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UK HPR1000 GDA Project

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Preliminary Safety Report
Chapter 12
Design Basis Conditions Analysis

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12.1 List of Abbreviations and Acronyms

AAD	Startup and Shutdown Feedwater system[SSFS]
ACC	Accumulator
ARE	Main Feedwater Flow Control System [MFFCS]
ASG	Emergency Feedwater System[EFS]
CPR1000	Chinese Pressurized Reactor
DBA	Design Basis Analysis
DBC	Design Basis Condition
DN	Nominal Diameter
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOC	End of Cycle
FC1	Safety Category 1 Function
FC2	Safety Category 2 Function
FCG Unit3	Fangchenggang Nuclear Power Plant Unit 3
FMEA	Failure Modes and Effects Analysis
FP	Full Power
GDA	Generic Design Assessment
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
LB-LOCA	Large Break(Loss of Coolant Accident)
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident

LOOP	Loss of Off-Site Power
MHSI	Medium Head Safety Injection
MSIV	Main Steam Isolation Valve
NNSA	National Nuclear Safety Administration
PCSR	Pre-Construction Safety Report
PCT	Peaking Cladding Temperature
PIE	Postulated Initiating Events
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSR	Preliminary Safety Report
PTR	Fuel Pool Cooling and Treatment System[FPCTS]
RBS	Emergency Boration System[EBS]
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant System[RCS]
RCPB	Reactor Coolant Pressure Boundary
RCV	Chemical and Volume Control System[CVCS]
RHR	Residual Heat Removal
RIS	Safety Injection System[SIS]
SFC	Single Failure Criterion
SG	Steam Generator
SLB	Steam Line Break
SSCs	Structures, Systems and Components
UK HPR1000	The UK version of the Hua-long Pressurized Reactor
URWP	Uncontrolled RCCA bank Withdrawal at Power
VDA	Main Steam Relief Train[MSRT]
WR	Wide Range

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

12.2 Introduction

This chapter supports the following high level objective: the UK version of the Hua-long Pressurized Reactor (UK HPR1000) design will be developed in an evolutionary manner, using robust design processes, building on relevant good international practice, to achieve a strong safety and environmental performance.

This chapter will demonstrate the following:

a) All initiating faults with the potential to lead to significant radiation exposure or release of radioactive material will be identified in the Fault Schedule;

b) The Design Basis Analysis (DBA) provides a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety measures.

In addition, this chapter includes the initiating events list for DBA required to be considered for the assessment of Hua-long Pressurized Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)), along with example analysis results for three key initiating events.

The scope of fault identification for the DBA and a summary of the methodology of fault identification and fault grouping required for the Hua-long Pressurized Reactor (HPR1000) safety case are addressed in sub-chapter 12.3.

In sub-chapter 12.4, an outline of the process to be followed to produce the fault schedule for UK HPR1000 and key elements to be included in the resultant fault schedule are discussed.

In sub-chapter 12.5, the DBA methodology and assumptions, analysis rules and main assumptions considered in the DBA for HPR1000 are briefly described.

In sub-chapter 12.6, the key requirements on the DBA, a discussion on the scope of assessment and a brief summary of the DBA results are presented. As the detailed analysis procedure and results to support the GDA assessment of the UK HPR1000 will be presented within the Generic Design Assessment (GDA) Pre-Construction Safety Report (PCSR), assessments of several typical events for FCG Unit3 are shown as examples to illustrate the performance of HPR1000 following design basis faults and to support the requirements discussed above.

12.3 Fault Identification and Fault Grouping Methodology

12.3.1 Scope of Fault Identification

A Postulated Initiating Event (PIE) is an event that leads to anticipated operational occurrences or accident conditions. The PIEs considered at 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 some other activity containing sources.

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

12.3.2 Fault Identification and Fault Grouping Methodology

The PIEs for HPR1000 (FCG3) are identified on the basis of engineering judgements and a combination of deterministic assessment and probabilistic assessment. In China, the Chinese Pressurised Reactor (CPR1000) has decades of operational experience, which is widely acknowledged by the public and the industry. For the HPR1000, which evolved from the CPR1000, the PIE list for Design Basis Condition (DBC) analysis is based on the CPR1000 PIE list and is combined with the results of a review of the design features of the HPR1000 (FCG3) plant with the addition of the PIE occurring during low power operation and shutdown states and the PIE relevant to the spent fuel pool.

A categorisation system groups PIE into four categories according to their anticipated frequency of occurrence and potential radiological consequences to the public. These categories are used to define the success criteria that must be demonstrated to be met following events within that categorisation group. 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: Infrequent faults

Conditions that may occur once during the lifetime of a fleet of operating plants. These conditions 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: Limiting accidents

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 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 HPR1000.

