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General Nuclear System Ltd.

UK HPR1000 GDA Project

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Pre-Construction Safety Report

Chapter 12

Design Basis Condition Analysis

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12.1 List of Abbreviations and Acronyms

AAD	Startup and Shutdown Feedwater system [SSFS]
ALARP	As Low As Reasonably Practicable
APG	Steam Generator Blowdown System
ARE	Main Feedwater System [MFFCS]
ASG	Emergency Feedwater System [EFWS]
BOC	Beginning of Cycle
DBA	Design Basis Analysis
DBC	Design Basis Condition
DEC-A	Design Extension Condition A
DN	Nominal Diameter
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DWK	Fuel Building Ventilation System [FBVS]
DWL	Safeguard Building Controlled Area Ventilation System [SBCAVS]
EBA	Containment Sweeping and Blowndown Ventilation System [CSBVS]
ECCS	Emergency Core Cooling System
EDE	Annulus Ventilation System [AVS]
EDG	Emergency Diesel Generator
EOC	End of Cycle
FC1	Safety Category 1 Function
FC2	Safety Category 2 Function
FCG3	Fangchenggang Nuclear Power Plant Unit 3
FMEA	Failure Modes and Effects Analysis
FP	Full Power
GCT	Turbine Bypass System [TBS]

GDA	Generic Design Assessment
HP	High Pressure
HPR1000	Hua-long Pressurized Reactor
HPR1000 (FCG3)	Hua-long Pressurized Reactor under construction at Fangchenggang nuclear power plant unit 3
I&C	Instrumentation and Control
IRWST	In-containment Refuelling Water Storage Tank
KRT	Plant Radiation Monitoring System [PRMS]
LB-LOCA	Large Break(Loss of Coolant Accident)
LCO	Limiting Condition of Operation
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-Site Power
MCS	Maintenance Cold Shutdown
MHSI	Medium Head Safety Injection
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
NC	Non-Classified
NR	Narrow Range
NS/RIS-RHR	Normal Shutdown with RIS-RHR
NSSS	Nuclear Steam Supply System
NS/SG	Normal Shutdown with Steam Generators
OTS	Operating Technical Specification
PCSR	Pre-Construction Safety Report
PIE	Postulated Initiating Events
PSA	Probabilistic Safety Assessment
PTR	Fuel Pool Cooling and Treatment System [FPCTS]

PZR	Pressuriser
RBS	Emergency Boration System [EBS]
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCP	Reactor Coolant System [RCS]
RCPB	Reactor Coolant Pressure Boundary
RCV	Chemical and Volume Control System [CVCS]
RCD	Reactor Completely Discharge
REA	Reactor Boron and Water Makeup System [RBWMS]
REN	Nuclear Sampling System [NSS]
RHR	Residual Heat Removal
RIS	Safety Injection System [SIS]
RRI	Component Cooling Water System [CCWS]
RT	Reactor Trip
SADV	Severe Accident Dedicated Valve
SF	Safety functions
SFC	Single Failure Criterion
SG	Steam Generator
SGa	Affected Steam Generator
SGTR	Steam Generator Tube Rupture
SSCs	Structures, Systems and Components
SWI	Seal Water Injection
UK HPR1000	The UK version of the Hua-long Pressurized Reactor
VDA	Atmospheric Steam Dump System [ASDS]

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Emergency Feedwater System (ASG [EFWS]).

12.2 Introduction

12.2.1 Objective

The performance of the UK version of the Hua-long Pressurised Reactor (UK HPR1000) under Postulated Initiating Events (PIE), is studied and demonstrated by the design basis condition analysis presented in this Chapter. This is a central source of evidence to support the fundamental objective of the PCSR and relevant PCSR claims (see Sub-chapter 1.6) listed as following:

a) ***Claim 3:*** *The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is as low as reasonably practicable;*

b) ***Claim 3.2****: A comprehensive fault and hazard analysis has been used to specify the requirements on the safety measures;*

c) ***Claim 3.4:*** *The safety assessment shows that the nuclear safety risks are ALARP;*

The plant with its design basis safety systems can respond to any fault with an initiating event frequency > 1E-5/year and achieve a controlled and safe state with a tolerable radiation exposure or release of radioactive material.

All postulated initiating events or fault sequences of the UK HPR1000 are classified into four design basis conditions (DBC-1 to DBC-4) and two Design Extension Conditions (DEC-A and DEC-B). The analysis of design basis conditions is presented in Chapter 12, and that of Design Extension Conditions in Chapter 13.

12.2.2 Scope

12.2.2.1 Sub-Claims and Arguments

The purpose of this chapter is to demonstrate the high-level claim using the following sub-claims and arguments:

a) Sub-Claim 1 (Claim 3.2.1): All initiating events with the potential to lead to significant radiation exposure or release of radioactive material, including the effects of internal and external hazards have been identified.

    1) Argument 1.1: A systematic process has been defined to identify all PIE that could lead to a release of radioactive material. (See Sub-chapter 12.4 - Fault Identification and Fault Grouping)

    2) Argument 1.2: The systematic PIE identification methodology has been applied to produce the PIE list which is a common input for both DBA and PSA grouping process. (See Sub-chapter 12.4 - Fault Identification and Fault Grouping)

3) Argument 1.3: All operating states have been considered in the identification of PIE. (See Sub-chapter 12.4 - Fault Identification and Fault Grouping)

4) Argument 1.4: All potential sources of radioactivity have been assessed in the derivation of the PIEs. (See Sub-chapter 12.4 - Fault Identification and Fault Grouping)

5) Argument 1.5: The frequency of each PIE has been derived in a systematic manner. (See Sub-chapter 12.4 - Fault Identification and Fault Grouping)

6) Argument 1.6: The PIEs have been grouped to produce initiating events for detailed assessment. (See Sub-chapter 12.4 - Fault Identification and Fault Grouping)

b) Sub-Claim 2 (supporting Claim 3.2.2: Design basis events have been appropriately assessed to specify requirements on safety functions and on safety measures and assess their effectiveness.): The appropriate analysis approach has been applied for each fault;

1) Argument 2.1: Appropriate acceptance criteria have been identified that, if met, ensure the delivery of the three essential safety functions following any initiating event.

- Acceptance criteria have been defined that demonstrate the continuing delivery of the reactivity control essential safety function (see Chapters 5, 6, 7 and Sub-chapter 12.5 - DBC Accident Analysis Rules);
- Acceptance criteria have been defined that demonstrate the continuing delivery of the heat removal essential safety function (see Chapters 5, 6, 7 and Sub-chapter 12.5 - DBC Accident Analysis Rules);
- Acceptance criteria have been defined that demonstrate the continuing delivery of the confinement essential safety function (see Chapters 5, 6, 7, 16 and Sub-chapter 12.5 - DBC Accident Analysis Rules).
- The conditions arising from the identified initiating events will not result in the consequential failure of other safety related equipment (see Chapter 19).

2) Argument 2.2: Appropriate analysis methods have been identified, verified and validated to assess the consequences of the design basis initiating events identified in the fault schedule.

- A review of potential analysis codes has identified a set of suitable analysis codes to perform the fault analysis for the UK HPR1000; (see Appendix A. Computer Codes Used in the Fault Studies and Radiological Release Analysis)

- Each of the codes identified for use in the fault analysis has been verified to confirm the correctness of the coding; (see Appendix A Computer Codes Used in the Fault Studies and Radiological Release Analysis)

- For each fault analysis code, a detailed validation matrix has been made showing all the validation cases and the related validated models or phenomena; (see Appendix A Computer Codes Used in the Fault Studies and Radiological Release Analysis)

- Appropriate validation results have been provided against the requirements of the validation matrix to justify the use of the fault analysis codes and related analysis method; (see Appendix A Computer Codes Used in the Fault Studies and Radiological Release Analysis)

- Appropriate guidance has been provided to the analysts to undertake the fault analysis codes. (see Appendix A Computer Codes Used in the Fault Studies and Radiological Release Analysis)

3) Argument 2.3: Appropriate analysis rules have been applied, including conservative initial conditions, plant operation parameters, operator actions and the worst single failure, in the analysis of design basis faults.

   - Conservative plant initial conditions have been defined for the DBA (see reference report for each DBC event analysis);

   - Performance requirements for the safety systems are defined for use in the fault analysis and the design of the safety systems (see reference report for each DBC event analysis);

   - A detailed assessment of the progression of each fault is performed to identify the worst single failure (see reference report for each DBC event analysis).

c) Sub-Claim 3 (supporting Claim 3.4.3: Design Basis Analysis demonstrates that the design has suitable safety measures to ensure that it is tolerant to faults, meeting the acceptance criteria for design basis faults.): Depending on the frequency of the initiating fault, an appropriate level of diversity, redundancy, and reliability of the individual SSCs is provided to achieve safety measures for a robust fault-tolerant plant, and identified in a comprehensive fault schedule.

   1) Argument 3.1: An appropriate methodology has been developed to produce the fault schedule (See Sub-chapter 12.12);

   2) Argument 3.2: The fault schedule methodology has defined the process for identifying SSCs of the required classification for each of the safety measures (See Sub-chapter 12.12);

   3) Argument 3.3: The safety measures required to protect against the fault

following each initiating event have been identified (See Sub-chapter 12.12).

d) Sub-Claim 4 (supporting Claim 3.4.5: The risk to workers and members of the public from the potential harmful effects of ionising radiation resulting from fault and accident conditions complies with UK legal requirements and is ALARP.): Analysis results demonstrated that the UK HPR1000 is able to achieve the controlled state and the safe state after any identified initial events without leading to an off-site release that would require significant off-site counter-measures (such as evacuation) or to an excessive dose to workers

1) Argument 4.1: The plant can be led to the controlled state and the safe state using the safety systems identified in the fault schedule without violation of the identified acceptance criteria. The achievement of safe state may require operator actions by the operator to achieve it. (See Sub-chapters 12.7, 12.8, 12.9 and 12.10)

- The fault analysis shows that the plant can be transferred to the controlled state under all faults in the fault schedule using only Class 1 or Class 2 safety systems without violating the defined acceptance criteria;

- The operator actions required to support the transition of the plant from the controlled state to the safe state have been identified;

- The systems required to support the transition from the controlled state to the long term safe state have been identified;

- The operator actions required to support the transition of the plant from the controlled state to the safe state have been appropriately justified.

2) Argument 4.2: The radiological release and exposure or direct radiation exposure following design basis events meets the acceptance criteria. (See Sub-chapter 12.11)

- The consequences for the public under design basis faults have been evaluated and show that the required success criteria have been met (Generic site radiological consequence analysis results);

- The consequences for workers under design basis faults have been evaluated and show that the required acceptance criteria have been met (Generic site worker dose assessment).

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12.2.2.2 Structure of This Chapter

This chapter includes 15 subchapters and an appendix:

a) Sub-chapter 12.1 gives a list of abbreviations and acronyms.

b) Sub-chapter 12.2 gives the introduction.

c) Sub-chapter 12.3 presents the applicable codes and standard

d) Sub-chapter 12.4 contains the methodologies and results of fault identification and fault grouping.

e) Sub-chapter 12.5 addresses the analysis rules applied in DBA.

f) Sub-chapter 12.6 addresses the plant characteristics taken into account during analysis.

g) Sub-chapter 12.7 contains the analysis of DBC-2 events.

h) Sub-chapter 12.8 contains the analysis of DBC-3 events.

i) Sub-chapter 12.9 contains the analysis of DBC-4 events.

j) Sub-chapter 12.10 provides a summary of the analysis from controlled state to safe state.

k) Sub-chapter 12.11 contains the radiological consequences of the DBA, which is performed in a way similar to the HPR1000 (FCG 3) DBA.

l) Sub-chapter 12.12 describes the design methodology of the Fault Schedule.

m) Sub-chapter 12.13 addresses the role of DBC analysis in the whole ALARP evaluation.

n) Sub-chapter 12.14 presents the concluding remarks.

o) Sub-chapter 12.15 provides the references.

p) Appendix A is the introduction of computer codes used in the fault studies and radiological release analysis.

12.2.2.3 Interfaces with Other Parts of the PCSR

The interfaces to other parts of the PCSR are as below:

T-12.2-1 Interfaces to Other Parts of the PCSR

PCSR Chapter	Interface
Chapter 1 Introduction	Chapter 1 provides the fundamental objective, level 1 claims and level 2 claims. Chapter 12 provides chapter claims, arguments to support relevant claims that are addressed in Chapter 1.
Chapter 4 General Safety and Design Principles	Chapter 4 addresses the methods adopted for the assessment and substantiation of the plant systems
Chapter 5 Reactor Core	Chapter 5 provides the acceptance criteria related to core and fuel under accidents
Chapters 6\7\8\9\10\11 Reactor Coolant System\Safety Systems\Instrumentation and Control\Electric Power\Auxiliary Systems\Steam and Power Conversion System	Chapters 6\7\8\9\10\11 provide the substantiation of the Reactor Coolant System, Safety Systems, Instrumentation & Controls, Electric Power, Auxiliary Systems and Steam & Power Conversion System which are taken into consideration of fault analysis
Chapter 13 Design Extension Conditions and Severe Accident Analysis	Chapter 13 provide the Design Extension Condition A (DEC-A) analyses which will support Fault Schedule.
Chapter 14 Probabilistic Safety Assessment	Chapter 14 provides PSA results related to the Fault Identification and Fault Grouping
Chapter 15 Human Factors	Chapter 15 substantiate the claims on operator actions under DBC conditions.
Chapters 18\19 External Hazards\ Internal Hazards	Chapters 18\19 address whether there is any fault need to be considered in DBC Accident Analysis
Chapter 20 MSQA and Safety Case Management	The organisational arrangements and quality assurance arrangements set out in Chapter 20 are implemented in the design process and production of all PCSR
Chapter 21 Reactor Chemistry	Chapter 12 and its supporting documents contain description of chemical effects that related to Design Basis Condition (DBC) source term analysis, including fission product control and iodine retention and transport.
Chapter 22 Radiological Protection	Chapter 22 provides input for radiological consequence analysis under DBC accidents
Chapter 23 Radioactive Waste Management	Chapter 12 provides the Design Basis Condition (DBC) analysis related to radioactive waste management systems Chapter 23 provides the specific design of radioactive waste

PCSR Chapter	Interface
	management systems, which in taken into consideration of fault analysis.
Chapter 28 Fuel Route and Storage	Chapter 28 demonstrates that fuel handling and storage system have been substantiated. Analysis related to Fuel Route and Storage fault in Chapter 12 supports the As Low As Reasonably Practicable (ALARP) assessment of fuel handling and storage related operations in Chapter 28.
Chapter 31 Operational Management	Chapter 31 provides the arrangement of operating limits and conditions consistent with the safety analysis
Chapter 33 ALARP Evaluation	Chapter 33 provides the overview of ALARP assessment. Chapter 12 provides the input for risk assessment to support the ALARP assessment in ALARP evaluation.

12.3 Applicable Codes and Standards

For design basis conditions analysis, IAEA codes and standards are regarded as widely accepted RGP around the world. These codes and standards are also followed in the Chinese domestic approach. The following codes and standards will be considered in this chapter (Reference [1-9]):

[1] IAEA, Format and Content of the Safety Analysis Report for Nuclear Power Plants Safety Guide, Series No. GS-G-4.1, April 27, 2004.

[2] IAEA, Safety Assessment for Facilities and Activities General Safety Requirements Part 4, Series No. GSR Part 4, February, 2016.

[3] IAEA, Design of Reactor Containment Systems for Nuclear Power Plants Safety Guide, Series No. NS-G-1.10, September 22, 2004.

[4] IAEA, Design of the Reactor Core for Nuclear Power Plants Safety Guide, Series No. NS-G-1.12, May 26, 2005.

[5] IAEA, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety Guide, Series No. NS-G-1.9, September 23, 2004.

[6] IAEA, Storage of Spent Nuclear Fuel, Series No. SSG-15, March 27, 2012.

[7] IAEA, Deterministic Safety Analysis for Nuclear Power Plants Specific Safety Guide, Series No. SSG-2, January 05, 2010.

[8] IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, Series No. SSG-30, May 22, 2014.

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

12.4 Fault Identification and Fault Grouping

12.4.1 Scope of Fault Identification

A Postulated Initiating Event (PIE) is an event that has the potential to lead to anticipated operational occurrences or accident conditions. The PIEs considered for the UK HPR1000 will include all foreseeable failures of Structures, Systems and Components (SSCs) of the plant, as well as operating errors and possible failures arising from internal and external hazards, whether in full power, low power or shutdown states, in the reactor, the fuel pool or any other activity containing sources of radioactivity.

The PIEs are selected if they could lead to a potential risk to the fundamental safety functions:

a) Control of reactivity;

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

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

For the UK HPR1000, the PIE list will be reviewed and developed to ensure that it considers all potential sources of activity and all risks to the public and the environment are appropriately considered in the design process and safety demonstration.

12.4.2 Fault Identification and Fault Grouping Methodology

The PIEs can be identified by the methods presented below.

a) Use of analytical methods such as hazard and operability analysis (HAZOP), failure mode, effect analysis (FMEA), and master logic diagrams;

b) Comparison with the list of PIEs developed for safety analysis of similar plants (although this method should not be exclusively used since prior mistakes could also be transferred);

c) Analysis of operating experience data for similar plants.

Due to the limitation of each method mentioned above, several methods were adopted in the identification of PIEs to assure the completeness of events analysis.

The identification of PIEs for UK HPR1000 is a combination of several methods which is initially identified by using master logic diagram analysis and FMEA. A number of PIEs are identified by the above two methods, but it is not necessary to analyse all PIEs in event analysis. PIEs are grouped according to the similarity of the plant (including operator) response, accident mitigation measures, allowable response time, etc. The selection of bounding initiating events is further undertaken for each PIEs group. Thus, the grouping process will also be performed for PSA.

Following the grouping process, a DBC list will be obtained. Then the DBC list will be compared with the list from similar plants or standards to ensure completeness

The detailed PIE identification method and process are described in reference [10].

A categorisation system groups PIE into four categories according to their anticipated frequency of occurrence and potential radiological consequences to the public. Acceptance criteria are defined for each category of which the events must meet. The four categories are as follows:

a) DBC-1: Normal operation

Operation within specified operational limits and conditions.

b) DBC-2: Anticipated operational occurrences

An operational process deviating from normal operation which is likely to occur at least once during the operating lifetime of a single unit facility but which, because of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

c) DBC-3: Design Basis Condition category 3

Conditions that may occur once during the lifetime of a fleet of operating plants and may result in the failure of a small fraction of the fuel rods but do not generate a Design Basis Category 4 Condition or result in the consequential loss of function of the Reactor Coolant System (RCP [RCS]) or Containment System.

d) DBC-4: Design Basis Condition category 4

Conditions which are not expected to occur but are postulated because their consequences could include the potential release of significant amounts of radioactive material; they are the most extreme conditions which must be considered in the design and they represent limiting cases.

Within each categorisation group, representative events are identified for detailed assessment and these form the fault list for the detailed analysis undertaken for the UK HPR1000.

12.4.3 Fault List

The identification of PIEs for the UK HPR1000 is still ongoing. The fault list in this document version refers to the Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)). It is acknowledged that there are still some gaps compared to the requirements of UK context such as the PIEs within fuel route or faults that only result in radiation exposure.

The list of PIEs will be completed systematically during the GDA with consideration made to previous GDA project experience.

12.4.3.1 Safety Analysis States

Events postulated in the safety analysis are supposed to occur during normal plant operating states. The initiating conditions assumed in safety analyses cover all the possible standard conditions from full power operation to cold shutdown. The definitions of the safety analysis domains for the UK HPR1000 are provided below.

a) State A: power states, hot and intermediate shutdown states.

In these states, all the necessary automatic reactor protection functions are available in power states. Some protection functions might be deactivated at low power, but there are always enough automatic protection functions to meet the acceptance criteria in case of a transient condition.

b) State B: Intermediate shutdown with temperature above 140℃.

When the temperature is above 140℃, in normal operation, the Safety Injection System (RIS [SIS]) is not connected in residual heat removal (RHR) mode to the RCP [RCS]. It should be noted that when the temperature reaches 180℃, the RIS [SIS] in RHR mode can be connected with the RCP [RCS] as needed. In this state, some automatic reactor protection functions available in state A may be deactivated.

c) State C: Intermediate shutdown and cold shutdown conditions when RIS [SIS] is under RHR operation mode.

In this state, the RCP [RCS] is closed or can be closed quickly (e.g., when the ventilation pipe is open) so that the SGs can be used for core heat removal if needed.

d) State D: cold shutdown with RCP [RCS] open.

Due to the open status of the RCP [RCS], the SGs cannot be used for core decay heat removal.

e) State E: cold shutdown during refuelling.

f) State F: cold shutdown with the fuel fully unloaded.

During this state, works are undertaken on RCP [RCS] components. This state does not have to be analysed with regard to core protection.

The concrete safety analysis state breakdown is presented in T-12.4-1.

The list of DBC events for the UK HPR1000 is provided in the subsections below.

12.4.3.2 DBC-1: Normal Operation

For the UK HPR1000, The DBC-1 list for the UK HPR1000 is as follows:

a) All steady-state operation, startup and shutdown processes permitted by the nuclear power plant technical specifications during:

1) Power operation;

2) Hot standby;

3) Hot shutdown;

4) Cold shutdown;

5) Reactor refuelling;

6) Reactor start-up and load increase;

7) Reactor power-reducing and shutdown processes.

b) Permitted operation with the temporary deviation in plant parameters or equipment unavailability (or defects) permitted by the power plant technical specifications:

1) Within shutdown equipment or systems;

2) Fuel clad defect;

3) Steam Generator (SG) tube leakage;

4) Reactor coolant radioactive substance (fission products, corrosion products and tritium) concentration increases;

5) Tests permitted by the technical specifications.

c) Operating transient:

1) Change in reactor coolant temperature within the rate specified by the technical specifications (excluding normal startup and shutdown);

2) Load ramp within the rates specified by the Technical Specifications;

3) Step change of load within the magnitude specified by the Technical Specifications;

4) Load shedding (including full load shed to auxiliary power load).

12.4.3.3 DBC-2: Anticipated Operating Occurrences

The DBC-2 list (see Reference [11]) for UK HPR1000 is provided in T-12.4-2.

12.4.3.4 DBC-3: Design Basis Condition category 3

The DBC-3 list (see Reference [11]) for UK HPR1000 is provided in T-12.4-3.

12.4.3.5 DBC-4: Design Basis Condition category 4

The DBC-4 list (see Reference [11]) for UK HPR1000 is provided in T-12.4-4.

T-12.4-1 Definition of Safety Analysis Domains

Operating mode	Safety analysis domain		Standard state	RCP [RCS] state	Primary coolant inventory	RCPs in operation	RCP [RCS] average temperature, ℃	RCP [RCS] Pressure, MPa	RCP boron concentration, ppm	Power
Reactor in Power (RP)	A		Reactor in Power	Closed	Full	3	295≤T≤307	15.5	Critical CB	2%Pn ~ 100%Pn
			Hot Standby			3	295	15.5	Critical CB	< 2%Pn
Normal Shutdown with SGs (NS/SG)			Hot Shutdown			3	295	15.5	≥CB of Hot Shutdown	0
			Intermediate Shutdown with SG PRCP≥13 MPa abs			3	T<295	13≤P≤15.5	≥CB of Cold Shutdown	0
	B	B1	Intermediate Shutdown with SGs PRCP<13 MPa abs	Closed	Full	3	T<295	P<13 7≤P_{HL}	≥CB of Cold Shutdown	0
		B2					135≤T≤245 If 135≤T≤140	P_{HL}<7 3.2<P		
			Intermediate Shutdown with SGs at RIS in RHR Mode Conditions of Connection			3	135≤T≤140	2.4≤P≤3.2	≥CB of Cold Shutdown	0
Normal Shutdown with RIS-RHR (NS/RIS-RHR)	C	C1	Intermediate Shutdown with RIS-RHR	Closed	Full	≥1	100≤T≤140	2.4≤P≤3.2	≥CB of Cold Shutdown	0
		C2				≥1	10≤T≤100		≥CB of Cold Shutdown	
						≥0	10≤T≤60	P≤3.2	≥CB of refuelling phase	
		C3a	Normal Cold Shutdown (RCP [RCS] pressurisable)	Non-Closed RCP [RCS] pressurisable	≥3/4 loop level	0	10≤T≤60	P≤3.2	≥CB of refuelling phase	0
Maintenance Cold Shutdown (MCS)	D	C3b	Normal Cold Shutdown for Maintenance (RCP [RCS] not pressurisable)	Non-Closed RCP [RCS] not pressurisable Reactor cavity not fillable	≥3/4 loop level	0	10≤T≤60	Atmospheric	≥CB of refuelling phase	0
Refuelling Cold Shutdown (RCS)	E		Normal Cold Shutdown for Refuelling	Non-Closed RCP [RCS] not pressurisable Reactor cavity fillable	Reactor Cavity Flooded	0	10≤T≤60	Atmospheric	≥CB of refuelling phase	0
Reactor Completely Discharge (RCD)	F		Core Totally Unloaded	---	---	---	---	---	---	---

T-12.4-2 DBC-2 Events Considered in the UK HPR1000

No.	Event description
1	Feedwater system malfunctions causing a reduction in feedwater temperature (State A\B)
2	Feedwater system malfunctions causing an increase in feedwater flow (State A\B)
3	Excessive increase in secondary steam flow (State A\B)
4	Turbine trip (State A)
5	Loss of condenser vacuum (State A)
6	Short term loss of off-site power (< 2 hours) (State A)
7	Loss of normal feedwater flow (loss of all main feedwater pumps and Startup and Shutdown Feedwater System (AAD [SSFS]) pumps) (State A)
8	Loss of one cooling train of the Safety Injection System (RIS [SIS]) in Residual Heat Removal (RHR) mode (State C\D)
9	Partial loss of core coolant flow (loss of one main coolant pump) (State A)
10	Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at a subcritical or low power startup condition (State A)
11	RCCA bank withdrawal at power (State A)
12	RCCA misalignment up to rod drop without limitation (State A)
13	Startup of an inactive reactor coolant loop at an improper temperature (State A)
14	Chemical and Volume Control System (RCV [CVCS]) malfunction that results in a decrease in boron concentration in the reactor coolant (State A to E)
15	Spurious reactor trip (State A)
16	RCV [CVCS] malfunction causing an increase in (RCP [RCS]) inventory (State A)
17	RCV [CVCS] malfunction causing a decrease in (RCP [RCS]) inventory (State A)
18	Uncontrolled RCP [RCS] level drop in shutdown states with RIS [SIS] connected in RHR mode (State C\D)

No.	Event description
19	Spurious pressuriser heater operation (State A)
20	Spurious pressuriser spray operation (State A)
21	Loss of one train of the Fuel Pool Cooling and Treatment System (PTR [FPCTS]) or of a supporting system (State A)

T-12.4-3 DBC-3 Events Considered in the UK HPR1000

No.	Event description
1	Inadvertent opening of an SG relief train or of a safety valve (State A)
2	Small steam system piping break including breaks in connecting lines (State A\B)
3	Inadvertent closure of all or one main steam isolation valves (State A)
4	Long term LOOP (> 2 hours) (State A)
5	Small feedwater system piping break including breaks in connecting lines to SG (State A\B)
6	Forced reduction in reactor coolant flow (3 pumps) (State A)
7	Inadvertent loading of a fuel assembly in an improper position (State E)
8	Uncontrolled RCCA bank withdrawal (State B\C\D)
9	Uncontrolled single RCCA withdrawal (State A)
10	SG Tube Rupture (SGTR) (one tube) (State A)
11	Inadvertent opening of a pressuriser safety valve (State A)
12	Rupture of a line carrying primary coolant outside containment (e.g. nuclear sampling line) (State A)
13	Small Break (Loss of Coolant Accident) (SB-LOCA) (at power) including a break in the Emergency Boration System (RBS [EBS]) injection line (State A)

No.	Event description
14	Small break LOCA (at shutdown, RIS [SIS] not connected in RHR mode) including a break in the RBS [EBS] injection line (State A\B)
15	Gaseous waste tank break (State A to F)
16	Liquid waste effluent tank break (State A to F)
17	Volume control tank break (State A to F)
18	LOOP (>2 hours) affecting fuel pool cooling (State A)
19	Loss of one train of the PTR [FPCTS] or of a supporting system (with the reactor core offloaded to the fuel pool) (State F)
20	Isolatable piping failure on a system connected to the spent fuel pool (State A to F)

T-12.4-4 DBC-4 Events Considered in the UK HPR1000

No.	Event description
1	Large steam system piping break (State A\B)
2	Inadvertent opening of an SG relief or safety valve (State B)
3	Large feedwater system piping break (State A\B)
4	Long term LOOP (State C)
5	Reactor coolant pump seizure (locked rotor) or Reactor coolant pump shaft break (State A)
6	Spectrum of RCCA ejection accidents (State A)
7	Boron dilution due to a non-isolatable rupture of a heat exchanger tube (State C\D\E)
8	SGTR (two tubes in one SG) (State A)
9	Large Break (Loss of Coolant Accident) (LB-LOCA) (up to double-ended break) (State A)
10	Intermediate Loss of Coolant Accident (LOCA) (State A\B)

No.	Event description
11	SB-LOCA, including a break in the emergency boration system injection line (State C\D)
12	RHR system piping break inside (outside) containment (≤DN 250) (State C\D)
13	Inadvertent opening of the dedicated depressurisation device (State A\B)
14	Fuel handling accident (State A to F)
15	Spent fuel transport cask drop (State A to F)
16	Failure of radioactivity containing equipment in nuclear auxiliary building (State A to F)
17	Non-isolatable small break or isolatable RIS [SIS] break (≤DN 250) in RHR mode affecting fuel pool cooling (during refuelling) (State E)

DN: Nominal Diameter

12.5 DBC Accident Analysis Rules

This sub-chapter describes the rules which will be followed for the DBC accident analysis. Sub-chapters 12.5.1 to 12.5.8 are the general requirements except for faults related to the fuel storage pool. Sub-chapter 12.5.9 presents the specific rules for events associated with the fuel storage pool.

12.5.1 Acceptance Criteria

Acceptance criteria are assigned to each DBC accident or incident (or family of accidents or incidents). Compliance with these acceptance criteria ensures that the safety objectives relevant to the DBC accident or incident are met.

The acceptance criteria include safety criteria and decoupling criteria.

a) Safety criteria

Safety criteria are defined in terms of radiological limits. They have to be met in the safety analysis.

In detail, the radiological consequences will be evaluated against Numerical Target 4.

Data and assumptions used for radiological calculations are described in the section related to radiological consequences (see Sub-chapter 12.11).

b) Decoupling criteria

In addition to safety criteria, it is convenient for practical purposes to introduce some decoupling criteria, which can be applied to the thermal hydraulic and neutronic calculations. In this way, the thermal hydraulic and neutronic calculations, and the radiological calculations, can be decoupled and carried out separately.

Decoupling criteria are defined in such a way that meeting them guaranties that safety criteria, i.e. radiological limits, are also met.

Decoupling criteria must be met under application of all “DBC analysis rules”.

The following decoupling criteria are used in the DBC-2 analyses:

Fuel integrity shall be ensured (no Departure from Nucleate Boiling (DNB) and no fuel melting).

1) The Departure from Nucleate Boiling Ratio (DNBR) shall be greater than the design limit which is described in Sub-section 5.6.2.1;

2) The fuel temperature shall be lower than the fuel melting temperature 2590℃.

The following decoupling criteria are used in the DBC-3 and DBC-4 analyses:

1) For the DBC-3 and DBC-4 events (LOCA excluded), the amount of fuel rods

experiencing DNB must remain lower than 10% (DBC-3 and DBC-4).

2) In the DBC-3 and DBC-4 analyses, the fuel pellet melting at the hot spot must not exceed 10% (DBC-3 and DBC-4 excluding LOCA) by volume, i.e. considering a cross section of the hottest fuel rod at the elevation of the power peak, less than 10% (DBC-3 and DBC-4) of this area is allowed to reach the melting temperature. However, for the analysis of some DBC-3 and DBC-4 faults, the criteria of no DNB and no fuel melting are still applied;

3) For cases not involving the rapid transient of oxidation of the cladding, the peak cladding temperature must remain lower than 1482℃;

4) Moreover, the decoupling criteria for a rod ejection accident are:

   - The enthalpy of fuel pellet must be less than design limit (942 J/g for non-irradiated fuel and 837 J/g for irradiated fuel).

5) The decoupling criteria for LOCA events:

   - The peak cladding temperature must remain lower than 1204℃;
   - The maximum cladding oxidation must remain lower than 17% of the cladding thickness;
   - The maximum hydrogen generation must remain lower than 1% of the amount that would be generated if the whole active part of the cladding had reacted;
   - The ability to cool the core shall be maintained, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling;
   - The long-term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed.

6) Under the safe state, the plant shall be maintained in a sub-critical state.

In addition, protection against primary and secondary system overpressures is discussed in Sub-chapter 6.6.

For the DBC events occurring in a cold shutdown state, the initial state of the various barriers can be different from their state at power. For example, the reactor containment or the RCP [RCS] can be open. The decoupling criteria for these events are adapted accordingly.

12.5.2 Safe Conditions

The safety analysis must be performed up to a safe condition. Two states are defined: the controlled state and the safe state.

For each DBC-2/3/4 event, it shall be demonstrated that the controlled state can be reached. The analysis of the transition from controlled state to safe state can be performed by grouping similar transient causes and PIEs.

a) Controlled state

Plant state, following an anticipated operational occurrence or accident conditions, in which the fundamental safety functions can be ensured and which can be maintained for a time sufficient to effect provisions to reach a safe state. Under this state, the main characteristics are as follows:

1) The core remains sub-critical;

2) The residual heat removal can be ensured;

3) The core coolant volume remains stable;

4) Radioactive releases remain tolerable.

For the assumptions of most events, the reactor is at a sub-critical state when the emergency shutdown happens, and the core remains sub-critical after shutdown. Examples are provided below:

1) For DBC-2 events, the controlled state is:

- During a controlled state, the core remains sub-critical by inserting the control rods under reactor trip signal;
- Between the controlled state and the safe state, during the RCP [RCS] cooling process, the operators can maintain the sub-critical state.

2) For small break LOCA cases, the controlled state is:

- The break flow can be compensated;
- The heat in RCP [RCS] can be evacuated;
- The heat in the In-containment Refuelling Water Storage Tank (IRWST) can be evacuated;
- Under a controlled state and safe state, core sub-criticality needs to be ensured.

b) Safe state

Plant state, following an anticipated operational occurrence or accident conditions, in which the reactor is subcritical and the fundamental safety functions can be ensured and maintained as stable in the long term. Under this state:

1) The core remains sub-critical, even after Xenon burn-off;

2) The residual heat is evacuated;

3) Radioactive releases remain tolerable.

12.5.3 Methodology

A methodology may be defined as a set of procedures, or rules, for the calculation methods to be implemented to ensure the conservative nature of the results. The transient analysis must use calculation codes that are appropriate for the relevant physical phenomena.

The methodology of DBC analysis is developed in the following steps:

1) Define initiating event;

2) Identify key physical phenomena and confirm selected computer codes are appropriate to perform the studies;

3) Identify analysis assumptions including dominant parameters, uncertainties and penalties, etc., within the calculations.

Uncertainties must be considered:

1) Either in a deterministic manner for each dominant parameter considered as conservative value; or

2) In a statistical manner with the uncertainties in several parameters statistically combined.

The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip.

Once this scenario is established, the systems claimed during the transient, as identified in Sub-chapter 12.5.6, are assumed to work properly.

A DBC study must show that the safety criteria are met with a high confidence level of at least 95%.

12.5.4 Initial Conditions

The initial conditions for DBC analyses correspond to a steady state operation.

The definition of the DBC involves the definition of safety analysis domains to be considered. The safety analysis domains are described in Sub-chapter 12.4.

Within the given safety analysis domain, the most pessimistic operating condition is considered with regard to the fulfilment of the DBC acceptance criteria, e.g. full power operation for LOCA in state A, or the maximum RCP [RCS] pressure of 3.2 MPa for LOCA in state C (RIS [SIS] operation in RHR mode).

The physical parameters are within the limits provided by the plant controls. A conservative combination of parameters is considered including uncertainties, dead bands and response times. For each DBC event, the most conservative case is analysed.

The list of DBC events covers all plant operating conditions, including shutdown states, as potential initial conditions before the occurrence of initiating events.

12.5.5 Rules for Operator Actions

The transient can be divided into automatic phase and manual phase. There are obvious differences between these two phases:

a) Automatic phase means the transient between occurrence of an incident and the first manual action;

b) Manual phase means the transient between the first manual action (after a grace period) and the safety shutdown.

During the manual phase, in addition to the consideration of the automatic actions, the manual actions need to be considered. The definition of “grace period” of the operator actions is:

1) A manual action from the MCR is assumed to take place, no earlier than 30 minutes after the first significant information is transmitted to the operator;

2) A local manual action, i.e. a manual action that must be performed outside the MCR, is assumed to take place no earlier than one hour after the first significant information is transmitted to the operator.

For most cases, the controllable state can be achieved with automatic actions. However, manual actions are also allowable when the grace period condition is met.

The operators must follow the rules for emergency operation. In the DBC analysis, operator errors are not considered.

12.5.6 Safety Classification of Mechanical, Electrical and I&C Systems

The safety classification concept and the related wording are defined in Chapter 4.

In DBC analysis, the safety classification of systems can be divided as following:

a) Safety Category 1 Function (FC1) systems and Safety Category 2 Function (FC2) systems;

b) Safety Category 3 Function (FC3) systems and Non-Classified (NC) systems.

12.5.6.1 FC1, FC2 Systems and Functions

Conservative performances of the FC1 and FC2 systems are considered in DBC analyses, i.e., the most pessimistic system efficiencies of the FC1 and FC2 systems are considered including following aspects:

a) Conservative uncertainties on equipment characteristics;

b) Conservative uncertainties on actuation of signals;

c) The most pessimistic environmental conditions.

It shall be shown in the DBC analyses that:

a) The controlled state can be reached relying only on FC1 systems;

b) The transition from the controlled state to the safe state can be done relying only on FC1 and/or FC2 systems.

12.5.6.2 FC3 and NC Systems

The following principles apply to FC3 and NC systems used in the DBC analysis:

a) If the transient leads to the actuation of an FC3 or NC system, and if the operation of this system would have a beneficial effect with regard to a safety criterion, the DBC analysis shall be performed without considering this system;

b) If the transient leads to the actuation of an FC3 or NC system, and if this system worsens the consequences of the transient with regard to a safety criterion, the DBC analysis shall be performed assuming the system is operating normally;

For example: Following loss of main feedwater, the pressuriser main spray should be considered to operate normally when calculating the minimum DNBR.

c) If the transient has no impact on FC3 or NC system performance (no change of status, no change of operating and environmental conditions), and if the system were operating prior to the initiating event, the system is assumed to continue in normal operation.

For example: RCV [CVCS] charging is FC3 safety classified. If RCV [CVCS] charging is supposed to be in operation before SGTR happens, it is assumed to continue in operation during the transient until it is isolated.

d) More generally, an FC3 or NC system is assumed either to work correctly or not to work at all. Spurious operation is not considered in the DBC analyses.

For example, when considering the RCV [CVCS], the charging pumps are supposed to function as intended, or not at all.

e) Turbine isolation valves are not FC1 or FC2 classified, but they are considered to close normally after a reactor trip. This is justified because they are redundant in series. The situation described above is its design condition, and they are designed as “fail safe”. After the turbine trip, the disconnection of the main power generator is also assumed to be effective.

12.5.7 Application of the Single Failure Criterion (SFC) in the Safety Analysis

The single failure criterion (SFC) is addressed in the section related to the general design safety bases (see Chapter 4).

For the DBC analyses, the term single failure is considered as any active or passive failure, independent of the postulated initiating event, which affects all or part of equipment used in the analysed transient. Active single failure applies to equipment that needs a change of state to fulfil its function and that has beneficial effects on the transient.

In system design, this concept is considered under the same conditions as described in the universal design safety criterion (see Chapter 4), and the following aspects are considered:

a) If FC1 and FC2 functions can be fulfilled by more than one safety system, including equipment or auxiliary systems, the single failure must be applied to these systems only once;

b) Concerning passive single failure, it must be verified in the DBC analysis that a single failure in the form of a leak at any location in the pressure boundary and its consequential failure do not prevent the performance of the required safety function.

In DBC analyses, the following additional rules must be applied:

a) The most pessimistic single failure must be assumed to occur anywhere in the systems needed to perform the safety function;

b) Consequential failures resulting from the assumed failure must be considered in single failure criteria;

c) If necessary, a sensitivity analysis must be performed for a given initial event with the application of the SFC to different components. The purpose is to determine the worst-case single failure with regard to safety criteria;

d) The active single failure must be considered from the beginning of analysis. The passive single failure is considered 24 hours after the transient. The assumption for passive single failure is consistent with Chinese domestic requirements and additional analysis for the UK HPR1000 will be carried out in the future.

e) Any exception with respect to the single failure must be stated and justified;

f) The spurious opening of a safety valve is considered as an initial event;

g) The non-closure of a safety valve after actuation is considered as an application of the SFC.

12.5.8 Loss of Off-Site Power (LOOP)

LOOP due to turbine trip shall be considered for DBC-2, DBC-3 and DBC-4 accidents at power if it is conservative.

12.5.9 Analysis Rules Specific to DBC Events Associated with the Fuel Storage Pool

This section involves acceptance criteria, states definitions, initial conditions and the analysis rules of accidents associated with the fuel storage pool. Due to the specific characteristics of the fuel storage pool, such as operating at low pressure, the physical parameter changes in the accidental transient are very low when compared to those affecting the core in the reactor building. The acceptance criteria and analysis rules are consequently adapted and are described in the sub-sections below.

12.5.9.1 Acceptance Criteria

The acceptance criteria for the DBC-2/3/4 transients related to the fuel storage pool are as follows:

a) Permanent maintenance of sub-criticality;

b) The fuel assemblies in the fuel pool remain covered;

c) Removal of decay heat from the spent fuel pool.

   1) For DBC events without spent fuel pool accidental drainage, the removal of decay heat from the fuel storage pool is ensured through a decoupling criterion: the fuel storage pool temperature must remain lower than 80℃.

   2) For DBC with accidental fuel storage pool drainage events, the removal of decay heat from the fuel storage pool is ensured through a decoupling criterion: the fuel storage pool water temperature must remain below 95℃.

12.5.9.2 States Definition

a) Controlled state

The controlled state is a state in which the decay heat removal is ensured in the short term. For the fuel storage pool without draining events, because of the long grace period before fuel exposure, the controlled state can be considered as reached from the initial time. For the fuel storage pool with draining events, the controlled state is reached when the fuel pool water inventory is stabilised, after the draining has been stopped and no fuel assemblies have been exposed.

b) Safe state

The safe state is a state where the decay heat of the fuel assemblies in the fuel storage pool is reliably removed by at least one PTR train, with a significant margin to boiling.

For each DBC-2/3/4 fuel storage pool related transient, it must be demonstrated that the safe state can be reached and maintained.

