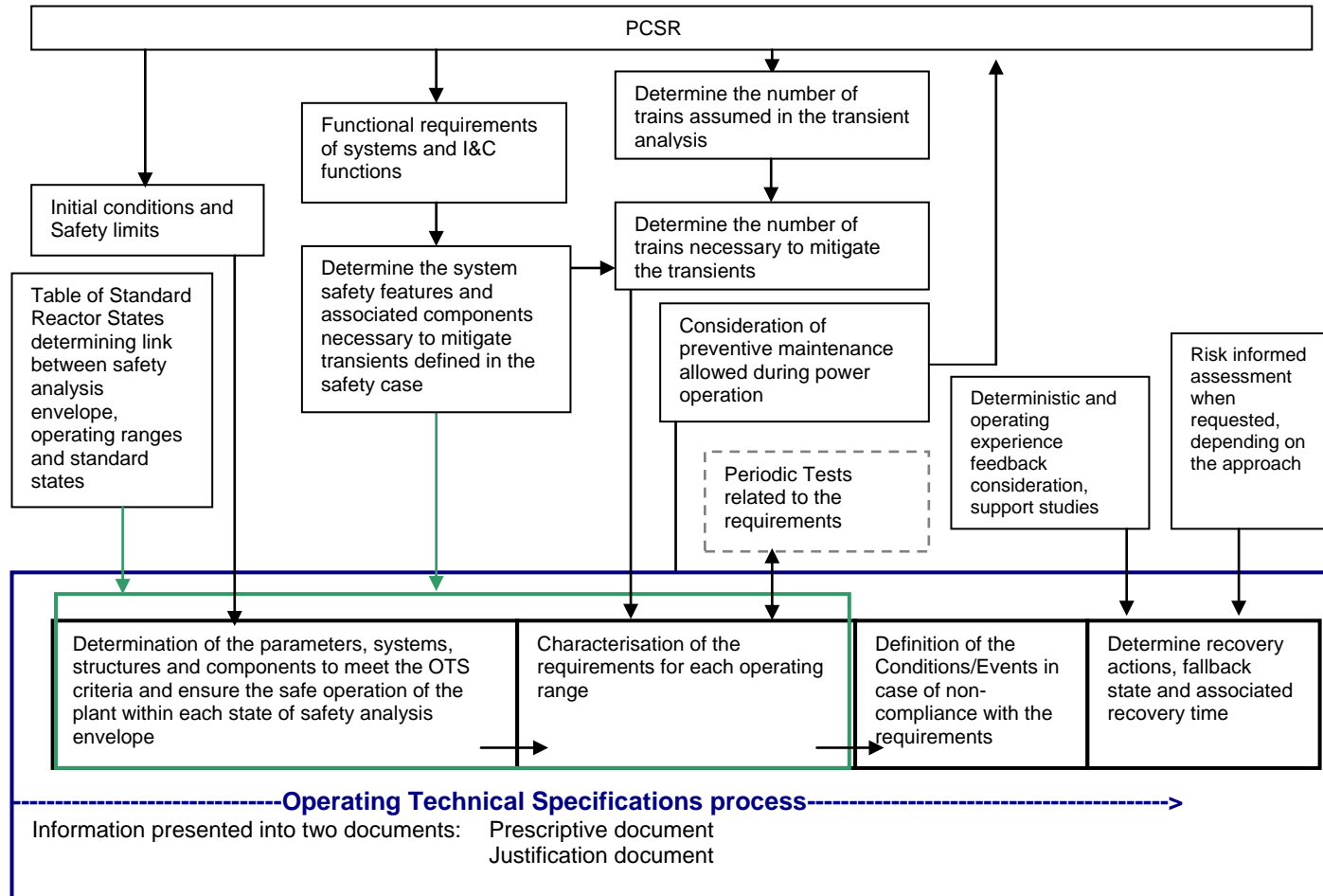
N°	Event - Upset Conditions	N° of occurrences
12	<p><b>Reactor Trip</b></p> <p>12.1)Normal Operation</p> <p>a) with subsequent start-up to full load</p> <p>b) with subsequent shutdown to cold shutdown</p> <p>12.2)Stretch-out operation</p> <p>a) with subsequent start-up to full load</p>	<p>55</p> <p>15</p> <p>20</p>
13	<p><b>Turbine trip with failure of transfer to house load</b></p> <p>The plant is tripped to hot shutdown, with subsequent start-up to full load</p>	60
14	<p><b>LOOP with failure of transfer to house load (short term Emergency Power Mode)</b></p> <p>The plant is tripped to hot shutdown, with subsequent start-up to full load</p>	30
15	<p><b>Loss of Feed-water (loss of 4 ARE[MFWS]-pumps)</b></p> <p>The plant is tripped to hot shutdown, with subsequent start-up to full load</p>	60
16	<p><b>Spurious RCP[RCS] depressurisation (faulty spraying)</b></p> <p>The plant is tripped to hot shutdown, with subsequent start-up to full load</p>	15
17	<p><b>Full load rejection with excessive secondary side heat removal</b></p> <p>Reactor trip with excessive cool-down, with subsequent start-up to full load</p>	15
18	<p><b>Excessive feed-water supply at hot shutdown</b></p>	15
19	<p><b>Significant depressurisation in the secondary side</b></p> <p>leading to significant pressure difference between CPP [RCPB] and CSP [SSPB]</p>	15
20	<p><b>Unscheduled fluctuations in temperature and pressure between cold and hot shutdowns</b></p> <p>Fast fluctuations, low magnitude</p> <p>Ramps of large amplitude</p> <p>Fast fluctuations, large magnitude</p> <p>Larger ramps with larger magnitude</p>	4010
21	<p><b>Maximum SG pressure with an open RCP[RCS]</b></p>	30
22	<p><b>Secondary overpressure: turbine trip at 60% FP</b></p>	15

**SECTION 18.2.4 - TABLE 4****List of Hydraulic Pressure Tests**

<b>N°</b>	<b>Event – Hydraulic Pressure Tests</b>	<b>No of occurrences</b>
	<b>Hydraulic test of individual component before installation</b>	<b>3</b>
	<b>Hydraulic test during commissioning</b>	<b>3</b>
	<b>Periodic hydraulic test</b>	<b>15</b>
	<b>Leak tightness RCP[RCS] test</b>	<b>15</b>

**SECTION 18.2.4 - FIGURE 1**

**Generic Operating Technical Specifications Process**



: Information used as an input data



↔ Consistency ensured between both sets of information



## 5. PERIODIC TESTING [REF-1]

This section presents generic principles and requirements that are used as a basis for producing the Periodic Tests (PT) of the EPR. This section also describes the process used to produce the PT and demonstrates how the results obtained through the safety analyses contained in the PCSR are used to establish the parameters that must be tested. In this way, when the PTs are performed adequately, the safe operation of the plant, within the assumptions contained in the safety case, is guaranteed.

This section also presents the boundary between GDA and Licensee PT documentations.

### 5.1. GENERIC PRINCIPLES FOR PT PRODUCTION

Periodic tests are regulatory tests which must be identified in general operational documentation. The tests form part of the regulatory inspection and maintenance programme that a future Licensee will be required to implement to meet appropriate Nuclear Site Licence Conditions and regulatory requirements.

The object of periodic testing is to verify that, for safety features contributing to safety related functions, the safety criteria defined at the design stage are complied with during the operational lifetime of the unit. In other words, the PTs allow verification that the performance of the system safety feature considered in the fault studies is being maintained over the plant lifetime.

The verifications are carried out in pre-defined configurations, according to a predefined frequency and method.

The main objectives for developing a PT programme in the detailed design phase of the EPR are:

- The optimisation of the total number of tests in operations and therefore of PT,
- The possibility of carrying out PT during power operation which is made possible thanks to:
  - the consideration of an additional train in maintenance in safety analyses,
  - the use of specific test lines (e.g. a line with full flow rate),
  - the use of special requirements or (exceptionally) of limiting conditions for the OTS,
- The limitation of constraints related to the execution of PTs, in particular if they may reduce plant unit availability by:
  - selecting a suitable execution phase (outside the critical unit outage phases or, if needed, in a window compatible with respect to the scheduled outages),
  - adapting the execution conditions by transposing accident studies to actual conditions, in a test with several parts with functional overlap,

- the optimisation of the execution times via automatic sequences, centralised summary information, remote control equipment,
- excluding special measures and resources (as far as possible), adapting the OTS (special conditions or limiting conditions), via equipment dedicated to PTs (test line, PT instruments, etc.), automatic controls, dividing PT into “Sub-PT”, etc.
- To guarantee sufficient margins between equipment performance and the PT criteria in order to avoid problems concerning attainment of operating criteria via sufficient equipment design margins (by taking into account the possibility of performance degradation over time and PT measurement uncertainties), suitable instrumentation, and by taking into account measurement uncertainties during the component design phase.