12.3.3 Fault List

12.3.3.1 Safety Analysis Conditions

Events postulated in safety analysis are supposed to occur during normal plant operation. The initiating conditions assumed in safety analyses cover all the possible standard conditions from full power operation to cold shutdown. Following are definitions of the safety analysis domains based on FCG Unit3 practice.

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

In these shutdown states, all the necessary automatic reactor protection functions are available as in power state. In fact, 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.

State B: Intermediate shutdown with temperature above 140℃.

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

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

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

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

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

State E: cold shutdown during refuelling.

State F: cold shutdown when the pressure vessel is fully open. During this state, works are performed on RCP[RCS] components. This state does not have to be analysed with regard to core protection.

The list of DBC events based on FCG Unit3 practice is provided below.

12.3.3.2 DBC-1: Normal Operation

For HPR1000 (FCG3), the DBC-1 list is provided below, which will be the basis of the UK HPR1000 DBC-1 fault list to be assessed in the GDA PCSR.

a) All steady-state operation, start-up 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 power-raising processes;

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 start-up and shutdown);

2) Continuous changes of load 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).

12.3.3.3 DBC-2: Anticipated Operating Occurrences

For HPR1000 (FCG3), the DBC-2 list is provided in T-12.3-1 below, which will be the basis of the UK HPR1000 DBC-2 fault list to be assessed in the GDA PCSR.

T-12.3-1 DBC-2 sequences considered in the HPR1000 (FCG3)

No.	Sequence 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 offsite 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, B)
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)

No.	Sequence description
15	Spurious reactor shutdown (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)
19	Spurious pressuriser heater operation (State A)
20	Spurious pressuriser spray operation (State A)
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)

12.3.3.4 DBC-3: Infrequent Faults

For HPR1000 (FCG3), the DBC-3 list is provided in T-12.3-2 below, which will be the basis of the UK HPR1000 DBC-3 fault list to be assessed in the GDA PCSR.

T-12.3-2 DBC-3 sequences considered in the HPR1000 (FCG3)

No.	Sequence description
1	Inadvertent opening of a 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 loss of offsite power (> 2 hours) (State A)
5	Small feedwater system piping break including breaks in connecting lines to SG (State A, B)

No.	Sequence description
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 (during shutdown conditions) (State A)
9	Uncontrolled single RCCA withdrawal (State A)
10	SG tube rupture (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 (LOCA) (at power) including a break in the Emergency Boration System (RBS[EBS]) injection line (State A)
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	Long term loss of offsite power (>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)

12.3.3.5 DBC-4: Limiting Accidents

For HPR1000 (FCG3), the DBC-4 list is provided in T-12.3-3 below, which will be the basis of the UK HPR1000 DBC-4 fault list to be assessed in the GDA PCSR.

T-12.3-3 DBC-4 sequences considered in the HPR1000 (FCG3)

No.	Sequence description
1	Large steam system piping break (State A, B)
2	Inadvertent opening of a SG relief or safety valve (State B)
3	Large Feedwater system piping break (State A, B)
4	Long term loss of offsite power (in intermediate shutdown and cold shutdown conditions) (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 (during shutdown conditions) (State C, D, E)
8	Steam Generator tube rupture (two tubes in one SG) (State A)
9	Large break LOCA (LB-LOCA) (up to the surge line break, at power) (State A)
10	Intermediate LOCA (at power and in a shutdown condition) (State A, B)
11	Small break LOCA, including a break in the emergency boration system injection line (during shutdown conditions, and RIS[SIS] under RHR operation mode) (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 vessel drop (State A to F)

No.	Sequence description
16	Failure of radioactivity containing equipment in nuclear auxiliary building (State A to F)
17	Non isolable small break or isolable RIS[SIS] break (≤250 mm) in RHR mode affecting fuel pool cooling (during refuelling) (State E)

12.4 Fault Schedule Methodology

For the UK HPR1000, the PIE list will be identified using a combination of deterministic assessment with engineering judgements via a systematic review of all SSCs that support the normal operation of the plant. This will include all SSCs that may contain radioactive material or whose failure or mal-operation may result in a transient on the plant that could result in the release of radioactive material. The review will be based on the current good practices in PIE identification e.g. Failure Modes and Effects Analysis (FMEA) or other appropriate approaches combined with engineering judgements.

For the UK HPR1000, the list of events will be produced based on that of the HPR1000 and will include potential additional events identified by the systematic review of the design of the UK HPR1000 and the operating experience feedback of the similar units.

For each PIE identified the review will also identify the safeguards required to deliver the three fundamental safety functions of

a) Control of reactivity;

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

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

A fault schedule for the UK HPR1000 will be addressed in the GDA PCSR. A process for the fault schedule production will also be outlined in the GDA PCSR. According to Reference [1], it is likely that the fault schedule will contain the following information:

a) A list of all PIEs with their frequency identified;

b) The safety function, the associated safety system(s) and their safety classification for each fault;

c) For frequent faults, a second line of protection to satisfy the requirement of diversity protection;

d) Reference to the fault analysis that demonstrates that the UK HPR1000 plant is adequately protected for the events listed.