12.5.9.3 Initial Conditions

The initial conditions for DBC analyses correspond to a steady state operation. Three initial conditions are considered in the spent fuel pool DBC events.

a) Refuelling

After the total unloading of the core, the pool is supposed to be filled with the following fuel assemblies: fuel assemblies just unloaded, new fuel assemblies (for the following cycle) and fuel assemblies unloaded from cycles previous to the most recent.

b) BOC

The fuel assemblies in the fuel storage pool are as per the ‘refuelling’ state, minus those that have just been loaded in the core for the cycle to come.

c) EOC

Fuel assemblies are the same as those at the BOC, but preventive maintenance performed at this stage when the decay heat is at its lowest will be considered.

To maximise the pool temperature reached at the end of the transient, the transient is analysed considering uncertainties in the decay heat value.

12.5.9.4 Analysis Rules

12.5.9.4.1 Rules for Operator Actions

In the DBC safety case analysis associated with the fuel storage pool, the rules for operator action are consistent with the rules specified for DBC accident analysis in Sub-section 12.5.5.

12.5.9.4.2 Mechanical, Electrical and I&C Systems used in DBCs Analysis

It shall be shown in the DBC analyses associated with the fuel storage pool that:

a) The controlled state can be reached relying only on FC1 systems;

b) The transition from the controlled state to the safe state can be done relying only on FC1 and/or FC2 systems.

12.5.9.4.3 Application of the Single Failure Criterion (SFC) in the Safety Analysis

For the DBC analysis associated with the spent fuel storage pool, the term single failure will be understood as any active failure, independent of the postulated initiating event, which affects all or part of the equipment used in the analysed transient. It applies to equipment that needs a change of state to fulfil its function and that has beneficial effects on the transient.

Passive single failures are not considered for the PTR [FPCS] as part of the DBC analysis associated with fuel storage pool because of the specific characteristics of the fuel storage pool cooling system, such as operating at low pressure, and use of an in-service inspection program.

12.5.9.4.4 Preventive Maintenance

PTR [FPCS] preventive maintenance is programmed when the grace period before boiling in the fuel storage pool is long enough. The time taken for boiling to start depends on both the decay heat and the cooling water temperature.

In the DBC analysis, preventive maintenance for the PTR [FPCS] train is assumed to be performed under EOC conditions, when the decay heat in the fuel storage pool is at its lowest, but whilst assuming a conservatively high cooling water temperature.

During the refuelling stage, PTR [FPCS] preventive maintenance is not performed. However, maintenance of the supporting systems may be carried out during these periods. To keep the trains separate and independent, suitable measures must be implemented on the support systems.

Periodic tests are assumed to be performed during power operation, by switching from one PTR [FPCS] train to another. As a result, the periodic tests have no impact on the accident studies.

12.5.9.4.5 Loss of Offsite Power (LOOP)

Loss of offsite power is not considered in the DBC safety analysis associated with fuel storage pool. The occurrence of initiating events associated with fuel storage pool does not affect the core in the reactor building and does not cause the turbine trip. LOOP as an initiating event has been included in the fault list associated with the fuel storage pool.

12.6 Plant Characteristics taken into account in Accident Analyses

12.6.1 Plant Level Parameters

12.6.1.1 Plant Geometrical Data

T-12.6-1 lists the main geometrical data of the RCP [RCS].

12.6.1.2 Plant Initial Conditions

Initial conditions of the fault analysis are obtained by considering maximum steady state uncertainties in a pessimistic way: adding or subtracting maximum steady state uncertainties to or from nominal values. The steady state uncertainties include the measurement uncertainties, the steady state fluctuations, and the control dead band (if applicable).

T-12.6-2 lists the nominal values and associated maximum uncertainties for all the relevant parameters:

a) Core power;

b) PZR pressure;

c) RCP [RCS] average temperature;

d) PZR level;

e) SG level.

The transient response of the RCP [RCS] is related to the initial power distribution. The nuclear design of the reactor core minimises adverse power distribution through adequate positioning of the control rods and adherence to operating instructions.

The most unfavourable power distributions that can occur during normal operation are considered as initial conditions for the transient studies.

T-12.6-3 provides additional information of other relevant parameters for plant initial conditions.

T-12.6-4 provides the maximum steady state uncertainties for initial conditions adopted in accidents analysis.

T-12.6-1 Main Geometrical Data of RCP [RCS]

Number of fuel assemblies in reactor core	177(17×17 type)
Number of fuel rods per assembly	264
Core active height	3.6576 m
Volume of PZR	67 m^3
Total volume of RCP [RCS]	357.5 m^3
Inner diameter of surge line	0.284 m
Inner diameter of Hot/crossover/cold legs	0.760 m

Note: The above data is based on the cold conditions.

T-12.6-2 Plant Initial Conditions

Parameter	Nominal values (thermal hydraulic flowrate)	
	0 %FP	100 %FP
Core power	0 %FP	100 %FP
PZR pressure	15.5 MPa	15.5 MPa
RCP [RCS] average temperature	295.0℃	307.0℃
PZR level	36 %R	53.1 %R
SG level	50 %NR	50 %NR

FP: Full Power

R: Range

NR: Narrow Range

T-12.6-3 Plant Initial Conditions (other relevant parameters)

Nominal NSSS thermal power output	3160MWth		
Nominal core thermal power	3150MWth		
ARE [MFFCS]flow temperature at SG inlet	228℃		
	For Thermal hydraulic flowrate	For Best estimate flowrate	For Mechanical design flowrate
RCP [RCS] flowrate for each loop	24000 m^3/h	25450m^3/h	26500 m^3/h
Core average temperature	308.1℃	307.7℃	307.5℃
RPV inlet temperature	288.6℃	289.5℃	290.2℃
RPV outlet temperature	325.4℃	324.5℃	323.8℃
RPV average temperature	307.0℃	307.0℃	307.0℃

T-12.6-4 Maximum Steady State Uncertainties for Plant Initial Conditions Adopted in Accidents Analysis

Parameter	Uncertainty	Comment
Power	±2 %FP	Tolerance error for power measurement
Primary pressure	±0.25 MPa	Tolerance error for steady state fluctuation and measurement uncertainties
Primary temperature	±2.5℃	Tolerance error for the control dead band and measurement uncertainties
PZR level	±7%	Tolerance error of measurement techniques, sensors and control channel techniques
SG level	±10%	Tolerance error of measurement techniques, sensors and control channel techniques

12.6.2 Core related Parameters

12.6.2.1 Reactivity Coefficients

The transient response of the reactor is related to reactivity feedback effects, in particular the moderator density coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in Chapter 5.

In the analysis of some events, conservatism requires the use of maximum reactivity coefficient values, whereas for others, conservatism requires the use of minimum reactivity coefficient values. The values used are given in T-12.6-5.

RCP [RCS] boron concentration for the initial state (nominal operating conditions), and required boron concentration for the RIS [SIS] in RHR mode are provided in T-12.6-6.

T-12.6-5 Reactivity Coefficients (conservative values set for point kinetic model for UO_2 fuel management)

	Minimum	Maximum
Prompt neutron generation time	13.5 μs	31 μs
Fraction of delayed neutrons	440 pcm	750 pcm
Moderator density coefficient	0 $(\Delta k/k)/(g/cm^3)$	0.58$(\Delta k/k)/(g/cm^3)$
Doppler temperature coefficient	-4.65pcm/℃	-1.8pcm/℃
Doppler power coefficient	-12.4 pcm/%FP at 100%FP	-5.2 pcm/%FP at 100%FP
	-12.9 pcm/%FP at 80%FP	-5.5 pcm/%FP at 80%FP
	-15.0 pcm/%FP at 60%FP	-5.7 pcm/%FP at 60%FP
	-18.6 pcm/%FP at 40%FP	-6.1 pcm/%FP at 40%FP
	-23.5 pcm/%FP at 20%FP	-6.5 pcm/%FP at 20%FP
	-29.8 pcm/%FP at 0%FP	-6.9 pcm/%FP at 0%FP

T-12.6-6 RCP [RCS] Boron Concentrations

Plant conditions	EOC	BOC
Nominal operating conditions (full power)	~ 6 ppm (35% enriched ^{10}B)	567 ppm(first cycle) / 1011 ppm (equilibrium cycle) (35% enriched ^{10}B)
RIS [SIS] in RHR mode	1300ppm (35% enriched ^{10}B)	1300ppm (35% enriched ^{10}B)

12.6.2.2 Design Hot Channel Factor

The design hot channel factor used in the safety analysis depends on a transient study (related to uncertainties). The power distribution may be characterised by the enthalpy rise hot channel $F_{\Delta H}$ and the heat flux hot channel factor F_Q. The values are given below:

a) Enthalpy rise hot channel factor $F_{\Delta H}$=1.65;

b) Heat flux hot channel factor F_Q=2.45.

12.6.2.3 Power Distribution

The transient response of the reactor system is related to the initial power distribution. The nuclear design of the reactor core minimises adverse power distribution by adequate positioning of control rods and adherence to operating instructions. The power distribution may be characterised by the total enthalpy rise factor $F_{\Delta H}$ and the total peaking factor F_Q. Peak factor limits are defined in Chapter 5.

For transients which may be DNB limited, $F_{\Delta H}$ is of importance. The $F_{\Delta H}$ increases with decreasing power levels due to control rod insertion. All transients that may be DNB limited are assumed to begin with an $F_{\Delta H}$ value consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is discussed in Sub-chapter 5.5.

For transients, which may be overpower limited, the F_Q is of importance. These transients are assumed to begin with plant conditions, including power distributions, which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients which are slow with respect to the fuel rod thermal response, fuel rod thermal evaluations are determined as discussed in Sub-chapter 5.4.

For overpower transients which are fast with respect to the fuel rod thermal response (for example, the uncontrolled rod cluster control assembly bank withdrawal during subcritical conditions and rod cluster control assembly ejection incidents which result in a large power rise over a few seconds), a detailed fuel heat transfer calculation is performed.

12.6.2.4 Fission Power and Decay Heat after Reactor Trip (RT)

This sub-chapter describes the method for fission power and decay heat calculation after reactor trip.

The residual heat in a subcritical core can be split into three terms:

a) The residual thermal power generated by residual fissions (A term);

b) The heat came from the decay of capture products U-239 and Np-239 (B term);

c) The heat produced from the decay of fission products and actinides excluding U-239 and Np-239 (C term).

The residual thermal power of A term is defined by the number of neutrons, whose sources are classified as following list:

a) The decay of delayed neutron precursors;

b) The spontaneous fissions of actinides;

c) The (α, n) reactions.

The residual thermal power due to the decay of fission products and actinides (B+C term) depends on the amount of these products at the moment of RT. The main parameters that have an influence on the nuclide composition in the core are linked to the fuel type and to the fuel management. They are:

a) The initial fuel enrichment;

b) The number of fuel assemblies in the core;

c) The enrichment and the burnup of the different fuel assemblies;

d) The burnup history (irradiation sequences) of each fuel assembly (cycle length, specific irradiation power).

Uncertainties are considered in the calculation of the residual heat of B+C terms:

1) +1.645σuncertainty;

2) +2σuncertainty.

A time-dependent rod reactivity worth curve based on the decoupling dropping characteristics shown in T-12.6-7 .

T-12.6-7 RCCA Dropping Characteristics

Total dropping time (maximum)	4.20s (with earthquake) 3.00s (without earthquake)

12.6.3 I&C Signals

12.6.3.1 Primary and Secondary I&C Signals

I&C signals considered in the DBC accident analyses refer either to RT actuation or to FC1 and FC2 classified systems actuation (some non-FC1 and FC2 classified signals may be accounted for if in accordance with the accident analyses rules).

The FC1 and FC2 signals related to the Primary and Secondary (P/S) systems used in the DBC accident analyses are listed in Table

T-12.6-8, with mention of the setpoints and their associated uncertainties, indicates the FC1 systems that are not actuated by the I&C signals, such as the PZR safety valves and main steam safety valves (MSSVs).

The list of FC1 and FC2 signals does not provide the manual FC2 actions. These actions are mentioned in the relevant sub-chapters on accident analysis.

The maximum time delay is considered for the actuation of a signal and completion of the resulting action.

T-12.6-9 defines the delays considered in the DBC accident analyses.

T-12.6-8 FC1 and FC2 Signals (P/S related)

PZR pressure		
Action	Threshold	Uncertainties
3^{rd} PSV opening (closing)	17.7 MPa (90% opening threshold)	± 0.15 MPa
2^{nd} PSV opening (closing)	17.4 MPa (90% opening threshold)	± 0.15 MPa
1^{st} PSV opening (closing)	17.1 MPa (90% opening threshold)	± 0.15 MPa

SG pressure			
	Action	Threshold	Uncertainties
High 1	RT, TT	8.6 MPa	normal conditions: 0.15 MPa degraded conditions: 0.5 MPa
Low 1	RT, TT	5 MPa	normal conditions: 0.15 MPa degraded conditions: 0.5 MPa
Low 2	ARE isolation	4 MPa	normal conditions: 0.15 MPa degraded conditions: 0.5 MPa
Low 3	VDA isolation	4 MPa	normal conditions: 0.15 MPa degraded conditions: 0.5 MPa
Low 4	Charging line isolation	3.5 MPa	normal conditions: 0.15 MPa degraded conditions: 0.5 MPa

T-12.6-9 Safeguard Actions Delay

Safeguard action	Delay (maximum)	Note
Medium Head Safety Injection (MHSI) & Low Head Safety Injection (LHSI)	17 s	Delay in SI pump start-up to full flowrate after SI signal (without LOOP)
MHSI	38 s	Delay in SI pump start-up to full flowrate after SI signal (with LOOP)
LHSI	33 s	Delay in SI pump start-up to full flowrate after SI signal (with LOOP)
ASG [EFWS] actuation	15 s	Delay in pump start-up to full flowrate after ASG signal (without LOOP)
	55 s	Delay in pump start-up to full flowrate after ASG signal (with LOOP)
ARE full load isolation	5.0 s	Valves closing delay
ARE low load isolation	20 s	Valves closing delay
ASG [EFWS] isolation	60 s	Valves closing delay
RCV [CVCS] letdown line isolation	35 s	Valves closing delay
RCV [CVCS] charging line isolation	40 s	Valves closing delay
MSIV closure	5.0 s	Valves closing delay
VDA [ASDS] opening	1.8 s	VDA IV opening delay

Safeguard action	Delay (maximum)	Note
VDA [ASDS] isolation	20 s	VDA IV closing delay
RCPs cut-off	0.15 s	Breaker opening delay
Turbine trip	2.2 s (minimum) 4.2 s (maximum)	Delay between RT signal and TT signal

12.6.3.2 Core Related I&C Signals

This sub-chapter lists the setting values of the reactor trip signals and the temperature difference protection values used in the DBC analysis.

T-12.6-10 provides the signals and thresholds for reactor trip considered in the DBC accident analyses.

T-12.6-10 Reactor Trip Protection Thresholds

Signal	Threshold	Delay, s	Uncertainties
High neutron flux (intermediate range)	25 %FP	0.7	10 %FP
High neutron flux (power range, low setpoint)	25 %FP	0.7	10 %FP
High neutron flux (power range, high setpoint)	109 %FP	0.7	9 %FP
High positive neutron flux rate	5 %FP	0.7	1.00 %FP
High negative neutron flux rate	-5 %FP	0.7	1.00 %FP
Overtemperature ΔT and Overpower ΔT	See reference [12]	1.0	See reference [12]
Low flow rate in two primary loops	88.80% nominal flowrate	1	3% nominal flowrate
Low flow rate in one primary loop	88.80% nominal flowrate	1	3% nominal flowrate
Low RCP speed	91.90% nominal speed	0.7	0.70% nominal speed
PZR pressure low 2	13.5 MPa	1.3	normal conditions: 0.1 MPa degraded conditions: 0.5 MPa

Signal	Threshold	Delay, s	Uncertainties
PZR pressure high 2	16.62 MPa	1.3	normal conditions: 0.1 MPa degraded conditions: 0.5 MPa
PZR level high 1	80.00%	1.9	normal conditions: 2.5% degraded conditions:11%
SG pressure drop high 1	-0.5 MPa/min, initial pressure - 0.7 MPa	1.3	normal conditions: 0.15 MPa degraded conditions: 0.5 MPa
SG pressure low 1	5.0 MPa	1.3	normal conditions: 0.15 MPa degraded conditions: 0.5 MPa
SG pressure high 1	8.6 MPa	1.3	normal conditions: 0.15 MPa degraded conditions: 0.5 MPa
Containment pressure high 1	0.123 MPa	1.3	normal conditions: 0.02 MPa degraded conditions: 0.02 MPa

12.6.4 Safety Systems Functions and Characteristics

This sub-chapter lists the characteristics of the safety systems used to mitigate the consequences of a DBC accident.

In DBC accident analyses, the systems mitigating the consequences of an event are FC1 or FC2 classified:

a) The controlled state can be reached relying only on FC1 systems;

b) The transfer from the controlled state to the safe state can be done relying only on FC1 and/or FC2 systems.

The FC1 and FC2 mechanical systems accounted for in the DBC accident analyses include:

a) The core control and shutdown rods, performing RTs;

b) The RCP [RCS] and SG isolation valves;

c) The RCP [RCS] and SG fluid systems performing injection;

d) The RCP [RCS] and SG fluid systems performing pressure relief;

e) The RCV [CVCS] control tank isolation valves.

Those systems are designed according to the conservative DBC analyses rules defined in Sub-chapter 12.5:

a) Minimum guaranteed efficiency;

b) Consideration of the worst-case single failure.

The FC1 systems/functions involved in accident analyses are:

a) RT;

b) RIS [SIS] (MHSI, LHSI, accumulators, In-containment Refuelling Water Storage Tank (IRWST));

c) PSVs;

d) VDA [ASDS];

e) MSSV;

f) MSIV;

g) ASG [EFWS] actuation;

h) ASG [EFWS] isolation;

i) Containment/RCP [RCS] isolation;

j) ARE [MFFCS] full load isolation;

k) ARE [MFFCS] low load isolation;

l) SG blowdown isolation

m) RCPs trip;

n) Emergency Diesel Generator (EDG) startup.

The FC2 systems/functions involved in accident analyses (other than FC1 instances previously listed) are:

a) RBS [EBS] startup;

b) LHSI in hot leg SI mode (switchover to hot leg injection);

c) LHSI in RHR mode;

d) RCPs manual cut-off;

e) MHSI manual cut-off;

f) Accumulators manual isolation;

g) VDA [ASDS] manual operation;

h) MSIV bypass;

i) SG blowdown between two SGs;

In addition to the above FC1 and FC2 systems, non-FC1 or FC2 systems may be considered in course of the DBC accident analyses according to the DBC analyses rules (either negative impact, or positive impact but not experiencing any discontinuity in their operation).

12.7 Analyses of DBC-2 Events

12.7.1 Increase in Heat Removal by the Secondary Circuit

12.7.1.1 Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature (State A\B)

12.7.1.1.1 Initiating Event

This event results in a sudden reduction in the feedwater temperature of the SGs. The reduction in feedwater temperature increases the heat transfer from the primary circuit to the secondary circuit, which results in a decrease of the primary coolant temperature and an insertion of positive reactivity. The core power increases, which potentially causes DNB.

Feedwater system malfunctions causing a reduction in feedwater temperature may be caused by:

a) Failure of one feedwater heater;

b) Spurious opening of a feedwater bypass valve.

12.7.1.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.1.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

   1) Overtemperature ΔT;

   2) Overpower ΔT;

   3) High neutron flux (power range, high setpoint).

b) After reactor trip:

   1) Turbine trip;

   2) The ARE [MFFCS] full load lines are isolated;

   3) The RCP [RCS] heat is removed by the VDA [ASDS];

   4) The SGs are fed by the ASG [EFWS].

12.7.1.1.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

The reduction in feedwater temperature increases the heat transfer from the primary circuit to the secondary circuit, which results in a decrease of the primary coolant temperature and an insertion of positive reactivity. The core power is increased. The consequence of this event can be attenuated by the thermal capacity of the primary and secondary circuit. The reactor finally reaches a new equilibrium. The reactor can be tripped by the operator manually, and the reactor can be stabilized in the hot shutdown state after reactor trip.

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.1.1.5 Analysis Assumptions

The detailed assumptions are presented in the Reference [13]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition. The hot shutdown condition is not explicitly analysed because the peaking nuclear power is low. There is no DNB or fuel melting risk.

b) A conservative decrease of feedwater temperature is considered.

c) The Doppler power coefficient is considered as its minimum absolute value, and the moderator density coefficient is considered as its maximum absolute value.

d) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

e) The single failure is not considered in this event because no protection signal is actuated.

12.7.1.1.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [13]) shows that the minimum DNBR is greater than the design limit which is described in Sub-section 5.6.2.1. The nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by other events. The description is provided in Sub-chapter 12.10.

12.7.1.2 Feedwater System Malfunctions Causing an Increase in Feedwater Flow (State A\B)

12.7.1.2.1 Initiating Event

This event leads to an increase in heat removal from the secondary circuit. During power operation, this increase in heat removal leads to an increase in the core power due to the moderator feedback or by the RCP [RCS] average coolant temperature control. In the hot shutdown state, the heat removal increase leads to a return to power due to the moderator feedback. Therefore, feedwater system malfunction causing an increase in feedwater flow may lead to the DNB.

Feedwater system malfunctions causing an increase in feedwater flow may be caused by:

a) Malfunctions of the ARE [MFFCS] leading to full opening of a feedwater control valve;

b) Spurious actions of the operator leading to full opening of a feedwater control valve.

c) Spurious start of backup APA [MFPS] pump.

12.7.1.2.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.1.2.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) Full power operation

    1) RT is actuated on receipt of any of the following signals:

        - High neutron flux (power range, high setpoint);

        - Overtemperature ΔT;

        - Overpower ΔT;

        - SG level (narrow range) high 1.

    2) After reactor trip:

        - Turbine trip;

        - The ARE [MFFCS] full load lines are isolated;

        - The RCP [RCS] heat is removed by the VDA [ASDS];

- The SGs are fed by the ASG [EFWS].

3) The ARE [MFFCS] full load isolation is actuated by "SG level (narrow range) high 1" or RT signal.

4) The ARE [MFFCS] low load isolation is actuated by "SG level (narrow range) high 0" and RT signal.

b) Hot shutdown state

1) The ARE [MFFCS] full load isolation is actuated by "SG level (narrow range) high 1" or RT signal.

2) The ARE [MFFCS] low load isolation is actuated by "SG level (narrow range) high 0" and RT signal.

3) The main steam isolation valves are closed by "Pressure drop of SG high 1" signal.

4) The RIS [SIS] is triggered by "Pressuriser pressure low 3" signal.

12.7.1.2.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

During power operation, an increase in feedwater flow causes a reduction of the primary average coolant temperature and pressure. Due to the effects of negative moderator feedback, the core power increases. This event can lead to RT triggered by "SG level (narrow range) high 1" signal. The ARE [MFFCS] full load isolation is actuated by "SG level (narrow range) high 1" or RT signal. If the SG level remains above the threshold of "SG level (narrow range) high 0" after a period of time delay, the low load main feedwater lines are isolated.

In hot shutdown state, an increase in feedwater flow causes a reduction of the primary average coolant temperature and pressure. Due to the effects of negative moderator feedback, core cooling leads to a decrease of the core shutdown margin, and then the core may return to criticality. The ARE [MFFCS] full load isolation is actuated by "SG level (narrow range) high 1" or RT signal. The main feedwater low load control and isolation valves are closed on RT signal and "SG level (narrow range) high 0" signal. The MSIVs are closed by "pressure drop of SG high 1" signal, which reduces the main steam flowrate to zero.

Later on, the controlled state is reached. Under this state, the residual heat is removed via the VDA [ASDS] of all SGs. And the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.1.2.5 Analysis Assumptions

The detailed assumptions are presented in the reference report [14]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition and in hot shutdown condition respectively.

b) A conservative increase in main feedwater flowrate is considered.

c) The Doppler power coefficient is considered as its minimum absolute value, the moderator density coefficient is considered as its maximum absolute value.

d) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

e) The single failure is applied on one “SG level (narrow range) high 1” channel in power operation condition. The single failure is applied on one train of MHSI in hot shutdown condition.

12.7.1.2.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

In power operation conditions, the detailed analysis of this fault (see Reference [14]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before the reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

In hot shutdown state, the reactor returns to criticality. However, the peaking nuclear power is low. There is no DNB or fuel melting risk.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.1.3 Excessive Increase in Secondary Steam Flow (State A\B)

12.7.1.3.1 Initiating Event

This event is defined as a rapid increase in the steam flow that causes a power mismatch between reactor power and steam flow demand. An excessive increase in steam flow may thus lead to an inadequate cooling of the fuel cladding by DNB.

An excessive increase in secondary steam flow may be caused by:

a) Spurious opening of the turbine inlet valve;

b) Spurious opening of one GCT [TBS] control valve.

12.7.1.3.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.1.3.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

    1) Overtemperature ΔT;

    2) Overpower ΔT;

    3) High neutron flux (power range, high setpoint).

b) After reactor trip:

    1) Turbine trip;

    2) The ARE [MFFCS] full load lines are isolated;

    3) The RCP [RCS] heat is removed by the VDA [ASDS];

    4) The SGs are fed by the ASG [EFWS].

12.7.1.3.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

An excessive increase in secondary steam flow induces an overcooling of the primary circuit. This overcooling leads to a core power increase due to moderator feedback or the RCP [RCS] average coolant temperature control. Following this, the reactor can be protected by RT following receipt of the “overpower ΔT”, “overtemperature ΔT” or “high neutron flux (power range, high setpoint)” protection signals.

If the increase in steam flow is limited, the event may not result in a reactor trip, and the reactor stabilises at a higher power level. The alarm for high core thermal power can alert the operator to the occurrence of such an accident. Once the source of the alarm is identified, the operator attempts to stop the excessive increase in steam flow, and the reactor returns to normal power operation. Otherwise the operator needs to transfer the reactor to the controlled state.

After reactor trip, turbine trip and the ARE [MFFCS] full load isolation are initiated. The plant stabilises in the hot shutdown state.

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.1.3.5 Analysis Assumptions

The detailed assumptions are presented in Reference [15]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition. The core power which is lower than full power can be bounded by this condition in State A and State B since the thermal parameters are more pessimistic.

b) A conservative step increase in steam flow extracted from the SGs is assumed at the start of the transient.

c) Plant behaviour is analysed using four cases:

1) BOC, manual reactor control;

2) EOC, manual reactor control;

3) BOC, automatic reactor control;

4) EOC, automatic reactor control.

d) The Doppler power coefficient is considered as its minimum absolute value, the moderator density coefficient is considered to be zero during BOC and the maximum in absolute value during EOC.

e) The single failure is not considered in this event because no protection signal is actuated.

12.7.1.3.6 Results and Conclusions

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [15]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. The nuclear power increases by a small amount from the initial moment, and a limited increase in the fuel temperature is induced. However, the fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.2 Decrease in Heat Removal from the Primary Circuit

12.7.2.1 Turbine Trip (State A)

12.7.2.1.1 Initiating Event

For a turbine trip event, the reactor could be tripped directly if power > 10% (P10) and some conditions regarding condenser unavailability are satisfied. The turbine stop valves close rapidly on loss of trip-fluid pressure actuated by one of a number of trip signals. The steam flow to the turbine decreases abruptly following the stop valve closure. Sensors on the stop valves detect the turbine trip and steam dump is initiated. The loss of steam flow results in a rapid rise in the secondary circuit temperature and pressure. Consequently, the pressure also increases in the primary circuit. Because of the rapid reduction in steam flow, the capacity of the secondary system to remove the core heat decreases, further potentially causing DNB and insufficient cooling of the fuel cladding.

The turbine trip may be caused by:

a) A spurious closing signal of the turbine stop valves;

b) The disruption of the external grid (causing a fluctuation of the voltage and frequency);

c) The power plant breaker triggering signal.

12.7.2.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.2.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

   1) Overtemperature ΔT;

   2) Pressuriser level high 1;

   3) SG pressure high 1;

   4) Pressuriser pressure high 2;

   5) Nuclear power is higher than 10%FP (P7 signal)

      - Low flow rate in two primary loops;

      - Low RCP speed in two primary loops.

   6) Nuclear power is higher than 30% FP (P8 signal):

- Low flow rate in one primary loop.

b) The ASG [EFWS] is actuated by an “SG level (wide range) low 2” signal.

c) The VDA [ASDS] is actuated by an “SG pressure high 1” signal.

d) The PSVs are opened when the PZR pressure reaches the setpoint.

e) After reactor trip:

1) The ARE [MFFCS] full load lines are isolated.

2) The RCP [RCS] heat is removed by the VDA [ASDS].

3) The SGs are fed by the ASG [EFWS].

12.7.2.1.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

After turbine trip, the loss of steam flow results in a rapid rise in secondary circuit temperature and pressure. Consequently, the temperature and pressure also increases in the primary circuit. The reactor trip is automatically triggered on “Pressuriser pressure high 2” signal.

If turbine trip results in the loss of offsite power, it leads to the decrease of the heat removal capacity of the fuel cladding. After the loss of power supply, the RCPs begin to coast down. The coast down of the reactor coolant pump speed takes several seconds due to the inertia of the flywheel. Furthermore, the main feedwater pumps are tripped after the loss of offsite power. When the speed of the reactor coolant pump reaches the “Low RCP speed” setpoint, RT is actuated so as to protect the core. During the whole transient, the instruments and the most important safety related motors are supplied by the uninterruptable power supply batteries. The EDGs are initiated by the low-voltage signal of the emergency busbar.

After RT, the RCCAs drop into the core and the core power decreases dramatically.

Since the secondary circuit pressure rises continuously after reactor trip, the VDA [ASDS] is automatically opened (triggered by “High steam generator pressure 1” signal), and the steam is discharged into the atmosphere. The GCT [TBS] is unavailable because of the loss of condensate pumps. During this period, the VDA [ASDS] may be unable to remove all the residual heat of the core. The PSVs may open to limit the increase in primary pressure, since the normal spray is unavailable. Then, the ASG [EFWS] is actuated by the “Low SG level low 2” signal.

Finally, the controlled state is reached. The residual heat is removed by the VDA [ASDS] of all SGs and the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to

the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.2.1.5 Analysis Assumptions

The detailed assumptions are presented in Reference [16]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition.

b) The turbine is tripped at time zero.

c) Plant behaviour is analysed using the following cases:

   1) Turbine trip without LOOP;

   2) Turbine trip with LOOP.

d) The Doppler power coefficient is considered as its maximum absolute value, the moderator density coefficient is considered as its minimum absolute value.

e) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

f) The single failure is applied on one “Pressuriser pressure high 2” channel (without LOOP) or one “Low RCP speed” channel (with LOOP).

12.7.2.1.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [16]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before the reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.2.2 Loss of Condenser Vacuum (State A)

Loss of condenser vacuum is one of the events that could cause turbine trip. The event of turbine trip is described in Sub-section 12.7.2.1.

After loss of condenser vacuum, the steam dump to the condenser is unavailable. Since steam dump is assumed unavailable in the turbine trip analysis, there are no additional adverse effects if the turbine trip is caused by loss of condenser vacuum. Therefore, the results and conclusions contained in Sub-section 12.7.2.1 apply to the loss of condenser vacuum accident.

12.7.2.3 Short Term Loss of Off-site Power (< 2 hours) (State A)

12.7.2.3.1 Initiating Event

A loss of offsite power leads to the loss of power supply to all RCPs, main feedwater pumps and condensate pumps. Because of the decrease of reactor coolant flow and the decrease of the heat removal capacity by the secondary circuit, the capacity of the primary coolant to remove the core heat decreases, potentially resulting in DNB and insufficient cooling of the fuel cladding.

The loss of offsite power may be caused by:

a) A complete loss of offsite grid;

b) An onsite AC power distribution system failure;

c) An external grid disturbance (dropped voltage or frequency).

12.7.2.3.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.2.3.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

  1) Nuclear power is higher than 10% full power (FP) (P7 signal)

    - The flow rate is low in two primary loops;

    - Low RCP speed in two primary loops.

  2) Nuclear power is higher than 30% FP (P8 signal)

    - The flow rate is low in one primary loop.

b) The ASG [EFWS] is actuated by the “SG level (wide range) low 2” signal.

c) The VDA [ASDS] is actuated by the “SG pressure high 1” signal.

d) The PSVs are opened when the PZR pressure reaches the setpoint.

e) After reactor trip:

  1) Turbine trip;

  2) The ARE [MFFCS] full load lines are isolated;

  3) The RCP [RCS] heat is removed by the VDA [ASDS];

  4) The SGs are fed by the ASG [EFWS].

12.7.2.3.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

After the loss of power supply, the RCPs begin to coast down. The coast down of the reactor coolant pump speed takes several seconds due to the inertia of the flywheel. Furthermore, the turbine and the main feedwater pumps are tripped after the loss of offsite power. The capacity of the primary coolant to remove the core heat decreases.

When the speed of reactor coolant pump reaches the "Low RCP speed" setpoint, the reactor trip is actuated so as to protect the core. After the reactor trip, the RCCAs drop into the core and the core power decreases dramatically.

During the whole transient, the instruments and the most important safety related motors are supplied by the uninterruptable power supply batteries.

Since the secondary circuit pressure rises continuously after the reactor trip, the VDA [ASDS] is automatically opened, and the steam is discharged into the atmosphere. The GCT [TBS] is unavailable because of the loss of condensate pumps. During this period, the VDA [ASDS] may be unable to remove all the residual heat from the core. The PSVs may open to limit the increase in primary pressure, since the normal spray is unavailable. Then, the ASG [EFWS] supplied by EDG is actuated by the "SG level (wide range) low 2" signal.

Finally, the controlled state is reached. The residual heat is removed by the VDA [ASDS] of all SGs and the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the PSVs.

12.7.2.3.5 Analysis Assumptions

The detailed assumptions are presented in Reference [17]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition.

b) The non-emergency AC power is lost at time zero, and then the RCPs begin to coast down.

c) The Doppler power coefficient is considered as its maximum absolute value, the moderator density coefficient is considered as its minimum absolute value.

d) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

e) The single failure is applied on one “Low RCP speed” channel.

12.7.2.3.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [17]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before the reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.2.4 Loss of Normal Feedwater Flow (Loss of All Main Feedwater Pumps and Startup and Shutdown Feedwater System (AAD [SSFS]) pumps) (State A)

12.7.2.4.1 Initiating Event

Loss of normal feedwater flow is an overheating event, leading to the reduction in capability of the secondary circuit to remove the heat generated in the primary circuit, and inadequate cooling of fuel cladding due to DNB.

The loss of normal feedwater flow may be caused by:

a) Feedwater lines isolation or control valves malfunction;

b) Failures of main feedwater pumps, valves, or other signals;

c) Spurious shutdown of the APA [MFPS] pumps and the AAD [SSFS] pumps.

12.7.2.4.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.2.4.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

   1) SG level (narrow range) low 1;

   2) Pressuriser pressure high 2;

   3) Overtemperature ΔT.

b) The ASG [EFWS] is actuated by the “SG level (wide range) low 2” signal.

c) The VDA [ASDS] is actuated by the “SG pressure high 1” signal.

d) The PSVs are opened when the PZR pressure reaches the setpoint.

e) After reactor trip:

   1) Turbine trip;

   2) The ARE [MFFCS] full load lines are isolated;

   3) The RCP [RCS] heat is removed by the VDA [ASDS];

   4) The SGs are fed by the ASG [EFWS].

12.7.2.4.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

After the loss of normal feedwater flow, the water inventory of SGs decreases while the core power remains unchanged, leading to the increase in the primary temperature and pressure.

Afterwards, the “SG level (narrow range) low 1” signal triggers RT automatically, thus leading to turbine trip. The secondary pressure is limited by the GCT [TBS] if it is available. Otherwise, it is limited by the VDA [ASDS]. The SG levels continuously decrease until the “SG level (wide range) low 2” signal occurs, and the ASG [EFWS] starts to remove the core residual heat.

Finally, the controlled state is reached. The residual heat is removed by the VDA [ASDS] of all SGs and the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.2.4.5 Analysis Assumptions

The detailed assumptions are presented in Reference [18]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition.

b) The main feedwater flowrate is completely lost at time zero.

c) The Doppler power coefficient is considered as its maximum absolute value; the moderator density coefficient is considered as its minimum absolute value.

d) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

e) The single failure is applied on one “SG level (narrow range) low 1” channel.

12.7.2.4.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [18]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before the reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.2.5 Loss of One Cooling Train of the Safety Injection System (RIS [SIS]) in Residual Heat Removal (RHR) Mode (State C\D)

12.7.2.5.1 Initiating Event

In this event, one RIS [SIS] train in RHR mode is lost, and the primary average coolant temperature increase leads to a reduction of the capability of the heat removal.

The loss of one RIS [SIS] train in RHR mode may be caused by:

a) Failure of one LHSI pump;

b) A malfunction of one RRI [CCWS]/RIS [SIS] heat exchanger;

c) Spurious closure of one valve at the RIS [SIS] point of suction from the hot leg, or of one valve at the point of injection;

d) Spurious RIS [SIS] train isolation signal.

12.7.2.5.2 Acceptance Criteria

The acceptance criteria to be considered for this event are:

a) Core water inventory is stable;

b) Heat removal is ensured on a long term basis.

12.7.2.5.3 Main Safety Functions

The following plant safety function can mitigate the event:

The safety injection is actuated on receipt of the “RCP [RCS] loop level low 1” signal.

12.7.2.5.4 Typical Events Sequences

The plant operating in RCP [RCS] ¾ loop level is the most pessimistic condition with respect to heat removal, because the RCP [RCS] water inventory is small. In this condition, two out of three RIS [SIS] trains in RHR mode are required to be in operation to maintain the primary average coolant temperature below 60℃. The third RIS [SIS] train is in standby in SI mode.

After the loss of one of the RIS [SIS] cooling trains, the residual heat is removed by the one remaining RIS [SIS] train in RHR mode.

12.7.2.5.5 Analysis Assumptions

The detailed assumptions are presented in Reference [19]. The main assumptions are listed as follows:

a) The plant operates in RCP [RCS] ¾ loop level condition.

b) The failure of one RIS [SIS] train in RHR mode is assumed.

c) The maximum decay heat is considered.

d) The single failure is applied to one MHSI pump.

12.7.2.5.6 Result and Conclusion

The detailed analysis of this fault [19] shows that one remaining RIS [SIS] train in RHR mode is able to remove the total RCP [RCS] primary power. The maximum primary average coolant temperature during the event is lower than the saturation temperature, and all the fuel assemblies are covered with water during the entire transient.

Thus the acceptance criteria for this event are met.

12.7.2.6 Spurious reactor trip (State A)

Spurious reactor trip can be initiated by either a manual shutdown due to the spurious actuation of an automatic RT signal or an operator error. This is covered by other DBC-2 events resulting in RT, in Sub-chapter 12.7.2.

12.7.3 Decrease in Core Coolant System Flow

12.7.3.1 Partial Loss of Core Coolant Flow (Loss of One Main Coolant Pump) (State A)

12.7.3.1.1 Initiating Event

Partial loss of core coolant flow leads to an increase in the primary coolant temperature. The decrease of core coolant flow causes a reduction of the capacity of the primary coolant to remove heat from the core. That can lead to an inadequate cooling of the fuel cladding through DNB.

The partial loss of core coolant flow may be caused by:

a) Mechanical or electrical failure in a reactor coolant pump;

b) Failure of the busbar supplying a reactor coolant pump.

12.7.3.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.3.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

   1) Low flow rate in one primary loop and P8 signal.

   2) Overtemperature ΔT.

b) The ASG [EFWS] is actuated by the “SG level (wide range) low 2” signal.

c) The VDA [ASDS] is actuated by the “SG pressure high 1” signal.

d) After reactor trip:

   1) Turbine trip;

   2) The ARE [MFFCS] full load lines are isolated;

   3) The RCP [RCS] heat is removed by the VDA [ASDS];

   4) The SGs are fed by the ASG [EFWS].

12.7.3.1.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

At the initial moment, one reactor coolant pump is assumed to fail. The total core flowrate decreases gradually over time. RT is caused by “Low flow rate in one primary loop” signal.

After RT, the RCCAs drop into the core and the core power decreases dramatically. Turbine trip and the ARE [MFFCS] full load isolation are initiated by the RT signal. The secondary circuit pressure is limited by the VDA [ASDS] if the GCT [TBS] is unavailable. Then, the ASG [EFWS] is actuated by the "SG level (wide range) low 2" signal. During this period, the VDA [ASDS] may be unable to remove all the residual heat from the core. The PSVs may open to limit the increase in primary pressure, if the normal spray is unavailable.

Finally, the controlled state is reached with a reduced core coolant flow.

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.3.1.5 Analysis Assumptions

The detailed assumptions are presented in Reference [20]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition.

b) The loss of one reactor coolant pump is assumed at time zero.

c) The Doppler power coefficient is considered as its maximum absolute value; the moderator density coefficient is considered as its minimum absolute value.

d) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

e) The single failure is applied on one "Low flow rate in one primary loop" channel.

12.7.3.1.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [20]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before the reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.4 Reactivity & Power Distribution Anomalies

12.7.4.1 Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at a Subcritical or Low Power Startup Condition (State A)

12.7.4.1.1 Initiating Event

An RCCA withdrawal event is defined as an uncontrolled addition of reactivity to the reactor core resulting in a power excursion.

Uncontrolled RCCA bank withdrawal at a subcritical or low power startup condition may be caused by:

a) Electrical failure of RCCA control system;

b) Spurious actions of the operator;

c) Spurious I&C signal on RCCA control system.

12.7.4.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.4.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of the “High neutron flux (power range, low setpoint)” signal.

b) After reactor trip:

   1) The RCP [RCS] heat is removed by the VDA [ASDS];

   2) The SGs are fed by the ASG [EFWS].

12.7.4.1.1 Typical Events Sequences

a) From the Initiating Event to the Controlled State

At the start of the transient, the two banks with the highest worth are fully inserted and assumed to be withdrawn simultaneously at the maximum speed. The core reactivity is uncontrollably increased, causing an abnormal power distribution.

With the continuous reactivity insertion, the neutron flux rises rapidly until the Doppler negative feedback stops it. This self-limitation of the power excursion is of primary importance because it limits the power during the delay time for protection actions. The reactor trip is triggered by “high neutron flux (power range, low setpoint)” signal.

Finally, the controlled state is reached. The residual heat is removed by the VDA

[ASDS] of all SGs and the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.4.1.2 Analysis Assumptions

The detailed assumptions are presented in Reference [21]. The main assumptions are listed as follows:

a) The two banks with the highest worth are fully inserted and assumed to be withdrawn at the maximum speed (72 steps/minute). Under the assumed condition, the uncontrolled withdrawal of RCCA banks induces a very conservative reactivity insertion rate.

b) The delayed neutron fraction and the prompt neutron lifetime used in the analysis are set to the maximum envelope values, i.e. 750 pcm and 31 μs respectively, to ensure the maximum energy stored in the fuel pellet.

c) The Doppler power coefficient is considered as its minimum value to maximize the peaking nuclear power.

d) The axial power distribution greatly tilting towards the core bottom is selected, maximizing the differential worth of the withdrawn RCCAs as well as the axial peaking power factor.

e) The cladding-pellet gap heat transfer coefficients are assumed to be the minimum value to maximize the fuel pellet temperature.

f) The most pessimistic position of the withdrawal banks between the full insertion and total withdrawal is used in the $F_{\Delta H}$ calculation. The $F_{\Delta H}$ is assumed to remain unchanged and to equal the maximum value during the RCCA bank withdrawal.

g) The single failure is applied on one "High neutron flux (power range, low setpoint)" channel.