It should be noted that the automation of PTs is not in itself a goal for the EPR, however it is a possibility available to designers to try to meet the previously defined requirements in the best possible way.

## **5.2. GENERIC PROCESS OF PT PRODUCTION**

This section aims to justify why the PTs carried out by the operator are adequate to ensure the safety limits during the plant lifetime.

The methodology described to produce the preliminary PTs is intended to be generic, reflecting the main input data which should be used to produce the PTs. The main input data which should be used, and their contribution to the PT documents, are described in the Section 18.2.5 - Figure 1 and are discussed below.

### **5.2.1. List of Safety Features Subject to PT**

Safety features used for mitigation of faults are the key elements subject to Periodic Testing. A safety feature is a collection of similar SSCs that contribute to a safety function. For example, the safety feature ‘MHSI start-up’ ensures the RCP [RCS] safety function of ‘water inventory control’ in accident conditions – it is hence a front-line mechanical safety feature [Ref-1]. The UK EPR safety features are identified in the fault studies analysis and the fault schedule.

A system safety feature is not subject to Periodic Testing if:

- It is regularly required to operate under normal operations that are comparable to those in which it would be used during incident or accident scenarios and it is monitored continuously (either by computer systems or by common operations controls during rounds, shifts, etc.),
- It is entirely actuated by programmed machines with self-monitoring functions and functions for reporting any detected malfunctions,
- It is tested by another system.

Developing the list of safety features involves an exhaustive analysis to ensure that any safety feature important to safety and relevant for PT is captured. A justification for not considering a safety feature in the PT list must be provided during the exhaustive analysis.

## 5.2.2. Definition of Safety Feature Criteria to be subjected to PT

### 5.2.2.1. Basis

A detailed description of the safety features including the main components must be produced in order to assign accurate criteria. This involves mechanical front-line safety features and Instrumentation & Control (I&C) safety features (signals with sensors) required to produce the safety feature signals.

The numerical values for the mitigative safety features are those provided in Sub-chapter 14.1, which may be supplemented by data from System Design Manuals. The numerical values form the basis for determining the criteria that should be checked under normal operation: it must also be determined whether or not the criteria are accessible in the design states or in the operating state of the system feature. If not, it will be necessary to transpose the criteria into an accessible condition.

The transposition of different criteria may lead to the establishment of a test by part (basic safety feature). For example testing of the ASG [EFWS] the system safety feature ASG-Fs-01 [Ref-1] requires the verification of the following criteria:

- Flow rate is established within 15 seconds.
- Flow rate criteria (90t/h at 97bar abs) → criteria transposed to a HMT(Q) criteria in zero flow (1 line/month) and full flow (all cycles)

For any given system safety feature, several PTs may therefore be necessary, each to verify one or more criteria. One or more values may be associated with each criteria depending on the configuration used. If this is the case, the aim of the analyses will be to define one or more PTs containing these different values.

The criterion chosen defines a binary PT. A test can only be positive or negative. If the value checked fails to verify the criterion defined, the feature or part-feature tested is considered to be unavailable, even if some criteria relating to the feature or part-feature meet the reference values.

### 5.2.2.2. Establishing Original Measured Values

It is necessary to establish original values for criteria during the commissioning phase in order to:

- Allow the designer to approve the design (i.e. to ensure that there is a sufficient margin between the measured value and the reference value),
- To allow periodic assessment of changes in measured values over time during operation.

Consequently, all measured original values that correspond to the criteria that must be complied with in terms of PTs, must be measured and archived during the first commissioning tests of the plant.

Values measured whilst carrying out PTs are compared with the criteria reference values (safety criteria) to assess whether or not criteria have been met and also with the original measured values to determine if there has been any possible degradation.

### 5.2.2.3. Uncertainty of Measurements and Values of Criteria

Uncertainties on measurements shall be taken into account as much as possible in the definition of operational documentation. However, it is not considered as part of the exhaustive analysis as presented in this section.

The criteria and their associated values for verification in terms of the PTs are either analogue (e.g. flow rate), or binary (for example the opening of a valve). Analogue values can be defined, either as a single value or a value range with an upper and lower limit. For analogue values uncertainty due to instrumentation systems may be taken into account. This implies the raw reference values from the safety report must therefore be corrected if metrological uncertainty has not been taken into account.

### 5.2.3. Pre-defined Configuration for PT

The link with the fault studies determines the operating conditions of the safety features as well as their duration.

The definition of these configurations must take into account the following two elements:

- The specific criteria defined in the safety case for the particular theoretical configurations (notably incidental or accidental configurations, limited or compromised operation) must be transposed into normal operating conditions.
- PTs that may affect the duration of reactor shutdown must be minimised. Considering the number of trains available, tests during operation are preferred. Safety criteria resulting from the safety case must be transposed into this state.

Justification for the configurations used and criteria transposed should be subject to justification documents. When the tests are not carried out according to safety case conditions, specific studies justifying the change in criteria and their values must be produced. Such documents should state how the transposition values were derived.

For each criteria or value to be measured, it is necessary to determine the appropriate reactor state by taking the following into consideration:

- The PT should be scheduled as far as possible whilst the reactor is at power,
- If the PT is scheduled during a reactor outage then an analysis of the consequences of the test on the reactor outage schedule should be carried out.
- The list of the reactor states should be as broad as possible.

If some PTs lead to exceedance of the OTS requirements, the Licensee should put in place a mechanism to manage any conflicts between OTS and PT requirements.

### 5.2.4. Frequency of PT

Since determination of the frequency involves the verification of the availability of the safety features or part-features, it must be carried out by the designer using feedback from similar facilities, the provisional reliability of similar features, and the impact of the loss of classified safety features with regards to the risk of core damage or loss of containment (PSA based).

The frequency of a PT may be:

- Event-based: these are tests to be carried out according to a schedule, at each reloading (1 Reactor Trip (RT), 4 RT, etc), by cycle (1 cycle, 2 cycles, etc: the test can be carried out while the unit is in operation), after loading a piece of equipment, at restart after shutdown for reloading, during 10 year maintenance, etc.

For this type of frequency no tolerance is permitted in respect of frequency.

- Calendar-based frequency: these are tests which are to be carried out according to a daily, weekly, monthly, annual or five-yearly frequency.

For a calendar-based frequency, dates for tests are fixed on set dates at the required intervals. Tolerance is accepted, depending on the frequency, around each of these dates to provide the necessary flexibility to carry out the test effectively, in order to take account of the state of the facility and technical contingencies. Use of this tolerance must not under any circumstances lead to a delay in the planned date of the next test.

The definition of the periodicity of the test is based on three approaches:

- Consideration of existing plant feedback,
- Regulations and legislation,
- The use of Probabilistic Safety Analysis (PSA).

#### **Existing plant feedback**

In determining the frequency of a periodic test any relevant operational feedback that is available should be taken into account. This operational feedback may help to decrease the PT frequency of a given component already used for many years on NPPs providing adequate justification is supplied.

#### **Regulations**

The PT frequency may also be subject to a specific regulation (fire, electrical, switchboards, etc.).

#### **Probabilistic Safety Analysis (PSA)**

The aim of PT is to check that the system safety features adhere to the initial design criteria. The test frequency can therefore be determined based on the probabilistic evaluation of the influence that the unavailability of a certain system safety feature would have on the risk of core damage (PSA level 1).

It should be noted that the PSA used to determine the PT frequency does not take into account all equipment (e.g. polar crane, etc.) and can therefore be considered as a tool to help the relevant subject matter rather than to determine the frequency. Consequently, the use of the PSA may justify some PT frequencies.

Special attention will be paid to the PT frequency of I&C systems as the reliability claims [Ref-1] placed on the I&C platforms (software and hardware) need to be substantiated in the EPR detailed design phase. This link between reliability claims and PT is captured in the safety requirements of the Non-Computerised Safety System (NCSS) [Ref-2].

### **5.3. DEFINITION OF RULES AND REQUIREMENTS FOR PT**

In order to ensure consistency between the design phase and the PTs made during plant lifetime, it is necessary to define some requirements for the operator to follow.

#### **5.3.1. PT Validity Conditions**

The criteria, associated values, frequency and configurations of the PT are defined at the design stage and validated during first commissioning tests or modification of a system. These prescribed criteria, values etc. cannot be changed or deviated from at the time of carrying out PTs.

A PT can only be carried out if all the equipment required for the system safety feature or part-feature to be tested is available.

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

If the PT procedure cannot be entirely applied, due to a failure resulting in some values not being recorded, it is possible to continue the PT, according to tolerance in its frequency, after analysis of the configuration necessary to acquire the missing required criteria.

In the event of the modification to the facility, the facility must be qualified and a review carried out of the design of the PTs relative to the modified function (criteria and values, configuration, frequency, operating procedure).