12.7.4.1.3 Results and Conclusions

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [21]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. The maximum fuel temperature is lower than the fuel melting temperature.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.4.2 RCCA Bank Withdrawal at Power (State A)

12.7.4.2.1 Initiating Event

This event is defined as a continuous uncontrolled RCCA bank withdrawal at power. The insertion of positive reactivity results in an increase in the core power; before the secondary circuit pressure reaches the setpoint of the relief valve or safety valve, the heat removed from the SGs lags behind the increase in core power. Therefore, there is a significant increase in the reactor coolant temperature and pressure. If the event is not controlled manually or automatically, the DNB may be initiated by the power mismatch between the primary and secondary circuits as well as the resulting increase in primary average coolant temperature.

RCCA bank withdrawal at power may be caused by:

a) The failure of Control Rod Drive Mechanism (CRDM);

b) Spurious actions of the operator;

c) Spurious I&C signal.

12.7.4.2.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.4.2.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

   1) High neutron flux (power range, high setpoint);

   2) Overtemperature ΔT;

   3) Overpower ΔT;

   4) Pressuriser pressure high 2;

   5) Pressuriser level high 1;

   6) High positive neutron flux rate.

b) The PSVs are opened when the PZR pressure reaches the setpoint.

c) After reactor trip:

   1) Turbine trip;

   2) The ARE [MFFCS] full load lines are isolated;

   3) The RCP [RCS] heat is removed by the VDA [ASDS];

4) The SGs are fed by the ASG [EFWS].

12.7.4.2.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

The induced reactivity insertion results in an increase in the core heat flux. The heat extraction from the SGs lags behind the core power generation. The pressure and coolant temperature of the primary and secondary circuit increases until reactor trip. Before reactor trip, the PSVs may be opened to limit the RCP [RCS] pressure and the VDA [ASDS] may be opened to limit the secondary circuit pressure.

For high reactivity insertion rates, RT is initiated by the "High neutron flux (power range, high setpoint)" signal. The neutron flux in the core rises rapidly for these insertion rates while core power and primary coolant temperature lag behind due to the thermal capacity of the primary circuit. With further decrease in reactivity insertion rate, the "Overtemperature ΔT" and "High neutron flux (power range, high setpoint)" trips become equally effective in terminating the transient. For further reduction in reactivity insertion rates, RT is initiated by the "Overtemperature ΔT" signal.

After RT, the controlled state is reached. Under this state, the residual heat is removed via the VDA [ASDS] of all SGs. And the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.4.2.5 Analysis Assumptions

The detailed assumptions are presented in Reference [22]. The main assumptions are listed as follows:

a) The initial power is assumed to be each 10% step from 10%FP to 100%FP inclusive.

b) The reactivity insertion rates (0.1 pcm/s ~ 100 pcm/s) assumed in the analysis envelop all possible conditions. The maximum positive reactivity insertion rate analysed is greater than that for the simultaneous withdrawal at maximum speed of the two control banks having the maximum combined worth and the maximum overlap.

c) Two sets of reactivity coefficients are considered for each reactivity insertion rate:

   1) Minimum reactivity feedback: The Doppler power coefficient is considered as its minimum absolute value; the moderator density coefficient is considered as its minimum absolute value.

   2) Maximum reactivity feedback: The Doppler power coefficient is considered as its maximum absolute value, the moderator density coefficient is considered as its maximum absolute value.

d) Pressuriser spray and PSVs which limit the RCP [RCS] pressure increase are available.

e) RT actuated by “Pressuriser pressure high 2”, “Pressuriser level high 1” and “Overpower ΔT” signals are not considered. These protection channels may be triggered under low initial power level and small reactivity insertion rate.

f) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

g) The single failure is applied on one “High neutron flux (power range, high setpoint)” channel or one “Overtemperature ΔT” channel.

12.7.4.2.6 Results and Conclusions

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [22]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before reactor trip, the nuclear power increases by a small amount from the initial moment, and a limited increase in the fuel temperature is induced. However, the fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.4.3 RCCA Misalignment up to Rod Drop without Limitation (State A)

12.7.4.3.1 Initiating Event

One or more RCCA(s) in one bank dropping into the core may cause a negative reactivity insertion, thus leading to a decrease in the primary average coolant temperature. If RT is not triggered, the core power decrease and primary-secondary power mismatch result in a thermal-hydraulic transient governed by the reactivity feedback and by the RCP [RCS] average coolant temperature control. The core power may return to the initial level and may exhibit an overshoot for a short period. The combination of the high power level and the distorted power distribution caused by the rod drop may lead to a DNB if the reactor core is not protected.

RCCA misalignment up to rod drop may be caused by:

a) Electrical failure of CRDM or mechanical failure of CRDM;

b) Spurious actions of the operator;

c) Spurious I&C signal.

12.7.4.3.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.4.3.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

  1) High negative neutron flux rate;

  2) Overtemperature ΔT.

b) After reactor trip:

  1) Turbine trip;

  2) The ARE [MFFCS] full load lines are isolated;

  3) The RCP [RCS] heat is removed by the VDA [ASDS];

  4) The SGs are fed by the ASG [EFWS].

12.7.4.3.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

At the beginning of the transient, the nuclear power drops rapidly. The RCP [RCS] pressures and the primary average coolant temperature decrease.

If the reactor trip is triggered, the power plant reaches the controlled state. Under this state, the residual heat is removed via the VDA [ASDS] of all SGs. And the feedwater is supplied by the ASG [EFWS].

For the transient without RT, the nuclear power increases to a new primary-secondary equilibrium if the control banks are in automatic control mode. If the reactivity inserted by the temperature control banks is sufficient, the core power may return to the initial level and might exhibit an overshoot for a short period. If the rod control system is in the manual control mode, the core power might return monotonously to a new balance, and the primary coolant temperature remains lower than the initial value. The reactor can be tripped by the operator manually, and the reactor can be stabilised in the hot shutdown state after reactor trip.

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.4.3.5 Analysis Assumptions

The detailed assumptions are presented in Reference [23]. The main assumptions are listed as follows:

a) Initial event is considered in the full power condition.

b) The Doppler power coefficients and the moderator density coefficients are considered as the maximum absolute values for the cases which are used to verify whether a RCCA drop is detectable;

c) For transient analysis, the Doppler power coefficients and the moderator density coefficients are considered as the minimum absolute valves;

d) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

e) The single failure is applied on one “high positive neutron flux rate” channel.

f) Drop cases of one to four RCCA(s) are considered in transient analysis.

12.7.4.3.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [23]) shows that the minimum DNBR is greater than the design limit which is described in Sub-section 5.6.2.1. The nuclear power increases slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.4.4 Startup of an Inactive Reactor Coolant Loop at an Improper Temperature (State A)

Operation with one pump out of service is prevented through use of administrative controls if the reactor is at power or at hot standby. The reactor is automatically tripped if nuclear power is higher than 30% of nominal power (permissive signal P8). However, in this event, the Technical Specifications require a rapid return to the hot shutdown state for all power levels. In hot shutdown state, there is no temperature difference between the loops. Therefore, this event would not have any consequence for safety.

12.7.4.5 Chemical and Volume Control System (RCV [CVCS]) Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (State A to E)

12.7.4.5.1 Description

The boron dilution accident is hypothesised to occur as a result of operator error or failure of a component in the REA [RBWMS] or in the RCV [CVCS].

The RCV [CVCS] can increase core reactivity by injecting water with a lower boron concentration into the RCP [RCS] using the REA [RBWMS].

Uncontrolled boron dilution can potentially lead to unintentional criticality if it occurs during shutdown, or to DNB, if it occurs during power operation.

12.7.4.5.2 Acceptance Criteria

The boron dilution accident is considered as a DBC-2 event (an incident of moderate frequency). The acceptance criteria are:

a) At power operation for manual control operating mode, the operator has more than 30 minutes to perform dilution source isolation before the reactor core is returned to criticality;

b) At power operation for automatic control operating mode, the maximum allowable delay time after the isolation signal is generated shall be larger than the delay of the isolation used in the accident analysis (the delay of this isolation is equal to { } seconds);

c) In all shutdown states, the core sub-criticality still remains after reactor trip and isolation of dilution source is achieved;

d) At power operation, the DNBR shall be greater than the design limit.

12.7.4.5.3 Protection against the Fault

a) Precautions to reduce the accident probability

The "Low volume control tank level" signal triggers the makeup action. This signal actuates a boric acid transfer pump and a makeup water pump, opens boric acid and demineralised water makeup isolation valves, actuates valve regulators to control boric acid and demineralised water flow rates and controls the injection into the reactor coolant system. The makeup stops when the low volume control tank level signal disappears.

Other possible dilution methods for the boron and demineralised water makeup system are as follows:

1) Dilution

The operator shall perform the following actions:

- Selecting “Dilution” for the REA [RBWMS] operation mode;

- Setting the volume of water to be drawn;

- Setting the setpoint of demineralized water flow rate;

- Giving a startup command.

After the required water volume is reached, the dilution will stop automatically or be stopped by an operator.

2) Manual makeup

The operator shall perform the following actions:

- Selecting “Manual” for the REA [RBWMS] operation mode;

- Setting demineralised water and boric acid flow rates;

- Setting demineralised water and boric acid volumes;

- Giving a startup command.

After the required water and boric acid volumes are reached, the makeup will stop automatically or be stopped manually.

Considering numerous independent and diverse operations, there is few possibilities that accidental boron dilution happens.

b) Management procedure

The management procedures decide the RCCA bank position under different initial conditions, and also ensure that the valves of the dilution subsystems which are not required remain closed or locked during plant operation.

c) Automatic protection measures

During refuelling and steam generator inspection, the coolant is supplied by the IRWST. The reactor coolant system is isolated from the source of non-borated water so the boron dilution transient is not considered under these two modes of operation.

There are two ways that the plant can be operated at power: automatic T_{avg}/rod control and manual control.

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With the reactor in manual control, dilution causes a slow reactivity insertion which leads to the reactor power level and the average coolant temperature rising slowly, finally triggering over-temperature ΔT protection or over-power ΔT protection.

In all shutdown states, the reactor core is protected by the source range high neutron flux signal. This signal leads to the reactor trip, the isolation of the RCV [CVCS] system to cut off the dilution source, and the injection of higher-concentration borated water from the IRWST automatically. This anti-reactivity insertion has to be sufficient to avoid the return to criticality after the effective isolation of dilution.

12.7.4.5.4 Analysis Method

a) Methods

To cover all plant operation conditions, the uncontrolled boron dilution in the following operation modes is analysed:

1) Shutdown;
   - Normal cold shutdown;
   - Hot shutdown;
   - Transition from hot shutdown to cold shutdown.
2) 100% power operation in the automatic control mode and the manual control mode.

b) Analysis assumptions

The detailed assumptions are presented in Reference [24].The beginning of the cycle is considered because of the high initial boron concentration. Main assumptions in this mode are:

1) Uncontrolled boron dilution during shutdown;
   - The technical specification stipulates that for the withdrawal of certain RCCA banks, the initial boron concentration shall correspond to the minimum shutdown margin required during the withdrawal of these banks;
   - The mass of borated water in the reactor coolant system is 255 t (normal cold shutdown) and 183 t (hot shutdown). Reactor coolant pumps or residual heat removal system pumps ensure the continuous blending in the reactor pressure vessel;
   - There is a dilution flow of 87 t/h in all shutdown states;

- The most reactive RCCA is assumed to be stuck out of core;
- Only the signal of "High neutron flux (source range)" triggering the reactor trip is taken into consideration.

2) Uncontrolled boron dilution during the power operation

- The reactivity insertion rate depends on the coolant temperature and the initial boron concentration. The most conservative condition is that power operation occurs with a high initial boron concentration.
- The mass of borated water in the reactor coolant system is 178 t and all the reactor coolant pumps are working to ensure that the boric acid and water are fully mixed.
- There is a dilution flow of 87 t/h in manual control mode and 84.18 t/h in automatic control mode.
- For manual control mode, conservative values of high reactor coolant system critical boron concentrations, high boron worth and minimum shut down margins are used.

12.7.4.5.5 Results and Conclusions

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With the reactor in manual control, the operator has more than 30 minutes to perform dilution source isolation before the reactor core is returned to criticality.

During power operation, the DNBR is always greater than the design limit.

In all shutdown states, the core sub-criticality is retained after reactor trip and isolation of dilution source.

The analysis results show that for all operation modes, no violation of acceptance criteria occurs. Therefore, the uncontrolled dilution will not cause damage to the fuel because the protection system and operator have enough time to react. The operator has sufficient information to identify the event and sufficient time to take measures.

12.7.5 Increase in Reactor Coolant System Inventory

12.7.5.1 RCV [CVCS] Malfunction Causing an Increase in (RCP [RCS]) Inventory (State A)

12.7.5.1.1 Initiating Event

An RCV [CVCS] malfunction causing an increase in RCP [RCS] inventory may result in the increase in pressuriser pressure and level. The cooling down of the RCP [RCS] causes a core power increase due to moderator feedback.

The RCV [CVCS] malfunction causing an increase in reactor coolant inventory may be caused by:

a) Control failure of high pressure control valves in the RCV [CVCS] high pressure letdown line;

b) Control failure of flow control valves in the RCV [CVCS] charging line;

c) Control failure of flow control valves in the seal injection line;

d) Inadvertent closure of isolation valves in the RCV [CVCS] high pressure letdown line;

e) Spurious activation of the RCV [CVCS] charging pump.

12.7.5.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.5.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

   1) Pressuriser pressure high 2;

   2) Pressuriser level high 1.

b) When the pressuriser level reaches the “Pressuriser level high 1” setpoint, the RCV [CVCS] charging line is isolated.

c) When the “Pressuriser level high 2” setpoint is reached, the SWI for the RCPs is isolated.

d) The PSVs are opened when the PZR pressure reaches the setpoint.

e) After reactor trip:

   1) Turbine trip;

   2) The ARE [MFFCS] full load lines are isolated;

3) The RCP [RCS] heat is removed by the VDA [ASDS];

4) The SGs are fed by the ASG [EFWS].

12.7.5.1.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

Here, the PZR level increases. If the PZR level control fails to maintain the water level, RT is triggered on “Pressuriser level high 1” signal or “Pressuriser pressure high 2” signal.

Turbine trip and the ARE [MFFCS] full load isolation are initiated on receipt of the RT signal. The secondary pressure is limited by the VDA [ASDS] if the GCT [TBS] is unavailable. The RCV [CVCS] charging line is isolated following the “Pressuriser level high 1” signal. The SWI for RCPs is isolated on “Pressuriser level high 2” signal. The PSVs may open if the normal spray is unable to limit the increase in primary pressure.

Finally, the controlled state is reached. The residual heat is removed by the VDA [ASDS] of all SGs and the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.5.1.5 Analysis Assumptions

The detailed assumptions are presented in Reference [25]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition.

b) The Doppler power coefficient is considered as its minimum absolute value, the moderator density coefficient is considered as its maximum absolute value.

c) The PZR pressure control is considered, because it limits the increase in primary pressure. The maximum spray capacity is considered.

d) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

e) The single failure is applied on one “Pressuriser level high 1” channel.

12.7.5.1.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [25]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before the reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.6 Decrease in Reactor Coolant System Inventory

12.7.6.1 RCV [CVCS] Malfunction Causing a Decrease in (RCP [RCS]) Inventory (State A)

12.7.6.1.1 Initiating Event

This event induces an RCP [RCS] pressure decrease and may result in inadequate cooling of the fuel cladding through a DNB.

The RCV [CVCS] malfunction causing a decrease in reactor coolant inventory may be caused by:

a) Failure of the pressuriser level control channel;

b) Malfunction of one or two HP reducing station valves;

c) Spurious isolation of the RCV [CVCS] charging line;

d) Spurious shutdown of the RCV [CVCS] charging pump.

12.7.6.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.6.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

    1) Pressuriser pressure low 2;

    2) Overtemperature ΔT.

b) The RCV [CVCS] letdown line is isolated when the “Pressuriser level low 1” and RT signals occur.

c) After reactor trip:

    1) Turbine trip;

    2) The ARE [MFFCS] full load lines are isolated;

    3) The RCP [RCS] heat is removed by the VDA [ASDS];

    4) The SGs are fed by the ASG [EFWS].

12.7.6.1.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

After the RCV [CVCS] malfunction, the reactor coolant inventory begins to decrease, When the "Pressuriser pressure low 2" setpoint or "Overtemperature ΔT" setpoint is reached, the reactor trip is triggered to protect the core.

Turbine trip and the ARE [MFFCS] full load isolation are initiated on RT signal. The secondary pressure is limited by the VDA [ASDS] if the GCT [TBS] is unavailable. When the "Pressuriser level low 1" and RT signal occurs, the RCV [CVCS] letdown line is isolated.

Finally, the controlled state is reached. The residual heat is removed by the VDA [ASDS] of all SGs and the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.
2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.
3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.6.1.5 Analysis Assumptions

The detailed assumptions are presented in Reference [26]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition.
b) The Doppler power coefficient is considered as its maximum absolute value, the moderator density coefficient is considered as its minimum absolute value.
c) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.
d) The single failure is applied on one "Pressuriser pressure low 2" channel.

12.7.6.1.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [26]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before the reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.6.2 Uncontrolled RCP [RCS] Level Drop in Shutdown States with RIS [SIS] Connected in RHR Mode (State C\D)

12.7.6.2.1 Initiating Event

This event reduces the RCP [RCS] water level which may lead to the trip of the RIS [SIS] pumps. Therefore, the RCP [RCS] cooling may not be ensured.

An uncontrolled RCP [RCS] level drop in a shutdown state with the RIS [SIS] connected in RHR mode may be caused by:

a) Spurious operator action;

b) An RCP [RCS] level control malfunction;

c) A malfunction of the RCV [CVCS] low pressure reducing station valve.

12.7.6.2.2 Acceptance Criteria

The acceptance criteria to be considered for this event are:

a) Core water inventory is stable;

b) Residual heat removal is ensured on a long term basis.

12.7.6.2.3 Main Safety Functions

The reactor is protected by the following mitigation actions:

a) The safety injection is actuated by the “RCP [RCS] loop level low 1” signal.

b) The RIS [SIS] pumps are tripped by the “RCP [RCS] loop level low 2” signal.

c) Isolation of the RCV [CVCS] letdown line is initiated by the SI signal.

12.7.6.2.4 Typical Events Sequences

At the start of the transient, the RCP [RCS] level begins to drop. The safety injection is actuated on receipt of “RCP [RCS] loop level low 1” signal. The RCV [CVCS] letdown line is isolated by the SI signal. The primary water inventory drop is stopped before the RIS [SIS] pumps trip threshold is reached. Then the safety state is reached.

12.7.6.2.5 Analysis Assumptions

The detailed assumptions are presented in Reference [27]. The main assumptions are listed as follows:

a) The plant operates in RCP [RCS] ¾ loop level condition.

b) The primary average coolant temperature is considered as its maximum value.

c) The single failure is applied on one MHSI pump.

d) The maximum RCV [CVCS] letdown flowrate is 41 kg/s.

12.7.6.2.6 Result and Conclusion

The detailed analysis of this fault [27] shows that the primary water inventory can be restored and the RIS [SIS] pumps are maintained in operation.

Thus the acceptance criteria for this event are met.

12.7.7 Increase/Decrease in Reactor Coolant System Pressure

12.7.7.1 Spurious Pressuriser Heater Operation (State A)

12.7.7.1.1 Initiating Event

A spurious pressuriser heating event leads to a pressure increase in the RCP [RCS].

Spurious pressuriser heating can be caused by:

a) Spurious I&C actuation signal on the heaters;

b) An automatic pressure control malfunction;

c) Spurious operator action.

12.7.7.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.7.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

   1) Pressuriser pressure high 2;

   2) Overtemperature ΔT.

b) After reactor trip:

   1) Turbine trip;

   2) The ARE [MFFCS] full load lines are isolated;

   3) The RCP [RCS] heat is removed by the VDA [ASDS];

   4) The SGs are fed by the ASG [EFWS].

12.7.7.1.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

If a spurious pressuriser heating event occurs, the PZR pressure control system automatically switches off the heaters, and opens the normal spraying valves. The pressuriser pressure can be stabilised around its setpoint. No protection signal is actuated and the pressuriser safety valves are not open.

If the PZR heaters are not switched off automatically or manually, RT is actuated by the “Pressuriser pressure high 2” signal or “Overtemperature ΔT” signal.

After RT, the RCCAs drop into the core and the core power decreases dramatically.

Turbine trip and the ARE [MFFCS] full load isolation are initiated by RT signal. The secondary circuit pressure is limited by the VDA [ASDS] if the GCT [TBS] is unavailable. Then, the ASG [EFWS] is actuated by the “SG level (wide range) low 2” signal. The PSVs may open if the normal spray is unable to limit the increase in primary pressure.

Finally, the controlled state is reached. The residual heat is removed by the VDA [ASDS] of all steam generators and the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.7.1.5 Analysis Assumptions

The detailed assumptions are presented in Reference [28]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition.

b) The Doppler power coefficient is considered as its maximum absolute value, the moderator density coefficient is considered as its minimum absolute value.

c) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

d) The single failure is applied on one “Pressuriser pressure high 2” channel.

12.7.7.1.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [28]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature increases slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.7.2 Spurious Pressuriser Spray Operation (State A)

12.7.7.2.1 Initiating Event

A spurious pressuriser spraying event induces a pressure decrease on the primary circuit and might lead to inadequate cooling of the fuel cladding.

A spurious pressuriser spraying event can be caused by:

a) Spurious opening of a normal spray control valve;

b) Spurious opening of an auxiliary spray control valve;

c) An automatic pressure control malfunction;

d) Spurious operator action.

12.7.7.2.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.7.7.2.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated on receipt of any of the following signals:

   1) Pressuriser pressure low 2;

   2) Overtemperature ΔT.

b) After reactor trip:

   1) Turbine trip;

   2) The ARE [MFFCS] full load lines are isolated;

   3) The RCP [RCS] heat is removed by the VDA [ASDS];

   4) The SGs are fed by the ASG [EFWS].

12.7.7.2.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

In this event, the RCP [RCS] pressure decreases. The pressure control system automatically switches off both normal and auxiliary spraying valves, and switches on the heaters. The pressuriser pressure can be stabilised around its setpoint. No protection signal is actuated.

If the pressure decrease cannot be stopped by the PZR pressure control system, RT is triggered by “Pressuriser pressure low 2” signal or “Overtemperature ΔT” signal.

After RT, the RCCAs drop into the core and the core power decreases dramatically. Turbine trip and the ARE [MFFCS] full load isolation are initiated by the RT signal. The secondary circuit pressure is limited by the VDA [ASDS] if the GCT [TBS] is unavailable. Then, the ASG [EFWS] is actuated by the “SG level (wide range) low 2” signal.

Finally, the controlled state is reached. The residual heat is removed by the VDA [ASDS] of all steam generators and the feedwater is supplied by the ASG [EFWS].

b) From the Controlled State to the Safe State

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted.

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.7.7.2.5 Analysis Assumptions

The detailed assumptions are presented in Reference report [29]. The main assumptions are listed as follows:

a) The initial event is considered under full power condition.

b) The Doppler power coefficient is considered as its maximum absolute value, the moderator density coefficient is considered as its minimum absolute value.

c) The RCCA having the highest worth is assumed to be stuck out of the core to minimise the negative reactivity insertion after the reactor trip. The most conservative negative reactivity insertion curve as a function of time is used.

d) The single failure is applied on one “Pressuriser pressure low 2” channel.

12.7.7.2.6 Result and Conclusion

a) From the Initiating Event to the Controlled State

The detailed analysis of this fault (see Reference [29]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. Before the reactor trip, the nuclear power changes slightly from the initial moment, and the fuel temperature changes slightly. The fuel melting temperature limit is not challenged.

Thus the acceptance criteria for this event are met.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.7.8 Fuel Pool Accidents

12.7.8.1 Loss of One Train of the Fuel Pool Cooling and Treatment System (PTR [FPCTS]) or of a Supporting System (State A)

12.7.8.1.1 Initiating Event

This event may cause inadequate cooling of the fuel assemblies in SFP.

Loss of one train of the PTR [FPCTS] or of a supporting system may be caused by:

a) PTR [FPCTS] system failure;

b) Loss of cooling chains;

c) Loss of water intake.

12.7.8.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are:

a) The SFP water average temperature shall remain lower than 80℃;

b) All the fuel assemblies shall be covered with water.

SFP subcritical criteria can be met by the design of the fuel assembly storage grid.

12.7.8.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

Each PTR [FPCTS] train is designed to have enough capacity to remove residual heat from the SFP.

12.7.8.1.4 Typical Events Sequences

Generally, one PTR [FPCTS] train is in operation during State A, the other two are backups. After loss of the operating train, the other PTR [FPCTS] trains begin to start up. However, one backup train is not considered as this is based on the single failure assumption. The third train can remove the residual heat from the SFP.

If one PTR [FPCTS] train is in maintenance during State A, the other two shall be in operation. After loss of one operating train of the PTR [FPCTS], the last train is able to remove the residual heat from the SFP.

12.7.8.1.5 Analysis Assumptions

The detailed assumptions are presented in Reference [30]. The main assumptions are listed as follows:

a) Conservative fuel decay heat is considered.

b) SFP and PTR [FPCTS] pipes are considered as adiabatic.

c) All decay heat released from the spent fuel is absorbed by the water in the SFP.

d) Single failure is applied on one PTR [FPCTS] train.

12.7.8.1.6 Result and Conclusion

The detailed analysis of this fault (see Reference [30]) shows that the water temperature of the SFP does not exceed the limit, and all the fuel assemblies are covered with water during the transient.

Thus the acceptance criteria for this event are met.

12.8 Analyses of DBC-3 events

12.8.1 Increase in Heat Removal by the Secondary System

12.8.1.1 Inadvertent Opening of an SG Relief Train or of a Safety Valve (State A)

12.8.1.1.1 Initiating Event

The inadvertent opening of an atmospheric steam dump train (VDA [ASDS]) or of a safety valve during power operation can be caused by the spurious opening of a VDA isolation valve, spurious opening of a main steam safety valve or a main steam safety valve seizing open. This is defined as a DBC-3 condition, and leads to an increase in the heat removal from the RCP [RCS]. The core power increases due to the negative moderator temperature coefficient. This could lead to the occurrence of DNB and subsequently to fuel damage.

The inadvertent opening of a VDA [ASDS], which is the valve with the maximum discharge capacity in the main steam line, is the worst-case condition. This is, therefore, simulated in the accident analysis.

12.8.1.1.2 Acceptance Criterion

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.8.1.1.3 Main Safety Functions

Any of the following signals can trigger the reactor trip to maintain core sub-criticality:

a) High neutron flux (power range, high setpoint);

b) Overpower ΔT;

c) Pressuriser pressure low 2;

d) Safety injection (SI) signal.

The safety injection system can be actuated by the “Pressuriser pressure low 3” signal.

Any of the following signals can lead to the quick closure of the MSIV to protect the RCP [RCS] against overcooling:

a) SG pressure low 1;

b) Pressure drop of SG high 1.

Any of the following signals can lead to ARE [MFFCS] isolation to protect the RCP [RCS] against overcooling:

a) ARE [MFFCS] full load lines isolated on the “RT” signal;

b) ARE [MFFCS] low load lines isolated on the “SG pressure low 2” signal;

c) ARE [MFFCS] low load lines isolated on the “Pressure drop of SG high 2” signal.

12.8.1.1.4 Typical Events Sequences

a) From the initiating event to the controlled state

The inadvertent opening of an atmospheric steam dump train (VDA [ASDS]) or a safety valve during power operation leads to a steam flow increase, and then to a steam flow reduction due to the decrease in steam pressure.

The increase in heat removal from the RCP [RCS] during the accident causes coolant temperature and pressure to decrease. Due to the negative moderator temperature coefficient, a decrease of RCP [RCS] temperature can induce positive reactivity insertion, resulting in a core power increase.

If the blowdown size is not large enough, this accident may not cause a RT, and the core power will remain at a higher level. The operator can identify this event via the “high core power” alarm signal. Once confirming the alarm, the operator will transition the reactor into the controlled state (i.e. hot showdown state).

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray and the PSVs can be used when the pressuriser sprays are unavailable.

12.8.1.1.5 Analysis Assumptions

a) Initial condition

1) The initial power is the nominal value plus uncertainty;

2) Initial average temperature of reactor coolant is the nominal value plus uncertainty;

3) Initial pressuriser pressure is the nominal value minus uncertainty;

4) Initial pressuriser level is the nominal value minus uncertainty.

b) Core parameters

1) The moderator density coefficient is considered as maximum absolute value;

2) The Doppler power coefficient is considered as minimum value.

c) Other assumptions

1) The analysis adopts the maximum heat exchange coefficient between the primary side and secondary side;

2) The steam flow passing through the opening valve for each step is calculated with an assumption where the back pressure is at atmospheric pressure.

d) Single failure

The single failure is applied to the “high neutron flux (power range, high setpoint)” signal (sensor or channel).

e) LOOP

The impact of LOOP is considered in this analysis.

12.8.1.1.6 Results and Conclusions

a) From the initiating event to the controlled state

The detailed analysis of this fault (see reference [32]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. The fuel temperature increase but does not challenge the fuel melting temperature limit.

Thus the acceptance criteria for this event are met.

b) From the controlled state to the safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.8.1.2 Small Steam System Piping Break Including Breaks in Connecting Lines (State A\B)

12.8.1.2.1 Initiating Event

The small steam system piping break (≤DN50), including break of connecting lines (≤DN50) in power operation state or cold shutdown state, is classified as a DBC-3 event.

12.8.1.2.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.8.1.2.3 Main Safety Functions

Any of the following signals can trigger the reactor trip to maintain core sub-criticality:

a) High neutron flux (power range, high setpoint);

b) Overpower ΔT;

c) Pressuriser pressure low 2;

d) Safety injection (SI) signal.

The safety injection system can be actuated by “Pressuriser pressure low 3” signal.

Any of the following signals can lead to the quick closure of the MSIV to protect the RCP [RCS] against overcooling:

a) SG pressure low 1;

b) Pressure drop of SG high 1.

Any of the following signals can lead to the ARE [MFFCS] isolation to protect the RCP [RCS] against overcooling:

a) ARE [MFFCS] full load lines isolated on “RT” signal.

b) ARE [MFFCS] low load lines isolated on “SG pressure low 2” signal;

ARE [MFFCS] low load lines isolated on “Pressure drop of SG high 2” signal.

12.8.1.2.4 Typical Events Sequences

a) From the initiating event to the controlled state

The Small steam system piping break including breaks in connecting lines (state A\B) leads to a steam flow increase, and then to a steam flow reduction due to the decrease in steam pressure.

The increase in heat removal from the RCP [RCS] during the accident causes coolant temperature and pressure to decrease. Due to the negative moderator temperature coefficient, a decrease of RCP [RCS] temperature can induce the positive reactivity insertion, and increase the core power.

If the break size is not large enough, this accident may not cause a RT, and the core power will remain at a higher level. The operator can identify this event via the “high core power” alarm signal. Once confirming the alarm, the operator will bring the reactor to the controlled state (i.e. hot showdown state)

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.8.1.2.5 Result and Conclusion

a) From the Initiating Event to the Controlled State

The results of this event can be bounded by the “Large steam system piping break” accident (DBC-4) in Sub-chapter 12.9.1.1.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.8.2 Decrease in Heat Removal by the Secondary System

12.8.2.1 Inadvertent Closure of All or One Main Steam Isolation Valves (State A)

12.8.2.1.1 Initiating Event

This event is defined as the inadvertently closure of all main steam isolation valves (MSIVs) on the steam line all at once. This accident is typically caused by a spurious Instrumentation and Control (I&C) closure signal on the four MSIVs. It is an overheating event, and it may cause the occurrence of DNB and result in fuel cladding failure.

The inadvertent closure of one MSIV is also typically caused by a spurious closure I&C signal, and it too is an overheating event, which can reduce the steam flowrate and the heat removal capability of the affected SG. Therefore, the pressure and temperature in the primary side of this SG increase, as well as the temperature of the corresponding loop.

12.8.2.1.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.8.2.1.3 Main Safety Functions

The following plant safety functions can mitigate the event:

a) RT is actuated by any of the following signals:

    1) SG pressure high 1;

    2) Pressuriser pressure high 2.

b) VDA [ASDS] is opened by the “SG pressure high 1” signal.

c) ASG [EFWS] is actuated by the “SG level (wide range) low 2” signal.

d) PSVs are opened when the PZR pressure reaches the setpoints.

e) Turbine trip on receipt of RT signal.

12.8.2.1.4 Typical Events Sequences

a) From the initiating event to the controlled state

During the transient, the primary pressure and secondary pressure increase gradually, and RT can be triggered by the “SG pressure high 1” signal or “Pressuriser pressure high 2” signal.

After RT, residual heat is removed by the VDA [ASDS] of all available SGs. If the ARE [MFFCS] is not available, the feedwater supply is ensured by ASG [EFWS]. The RCP [RCS] will remain stable and the controlled state is reached.

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.8.2.1.5 Analysis Assumptions

a) Initial state

1) The initial power is set to the full power plus uncertainty.

2) The initial coolant temperature is set to nominal value plus uncertainty.

3) The initial PZR pressure is set to nominal value minus uncertainty.

4) The initial flow is set to the thermal-hydraulic design flow.

b) Analysis conditions

The analysis only considers the process in the short-term phase, and includes following two cases:

1) Case 1: all main steam isolation valves are inadvertently closed;

2) Case 2: one main steam isolation valve is inadvertently closed.

c) LOOP

LOOP is assumed to occur at the time of turbine trip.

d) Single failure

The single failure is applied to one “SG pressure high 1” signal (sensor or channel).

12.8.2.1.6 Result and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see reference [33]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. The fuel temperature increases but does not challenge the fuel melting temperature limit.

Thus the acceptance criteria for this event are met.

b) From the initiating event to the controlled state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.8.2.2 Long Term LOOP (> 2 hours) (State A)

12.8.2.2.1 Initiating Event

LOOP leads to the loss of power supply to all RCPs, feed water pumps and condensate pumps. Because of the decrease in reactor coolant flow and the reduction of heat removal capacity of the secondary circuit, the capacity of the primary coolant to remove the core heat decreases, potentially causing DNB and insufficient cooling of the fuel cladding.

The loss of offsite power may be caused by:

a) A complete loss of offsite grid;

b) An onsite AC power distribution system failure;

c) An external grid disturbance (dropped voltage or frequency).

This sub-chapter addresses the analysis of a long term loss of offsite power accident.

12.8.2.2.2 Acceptance Criteria

The challenge to fuel integrity in a long term LOOP (> 2 hours) accident is considered in the analysis of short term LOOP (< 2 hours) in Sub-chapter 12.7.2.3. Thus this event is analysed with respect to the following decoupling criteria: The plant can be taken to, and maintained in, safe state.

12.8.2.2.3 Main Safety Functions

The following safety functions are capable of bringing the plant to the safe state:

a) RT can be actuated by the “Low RCP speed” signal or “Low flow rate” signal;

b) VDA [ASDS] is actuated by the “SG pressure high 1” signal;

c) ASG [EFWS] is actuated by the “SG level (wide range) low 2” signal;

d) PSVs are opened when the PZR pressure reaches the setpoint.

12.8.2.2.4 Typical Events Sequences

a) From the initiating event to the controlled state

The LOOP leads to a reactor coolant flow decrease, and then the reduction of heat removal from the RCP [RCS] during the accident causes coolant temperature and pressure increases. After the LOOP, the EDGs come into service, supporting the operation of the main systems related to the automatic protection functions which are discussed in Sub-section 12.8.2.2.3.

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to

the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the PSVs.

12.8.2.2.5 Analysis Assumptions

a) Initial state

1) The initial power is set to the full power plus uncertainty.

2) The initial coolant temperature is set to nominal value plus uncertainty.

3) The initial PZR pressure is set to nominal value plus uncertainty.

4) The initial flow is set to the thermal-hydraulic design flow.

5) The initial PZR level is set to nominal value plus uncertainty.

6) The initial SG level is set to nominal value minus uncertainty.

b) Single failure

One EDG failure is taken into account.

12.8.2.2.6 Result and Conclusion

a) From the initiating event to the controlled state

The results of this fault are showed in the analysis of short term LOOP (< 2 hours) in Sub-chapter 12.7.2.3.

b) From the controlled state to the safe state

The description which is shown in Sub-chapter 12.10 demonstrates that the plant can be taken to safe state.

12.8.2.3 Small Feedwater System Piping Break Including Breaks in Connecting Lines to SG (State A\B)

12.8.2.3.1 Initiating Event

This event is defined as the small break occurring in feedwater system pipeline or in an SG connecting pipeline (DN<50mm). Depending upon the break size and the plant operating conditions, the event could cause either an RCP [RCS] overcooling event (by excessive steam discharge through the break) or an RCP [RCS] overheating event (by excessive liquid discharge through the break). Only the RCP [RCS] heat-up effects are evaluated in this section, as the "Large Steam System Piping Break" analysis can cover the effects of RCP [RCS] overcooling.

12.8.2.3.2 Acceptance Criteria

The small feedwater system piping break is classified as a DBC-3 event.

The aim of the study is to demonstrate that the core and reactor coolant system integrity are maintained during the transient.

12.8.2.3.3 Main Safety Functions

a) Any of the following signals can trigger the reactor trip to maintain core sub-criticality:

    1) Pressuriser pressure high 2;

    2) SG level (narrow range) low 1;

    3) Overtemperature ΔT;

    4) Pressuriser pressure low 2.

b) ASG [EFWS] is actuated by the "SG level (wide range) low 2" signal;

c) VDA [ASDS] is opened by the "SG pressure high 1" signal.

12.8.2.3.4 Typical Events Sequences

a) From the initiating event to the controlled state

At the beginning of the transient, the affected SG feedwater flowrate will decrease due to the feedwater line break and thus the affected SG level will decrease, which reduces the capacity of heat removal from the RCP [RCS]. Therefore, the RCP [RCS] pressure and temperature will increase.

The affected SG level will continue to decrease and once the level reaches to the setpoint of the "SG level (narrow range) low 1" signal, the RT will be triggered, resulting in the turbine trip and main feedwater isolation. When the SG level decreases to the setpoint of "SG level (wide range) low 2", the ASG [EFWS] will be activated to increase the SG water level. Meanwhile, the RCP [RCS] decay heat will be removed by the VDA [ASDS]. Therefore, the reactor can be taken to the controlled state.

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to

the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray and the PSVs can be used when the pressuriser sprays are unavailable.

12.8.2.3.5 Analysis Assumptions

a) Initial state

1) The initial power is set to the full power plus uncertainty;

2) Initial average temperature of reactor coolant is set to the nominal value plus uncertainty;

3) Initial pressuriser pressure is set to the nominal value minus uncertainty;

4) Initial pressuriser level is set to the nominal value minus uncertainty;

5) The initial SG level is set to the nominal value minus uncertainty.

b) Single failure

One ASG [EFWS] is assumed to be unavailable. Moreover, the ASG [EFWS] flow is set to the minimum value.

c) LOOP

The impact of the LOOP is considered in this analysis.

d) Other assumptions

1) After the reactor coolant pumps trip caused by LOOP due to turbine trip, heat transfer is ensured by natural circulation;

2) No credit is taken for heat transfer to metal of reactor coolant system during the reactor coolant system heat-up;

3) Only one ASG [EFWS] pump supplies water to one unaffected SG with a minimum flowrate.

12.8.2.3.6 Results and Conclusions

a) From the initiating event to the controlled state

The results of this event can be bounded by the "Large feedwater system piping break" accident (DBC-4) in Sub-chapter 12.9.2.1.

b) From the controlled state to the safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.8.3 Decrease in Reactor Coolant System Flowrate

12.8.3.1 Forced Reduction in Reactor Coolant Flow (3 Pumps) (State A)

12.8.3.1.1 Initiating Event

The forced reduction in reactor coolant flow is caused by a simultaneous loss of electrical supplies to all the reactor coolant pumps. A fast decrease in the off-site grid supply frequency can lead to a reversal of motor torque and thus to decrease the reactor coolant pumps speed, which rapidly reduces the coolant flow.

The decrease in reactor coolant flow causes a rapid increase in coolant temperature and pressure, potentially resulting in DNB and subsequent fuel damage.

12.8.3.1.2 Acceptance Criteria

The forced reduction in reactor coolant flow is classified as a DBC-3 event.

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.8.3.1.3 Main Safety Functions

Any of the following signals can trigger the reactor trip:

a) For nuclear power higher than 10%FP:

   1) Low flow rate in two primary loops;

   2) Opening of two primary coolant pump breakers;

   3) Low RCP speed in two loops.

b) For nuclear power higher than 30%FP

   1) Low flow rate in one primary loop;

   2) Opening of one primary coolant pump breaker.

12.8.3.1.4 Typical Events Sequences

a) From the initiating event to the controlled state

After a frequency drop on the off-site grid, the speed of the RCPs decreases, and when it reaches the threshold of “Low RCP speed” signal, RT is triggered and then the turbine trips automatically. After this, the ARE [MFFCS] full load lines are automatically closed.

The EDGs come into service following the failure of the off-site grid to support the operation of the main systems related to the automatic protection functions.

During the transient, the ASG [EFWS] can be actuated when SG level reaches the setpoint of the “SG level (wide range) low 2” signal, and the PSVs will open when the pressuriser pressure exceeds the required thresholds. Moreover, the VDA [ASDS] will automatically open if the secondary pressure exceeds its threshold. Therefore, the reactor can be maintained in the controlled state.

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.8.3.1.5 Analysis Assumptions

a) Initial conditions

1) The initial operating power is the nominal value.

2) The initial average coolant temperature is the nominal value.

3) The reactor coolant flowrate is the thermal-hydraulic design flowrate.

4) The initial pressuriser pressure is the nominal value.

b) Single failure

The single failure is set so that one of the measurement channels for “Low RCP speed” is invalid.

c) LOOP

The impact of the LOOP is considered in this analysis.

12.8.3.1.6 Results and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see Reference [34]) shows that the minimum DNBR is greater than the design limit which is described in Sub-section 5.6.2.1. Thus the acceptance criteria for this event are met.

b) From the controlled state to the safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.8.4 Reactivity & Power Distribution Anomalies

12.8.4.1 Inadvertent Loading of a Fuel Assembly in an Improper Position (State E)

The inadvertent loading of a fuel assembly in an improper position is defined as a DBC-3 event (infrequent event). One of the most important objectives in loading scheme design of a given cycle is to minimise the nuclear enthalpy rise factor $F_{\Delta H}$. However, the inadvertent loading may disturb the expected power distribution.

The inadvertent loading of fuel assembly includes one or more fuel assemblies placed in improper positions, one or more fuel rods made of pellets with improper enrichment or the whole fuel assembly made of pellets with improper enrichment during the manufacturing process. If the enrichment of the fuel assembly misplaced at a certain place in the core is higher than that of the expected fuel assembly, an increase in the neutron flux at the position will be caused. The possible inadvertent loading also includes no burnable poison rods loaded for the initial core in need.

The quality control process during the fuel manufactory process limits the possibility of fuel rods or fuel assembly made of pellets with improper enrichment, and also limits the possibility of burnable poison rods be placed in wrong place. To minimise the possibility of the inadvertent loading, each fuel assembly is marked with a code and loaded as the core loading chart indicates. During the loading, each assembly shall be checked for its identification number before being carried to the core. After the fuel assembly is placed in core, its number will be recorded in the loading chart so that the core will be checked further for its correctness after the loading.