#### **5.3.2. Measures Replacing the Need for a PT**

The automatic monitoring of a safety feature can replace the need for PT of this safety feature (e.g. automatic self test of I&C systems).

Furthermore, inadvertent actuation of a system safety feature may, where applicable, after analysis of the situation and the sequence of the action, replace the need for a PT of a system safety feature provided the performance conditions meet the PT requirements and the values recorded conform to the validation criteria of the PT.

#### **5.3.3. Functions Subject to PT Part by Part**

The safety features may be tested part by part as long as the various parts overlap functionally. It is recognised that verification of the criteria corresponding to the safety feature is then carried out via verification of elementary criteria of each of the parts, possibly at different frequencies.

Despite the possibility of dividing the PT of a safety feature into PT of individual parts, the resulting tests ensure the performance of the safety feature in its totality.

### **5.4. CORE PHYSICS TESTS (RELOADING TESTING)**

The methodology and principles taken into account in the definition of specifications for the Core Physics Tests and the Core Physics Tests that must be performed after each refuelling are described in [Ref-1].

Start-up physics testing is part of the overall start-up phase. The contents of tests must be defined consistent with the purpose of the start-up phase i.e. for the initial core loading, or subsequent to each new fuel reload.

For each fuel reload, nuclear design calculations are performed to ensure that the reactor physics characteristics of the new core are consistent with the safety limits. Therefore, prior to a return to normal operation, successful execution of a physics test programme is required to determine if the operating characteristics of the core are consistent with the design predictions and to ensure that the core can be operated as designed.

The General Principles for Definition of Core Physics Tests [Ref-1] define the acceptance criteria for the core physics parameter measurements.

One of the main purposes of the Core Physics Tests that are performed after each refuelling is to check consistency of the newly loaded core with the design studies, to confirm that the reactor can operate safely.

As long as no basic change has occurred since commissioning, in the systems which could affect protection, surveillance, limitation and control, the overall demonstration done during commissioning remains applicable, and the core physics tests carried out after refuelling, for core related I&C functions are limited to the updating of calibration coefficients only. In the plant schedule after each refuelling outage, the main phases where Core Physics Tests take place are:

- Approach to criticality and zero power tests,
- At power tests

Tests are carried out at initial criticality and initial low power to:

- Confirm that the reactor physics parameters are in accordance with predictions made in the design,
- To measure the moderator temperature reactivity over the temperature range in which the reactor may become critical,
- To determine the reactivity worth for control rods and the control rod bank,
- To compare the actual critical configuration with the predicted configuration

The Core Physics Tests carried out at power during power raise after refuelling are limited to core power distribution checks and to determination of calibration coefficients of I&C Core Functions.

During the latter phase the main measurements and checks carried out to evaluate the core performance are:

- Reactor power measurements,
- Verification of the calibration of I&C "Core Functions",
- Verification of the core power distribution.

## 5.5. GDA/LICENSEE DOCUMENTATION BOUNDARY

The exhaustive analyses and reload testing as described above form the interface between the EPR designer and the future Licensee.

A complete set of exhaustive analyses documents will be produced covering all system safety features. This in turn will form the basis of the recommended PT programme for the EPR. The documents will be submitted during the NSL phase and will be consistent with the design freeze.

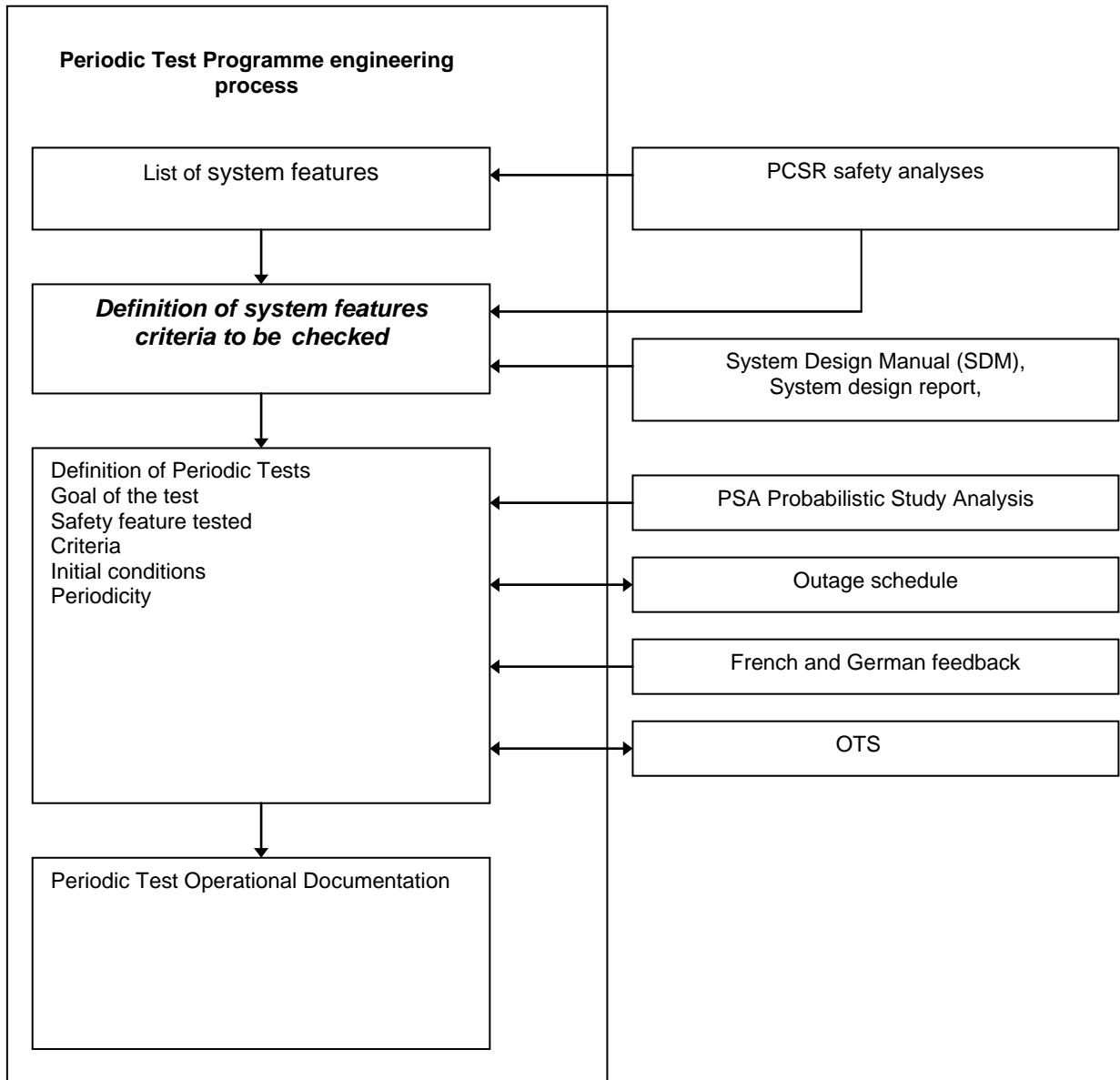
The transformation of the analysis documents into PT operational documentation is the responsibility of the Licensee. The implementation of PTs recommended by the designer and any associated Maintenance Schedule requirements is also the responsibility of the Licensee.

The actual PT and maintenance intervals underpinning EDF derived component failure probabilities are out of scope for GDA, but will need to be adhered to in practice on a UK EPR, or alternatives justified.



**SECTION 18.2.5 - FIGURE 1**

**Periodic Test Programme Engineering Process**



## 6. IN SERVICE INSPECTION AND MAINTENANCE

### 6.1. IN-SERVICE INSPECTION

As a part of the second level of the defence in depth approach adopted for the FA3 EPR, and proposed for the UK EPR, to prevent any loss of integrity, In-Service Inspection (ISI) is a preventive maintenance process focused on non-destructive examinations for equipment. These examinations constitute a monitoring and maintenance programme that is scheduled and implemented during planned outages.

#### 6.1.1. Pre-Service Inspection (PSI)

Regulatory requirements applied for the FA3 design, corresponding to the application of regulations for monitoring the operation of the Main Primary System (MPS) and Main Secondary System (MSS) in nuclear plants, places an obligation on the operator to proceed with inspection of the components making up these systems before the first fuel loading in order to:

- Provide a reference for future comparison and interpretation,
- Check that the measures taken in the design process and during manufacturing and erection are adapted to provide satisfactory monitoring during operation (accessibility of the areas to be controlled, satisfying surface quality...),
- Carry out an ultimate examination of the components before their start-up.

This visit is called Pre-Service Inspection (PSI). The FA3 PSI will contribute to and form the basis of the intended UK EPR Pre-Service Inspection requirements. The UK programme will address the requirements in the UK regulatory context.

For the main primary and secondary systems (MPS & MSS), Pre-Service Inspection (PSI) consists of implementing initial controls, the outcomes of which will be the reference for ISI during plant lifetime.