The inadvertent loading of fuel assembly can cause significant changes in core power distribution. Various loading errors will increase the peak of power distribution significantly, and will be detected by the in-core neutron flux detector.

The purpose of the inadvertent loading analysis is to show that a loading error will not influence the safety operation of the plant for it can be found easily or has little impact on the nuclear enthalpy rise factor.

The following aspects shall be taken into consideration for the inadvertent loading scheme:

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For each possible inadvertent loading scheme, the following calculations are performed:

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The analyses show that inadvertent loading of a fuel assembly in an improper position will not threaten reactor core safety, since it either can be detected by detectors during the start-up due to great power variations or will not affect the core safety due to little disturbance.

12.8.4.2 Uncontrolled RCCA bank withdrawal (State B\C\D)

12.8.4.2.1 Initiating Event

This fault is defined as a continuous uncontrolled RCCA bank withdrawal during shutdown conditions. Due to the insertion of positive reactivity, there is a significant increase in the core power.

Uncontrolled RCCA bank withdrawal during shutdown conditions may be caused by:

a) Electrical failure of RCCA control system;

b) Human failure;

c) Spurious I&C signal from the RCCA control system.

12.8.4.2.2 Acceptance Criteria

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.8.4.2.3 Main Safety Functions

RT is actuated by the "High neutron flux (intermediate range)" signal or "High neutron flux (source range)" signals.

12.8.4.2.4 Typical Events Sequences

The uncontrolled withdrawal of RCCA banks induces a very conservative reactivity insertion rate. The RT is initiated by the "High neutron flux (intermediate range)" signal or "High neutron flux (source range)" signal, and then, the controlled state is reached. Under State B, the residual heat is removed via the VDA [ASDS] of all SGs. and the feedwater is supplied by the ARE [MFFCS] low load lines without LOOP or ASG [EFWS] with LOOP. Under State C\D, the residual heat is removed via the RIS [SIS] in RHR Mode.

12.8.4.2.5 Results and Conclusions

a) From the Initiating Event to the Controlled State

The results of this event can be bounded by the "Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at a subcritical or low power startup condition (State A)" accident in Sub-chapter 12.7.4.1 since the initial sub-criticality margin is higher.

b) From the Controlled State to the Safe State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.8.4.3 Uncontrolled Single RCCA Withdrawal (State A)

12.8.4.3.1 Initiating Event

This event refers to the uncontrolled withdrawal of a single control rod under full power reactor operation. It can only occur in two unlikely cases:

a) Case 1: In manual mode, a single control rod can be withdrawn through operator error, based on their misunderstanding that the clusters are misaligned or dropped. In this case, it is assumed there is a:

    1) Malfunction of a single rod position detector;

    2) Disregard of operating procedures and misunderstanding of the alarm signal produced during RCCA withdrawal.

a) Case 2: Under automatic control, several simultaneous electrical or mechanical failures can cause withdrawal of a single control rod. This assumes:

    1) Simultaneous and independent electrical or mechanical failures;

    2) Misunderstanding of the respective alarm signals.

The uncontrolled withdrawal of a single control rod causes an insertion of reactivity, resulting in an increase in the core power, coolant temperature and hot channel factor. Around the withdrawal position, the local power peak may lead to a low DNBR. The DNBR could drop below the design limit, potentially leading to fuel cladding failure.

12.8.4.3.2 Acceptance Criteria

The following criteria are used:

a) The amount of fuel rods experiencing DNB must remain lower than 10%;

b) The peak cladding temperature must remain lower than 1482℃.

12.8.4.3.3 Main Safety Functions

For this event, the reactor trip can be triggered by the following signals:

a) High neutron flux (power range, high setpoint);

b) Overtemperature ΔT;

c) Overpower ΔT;

d) Pressuriser pressure high 2.

12.8.4.3.4 Typical Events Sequence

a) From the initiating event to the controlled state

After the withdrawal of a single control rod, RT can be triggered by numerous signals.

This limits the local power peak and mitigates the consequence of the accident. After RT, the ARE [MFFCS] full load lines are isolated. The decay heat is removed via the VDA [ASDS] of all SGs and the feedwater is supplied by the ARE [MFFCS] low load lines without LOOP or ASG [EFWS] with LOOP. The RCS will remain stable and the controlled state is subsequently reached.

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.8.4.3.5 Analysis Assumptions

a) Initial Condition

1) The initial power is set to the full power plus uncertainty;

2) The initial coolant temperature is set to the nominal value plus uncertainty;

3) The initial PZR pressure is set to the nominal value minus uncertainty;

4) The initial flow is set to the thermal-hydraulic design flow;

5) The initial state corresponds to the cases with the R bank at the insertion limit when the Axial Offset Anomaly (AOA) is in the boundary of the LCO.

b) The uncontrolled single control rod is withdrawn at the initial time. Two withdrawal speeds are studied: 8 steps/min and 72 steps/min.

c) Two sets of reactivity coefficients are considered:

1) Minimum reactivity feedback: assuming the minimum moderator density coefficient and the minimum Doppler power coefficient;

2) Maximum reactivity feedback: assuming the maximum moderator density coefficient and the maximum Doppler power coefficient.

d) Single failure

The single failure assumption is applied to one “Overtemperature ΔT” signal (the sensor or channel).

e) LOOP

The impact of the LOOP is considered in this analysis.

12.8.4.3.6 Results and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see reference [36]) shows that the number of rods experiencing a DNB is less than the acceptance criteria. The peak cladding temperature is lower than the design limit.

Thus the acceptance criteria for this event are met.

b) From the controlled state to the safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.8.5 Decrease in Reactor Coolant System Inventory

12.8.5.1 Rupture of a Line Carrying Primary Coolant outside Containment (e.g. Nuclear Sampling Line) (State A)

12.8.5.1.1 Initial Event

The loss of primary coolant outside containment may result from the failure of:

a) The RCV [CVCS], including the RCV [CVCS] connecting lines;

b) The nuclear sampling system (REN [NSS]).

The maximum break equivalent area with regards to reactor coolant depletion corresponds to the minimum area of the RCV [CVCS] letdown line (i.e. 60 cm^2). Small breaks that could be compensated by the RCV [CVCS] are not analysed in this sub-chapter. Therefore, only the largest breaks occurring on the RCV [CVCS] letdown line are analysed.

Austenitic stainless steels are used for the components connecting the RCV [CVCS] and the REN [NSS] that are used for the primary coolant because of its high resistance to generalised corrosion during operation and cold shutdown conditions. Precautions are taken to avoid other sources of localised corrosion by means of monitoring and optimising the chemical composition of the primary coolant and austenitic stainless steels. The chloride and oxygen content in the primary coolant is controlled to avoid pitting of materials during operation, which can also protect the stainless steel against corrosion cracking. The risk of rupture caused by corrosion on the RCV [CVCS] and the REN [NSS] is therefore reduced.

A loss of coolant accident (LOCA) occurring on a line carrying primary coolant outside the containment induces a loss of primary coolant, potential decrease in RCP [RCS] pressure (if the loss flowrate cannot be compensated by the RCV [CVCS]) and radioactive release to the environment. This accident may lead to a decrease in reactor coolant inventory and potential core overheating.

12.8.5.1.2 Acceptance Criteria

The decoupling criteria for LOCA events are described in Sub-chapter 12.5.1

12.8.5.1.3 Main Safety Functions

During this event, reactor protection is provided by the following signals and actions:

a) Reactor trip (RT) is triggered by the “Pressuriser pressure low 2” signal.

b) Safety injection (SI) signal is actuated by the “Pressuriser pressure low 3” signal.

c) The SI signal automatically starts the medium head safety injection (MHSI) and the low head safety injection (LHSI) pumps, and initiates a medium-pressure rapid cooldown (MCD).

d) The SI signal induces the automatic isolation of the RCV [CVCS] letdown line. It isolates the break and stops RCP [RCS] coolant inventory depletion.

e) The reactor coolant pumps (RCPs) are tripped by the "RCP ∆P low 1 and SI" signal.

f) The ASG [EFWS] pumps are actuated by the "SG level (wide range) low 2" signal.

12.8.5.1.4 Typical Events Sequences

a) From the initiating event to controlled state

The consequences of the rupture of a line carrying primary coolant outside containment are covered by the small break LOCA (SB-LOCA) presented in Sub-section 12.8.5.3, because the safety criteria, initial state and main assumptions are the same. As a consequence, based on the SB-LOCA results, it can be concluded that the acceptance criteria related to the core are met for the short term analysis of the rupture of a line carrying primary coolant outside containment.

Core sub-criticality is ensured by the rod drop negative reactivity effects, which acts in addition to the RIS [SIS] boron injection.

The controlled state is reached when the following conditions are met:

1) The core is sub-critical;

2) The RCP [RCS] coolant inventory is stable or increasing to the correct level;

b) From controlled state to safe state

In controlled state, the reactor coolant inventory is stable or increasing and the core decay heat is removed via the SGs until the situation where the connecting conditions for the RIS [SIS] to operate in RHR mode are reached. Once one RIS [SIS] train is connected to the RCP [RCS] in RHR mode, the following conditions are met:

1) The core is sub-critical due to the boration performed by the operator.

2) The decay heat is removed via the RIS [SIS] in RHR mode.

12.8.5.1.5 Analysis Assumptions

The rupture of a line carrying primary coolant outside containment in state A is classified as a DBC-3 event. This fault is bounded by the analysis of SB-LOCA, IB-LOCA and LB-LOCA. Considering the LOCA criteria, the demonstration relies on qualitative explanations. The source terms are analysed in Sub-chapter 12.11.

12.8.5.1.6 Result and Conclusion

a) From the initiating event to the controlled state

The rupture of a line carrying primary coolant outside containment in state A is classified as a DBC-3 event. This fault in terms of decoupling acceptance criteria is bounded by the analysis of SB-LOCA, IB-LOCA and LB-LOCA. Thus the decoupling acceptance criteria are met. Radiological consequences are analysed in Sub-section 12.11.5.8.

b) From the controlled state to the safe state

The description from the controlled state to the safe state is provided in Sub-chapter 12.10.

12.8.5.2 SG Tube Rupture (SGTR) (one tube) (State A)

12.8.5.2.1 Initiating Event

This sub-chapter describes the thermal-hydraulic analysis of a steam generator tube rupture event. The steam generator tube rupture event is defined as the double-ended guillotine rupture of a single steam generator (SG) tube.

This accident will lead to an increase in radioactivity in the secondary system due to the leak of radioactive coolant to the secondary system from the reactor coolant system (RCP [RCS]). If a loss of offsite power or failure of the condenser steam dump system happens during this event, discharges of steam or liquid from the main steam safety valve (MSSV) and/or the VDA [ASDS] will lead directly to a discharge of activity to the atmosphere, as the damaged steam generator will be contaminated. The radioactivity of the primary side coolant is caused by corrosion and fission products generated through the continuous operation of the reactor with a limited number of damaged fuel rods.

The probability and risk of steam generator tube rupture (SGTR) event is reduced through the following precautions:

a) High ductility of SG tube material;

b) Blowdown system location at the bottom of SG tube bundle to prevent solid deposits on SG tube plate;

c) Chemically conditioned secondary water to protect SG tubes from corrosion;

d) Prevention of projectiles originating from the main feedwater;

e) Specification of SG support plates to prevent tube damage and pipe whip (of neighbouring tubes) following tube rupture;

f) Continuous monitoring and control of secondary side activity.

The SGTR of one tube is classified as a DBC-3 event.

The cases studied in this sub-chapter correspond to the double-ended guillotine rupture of one tube in one SG, which allows unimpeded blowdown from both ends of the tube.

The rupture is located in the lower part of the SG tubes bundle, close to the tubesheet, on the cold side. This location maximises the SGTR leak flowrate. For DBC-3 transients, a LOOP is considered part of the event, if it adds further pessimism.

This sub-chapter aims to quantify the maximum amount of radioactivity release to the environment. If overfilling occurs, contaminated water will be directly released to the environment and the radioactivity release will be significant. Therefore, an SG overfilling assessment is performed (see Reference [37]) and the analysis result demonstrates that, even if under the most onerous conditions, overfilling will not occur. Without overfilling, the radioactivity is released to the environment via steam. The case presenting the most pessimistic radioactivity release is analysed hereafter.

12.8.5.2.2 Acceptance Criteria

The consequences of an SGTR (one tube) are analysed with respect to the following decoupling criteria:

a) There shall be no core damage;

b) There shall be no water flowing through the MSSV, to prevent the MSSV from seizing open;

c) The plant shall be brought to and maintained in safe state.

The radiological criterion is the dose equivalent released to the environment, and will be discussed in Sub-chapter 12.11.

12.8.5.2.3 Main Safety Functions

In case of an SGTR, automatic protection systems and operator actions will trip the reactor, remove the RCP [RCS] residual heat, eliminate the leak flowrate from the primary side to secondary side and bring the plant to safe state.

a) FC1 system/function

1) SGTR detection on receipt of “High activity in the VVP KRT [PRMS]” signal:

   On high activity detection in the VVP KRT [PRMS], an alarm is actuated.

2) Reactor trip following “Pressuriser pressure low 2” signal:

   RT is triggered by the “Pressuriser pressure signal low 2” signal. The related uncertainties are taken into account to bring forward the RT. This maximises the release of radioactive steam to the environment upon VDA [ASDS] opening.

3) Turbine trip:

   The RT signal triggers the turbine trip.

4) ARE [MFFCS] full load line isolation:

   Following the RT signal, the ARE [MFFCS] full load line for all SGs is isolated. In a penalizing way, the ARE [MFFCS] low load line is supposed to be isolated at the RT signal.

5) APG [SGBS] isolation:

   APG [SGBS] discharge is isolated when ASG [EFWS] is actuated.

6) VDA [ASDS]:

The VDA [ASDS] opening setpoint is minimised on the affected SG (SGa) and maximised on the unaffected SGs to maximise the steam release from the SGa before the end of MCD.

At the beginning of the operator actions, the set value of the SGa VDA [ASDS] is manually increased above the MHSI injection head and below the MSSV opening set value, i.e. { }.

7) Safety injection:

SI is actuated by the "Pressuriser pressure low 3" signal or "SG level (narrow range) high 2" signal. This initiates a medium pressure rapid cooldown. The maximum MHSI flowrate curve is chosen to maximise the pressure difference between the RCP [RCS] and the SGa.

8) Medium pressure rapid cooldown:

Following the SI signal, a medium pressure rapid cooldown (performed by all SGs, including the SGa) is initiated with a rate of { }, to cool down the primary circuit.

9) ASG [EFWS] startup:

The ASG [EFWS] is actuated automatically by the "LOOP and SI" signal or by "SG level (wide range) low 2" signal. Only two unaffected SGs are fed, since the other ASG [EFWS] pump is affected by the single failure. After ASG [EFWS] actuation, the SG level in the unaffected SGs is maintained at the reference water level.

The ASG [EFWS] is operated with its minimum flowrate.

During the long-term, the ASG [EFWS] is manually controlled by the operator to maintain the levels of the two unaffected SGs to their nominal value.

10) MSIV closure:

The MSIV closure is manually initiated when one SG becomes unavailable. All the main steam lines are then isolated.

b) Other systems:

SG level control is in operation from accident initiation and keeps operating normally until RT. This maximises the activity released from the SGa.

c) Operator actions

The main operator actions include:

1) SGa isolation:

   After SGTR detection, the operator isolates the SGa (if not already done), comprising ARE [MFFCS] and ASG [EFWS] isolation, MSIV closure and an increase in the VDA [ASDS] opening set value.

2) RCP [RCS] boration:

   The RBS [EBS] pumps are started by the operator at the beginning of the RCP [RCS] cooldown. The RBS [EBS] injection is stopped when the primary cold shutdown state boron concentration is reached.

3) RCP [RCS] cooldown by unaffected SGs:

   The cooldown is performed using the VDA [ASDS]. The cooling rate is { }.

4) MHSI stop:

   Two of the three MHSI pumps are stopped. The last MHSI pump stops when the core outlet temperature TRIC is reduced to { }.

5) RCP [RCS] and SGa depressurisation:

   During RCP [RCS] depressurisation, the accumulators are isolated when the RCP [RCS] pressure has decreased below { }.

   RCP [RCS] is then depressurised until the low head safety injection (LHSI) injection pressure is reached (RCP [RCS] pressure is stabilised). In the event of LOOP, the normal spray is unusable. The RCP [RCS] depressurisation is then realised by opening the VDA [ASDS] of the SGa.

6) RIS [SIS] in RHR mode connection:

   The connection of the RIS [SIS] in RHR mode guarantees continuous heat removal and core long term cooling.

12.8.5.2.4 Typical Events Sequences

A typical sequence, the most likely to occur during the transient, is described hereafter. Within this sub-chapter, the following description considers only the full power condition for this is the most onerous case for radiological release, and the overfilling case is described in Reference [37].

The sequence of events consists of two phases: the short-term phase until leak elimination and the long-term phase where the plant is operated from the leak elimination to the safe state.

a) From the initiating event to leak elimination (short term)

   1) From the initiating event to the controlled state

The controlled state for SGTR is defined as a state when core coolant inventory remains stable and residual heat removal can be ensured via the SGs.

At the beginning of this event, primary coolant leaks to the secondary side through the break, which leads to the contamination of the affected SG. Meanwhile, the primary pressure decreases. The reactor trip signal triggered depends on the initial conditions. For full power operation: The SGTR can be identified by the operator receiving the “High activity in the main steam system (VVP [MSS]) radiation monitoring system (KRT)” signal. Reactor trip is triggered by the “Pressuriser pressure low 2” signal. Turbine trip and isolation of ARE [MFFCS] full and low load lines for all SGs are initiated following the RT signal.

After RT and turbine trip, the secondary pressure increases and rapidly reaches the setpoint of the VDA [ASDS], if the GCT [TBS] is unusable. Contaminated steam is thus released to the environment and decay heat is removed.

The continuous leakage to secondary side and the decrease in decay heat after reactor trip leads to a primary depressurisation. It is likely the “Pressuriser pressure low 3” signal will be triggered. Within this sub-chapter, the following description assumes the break is large enough to lead to the “Pressuriser pressure low 3” signal.

Following the safety injection signal triggered by the “Pressuriser pressure low 3” signal, the Medium pressure rapid cooldown (MCD) is actuated. {

} The medium-head safety injection (MHSI) pumps are actuated by the SI signal and start injecting when the primary pressure is lower than their injection head. The MHSI injection flow can compensate for the leak flow from the SGTR and the controlled state is reached.

2) From the controlled state to the leak elimination

To stop the leak, the operator isolates the SGa from both the steam side and feed side. In other words, the operator closes the ASG [EFWS] and the MSIV. The VDA [ASDS] setpoint of the SGa is adjusted up to a value between the MHSI injection head and MSSV setpoint in order to limit the radioactive release.

The injection of MHSI maintains the primary pressure at a stable level. To reduce the leakage flow, the operator shuts down two of the three MHSI pumps (with the remaining left operating). Due to the isolation of the SGa, the pressure of SGa increases until it reaches primary pressure level and the leak is stopped. Before the leak is stopped, the SGa is not overfilled and only steam is released to the environment.

b) From the leak elimination to the safe state (long term)

The safe state for SGTR is defined as a state where the RIS [SIS] train is connected to the RCP [RCS], in residual heat removal (RHR) mode, and the SGa remains isolated. The operator performs primary cooldown and depressurisation to reach the RHR connection conditions, which are:

1) RCP [RCS] hot leg pressure < { };

2) RCP [RCS] hot leg temperature < { };

3) RCP [RCS] hot leg saturation margin (ΔTsat) and Reactor Pressure Vessel Level (RPVL) consistent with RIS [SIS] in RHR mode suction conditions from the hot leg.

In order to cool down the primary coolant, the operator uses the RBS [EBS] to compensate for the reactivity insertion resulting from the RCP [RCS] cooldown. Unaffected SGs and the MHSI are used to cool down the primary side at a rate of { } with two or three RBS [EBS] trains in operation or at a rate of { } with one RBS [EBS] train in operation.

When the primary temperature is lower than { }, the last MHSI injection is stopped and the operator prepares to perform final depressurisation via the SGa. Prior to that, the operator shall confirm that the SGa level is lower than the relevant limits. Otherwise, the partner SG shall be isolated, with its level limited to a lower value so that the SGa water inventory can be transferred to the partner SG via the steam generator blowdown system (APG [SGBS]) transfer line. The RIS [SIS] in RHR mode can finally be connected and the safe state is reached.

12.8.5.2.5 Analysis Assumptions

a) Single failure

The failure of SGa ASG [EFWS] pump is assumed. This assumption worsens the radioactive release from the SGa as it increases the risk of heat transfer tube exposure.

b) Initial state

The conditions for initial state are chosen to maximise the RCP [RCS] heat to be removed after RT and to assume the SGa tubes are exposed as soon as possible.

c) Neutronic parameters

Core power is assumed constant at 102%FP until RT. Following RT, the maximum decay heat curve with an uncertainty of 1.645σ is used.

d) Loss of offsite power (LOOP)

The LOOP at turbine trip is assumed to occur if it is conservative.

e) Other assumptions: operator actions

The first operator action is assumed to be performed at 30 minutes after the “high activity in the VVP KRT [PRMS]” signal. The first local manual operator action is assumed to be performed one hour after this signal.

For long-term mitigation, the operator actions aim at reaching the safe state.

12.8.5.2.6 Result and Conclusion

a) Results

The detailed analysis of this fault see Reference [37] shows that the controlled state can be reached after injection of MHSI and the leak elimination can be achieved by operator actions.

The RIS [SIS] in RHR mode connection conditions are met roughly 4 hours after SGTR initiation. Once the RHR is underway, the safe state is reached. The description for the long-term analysis is provided in Sub-chapter 12.10.

b) Conclusion

As the analysis shows, the reactor can be taken to the safe state. Radiological consequences are analysed in Sub-chapter 12.11.

12.8.5.3 Small Break (Loss of Coolant Accident) (SB-LOCA) (at Power) including a Break in the Emergency Boration System (RBS [EBS]) Injection Line (State A)

12.8.5.3.1 Initiating Event

A small break loss of coolant accident (SB-LOCA) is defined as an accident in which a small break occurs on the pipes of the reactor coolant system or on the upstream line of the second isolation valve connected to it. These accidents lead to pressuriser level instability and RCP [RCS] depressurisation with a possible core heat up due to lack of cooling.

12.8.5.3.2 Acceptance Criteria

The decoupling criteria for LOCA events:

a) The peak cladding temperature must remain lower than 1204℃;

b) The maximum cladding oxidation must remain lower than 17% of the cladding thickness;

c) The maximum hydrogen generation must remain lower than 1% of the amount that would generate if the whole active part of the cladding had reacted;

d) The ability to cool the core geometry shall be retained, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling;

e) Long-term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed.

12.8.5.3.3 Main Safety Functions

In this event, reactor protection is provided by the following signals and actions:

a) Reactor trip (RT) is triggered by the “Pressuriser pressure low 2” signal;

b) The safety injection signal is activated by:

   “Pressuriser pressure low 2” signal;

c) The injection by the RIS [SIS] accumulator to the RCP [RCS] loop is activated when:

   The primary pressure is lower than the accumulator injection pressure;

d) Action induced by the reactor trip is:

   Close of the turbine inlet valve;

e) The emergency feedwater system is actuated by the:

   “SG level (wide range) low 2” signal;

f) The main feedwater system is isolated by the:

- SI signal;
- RT signal.

During the long-term phase, the plant is brought to the safe state via MHSI and LHSI.

12.8.5.3.4 Typical Events Sequences

a) From the initiating event to controlled state

The small break results in loss of primary coolant, a potential decrease in RCP [RCS] pressure (if the break flowrate cannot be compensated by RCV [CVCS]) and radioactive release to the environment. This accident may lead to a decrease in reactor coolant inventory and potential core overheating. The SB-LOCA accident is mainly a gravity-driven accident, in which the RCP [RCS] discharges slowly with the evident formation of mixing layers throughout the RCP [RCS]. These mixing layers change over time, depending on the transient of two phase mass and energy mutual transfer. The first heat-up event results from the core level decrease and the formation of a loop seal, and can be mitigated by loop seal clearance during the accident. The second heat-up event is due to the boiling and evaporation of the core coolant. During this event, the flows from the MHSI, accumulators and LHSI enter the core and cool the fuel cladding to prevent further temperature increase, and eventually achieve the controlled state.

b) From controlled state to safe state

The safe state is reached once the RIS [SIS] is in operation in RHR mode. The following actions need to be performed (initiated by the operator) in order to reach the safe state:

1) Reactor coolant boration

   During the RCP [RCS] cooldown, boration is performed via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) Reactor coolant cooldown

   The cooldown, to achieve suitable connecting conditions for the RIS [SIS] in RHR mode, is performed for the units via the secondary side by reducing the VDA [ASDS] opening pressure set value. The RBS [EBS] is designed so that the boration matches the reactivity insertion resulting from the RCP [RCS] cooling. The cooling rate is consistent with the ASG [EFWS] tank capacity, which means that the RIS [SIS] can begin operation in RHR mode before the ASG [EFWS] tanks are emptied.

3) Reactor coolant depressurisation

   The reactor coolant depressurisation, achieving connecting conditions for the RIS [SIS] in RHR mode, is performed by the operator with the activation of the normal or auxiliary pressuriser spray. If the pressuriser sprays are unavailable, the PSVs can be used to depressurise the RCP [RCS].

12.8.5.3.5 Analysis Assumptions

a) Initial condition

The break is assumed to be located in the cold leg of the reactor coolant system. A break with an equivalent diameter lower than 5.0 cm is considered. This condition results in a covered core and a normal heat-carrying ability for the SG after the event.

Assumptions for the initial condition are shown as follows:

1) Initial reactor power is the nominal power plus the maximum error of the steady state measurement;

2) The initial coolant flowrate is the thermal design flowrate;

3) The average temperature of the coolant is the rated value plus the maximum steady state control range and measurement error;

4) The initial pressure of the pressuriser is the rated value plus the maximum steady state surge and measurement error so as to delay the reactor trip and safety injection signals;

5) The initial level of the pressuriser is the rated level at power minus 7% based on uncertainties;

6) The core bypass flowrate is 6.5%.

b) Single failure

Considering single failure, the flow in only one safety injection train (1MHSI+1LHSI) is taken into account for the RIS [SIS], and the 3rd MHSI/LHSI train is assumed lost at the break and not considered.

c) LOOP assumption

It is assumed that LOOP occurs at the time of turbine shutdown.

d) Core related assumptions

In thermal-hydraulic analysis, the reactor average core is simulated as a core with a severe axial power distribution.

In hot rod analysis, the core enthalpy rise factor $F_{\Delta H}$ is set at its maximum value (1.65), and the peaking factor F_Q is set at its maximum value (2.45).

12.8.5.3.6 Results and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see Reference [38]) shows that for the break conditions with the equivalent diameters of 2.5 cm and 5.0 cm, the SI system is able to provide sufficient flow to keep the core covered and continuously cooled. The relevant safety acceptance criteria are thus met.

b) From the controlled state to the safe state

After the controlled state, the RHR connecting condition is reached through the primary boration, cooling and depressurisation with at least one LHSI train in RIS [SIS] mode and at least one LHSI train in RHR mode, which brings the reactor to a safe state. The description is provided in Sub-chapter 12.10.

Radiological consequences of LOCA are analysed in Sub-chapter 12.11.

12.8.5.4 Small Break LOCA (at Shutdown, RIS [SIS] not Connected in RHR Mode) including a Break in the RBS [EBS] Injection Line (State A\B)

12.8.5.4.1 Initiating Event

A small break loss of coolant accident (SB-LOCA) is defined as an accident in which a small break occurs on the pipes of the reactor coolant system or on the upstream line of the second isolation valve connected to it.

A SB-LOCA on the RCP [RCS] induces a loss of coolant inventory, potential pressure decrease of the RCP [RCS] and potential core overheating.

The small break loss of coolant accident (at shutdown conditions) can be divided into two conditions regarding to the availability of accumulators.

For cases with the initial primary pressure higher than { }, the accumulators are not isolated. In this condition, safety functions available are the same as cases at State A, while the residual heat in the primary coolant and the core power is lower. Therefore, the analysis in Sub-section 12.8.5.3.1 is bounded by this condition.

For cases with the initial primary pressure lower than { }, the accumulators are isolated. This condition is analysed in this section. The following differences in terms of FC1 mitigation methods are considered compared to state A:

a) The change of SI signal.

b) The isolation of accumulators.

12.8.5.4.2 Acceptance Criteria

The SB-LOCA analyses should meet the following acceptance criteria:

a) The peak cladding temperature shall not exceed 1204℃;

b) Maximum clad oxidation. The total oxidation of the cladding at the limiting point shall not exceed 17% of the total clad thickness before oxidation;

c) Maximum hydrogen generation. The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the amount that would be generated if all of the cladding material in the active core, were to react;

d) Coolable geometry. Changes in core geometry shall be such that the core remains capable of being cooled;

e) Long-term cooling. After any successful initial operation of the Emergency Core Cooling System (ECCS), the core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

12.8.5.4.3 Main Safety Functions

For this event, the following plant safety functions can mitigate the event:

a) Safety injection (SI) signal is actuated by:

“Hot leg ΔP_{sat} low 1” signal;

b) The medium head safety injection (MHSI) and low head safety injection (LHSI) pumps is actuated by:

SI signal;

c) Medium pressure rapid cooldown (MCD) is actuated by:

SI signal;

d) The reactor coolant pumps (RCPs) are tripped by:

“RCP ΔP low 1 over two loops and SI” signal;

e) Containment isolation is trigged by:

SI signal.

12.8.5.4.4 Typical Events Sequences

The SB-LOCA in state B is classified as a DBC-3 event. A typical sequence of events is described below.

a) From the initiating event to controlled state

In state B, safety injection (SI) signal is actuated by the “Hot leg ΔP_{sat} low 1” signal, inducing the following actions:

1) Starting of medium head safety injection (MHSI) and low head safety injection (LHSI) pumps;

2) { } After MCD, the RCP [RCS] pressure decreases sufficiently to allow MHSI pumps injection into the cold legs;

3) Containment isolation stage: the reactor coolant pressure boundary (RCPB) is isolated, in particular the RCV [CVCS] letdown line and the steam generators (SG) blowdown lines are isolated.

Reactor coolant pumps (RCPs) are tripped following the “RCP ΔP low 1 over two loops and SI” signal.

After blowdown and medium pressure rapid cooldown phase, the primary pressure stabilises to a value that is higher or equal to the secondary side pressure, until the break energy flowrate becomes sufficient to remove the decay heat. If the safety injection flow cannot compensate for the break flowrate, RCP [RCS] coolant water inventory continues to decrease. During this phase, the break flow is sub-saturated and eventually reaches saturation state.

The break flowrate decreases as the void fraction in the affected cold leg increases. Eventually, the break flow changes to a single vapour phase. The primary coolant inventory depletion stops when the SI flowrate is able to compensate for the break flowrate. If the accident occurs before the isolation of the accumulators by the operator, they may discharge borated water into the RCP [RCS].

Then, the controlled state is reached. It corresponds to the achievement of stable heat removal conditions, via the operation of the MHSI and the GCT [TBS] or VDA [ASDS] of all SGs, core sub-criticality and a stabilised or increasing core coolant inventory due to SI flow. The feedwater supply is ensured by the SG main feedwater system or the emergency feedwater system if ARE [MFFCS] is not available.

b) From Controlled State to Safe State

The controlled state cannot be sustained due to the following reasons:

1) The ASG [EFWS] system tanks will empty;

2) The containment pressure and temperature will increase.

Thus, the transition to safe state is needed. The safe state is reached through several key means:

1) RCP [RCS] boration to keep core sub-criticality;

2) RCP [RCS] cooling to ensure LHSI injection;

3) RCP [RCS] depressurisation to reach RHR connection condition;

4) RHR connection.

12.8.5.4.5 Analysis Assumptions

The break with an equivalent diameter of 5.0 cm is assumed to be located in the cold leg of the coolant system, between the outlet of the main coolant pump and inlet of the reactor pressure vessel.

The most pessimistic single failure is loss of one RRI [CCWS] train in one unaffected loop. As a consequence, one RIS [SIS] train (one MHSI pump and one LHSI pump) is unavailable.

12.8.5.4.6 Results and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see Reference [39]) shows that, for the SB-LOCA with an equivalent diameter of 5.0 cm in state B, the SI system can provide sufficient flow rates to compensate for the loss of the coolant flow rates at the break and ensure that the core remains covered. Thus, the decoupled acceptance criteria are met. Radiological consequences of LOCA are analysed in Sub-chapter 12.11.

b) From the controlled state to the safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.8.5.5 Inadvertent Opening of a Pressuriser Safety Valve (State A)

The inadvertent opening of a Pressuriser Safety Valve (PSV) event is defined as the spurious opening of a pressuriser safety valve, which is normally closed during plant operation, and cannot be isolated or reclosed. It is similar to a small break loss of coolant accident (SB-LOCA) on the hot leg of the reactor coolant system. An inadvertent depressurisation of the reactor coolant system can occur as a result of an inadvertent opening of a PSV.

An inadvertent opening of a pressuriser safety valve is a DBC-3 event. The maximum opening area of the PSV is approximately 28.3 cm^2.

The maximum opening area of the PSV is larger than the largest SB-LOCA event, with an equivalent break area of roughly 20.0 cm^2. However, the effects and consequences of the inadvertent opening of a PSV are still bounded by the other PCC-3 event, “SB-LOCA DN < 50”, for the following reasons:

a) For the inadvertent opening of a PSV, the minimum effective injection capacity is two times as large as that of cold leg SB-LOCA for which the single failure is considered. Safety injection of the broken loop can be claimed for the PSV leak and two MHSIs are available versus one MHSI for the cold leg SB-LOCA.

b) The size of the PSV is only about 50% larger than the break size of SB-LOCA.

c) The leak mass flowrate of steam from the hot leg under this situation leads to less severe consequences than that of the cold leg break.

As above, the response of this event is bounded by the SB-LOCA analysis. Thus the plant can be taken to the controlled state and safe state under inadvertent opening of a pressuriser safety valve (State A), which is demonstrated by the analysis of SB-LOCA at power in Sub-section 12.8.5.3

12.8.6 Radiological Release of Systems or Components

12.8.6.1 Gaseous Waste Tank Break (State A to F)

A Gaseous Waste Tank Break accident is caused by leakage of or damage to the Gaseous Waste Treatment System (TEG [GWTS]), resulting in release of radioactive nuclides to the environment. The analysis in terms of source terms and radiological consequences is addressed in Sub-chapter 12.11.5.10.

12.8.6.2 Liquid Waste Tank Break (State A to F)

The analysis of this accident is to demonstrate that the release of radioactive substances into the environment is sufficiently limited. The analysis in terms of source terms and radiological consequences is provided in Sub-chapter 12.11.5.11.

12.8.6.3 Volume Control Tank Break (State A to F)

Leakage of or damage to the RCV [CVCS] tank will cause leakage of untreated primary coolant which may cause radioactivity release to the environment. The analysis in terms of source terms and radiological consequences is provided in Sub-chapter 12.11.5.12.

12.8.7 Fuel Pool Accidents

12.8.7.1 LOOP (>2 hours) affecting fuel pool cooling (State A)

12.8.7.1.1 Initiating Event

LOOP leads to the loss of electrical supply for all plant auxiliary systems including all pumps and the supporting systems of PTR [FPCTS] trains which provide cooling for the SFP. As the SFP temporarily loses cooling, the temperature, and the risks associated, may increase. An analysis is performed to demonstrate that cooling of SFP can be recovered and that safety state of SFP is ensured.

Two pre-accident conditions, End of Cycle (EOC) and Beginning of Cycle (BOC), are analysed in this accident. For each case, the spent fuel pool is cooled by the main PTR [FPCTS] train, and the PTR heat exchanger is cooled by a component cooling water system [CCWS] train. At the EOC condition, preventive maintenance for the PTR [FPCTS] train can be performed as the decay heat in the fuel storage pool is at its minimum in state A. At the BOC condition, maintenance for the supporting systems of the PTR [FPCTS] train, for example, the component cooling water system [CCWS], can be performed. If one PTR [FPCTS] train is in maintenance during power operation, hot shutdown and intermediate shutdown conditions, the backup PTR train shall be operated in its place.

12.8.7.1.2 Acceptance Criteria

The safety criteria for the DBC accidents associated with spent fuel storage pool are as follows:

a) Permanent maintenance of sub-criticality;

b) The fuel assemblies in the fuel pool remain covered;

c) Removal of decay heat from the spent fuel pool.

For a DBC event without spent fuel pool accidental drainage, the removal of decay heat from the fuel storage pool is ensured through a decoupling criterion: the fuel storage pool temperature must remain lower than 80℃ for all heat load conditions.

12.8.7.1.3 Main Safety functions

Every PTR [FPCTS] train is designed to remove decay heat from the SFP.

12.8.7.1.4 Typical Events Sequence

After the loss of offsite power (LOOP), all PTR [FPCTS] trains are lost and SFP water temperature increases until EDGs are started up.

Once the EDGs are started up, at least one PTR [FPCTS] train, cooled by an intact RRI [CCWS] train, becomes available. Powered by the EDGs, cooling for SFP is recovered, the temperature increase is stopped and safety is ensured.

12.8.7.1.5 Analysis Assumptions

Conservative assumptions are adopted in the analysis as follows:

a) The bounding decay heat of the fuel assemblies in SFP at BOC is assumed.

b) The initial water temperature of the SFP is assumed to be 50℃, which covers all the normal operating conditions.

c) The initial water volume of the SFP is 1265.8 m^3 corresponding to a water level of 16.9m.

d) The SFP and the PTR [FPCTS] pipes are considered as adiabatic.

e) Water heating is only localized in the SFP compartment, and the water is heated homogeneously in this area.

f) Once the SFP cooling system is recovered, a maximum RRI [CCWS] inlet temperature of 45℃ is used.

g) EDGs are started up 1 hour after the accident.

h) One cooling train of the PTR [FPCTS] is considered unavailable due to maintenance to PTR train or its supporting system.

i) Since the third backup PTR [FPCTS] train shall be started up first before the maintenance to one PTR [FPCTS] train, there is no requirement for change of state for the PTR [FPCTS]. Single Failure Criterion is not considered.

12.8.7.1.6 Results and Conclusion

The analysis demonstrated that the stabilized SFP water temperature is below 80℃ and the fuel assemblies remain covered during the entire transient. Sub-criticality can be ensured by the design of the storage rack which is analysed in detail in Reference [40].

All acceptance criteria are met for this accident.

12.8.7.2 Loss of One Train of the PTR [FPCTS] or of a Supporting System (with the Reactor Core Offloaded to the Fuel Pool) (State F)

12.8.7.2.1 Initiating Event

During refuelling, two PTR [FPCTS] cooling trains are operating to maintain cooling of the SFP, into which fuel assemblies are unloaded from the reactor vessel. If one operating cooling train of the PTR [FPCTS] is lost, caused by failure of the pump or heat exchanger for example, SFP temperature may increase. Therefore, an analysis has been performed to demonstrate that SFP cooling can be recovered and the safety state of SFP can be ensured.

12.8.7.2.2 Acceptance Criteria

The safety criteria for the DBC accidents associated with spent fuel storage pool are as follows:

a) Permanent maintenance of sub-criticality;

b) The fuel assemblies in the fuel pool remain covered;

c) Removal of decay heat from the spent fuel pool.

For a DBC event without spent fuel pool accidental drainage, the removal of decay heat from the fuel storage pool is ensured through a decoupling criterion where the fuel storage pool temperature must remain lower than 80℃ for all heat load conditions.

12.8.7.2.3 Main Safety functions

Every PTR [FPCTS] train is designed to remove decay heat from the SFP.

12.8.7.2.4 Typical Events Sequence

Under the most penalizing case, cooling train A of the PTR [FPCTS] is lost as the initial event and cooling train B is also lost due to the maintenance to the PTR [FPCTS] supporting systems. PTR train C, is already in service, remains available for SFP cooling.

12.8.7.2.5 Analysis Assumptions

The main assumptions adopted in the analysis are listed as follows:

a) The maximum decay heat of the SFP for the last fuel assemblies unloaded into the SFP in state F is taken into account;

b) The initial water temperature of the SFP is assumed to be 50℃, which covers all the normal operating conditions;

c) The initial water volume of SFP is 1265.8m^3 corresponding to a water level of 16.9m;

d) The SFP and the PTR [FPCTS] pipes are considered as adiabatic;

e) Water heating is only localized in the SFP compartment, and the water is heated homogeneously in this area;

f) Once the SFP cooling system has been recovered, a maximum RRI [CCWS] inlet temperature of 45℃ is used;

g) One cooling train of the PTR is considered unavailable due to the maintenance on its supporting systems.

12.8.7.2.6 Results and Conclusion

When cooled by PTR [FPCTS] train C, the SFP water temperature increases at first and then stabilized below 80℃. The fuel assemblies remain covered during the entire transient and the sub-criticality of fuel assemblies is ensured by the design of the storage rack which is analysed in detail in Reference [40].

All acceptance criteria are met for this accident.

12.8.7.3 Isolatable Piping Failure on a System Connected to the Spent Fuel Pool (State A to F)

12.8.7.3.1 Initiating Event

Isolatable piping failure on a system connected to the spent fuel pool could happen in all states. The break is assumed to be located downstream of the second isolation valve of a PTR train. The break leads to the drainage of the SFP and connected compartment. In state A, one of the PTR cooling trains is used to cool the spent fuel pool. In states E and F, two PTR cooling trains are used to cool the pool.

12.8.7.3.2 Acceptance Criteria

The safety criteria for DBC accidents associated with the spent fuel storage pool are as follows:

a) Permanent maintenance of sub-criticality;

b) The fuel assemblies in the fuel pool remain covered;

c) Removal of decay heat from the spent fuel pool.

For the DBC accidental fuel storage pool draining events, the removal of decay heat from the fuel storage pool is ensured through a decoupling criterion: the fuel storage pool water temperature must remain below 95℃.

12.8.7.3.3 Main Safety functions

For this event, the following plant safety functions can mitigate the event:

a) Every PTR [FPCTS] train is designed to remove decay heat from the SFP.

b) Isolation of the PTR line following the signal “SFP water low-level L4”.

12.8.7.3.4 Typical Events Sequence

a) For accident happened in State A:

In state A, PTR train A is in operation, and PTR train B is considered to be unavailable for maintenance in the accident.

A break in the PTR train leads to the drainage of the SFP. The break is assumed to be located downstream of the second isolation valve of PTR train A. When the water level drops to “low-level L4” (+16.0m), the isolation signal of the PTR trains is triggered. 80 seconds later (signal time delay and time for valve isolation action), the PTR trains are isolated. The result shows that the water level is stabilized at a level above the lowest acceptable water level for PTR train C operation.

b) For accident happened in State E or F:

In state E or F, PTR train A and B are used in operation. PTR train C is considered to be unavailable due to maintenance.

A break in the PTR train leads to the drainage of the SFP. The break is assumed to be located downstream of the second isolation valve of PTR train A. When the water level drops to "low-level L4" (+16.0m), the isolation signal of the PTR trains is triggered. 80 seconds later (signal time delay and time for valve isolation action), the PTR trains are isolated. The result shows that the water level is stabilized at a level above the lowest acceptable water level for PTR train B operation.