PSI provides the baseline for surveillance tests performed during full and partial outage inspections. It consists of a series of non-destructive tests after the completion of manufacturing and erection operations, in order to define an initial state of components involved.

#### 6.1.2. In-Service Inspection (ISI)

In-Service Inspection implemented during operation is defined on the basis of the engineering design, in particular the mechanical analyses and feedback concerning the risk of damage in specific areas. All class 1 mechanical components, which require in-service inspection, are designed, manufactured and assembled so that all of the welds can be inspected. It also considers the maintenance policies and the operational experience feedback from the N4 and Konvoi plants, from which the EPR design has been developed.

- In-service inspection must be carried out in the areas that may be sensitive to fatigue damage with the appearance of defects that can be initiated and be propagated in operation. The location of fatigue damage areas depends on the design of components and on their operating conditions. These areas must be identified at the design stage.
- For the areas considered as non-sensitive to fatigue damage at the beginning of the design, in service inspection is carried out as a part of the defence in depth approach. The selection of these areas depends on the feedback from the N4 and Konvoi with regard to the risk of damages.
- For all the areas to be inspected, it is verified that there are no accessibility and controllability problems. It is also checked that the performances expected during the non-destructive tests of in-service inspection are in accordance with the design and manufacture. The components must be designed and installed to ease the inspection of these areas.

In-service inspection is also intended to confirm the absence of notable defects in a number of RCP [RCS] boundary locations selected in advance. This is one of the components of in-service surveillance for the second line of defence in depth to underpin the break preclusion safety case, and is therefore independent of the design and manufacturing quality process.

The detailed programme will be finalised during the detailed design phase and will take into account feedback experience, available non-destructive testing (NDT) techniques, the recommendations of the Safety Authority, and the results of Pre-Service Inspection (PSI) before start-up.

The PSI and ISI programme for UK EPR will be developed in the Nuclear Site Licence (NSL) Phase.

Typical examples of ISI controls are the following:

- Ultrasonic Testing of the core shell welds,
- Radiographic Testing of SG nozzle welds,
- Remote Visual Tests of RPV or SG internal cladding,
- Eddy Testing of RPV nuts or SG tubes

### 6.1.3. Reactor Coolant Pressure Boundary (RCPB) In-Service Inspection

This section mainly deals with ISI of the Main Reactor Coolant Pipe-work the details for which are provided in Sub-chapter 5.2 and Sub-chapter 5.3.

ISI is intended to confirm the absence of notable defects in a number of locations selected in advance. This is one of the components of in-service surveillance for the second line of defence-in-depth to underpin the break preclusion safety case, and is therefore independent of the design and manufacturing quality process.

The detailed programme will be finalised during the detailed design phase and will take into account feedback experience, available non-destructive testing (NDT) techniques, the recommendations of the Safety Authority, and the results of pre-service inspection (PSI) before start-up.

The In-Service Inspection (ISI) programme is a surveillance programme optimised by the break preclusion principle by permitting the inspection of areas that would not be accessible if anti-whip mechanisms were installed.

All welds, and more generally all areas of the reactor coolant system loops subject to an in-service inspection programme, will be accessible and be inspected in accordance with the technical guideline requirements. This has been achieved and demonstrated for FA3 EPR In-Service Inspection [Ref-1] [Ref-2].

In particular, a comprehensive study of the end-of-manufacturing accessibility and inspectability of the Main Coolant Line (MCL) welds has been performed [Ref-3], especially for relevant UT techniques. This study covers the MCL design state after modification of the crossover leg as described in Sub-chapter 5.4 section 3. It has also covered in-service accessibility and shows that this accessibility is sufficient to enable the deployment of the Pre-Service / In-Service Inspections yet to be developed.

In such areas, the non-destructive testing equipment used will be qualified. All homogenous welds will allow inspection by at least one volume inspection method. Heterogeneous welds will allow inspection by two volume inspection methods in accordance with the technical guideline requirements in the FA3 PSI.

#### **6.1.3.1. In-Service Inspection of the Reactor Coolant System (RCP [RCS])**

This section concerns the In-Service Inspections that are carried out in general on the Reactor Coolant System (RCP [RCS]) as part of ensuring the integrity of its pressure boundary. It also applies to the secondary side of the steam generators.

#### **6.1.3.2. Main Potential Damage Mechanisms Taken into Account**

The structural integrity of the RCP [RCS] pipe-work is based upon a design that reduces the likelihood of any damage occurring. The following types of damage to the RCP [RCS] pipe-work are considered:

- General damage, such as excessive/progressive deformation, or plastic instability, which may result from thinning of the wall through corrosion or general wear and tear
- Local damage or a pre-existing defect (or one initiated by fatigue or corrosion), which may result in a sudden leak or fast fracture risk, after a propagation phase

#### **6.1.3.3. RCPB In-Service Inspection Programme**

All class 1 mechanical components, such as the reactor pressure vessel (RPV), the main coolant lines (including the surge line), the steam generators (SG) and the pressuriser (PZR), which require ISI, are designed, manufactured and assembled to permit all welds and all areas to be inspected.

Section 18.2.6.1 - Table 1 establishes an initial provisional list of the typical areas that could be subject to ISI. This analysis is based both on the experience gained from similar designs and on specific analysis carried out for the EPR.

The ISI programme has been based on the mechanical analysis results (fatigue, fast fracture, etc) and on feedback knowledge in specific areas (mechanical problems, for example). It has been verified that there are no accessibility problems for any area. In addition, it has been verified that the expected performances during the non-destructive ISI tests are consistent with design and manufacture (surface quality, geometry, etc.). Non-destructive tests during manufacture must show that no unacceptable defects exist. For all workshop welds that require ISI, non-destructive tests are carried out using the same method. They have the status of "preliminary inspections" which provide a reference point for subsequent inspections.

Detailed analyses identify the areas that are potentially sensitive to damage such as fatigue or fast fracture. In this case, these areas are included in the ISI programme.

A selection of welds where the combination of loads and material properties is the most unfavourable, along with a selection of less sensitive welds, are included within the ISI programme as part of the defence-in-depth approach. The ISI programme is reduced compared to the equivalent programme for sensitive areas if it has been confirmed during the analysis that:

- There is no risk of damage during operation,
- The design fulfils the RCC-M criteria [Ref-1] (see Sub-chapter 3.8),
- During manufacture, and based on non-destructive tests, there are no unacceptable defects in these areas.

The non-destructive tests are qualified in accordance with relative regulatory rules.

#### **6.1.3.4. Access Necessary to Inspect RCPB**

Welds forming part of the RCPB must be accessible and available for inspection.

Pipe-work systems that require surface or visual inspection are designed to enable suitable access and visibility to allow such inspections to be carried out properly. Access for ISI of the key components of the reactor coolant system other than the reactor vessel will be provided as follows:

- In general, work platforms or temporary scaffolding will be supplied to facilitate access to the areas to be inspected,
- Manholes are designed for entry into the steam generator water chamber to provide access for internal inspection,
- A manhole is built in to the upper spherical head of the pressuriser to permit internal inspection,
- The insulation covering all component welds and the adjoining pipe-work is removable in those areas where external inspection is planned.

The reactor pit is designed with an access area reserved for staff during refuelling operations to permit external inspection of pipe-work and heavy components

#### 6.1.4. Reactor Pressure Vessel In-Service Inspection (see Sub-chapter 5.3)

The internal surface of the vessel may be inspected during the ten-yearly ISI, using the In-Service Inspection Machine (MIS). The lower core support structure may be removed, making the entire internal surface of the vessel accessible.

The closure head is examined during each refuelling outage. Optical devices permit a selective inspection of the cladding and of the seal-seating surface.

The closure studs must be inspected periodically by visual examination, by dye penetrant tests and by eddy current tests.

During the detailed analysis, an "Initial Scheme of Maintenance" will be established for the in-service surveillance of the RPV and the closure head. This will allow a precise definition of the zones to be inspected and the types of non-destructive examination (NDE) to be used. All the following welds are accessible for ISI:

- Vessel shells – from the inside surfaces,
- Lower head – from the inside surfaces,
- Dissimilar welds between the reactor vessel nozzles and the reactor coolant pipe-work to examine the welds through the cladding from inside.

The design considerations that were incorporated into the system design to permit the above inspections are as follows:

- All reactor internals are completely removable,
- The closure head is stored dry on a stand during refuelling to facilitate direct visual inspection,
- All reactor vessel studs, nuts, and washers can be removed to dry storage during refuelling,
- The insulation covering the welds between nozzles and reactor coolant pipe-work may be removed.

The insulation of the bottom head has removable panels which provide access to enable visual inspection of the bottom head and can facilitate, if required, the installation of an automatic inspection device on the outside.

Access to the reactor vessel body is limited because of radiation levels and remote underwater accessibility to this component. For these reasons, several steps have been incorporated in the design and manufacturing procedures to facilitate periodic non-destructive tests, which the regulatory guides require. These are:

- In-shop ultrasonic examinations are performed on the most critical internally clad surfaces to assure an adequate cladding bond to allow later ultra-sonic testing of the base metal from the inside surface,
- The design of the reactor vessel core shell is an uncluttered cylindrical surface to allow future positioning of the test equipment without obstruction,

- The weld deposited clad surface on both sides of the welds to be inspected are specifically prepared to ensure the effectiveness of ultrasonic examinations,
- During fabrication, all full penetration ferritic pressure boundary welds are ultrasonically examined.

The vessel design and construction are intended to allow inspection as required by the regulations that apply to French nuclear power plants and it is expected to meet UK regulatory requirements.

### 6.1.5. Main Secondary System In-Service Inspection

This section mainly deals with ISI of the main secondary pipework, the details for which are provided in Sub-chapter 10.3.

ISI is intended to confirm the absence of notable defects in a number of locations selected in advance. This is one of the components of in-service surveillance for the second line of defence-in-depth to underpin the break preclusion safety case, and is therefore independent of the design and manufacturing quality process.

The detailed programme will be finalised during the detailed design phase and will take into account feedback experience, available NDT techniques, the recommendations of the Safety Authority, and the results of PSI before start-up.

The In-Service Inspection (ISI) programme is a surveillance programme optimised by the break preclusion principle by permitting the inspection of areas that would not be accessible if anti-whip mechanisms were installed.

All Main Steam Line (MSL) welds subject to an in-service inspection programme will be accessible and be inspected in accordance with the technical guideline requirements. This has been achieved and demonstrated for FA3 EPR In-Service Inspection [Ref-1] [Ref-2].

In particular the systematic review of MSL welds confirms that all welds are physically accessible and are designed to facilitate the in-service inspections of the whole volume of the welds [Ref-3]:

- The examinations of the MSL welds are possible by scanning from outside of the pipe as is the case for in-service inspection.
- The full volume of the welds are accessible for scanning with angle beam 45° - 65° - 70° from both sides of the welds for the majority of welds or at least from one side with a complete examination in the other cases.
- The counterbores do not interfere with the NDT examinations because the counterbores are adequately long.
- The influence of the tapered surface has been taken into account by the use of a gentle and continuous slope in the design of certain MSL components.

### 6.1.6. In-Service Inspection Principles Excluding Main Primary and Secondary Systems (MPS/MSS) (see Sub-chapter 6.5)

ISI is a preventive maintenance operation involving NDE and checks on equipment. These checks and examinations constitute a monitoring and maintenance programme (or inspection plan), which is systematically scheduled and implemented during planned outages. This section addresses ISI carried out on the nuclear island pressurised equipment, excluding the MPS/MSS. This includes pressurised auxiliaries, safety auxiliaries and parts welded to pressurised components.

#### 6.1.6.1. Safety Requirements

In principle, ISI are performed in areas that are sensitive to the development of different kinds of damage (e.g. crack initiation and propagation under fatigue conditions, corrosion, vibratory fatigue, radiation damage, fast fracture damage, etc.).

These sensitive areas depend on the design of the equipment and its operating conditions. They are identified at the design stage.

Equipment is designed and installed to facilitate the inspection of sensitive areas (i.e. it is accessible and can be inspected).

For areas not considered to be design-sensitive, in-service inspection is performed by sampling, based on the defence-in-depth principle. This type of inspection is also performed in design sensitive areas to meet regulations relating to pressure retaining boundaries.

For areas where radioactivity is important, design, construction and installation provisions ensure that the collective dose impact of in-service inspections is minimised as far as reasonably practicable.

#### 6.1.6.2. Areas Considered

The areas considered for ISI are those, for example:

- Containing welded joints,
- Potentially presenting a risk of in-service damage, linked in particular to:
  - the thermal, thermal-hydraulic and mechanical loads applied to the equipment,
  - the action of the fluids in contact with the equipment.
- Which are subjected to periodic inspection in accordance with pressurised equipment regulations.

### 6.1.7. Principles

#### 6.1.7.1. Scope, Type and Frequency of In-Service Inspection

The scope, type and frequency of ISI depends on:

- The level of potential damage during service and the dynamics of the potential development of the damage,



- The importance to safety of the consequences of equipment failure,
- Operational feedback,
- Regulatory requirements,
- The equipment state.

#### **6.1.7.2. Principles Applied at the Design Stage**

The designer is required to:

- Anticipate the damage mechanisms that are likely to affect the equipment, in order to eliminate or minimise them as much as possible,
- Identify items of equipment that need to be inspected and to specify the means of inspection. This will take into account the residual risks not eliminated at the design stage as well as applicable regulations,
- Make provision in the design for easy access to items of equipment to be inspected. Access should allow detection and characterisation of any cracks taking into account the materials used, surface conditions, the geometry of the items to be inspected and location and geometry of the welds themselves,
- Recommend examination methods to be used in specific inspections. In principle, visual and dye penetrant examinations are preferred. Televisual examinations may be used, particularly on surfaces that may be contaminated with radioactive particles. Radiographic or ultrasonic examination of the main welds in pressurised items may be required, as may automatic or remote controlled examinations. It is expected that the design will minimise the necessity for such techniques.

#### **6.1.7.3. Principles Used for the Operational Phase**

The plant operator establishes maintenance and monitoring programmes including appropriate frequency of implementation, based on:

- Initial maintenance plans, by type and group of equipment, taking into account operational feedback (reported cracks and degradation, material properties, operating incidents), statutory requirements and the importance to safety of the consequences of equipment failure,
- Manufacturers' Instructions which take statutory hazard analysis results into account together with the associated recommended in-service inspections,
- The requirement for the equipment to remain effective.

#### **6.1.7.4. Measures for Facilitating Inspections**

Measures provided to facilitate the inspections, in addition to the design, construction and installation provisions already made, are:

- For equipment, provision of:
  - sufficiently large manholes and hand-holes,
  - eyeholes on pressure housings to allow visual examinations from a distance (e.g. shell side of the heat exchangers),
  - access areas for remote examination of the insides of heat exchangers,
  - plugs for radiographic inspection access holes,
  - locally removable insulation,
  - means for full venting and drainage.
- For the area around equipment, provision of:
  - temporary or permanent means for accessing the areas to be controlled (cellular metal floor, etc.),
  - biological shielding in the vicinity of radiological hot spots,
  - automatic or remote control inspection means in order for personnel to remain distant from radiologically active areas,
  - sufficient clearance to allow examination of items to be inspected.

#### **6.1.8. GDA/Licensee Documentation Boundary**

The output of the generic design assessment for ISI will be an outline of the PSI and the ISI programmes which will confirm that equipment identified in these programmes is accessible and that inspection of the equipment is feasible, due to measures that have been incorporated into the EPR design.

The detailed ISI/PSI programme will be defined in the NSL Phase of the UK EPR licensing process.

**SECTION 18.2.6.1 - TABLE 1****Initial provisional list of typical areas that could be subject to In-Service Inspection**

<b>Component</b>	<b>Sensitive area</b>	<b>Damage</b>
<b>Reactor Pressure Vessel</b>	Stud and threaded holes	Fatigue
<b>Main Coolant Lines</b>	RCV [CVCS] charging nozzles	Fatigue
	Surge Line nozzle	Fatigue
<b>Steam Generator</b>	Tube sheet / channel head weld	Fast fracture
	Tube sheet / secondary shell weld	Fast fracture
	Channel head / partition plate weld	Fatigue

## **6.2. MAINTENANCE PROGRAMME**

The maintenance regime is operator dependent. For UK EPR equipment details supplied by the manufacturer and environmental qualification and operational experience would be used to generate maintenance schedules in compliance with requirements of the Nuclear Site Licence Conditions (predominantly, Licence Conditions 28 and 29).

The maintenance regimes will be established as the plant is commissioned and put into service. The aim of the following section is to provide a general overview.

### **6.2.1. Safety requirements**

#### **6.2.1.1. Objectives and Definitions**

Preventive maintenance includes inspections, tests, maintenance, and repair and replacement activities aimed at reducing the frequency of equipment failure. These operations imply planned unavailability of equipment for maintenance purposes during unit operation or outages, irrespective of the occurrence of failures.

Preventive maintenance is considered in the plant safety analyses.

#### **6.2.1.2. Regulatory Framework**

The EPR Technical Guidelines presented in PCSR Sub-chapter 3.1 address the issue of the preventive maintenance (see Chapter C.2.1, single-failure criterion and preventive maintenance). In addition, the principles of preventive maintenance must be consistent with requirements of Chapters C.2.2 (Probabilistic Safety Analyses and diversity), C.3 (Human Factors), C.4 (Radiation protection for workers and the public) and D.2.1 (Safety Analysis Rules) of the Technical Guidelines.