12.8.7.3.5 Analysis Assumptions

Main assumptions adopted in the analysis are listed as follows:

a) The decay heat of the SFP in state A, E and F are conservatively assumed respectively.

b) The break is conservatively assumed to be located in the lowest position of the PTR train, to maximize the break flowrate.

c) The initial SFP water temperature is assumed to be 50℃, which covers all the normal operating conditions.

d) The initial water volume of the SFP is 1265.8m^3 corresponding to a water level of 16.9m.

e) The SFP and the PTR [FPCTS] pipes are considered as adiabatic.

f) The water heating is only localized in the SFP compartment, and the water is heated homogeneously in this area.

g) Once the SFP cooling system is recovered, a maximum RRI [CCWS] inlet temperature of 45℃ is used;

h) One cooling train of the PTR [FPCTS] is considered to be unavailable for its preventive maintenance in state A, and for maintenance to its supporting system in state E/F.

12.8.7.3.6 Results and Conclusion

In all states, the temperature of the SFP will not exceed 95℃, and the fuel assemblies remain covered during the entire transient. The sub-criticality of the fuel assemblies is ensured by the design of spent fuel storage rack which is analysed in detail in Reference [40].

All acceptance criteria are met in this accident.

12.9 Analyses of DBC-4 events

12.9.1 Increase in Heat Removal by the Secondary Side

12.9.1.1 Large Steam System Piping Break (State A\B)

12.9.1.1.1 Large Steam System Piping Break (State A)

12.9.1.1.1.1 Initiating Event

The steam system piping break induces an initial increase in the steam flow which then decreases during the accident as the steam pressure falls.

The energy removed from the RCP [RCS] increases, causing decreases in RCP [RCS] coolant temperature and pressure, which lead to the core overcooling and an insertion of positive reactivity caused by a negative moderator temperature coefficient. The increase in reactivity in the core induces a rise in nuclear power at power operation or results in a return to criticality during hot shutdown state.

12.9.1.1.1.2 Acceptance Criteria

The safety criteria are the radiological limits for DBC-4 events.

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.9.1.1.1.3 Main Safety Functions

For this event, the following plant safety functions can mitigate the event:

a) RT is actuated by any of the following signals:

    1) High neutron flux (power range, high setpoint);

    2) Pressuriser pressure low 2;

    3) Pressure drop of SG high 0;

    4) Pressure drop of SG high l.

b) MSIV isolated by the “SG pressure low 1” signal or “Pressure drop of SG high 1” signal.

c) ARE [MFFCS] full load line isolated by the “Pressure drop of SG high 1” signal.

d) ARE [MFFCS] low load lines isolated after the“Pressure drop of SG high 2” signal.

e) Safety injection triggered by the “Pressuriser pressure low 3” signal.

f) ASG [EFWS] triggered by the “SG level (wide range) low 2” signal.

12.9.1.1.1.4 Typical Events Sequences

a) From the initiating event to the controlled state

Initially, the fault leads to the secondary side depressurisation. RT and the closure of all MSIVs are triggered by the "SG pressure low 1" or "Pressure drop of SG high 1" signals, thus the ARE [MFFCS] full load lines for all SGs are isolated. ARE [MFFCS] low load lines may also be isolated following the "Pressure drop of SG high 2" signal. The "Pressuriser pressure low 3" signal would trigger the safety injection (RIS [SIS]). After the ARE [MFFCS] low load lines have been isolated, the "SG level (wide range) low 2" signal should initiate the ASG [EFWS] for the affected SG.

The RCP [RCS] cooldown induces positive reactivity in the core, and the reactor may return to criticality. However, the Doppler effect may limit the power excursion.

The RIS [SIS] supplies sufficient boron to compensate for the reactivity insertion, bringing or maintaining the core sub-critical.

Thereafter, the controlled state is reached, where the heat removal is ensured by the VDA [ASDS] of the unaffected SGs, and feedwater is supplied by the ASG [EFWS].

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, the affected SG will be isolated when the first operator action happens. After that, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurization via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.9.1.1.1.5 Analysis Assumptions

a) Studied case

The studied case is the double-ended guillotine break of the main steam line (MSL) and the break locates at the upstream of MSIVs.

b) Single failure

The single failure is considered as the failure of one train of medium head safety injection (MHSI), as the boration function it provides is key to limiting the reactivity insertion and power excursion.

c) Initial state

The initial condition is assumed for a hot shutdown state, as this event would have more severe consequences if it happens at 0% NP than at power. The consequences of the accident are less severe when the reactor is at power as the reactor coolant system contains more energy than at zero power, and some energy is contained in the fuel. This remaining energy provides thermal inertia and delays the time at which temperature and reactivity shutdown margins corresponding to zero power are reached.

Moreover, the secondary system has a greater initial inventory and higher pressure within the SG under these conditions, causing a more dramatic overcooling of the RCP [RCS] overcooling and a lengthening of the process.

d) Specific assumptions

1) The moderator density coefficient is set to the maximum absolute value to promote the nuclear power increase.

2) The boron differential worth is set to the minimum absolute value.

3) The Doppler power coefficient, which tends to slowdown the nuclear power increase, is set to the minimum value to maximise the core power.

4) The shutdown margin is set to the minimum value to maximise the core power.

12.9.1.1.1.6 Results and Conclusion

a) From the initiating event to the controlled state

The analysis of this fault (see reference [41]) shows that the minimum DNBR is greater than the design limit described in Sub-section 5.6.2.1. The fuel temperature increase but does not challenge the fuel melting temperature limit.

Thus the acceptance criteria for this event are met.

b) From the controlled state to the safe state

The description is provided in Sub-chapter 12.10.

12.9.1.1.2 Large steam system piping break (State B)

12.9.1.1.2.1 Initiating Event

State B is an intermediate shutdown, with heat removal performed by the three SGs.

A steam system piping break in state B is classified as a DBC-4 event.

The steam system piping break induces an initial increase in steam flow which then decreases during the accident as the steam pressure falls.

The energy removed from the RCP [RCS] increases, causing a decrease in RCP [RCS] coolant temperature and pressure, which leads to core overcooling and an insertion of positive reactivity caused by negative moderator temperature coefficient.

12.9.1.1.2.2 Acceptance Criteria

The safety criteria and decoupling criteria are the same as those described in Sub-chapter 12.9.1.1.1.2 for a steam piping break in state A.

12.9.1.1.2.3 Main Safety Functions

For this event, the following plant safety functions can mitigate the event:

a) RT is actuated by the"Pressure drop of SG high 0" signal or the "Pressure drop of SG high 1" signal.

b) MSIV isolated after the"Pressure drop of SG high 1".

c) ARE [MFFCS] low load lines isolated after the"Pressure drop of SG high 2".

d) Safety injection triggered by the "Hot leg ∆Psat low 1".

e) ASG [EFWS] triggered by the"SG level (wide range) low 2".

12.9.1.1.2.4 Typical Events Sequences

a) From the initiating event to the controlled state

   1) Large breaks for which the "Pressure drop of SG high 1" or "Pressure drop of SG high 2" signals are transmitted:

      Initially, the fault leads to the secondary side depressurisation. RT and the closure of all MSIVs are triggered by the "Pressure drop of SG high 1" signal. ARE [MFFCS] low load lines is isolated on "Pressure drop of SG high 2" signal. The "Hot leg ∆Psat low 1" signal will trigger the safety injection (RIS [SIS]). After the ARE [MFFCS] low load lines have been isolated, the "SG level (wide range) low 2" signal will initiate the ASG [EFWS] for the affected SG.

      The RCP [RCS] cooldown induces positive reactivity in the core, and the reactor may return to criticality. However, the Doppler effect can limit the power excursion.

The RIS [SIS] supplies sufficient boron to compensate for the reactivity insertion, bringing or maintaining the core sub-critical.

Thereafter, the controlled state is reached, where the heat removal is ensured by the VDA [ASDS] of the unaffected SGs, and the feedwater is supplied by the ASG [EFWS].

2) Small breaks for which the "SG pressure drop high 1" and "SG pressure drop high 2" signals are not generated:

Initially, the fault leads to secondary side depressurisation, which is not rapid enough to trigger the "SG pressure drop" signals. The MSIVs are not closed and the ARE [MFFCS] low load is not isolated since the "SG pressure drop high 1" and "SG pressure drop high 2" signals have not been generated and the "low SG pressure" signals are deactivated.

Under these conditions, even if all SGs cool down the RCP [RCS] with a maximum rate, the SG pressure drops signals will not be triggered. Due to a high shutdown margin and an initial high core boron concentration, core damage and a DNB event is prevented.

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, the affected SG will be isolated when the first operator action happens. After that, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable

12.9.1.1.2.5 Results and Conclusion

This fault is not quantitatively analysed as the consequences of large breaks are bounded by the analysis in State A since the initial sub-criticality margin and the initial boron concentration are higher, which weakens the moderator effect. In addition, events with small breaks are not onerous compared to those with large breaks. Thus, the acceptance criteria for both the short term and long term under a Steam System Piping Break [State B] event can be met.

12.9.1.2 Inadvertent Opening of an SG Relief or Safety Valve (State B)

12.9.1.2.1 Initiating Event

The inadvertent opening of an atmospheric steam dump train (VDA [ASDS]) or a safety valve during shutdown conditions is caused by the spurious opening of a VDA isolation valve, spurious opening of a main steam safety valve or by a main steam safety valve seizing open. This leads to an increase in the heat removal from the RCP [RCS], and then causes the RCP [RCS] to overcooling. The core power will increase due to the negative moderator temperature coefficient. This event could lead to the occurrence of DNB and subsequently to fuel damage.

12.9.1.2.2 Acceptance Criteria

The inadvertent opening of an SG relief or safety valve (during shutdown conditions) is defined as a DBC-4 condition.

The acceptance criteria to be considered for this event are: Fuel integrity shall be ensured (no DNB and no fuel melting).

12.9.1.2.3 Main Safety Functions

For this event, the following plant safety functions can mitigate the event:

a) MSIV isolated after the “SG pressure low 1” signal or “Pressure drop of SG high 1” signal.

b) ARE [MFFCS] full load line isolated after the “Pressure drop of SG high 1” signal.

c) ARE [MFFCS] low load lines isolated after the “Pressure drop of SG high 2” signal.

d) Safety injection triggered by the “Pressuriser pressure low 3” signal.

e) ASG [EFWS] triggered by the “SG level (wide range) low 2” signal.

12.9.1.2.4 Typical Events Sequences

a) From the initiating event to the controlled state

The inadvertent opening of an atmospheric steam dump train (VDA [ASDS]) or a safety valve during shutdown conditions leads to a steam flow increase. Then the increase in heat removal from the RCP [RCS] caused by the increase in the steam flow causes coolant temperature and pressure to decrease. The temperature decrease during the transient is not as severe as the temperature decrease caused by the main steam line break (MSLB).

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, the affected SG will be isolated when the first operator action happens. After that, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray and the PSVs can be used when the pressuriser sprays are unavailable.

12.9.1.2.5 Results and Conclusions

a) From the initiating event to the controlled state

This fault is covered by the consequences of the MSLB accident. Based on MSLB analysis, neither fuel damage nor DNB happens during this transient. Thus, the decoupling criterion is met.

b) From the controlled state to the safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.9.2 Decrease in Heat Removal by the Secondary System

12.9.2.1 Large Feedwater System Piping Break (State A\B)

12.9.2.1.1 Initiating Events

A large feedwater system piping break is defined as a feedwater line break (FLB) which is large enough to prevent the feedwater from reaching the SGs. If the break is postulated to be between the check valve and the SG, the fluid in this SG may also be discharged through the break, resulting in depressurisation and a reversal of steam flow from the two unaffected SGs to the affected SG.

Depending on the break size and the plant operating conditions at the time of break, this event could either cause RCP [RCS] overcooling (by excessive steam discharge through the break) or RCP [RCS] overheating (by excessive liquid discharge through the break).

During the feedwater system piping break transient, saturated water could leak from the break. In a “Large steam system piping break” transient, saturated steam is released. Since saturated steam can remove more heat from the primary circuit than saturated water, overcooling from a “Large steam system piping break” transient is the more severe case and thus, the analysis of the former can bound the effects of RCP [RCS] overcooling caused by a feedwater system piping break. Therefore, only the RCP [RCS] heat-up effects of a feedwater line rupture are evaluated in this section.

12.9.2.1.2 Acceptance Criteria

The feedwater line break accident is classified as a DBC-4 event.

The aim of the study is to demonstrate that the core and reactor coolant system integrity are maintained during the transient.

12.9.2.1.3 Main Safety Functions

a) Reactor trip

Any of the following signals can trigger the reactor trip:

1) Pressuriser pressure high 2;

2) SG level (narrow range) low 1;

3) Overtemperature ΔT;

4) Pressuriser pressure low 2.

b) ASG [EFWS] is actuated after the “SG level (wide range) low 2” signal;

c) Main steam isolation can be triggered by any of the following signals:

1) SG pressure low 1;

2) Pressure drop of SG high 1.

12.9.2.1.4 Typical Events Sequences

a) From the initiating event to the controlled state

After the feedwater line break, the water levels in the unaffected SGs will decrease before the affected SG has been isolated, leading to primary side heating. Several RT signals can be triggered by this accident, after which the primary temperature and pressure continue to increase due to the core decay heat. Bulk boiling may occur and the pressuriser may be filled.

However, RCP [RCS] pressure can be limited by opening the PSVs. Then the main steam line will be isolated and the ASG will be actuated. Therefore, the residual heat can be continuously removed and the reactor can be taken to the controlled state.

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, the affected SG will be isolated when the first operator action happens. After that, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;
2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;
3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray and the PSVs can be used when the pressuriser sprays are unavailable.

12.9.2.1.5 Analysis Assumptions

a) Studied case

The break is assumed to be the in the most conservative area, corresponding to the most conservative rupture.

Main feedwater to all SGs is assumed to be isolated after the break occurs. All main feedwater flows out through the break.

Saturated liquid is assumed to be released through the break until all the water is drained from the affected SG, which maximises the loss of secondary fluid and thus minimises the heat removal capability of the affected SG.

b) Initial state

1) The initial power is set to full power plus uncertainty;

2) The initial coolant temperature is set to its nominal value plus uncertainty;

3) The initial PZR pressure is set to its nominal value minus uncertainty;

4) The initial pressuriser level is set to its nominal value minus uncertainty;

5) The initial SG water level is assumed to be the nominal level minus uncertainty.

c) Single failure

The single failure postulated is where one ASG [EFWS] is unavailable. Moreover, the ASG [EFWS] flow is set to the minimum value.

d) LOOP

The impact of the loss of offsite power (LOOP) is considered in this analysis.

e) Other assumptions

1) After the reactor coolant pumps trip caused by LOOP due to turbine trip, heat transfer is ensured through natural circulation;

2) No credit is taken for heat transfer to the metal of reactor coolant system during the reactor coolant system heat-up;

3) Only one ASG [EFWS] pump supplies water to one unaffected SG with a minimum flowrate.

12.9.2.1.6 Result and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see reference [42]) shows that:

1) Bulk boiling does not occur in the RCP [RCS];

2) There is no risk of the core becoming exposed.

Thus, the acceptance criteria for this event are met.

b) From the controlled state to the safe state

The description is provided in Sub-chapter 12.10.

12.9.2.2 Long Term LOOP (State C)

12.9.2.2.1 Initiating Event

The loss of offsite power (LOOP) may be caused by the following events:

a) A complete loss of the offsite grid;

b) A loss of the onsite alternating current distribution system;

c) An external power grid disturbance (degraded voltage or frequency).

LOOP can lead to the loss of power supply to all RCPs, feed water pumps, condensate pumps and RIS [SIS] pumps. As a consequence, the capacity of heat removal from the reactor core reduces, causing overheating on the primary side.

During normal conditions in State C, plant cooling is ensured by the RIS [SIS] in RHR mode. State C contains three sub-states:

a) When the RCP [RCS] temperature is between 100℃ and 140℃,RIS [SIS] train A and train B are in service (state C1);

b) When the RCP [RCS] temperature is lower than 100℃,RIS [SIS] train C is additionally connected (state C2);

c) When the RCP [RCS] temperature is below 60℃,two RIS [SIS] trains are sufficient to maintain the current RCP [RCS] temperature so the third train is left on standby (state C3).

12.9.2.2.2 Acceptance Criteria

For this event, the acceptance criteria are as follows:

a) Core water inventory is stable.

b) Heat removal is ensured in the long term.

12.9.2.2.3 Main Safety Functions

In state C, automatic actuation of EDGs following the LOOP signal will initiate the supply of electricity to the RIS pumps and thus ensure the recovery of RIS in RHR mode.

a) The safety injection system can be actuated by any of the following signals:

   1) Hot leg ΔPsat low 1;

   2) RCP [RCS] loop level low 1.

b) The RIS pumps in RHR mode trip is actuated on any of the following signals:

   1) Hot leg ΔPsat low 2;

2) RCP [RCS] loop level low 2.

12.9.2.2.4 Typical Events Sequences

The LOOP causes the failure of the power supply to all the RCPs, as well as the temporary loss of the heat removal function performed by the RIS in RHR mode.

The power supply to the RIS in RHR modes is restored by the automatic start-up of EDGs following the LOOP, thus the RHR function is re-established in the long term and the plant reaches safe state.

12.9.2.2.5 Analysis Assumptions

The main assumptions are listed as follows:

a) The initial primary temperature is the maximum value.

b) The maximum decay heat is used, and the residual power is assumed to be constant during the accident.

c) The single failure is assumed to be the failure of one EDG.

12.9.2.2.6 Result and Conclusion

The detailed analysis of this fault (see reference [43]) shows that the remaining RIS [SIS] train in RHR mode is able to remove the core decay heat in state C.

Thus the acceptance criteria for this event are met.

12.9.3 Decrease in Reactor Coolant System Flowrate

12.9.3.1 Reactor Coolant Pump Seizure (Locked Rotor) or Reactor Coolant Pump Shaft Break (State A)

12.9.3.1.1 Initiating Event

This is defined as a seizure or a shaft break of the rotor for the reactor coolant pump under normal operation. A seized rotor or shaft break is due to mechanical failure. During the transient, the core flow reduction causes a rapid increase in reactor coolant temperature. This increase may result in fuel rods experiencing DNB, and subsequent fuel damage.

The result of a reactor coolant pump shaft break is bounded by locked rotor, as the coolant flow reduction rate is lower for the former. Therefore, only the seized rotor is considered further.

12.9.3.1.2 Acceptance Criteria

The reactor coolant pump locked rotor accident is classified as a DBC-4 condition. The following acceptance criteria are required:

a) The amount of fuel rods experiencing DNB must remain less than 10%;

b) The peak cladding temperature must remain less than 1482℃.

12.9.3.1.3 Main Safety Functions

Any of the following signals can trigger reactor trip to maintain core sub-criticality:

a) Nuclear power is higher than 10%FP

   1) Low flow rate in two primary loops;

   2) Opening of two primary coolant pump breakers;

   3) Low RCP speed in two loops.

b) Nuclear power is higher than 30%FP

   1) Low flow rate in one primary loop;

   2) Opening of one primary coolant pump breaker.

12.9.3.1.4 Typical Events Sequences

a) From the initiating event to the controlled state

At the beginning of the transient, the flow rates in the core and the affected loop decrease rapidly. The decrease in core flow will result in rapid rise in coolant temperature and pressure. When the primary flowrate reduces to the setpoint of the “Low flow rate in one primary loop” signal, reactor trip occurs, after which the turbine trip occurs, leading to the LOOP. Subsequently, the remaining two unaffected pumps begin to coast down and the main feedwater will be lost. The decay heat is removed via the VDA [ASDS], and the feedwater is supplied by ASG [EFWS]. The RCP [RCS] will remain stable and the controlled state is reached.

b) From the controlled state to the safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using the ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray and the PSVs can be used when the pressuriser sprays are unavailable.

12.9.3.1.5 Analysis Assumptions

a) Initial conditions

1) Initial reactor power is the nominal value plus uncertainty;

2) Initial average temperature of reactor coolant is the nominal value plus uncertainty;

3) Initial pressuriser pressure is the nominal value minus uncertainty.

b) Core parameters

1) Moderator density coefficient is set to the minimum absolute value;

2) Doppler power coefficient is set to the maximum absolute value;

3) The specific axial power distribution and radial power distribution will be adopted in the DNBR calculation.

c) Single failure

One of the measurement channels or sensors for the “Low flow rate in one primary loop” singal has failed.

d) LOOP

The impact of LOOP is considered in this analysis.

e) Specific assumptions

1) Pressuriser heater is unavailable;

2) The opening pressure of the PSV is set to the minimum value;

3) The RCCA rod with the maximum worth is assumed to be stuck out of the core;

4) In the fuel thermal transient analysis, it is assumed that DNB occurs at the beginning.

12.9.3.1.6 Results and Conclusion

a) From the Initiating Event to the Controlled State

The analysis of this fault (see reference [44]) shows that amount of fuel rods experiencing DNB remain less than 10%, and that the cladding temperature increase does not challenge the design limit.

Thus, the acceptance criteria for this event are met.

b) From the Initiating Event to the Controlled State

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.9.4 Reactivity & Power Distribution Anomalies

12.9.4.1 Spectrum of RCCA Ejection Accidents (State A)

12.9.4.1.1 Initiating Event

The rod cluster control assembly (RCCA) ejection accident is defined as the scenario where an RCCA is vertically ejected from the reactor core due to the mechanical failure of the RCCA drive mechanism housing at the top of the pressure vessel.

This accident leads to a rapid reactivity transient by inducing an uncontrolled positive reactivity, followed by a strong nuclear power increase and a power distribution disturbance. The magnitude of the inserted reactivity is related to the inserted depth of the ejected RCCA. The nuclear power increase is limited by reactivity feedback effects, such as the Doppler feedback effect caused by the fuel temperature increase. The energy emitted by the fuel is transferred to the primary coolant. This may result in Departure from Nucleate Boiling (DNB) among the adjacent fuel channels, and the potential for the fuel cladding failure.

12.9.4.1.2 Acceptance Criteria

Specific criteria are also considered in this study. These are as follows:

a) The enthalpy of fuel pellet must be less than design limit (942 J/g for non-irradiated fuel and 837 J/g for irradiated fuel);

b) The fuel cladding temperature must not be greater than design limit (1482℃);

c) The amount of fuel rods experiencing DNB must not exceed design limit (10%);

d) The melting fuel pellet amount at the hot spot must remain below the design limit (10%);

e) The coolant pressure peak must be less than design limit.

12.9.4.1.3 Main Safety Functions

a) At the beginning of the accident, the Doppler feedback effect limits the increase in core nuclear power. Then, the RT is actuated by the following signals:

   1) High neutron flux;

   2) High positive neutron flux rate.

b) PSVs are opened when the PZR pressure reaches the setpoint.

c) After reactor trip:

   1) Turbine trip and ARE [MFFCS] full load lines isolation.

   2) RCP [RCS] heat removal by the VDA [ASDS].

3) The SGs are fed by the ASG [EFWS].

12.9.4.1.4 Typical Events Sequences

a) From the transition to the controlled state

After the RCCA is ejected vertically from the reactor core, an uncontrolled positive reactivity insertion will be induced in the core, leading to a strong nuclear power increase and a power distribution disturbance. During the initial phase, the nuclear power increase will be limited by reactivity feedback effect, i.e. the Doppler feedback effect. Then, the reactor trip will be triggered by the “High neutron flux” or “High positive neutron flux rate” signals, causing a rapid decrease in nuclear power. The reactor core controlled state is reached by the actuation of the reactor protection system.

b) From the controlled state to safe state

The safe state is defined as a state where at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. To reach the safe state, operators perform primary cooldown and depressurisation according to specific rules:

1) Reactor coolant boration via the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable;

2) Reactor coolant cooldown via the secondary circuit using ASG [EFWS] and VDA [ASDS] by means of reducing the VDA [ASDS] opening pressure setpoint. The cooldown rate depends on the number of available RBS [EBS]. ASG [EFWS] tank capacity is designed to ensure that the RIS [SIS] can be connected in RHR mode before the ASG [EFWS] tank inventory is exhausted;

3) Reactor coolant depressurisation via the normal or auxiliary pressuriser spray, and the PSVs can be used when the pressuriser sprays are unavailable.

12.9.4.1.5 Analysis Assumptions

a) Calculations steps

The RCCA ejection analysis is divided into three stages:

1) Nuclear data calculation. The aim of this is to calculate the nuclear data such as the ejected rod worth, hot channel enthalpy rise factor $F_{\Delta H}$ and hotspot factor F_Q, based on the cases to be considered for the nuclear transient.

2) Nuclear transient for clad and fuel thermal analysis. The aim of this part is to verify if the cladding and fuel thermal criteria are met by evaluating the maximum clad temperature, the peak radial fuel enthalpy and the peak fuel temperature.

3) Nuclear transient for thermal hydraulic analysis. The aim of this part is to verify if the thermal hydraulic criteria are met. The input data such as thermal power and axial power distribution are obtained through transient analysis and the hot channel enthalpy rise factor is obtained through the nuclear data calculation.

b) Initial conditions

The inserted reactivity during the RCCA ejection accident is closely related to RCCA configuration and therefore to the power level. Thus, all the fuel cycles, the typical burnup and power levels (0%FP, 100%FP and other intermediate power levels) are considered.

c) Main assumptions

1) Delayed neutron fraction: the adopted delayed neutron fraction value is conservatively set as the minimum value.

2) Moderator temperature coefficient: the value is set to the minimum absolute value for each burnup.

3) Doppler temperature coefficient: minimum absolute value is adopted in the analysis.

4) Prompt neutron life: the minimum absolute value is applied.

5) Insertion of negative reactivity during RT: the most reactive RCCA is assumed to be stuck during RT. The control rod drop time, consistent with the safety shutdown during an earthquake, is used in the event analysis.

6) The RCCA ejection time is conservatively assumed, i.e. the RCCA is assumed to be ejected within 0.1s.

d) Single Failure

Single failure is applied to one “High neutron flux” (or on “High positive neutron flux rate”) signal (sensor or channel).

e) LOOP

The impact of the loss of offsite power (LOOP) is considered in this analysis.

12.9.4.1.6 Results and Conclusions

a) From the initiating event to the controlled state

The analysis of this fault (see reference [45]) shows that the calculated amount of fuel rods experiencing DNB in the worst case is less than the design limit. The maximum fuel temperature, maximum fuel enthalpy, and maximum cladding temperature are all under the design limit.

Thus the acceptance criteria for this event are met.

b) From the controlled state to the safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.9.4.2 Boron Dilution due to a Non-isolatable Rupture of a Heat Exchanger Tube (State C\D\E)

12.9.4.2.1 Initiating Event

This is hypothesised to be a homogenous boron dilution event, resulting from a non-isolatable reverse leak of water from the RRI [CCWS] to the RCP [RCS] or other connecting systems through damaged heat exchanger tubes. The water finally injected into RCP [RCS] can result in a rise in core reactivity and subsequent core criticality.

12.9.4.2.2 Acceptance criteria

This is classified as a DBC-4 event. The acceptance criterion is that the operator has more than 60 minutes to perform dilution source isolation before the reactor core returns to criticality.

12.9.4.2.3 Main Safety Functions

The operator will be informed of the dilution caused reactivity increase by the "shutdown high neutron flux" alarm (3Φ alarm) initiated by the source range detectors. This alarm is displayed in the reactor building and in the main control room and will be triggered when the threshold is exceeded.

The most pessimistic scenario corresponds to a ¾ loop operation level where the total coolant inventory is low.

12.9.4.2.4 Analysis Method and Assumptions

a) Analysis Method

For each case studied, the time interval between the beginning of boron dilution and the return to criticality is calculated.

Results show that the time intervals are compatible with the following movements:

1) Boron dilution alarm ("shutdown high neutron flux" alarm initiated by the source range detectors);

2) Isolation of the dilution source by the operator from the main control room or by local action outside of the main control room.

b) Analysis assumptions

1) Continuous mixing in the reactor vessel is maintained by one residual heat removal pump. The analysis uses a maximum dilution flow rate of 14 t/h;

2) The initial boron concentration is the minium boron concentration of the in-containment refuelling water storage tank: enriched boron of 1300 ppm with boron-10 abundance of 35%;

3) The volume of borated water in the reactor is 110 t at ¾ loop operation level

(pessimistic case).

12.9.4.2.5 Results and Conclusions

The minimum time interval between the alarm (“shutdown high neutron flux” alarm) and a subsequent return to criticality is analysed for all the calculated cycles (see Reference [46]). This time interval is greater than the maximum grace period of one hour from the receiving boron dilution alarm to operator action (i.e. isolation of the dilution source executed from the main control room or by local action outside of the main control room).

Since the RIS [SIS] in RHR mode remains in operation, a safe state is reached as soon as the dilution source is isolated, signifying that the core will remain sub-critical.

The analysis performed shows that the safety of the plant can be ensured under during this accident.

12.9.5 Decrease in Reactor Coolant System Inventory

12.9.5.1 SGTR (two tubes in one SG) (State A)

12.9.5.1.1 Initiating Event

This sub-chapter describes the thermal-hydraulic analysis of a steam generator tube rupture event (two tubes in one SG). A steam generator tube rupture event (two tubes in one SG) is defined as the double-ended guillotine rupture of two steam generator (SG) tubes in one SG.

This accident will lead to an increase in the radioactivity in the secondary system due to the leak of radioactive coolant from the reactor coolant system (RCP [RCS]). If the loss of offsite power or a failure of the condenser steam dump system occurs during the event, discharges of steam or liquid from the main steam safety valve (MSSV) and/or the VDA [ASDS] will lead to a direct activity discharge to the atmosphere since the damaged steam generator has been contaminated. The radioactivity of the primary side coolant is due to corrosion and fission products generated by the continuous operation of the reactor with a limited number of damaged fuel rods.

The probability and risk of a steam generator tube rupture (SGTR) event is reduced through the following precautions:

a) High ductility of the SG tube material;

b) Blowdown system location at the bottom of SG tube bundle to prevent solid deposits on SG tube plate;

c) Chemically conditioned secondary water to protect SG tubes from corrosion;

d) Prevention of projectiles from originating from the main feedwater;

e) Designation of SG support plates to prevent tube damage and pipe whip of neighbouring tubes;

f) Continuous monitoring and control of secondary side activity.

The SGTR of one tube is classified as a DBC-4 event.

The cases studied in this sub-chapter correspond to the double-ended guillotine rupture of two tubes in one SG, which allows unimpeded blowdown from both ends of the tube.

The rupture is located in the lower part of the SG tubes bundle, close to the tubesheet, on the cold side. This location maximises the SGTR leak flowrate. For the DBC-4 transients, a LOOP is superimposed on the accident, if conservative.

This sub-chapter aims to quantify the maximum radioactivity release to the environment. If overfilling occurs, contaminated water will be directly released to environment and the radioactivity release will be significant. The SG overfilling assessment (see Reference [47]) demonstrates that even under the most onerous conditions, overfilling will not occur. Without overfilling, the radioactivity release is carried by steam released to the environment. The case which analyses the most pessimistic radioactivity release is presented hereafter.

12.9.5.1.2 Acceptance Criteria

The consequences of the SGTR (one tube) are analysed with respect to the following decoupling criteria:

a) There shall be no core damage;

b) There shall be no water flowing through the MSSV, to prevent the MSSV seizing open;

c) The plant shall be brought to and maintained in safe state.

The radiological criterion is the dose equivalent released to the environment and will be discussed in Sub-chapter 12.11.

12.9.5.1.3 Main Safety Functions

In the event of SGTR, automatic protection systems and operator actions are aimed at tripping the reactor, removing the RCP [RCS] residual heat, stopping the leak flow from the primary to the secondary side and bringing the plant to safe state.

a) FC1 system/function

   1) SGTR detection following “High activity in the VVP KRT [PRMS]” signal:

      On high activity detection in the VVP KRT [PRMS], an alarm is actuated.

   2) Reactor trip on “Pressuriser pressure low 2” signal:

      RT is triggered by the “Pressuriser pressure signal low 2” signal. The related uncertainties are taken into account to bring forward the RT. This maximises the release of radioactive steam to the environment on VDA [ASDS] opening.

   3) Turbine trip:

      The RT signal triggers the turbine trip.

   4) ARE [MFFCS] full load line isolation:

      Following an RT signal, the ARE [MFFCS] full load line for all SGs is isolated. In a penalizing way, the ARE [MFFCS] low load line is supposed to be isolated by the RT signal.

   5) APG [SGBS] isolation:

APG discharge is isolated when the ASG is actuated.

6) VDA [ASDS]:

The VDA [ASDS] opening setpoint is minimised on the SGa and maximised on the unaffected SGs to maximise the steam release from the SGa before the end of MCD.

At the beginning of the operator actions, the set value of the SGa VDA [ASDS] is manually increased above the MHSI injection head and below the MSSV opening set value, i.e. { }.

7) Safety injection:

SI is actuated by the "Pressuriser pressure low 3" signal or "SG level (narrow range) high 2" signal. This signal initiates a medium-pressure rapid cooldown. The maximum MHSI flowrate curve is chosen to maximise the pressure difference between the RCP [RCS] and the SGa.

8) Medium pressure rapid cooldown:

Following the SI signal, a medium pressure rapid cooldown (performed by all SGs, including the SGa) is initiated with a rate of { }, so as to cool down the primary circuit.

9) ASG [EFWS] startup:

The ASG [EFWS] is actuated automatically by the "LOOP and SI" signal or by "SG level (wide range) low 2" signal. Only two unaffected SGs are fed, since the other ASG [EFWS] pump is affected by the single failure. After ASG [EFWS] actuation, the SG level in the unaffected SGs is maintained at the reference water level.

The ASG [EFWS] is actuated with its minimum flowrate.

During the long-term phase, the ASG [EFWS] is manually controlled by the operator to maintain the SG levels in the two unaffected SGs at their nominal values.

10) MSIV closure:

MSIV closure is manually initiated when one SG becomes unavailable. All the main steam lines are then isolated.

b) Other systems:

SG level control is in operation from accident initiation and keeps operating normally until RT. This maximises the activity released from the SGa.

c) Operator actions

The main operator actions include:

1) SGa isolation:

After SGTR detection, the operator isolates the SGa (if not already done): ARE [MFFCS] and ASG [EFWS] isolation, MSIV closure and increase in the VDA [ASDS] opening set value.

2) RCP [RCS] boration:

The RBS [EBS] pumps are started by the operator at the beginning of the RCP [RCS] cooldown. The RBS [EBS] injection is stopped when the primary cold shutdown state boron concentration is reached.

3) RCP [RCS] cooldown by unaffected SGs:

The cooldown is performed via the VDA [ASDS]. The cooling rate is { }.

4) MHSI stop:

Two of the three MHSI pumps are stopped. The last MHSI pump stops when the core outlet temperature TRIC is reduced to { }.

5) RCP [RCS] and SGa depressurisation:

During the RCP [RCS] depressurisation, the accumulators are isolated when the RCP [RCS] pressure has decreased below { }.

RCP [RCS] is then depressurised until the low head safety injection (LHSI) injection pressure is reached (RCP [RCS] pressure is stabilised). Because of LOOP, the normal spray is not available. RCP [RCS] depressurisation is then realised by means of opening the VDA [ASDS] of the SGa.

6) RIS [SIS] in RHR mode connection:

The connection of the RIS [SIS] in RHR mode guarantees heat removal and core long term cooling.

12.9.5.1.4 Typical Events Sequences

A typical sequence, the most likely to occur during the transient, is described hereafter. Within this sub-chapter, the following description considers only the full power condition for this is the most onerous case for radiological release, and the overfilling case is described in Reference [47].

The sequence of events consists of two phases: the short-term phase, until leak elimination, and the long-term phase, where the plant is operated from leak elimination to the safe state.

a) From the initiating event to the leak elimination (short term)

1) From the initiating event to the controlled state

The controlled state for an SGTR is defined as a state where the core coolant inventory remains stable and residual heat removal can be ensured via the SGs.

At the beginning of this event, primary coolant leaks to the secondary side through the break, which leads to the contamination of the affected SG. Meanwhile, the primary pressure decreases. The reactor trip signal triggered depends on the initial conditions. For full power operation: The SGTR can be identified by the operator on receipt of the "High activity in the main steam system (VVP) radiation monitoring system (KRT)" signal. Reactor trip is triggered by the "Pressuriser pressure low 2" signal. Turbine trip and isolation of the ARE [MFFCS] full and low load lines for all SGs are initiated following the RT signal.

After RT and turbine trip, the secondary pressure increases and rapidly reaches the VDA [ASDS] setpoint if GCT [TBS] is unusable. Contaminated steam is thus released to the environment and decay heat is removed.

The continuous leak to secondary side and the decrease in decay heat after reactor trip lead to a primary depressurisation. It is likely the "Pressuriser pressure low 3" signal will be triggered. Within this sub-chapter, the following description assumes the break is large enough to lead to the "Pressuriser pressure low 3" signal.

Following the safety injection signal triggered by the "Pressuriser pressure low 3" signal, the Medium pressure rapid cooldown (MCD) is actuated. {

} The medium head safety injection (MHSI) pumps are actuated following the SI signal and start injecting when the primary pressure is lower than their injection head. The MHSI injection flow can compensate the leak flow from an SGTR and the controlled state is reached.

2) From the controlled state to the leak elimination

To stop the leakage, the operator isolates the SGa both on steam side and on feed side. In other words, the operator closes the ASG [EFWS] and the MSIV. In addition, the VDA [ASDS] setpoint for the SGa is adjusted up to a value between the MHSI injection head and MSSV setpoint in order to limit the radioactivity release.

The injection of MHSI maintains the primary pressure at a stable level. To reduce the leakage flow, the operator shuts down two of the three MHSI pumps (with the remaining left operating). Due to the isolation of the SGa, the pressure of SGa increases until it reaches primary pressure level and the leak is stopped. Before the leak is stopped, the SGa is not overfilled and only steam is released to the environment.

b) From the leak elimination to the safe state (long term)

The safe state for an SGTR is defined as a state when the RIS [SIS] train is connected to the RCP [RCS] in residual heat removal (RHR) mode and the SGa remains isolated. The operator performs a primary cooldown and depressurisation to reach the RHR connection conditions, which are:

1) RCP [RCS] hot leg pressure < { };

2) RCP [RCS] hot leg temperature < { };

3) RCP [RCS] hot leg saturation margin (ΔTsat) and Reactor Pressure Vessel Level (RPVL) consistent with RIS [SIS] in RHR mode suction conditions from the hot leg.

In order to cool down the primary coolant, the operator uses the RBS [EBS] to compensate for the reactivity insertion resulting from the RCP [RCS] cooldown. Unaffected SGs and the MHSI are used to cool down the primary side at a rate of { } with two or three RBS [EBS] trains in operation or at a rate of { } with one RBS [EBS] train in operation.

When the primary temperature is lower than { }, the last MHSI injection is stopped and the operator prepares to perform final depressurisation via the SGa. Prior to that, the operator shall confirm that the SGa level is lower than the relevant limits. Otherwise, the partner SG shall be isolated, with its level limited to a lower value so that the SGa water inventory can be transferred to the partner SG via the steam generator blowdown system (APG [SGBS]) transfer line. The RIS [SIS] in RHR mode can finally be connected and the safe state is reached.

12.9.5.1.5 Analysis Assumptions

a) Single failure

The failure of an ASG [EFWS] pump is assumed. This assumption worsens the radioactive release from the SGa as it increases the risk of heat transfer tube exposure.

b) Initial state

The conditions for the initial state are chosen to maximise the RCP [RCS] heat to be removed after RT and to make the SGa tubes exposed as soon as possible.

c) Neutronic parameters

Core power is assumed constant at 102%FP until RT. Following RT, the maximum decay heat curve with an uncertainty of 1.645σ is used.

d) Loss of offsite power (LOOP)

The LOOP at turbine trip is assumed to occur if it is conservative.

e) Other assumptions: operator actions

The first operator action is assumed to be performed 30 minutes after the “High activity in the VVP KRT [PRMS]” signal. The first local manual operator action is assumed to be performed one hour after this signal.

For long-term mitigation, the operator actions aim at reaching the safe state.

12.9.5.1.6 Result and Conclusion

a) Results

The detailed analysis of this fault (see Reference [47]) shows that the controlled state can be reached after injection by the MHSI and the leak can be stopped by operator actions.

The RIS [SIS] in RHR mode connection conditions are met after SGTR initiation. Once the RHR is underway, the safe state is reached. The description for long-term analysis is provided in Sub-chapter 12.10.

b) Conclusion

As the analysis shows, the reactor can be taken to the safe state. Radiological consequences are analysed in Sub-chapter 12.11.

12.9.5.2 Large Break (Loss of Coolant Accident) (LB-LOCA) (up to double-ended break) (State A)

12.9.5.2.1 Initiating Event

A large break loss of coolant accident (LB-LOCA) is defined as an accident in which a large break occurs on the pipes of the reactor coolant system or on the lines connected to this system before the first isolation valve.

A LB-LOCA at power induces a loss of primary coolant. It results in an abrupt decrease in the RCP [RCS] pressure and in the pressuriser (PZR) level with a possible core heat up due to the lack of heat removal.

The consequences of this event are considered in the plant design and can be managed with the proper acceptance criteria.

12.9.5.2.2 Acceptance Criteria

The LOCA analyses should meet the following acceptance criteria:

a) The peak cladding temperature must remain lower than 1204℃;

b) The maximum cladding oxidation must remain lower than 17% of the cladding thickness;

c) The maximum hydrogen generation must remain lower than 1% of the amount that would generate if the whole active part of the cladding had reacted;

d) The ability to cool the core geometry shall be retained, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling;

e) Long-term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed.

12.9.5.2.3 Main Safety Functions

For this event, the following plant safety functions can mitigate the event:

a) Reactor trip (RT) triggered by the “Pressuriser pressure low 2” signal.

b) Turbine trip and ARE [MFFCS] full load lines isolation after RT.

c) The SI signal is triggered by the “Pressuriser pressure low 3” signal. In case of a LOOP, the RCP [RCS] system is assumed conservatively to be injected by the MHSI after the actuation of the SI signal, and LHSI after the SI signal. Meanwhile, a loss of the SI flowrate in the affected loop is also postulated.

d) The main feedwater system is isolated after the SI signal

e) The emergency feedwater system is put into operation after the SI signal.

f) The accumulator will inject into the RCP [RCS] automatically while the RCP

[RCS] pressure is less than the initial accumulator pressure.

12.9.5.2.4 Typical Events Sequences

The description presented hereafter is a typical sequence, i.e. the most likely to occur during the transient, and both operational and safety system actions are mentioned.

The sequence of events consists of two phases: the short-term phase until reaching of the controlled state by use of automatic actions, and the long-term phase when the plant is operated from the controlled to the safe state.

When a break occurs on the RCP [RCS], the moderator density decreases sharply due to the system depressurisation and formation of steam in the core, thereby it causes an abrupt decrease of the reactivity. Following that, the core neutron reaction stops and the instantaneous power reduces rapidly. The core power is then reduced to a shutdown level by the core void effect, reactor trip (RT) and borated water injected via the accumulator and SI pumps.

During the blowdown phase, a rapid depressurisation of primary side to the saturation pressure happens and produces liquid flashing and coolant vaporisation. An excursion of the fuel cladding temperature may be induced due to departure from nucleate boiling (DNB) in core. The competition effect between the break flowrate at the pressure vessel side and pump flowrate shall determine the core flowrate at this phase. Initially, the break flowrate is greater than the total flow rates of the two intact cold legs driven by pumps, leading to a loss of core coolant from the bottom and top of the core simultaneously. Because of a rapid RCP [RCS] depressurisation, the break flowrate at the pressure vessel side, which is determined by the upstream fluid pressure and enthalpy under choked conditions, also decreases and becomes temporarily lower than the flow rate of the two intact cold legs. This induces a net positive flow rate at the inlets of the pressure vessel and core, and the core resumes some positive flow period. However, this situation is short-lived. The positive flow rates into the core and the core steam generation are also lowered due to an assumed loss of the RCP power supply and a quick decrease in the RCP efficiency resulting from the void fraction effect at the inlet of the RCP. Since the hydraulic resistance of the fluid passing through the core and downcomer is lower than that of the fluid passing through the steam generator tubes and the pump, the flow reverses in the core. The reverse core flow can contribute to cooling of the core.

During the blowdown phase, the RCP [RCS] inventory gradually turns entirely into steam, with only the lower plenum retaining some liquid. While the RCP [RCS] pressure decreases rapidly, it essentially reaches the initial accumulator pressure allowing their automatic discharge into the cold legs where the large flowrate of highly sub-cooled water condenses the steam and creates water plugs near the injection port. The water flow injected to the downcomer by the ECCS begins to fluctuate due to the water plug oscillation. The accumulator flowrate from the penetration to the downcomer is blocked by the counter-current steam flow in the core and downcomer, and the SI water in the accumulator is allowed to flow into the downcomer until the reverse steam flow is low enough at the end of the blowdown phase. This corresponds to the so-called “end of bypass”, i.e., the phase where the accumulator flowrate bypasses the core and flows directly into the cold leg break. At the end of this phase, the accumulator flowrate continuously injects into the vessel lower plenum up to the bottom of the core barrel, and then to the bottom of the active core section, within a few seconds after the end of depressurisation.

The downcomer level increases rapidly due to high ECCS flow rates from the accumulator in the earlier stage of reflooding, and water overflows from the break of the cold leg, which stabilises the water level in the cold leg. The reflooding phase begins as a large amount of subcooled liquid is injected to the core. In the next few seconds, the pressure peak appears because of the steam flashing, allowing the core water to discharge to the upper and lower plenums. This process may be repeated several times, which fluctuates the flow rates of the core and the descending plenum. As the energy stored in the core reduces, the fluctuation gradually slows down. At this phase, the core is cooled effectively and the quench front progresses quickly. The water drained from the core is brought to the upper plenum or flows into the inlet plenums of the cold leg and SGs. The steam flow in an intact loop cold leg is completely compensated by a large amount of subcooled SI water.

When all water from accumulators has been injected, the nitrogen in the water may enter the RCP [RCS] via the SI lines. At the later stage of reflooding, the pressure distribution between the core and downcomer depends upon the driven head and steam choking at the downcomer, which is the main phenomenon to determine the quench front progression. The heat and energy stored in the core continue to be released.

During the transient, the reactor trip (RT) is triggered by the “Pressuriser pressure low 2” signal. The RT signal triggers the turbine and isolates the main feedwater system full load lines.

The “Pressuriser pressure low 3” signal starts the safety injection (SI). The SI signal automatically starts the medium head safety injection (MHSI) and the low head safety injection (LHSI) pumps, and initiates the emergency feedwater system. The flows from the MHSI, accumulators and LHSI enter the core and cool the fuel cladding to stop its temperature rise and eventually achieve the controlled state. The controlled state corresponds to the achievement of a stable heat removal condition via the operation of the RIS [SIS] and the GCT [TBS], or VDA [ASDS] of all SGs, core sub-criticality and a stabilised or increasing core coolant inventory due to SI.

At the later stage of the accident, the LHSI shall be switched from the cold leg injection to the simultaneous injection to the cold leg and hot leg, so as to avoid the core boron concentration reaching the crystallisation limit. This can occur when steam leaks from the core in the long-term after the cold/hot leg breaks, and the core boron concentration increases due to safety injection of boron at the cold leg. To prevent an unacceptable increase, it is necessary to switch the LHSI pumps to the hot leg so that the borated liquid is discharged from the core. This shall be adopted as early as possible to prevent a containment overpressure resulting from the discharge of excessive steam from the break.

The long-term core cooldown can be guaranteed by the following systems: at least one RIS [SIS] train in SI mode, and the LHSI pumps injecting to both cold and hot legs.

The safe state is defined as a state for which the break flowrate is compensated by the RIS [SIS] flowrate with long-term core cooling ensured. In the later phase of LB-LOCA, the primary state does not meet the required RIS [SIS] in RHR mode connection conditions. The operator switches the LHSI pumps to the RCP [RCS] hot and cold legs injection at the later stage of the accident, and the MHSI pumps continue to inject into the cold leg. Long-term core cooldown can be guaranteed

12.9.5.2.5 Analysis Assumptions

a) Conservative calculation combination

The conservative calculation combination is determined through the sensitivity analysis, and the clarification of the conservative input parameters' effects on the results. The conservative calculation is divided into four categories, as shown below:

1) Operating point and initial conditions;
2) Accident conditions;
3) Safety systems;
4) Core data.

b) Initial Conditions

Important input parameters and initial conditions used in the analysis are presented as follows:

1) The initial operating power is full power plus the maximum uncertainty of the steady state power measurement;
2) The initial primary temperature is the rated temperature at power minus the maximum temperature surge and measurement uncertainty (the conservative assumption validated through the sensitivity analysis);
3) The initial pressure of the pressuriser is the rated value plus the maximum measurement uncertainty so as to delay the reactor trip and safety injection signals;

4) The primary flowrate is the thermal design flowrate;

5) The total core bypass flowrate takes the maximum value (6.5%) to minimise the flow rate passing through the core;

6) Plugging of 10% of the SG tubes is assumed for the primary mass calculation, while the heat exchange calculation at the primary and secondary sides of SGs does not factor in this assumption.

c) Accident Conditions

1) The break is assumed to be located in the cold leg between the pumps and the reactor pressure vessel.

2) Break size sensitivity analysis (up to double-ended cold leg rupture) are performed;

3) In general, LOOP is assumed following turbine trip. To be conservative, here LOOP is assumed to occur at the beginning.

d) Safety Systems

1) Safety injection

- According to the single failure (SF) criterion, a failure is assumed to occur on the emergency diesel generator (EDG) in one unaffected loop. The break is assumed to be located at the safety injection point. As a consequence, one RIS [SIS] train (one MHSI pump and one LHSI pump) and one ASG [EFWS] system are unavailable.

2) Accumulator

- The accumulator temperature is assumed to be the highest according to results of the sensitivity analysis.

e) Core Data

1) The enthalpy rise factor ($F_{\Delta H}$) is 1.65;

2) The core peaking factor (F_Q) is 2.45;

More detailed analysis assumptions are in Reference [48].

12.9.5.2.6 Results and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see Reference [48]) shows that the criteria are met.

b) From the controlled state to the safe state

In the long term of LB-LOCA, the operator brings the plant to a safe state through appropriate actions to ensure the break flowrate is compensated for by the RIS [SIS] flowrate with the long-term core cooling ensured. A description of the long-term analysis is provided in Sub-chapter 12.10.

Radiological consequences are analysed in Sub-chapter 12.11.

12.9.5.3 Intermediate Loss of Coolant Accident (LOCA) (State A\B)

12.9.5.3.1 Analysis of Intermediate Break Loss of Coolant Accident (State A)

12.9.5.3.1.1 Initiating Event

An intermediate break loss of coolant accident (IB-LOCA) is defined as an accident in which an intermediate break occurs on the pipes of the reactor coolant system or on the line in front of the first isolation valve connected to it. A risk of core uncovery exists, in which the efficiency of the Safety Injection system decreases and the core is unable to be maintained at its nominal level.

12.9.5.3.1.2 Acceptance Criteria

This accident is classified as a DBC-4 event. The safety criteria are the radiological limits for DBC-4 events.

The LOCA analyses should meet the following acceptance criteria:

a) The peak cladding temperature must remain lower than 1204℃;

b) The maximum cladding oxidation must remain lower than 17% of the cladding thickness;

c) The maximum hydrogen generation must remain lower than 1% of the amount that would generate if the whole active part of the cladding had reacted;

d) The ability to cool the core geometry shall be retained, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling;

e) Long-term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed.

12.9.5.3.1.3 Main Safety Functions

In this event, reactor protection is provided by the following signals and actions:

a) Reactor trip (RT) is triggered by the “Pressuriser pressure low 2” signal;

b) The safety injection system is actuated by:

   “Pressuriser pressure low 3” signal;

c) The RCPs are stopped by:

   “RCP ΔP low 1” signal cumulated with the “Safety injection” signal;

d) The injection of the RIS [SIS] accumulator to the RCP [RCS] is actuated when:

   The pressure of RCP [RCS] is lower than 4.5 MPa;

e) The following actions are induced by the reactor trip:

Closure of the turbine inlet valve;

f) The emergency feedwater system is actuated by:

"SG level (wide range) low 2" signal;

g) The main feedwater system is isolated by:

RIS [SIS] action signal or Reactor trip (RT).

12.9.5.3.1.4 Typical Events Sequences

a) Analysis from the initiating event to the controlled state

The IB-LOCA is mainly a gravity-driven accident, in which the RCP [RCS] discharges slowly with the evident formation of mixing layers over the RCP [RCS]. These mixing layers change over time, depending on the transient of the mass and energy mutual transfer. Two core heat-up events may occur. The first heat-up event results from the core level decrease and the formation of a loop seal, and can be mitigated by loop seal clearance during the accident. The second heat-up event results from the boiling and evaporation of the core coolant. In this event, the primary pressure drops to the accumulator setpoint, and the steam generated escapes from the system via the break. Core heat-up during this period can be mitigated by the reflooding of the accumulator injection system. The core temperature increase is potentially affected by various factors. These may be the break size, safety injection flowrate, bypass flowrate from the upper head to the descending section, hot rod burnup, etc.

The accident process can be divided into two phases according to the characteristics of the transient phenomenon:

1) Core heating resulting from the level decrease;

2) Core heat up resulting from the core coolant evaporation.

These two temperature rises are the most obvious upon the immediate trip of the RCPs after the RT signal. The accident process is complex if the RCPs keep running for some time after the RT signal. The loss of offsite power (LOOP) is not considered, and the pump trip signals of the RCPs are triggered by the "RCP ΔP low 1 and SI" signal.

Medium pressure rapid cooldown (MCD) is initiated on receipt of the safety injection (SI) signal (on "Pressuriser pressure low 3" signal) in all SGs. {

}

On SI signal, the medium head safety injection (MHSI) pumps are actuated but may not inject immediately. The MHSI pumps start injecting when the RCP [RCS] backpressure is below their injection head.

The controlled state is reached when MHSI is able to compensate for the break flowrate.

b) Analysis from controlled state to safe state

After the controlled state, the safety function of the plant has been recovered, but cannot be maintained for a long time due to the following reasons:

1) Water consumption in the ASG [EFWS] tank;
2) Increase in the pressure and temperature within the containment.

As a consequence, this state cannot be sustained and the plant needs to be brought to a safe state. To achieve this, the following conditions shall be met:

1) The core is kept sub-critical;
2) The RIS [SIS] flowrate is sufficient to compensate for the break flowrate;
3) The decay heat is removed via RHR, and the remaining decay heat is removed through the break flowrate;
4) Radioactive discharges are controlled within the limit of DBC-4.

Strategies used to meet the above conditions are:

1) Primary boration shall be used to maintain core sub-criticality;
2) Reduce the primary temperature;
3) Reduce the primary pressure;
4) Connection of RIS [SIS] in RHR mode;
5) Switchover to simultaneous injection to hot leg and cold leg (if necessary).

12.9.5.3.1.5 Analysis Assumptions

a) Initial condition assumptions

The break is assumed to be located on the cold leg of the reactor coolant system. Assumptions for the initial conditions are shown as follows:

1) The reactor initial power is the nominal power plus the maximum error of the steady state measurement;
2) The initial coolant flowrate is the thermal design flowrate;
3) The average temperature of the coolant is the rated value plus the maximum steady state control range and measurement error;
4) The initial pressuriser pressure is the rated value plus the maximum steady

state control range and measurement error;

5) The initial pressuriser level is the rated level at power minus maximum steady state uncertainties.

b) Single failure

Considering single failure, the flow in only one safety injection train (1MHSI+1LHSI) is taken into account for the RIS [SIS], and the 3rd MHSI/LHSI train is assumed lost at the break and not considered.

c) Accident conditions

1) The break is assumed to be located in the cold leg between the pumps and the reactor pressure vessel.

2) Break size sensitivity analyses (equivalent diameter up to 27.5 cm) are performed;

d) Core-related assumptions

In the system thermal-hydraulic analysis, the core is simulated as a typical core with a severe axial power distribution.

For the hot rod analysis, the core enthalpy rise factor $F_{\Delta H}$ is set at its maximum value (1.65), and the hot spot factor F_Q is set at its maximum value (2.45).

12.9.5.3.1.6 Result and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see Reference [49]) shows that the criteria are met.

b) From the controlled state to the safe state

The description from the controlled state to the safe state is provided in Sub-chapter 12.10, which demonstrates that the safe state can be achieved.

12.9.5.3.2 Intermediate Break Loss of Coolant Accident (State B)

12.9.5.3.2.1 Initiating Event

The intermediate break loss of coolant accident (State B) can be divided into two conditions regarding to the availability of accumulators.

For cases with the initial primary pressure higher than { }, the accumulators are not isolated. Here, the safety functions available are the same as those for cases in State A, while the residual heat in the primary coolant and the core power are both lower and so the analysis in Sub-section 12.9.5.3.1.1 bounds this condition.

For cases with the initial primary pressure lower than { }, the accumulators are isolated. This condition is analysed in this section. The following differences in terms of FC1 mitigation methods, when compared to state A, are considered:

a) The triggering signal of SI signal.

b) The isolation of accumulators.

12.9.5.3.2.2 Acceptance Criteria

This accident is classified as a DBC-4 event. The safety criteria are the radiological limits for DBC-4 events and the LOCA acceptance criteria described in Sub-chapter 12.5.1 are also applied.

12.9.5.3.2.3 Main Safety Functions

During this event, reactor protection is provided by the following signals and actions:

a) The RCPs are tripped on:

"ΔP low 1 over two RCPs" signal cumulated with the "Safety injection" signal;

b) The safety injection system is actuated by:

"Hot leg ∆Psat low 1" signal;

c) The emergency feedwater system is actuated by:

"SG level (wide range) low 2" signal;

d) The main feedwater system is isolated by:

1) RIS [SIS] action signal;

2) Reactor trip (RT).

12.9.5.3.2.4 Typical Events Sequences

a) From the initiating event to the controlled state

An IB-LOCA in normal shutdown state where the RIS [SIS] in RHR mode is not connected induces a loss of coolant inventory. As it is assumed not compensable by the RCV [CVCS], it results in an abrupt RCP [RCS] pressure and pressuriser level decrease with a possible core heat up.

The safety injection (SI) signal is emitted on "Hot leg ∆Psat low 1" signal, inducing the following actions:

1) Starting of medium head safety injection (MHSI) and low head safety injection (LHSI) pumps;

2) {

}

3) Containment isolation stage: the reactor coolant pressure boundary is isolated, in particular the RCV [CVCS] letdown line and the steam generator (SG) blowdown lines are isolated.

Reactor coolant pumps (RCPs) are tripped by the "ΔP low 1 over two RCPs" signal cumulated with the "Safety injection" signal.

As long as the SI flowrate does not compensate for the break flowrate, RCP [RCS] coolant water inventory continues to decrease. During this phase the break flow is subsaturated and eventually reaches saturation conditions.

The break flowrate decreases as the void fraction in the affected leg increases. Eventually, the break flow changes to single steam phase. The primary coolant inventory depletion stops when the SI flowrate compensates for the break flowrate. If the accident occurred before the isolation of the accumulators by the operator, they may discharge their content of borated water into the RCP [RCS]. Later on, the controlled state is reached. This corresponds to the achievement of stable heat removal conditions via the operation of MHSI and the GCT [TBS] or VDA [ASDS] of all SGs, core sub-criticality and a stabilised or increasing core coolant inventory through use of the SI. The feedwater supply is ensured by the SG main feedwater system (ARE [MFFCS]) or the emergency feedwater system (ASG [EFWS]) if the ARE [MFFCS] is not available.

b) From controlled state to safe state

The safe shutdown and the way to reach the safe state from the controlled state are the same as in state A, see Sub-section 12.9.5.3.1.4.

12.9.5.3.2.5 Analysis Assumptions

a) Initial condition assumptions

The break is assumed to be located in the cold leg of the reactor coolant system, with the accumulator having been isolated. Assumptions for the initial conditions are shown as follows:

1) The reactor initial power shall be the core power 5 hours after shutdown;

2) The initial coolant flowrate is the thermal design flowrate: 24000 m^3/h;

3) The average coolant temperature is 250℃;

4) The initial pressuriser pressure is 7.25 MPa;

5) The initial pressuriser level is 36%;

6) The core bypass flowrate is 6.5%.

b) Single Failure

Considering single failure, the flow in only one safety injection train (1MHSI+1LHSI) is taken into account for the RIS [SIS], and the 3rd MHSI/LHSI train is assumed lost at the break and not considered.

c) Core-related assumptions

In the system thermal-hydraulic analysis, the core is simulated as an average core with a severe axial power distribution.

In the hot rod analysis, the core enthalpy rise factor $F_{\Delta H}$ is set at its maximum value (1.65), and the hot spot factor F_Q is set at its maximum value (2.45).

12.9.5.3.2.6 Result and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see Reference [49]) shows that the criteria are met.

b) From the controlled state to the safe state

The safe state is reached with the operation in Sub-section 12.9.5.3.2.4. The description from the controlled state to the safe state is provided in Sub-chapter 12.10.

12.9.5.4 SB-LOCA, including a Break in the Emergency Boration System Injection Line (during Shutdown Conditions, and RIS [SIS] under RHR Operation Mode) (State C\D)

A small break loss of coolant accident (SB-LOCA) on the RCP [RCS] or on the RBS [EBS] injection line in normal shutdown state with RIS [SIS] under RHR Operation Mode (in state C), induces a loss of coolant inventory and an RCP [RCS] pressure decrease, thereby leading to a possible core heat-up.

During normal operation in states C and D, the cooling of the plant is performed by the RIS [SIS] in residual heat removal (RHR) mode. Primary fluid is extracted from the RCP [RCS] hot legs, cooled down and injected back into the RCP [RCS] cold legs through the low head safety injection (LHSI) pumps. The various states are described below:

a) When the RCP [RCS] temperature is between 100℃ and 140℃ (state C1), 2 trains of RIS [SIS] are used in RHR mode, and at least one RCP is running.

b) When the RCP [RCS] temperature is between 10℃ and 100℃ (state C2), 3 trains of RIS [SIS] are used in RHR mode and at least one RCP is running.

c) When the RCP [RCS] temperature is between 10℃ and 60℃ (state C3), 2 trains of RIS [SIS] are used in RHR mode. In this state, state C3a is defined as shutdown state with at least one RCP is running and state C3b is defined as shutdown state without RCP is running.

d) When the RCP [RCS] temperature is between 10℃ and 60℃ (state D), at least 2 trains of RIS [SIS] are used in RHR mode and all the RCPs stop.

12.9.5.4.1 Small Break (in Shutdown State, RIS [SIS] Connected in RHR mode) (Three RCPs are Running)

12.9.5.4.1.1 Initiating Event

A non-isolatable break with an equivalent diameter smaller than or equal to 5.0 cm (equivalent area smaller than or equal to 20 cm^2) on the RCP [RCS] or on the RBS [EBS] injection line in NS/RHR mode state C, induces a loss of coolant inventory and an RCP [RCS] pressure decrease, thereby leading to a possible core heat-up. This subsection addresses the analysis of a non-isolatable break with an equivalent diameter equal to 5.0 cm, in a cold leg on the RCP [RCS] in NS/RHR mode state C1. This location induces the most pessimistic consequences for core uncovery. Compared with the RCP [RCS] operating conditions, the primary initial temperature, pressure, and initial core power are lower under this condition, and the normal system protection function assumption will not be worse. Hence, the other cases' analysis results can be covered by the analysis specified in this subsection.

12.9.5.4.1.2 Acceptance Criteria

SB-LOCA in state C is classified as a DBC-4 event. The aim of the study is to demonstrate that core and reactor coolant system integrity are maintained even assuming a single failure in the protection or safeguard systems.

The LOCA analyses should meet the following acceptance criteria:

a) The peak cladding temperature must remain lower than 1204℃;

b) The maximum cladding oxidation must remain lower than 17% of the cladding thickness;

c) The maximum hydrogen generation must remain lower than 1% of the amount that would generate if the whole active part of the cladding had reacted;

d) The ability to cool the core geometry shall be retained, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling;

The long-term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed.

These criteria are met if the core remains covered. Moreover, it shall be demonstrated that the plant can be brought to a safe state.

12.9.5.4.1.3 Main Safety Function

In this condition, Safety injection startup is triggered by “Hot leg ΔPsat low 1”; RCP trip is triggered by “ΔP low 1 over two RCPs” signal cumulated with the “Safety injection” signal; RIS [SIS] pump is stopped by “Hot leg ΔPsat low 2”.

If the RIS [SIS] pumps in RHR mode are not stopped, the RIS [SIS] pumps in RHR mode ensure the RCP [RCS] heat removal. The MHSI compensates for the break flow and the RCP [RCS] water inventory is stabilised, enabling progression to the controlled state.

If the RIS [SIS] pumps in RHR mode are stopped, the RCP [RCS] heat is removed via the break and the pressuriser pressure decreases. As the pressure decreases, the MHSI begins to operate and the primary water inventory is compensated and stabilised, allowing the continuous removal of the core residual heat. This ensures that the controlled state is reached.

12.9.5.4.1.4 Typical Events Sequences

a) From the Initiating Event to the Controlled State

Following the break occurrence, RCP [RCS] pressure and PZR level decrease due to the loss of coolant water inventory. The Safety injection is triggered by “Hot leg ΔPsat low 1”. A few seconds later, RIS [SIS] pump is stopped by “Hot leg ΔPsat low 2”. RCPs trip are triggered by the “RCP ΔP low 1” and “SI” signals. After the delay, the MHSI starts to inject. As a result, the compensation for the break flowrate is ensured. With the trip of the RIS pumps in RHR mode, the RCP [RCS] temperature increases and the secondary pressure increases slowly until the VDA [ASDS] valve opening setpoint is reached. The VDA [ASDS] valve opens and removes heat from the RCP [RCS]. Then the heat removal is ensured by VDA [ASDS], break flow and MHSI. The water inventory is compensated by MHSI. Finally, the RCP [RCS] water inventory stabilises: the controlled state is reached.

b) From controlled state to safe state

The safe state is defined as a state for which the break flowrate is compensated by the RIS flowrate with the long-term cooling ensured.

1) If the RIS [SIS] pumps in RHR mode are not tripped:

Considering that the RIS [SIS] trains are operating in RHR mode, the safe shutdown is reached at the same time as the controlled state.

2) If the RIS [SIS] pumps in RHR mode are tripped:

The sequence of actions to be performed by the operator is the following:

- RCP [RCS] cooldown:

  The cooldown to RIS in RHR mode connecting conditions is performed via the secondary side by reducing the VDA [ASDS] opening pressure setpoint.

- Saturation margin (ΔT_{SAT}) restoration:

  If the RCV [CVCS] is available, the RCV [CVCS] pumps are started to help restore the saturation margin.

  If the saturation margin (ΔT_{SAT}) is restored, since the RCP [RCS] temperature is kept below { } and the RCP [RCS] pressure below { }, the RIS [SIS] trains in RHR mode can be re-started and the safe state is reached.

  If the saturation margin (ΔT_{SAT}) is not restored, the operator performs the water inventory restoration.

- Water inventory restoration:

  In addition to the MHSI injection, the available LHSI pumps on stand-by are started in SI mode to ensure the RCP [RCS] water supply. The RCP [RCS] heat removal is then ensured by the flowrate from the MHSI and LHSI pumps and through the break, and the safe state is reached.

12.9.5.4.1.5 Analysis Assumption

The initial state conditions correspond to state C1. The initial values of the thermal hydraulic parameters are set to be pessimistic regarding to the acceptance criteria. The initial core power is assumed at the beginning of state C1 (i.e. at the earliest 12 hours after reactor shutdown) and decreases according to the decay heat curve (2 σ) “B+C term”.

In this analysis, the most pessimistic single failure is the loss of one MHSI pump in an intact loop. The flow in only one safety injection train (1MHSI+1LHSI) is taken into account for the RIS [SIS], and the 3rd MHSI/LHSI train is assumed lost at the break and not considered. Preventive Maintenance is not considered.

12.9.5.4.1.6 Result and Conclusion

a) From the initiating event to the controlled state

The detailed analysis of this fault (see Reference [50]) shows that during the accident, the primary water inventory is maintained without exposing the core and the core heat is removed via the break and SI. The relevant safety acceptance criteria are thus met.

b) From the controlled state to the safe state

The description from the controlled state to the safe state is provided in Sub-chapter 12.10, which demonstrates that the safe state can be achieved.

12.9.5.4.2 Small Break (in Shutdown State, RIS [SIS] Connected in RHR Mode) (One RCP or No RCP is Running)

Compared with the RCP [RCS] operating conditions, the primary initial temperature, pressure, and initial core power are lower under this condition, and the normal system protection function assumption is the same as in Sub-section 12.9.5.4.1. The accident analysis results can be covered by the analysis specified in Sub-section 12.9.5.4.1.

12.9.5.5 RHR System Piping Break inside (outside) Containment (≤DN 250) (State C\D)

12.9.5.5.1 RHR System Break inside Containment (in Shutdown State)

12.9.5.5.1.1 Initiating Event

If a break occurs on the RIS [SIS] system inside containment, it leads to a loss of primary coolant inventory and a discharge of radioactive primary fluid into the containment. The studied accident is due to an isolatable break on one RIS [SIS] line during RHR mode operation, with an equivalent size smaller than or equal to DN250 (with a nominal diameter of 250 mm). The isolatable break inside containment can be located:

a) Downstream of the RIS [SIS] isolation valve which is the second closest to the RCP [RCS] on the RIS [SIS] suction line connected to an RCP [RCS] hot leg;

b) Upstream of the check valve which is the second closest to the RCP [RCS] on the RIS [SIS] injection line connected to an RCP [RCS] cold leg.

Austenitic stainless steels are used for the parts of the RCP [RCS] in contact with the primary coolant because of their high resistance to generalised corrosion at the service temperature and in conditions of cold shutdown. Precautions are taken to avoid any other localised corrosion, by conditioning of the primary coolant and appropriate chemical composition of the materials. With regards to pitting, the chloride content and oxygen content of the primary coolant is controlled to avoid this form of corrosion during service. This conditioning of the primary coolant also promotes resistance of the stainless steel to corrosion cracking.

The risk of breaks caused by corrosion on the RIS [SIS] is therefore reduced.

During normal operation in states C and D, the cooling of the plant is performed by the RIS [SIS] in residual heat removal (RHR) mode. Primary fluid is extracted from the RCP [RCS] hot legs, cooled down and injected back into the RCP [RCS] cold legs through the low head safety injection (LHSI) pumps. The various states are described below:

a) When the RCP [RCS] temperature is between 100℃ and 140℃ (state C1), 2 trains of RIS [SIS] are used in RHR mode, and at least one RCP is running.

b) When the RCP [RCS] temperature is between 10℃ and 100℃ (state C2), 3 trains of RIS [SIS] are used in RHR mode and at least one RCP is running.

c) When the RCP [RCS] temperature is between 10℃ and 60℃ (state C3), 2 trains of RIS [SIS] are used in RHR mode. In this state, state C3a is defined as shutdown state with at least one RCP is running and state C3b is defined as shutdown state without RCP is running.

d) When the RCP [RCS] temperature is between 10℃ and 60℃ (state D), at least 2 trains of RIS [SIS] are used in RHR mode and all the RCPs stop.

If a break occurs on the RIS [SIS] system inside containment, it may lead to a decrease in reactor coolant inventory, a discharge of radioactive primary fluid into the containment and potential core overheating.

12.9.5.5.1.2 Acceptance Criteria

The decoupling criteria for LOCA events:

a) The peak cladding temperature must remain lower than 1204℃;

b) The maximum cladding oxidation must remain lower than 17% of the cladding thickness;

c) The maximum hydrogen generation must remain lower than 1% of the amount that would generate if the whole active part of the cladding had reacted;

d) The ability to cool the core geometry shall be retained, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling;

e) The long-term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed.

12.9.5.5.1.3 Main Safety Functions

For this event, the following plant safety functions can mitigate the event:

a) Safety injection (SI) actuated on “Hot leg ΔPsat low 1” signal or on “Hot leg RCP [RCS] loop level low 1” signal.

b) The RCPs can be tripped by the “RCP ΔP low 1 and SI” signal.

c) The RIS [SIS] pumps can be tripped by the “Hot leg ΔPsat low 2” signal or “Hot leg RCP [RCS] loop level low 2” signal.

12.9.5.5.1.4 Typical Events Sequences

a) From the initiating event to the controlled state

At the beginning of this fault, the break causes a loss of primary coolant and results in an RCP [RCS] water inventory decrease. The SI signal is emitted on “Hot leg ΔPsat low 1” signal or on “Hot leg RCP [RCS] loop level low 1” signal. The SI signal emission induces the actuation of the safety injection system. The RCPs can be tripped by the “RCP ΔP low 1 and SI” signal. The RIS [SIS] pumps can be tripped by the “Hot leg ΔPsat low 2” signal or “Hot leg RCP [RCS] loop level low 2” signal:

1) If the RIS [SIS] pumps are not tripped:

The unaffected RIS [SIS] pumps ensure the RCP [RCS] heat removal. The MHSI then compensates for the break flowrate and the RCP [RCS] water inventory remains sufficient and is stabilised. This enables progression to the controlled state.

2) If the RIS [SIS] pumps are tripped:

Following the trip of the RIS [SIS] pumps, the decay heat can be removed by the break flow and the steam generators (SGs), depending on the break size and the plant state. The RCP [RCS] water inventory is controlled by the MHSI. This enables progression to the controlled state.

b) From controlled state to safe state

Since the break is inside containment, the break is not automatically isolated from the RCP [RCS]. Heat removal via the MHSI and break or SGs continues until the operator acts to take the plant to the long-term safe state.

Depending on the trip of the RIS [SIS] pumps, two methods to achieving the safe state are possible:

1) If the RIS [SIS] pumps have not tripped:

Considering the RIS [SIS] trains are operating in RHR mode, the safe state is reached at the same time as the controlled state.

2) If the RIS [SIS] pumps are tripped:

The sequence of actions to be performed by the operator is the following:

- RCP [RCS] integrity test / break isolation:

  Only for closed reactor pressure boundary and re-sealable states, the operator first starts the RCP [RCS] integrity test to identify which LHSI/RHR train is broken. After localisation, the operator isolates the break.

- RCP [RCS] temperature management

  In states C1 and C2, the RCP [RCS] is initially closed. The isolation of the break restores the saturation margin. The operator can use the available SGs to manage RCP [RCS] temperature to allow RIS [SIS] in RHR mode re-connection.

  In state C3a, the operator is assumed to use Pressuriser Safety Valves (PSVs) and LHSI to cool the RCP [RCS].

  In state C3b, since the temperature is lower than {   }, no operator action is required.

  In state D, the operator starts the MHSI and LHSI pumps to bring the

RIS to RHR mode.

- RIS [SIS] is connected to RCP [RCS] in RHR mode

When the connection conditions are reached, the operator connects the RIS [SIS] in RHR mode to the RCP [RCS], and the safe state is reached.

12.9.5.5.1.5 Analysis Assumptions

The studied cases correspond to an isolatable guillotine rupture of a RIS [SIS] pipe connected to the RCP [RCS] hot leg in state C1, inside the containment.

a) Single failure

The most pessimistic single failure is postulated: in the unaffected loop, only one safety injection train (MHSI) is taken into account for the RIS [SIS]. The 3rd MHSI/LHSI train is assumed lost at the break and not considered.

b) LOOP assumption

Because no reactor trip occurs, LOOP is not considered in this event.

12.9.5.5.1.6 Result and Conclusion

a) From the initiating event to the controlled state

An isolatable residual heat removal system break (≤DN250) located inside the containment in shutdown state with RIS [SIS] connected in RHR mode is classified as a DBC-4 event. The detailed analysis of this fault (see Reference [51]) shows that the core remains covered, so the acceptance criteria are not challenged.

b) From controlled state to safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.9.5.5.2 RHR System Line Break outside Containment (≤DN250) (in Shutdown State)

12.9.5.5.2.1 Initiating Event

If a break occurs on the RIS [SIS] system outside containment, it leads to a loss of primary coolant inventory and a discharge of radioactive primary fluid into the safeguard building. The studied accident is due to an isolatable break on one RIS [SIS] line during RHR mode operation, with an equivalent size smaller than or equal to DN250. The isolatable break outside containment can be located:

a) Downstream of the RIS [SIS] isolation valve second closest to the RCP [RCS] on the RIS [SIS] suction line connected to an RCP [RCS] hot leg;

b) Upstream of the check valve second closest to the RCP [RCS] on the RIS [SIS] injection line connected to an RCP [RCS] cold leg.

The precautions limiting the transient are the same as for a break inside containment, described in Sub-section 12.9.5.5.1.

During normal operation in states C and D, the cooling of the plant is performed by the RIS [SIS] in residual heat removal (RHR) mode. Primary fluid is extracted from the RCP [RCS] hot legs, cooled down and injected back into the RCP [RCS] cold legs through the low head safety injection (LHSI) pumps. The various states are described below:

a) When the RCP [RCS] temperature is between 100℃ and 140℃ (state C1), 2 trains of RIS [SIS] are used in RHR mode, and at least one RCP is running.

b) When the RCP [RCS] temperature is between 10℃ and 100℃ (state C2), 3 trains of RIS [SIS] are used in RHR mode and at least one RCP is running.

c) When the RCP [RCS] temperature is between 10℃ and 60℃ (state C3), 2 trains of RIS [SIS] are used in RHR mode. In this state, state C3a is defined as shutdown state with at least one RCP is running and state C3b is defined as shutdown state without RCP is running.

d) When the RCP [RCS] temperature is between 10℃ and 60℃ (state D), at least 2 trains of RIS [SIS] are used in RHR mode and all the RCPs stop.

An isolatable residual heat removal system break (≤DN250) located outside the containment in shutdown state with the RIS [SIS] connected in RHR mode is classified as a DBC-4 event.

12.9.5.5.2.2 Acceptance Criteria

The decoupling criteria for LOCA events:

a) The peak cladding temperature must remain lower than 1204℃;

b) The maximum cladding oxidation must remain lower than 17% of the cladding thickness;

c) The maximum hydrogen generation must remain lower than 1% of the amount that would generate if the whole active part of the cladding had reacted;

d) The ability to cool the core geometry shall be retained, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling;

e) The long-term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed.

12.9.5.5.2.3 Main Safety Functions

For this event, the following plant safety functions can mitigate the event:

a) Safety injection (SI) actuated on ‘‘Hot leg ∆Psat low 1’’ signal or on “Hot leg RCP [RCS] loop level low 1” signal.

b) The RCPs will be tripped by the “RCP ∆P low 1 and SI’’ signal.

c) The RIS [SIS] pumps will be tripped by the ‘‘Hot leg ∆Psat low 2’’ signal or “Hot leg RCP [RCS] loop level low 2” signal.

d) RIS/RHR array isolation will be triggered by ‘‘Safeguard building sump level high 1” signal or “Safeguard building pressure rise high 1” signal.

12.9.5.5.2.4 Typical Sequence of Events

The description presented hereafter is a typical sequence, i.e. the most likely to occur during the transient.

The sequence of events consists of two phases: the short-term phase until reaching of the controlled state using automatic actions from the initial event, and the long-term phase where the plant is operated from the controlled to the safe state.

a) From the initiating event to the controlled state

The break causes a loss of primary coolant and results in an RCP [RCS] water inventory decrease.

The SI signal is emitted following a ‘‘Hot leg ∆Psat low 1’’ signal or “Hot leg RCP [RCS] loop level low 1” signal. The SI signal emission induces the actuation of the safety injection system.

The RCPs will be tripped by the “RCP ∆P low 1 and SI” signal.

Since the break is outside the containment, the release of fluid in the safeguard building leads to the automatic isolation of the hot leg suction and the automatic pump trip of the RIS [SIS] train in RHR mode. The automatic isolation of the RIS [SIS] in RHR mode on the cold leg side is done through closure of the check valves.

The RIS [SIS] pumps may be tripped following a ‘‘Hot leg ∆Psat low 2’’ signal or “Hot leg RCP [RCS] loop level low 2” signal:

1) If the RIS [SIS] pumps are not tripped:

   The unaffected RIS [SIS] pumps ensure the RCP [RCS] heat removal. Before the break isolation, the MHSI then compensates for the break flowrate, and the RCP [RCS] water inventory remains sufficient, allowing progression to the controlled state. After break isolation, the MHSI pumps allow the primary water inventory to be replenished until the maximum MHSI pump head pressure is reached.

2) If the RIS [SIS] pumps are tripped:

Following the automatic isolation of the broken RIS [SIS] train in RHR mode, the RCP [RCS] is isolated from the break and the MHSI allows the primary water inventory to be replenished. Therefore, there is no further risk in terms of RCP [RCS] water inventory.

When the primary system is closed or re-sealable, since the break is automatically isolated, heat removal can be achieved using the VDA [ASDS] of the SGs, fed by their ASG [EFWS] trains. The controlled state is reached since the SGs are able to ensure the decay heat removal.

When the primary system is open (state D), heat can be removed by primary coolant bleed-off. The MHSI pumps guarantee a sufficient RCP [RCS] water inventory. This allows progression to the controlled state.

b) From the controlled state to the safe state

The safe state is defined as a state where the break flowrate is isolated or compensated by the MHSI flowrate and at least one RIS [SIS] train is connected to the RCP [RCS] in RHR mode. The connecting conditions are:

1) RCP [RCS] hot leg pressure < { } (absolute pressure);

2) RCP [RCS] hot leg temperature < { };

3) RCP [RCS] hot leg saturation margin ΔT_{SAT} and loop level consistent with RIS [SIS] in RHR mode suction from the hot leg.

Depending on whether the RIS [SIS] pumps trip, two cases are possible:

1) If the RIS [SIS] pumps are not tripped:

Considering the RIS [SIS] trains are operating in RHR mode, the safe state is reached at the same time as the controlled state.

2) If the RIS [SIS] pumps are tripped:

Since the break is outside containment, the break has been identified and isolated during the automatic phase. The operator does not perform the RCP [RCS] integrity test. The sequence of actions to be performed by the operator is the following:

- RCP [RCS] temperature management

  In states C1 and C2, the RCP [RCS] is initially closed. The isolation of the break restores the saturation margin. The operator can use the available SGs to manage RCP [RCS] temperature to allow RIS [SIS] in RHR mode re-connection.

  In state C3a, the operator is assumed to use Pressuriser Safety Valves (PSVs) and LHSI to cool the RCP [RCS].

In state C3b, since the temperature is lower than { }, no operator action is required.

In state D, the operator starts the MHSI and LHSI pumps to bring the RIS to RHR mode.

- RIS [SIS] is connected to RCP [RCS] in RHR mode

When the connection conditions are reached, the operator connects the RIS [SIS] in RHR mode to the RCP [RCS], and the safe state is reached.

12.9.5.5.2.5 Analysis Assumptions

The studied cases correspond to an isolatable guillotine rupture of an RIS [SIS] pipe connected to the RCP [RCS] hot leg in state C1, outside the containment.

a) Single failure

The single failure is postulated on the MHSI of the unaffected loop. As the break is assumed to be located at the safety injection point, only one safety injection train is taken into account for the RIS [SIS].

b) LOOP assumption

Because no reactor trip occurs, LOOP is not considered in this event.

12.9.5.5.2.6 Result and Conclusion

a) From the initiating event to the controlled state

The analysis performed (see Reference [51]) shows that the core remains covered following a residual heat removal system break accident outside the containment, so the acceptance criteria are not challenged.

b) From the controlled state to the safe state

This transient is not explicitly analysed as it is bounded by the other events. The description is provided in Sub-chapter 12.10.

12.9.5.6 Inadvertent Opening of the Dedicated Depressurisation Device (State A\B)

12.9.5.6.1 Inadvertent Opening of the Dedicated Depressurisation Device in State A

12.9.5.6.1.1 Initiating Event

This event is defined as the spurious opening of the pressuriser severe accident dedicated valve (SADV) in state A (at power), without rapid reclosing due to a spurious signal or operator error.

An inadvertent opening of the dedicated depressurisation device can be initiated by:

a) A spurious opening signal of two isolation valves in series;

b) An operator error causing the manual opening of the depressurisation device.

Precautions limiting the transient occurrence:

To open the severe accident dedicated valve, two valves in series must be sequentially opened, thus limiting the scenarios and occurrence probability. The position of the SADV is accessible to the operator.

This accident can lead to a loss of reactor coolant inventory and core heat-up. The consequences of the event are considered in the plant design and can be managed with proper acceptance criteria.

The inadvertent opening of the dedicated depressurisation device in state A is classified as a DBC-4 event.

12.9.5.6.1.2 Acceptance Criteria

This accident is classified as a DBC-4 event. The safety criteria are the radiological limits for DBC-4 events.

The decoupling criteria for LOCA events:

a) The peak cladding temperature must remain lower than 1204℃;

b) The maximum cladding oxidation must remain lower than 17% of the cladding thickness;

c) The maximum hydrogen generation must remain lower than 1% of the amount that would generate if the whole active part of the cladding had reacted;

d) The ability to cool the core geometry shall be retained, i.e. calculated changes in core geometry shall be such that the core remains amenable to cooling;

e) The long-term core cooling shall be ensured, i.e. the calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed.

These criteria are met if the core remains covered. Moreover, it shall be demonstrated that the plant can be brought to a safe state.

12.9.5.6.1.3 Main Safety Functions

In this event, reactor protection is provided by the following signals and actions:

a) Reactor trip (RT) is triggered after a “Pressuriser pressure low 2” signal;

b) The safety injection system is actuated by:

“Pressuriser pressure low 3” signal;

c) The RCPs are stopped by:

“RCP ΔP low 1” signal combined with the “Safety injection” signal;

d) The injection of the RIS [SIS] accumulator to the RCP [RCS] is actuated when:

The pressure of the RCP [RCS] is lower than 4.5 MPa;

e) Turbine trip is induced by:

Reactor trip;

f) The emergency feedwater system is actuated by:

“SG level (wide range) low 2” signal;

g) The main feedwater system is isolated by:

RIS [SIS] action signal or Reactor trip (RT).

12.9.5.6.1.4 Typical Events Sequence

The description presented hereafter is a typical sequence, i.e. the most likely to occur during the transient.

The sequence of events consists of two phases: the short-term phase until reaching the controlled state by use of automatic actions and the long-term phase where the plant is moved from the controlled to the safe state.

a) From the initiating event to the controlled state:

In state A, the spurious opening of the SADV induces a large pressure decrease in the RCP [RCS] and a pressuriser (PZR) level increase due to the steam released by the severe accident dedicated valve.