#### **6.2.1.3. Deterministic Safety Requirements**

The system function performed by equipment taken out of service for preventive maintenance is considered as inoperable. However, if the nature of the preventive maintenance is such that the system may be made operational within a suitable timeframe such that the system function can be performed if demanded, the system function may be considered as being operable.

With regard to PCC and RRC events, Chapters 14, 16, and 13 discuss how system unavailability due to preventive maintenance is taken into account in the related safety analyses.

When the unit is in operation, preventive maintenance may not be carried out on more than one train at a time.

Periodic tests are designed to avoid system safety functions being made unavailable on the systems being tested, except where a specific case is justified.

#### **6.2.1.4. Radiation Protection Requirements**

The conditions for carrying out preventive maintenance take into account the radiation protection measures set out in Sub-chapter 12.3.

### 6.2.1.5. Deterministic Human Factors Safety Requirements

Preventive maintenance activities take Human Factors into consideration, as described in Sub-chapter 18.1.

## 6.2.2. Definition of Preventive Maintenance

### 6.2.2.1. Objectives of Preventive Maintenance

By definition, 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 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 protection of the environment, staff safety, radiation protection and other regulations in force:

- The safety level defined in the design study phase is ensured by maintaining the equipment reliability at the required level
- Unit availability is optimised by:
  - Carrying out part of the preventive maintenance while the unit is in operation in line with the assumptions of the safety analysis
  - Enhancing the design and construction quality to minimise unplanned unit unavailability and meet the required outage durations when the unit is in a shutdown state
  - Enhancing design quality by incorporating operating experience feedback
  - Optimising the balance between preventive and corrective maintenance
  - Optimising the balance between systematic preventive maintenance and condition monitoring
  - Defining a provisional schedule for alternating maintenance activities between the different types of unit outage within the ten-yearly in-service inspection intervals
  - Studying at the design stage, exceptional maintenance situations (intervals in excess of 10 years or hypothetical occurrence).

Requalification tests are carried out at the end of all maintenance operations. Requalification tests after preventive maintenance work on equipment make it possible to check that the equipment has the same performance as it had prior to the preventive maintenance work being carried out. Consequently, these tests are sufficient to determine whether the equipment is operable after the maintenance operations.

The requalification tests, and criteria to be checked, are specific to the work carried out. Generally, the tests comprise two complementary parts:

- Intrinsic requalification: these tests are always required. They are limited to the equipment on which the intervention has taken place.
- Functional requalification: this covers testing of the equipment in the environment in which it must operate, including the system which contains the equipment. It is carried out in the normal operating configuration or a configuration representative of normal operation.

For preventive maintenance activities which are carried out while the unit is in operation, consistency between the requalification test and the maintenance activity will be sought. Therefore the level of preventive maintenance carried out during operation may be limited due to the restrictions in requalification tests possible on an operating unit. Similarly, the requalification scope will be adapted to the maintenance operations carried out.

#### **6.2.2.2. Objectives of Preventive Maintenance**

In order to ensure that the UK EPR design meets the ALARP objective, aspects related to availability, operating costs and radiation protection must be considered in the design process.

Consequently, in defining the principles of preventive maintenance, account needs to be taken of:

- the Probabilistic Safety Analyses (PSA) targets for the frequency of core melt (see Sub-chapter 15.0)
- Dosimetry limits (collective dose less than 0.35 mSv/year, see Sub-chapter 12.1)
- Optimised preventive maintenance programme
- The global aim of 91.1% availability for an 18-month cycle, for a unit service life of 60 years, incorporating the following sub-targets:
  - Unplanned unavailability of less than 2%
  - Normal Refuelling Outage (NRO) in 16 days
  - Refuelling Only Outage (ROO) in 11 days
  - 10-yearly In-Service Inspection Outage (ISIO) for complete overhaul in 40 days.

These sub-targets imply design requirements laid down in section 6.2.3.1 below. Achieving the outage duration objectives is based on the following requirements:

- Reducing maintenance activities during outages, by carrying out some preventive maintenance during operation (see section 6.2.3 below). Carrying out preventive maintenance on engineered safeguard systems during operation is possible because of the four-train EPR design
- Being able to access the reactor building during operation (see Chapter 12) before the outage for preparation work (polar crane, refuelling machine, etc.) and also after the end of the outage, as well as for certain preventive maintenance operations during the cycle

- Limiting the number of periodic tests during restart and reducing their impact on the restart critical path.

In addition, objectives related to human factors, environmental protection, radiological cleanliness and industrial safety are also taken into consideration when maintenance tasks are defined.

### **6.2.3. Identification of Preventive Maintenance Requirements**

#### **6.2.3.1. Maintenance Strategy**

The EPR maintenance strategy aims at permitting safe operation to maintain public confidence, while achieving economic efficiency by optimising availability and control of maintenance costs subject to the requirements of ALARP.

Design and construction quality (manufacture and assembly) must be high enough to make it possible to limit any differences between the final construction state and the safety requirements defined at the design stage.

The maintenance strategy implemented for the EPR complies with that is already in force in operational French nuclear power plants, but with the difference that the strategy has been incorporated at the design stage.

The Reliability Centred Maintenance (RCM) approach is one of the tools that has made it possible to structure the technical and economic choices which are a foundation of maintenance policy, providing rigour, rationality, efficacy and traceability in maintenance decisions. Reliability data used in RCM studies is obtained where possible from equipment manufacturers and from operating experience feedback.

However, this maintenance strategy is not entirely based on the RCM method. RCM is not applied in the following cases:

- To families of identical equipment where the maintenance strategy justifies a specific preventive maintenance programme e.g. by a sampling method,
- Equipment subject to Main Primary Circuit (RCPB) / Main Secondary Circuit regulations and to regulations applying to Pressurised Nuclear Containments
- Large primary or secondary components, or specific equipment for which the RCM method is not relevant
- Civil engineering structures.

Preventive maintenance programmes essentially concern equipment which is assessed as critical from an RCM point of view, with regards to safety, availability or maintenance. For equipment which is considered to be non-critical, preventive maintenance is limited to minor operations such as upkeep and lubrication essential for smooth running. On such equipment, it is legitimate to wait for a failure to occur before intervening. Naturally, maintenance will also take account of all regulatory requirements.

Technical and economic studies are carried out to optimise the choice between systematic preventive maintenance and condition monitoring, subject to the requirements of ALARP.

In conclusion, the maintenance strategy contributes towards the achievement of availability objectives for the unit by maintaining the required safety level while at the same time controlling maintenance costs, in conformance with the principle of ALARP.

### **6.2.3.2. Unavailability for Preventive Maintenance**

Because the EPR design provides four engineered safeguard trains, some preventive maintenance can be carried while the unit is in operation. This makes it possible to reduce the workload during outages and meet the objectives of outage duration discussed in section 6.2.2.2 of this sub-chapter.

Preventive maintenance operations will be carried out in compliance with OTS so as to comply with safety analysis assumptions.

Unavailabilities due to preventive maintenance while the unit is in operation do not contribute significantly to the global frequency of core meltdown. This is confirmed in PSA studies presented in Sub-chapter 15.1 of the PCSR based on an annual 28-day unavailability period for preventive maintenance of an engineered safeguard train while the unit is in operation.

The 28-day preventive maintenance period assumed in the PSA may be reviewed within the scope of in-depth studies consistent with meeting safety design objectives for the UK EPR.

EVU [CHRS] unavailability while the unit is in operation is necessary for carrying out preventive maintenance on the third PTR [FPCS] train. A 14-day timeframe is supported by PSA studies.

Preventive maintenance on a system during reactor operation is only permitted with the following nine provisions:

- 1) Equipment able to be isolated from the process when the unit is in operation.
- 2) Maintenance activities leading to F1- or F2-classified equipment unavailability must be included in PCC-2 to PCC-4 studies involving these systems. The design rules for PCC-2 to PCC-4 operating conditions require combination of unavailabilities due to the maintenance activities, single failure, and the loss of the external power supply as described in Chapter 14.
- 3) Maintenance activities during operation must not override the radioactive containment requirements stipulated in PCSR Sub-chapter 6.2.
- 4) Unavailabilities due to preventive maintenance while the unit is in operation must not contribute in any significant way to the global frequency of core meltdown. Acceptability of unavailability durations as a result of preventive maintenance is validated via PSA studies.
- 5) Provisions made at the design stage to reduce the risk due to human errors during the maintenance work. Realistic human error scenarios to be defined and included in safety analysis.
- 6) Rules governing allowed combinations of scheduled unavailabilities to be stipulated in the OTS. Scheduled combination scenarios to be identified and approved:
  - Through deterministic analysis (see provision 2), checking in particular those activities which result in exclusions between safety functions and support functions



- If necessary through probabilistic analysis to evaluate the impact on the global core melt frequency.
- 7) Equipment to be accessible and its environment suitable for preventive maintenance during operation (space, logistics etc.), including requirements for equipment overhaul or replacement. The maintenance activity must not lead to a hazard that could result in:
- An event which initiates a transient or accident
  - Damage to equipment in the locality of the maintenance work being carried out.
- 8) Electrical cross-connections installed during the maintenance work not to jeopardise electrical protection selectivity.
- 9) Any equipment on which maintenance has been carried out or whose operability is potentially affected by the work to be requalified to confirm its operability.