If the plant is initially at power, reactor trip (RT) is triggered by the “Pressuriser pressure low 2” signal or “Pressuriser level high 1” signal. The RT signal automatically trips the turbine and isolates the main feedwater system full load lines.

Following turbine trip, the possible secondary side pressure increase is limited by the steam generator (SG) relief devices: either the GCT [TBS] if available, or VDA

[ASDS].

The SGs are fed by the ARE [MFFCS] through the low load lines.

Safety injection (SI) signal is actuated by the "Pressuriser pressure low 3" signal. The SI signal automatically starts the MHSI and the LHSI pumps, and initiates an MCD. {

}

During the MCD, the RCP [RCS] pressure decreases sufficiently to allow MHSI injection into the cold legs. The released mass flowrate decreases as the PZR pressure decreases.

Further changes to the RCP [RCS] water inventory depend on the balance between the safety injection and heat removal system (RIS/RHR) injection, and the flowrate of the severe accident dedicated valve.

Before the RIS [SIS] injection is able to compensate for the flowrate of the severe accident dedicated valve, the core may be uncovered. This would result in a fuel clad temperature increase in the exposed area. The larger the area exposed, and the longer the duration, the higher the resulting clad temperatures.

The RCPs may be tripped by the "RCP ΔP low 1 over two loops and SI" signal. The RCP [RCS] inventory depletion stops when the MHSI and LHSI flowrate is sufficient to compensate for the flowrate of the severe accident dedicated valve.

The controlled state is reached following the achievement of stable heat removal conditions via the operation of MHSI and LHSI and the discharge of severe accident dedicated valve. The core is sub-critical and the reactor coolant inventory is stabilised or increasing due to RIS [SIS] injection.

b) From the controlled state to the safe state:

The safe state is defined as a state for which the flowrate of the SADV is compensated by the RIS [SIS] flowrate with long-term core cooling ensured.

The sequence of actions to be performed by the operator is as follows:

1) RCP [RCS] boration:

   The RCP [RCS] is borated sufficiently to keep the core sub-critical during the entire transient up to the safe state. Boration is performed by MHSI, or by additional boration using the RCV [CVCS], or via the RBS [EBS] if the RCV [CVCS] is unavailable.

2) RCP [RCS] cooldown:

   The RCP [RCS] cooldown is performed via the secondary side by reducing the VDA [ASDS] opening pressure setpoint.

3) Water inventory restoration:

After reaching the switching time of the hot and cold legs, the operator switches the LHSI trains from cold leg injection to simultaneous injection to the cold leg and hot leg,.

12.9.5.6.1.5 Analysis Assumptions

This transient is similar to the “Intermediate Break (at Power) (DBC-4)” presented in Sub-section 12.9.5.3. The maximum relief capacity of one severe accident dedicated valve line is limited by the severe accident discharge nozzle on the top of the pressuriser.

The analysis performed in Sub-section 12.9.5.3, which covers a break spectrum up to 27.5 cm, located on cold leg.

The case with a surge line break analysed in Sub-section 12.9.5.3 leads to a faster decrease of RCP [RCS] water inventory than the opening of a severe accident dedicated valve line.

The consequences of the inadvertent opening of the dedicated depressurisation device in state A are thus bounded by the analysis of the “Intermediate Break (at Power) (DBC-4)” presented in Sub-section 12.9.5.3. The acceptance criteria are thus met.

12.9.5.6.1.6 Result and Conclusion

The analysis performed shows that the transient is covered by IB-LOCA transient (at power) (DBC-4) (see Sub-section 12.9.5.3.). It demonstrates that the acceptance criteria presented in Paragraph 12.9.5.6.1.3 are met.

12.9.5.6.2 Inadvertent Opening of the Dedicated Depressurisation Device in State B

12.9.5.6.2.1 Initiating Event

The inadvertent opening of the severe accident dedicated depressurisation device in state B (in shutdown state) (DBC-4) is defined as the spurious opening of the pressuriser severe accident dedicated valve, without rapid reclosing due a spurious signal or operator error.

This accident can lead to a loss of reactor coolant inventory and core heat-up.

The consequences of the event are considered in the plant design and can be managed by meeting the proper acceptance criteria.

Identification of causes:

An inadvertent opening of the dedicated depressurisation device can be initiated by:

a) A spurious opening signal of two isolation valves in series;

b) An operator error causing the manual opening of the depressurisation device.

Precautions limiting the transient occurrence:

For the SADV to open, two valves in series must be sequentially opened by the operator, thus limiting the potential scenarios and occurrence probability. The position of the SADV is accessible to the operator.

The inadvertent opening of the dedicated depressurisation device in state B is classified as a DBC-4 event.

12.9.5.6.2.2 Acceptance Criteria

This accident is classified as a DBC-4 event. The safety criteria are the radiological limits for DBC-4 events. The acceptance criteria for LOCA are applied.

12.9.5.6.2.3 Main Safety Functions

In this event, reactor protection is provided by the following signals and actions:

a) The RCPs are tripped by:

“ΔP low 1 over two RCPs” signal cumulated with the “Safety injection” signal;

b) The safety injection system is actuated by:

“Hot leg ΔPsat low 1” signal;

c) The emergency feedwater system is actuated by:

“SG level (wide range) low 2” signal;

d) The main feedwater system is isolated by:

1) RIS [SIS] action signal;

2) Reactor trip (RT).

12.9.5.6.2.4 Typical Events Sequence

The description presented hereafter is a typical sequence, i.e. the most likely to occur during the transient.

The sequence of events consists of two phases: the short-term phase until reaching the controlled state through use of automatic actions and the long-term phase where the plant is operated from the controlled to the safe state.

a) From the initiating event to the controlled state:

In state B, the spurious opening of the severe accident dedicated valve (SADV) induces a large pressure decrease in the RCP [RCS] and a pressuriser (PZR) level increase due to the steam release by the severe accident dedicated valve.

SI signal is emitted after the “Hot leg ∆Psat low 1” signal, inducing the following actions:

1) Starting of MHSI and LHSI pumps;

2) Medium pressure rapid cooldown consisting in a controlled lowering of the GCT [TBS] (if available) or of the VDA [ASDS] setpoint;

3) Containment isolation stage: the reactor coolant pressure boundary is isolated, in particular the RCV [CVCS] letdown line and the steam generator (SG) blowdown lines are isolated.

The RCPs may be tripped by the “RCP ∆P low 1 over two loops” and “SI” signals.

As long as the RIS [SIS] flowrate does not compensate for the flowrate of the severe accident dedicated valve, RCP [RCS] coolant water inventory continues to decrease.

The flowrate of the SADV decreases as the PZR pressure decreases. The primary coolant inventory depletion stops when the RIS [SIS] flowrate compensates for the flowrate of the severe accident dedicated valve.

The controlled state is reached following achievement of stable heat removal conditions via the operation of MHSI and LHSI and the discharge of severe accident dedicated valve. The core is sub-critical and the reactor coolant inventory is stabilised or increasing due to RIS [SIS] injection.

b) From the controlled state to the safe state:

The means to reach the safe state from the controlled state are the same as in state A (see Paragraph 12.9.5.6.1.3).

12.9.5.6.2.5 Analysis Assumptions

This transient is similar to the “Intermediate Break (at shutdown condition) (DBC-4)” presented in Sub-section 12.9.5.3. The maximum relief capacity of one severe accident dedicated valve line is limited by the severe accident discharge nozzle on the top of the pressuriser.

The analysis performed in Sub-section 12.9.5.3, which covers a break spectrum up to 27.5 cm, located on cold leg.

The case with a surge line break analysed in Sub-section 12.9.5.3 leads to a faster decrease of RCP [RCS] water inventory than the opening of a severe accident dedicated valve line.

The consequences of the inadvertent opening of the dedicated depressurisation device in state B are thus bounded by the analysis of the “Intermediate Break (at shutdown condition) (DBC-4)” presented in Sub-section 12.9.5.3. The acceptance criteria are thus met.

12.9.5.6.2.6 Result and Conclusion

The analysis performed shows that the transient is bounded by the IB-LOCA transient (in shutdown state) (DBC-4) (see Sub-section 12.9.5.3). It demonstrates that the acceptance criteria presented in Sub-section 12.9.5.6.2.3 are met.

12.9.6 Radiological Release of Systems or Components

12.9.6.1 Fuel Handling Accident (State A to F)

Accidents occurring during fuel handling operations may be as follows:

a) Fuel assembly drop in the reactor pool.

b) Fuel assembly drop in the spent fuel pool.

c) Spent fuel transport cask drops outside the fuel building.

d) Spent fuel transport cask drops inside the fuel building.

The Spent fuel transport cask drop accident, with regards to radiological consequences, is analysed in Sub-chapter 12.11.5.13. For a fuel assembly drop in the reactor pool, the containment sweeping and blowdown ventilation system will shut down automatically. Therefore, in terms of the radiological consequences, a fuel assembly drop in the spent fuel pool will be more serious. The analysis in terms of source terms and radiological consequences for a fuel assembly drop during fuel handling operations is provided in Sub-chapter 12.11.5.6.

12.9.6.2 Spent Fuel Transport Cask Drop (State A to F)

A Spent fuel transport cask drop accident may occur when the crane is loading or unloading fuel which may result in radioactivity release to environment. The analysis in terms of source terms and radiological consequences is provided in Sub-chapter 12.11.5.13.

12.9.6.3 Failure of Radioactivity Containing Equipment in Nuclear Auxiliary Building (State A to F)

Failure of radioactivity containing equipment in nuclear auxiliary building may cause radioactivity release to the environment. The detailed analysis in terms of source terms and radiological consequences of this accident is addressed in Sub-chapter 12.11.5.14.

12.9.7 Fuel Pool Accidents

12.9.7.1 Non-isolatable Small Break or Isolable RIS [SIS] Break (≤DN 250) in RHR Mode Affecting Fuel Pool Cooling (during refuelling) (State E)

12.9.7.1.1 Initiating Event

A piping failure may occur on an RIS [SIS] line (DN<250mm) in RHR mode or a line (DN<50mm) connected to the primary cooling system upstream of the first isolation valve. Since the transfer tube may be opened in state E, such piping failure may affect the cooling function of the SFP.

12.9.7.1.2 Acceptance Criteria

The safety criteria for the DBC accidents associated with the spent fuel storage pool are as follows and are also described in Sub-chapter 12.5.9.1.

a) Permanent maintenance of sub-criticality;

b) The fuel assemblies in the fuel pool remain covered;

c) Removal of decay heat from the spent fuel pool.

For the DBC accidental fuel storage pool draining events, the removal of decay heat from the fuel storage pool is ensured through a decoupling criterion: the fuel storage pool water temperature must remain below 95℃.

12.9.7.1.3 Main Safety functions

For this event, the following plant safety functions can mitigate the event:

a) Every PTR [FPCTS] train is designed to remove decay heat from the SFP.

b) Isolation of the RIS/RHR line following indication that the reactor pool water low-level of 16.4m.

12.9.7.1.4 Typical Events Sequence

a) For non-isolable break (<50mm) on a line connected to the primary cooling system:

Since the break in this situation is non-isolatable, the resulting drainage leak cannot be passively stopped manually or automatically. Thus, it’s necessary to compensate for the leakage with a permanent water makeup, to stabilize the water level in the pools (i.e. to reach the controlled state) firstly, and then to restore the water level in the SFP to a level sufficient to start a PTR [FPCTS] train (i.e. to reach the safe state).

During the transient, the water loss is eventually collected in the IRWST due to its lower position in the containment.

Water makeup is performed with the MHSI pumps in recirculation mode between the

IRWST and the primary cooling system.

Before starting the MHSI pumps for injection, it is necessary to drain the lance compartment in the IRWST to ensure a sufficient IRWST water level for the MHSI pumps.

Related floor drainage/discharge lines are automatically isolated to prevent any loss of water outside the reactor building.

When the water makeup is supplied with the MHSI pumps, the water level remains a sufficiently high value and the PTR cooling train can be remained utilised. Thus, the controlled state and safe state are reached simultaneously.

b) For RIS/RHR lines isolatable break (<250mm)

1) transfer tube is open:

When the transfer tube remains opened, the reactor pool, the reactor internals storage compartment, transfer compartment and the spent fuel pool are connected. The break in the RIS/RHR line leads to drainage of the pools. When the water level in the reactor building transfer compartment drops to 16.4m, the RIS/RHR line isolation signal is triggered. The broken RIS/RHR line is fully isolated 90s later. The drainage of the pools is stopped and the controlled state is therefore automatically reached.

When the RIS/RHR line is isolated, the SFP water level is stabilized at a level higher than the lowest level required to restart the main cooling trains of PTR. Two PTR cooling trains can be restarted one hour after the transient. The SFP long-term cooling can be ensured after the restart of the PTR cooling train and safety state can be reached in this case.

2) transfer tube is closed:

In this case, when a low water level is detected in the reactor building transfer compartment, the leak through RIS [SIS] suction line is automatically isolated by closing motorized valves. The drainage through RIS [SIS] discharge line is automatically prevented by the check valves. Thus, the controlled state is reached automatically.

12.9.7.1.5 Analysis Assumptions

Main assumptions adopted in the analysis are listed as follows:

a) The decay heat of the SFP in State E is conservatively assumed.

b) The initial water temperature is 50℃, which can cover all normal operating conditions.

c) The initial water volume of the SFP is 1265.8m^3 corresponding to the water level of 16.9m.

d) The break is conservatively assumed to be located in the lowest position of the corresponding pipeline, to maximize the leak flowrate.

e) The water heating is considered to be only localized in the SFP compartment, and the water is homogeneously heated in the SFP compartment.

f) The SFP and the PTR pipes are considered as adiabatic.

g) Once the SFP cooling system is recovered, the mean SFP temperature is calculated using a maximum RRI inlet temperature of 45℃.

h) One PTR cooling train is considered unavailable due to maintenance on the supporting systems of the PTR [FPCTS] train. Single failure criterion will not be considered when maintenance is taken.

i) The LOOP is not considered in the analysis.

12.9.7.1.6 Results and Conclusion

In conclusion, for a non-isolatable small break or an isolatable RIS [SIS] break (<250mm) in RHR Mode affecting fuel pool cooling, the PTR cooling train can be used to remove the decay heat from the SFP. The temperature of the SFP will not exceed the limit. The fuel assemblies remain covered during the entire transient.The sub-criticality of the fuel assemblies is ensured by the design of spent fuel storage rack which is analysed in detail in Reference [40].

According to the analysis, all acceptance criteria are met in this accident.

12.10 Transfer of the Plant to the Long Term Safe State

In Sub-chapter 12.7 through 12.9 the analysis from controlled state to safe state has been described. As the detailed analysis of this phase for different events can be grouped, bounding events are identified for each group. This subchapter summarized the analysis of all the bounding events for analysis from controlled state to safe state.

As is shown in T-12.10-1 to T-12.10-3, following events are selected as bounding cases, for which the long term analysis will be specified (detailed analysis will be provided with reference):

a) Long term loss of offsite power (> 2 hours);

b) SG tube rupture (one tube);

c) Small break Loss of Coolant Accident (LOCA) (at power) including a break in the Emergency Boration System (RBS [EBS]) injection line;

d) Large feedwater system piping break;

e) Steam generator tube rupture (two tubes in one SG);

f) Intermediate break LOCA (at power and in a shutdown condition);

g) Small break LOCA, including a break in the Emergency Boration System injection line (during shutdown conditions, and RIS [SIS] under RHR operation mode);

h) Large break LOCA (LB-LOCA) (at power).

Detailed analysis will be provided with reference.

T-12.10-1 Bounding Cases for plant to transfer to the Long Term Safe State under DBC-2

No.	DBC events	Bounding cases regarding sub-criticality	Bounding cases regarding removal of heat	Bounding cases regarding confinement of radioactive substances
1	Feedwater system malfunctions causing a reduction in feedwater temperature (State A\B)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
2	Feedwater system malfunctions causing an increase in feedwater flow (State A\B)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
3	Excessive increase in secondary steam flow (State A\B)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
4	Turbine trip (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
5	Loss of condenser vacuum (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
6	Short term loss of offsite power (< 2 hours) (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)

No.	DBC events	Bounding cases regarding sub-criticality	Bounding cases regarding removal of heat	Bounding cases regarding confinement of radioactive substances
7	Loss of normal feedwater flow (loss of all main feedwater pumps and Startup and Shutdown Feedwater System (AAD [SSFS]) pumps) (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
8	Loss of one cooling train of the Safety Injection System (RIS [SIS]) in Residual Heat Removal (RHR) mode (State C\D)	NA	NA	NA
9	Partial loss of core coolant flow (loss of one main coolant pump) (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
10	Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at a sub-critical or low power startup condition (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
11	RCCA bank withdrawal at power (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
12	RCCA misalignment up to rod drop without limitation (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)

No.	DBC events	Bounding cases regarding sub-criticality	Bounding cases regarding removal of heat	Bounding cases regarding confinement of radioactive substances
13	Startup of an inactive reactor coolant loop at an improper temperature (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
14	Chemical and Volume Control System (RCV [CVCS]) malfunction that results in a decrease in boron concentration in the reactor coolant (State A to E)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
15	Spurious reactor trip (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
16	RCV [CVCS] malfunction causing an increase in (RCP [RCS]) inventory (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
17	RCV [CVCS] malfunction causing a decrease in (RCP [RCS]) inventory (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
18	Uncontrolled RCP [RCS] level drop in shutdown states with RIS [SIS] connected in RHR mode (State C\D)	NA	NA	NA

No.	DBC events	Bounding cases regarding sub-criticality	Bounding cases regarding removal of heat	Bounding cases regarding confinement of radioactive substances
19	Spurious pressuriser heater operation (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
20	Spurious pressuriser spray operation (State A)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
21	Loss of one train of the Fuel Pool Cooling System (PTR [FPCTS]) or of a supporting system (during power operation, hot shutdown and intermediate shutdown conditions) (State A)	NA	NA	NA

T-12.10-2 Bounding Cases for plant to transfer to the Long Term Safe State under DBC-3

No.	DBC events	Bounding cases regarding sub-criticality	Bounding cases regarding removal of heat	Bounding cases regarding confinement of radioactive substances
1	Inadvertent opening of an SG relief train or of a safety valve (State A)	Large steam system piping break (State A\B)	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
2	Small steam system piping break including breaks in connecting lines (State A\B)	Large steam system piping break (State A\B)	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
3	Inadvertent closure of all or one main steam isolation valves (State A)	Large steam system piping break (State A\B)	Large feedwater system piping break (State A\B);	Large steam system piping break (State A\B)
4	Long term loss of offsite power (> 2 hours) (State A)	Quantitatively anlaysed		
5	Small feedwater system piping break including breaks in connecting lines to SG (State A\B)	Large feedwater system piping break	Large feedwater system piping break (State A\B)	Large feedwater system piping break
6	Forced reduction in reactor coolant flow (3 pumps) (State A)	Large steam system piping break (State A\B)	Large feedwater system piping break (State A\B)	Reactor coolant pump seizure (locked rotor) or Reactor coolant pump shaft break (State A)
7	Inadvertent loading of a fuel assembly in an improper position	NA	NA	NA

	(State E)			
8	Uncontrolled RCCA bank withdrawal (State B\C\D)	RCV [CVCS] Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Large feedwater system piping break (State A\B)	Spectrum of RCCA ejection accidents (State A)
9	Uncontrolled single RCCA withdrawal (State A)	Large steam system piping break (State A\B)	Large feedwater system piping break (State A\B)	Spectrum of RCCA ejection accidents (State A)
10	SG tube rupture (SGTR) (one tube) (State A)	Quantitatively anlaysed		
11	Inadvertent opening of a pressuriser safety valve (State A)	SB/IB-LOCA	SB/IB-LOCA	LB-LOCA
12	Rupture of a line carrying primary coolant outside containment (e.g. nuclear sampling line) (State A)	SB/IB-LOCA	SB/IB-LOCA	Quantitatively anlaysed
13	Small break Loss of Coolant Accident (LOCA) (at power) including a break in the Emergency Boration System (RBS [EBS]) injection line (State A)	Quantitatively anlaysed	Quantitatively anlaysed	LB-LOCA
14	Small break LOCA (at shutdown, RIS [SIS] not connected in RHR mode) including a break in the RBS [EBS] injection line (State A\B)	SB/IB-LOCA	SB/IB-LOCA	LB-LOCA
15	Gaseous waste tank break (State A to F)	NA	NA	NA

16	Liquid waste effluent tank break (State A to F)	NA	NA	NA
17	Volume control tank break (State A to F)	NA	NA	NA
18	Long term loss of offsite power (>2 hours) affecting fuel pool cooling (State A)	NA	NA	NA
19	Loss of one train of the PTR [FPCTS] or of a supporting system (with the reactor core offloaded to the fuel pool) (State F)	NA	NA	NA
20	Isolatable piping failure on a system connected to the spent fuel pool (State A to F)	NA	NA	NA

T-12.10-3 Bounding Cases for plant to transfer to the Long Term Safe State under DBC-4

No.	DBC events	Bounding cases regarding sub-criticality	Bounding cases regarding removal of heat	Bounding cases regarding confinement of radioactive substances
1	Large steam system piping break (State A\B)	Quantitatively anlaysed		
2	Inadvertent opening of an SG relief or safety valve (State B)	Large steam system piping break (State A\B)	Large feedwater system piping break (State A\B)	Large steam system piping break (State A\B)
3	Large feedwater system piping break (State A\B)	Quantitatively anlaysed		
4	Long term LOOP (State C)	Quantitatively anlaysed		
5	Reactor coolant pump seizure (locked rotor) or Reactor coolant pump shaft break (State A)	Large steam system piping break (State A\B)	Large feedwater system piping break (State A\B)	Quantitatively anlaysed
6	Spectrum of RCCA ejection accidents (State A)	Large steam system piping break (State A\B)	Large feedwater system piping break (State A\B)	Quantitatively anlaysed
7	Boron dilution due to a non-isolatable rupture of a heat exchanger tube (during shutdown conditions) (State C\D\E)	For this transient, the safe state is characterised by the recovery of the initial boron concentration. This transient is covered by the DBC-2 dilution event for the boration capability.	no need, because focus on Boron dilution and the plant state is already in State C/D/E	NA

No.	DBC events	Bounding cases regarding sub-criticality	Bounding cases regarding removal of heat	Bounding cases regarding confinement of radioactive substances
8	SGTR (two tubes in one SG) (State A)	Quantitatively anlaysed		
9	Large break LOCA (LB-LOCA) (at power) (State A)	Quantitatively anlaysed		
10	Intermediate Loss of Coolant Accident (LOCA) (State A\B)	Quantitatively anlaysed		LB-LOCA
11	SB-LOCA, including a break in the emergency boration system injection line (State C\D)	Quantitatively anlaysed		LB-LOCA
12	RHR system piping break inside (outside) containment (≤DN 250) (State C\D)	Quantitatively anlaysed		NA
13	Inadvertent opening of the dedicated depressurisation device (State A\B)	bounded by Intermediate break LOCA (at power and in a shutdown condition)	bounded by Intermediate break LOCA (at power and in a shutdown condition)	LB-LOCA
14	Fuel handling accident (State A to F)	NA	NA	NA
15	Spent fuel transport cask drop (State A to F)	NA	NA	NA
16	Failure of radioactivity containing equipment in nuclear auxiliary building (State A to F)	NA	NA	NA

No.	DBC events	Bounding cases regarding sub-criticality	Bounding cases regarding removal of heat	Bounding cases regarding confinement of radioactive substances
17	Non-isolatable small break or isolatable RIS [SIS] break (≤DN 250) in RHR mode affecting fuel pool cooling (during refuelling) (State E)	Quantitatively anlaysed		

12.11 Radiological Consequences of Design Basis Accident

12.11.1 Safety Requirements

12.11.1.1 Safety Objectives

The nuclear safety objective of the UK HPR1000 is as follows, which is also defined in PCSR Chapter 4: The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is as low as reasonably practicable.

The safety objectives for DBCs radiological consequence analysis is to verify that the plant systems are appropriately designed and operated from the point of view of radiological release. It also aims to show that the doses to the public and on-site worker following a discharge of radioactivity of the plant meet the radiological consequence acceptance criteria.

12.11.1.2 Design Requirements

In order to judge the adequacy of radiological hazard control for DBC accidents, radiation protection targets are set in Reference [52]. This reference gives a set of targets for the effective dose received by any person from a DBC fault sequence and which are set in terms of a Basic Safety Level (BSL) and a Basic Safety Objective (BSO). The BSLs should be met in the first instance, as they are mandatory legal limits. The region under BSO represents the broadly acceptable region. For the UK HPR1000 nuclear power plant, the maximum effective dose received by any person arising from the DBCs should meet the requirement of these targets.

12.11.2 General Statements and Assumptions

The purpose of this section is to present the main assumptions and inputs to radiological consequence evaluation of the DBCs, including:

a) The activity inventory;

b) The activity release, including release fraction, timing of release phases, radionuclide composition and the chemical form of radioactivity release;

c) Radionuclides released into the environment;

d) Radionuclide release transfer in the Environment

e) Off-site dose calculations.

12.11.2.1 Activity Inventory

When the accident involves fuel failure, radioactivity may be released from the fuel, primary coolant or secondary coolant.

a) Core Inventory

When the integrity of fuel rods fails, the radioactivity in the fuel of the reactor core is released to the primary coolant, then to the containment or secondary side. Conservative reactor core activity inventory is considered for the source term calculation.

b) Primary Coolant Activity

The primary coolant source term for the UK HPR1000 is presented in *Primary Source Term Calculation Report* (see Reference [53]). The phenomenon of iodine spiking, which will cause the radioactive iodine concentration in the primary coolant to increase significantly, is considered in the source term analysis.

c) Secondary Coolant Activity

Under normal operating conditions, the secondary system is mostly free of radioactivity because of the leak-tight steam generator tubes. However, limited primary-to-secondary leaks through the SG tubes are considered in the analysis during normal plant operation. To calculate the radioactivity of secondary coolant in the accidental transient, the primary-to-secondary leak, decay of nuclides and the purification of the blowdown system of the secondary system are all taken into consideration. The maximum primary-secondary leakage, 44L/h per SG, is used.

12.11.2.2 Activity Release

a) Activity release following fuel failure

1) Release fraction

Based on the thermal-hydraulic transient analysis results of the design basis accidents, the radioactive release fractions from the fuel are conservatively considered.

For the DBC LOCAs, it is conservatively considered that 100% of the fuel cladding failure. The estimated release fractions for a fuel gap release are as follows:

- Noble Gases: 5%,
- Halogens: 5%,
- Alkali Metals: 5%.

For an RCCA ejection accident, the gap release fractions are 10% for noble gases and iodines, 12% for Alkali Metals.

For the other non-LOCA events, the gap release fractions are given below.

- Kr-85: 10%,

- Other Noble Gases: 5%,
- I-131: 8%,
- Other Halogens: 5%,
- Alkali Metals: 12%.

2) Timing of Release Phases

Generally, radioactivity release for the fuel failure occurs at the point of fuel failure. For DBC LOCAs, the assumed onset and duration of release is listed in T-12.11-1. The release can be modelled in a linear manner over the duration of the phase or be modeled to occur instantaneously with the onset. For non-LOCA DBCs where fuel damage occurs, the radioactivity from the damaged fuel is released instantaneously at the onset of the fuel damage.

3) Radionuclide Composition

The elements in each radionuclide group considered in the radiological consequence analysis are listed in T-12.11-2.

4) Chemical form

The iodine released from the RCP [RCS] to the containment in the postulated accident is assumed to be in various chemical forms, of which 95% is cesium iodine, 4.85% is elemental iodine, and 0.15% is organic iodine. The fission products released into the containment are all assumed to be in particulate form, except for elemental iodine, organic iodine and noble gases. During the radioactivity transporting process, the chemical form may be changed and will be analysed according to the specific accident.

b) Activity Release Following the Discharge of Liquid

Liquids discharged to the containment or peripheral buildings will partially flash into the atmosphere. For noble gases, 100% of the noble gases in the discharged liquids are released into the ambient atmosphere instantaneously.

For iodine and other solids in the discharged liquids, the release fraction to the atmosphere is determined by flash fraction and partition coefficient. The flash fraction of the discharged liquid is calculated assuming that the discharge is a constant enthalpy process.

c) Activity Release Following the Moisture Carryover from the Steam Generator

Radioactive release to the environment through the Steam Generator is determined by SG tube status. Two cases are considered:

1) Radioactivity release to the secondary system through operational SG

leakage;

2) Radioactivity release to the secondary system following SGTR.

For the first case, three following representative DBCs are considered:

- Rod Cluster Control Assembly (RCCA) ejection accident,
- Main steam line break accident,
- RCP locked rotor accident.

For the second case, two following representative DBCs are considered:

- SG Tube Rupture (SGTR) (one tube),
- SGTR (two tubes in one SG).

1) Activity Carryover in the case of operational leakage

In accidents with operational leakage in the SG, the primary coolant containing radionuclides passes from the primary side to the secondary side, due to the pressure difference between two circuits. The leak results in a build-up and an increase in radioactivity concentration of the secondary coolant during the transients.

The reactor coolant leaking into the steam generators is assumed to mix with the secondary water. As steam is released, a portion of the iodine and alkali metals in the coolant is released. Release from the secondary coolant is limited by the assumed SG moisture carryover. The noble gas activity entering the secondary side is assumed to be released to the environment directly. For the iodine and alkali metals, it is assumed that the leakage from the primary circuit to the secondary circuit enters into the liquid phase and mixes with the water, and which is then carried by secondary circuit steam and discharged into the environment. The release fraction is determined by the steam flowrate and carryover factors. The carryover factors are assumed to be 1% for iodines and 0.25% for alkali metals.

2) Activity Carryover in case of SGTR accident

In an SGTR accident, two different situations could arise, depending on whether the tube break is submerged. In either case, noble gases are conservatively assumed to be immediately transported to the main steam, released to the environment without reduction or mitigation.

- If the heat transfer tube is submerged, the primary-to-secondary leakage can be assumed to mix with the secondary water. Iodine and alkali metals are carried by steam and discharged into the environment. The carryover factors are assumed as 1% for iodine and 0.25% for alkali metals.

- During the periods where the water level drops below the break, a portion of the SGTR break flow would flash to vapour, according to the thermodynamic conditions in the RCS and the secondary side. The radioactivity in the flash coolant is directly released to the environment without mitigation.

12.11.2.3 Radionuclides Released into the Environment

a) Containment Leakage Rate

The containment is the third safety barrier of a nuclear power plant for preventing the release of radioactive material to the environment. It normally contains very small leaks originating from the manufacturing and construction processes. Appropriate leakage and strength tests, which are to be performed in the course of commissioning of the plant as well as at regular intervals during in-service inspections, would ensure that the realistic leak rate is lower than the limiting value in the Operation Technical Specification. The containment leakage rate considered is as follows:

1) 0-24h: peak pressure technical specification leak rate (0.3%/d) specified in the OTS;

2) >24h: reduced to 50% of the technical specification leak rate (namely 0.15%/d).

In accidental conditions, the containment and system design can ensure that the peak containment pressure is lower than the design criteria of the containment. The above leakage rate adopted in the radiological consequence analysis is conservative.

b) Ventilation System

During power operation, there is no sweeping ventilation of the containment atmosphere, and all the containment isolation valves are closed. However, when access to the service area is necessary, the low-capacity circuit of the Containment Sweeping and Blowdown Ventilation System (EBA [CSBVS]) is started for the purpose of sweeping, several days before and throughout access, in order to ensure a level of atmospheric contamination compatible with the staff access.

It is conservatively assumed that the EBA [CSBVS] low-capacity circuit is operating at its maximum flowrate. At the start of the accident, containment isolation is initiated by the reactor protection system (RPR). All containment isolation valves of the low-capacity EBA [CSBVS] circuit are closed. Therefore, the containment ventilation rate is stopped, and the containment leakage rate is subsequently assumed. After the isolation of the EBA [CSBVS] system, all containment leakages are assumed to be collected in the annulus.

The extracted air is released at the stack after being filtered by one of the EBA [CSBVS] filtration trains. However, the retention effects of the EBA [CSBVS] filters are conservatively not taken into account for the present analysis.

During normal operation, the normal-operational train of the Annulus Ventilation System (EDE [AVS]) operates continuously to create and maintain a sub-atmospheric pressure in the annulus. The exhaust air is discharged via filters to the vent stack. In the analysis, these filters are not taken into account.

During an accident, the annulus is maintained at sub-atmospheric pressure in order to collect the inner containment leakages including those collected by the containment leakage rate and monitoring system. During the accident, the EDE [AVS] switches automatically to the safety trains to process leakages from the reactor building containment. The normal-operational train is automatically isolated by motorized isolation dampers on the containment isolation signal. The fan on the safety train and the electric heater are started, thus maintaining the required sub-atmospheric pressure in the annulus. The leakage from the containment is mixed and diluted in the annulus before release via the EDE [AVS]. The maximum EDE [AVS] flowrate is considered in the accident.

The activity inside the annulus is assumed to be homogeneously distributed and to be discharged into the environment via the stack through accident filters at the ventilation rate of the EDE [AVS]. The filter efficiencies are postulated as follows:

1) Noble Gases: 0.0%,

2) Organic Iodine: 99.0%,

3) Elemental Iodine: 99.0%, and

4) Aerosols: 99.9%.

The predicted efficiency against elemental iodine is higher than the demonstrable efficiency against organic iodine compounds. However, a single conservative value is used in the present analysis.

12.11.2.4 Radionuclides Released Transfer in Environment

The assessment of radionuclides released to atmosphere by the HPR1000 (FCG3) is based on Reference [54] which identifies acceptable methods for calculating atmospheric relative concentration values, i.e. the atmospheric dispersion factor. A Gaussian model joint frequency of wind direction, wind speed and stability from site meteorological station is used for calculating the atmospheric dispersion factors in conservative and realistic cases for each 16 directions. The atmospheric diffusion coefficient observed for a specific site or the P-G curve is used.

12.11.2.5 Off-site Dose Calculations

The off-site dose calculation model used for the HPR1000 (FCG3) is based on U.S. RG1.195 (Reference [55]). This model contains the off-site thyroid doses and off-site whole body doses by assuming a semi-infinite cloud of photon emitters. There are two exposure pathways considered in off-site dose calculations for HPR1000 (FCG3), i.e. the internal exposure from inhalation of the airborne radionuclides and the external exposure from the radioactive cloud. Dose conversion factors for these 2 exposure pathways adopt the up-to-date International Commission on Radiological Protection, IAEA and Federal Guidance Report recommended values. It is conservatively assumed that the location of the off-site hypothetical person is on the boundaries of the exclusion area and the planning restricted area in the direction of most exposure. For the location at the exclusion area boundary, the exposure time is 2 hours, and for the location at the planning restricted area boundary, the exposure time is 30 days.

More detailed methods and assumptions will be presented in the off-site radiological consequences analysis for the HPR1000 (FCG3) which will be completed during step 3.

T-12.11-1 Onset and Duration of Release of DBC LOCAs

Phase	Onset	Duration
Gap release	30 s	0.5 hr

T-12.11-2 Elements in Each Radionuclide Group Discussed in the Radiological Consequence Analysis

Radionuclide group	Nuclide elements
Noble gases	Kr, Xe
Halogens	I, Br
Alkali metals	Cs, Rb
Tellurium group	Te, Sb, Se, Ba, Sr
Noble metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium	Ce, Pu, Np

12.11.3 Radiological Representative Design Basis Accidents

Representative design basis accidents are identified based on the following principles:

a) Assuring all the radiological release characteristics such as type of release, release paths, the potential mitigation system used, height of release and operational modes, etc., are included;

b) Covering locations of possible leakage on the site (Containment, Safeguard Building (BSX), Fuel Building (BFX), Nuclear Auxiliary Building (BNX), etc.).

c) Identifying the bounding accident from the quantities of radioactive release for each type of accident with the same radiological activity release characteristics described above.

Events with a potential radioactive release to the environment are checked based on the principles mentioned above and the representative radiological events are identified and listed as following:

a) Loss of coolant accident,

b) Rod Cluster Control Assembly (RCCA) ejection accident,

c) Main steam line break accident,

d) SG Tube Rupture (SGTR) (one tube),

e) SGTR (two tubes in one SG),

f) Fuel handling accident,

g) RCP locked rotor accident,

h) Rupture of a line carrying primary coolant outside containment,

i) Residual heat removal system break outside containment,

j) Gaseous waste tank break,

k) Liquid waste tank break,

l) Volume control tank break,

m) Spent fuel transport cask drop,

n) Failure of radioactivity containing equipment in nuclear auxiliary building.

12.11.4 Radiological Consequence Evaluation Process

To assess the radiological consequences for DBCs against the off-site radiation protection targets for a design basis fault sequence, source terms will be analysed first based on the thermal-hydraulic analysis, then the off-site radiological consequences analysis is undertaken. Conservative deterministic methods are used. Physical processes and phenomena are analysed with conservative, bounding assumptions.

The workstream for source term analysis is as follows:

a) Illustrate the activity source and source term release paths for each DBC scenario.

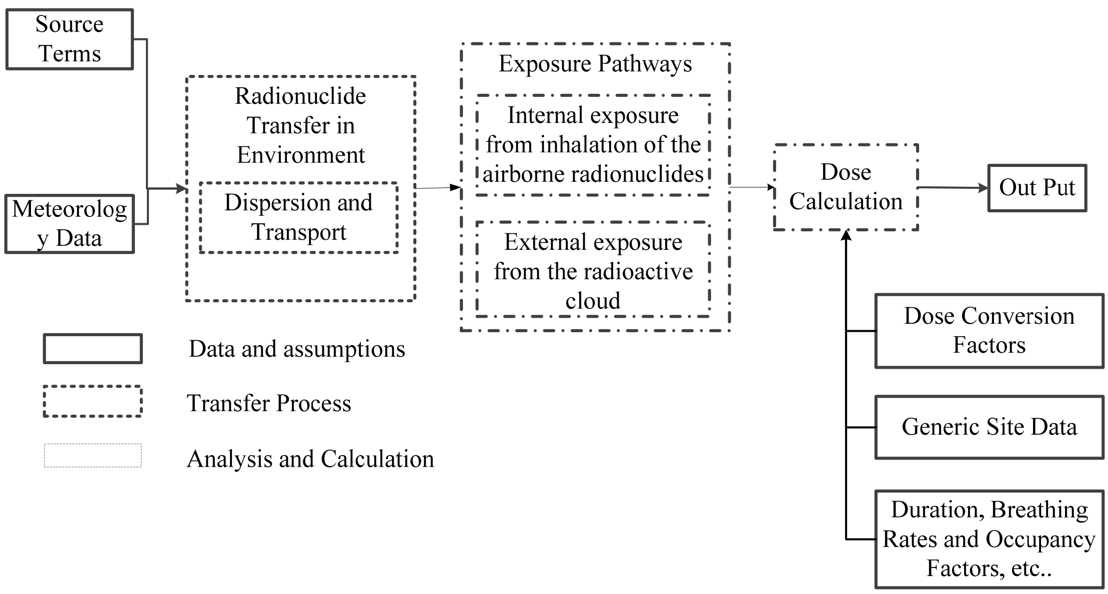
b) Identify the key inputs, assumptions and models which have a nuclear safety significance and impact on the radiological consequences analysis for DBC. An appropriate and prudent safety margin will be considered to compensate for uncertainties in facility parameter, accident progression, and radioactive material transport.

c) Provide appropriate and proportionate evidences to support the assumptions made and models used, which will be presented on a case-by-case basis.

d) Solve the radioactivity balance equations considering the source of release, activity transport and retention mechanism and get the source term results.

Source term results including the quantity of radioactivity, duration of release, rate of release, physical and chemical characteristics of the radionuclides released and height of release point combined with the frequency of the initiating event are used as inputs for dose evaluation to the off-site persons.

The off-site radiological consequence evaluation process is presented in F-12.11-1.



F-12.11-1 Radiological Consequences Analysis Scheme

12.11.5 Analysis for Radiological Representative DBC Accidents

12.11.5.1 Loss of Coolant Accident

12.11.5.1.1 Description

In this section, the release of radioactivity into the environment during LB-LOCA is analysed.

The description of the phenomena and evolution of an LB-LOCA is given in Sub-chapter 12.9.5.2.

In the LB-LOCA, a large amount of primary coolant is released into containment through the break, resulting in rapid increase in containment pressure and release of airborne radioactivity into the environment.

12.11.5.1.2 Analysis Methodology and Assumptions

The analysis of LOCA source terms is based on the Appendix A of Reference [56], considering specific conditions and system designs.

12.11.5.1.3 Release of Radioactivity to Containment

In the blowdown phase, it is conservatively assumed that all of primary coolant is released to containment where airborne radionuclides are uniformly mixed with air inside containment.

The cladding damage stage begins 30s after the accident. It is assumed that radionuclides in the fuel gaps are released to containment immediately and uniformly mixed with atmosphere. The fractions of core fission products released into containment are given in Sub-chapter 12.11.2.2.

12.11.5.1.4 Release of Radioactivity into Environment

The EBA [CSBVS] low-capacity circuit is operating at its maximum flowrate before the accident. All containment isolation valves of the low-capacity EBA [CSBVS] circuit are closed 20 s after the accident.

Before the accident, the normal-operational train of EDE [AVS] operates continuously at its maximum flowrate. During the accident, the EDE [AVS] switches automatically to the safety trains. The accident filters in the safety trains are taken into account after five minutes.

The other removal processes (such as deposition, sedimentation, wall condensation, etc.) are not taken into account. The impact of containment sprinkler system on containment depressurization and airborne radioactive material is not considered.

When the containment pressure has significantly decreased, the leak from inner containment to the containment annulus and that of from the containment annulus to the environment is gradually terminated. It is conservatively assumed that the leak from the containment and the ventilation of the containment annulus can last for 30 days after the accident.

Main assumptions and parameters used in the source term calculation are presented in Reference [57].

12.11.5.1.5 Source Terms

Source terms released through containment are calculated by using the method and parameters above, and are used for dose rate calculation of radiological consequences evaluation.

12.11.5.2 Rod Cluster Control Assembly (RCCA) Ejection Accident

12.11.5.2.1 Description

In Rod Cluster Control Assembly (RCCA) ejection accident, a sharp positive reactivity is rapidly induced into the reactor core, resulting in a distortion to power distribution, causing possible fuel damage and melting of pellets. The goal of the source terms and radiological consequences analysis is to demonstrate that in a rod cluster control assembly ejection accident with the evaluated amount of fuel damage and pellets melting, radioactivity released into environment is effectively limited by containment, related safety engineered features and a leakage rate from primary side to secondary side restricted by the OTS.

12.11.5.2.2 Analysis Methodology and Assumptions

The calculation of source terms in this section is based on the assumption given by Appendix F in Reference [56].

When the accident occurs, with the assumed control rod ejection, radionuclides in the damaged fuel assemblies are released into the primary coolant, resulting in a rapid increase in primary coolant activity.

It is assumed that 10% of the fuel rod cladding is ruptured and 10% of fuel pellets within the damaged fuel rods melt.

There are 2 pathways for the release of radionuclides into the environment:

a) Release From Containment

A rod cluster control assembly ejection accident may cause a break at the top of RPV, from which the radionuclide in primary coolant and damaged fuels may be released into containment atmosphere. The radionuclides from the inner containment leak to the annulus (radioactive materials may be released into environment from the inner containment through the ventilation before the isolation of the containment). The radionuclides in the annulus are released into the environment via ventilation system and filters.

b) Release From Secondary Circuit

Before RHR is connected and operational, the residual heat of the reactor core is removed by the secondary circuit.