In state E<sup>1</sup> of the reactor, preventive maintenance may only be scheduled on one electrical train at a time.

In state F<sup>2</sup> of the reactor, preventive maintenance may be scheduled on two electrical trains simultaneously. However, given the engineered safeguard train design, and the auxiliaries supplied by these safeguard trains, it is not possible to lock out electrical trains 1 and 2 or electrical trains 3 and 4 simultaneously while keeping the two main FPCS trains available. The PSA described in Chapter 15 incorporates these unavailability assumptions with regard to maintenance of the electrical trains during these outage phases.

During the outages, safeguard systems may be inoperable for reasons of preventive maintenance provided that the number of trains required by the OTS is available in the various plant states.

For F2-classified and non-classified systems, preventive maintenance may in general be authorised at any time. However, restrictions may be applied because of constraints related to unit availability or OTS requirements.

### 6.2.3.3. Implementation of System Preventive Maintenance

This section deals with the main characteristics of preventive maintenance on plant systems. For further information, reference should be made to the chapters of the PCSR describing the individual plant systems.

Following an initial examination of the different systems and equipment items, the following main conclusions have been drawn (non-exhaustive list):

- For fuel building systems:
  - PTR [FPCS]: preventive maintenance of the PTR [FPCS] system takes place while the unit is in operation. Preventive maintenance on the PTR [FPCS] is not permitted during outages, in particular during states E and F. In practice, the preventive maintenance will be scheduled when the decay heat level of the spent fuel in the pool is low, i.e. towards the end of cycle. Maintenance work may only be carried out on one train at a time (a main train or the third train)

<sup>1</sup> State E: cold shutdown for reloading

<sup>2</sup> State F: cold shutdown core fully unloaded

- DWK (fuel building ventilation): with regards to fuel building ventilation, preventive maintenance is scheduled while the unit is in operation, outside fuel handling periods when it is required.
- For systems located outside the reactor building, when it is allowed by safety design requirements, preventive maintenance may be carried out while the unit is in operation. Maintenance is carried out on one safety train at a time.
  - RIS [SIS] – Safety injection system and residual heat removal: the four-train RIS/ISMP [SIS/MHSI] and RIS/ISBP [SIS/LHSI] design enables preventive maintenance to be carried out while the unit is in operation, mainly on pumps and heat exchangers.
  - ASG [EFWS] – Auxiliary feedwater system: the ASG [EFWS] four-train design enables preventive maintenance to be carried out while the unit is in operation, limited mainly to pumps.

In the event of an accident, the steam generator (SG) of the train being maintained may be supplied by any other train, which makes it possible to meet the single failure criterion even in the case of isolation of a ruptured SG. Similarly, a passive header on the suction of each ASG [EFWS] pump makes it possible to use the capacity of all the emergency feedwater tanks.

- RRI [CCWS] – Component cooling water system: since maintenance of an RIS [SIS] train is possible while the unit is in operation, making an RRI [CCWS] train inoperable for preventive maintenance reasons is also allowed. Preventive maintenance is carried out in parallel with that on the same train of the RIS [SIS].

The systems which must be permanently cooled by the RRI [CCWS] are supplied by common sections which remain supplied by the available RRI [CCWS] train.

- SEC [ESWS] – Essential service water system: when preventive maintenance is carried out on the SEC [ESWS], the RRI [CCWS] of the same train is inoperable. Its four-train design makes it possible to carry out preventive maintenance on one train. Preventive maintenance is carried out at the same time as on the corresponding train of the RRI [CCWS].
- SRU [UCWS] – Ultimate cooling water system: SRU [UCWS] preventive maintenance is carried out train by train, while the unit is in operation. It may render the EVU [CHRS] and, for train 1, the third PTR [FPCS] train, inoperable. A probabilistic approach is used to justify the preventive maintenance timeframe.
- EVU [CHRS] – Containment heat removal: EVU [CHRS] preventive maintenance is possible whilst the unit is in operation and is carried out in parallel with that on the SRU [UCWS].

- EDG – Emergency diesel generators: maintenance of the diesel generator sets is carried out during power operation. The four-train design makes it possible for one diesel set to be inoperable whilst meeting the deterministic safety requirements. As for the engineered safeguard systems, the preventive maintenance timeframe for diesel generating sets is validated by probabilistic analysis. Power supply is ensured by dedicated interconnections to a neighbouring division during maintenance; for example in the case of maintenance of an emergency diesel generator set. The power supply to the containment isolation valves is also ensured via an interconnection.
- Electrical panels: maintenance and in particular regulatory inspections of the panel under voltage, are possible while the unit is in operation. Maintenance with isolation of a certain number of panels is also possible. The definitive list will be drawn up after in-depth studies. Control and protection panels are not isolated while the unit is in operation, nor are essential functions such as isolation of the primary system, containment isolation, steam generator isolation and steam dump.
- Station Blackout Diesel Generators (SBO-DG) – diesel generators to provide emergency power in the case of loss of off-site power coincident with failure of the four EDGs: preventive maintenance of these generator sets is possible while the unit is in operation if the timeframe is validated by PSA studies.
- DVL [SBVSE] – Safeguard Building Uncontrolled Area Ventilation System (unrestricted radiological zones): each of the four electrical and Instrumentation and Control divisions is ventilated by a safety-classified DVL [SBVSE] train cooled by the electrical building safety chilled water system (DEL [SCWS]) and a back-up safety classified train, called DVLnew, cooled by a new back-up train DELnew. A DVLnew train, common to two electrical trains, ensures ventilation during the preventive phases of maintenance on safety-classified trains. Preventive maintenance on safety-classified DVL [SBVSE] systems may be carried out subject to prescribed maximum external temperature limits (lower than around 25°C) while the unit is in operation by using DVLnew systems without the requirement for DELnew.
- DEL [SCWS] - Electrical building safety chilled water system: since preventive maintenance on the DVL [SBVSE] systems is allowed while the unit is in operation, maintenance is also possible on the DEL [SCWS] which cools it. The design of the other cooling systems cooled by the DEL [SCWS], such as the DCL [CRACS], allows a DEL [SCWS] train to be made inoperable.
 

Preventive maintenance of a DEL [SCWS] train will be carried out at the same time as that on the DVL [SBVSE]. It will be carried out when external temperatures are low so as to limit the consequences in the event of failure by using DVLnew and DELnew trains.
- DCL [CRACS] - Control room air conditioning system: the DCL [CRACS] design makes it possible to carry out preventive maintenance on an air conditioning train whilst the unit is in operation. With regard to train batteries, maintenance will be carried out at the same time as on the DEL [SCWS] in question.

- Containment isolation in specific conditions:

Generally, preventive maintenance whilst the unit is in operation is not possible on the isolation devices themselves because of their process or safeguard function.

In certain conditions, preventive maintenance on an isolation valve may be considered while the unit is in operation if the other valve can be locked closed during the maintenance operation and if the function or part of the function to which it contributes is not required while the unit is in operation either for safety or normal operation of the unit.

To comply with safety and operating requirements, preventive maintenance of numerous systems and equipment items is only permitted during outages. Such is the case for:

- The primary coolant system and its connected lines, and in particular the first isolation valves located at low levels of the primary loops
- Work on the secondary side of the SG and connected steam isolation or steam dumping equipment
- Systems ensuring a negative pressure in the containment annulus
- Maintenance of equipment that is borderline with the tagging 'bubble' necessary for a maintenance action while the unit is in operation, can only be carried out during an outage
- Parts of systems which cannot be requalified while the unit is in operation
- Preventive maintenance involving isolation on the busbars and electrical panel gear, in particular Instrumentation and Control equipment which is unable to be isolated whilst the unit is in operation
- In State E, when the reactor cavity is full, maintenance and inspection work on an electrical train is possible, if the other three trains are available
- In State F, when the reactor is fully unloaded, a second train may be isolated for maintenance and inspection work. However, trains 1 and 2 or trains 3 and 4 should not be isolated simultaneously. Trains 2 and 3 are also not isolated simultaneously for conventional island electrical supply reasons. Interconnection design between trains and auxiliaries supplied by these engineered safeguard trains is such that this requirement will be met in order to maintain permanent availability of the three PTR [FPCS] trains. These assumptions are taken into account in the probabilistic safety analyses.
- SG tube inspections are mainly carried out during state F when the core is totally unloaded.
- Using SG nozzle dams for this works during the fuel handling operations is also possible: the safety justification for use of nozzle dams will be assessed during the detailed design phase.

This list will be gradually added to as in-depth studies are carried out, in line with the conclusions from accident studies.

## 6.2.4. Allowing for Preventive Maintenance at the Design Stage

### 6.2.4.1. Design Requirements

Design requirements have already implemented to meet availability objectives while complying with safety objectives. In particular, these include:

- Design of the nuclear island with four divisions and four engineered safeguard trains giving total segregation of the four mechanical and electrical divisions, except for certain possible electrical interconnections. This divisionalisation facilitates preventive maintenance on electrical panels when the unit is shut down, and also enables maintenance work to be carried out on engineered safeguard trains when the unit is in operation
- Providing an equipment handling and storage area adjacent to the equipment hatch, which forms an extension of the reactor building containment during outage when the hatch is open
- Provision of two PTR [FPCS] trains which ensure cooling of the spent fuel pool. A third additional train may back up this cooling system. Power supply to the three PTR [FPCS] trains is possible via dedicated electrical connections when their electrical switchboards are isolated during outage
- As a result of its high-speed acquisition rate, the internal core instrumentation (aeroball-type measurement system) does not require long periods of core stabilisation during start-up before being able to perform flux mapping
- The SG inlet and outlet channel head design facilitates installation of nozzle dams using robots and fitting of a second section with a loose parts trap
- Carrying out of maintenance at the conventional island level while the unit is in operation is made possible by providing redundancy in the pumping station, in particular with regard to the APA [MFWPS] (motor-driven main feedwater pump system) and CEX (condenser extraction systems)
- By design, the EPR containment is permanently accessible. Conditions for carrying out work in the reactor building are acceptable in terms of temperature, humidity and noise. Access and work zones are classified green from a radiological point of view (see Chapter 12). For unit outages, after previously purging the containment atmosphere, provision has been made to carry out work in the reactor building in the seven days prior to, and the three days following, an outage. During a normal cycle, work inside the building reactor may also be carried out as part of preventive maintenance.

Sub-chapter 18.1 provides a discussion of Human Factors studies to support design for maintainability, and to substantiate Human Based Safety Claims associated with maintenance and testing.

#### 6.2.4.2. Maintenance Programme

Maintenance optimisation (RCM – Reliability Centred Maintenance approach: see section 6.2.3.1 of this sub-chapter) planned at the design phase, has provided a positive contribution to plant safety and availability objectives: by matching maintenance activities to the importance of the risk associated with actual systems and equipment; to the environment by limiting waste after a consistent and optimised maintenance programme; and to radiation protection and human factors through equipment maintainability inbuilt at the design stage.

The aim of the RCM approach is to optimise preventive maintenance programmes on equipment that has been declared safety critical. These maintenance programmes will meet the safety objectives defined at the design stage. With regards to these safety objectives, optimisation consists of enhancing availability whilst meeting maintenance-related constraints.

At the in-depth design study stage, adjustments can be made, mainly with regards to the choice of equipment technology (by making standard exchanges possible for sensitive equipment), on systems (by enhancing maintainability with a view to limiting the time needed for work to be carried out), instrumentation (by enabling monitoring of equipment so as to limit large-scale preventive maintenance operations), and by allowing condition monitoring to be implemented.

Maintenance programmes will be drawn up to achieve:

1) Compatibility of the duration of work carried out with scheduled maintenance timeframes:

- Within the framework of unit outages: Refuelling Only Outage (ROO) in 11 days, Normal Refuelling Outage (NRO) in 16 days, In-Service Inspection Outage (ISIO) in 40 days
- Within the framework of the duration of unavailability for preventive maintenance whilst the unit is in operation, validated by a probabilistic approach of engineered safeguard systems (see section 6.2.3.2 of this sub-chapter).

2) Feasibility with regards to:

- Accessibility of work zones, in particular, accessibility of the reactor building service zone during reactor operation
- Intervention dosimetry and any required decontamination
- Isolation: while the unit is in operation, the equipment on which maintenance work is scheduled to be carried out should be isolable from the adjacent system via suitable isolation devices that can be controlled from the Main Control Room or by local manual action (if such isolating devices are accessible while the unit is in operation)
- Isolation and draining durations
- Requalification after work has been carried out: work carried out while the unit is in operation should be requalifiable in the unit state in question
- System start-up (filling, venting, chemical and thermal treatment).

For each system, studies are performed to differentiate between equipment for which maintenance may be carried out while the unit is in operation and equipment for which maintenance must be carried out during a unit outage. Taking the above considerations into account at the design stage makes it possible to optimise maintenance programmes whilst ensuring that they are in line with the requirements for availability, safety, radiation protection, human factors, environmental protection, contamination control, and personnel safety.

### **6.2.5. Maintenance Schedule**

Licence Condition 28 of the UK HSE Nuclear Site Licence Conditions imposes specific duties on the Licensee to devise and implement adequate arrangements for the regular and systematic examination, inspection, maintenance and testing of all plant which may affect safety. These arrangements are contained within the Maintenance Schedule or equivalent document.

The maintenance schedule would typically be expected to contain the following types of activities, which are discussed below:

- Statutory Testing and Inspection,
- Periodic testing and inspection,
- In-Service Inspection, dealt with in section 6.1 of this sub-chapter,
- Preventive Maintenance, discussed in sections 6.2.2, 6.2.3 and 6.2.4 of this sub-chapter,
- Calibration.

The maintenance schedule presents the main operating parameters of equipment that need to be respected to ensure the correct operation and the absence of degradation of that equipment.

#### **6.2.5.1. Statutory Testing and Inspection**

The Licensee is responsible for developing and implementing arrangements for any testing and inspection that are statutory requirements in the UK.

The role of the EPR designer is to ensure that equipment subjected to statutory testing and inspection requirements is accessible and has been designed in order for these statutory requirements to be carried out.

#### **6.2.5.2. Periodic Testing and Inspection**

The definition of periodic testing requirements is discussed in section 5 of this sub-chapter. A future Licensee will be provided with exhaustive analysis reports for each system. These reports define the testing required to demonstrate the ability of SSCs to fulfil the requirements of a safety feature. Some testing is achieved through periodic tests, some through maintenance tests (for example a sensor calibration) and some through normal operation.

The exhaustive analysis will cover not only functional type testing such as flow-rate measurements, but also equipment requirements such as instrumentation. Testing of the functional and equipment components of the SSCs that contribute to a safety feature allow the availability of the safety feature to be confirmed.

This ability of SSCs to fulfil the requirements of a safety feature underpins the assumptions made in the safety case relating to fault mitigation and duty functions.

#### **6.2.5.3. Calibration**

Where instrumentation contributes to a safety feature, the periodic testing exhaustiveness analysis provided to the Licensee will define maintenance and calibration requirements.

#### **6.2.5.4. Corrective Maintenance**

The 4-train EPR design aims to optimise maintenance carried out at power. The principal consideration for the maintenance at power is that OTS requirements are met for the operability of safety-related systems and their associated support systems.

Maintenance of equipment at power must satisfy the following:

- Maintenance requirements of the system and associated operability requirements.
- Top level system operability requirements and the requirements for associated support systems.

In principle, and in accordance with the Preliminary OTS, an electrical division can be shut down for maintenance in outage states.

If necessary, corrective maintenance actions can be performed on equipment to achieve its expected performance, operational, availability, flexibility and to meet the safety requirements on the equipment (e.g. SG tube plugging). Corrective maintenance instructions are provided with detailed recommendations, spare parts and special tools lists, required to achieve operations under safe conditions. These details are provided by equipment manufacturers as part of equipment supply contracts. The information is provided to the Licensee in order that detailed corrective maintenance instructions can be developed. The supply of this information from equipment manufacturer to designer and then to Licensee, is subject to contract conditions which potentially result in the amount and type of information supplied differing from one Licensee to another.

#### **6.2.6. GDA/Licensee Maintenance Documentation Boundary**

The output of the generic design assessment for maintenance is expected to be:

- A design of UK EPR that can be maintained in terms of equipment accessibility for maintenance; redundancy of equipment to allow release of plant for maintenance where this is permitted in the design.

During the NSL process, recommendations will be provided by the designer to the Licensee on the examination, maintenance, inspection and testing (EMIT) activities that are required to underpin safety case assumptions on equipment performance and robustness

The Licensee will be responsible for putting in place adequate and appropriate arrangements for the development, implementation, management and update of the full EMIT programme.

The actual PT and maintenance intervals underpinning EDF derived component failure probabilities are out of scope for GDA, but will be specified during NSL. These will need to be included in the EMIT programme or alternatives justified for an operational UK EPR.



## SUB-CHAPTER 18.2 – REFERENCES

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

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### 1. INTRODUCTION

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