It is assumed that this accident occurs simultaneously with LOOP as well as failure of steam discharge condenser, so the residual heat of reactor core is removed by the secondary circuit through the VDA [ASDS]. Due to primary-to-secondary leakage, the radioactivity in secondary circuit increases and is released into the environment through the VDA [ASDS].

12.11.5.2.3 Release of Radioactivity from Containment

a) Release of Radioactivity into Containment

When this accident occurs, it is conservatively assumed that fission products in the damaged fuel rods are released to the primary coolant and then into containment atmosphere through the break of primary circuit and are uniformly mixed with atmosphere.

The fraction releases from the fuel gaps are given in Sub-chapter 12.11.2.2. It is assumed that all noble gases and 25% of the iodines and alkali metals in the damaged fuel rods are available for release from containment.

It is assumed that these radionuclides are released into the containment atmosphere instantaneously when the accident occurs.

b) Release of radioactivity to environment

The EBA [CSBVS] low-capacity circuit is operating at its maximum flowrate before the accident. All the containment isolation valves of the low-capacity EBA [CSBVS] circuit are closed 20 s after the accident.

Before the accident, the normal-operational train of the EDE [AVS] operates continuously at its maximum flowrate. During the accident, the EDE [AVS] switches automatically to the safety trains. The accident filters in the safety trains are taken into account after five minutes.

When the containment pressure has significantly decreased, the leaks from inner containment to the containment annulus and from the containment annulus to the environment are gradually terminated. It is conservatively assumed that the leak from the containment and the ventilation of containment annulus last for 30 days after the accident.

The main assumptions and parameters used in the calculation of containment release pathway are presented in Reference [58].

12.11.5.2.4 Release of Radioactivity from Secondary Circuit

When calculating the secondary circuit release, the release to the containment is not considered. The fraction releases from the fuel gaps are given in Sub-chapter 12.11.2.2. It is assumed that all noble gases and 50% of iodines and alkali metals in the damaged fuel rods are available for release to the primary coolant.

Due to a potential leak from the SG heat transfer tube, the radioactivity of the secondary coolant gradually increases. It is assumed that the leak rate from the primary side to the secondary side for each SG is 44.0 L/h.

All the noble gases released from the primary circuit to the steam generator are assumed to be released into the environment directly without reduction or mitigation.

For iodine and alkali metals, the release is determined by the steam flowrate through VDA [ASDS] and the associated carryover factors. The carryover factor of water in steam for iodine is assumed to be 1% and the entrainment factor of water in steam for cesium is assumed to be 0.25%

After the connection of the RHR, the steam release through VDA [ASDS] is terminated. There is no radioactive release into the environment though the secondary circuit. It is conservatively assumed that the RHR is connected 8 hours after the accident.

The chemical form of radioactive iodine released into the environment from the steam generators are listed as follows:

a) organic iodine: 3%

b) elemental iodine: 97%

The main assumptions and parameters for the source term calculation of secondary circuit are presented in Reference [58].

12.11.5.2.5 Source Terms

Source terms are calculated using the method and parameters above, and are used for dose rate calculation of radiological consequences evaluation.

12.11.5.3 Main Steam Line Break

12.11.5.3.1 Description

Thermal hydraulics analysis results demonstrate the fuel cladding damage will not happen in an MSLB. Mass and energy released into the containment are also analysed and the results demonstrate the mass and energy release into the containment or environment can be sufficiently restricted.

The goal of the source term and radiological consequences analysis is to demonstrate that in the event of an MSLB, radioactivity released into environment is effectively limited by engineered safety features, restricted primary coolant radioactivity and primary-to-secondary leakage rate as restricted by the OTS.

12.11.5.3.2 Analysis Methodology and Assumptions

The analysis method is based on Appendix C of Reference [56].

When an MSLB happens, all the water in the affected SG is released as steam into environment and the initial radioactive materials in the affected SG are released into environment.

The residual heat of the reactor core is removed by ASG [EFWS] and VDA [ASDS] through the two unaffected SGs. Due to the primary-to-secondary leakage, the radioactivity of the secondary side increases and radioactive material is released into the environment through the VDA [ASDS].

It is assumed that a power transient takes place before the MSLB and the primary coolant activity increases to the transient value.

The VDA [ASDS] will continue releasing steam to remove heat until the primary circuit reaches the condition that is suitable for RHR connection.

12.11.5.3.3 Mass Energy Release

Before RHR connection, the residual heat of the reactor core is removed through the secondary circuit.

It is conservatively assumed that all the initial water inventory in the affected SG, main feedwater and auxiliary feedwater pumped to the 3 SGs and primary-to-secondary leakage, are released into the environment during the entire evolution of the accident. It is conservatively assumed that the initial water inventory in the affected SG is 100t.

For the unaffected SG, after the connection of RHR, steam release through VDA [ASDS] is terminated. The radioactivity release to the environment through the secondary circuit is stopped.

It is assumed that the RHR is connected 8 hours after the accident.

12.11.5.3.4 Release of Radioactivity

For the affected SG, all radioactivity contained within this steam generator is released to the environment. The radionuclides contained in the secondary circuit, due to the primary-to-secondary leakage, are released directly to the environment.

For the unaffected SGs, it is assumed that all the noble gases released from primary circuit to secondary circuit are in the gas phase and directly released to the environment.

For iodine and alkali metals, the release is determined by the steam flowrate through the VDA [ASDS] and the associated carryover factors. The carryover factor of water in steam for iodine is assumed to be 1% and the entrainment factor of water in steam for caesium is assumed to be 0.25%

The fractions of different chemical forms of radioactive iodine released to the environment from secondary circuit are listed as follows:

a) organic iodine: 3%

b) elemental iodine: 97%

The main assumptions and parameters used in the source term calculation are presented in Reference [59].

12.11.5.3.5 Source Terms

Source terms released are calculated using the method and parameters above, and are used for the radiological consequences evaluation.

12.11.5.4 SG Tube Rupture (SGTR) (one tube)

12.11.5.4.1 Description

This accident is initiated by the double-ended guillotine rupture of one tube of a steam generator (SG) in full power operation condition.

It is assumed that the primary coolant is contaminated by activated corrosion and fission products at a level corresponding to continuous operation with a limited number of defective fuel rods. After the accident, the radioactive nuclides leak into the secondary side, increasing the activity concentration of the secondary circuit. During the accident, the contaminated steam is released through the atmospheric steam dump system (VDA [ASDA]).

The break mass flowrate and steam release mass flowrate are calculated through the thermal hydraulic analysis of this accident (see Sub-chapter 12.8.5.2).

12.11.5.4.2 Analysis Methodology

12.11.5.4.2.1 Assumptions

The main assumptions used in the SGTR source term analysis are presented in Reference [60].

12.11.5.4.2.2 Primary Coolant Activity

It is assumed that the reactor is in normal operation with 0.25% of the fuel rods containing cladding defects before the accident. The initial primary coolant activity in equivalent I-131 of { } is used.

An iodine spiking effect is initiated by the accident, assuming that the iodine release rate from the fuel rods increases to a value 335 times greater than the equilibrium release rate before the accident.

12.11.5.4.2.3 Activity Transport and Release into the Environment

The primary-to-secondary leakage of the unaffected SG is assumed to be constant during the transient. The break flowrate of the affected SG is determined according to the SGTR accident analysis.

Conservatively, all noble gases contained in the leaked primary coolant are assumed to be released directly to the atmosphere through the VDA [ASDA].

In the unaffected SG, the radionuclides released are determined by the steam flowrate through the VDA [ASDS] and the associated carryover factors. The carryover factors for iodines and alkali metals are 1% and 0.25% respectively.

In the affected SG, the submergence of the break is considered and two different situations are considered in the analysis, as given in Sub-chapter 12.11.2.2.

12.11.5.4.3 Source Terms

Source terms released are calculated using the method and parameters above, and are used for dose rate calculation of radiological consequences evaluation.

12.11.5.5 SGTR (two tubes in one SG)

In SGTR (two tubes in one SG), guillotine break occurs on 2 tubes inside an SG, which is a DBC-4 accident. Radioactive nuclides in primary circuit are released to the secondary circuit and then to the environment. The methodology and assumptions are exactly the same as those used for an SGTR.

12.11.5.6 Fuel Handling Accident

12.11.5.6.1 Description

Accidents occurred during fuel handling operations may be as follows:

a) Fuel assembly drop in the reactor pool;

b) Fuel assembly drop in the spent fuel pool;

c) Spent fuel transport cask drop outside the fuel building;

d) Spent fuel transport cask drop inside the fuel building.

In this section, the first two conditions are considered. The spent fuel transport cask drop accidents will further be analysed in Sub-chapter 12.11.5.13. If a fuel assembly dropped in the reactor pool, the containment sweeping and blowdown ventilation system will shut down automatically. Therefore, in terms of the radiological consequences, a fuel assembly drop in the spent fuel pool would be more serious.

The fuel handling accident analysed in this section is where one fuel assembly drop in the spent fuel pool during handling operations.

Assuming 100 hours after reactor shutdown, one fuel assembly is dropped into the spent fuel pool during transportation to the fuel storage area. All cladding of the fuel rods is assumed to be breached. The entire fission products inventory in the fuel cladding gap is released immediately. The noble gases released will escape from the pool to the fuel handling building. Because of the water solubility and decomposition, the large proportion of iodine will be trapped in the water. The escaped iodine and noble gases will be released to the environment through the fuel building ventilation system.

12.11.5.6.2 Analysis Methodology

12.11.5.6.2.1 Assumptions and Input Parameters

a) Fission Products Inventory of Damaged Fuel Assembly

It is assumed that the accident occurs 100 hours after reactor shutdown, as this is the shortest time for spent fuel being transported to the fuel storage area. The fission products inventory of damaged fuel assembly after 100 hours of shutdown is used in the analysis.

b) Release Fraction from the Fuel Cladding Gap

All the activity in the damaged fuel rods will be released. The release fraction for each nuclide group is presented in Reference [61].

c) Chemical Forms of Iodine

The chemical form of radioiodine released from the fuel to spent fuel pool is assumed to be 95% caesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The CsI will dissociate in the water, because of the low pH of the spent fuel pool, and subsequently form elemental iodine.

d) Decontamination Factors of Spent Fuel Pool Water

The decontamination factor for elemental and organic iodine is 500 and 1, respectively.

e) Fuel Building Ventilation System

Within the first 30 minutes of the accident, the fuel handling hall is ventilated by fuel building ventilation system (DWK [FBVS]). The maximum ventilation flowrate is used. There is no filtration to the radionuclides in this ventilation mode. After 30 minutes, the fuel handling hall is then ventilated by safeguard building controlled area ventilation system (DWL [SBCAVS]). The filtration factors for elemental iodine and organic iodide in the DWL are 1000 and 100, respectively. No filtration of other nuclides is considered.

12.11.5.6.3 Results and Conclusions

The activities of various nuclides released to the environment after fuel handling accident are calculated and will be used to calculate dose rate for the radiological consequences evaluation.

12.11.5.7 Reactor Coolant Pump Seizure (Locked Rotor) Accident

12.11.5.7.1 Description

For a locked rotor accident, Departure from Nucleate Boiling (DNB) may occur and lead to the fuel failure. The source term and radiological consequence analysis for reactor pump locked rotor accident is to verify that engineered safety features for mitigating the accident consequences, limitations on the primary coolant activity allowed by Operating Technical Specification (OTS) and restrictions on the primary/secondary side leakage rate can contribute to limiting the radioactive release to the environment.

12.11.5.7.2 Methodology and Assumptions

12.11.5.7.2.1 Assumptions

The analysis method used in this report is mainly based on Reference [56] which supports the development of the assumptions for evaluating the radiological consequences of a PWR locked rotor accident. According to specific analysis case and system design, these assumptions are applied to the source term analysis of a UK HPR1000 locked rotor accident.

In the analysis, 10% of fuel cladding is assumed to be ruptured. The radioactivity will be released into the reactor coolant. This is assumed to occur instantaneously and homogeneously throughout the primary coolant and dramatically increases the activity of the primary circuit.

Assuming that the steam dump condenser fails, residual heat is removed by the ASG [EFWS] and VDA [ASDS]. Considering the primary-to-secondary leakage, radioactivity in the secondary side will increase and the radiological material will be released to the environment through VDA [ASDS].

Steam from the secondary side will continue to be discharged until the RCP [RCS] reaches the temperature and pressure value for connection of the RIS [SIS] in RHR mode.

12.11.5.7.2.2 Activity Released to the Environment

a) Mass release

Before the connection of the RIS/RHR, steam is continuously released to the environment through the secondary system.

After the connection of the RIS/RHR, steam release through the VDA [ASDS], and therefore the radioactive release, is terminated.

Assuming the steam generator of the affected reactor coolant loop is free from heat removal, the RIS/RHR connection is finished after 8 hours of the accident.

b) Activity release

It is assumed all noble gases released from the primary circuit to the steam generator are released directly to the environment.

The radionuclides released are determined by the steam flowrate through VDA [ASDS] and carryover factors. The carryover factors for iodines and alkali metals are 1% and 0.25% respectively.

The chemical forms of radioactivity iodine released to the environment from the steam generators are listed as follows:

1) Organic iodine: 3%;

2) Elemental iodine: 97%.

The main assumptions and parameters used in the source term calculation are presented in Reference [62].

12.11.5.7.3 Results

Source terms released are calculated with the method and parameters above, and are used for the radiological consequences evaluation.

12.11.5.8 Rupture of a Line Carrying Primary Coolant Outside Containment

12.11.5.8.1 Description

During normal operation, rupture of a small line outside the containment which carries primary coolant will lead to release of radioactivity. The loss of primary coolant may be caused by failure of

- The nuclear sampling system (REN);
- The chemical and volume control system [CVCS].

The maximum break flowrates are well within the capacity of the normal operational coolant makeup using the RCV [CVCS].

a) Rupture in the Chemical and Volume Control System Discharge Line

Before diverting out of the containment, the coolant passes through the RCV [CVCS] heat exchanger and pressure reducing valve. The leak flowrate and coolant temperature are limited.

b) Rupture in the Reactor Coolant System Sample Line

The liquid sample lines that penetrate the containment contain two containment isolation valves, one located inside the containment and the other outside. The containment isolation valves are opened only when sampling is being performed.

Considering the low temperature of the flow and restricted flowrate, a rupture in the RCV [CVCS] discharge line is not analysed. However, the postulated sample line break is analysed, assuming that the failure of the sample line occurs downstream of the containment penetration.

12.11.5.8.2 Methodology and Assumptions

It is postulated that sampling is being performed before the accident during normal operation. The break is located just downstream of the containment penetration.

The initial primary coolant activity is assumed to be { } I-131 equivalent activity. An iodine spike occurs at the beginning of the accident, and a factor of 500 is considered for the release rate from the fuel.

A constant leak flowrate is considered during the transient.

The noble gases contained in the break flow are assumed to be released into the auxiliary building and then to the environment.

The fraction of iodine and caesium available for release to the environment is assumed to be equal to the coolant flash fraction. The flash fraction is determined by assuming the discharge is a constant enthalpy process. Retention within the auxiliary building is not considered.

The line is isolated 60 min after the accident (30 min for signal actuation and 30 min for manual operator action).

12.11.5.8.3 Results

Source terms released are calculated using the method and parameters above, and are used for the radiological consequences evaluation.

12.11.5.9 Residual Heat Removal System Break Outside Containment

This postulated accident is a double-ended rupture of a suction line on the residual heat removal system (RIS in RHR mode) just downstream of the containment penetration. The break is assumed to occur on a train operated in RHR mode.

12.11.5.9.1 Methodology and Assumptions

During a normal reactor cooling sequence, two RIS trains are connected at a pressure below 3.2 MPa and a temperature of approximately 120℃. The break occurs immediately when RHR conditions are reached in the RCP [RCS].

The RIS lines are automatically isolated to limit the coolant leak in the auxiliary building. The affected RIS is detected following a high sump level inside the affected auxiliary building. During the transient, the core remains fully covered. The break flowrate from the primary coolant is calculated.

In addition, the backflow of the coolant from the stationary volume within the RIS is taken into account.

The activity concentration of the primary coolant is assumed to be { } I-131 equivalent activity. The transient values are used considering the shutdown transient.

The noble gases contained in the discharged coolant are assumed to be released in to the auxiliary building and then to the environment.

The release fraction of the iodine and caesium to the environment is calculated considering that activity concentration in the steam is 10% of the concentration in the discharged coolant that has not evaporated. Retention within the auxiliary building is not considered.

12.11.5.9.2 Results

Source terms released are calculated using the method and parameters above, and are used for the radiological consequences evaluation.

12.11.5.10 Gaseous Waste Tank Break

A Gaseous Waste Tank Break accident is caused by a leak from or damage to the Gaseous Waste Treatment System (TEG [GWTS]), resulting in a release of radioactive nuclides. The complete rupture of the delay unit is regarded as the radiologically representative event because of the bounding quantity of radioactive gaseous waste released. The release continues until the affected pipes are isolated.

During stable-state operation of the TEG [GWTS], there will normally be no radioactive gas released into delay unit. Before shutdown, the Coolant Storage and Treatment System (TEP [CSTS]) is put into operation to reduce the radioactivity of primary coolant. The primary coolant is degassed by the degasifier column, and then the non-condensable gases are transferred into the TEG [GWTS] after being cooled by the gas cooler. Once the TEG [GWTS] switches to surge-state operation mode, the radioactive material will accumulate in the delay unit.

It is assumed that all non-condensable gases flow into the TEG [GWTS] delay unit after being degassed by the TEP [CSTS], and these are released immediately after a delay unit rupture without any retention by the delay beds being credited.

Radioactivity within the delay unit will continue to rise at the preliminary stage whilst non-condensable gases flow into TEG [GWTS]. Radioactivity within the delay unit will decrease after reaching to its maximum value. In the analysis, the delay unit rupture is assumed to occur at peak radioactivity.

12.11.5.10.1 Release Pathways

The radioactivity released includes:

a) Radioactivity accumulated in delay unit before rupture;

b) Radioactivity released from the TEP [CSTS] pipes prior to isolation.

The radioactivity released into the gaseous phase of the room is assumed to be immediately released into the environment through the ventilation system. In reality, after being released into the room, part of the radioactive material would be retained in the building and ventilation system. However, Retention is not considered in the analysis.

12.11.5.10.2 Main Assumptions

The main assumptions of accident of the radioactive gaseous waste treatment system leakage are presented in Reference [63].

a) Flow rate of the TEP[CSTS] degasifier column

The flow rate of the TEP [CSTS] degasifier column is consistent with the Chemical and Volume Control System RCV [CVCS] letdown flowrate, with a maximum flowrate of $75x10^3$ kg/h.

b) Primary coolant specific activity

The primary coolant specific activity after a power transient is considered in this accident ({ } I-131 Equivalent Specific Activity).

c) Decontamination factor of the RCV [CVCS] and TEP [CSTS] mixed bed

The decontamination effect on the radioactive nuclides by the RCV [CVCS] and TEP [CSTS] mixed bed demineraliser is not considered.

d) Partitioning coefficient in the TEP [CSTS] degasifier column

The partitioning coefficient of iodine and caesium in the TEP [CSTS] degasifier column is assumed to be { }.

e) Reactor coolant density

Reactor coolant density is conservatively assumed to be $1.0x\ 10^3$ kg/m^3.

f) TEP [CSTS] pipe isolation time

The isolation time of the TEP [CSTS] pipe is conservatively assumed to be 1hr (30 minutes for signal actuation and 30 minutes for operator action).

12.11.5.10.3 Source Terms

Source terms released into the environment are calculated using the method and parameters above, and are used for the radiological consequences evaluation.

12.11.5.11 Liquid Waste Tank Break

The whole liquid waste treatment system is located inside the radioactive waste processing building, including 5 subsystems: liquid waste storage subsystem, liquid waste treatment subsystem, discharge monitoring subsystem, chemical dosing subsystem and sampling analysis subsystem. These subsystems respectively perform the functions of collecting and storing liquid radioactive wastes, treating liquid wastes, monitoring discharge, injecting chemical reagents and sampling. With regards to this system, the most conservative potential radioactive material release would be a leak from the liquid waste storage subsystem storage tank.

The liquid waste storage subsystem has two process drains storage tanks, two chemical drains storage tanks, three floor drains storage tanks and two laundry liquid waste storage tanks to store different kinds of radioactive liquid wastes.

Each category of storage tanks is equipped with one liquid waste delivery pump and the flowrate of each pump is $50m^3/h$. During system operation, it must be ensured that at least one of the liquid waste storage tank of same type is reserved for collecting liquid waste. When the high level of the liquid waste storage tank is reached, the inlet isolation valve of this tank is closed and that of another one of same type opened by the operator.

The worst case is where a leakage occurs from the waste catch tank with the highest level of radioactivity. The capacity of one catch tank is limited to $50m^3$, and the radioactivity of the contents is also limited. Thus, the radioactive release resulting from a liquid waste tank break accident is also limited and it can be bounded by a volume control tank break accident (see Sub-chapter 12.11.5.12).

12.11.5.12 Volume Control Tank Break

Leakage from or damage to the RCV [CVCS] tank will result in a primary coolant leak. This falls under the category of loss of primary coolant outside containment and is a typical example of a system leak accident. Damage to the volume control tank is assumed.

The volume control tank is designed to compensate for volume changes to the primary coolant during various operating conditions. Part of the letdown flow and primary coolant pump shaft seal reflux, which converge on the letdown bypass line, will flow into the volume control tank. The mixture of these flows will ensure the boron density within the volume control tank is consistent with that of primary coolant.

12.11.5.12.1 Release Pathways

If volume control tank fails structurally, its contents will be released into the area in which the tank stands. Moreover, until the operator isolates the letdown line, a certain amount of liquids will also be released continuously. The radioactivity released includes:

a) Radioactivity from the gaseous phase of the tank;

b) Radioactivity from the liquid phase of the tank;

c) Radioactivity carried in liquid released from the letdown line before isolation.

The gaseous radioactive material released into the area is assumed to flow directly into the environment through the ventilation system. In reality, after being released into the room, part of the radioactive material would be retained by the building and ventilation system. However, retention is not considered in the analysis.

12.11.5.12.2 Main Assumptions

The main assumptions and parameters used in the volume control tank break accident are presented in Reference [64].

a) Primary coolant activity

The primary coolant activity after a power transient is considered in this accident ({ } I-131 Equivalent Specific Activity).

b) Gaseous phase and liquid phase volume of the volume control tank

The gaseous phase volume and liquid phase volume of the volume control tank are 8 m^3 and 7 m^3 respectively.

c) Flow rate into the volume control tank

The flow rate into the volume control tank considered in this accident is the maximum letdown flow rate 75 t/h.

d) Airborne release fraction

Part of the spilled liquids will become airborne. The maximum airborne release fraction during the free-fall of aqueous solutions onto a hard surface of $2x10^{-4}$ is used.

e) Partitioning coefficient in the volume control tank

The partitioning coefficients of iodine and caesium in the volume control tank are assumed to be $1.0x10^{-4}$.

f) Primary coolant density

Primary coolant density is assumed to be $1.0x10^{3}$ kg/m^3 conservatively.

g) Isolation time

The isolation time is conservatively assumed to be 1h from volume control tank damage (30 minutes for signal actuation and 30 minutes for operator action).

h) Decontamination factor of RCV [CVCS] mixed bed

The decontamination of radioactive nuclides by the RCV [CVCS] mixed bed demineraliser is conservatively not taken into consideration.

12.11.5.12.3 Source Terms

Source terms released into the environment are calculated by using the method and parameters above, and are used for dose rate calculation for radiological consequences calculation.

12.11.5.13 Spent Fuel Transport Cask Drop

The spent fuel transport cask drop accident can occur when the crane is loading or unloading fuel.

Spent fuel is stored in the spent fuel pool for more than 6 months before the transportation outside the fuel building. Concerning the radioactive decay of the inventory of the spent fuel assemblies in the cask, the fuel assembly inventory is very limited.

If the accident happened during transportation outside the fuel building, the cask will only be slightly above the ground and the drop height will be low and well within the capability of the cask. So it will not cause break of the cask.

If the accident happens inside the fuel building, the radioactive release is even more insignificant (due to filtration by the fuel building). And also, on the one hand, the cask must have satisfied the drop tests for demonstrating its ability to withstand a transport accident. The drop test consists of the falling of the specimen cask onto the target so as to suffer maximum damage; On the other hand, the fuel cask crane is designed to limit the risk of dropping the fuel casks during transportation and will also be appropriately tested and via a planned commissioning strategy prior to the start of commercial operation.

So the radiological consequences of both of the above cases can be limited.

12.11.5.14 Failure of Radioactivity Containing Equipment in Nuclear Auxiliary Building

Failure of equipment containing radioactive material within the nuclear auxiliary building (NAB) may result in a radioactive release to the environment. A leak from or failure of the upstream RCV [CVCS] lines in the nuclear auxiliary building is representative of the failure of equipment containing radioactive material in the NAB, and is analysed in Sub-chapter 12.11.5.12.

12.12 Fault Schedule Production

The production of a Fault Schedule is relevant good practice (RGP) in the UK. The Fault Schedule establishes an auditable link between initiating faults considered in the design and requirements to be applied to safety measures of a nuclear power plant. This visible link helps both designers and regulators to understand the status of design and work in a more efficient way. Thus a Fault Schedule for the UK HPR1000, which is a key tool for the UK HPR1000 design evolution during Generic Design Assessment (GDA), will be produced and provided to ONR. This document illustrates the methodology utilised to produce the Fault Schedule for the UK HPR1000.

The objectives of the Fault Schedule to be produced are:

a) Establish a visible link between initiating faults considered in the design and requirements to be applied to safety measures (golden thread);

b) Help provide confidence that probabilistic targets set for the UK HPR1000 will ultimately be met;

c) Provide a connection between the deterministic analysis and the probabilistic analysis;

Two versions of the Fault Schedule are to be produced:

1) One is the early version of the Fault Schedule. It is a start point for Fault Schedule evolution, which is based on the DBC list in the PSR and the diverse protection line is not demonstrated by the transient analysis. This Fault Schedule will be used to inform the transient analysis for diverse protection lines.

2) The second is the UK HPR1000 Fault Schedule, which will be based on the DBC List for the UK HPR1000. The diverse protection lines will have been demonstrated by the transient analysis when this version of the UK HPR1000 Fault Schedule is released.

Reference [19] provides the detailed production methodology for the Fault Schedule. The draft template of the Fault Schedule is presented in T-12.12-1. The following information is provided for each event:

a) All initiating faults considered in the design with their respective frequencies;

b) The safety functions involved for each initiating fault;

c) The safety systems to physically achieve the safety functions, including the mechanical systems, the signals to trigger the safety functions, the I&C platform to deal with the signal, the way of safety function actuation;

d) Categorisation and classification of the safety systems;

e) Links between the faults and their detailed analysis, which include the overall claim for the fault and the consequences.

The Fault Schedule will be continuously updated throughout GDA and will be used as an input for the design development. During production of the Fault Schedule, gaps including missing safety functions, insufficient categorisation/classification or lack of transient analysis will be identified.

T-12.12-1 Template of Fault Schedule

Event	Event category and frequency	Fundamental Safety Function	High Level Safety Function	Low Level Safety Function	Safety Function (*Launched Automatically or Manually*)	Signal	Safety system	I&C platform	Safety Class	Supporting studies
				Main Protection Line						
				Diverse Protection Line						

12.13 ALARP Evaluation

To support the overall ALARP demonstration addressed in Chapter 33, Chapter 12 provides the assessment of the effective dose received by any person arising from design basis fault sequence` in terms of Numerical Target 4.

The DBC analysis will be updated at the different stages of the GDA process to accurately reflect design changes and demonstrate the acceptance criteria are met.

Based on the appropriate analysis approach demonstrated in this chapter, the DBC analysis plays the following roles in the ALARP demonstration of the UK HPR1000 design:

a) Demonstration that the UK HPR1000 meets target 4 following a systematic and comprehensive DBC analysis;

b) Identification of improvements when the safety margin is low or the acceptance criteria cannot be met regarding the DBC analysis results;

c) Providing input for optioneering studies during the ALARP assessment to contribute to the determination of whether the implementation of improvements is reasonably practicable;

d) Outcomes from ALARP reviews will be incorporated into the DBC analysis and presented during various steps of the GDA process (and beyond)

12.14 Concluding Remarks

Chapter 12 is for demonstrating that the radiological consequences following design basis faults on the UK HPR1000 are within the acceptable criteria required in Chapter 4 and that the risks under design basis conditions will be demonstrated as ALARP with the design evolution of UK HPR1000. This Chapter provides evidence as follows:

a) All potential Initiating design basis faults are being identified and fault sequences have been developed which are listed in this Chapter. The SSCs required to protect the plant following the faults based on the DBC list from FCG 3 have been identified and will be recorded in a comprehensive Fault Schedule.

b) A suite of computer codes has been or will be identified with verification and validation evidence to justify their use in the analysis of design basis faults.

c) Detailed analysis of the design basis faults identified has been or will be undertaken using appropriate codes to confirm that the SSCs claimed in the fault schedule are sufficient to ensure that the radiological releases are limited to an acceptable level. The analyses demonstrate that the relevant acceptance criteria identified are met following identified design basis faults.

d) The radiological analysis methodology described in this Chapter has demonstrated that the radiological consequences of the design basis faults identified will be appropriately assessed to confirm whether the radiological acceptance criteria required in Chapter 4 can be met.

e) Potential design changes to reduce the consequences of design basis events will be identified in the individual system chapters (Chapters 5-11). This will provide input to the overall ALARP assessment discussed in Chapter 33.

12.15 References

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Appendix 12A: Computer Codes Used in the Fault Studies and Radiological Release Analysis

The computer codes used in the Fault Studies and Radiological Release Analysis are listed in T-12A-1 to T-12A-3. The brief descriptions of computer codes used are given as follows.

T-12A-1 Computer Codes Used in Chapter 12 (Analyses of DBC-2 Events)

Sub-chapter	DBC Faults	Computer codes for fault analysis
12.7.1.1	Feedwater system malfunctions causing a reduction in feedwater temperature (State A\B)	GINKGO LINDEN
12.7.1.2	Feedwater system malfunctions causing an increase in feedwater flow (State A\B)	GINKGO LINDEN
12.7.1.3	Excessive increase in secondary steam flow (State A\B)	GINKGO LINDEN
12.7.2.1	Turbine trip (State A)	GINKGO LINDEN
12.7.2.2	Loss of condenser vacuum (State A)	/
12.7.2.3	Short term loss of off-site power (< 2 hours) (State A)	GINKGO LINDEN
12.7.2.4	Loss of normal feedwater flow (loss of all main feedwater pumps and Startup and Shutdown Feedwater System (AAD [SSFS]) pumps) (State A)	GINKGO LINDEN
12.7.2.5	Loss of one cooling train of the Safety Injection System (RIS [SIS]) in Residual Heat Removal (RHR) mode (State C\D)	/
12.7.2.6	Spurious reactor trip (State A)	/
12.7.3.1	Partial loss of core coolant flow (loss of one main coolant pump) (State A)	GINKGO LINDEN
12.7.4.1	Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at a subcritical or low power startup condition (State A)	COCO POPLAR GINKGO LINDEN BIRCH
12.7.4.2	RCCA bank withdrawal at power (State A)	GINKGO LINDEN
12.7.4.3	RCCA misalignment up to rod drop without limitation (State A)	COCO POPLAR GINKGO LINDEN

Sub-chapter	DBC Faults	Computer codes for fault analysis
12.7.4.4	Startup of an inactive reactor coolant loop at an improper temperature (State A)	/
12.7.4.5	Chemical and Volume Control System (RCV [CVCS]) malfunction that results in a decrease in boron concentration in the reactor coolant (State A to E)	COCO, PINE
12.7.5.1	RCV [CVCS] malfunction causing an increase in (RCP [RCS]) inventory (State A)	GINKGO LINDEN
12.7.6.1	RCV [CVCS] malfunction causing a decrease in (RCP [RCS]) inventory (State A)	GINKGO LINDEN
12.7.6.2	Uncontrolled RCP [RCS] level drop in shutdown states with RIS [SIS] connected in RHR mode (State C\D)	/
12.7.7.1	Spurious pressuriser heater operation (State A)	GINKGO LINDEN
12.7.7.2	Spurious pressuriser spray operation (State A)	GINKGO LINDEN
12.7.8.1	Loss of one train of the Fuel Pool Cooling and Treatment System (PTR [FPCTS]) or of a supporting system (State A)	/

T-12A-2 Computer Codes Used in Chapter 12 (Analyses of DBC-3 Events)

Sub-chapter	DBC Faults	Computer codes for fault analysis
12.8.1.1	Inadvertent opening of an SG relief train or of a safety valve (State A)	GINKGO LINDEN
12.8.1.2	Small steam system piping break including breaks in connecting lines (State A\B)	/
12.8.2.1	Inadvertent closure of all or one main steam isolation valves (State A)	GINKGO LINDEN
12.8.2.2	Long term LOOP (> 2 hours) (State A)	LOCUST
12.8.2.3	Small feedwater system piping break including breaks in connecting lines to SG (State A\B)	/
12.8.3.1	Forced reduction in reactor coolant flow (3 pumps) (State A)	GINKGO LINDEN
12.8.4.1	Inadvertent loading of a fuel assembly in an improper position (State E)	COCO
12.8.4.2	Uncontrolled RCCA bank withdrawal (State B\C\D)	/
12.8.4.3	Uncontrolled single RCCA withdrawal (State A)	COCO POPLAR GINKGO LINDEN BIRCH
12.8.5.1	Rupture of a line carrying primary coolant outside containment (e.g. nuclear sampling line) (State A)	/
12.8.5.2	SG Tube Rupture (SGTR) (one tube) (State A)	LOCUST
12.8.5.3	Small Break (Loss of Coolant Accident) (SB-LOCA) (at power) including a break in the Emergency Boration System (RBS [EBS]) injection line (State A)	LOCUST
12.8.5.4	Small break LOCA (at shutdown, RIS [SIS] not connected in RHR mode) including a break in the RBS [EBS] injection line (State A\B)	LOCUST
12.8.5.5	Inadvertent opening of a pressuriser safety valve (State A)	/
12.8.6.1	Gaseous waste tank break (State A to F)	/
12.8.6.2	Liquid waste effluent tank break (State A to F)	/
12.8.6.3	Volume control tank break (State A to F)	/

Sub-chapter	DBC Faults	Computer codes for fault analysis
12.8.7.1	LOOP (>2 hours) affecting fuel pool cooling (State A)	/
12.8.7.2	Loss of one train of the PTR [FPCTS] or of a supporting system (with the reactor core offloaded to the fuel pool) (State F)	/
12.8.7.3	Isolatable piping failure on a system connected to the spent fuel pool (State A to F)	/

T-12A-3 Computer Codes Used in Chapter 12 (Analyses of DBC-4 Events)

Sub-chapter	DBC Faults	Computer codes for fault analysis
12.9.1.1	Large steam system piping break (State A\B)	COCO GINKGO LINDEN
12.9.1.2	Inadvertent opening of an SG relief or safety valve (State B)	/
12.9.2.1	Large feedwater system piping break (State A\B)	LOCUST
12.9.2.2	Long term LOOP (State C)	/
12.9.3.1	Reactor coolant pump seizure (locked rotor) or Reactor coolant pump shaft break (State A)	GINKGO LINDEN BIRCH
12.9.4.1	Spectrum of RCCA ejection accidents (State A)	COCO GINKGO LINDEN BIRCH
12.9.4.2	Boron dilution due to a non-isolatable rupture of a heat exchanger tube (State C\D\E)	COCO, PINE
12.9.5.1	SGTR (two tubes in one SG) (State A)	LOCUST
12.9.5.2	Large Break (Loss of Coolant Accident) (LB-LOCA) (up to double-ended break) (State A)	LOCUST
12.9.5.3	Intermediate Loss of Coolant Accident (LOCA) (State A\B)	LOCUST
12.9.5.4	SB-LOCA, including a break in the emergency boration system injection line (State C\D)	LOCUST
12.9.5.5	RHR system piping break inside (outside) containment (≤DN 250) (State C\D)	LOCUST
12.9.5.6	Inadvertent opening of the dedicated depressurisation device (State A\B)	/
12.9.6.1	Fuel handling accident (State A to F)	/
12.9.6.2	Spent fuel transport cask drop (State A to F)	/
12.9.6.3	Failure of radioactivity containing equipment in nuclear auxiliary building (State A to F)	/
12.9.7.1	Non-isolatable small break or isolatable RIS [SIS] break (≤DN 250) in RHR mode affecting fuel pool	/

Sub-chapter	DBC Faults	Computer codes for fault analysis
	cooling (during refuelling) (State E)	

a) PCM

PCM is a nuclear design code package that contains PINE and COCO. PINE is an advanced Pressurized Water Reactor (PWR) fuel assembly calculation code, and COCO is a three-dimensional (3-D) core calculation code. PINE generates two-group parameter tables for macroscopic cross-sections and the assembly discontinuity factors, which COCO uses to calculate these parameters.

1) PINE

PINE performs 2-D lattice calculation for single assembly and multiple assemblies of PWR and generates two-group parameter tables. The parameters include diffusion coefficients, macroscopic cross-section, surface dependent discontinuity factors, xenon and samarium microscopic densities, flux shape factor for power reconstruction and kinetic parameters.

PINE uses multi-group cross section databank of IAEA WIMS-D update program.

The physical models of PINE include resonance calculation, transport calculation, leakage correction and burn-up calculation.

The equivalence principle is applied to carry out resonance calculation. The Method of Characteristics (MOC) is applied to perform two-dimensional heterogeneous transport calculation. B1 approximation is applied to take into account the leakage effect. PINE uses three different advanced burn-up calculation strategies, which are Projected Predictor-Corrector (PPC) method, Linear Rate (LR) method and Log Linear Rate (LLR) method.

2) COCO

COCO is used for PWR nuclear reactor design. The main functions include loading pattern design, critical boron concentration search, evolution calculation, control rod worth assessment, reactivity coefficients calculation, shutdown margin calculation, etc. COCO is also used to perform transient calculations such as Reactivity Induced Accidents (RIA).

The solver of COCO is based on Nodal Expansion Method (NEM) which can handle 2-D and 3-D geometries. The NEM solver can provide flux distribution in full core and 1/4 core geometries. Furthermore, the NEM solver is accelerated using CMFD method.

The feedback of COCO includes a closed-channel thermal-hydraulic model, which is responsible for moderator temperature and density, and a fuel temperature calculation model.

Both microscopic and macroscopic burn-up models are developed. The former focuses on the fission products, minor actinides, etc. The latter handles the intra-node burn-up distribution. In macroscopic burn-up, nodal surface burn-up is calculated to correct cross-sections.

The PCM nuclear design code package has been intensively validated against Nuclear Power Plant (NPP) data, experimental data and benchmarks.

b) POPLAR

POPLAR is a 1-D neutron diffusion-depletion code. POPLAR is used to perform bite calculation, calibration calculation, xenon depletion calculation, transient xenon calculation, control rod worth calculation and control rod cross section modification. Furthermore, POPLAR is used for transient calculation.

POPLAR obtains relevant input of the core from COCO, and the tables of few-group parameters from PINE.

The physical models of POPLAR include cross section interpolation, 3-D to 1-D conversion, two-group 1-D diffusion solver, leakage correction, thermal feedback and 1-D control rod insertion.

The validation of the POPLAR includes NPP data validation.

c) GINKGO

GINKGO is a system transient analysis code, which is used to analyse PWR transients under normal operating conditions and accident conditions. For these transients, GINKGO simulates the reactor vessel and core, hot and cold legs, pressurizer, steam generator, reactor coolant pump in PWR plant. The modelling of Nuclear Steam Supply System (NSSS), Engineered Safety System (ESS), Reactor Protection System (RPS), Instrumentation and Control System (I&C) and secondary system components are also taken into account.

To account for the thermal-hydraulics features of the coolant in different transients, the separated phase model at thermal equilibrium is used in the code. Three governing mixture balance equations combined with a drift-flux model are applied. A reactor point kinetic model with six-group delayed neutron and a simplified decay heat model are combined to predict the core transients.

The validation of GINKGO includes separate effect validation and integral effect validation.

d) LINDEN

LINDEN is a sub-channel analysis code which is used for thermal-hydraulic design and safety analysis of reactor core. It calculates the thermal-hydraulic parameters of coolant in reactor core under various conditions, including pressure, mass flowrate, quality and void fraction, etc. Based on the calculated thermal-hydraulic parameters, the Departure from Nucleate Boiling (DNB) of reactor core can be predicted by using a specific Critical Heat Flux (CHF) correlation.

The two-phase flow in LINDEN is the four equations model combined with a drift-flux correlation. It takes into account the slip velocity between liquid and vapour phases and also the thermal non-equilibrium of liquid phase during sub-cooled boiling. The flow model has four conservation equations, including a mixture mass equation, a mixture energy equation, a mixture momentum equation and a liquid energy equation. Among which, the liquid energy equation is used to simulate the thermal non-equilibrium of liquid phase during sub-cooled boiling.

The validation of LINDEN includes experimental data validation and NPP operational data validation.

e) LOCUST

LOCUST is a system thermal-hydraulic code which has the capability of performing LOCA analysis. It focuses on the analysis of LBLOCA, IB/SBLOCA, SGTR, etc.

LOCUST is used to simulate two-fluid, non-equilibrium, and heterogeneous hydrodynamic conditions in various NPP transients. A six-equation model is employed in hydrodynamics model, which forms the trunk of LOCUST. Auxiliary models include heat structure model, trip system, control system, and point reactor kinetics model.

The most important features of LOCUST are flexible nodalization, capability to analyse two-fluid, thermal non-equilibrium in all fluid volumes. The code is incorporated with models to simulate special processes such as choked flow, thermal stratification, and counter-current flooding limitations.

The validation of LOCUST includes separate effect validation and integral effect validation.

f) BIRCH

BIRCH is a fuel rod temperature analysis code, mainly used to analyse the integrity of fuel rod in PWR under accident conditions. BIRCH calculates the radial temperature distribution of a fuel rod and the heat flux of cladding surface during transient conditions. In addition, it also calculates energy storage in the fuel pellet, gap heat transfer coefficient and thermal expansion of pellet and cladding, etc.

A 1-D heat conduction differential equation is implemented in heat conduction model together with auxiliary closure models, including cladding-coolant heat transfer model, gap conductance model, water-zircaloy reaction model and fuel pellet melting model, etc.

The physical properties of coolant and data of evolution of nuclear power required in the calculation are obtained from the system transient analysis code GINKGO or 3-D core calculation code COCO.

The validation of BIRCH includes separate effect validation and integral effect validation.

g) CATALPA

CATALPA is a containment analysis code. It is used to determine the change of pressure and temperature with time inside the containment under the accidents that result in significant release of high-energy fluid into the containment, such as Loss of Coolant Accident (LOCA) and Steam Line Break Accident (SLB).

The physical model of the code is a single-volume model. And the substances inside the containment are divided into two systems: a gas phase system and a liquid phase system. The mass, energy and volume conservation laws and the ideal-gas law are used to build conservation equations. In addition, steam condensation, heat conduction of structural components, heat transfer and energy absorption of containment wall are also considered in the physical models.

The code has been validated using experimental data, including separate effect experiments and integral effect experiments.