12.5 DBC Analysis Methodology and Assumptions

12.5.1 Analysis Rules

The analysis rules provide a conservative analysis methodology to validate the design of the safety systems provided to ensure the three fundamental safety functions are delivered following accidents and that sufficient safety systems are provided in the design. These rules are sufficiently conservative to demonstrate an appropriate design margin remains following the limiting faults.

This chapter describes the deterministic safety analysis performed for HPR1000 (FCG3) to illustrate the overall performance of the HPR1000 technology following accidents. The results of the analysis demonstrate that the current design provisions to protect against accidents are appropriate.

In parallel with the deterministic analysis， a comprehensive probabilistic safety analysis is undertaken. The Probabilistic Safety Assessment (PSA) analysis assesses the overall safety objectives of HPR1000. The PSA assessment is described in chapter 14.

In this sub-chapter, the rules defined for the assessment of HPR1000 (FCG3) are referred to as ‘DBC fault analysis rules’. The examples of the deterministic analysis presented in this chapter follow these rules.

Acceptance Criteria

The acceptance criteria for the assessment of the HPR1000 (FCG3) design are based on the applicable regulation in China. It is recognised that the legislative limits specified in China may not be sufficient to support the design of the UK HPR1000 with its more complex structure of consequence targets. Therefore a review of existing relevant UK and international best practice will be undertaken to produce UK HPR1000 specific quantitative safety targets for the development of the UK HPR1000 design from the HPR1000 design with the aim of completing this complex process prior to the start of detailed design work. These will then be utilised in the assessment of the design for the UK HPR1000 and provide part of the evidence to support the fundamental safety objectives reported in the GDA PCSR.

The detailed radiological release assessment of these events assumes data consistent with the success criteria for each category of event and demonstrates that the limits in the regulations are met. The safety criteria defined in terms of radiological limits are called Safety Criteria. For the DBA, the approach followed is to identify surrogate criteria that, if met, are consistent with the assumptions made in the identification of the safety systems required for each PIE and the radiological release assessment. These surrogate criteria, which are also called as Decoupled criteria, are either specified in Chinese legislation or have been agreed with the National Nuclear Safety Administration of China (NNSA). Therefore the decoupled criteria to be met in DBA analysis are:

a) No Departure from Nucleate Boiling (DNB) occurs in DBC-1 and DBC-2 events, or for DBC-3/4 events with a breach of the secondary circuit;

b) Less than 10% of the fuel rods experience DNB in DBC-3 events;

c) Less than 10% of the volume fraction of the fuel at the limiting point melts in DBC-3 or DBC-4 events (excluding LOCA);

d) Less than 10% of the fuel rods experience DNB in DBC-4 events (excluding LOCA), the core geometry is not affected and core cooling continues to remove decay heat;

e) Criteria for LOCA:

1) Peak clad temperature. The calculated maximum fuel element clad temperature shall not exceed 1204℃;

2) Maximum clad oxidation. The calculated total oxidation of the clad at the limiting point shall not exceed 17% of the total clad thickness before oxidation;

3) Maximum hydrogen generation. The calculated amount of hydrogen generated from the chemical reaction of the clad with water or steam shall not exceed 1% of the amount that would be generated if all of the clad material in the active core region, were to react;

4) Coolable geometry. Calculated changes in core geometry shall be such that the core remains capable of being cooled;

5) Long-term cooling. After any calculated successful initial operation of the Emergency Core Cooling System (ECCS), the calculated 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.

f) For transients without significant clad oxidation, the peak clad temperature shall not exceed 1482℃;

g) In safe shutdown conditions, the reactor core remains subcritical;

h) The maximum Linear Power Density at hot spot shall not exceed 590 W/cm for DBC-2.

In addition, specific criteria will be applied for events in cold shutdown conditions and faults related to the spent fuel pool.

To demonstrate that DNB does not occur, the ratio of the heat flux at which DNB would be predicted to occur in the current conditions and the current heat flux is calculated, known as the DNB Ratio (DNBR). A conservative value for this ratio has been established for the HPR1000 (FCG3) assessment, consistent with the methods used for its calculation. DNB is assumed to occur if the DNBR is below this limit value.

Codes approved by the NNSA have been used for transient simulation and minimum

DNBR calculations. Similar codes to those used at HPR1000 (FCG3) will be used for the UK HPR1000 analysis. The actual analysis codes to be used for transient simulation and DNBR calculation for the UK HPR1000 will depend on the fuel vendor chosen for the UK project. Details of the specific codes and methods used will be provided as part of the GDA PCSR. Therefore details of the codes and methods used for HPR1000 (FCG3) are not included in the Preliminary Safety Report (PSR).

Safe States

Analysis must be performed to show the plant is able to reach a safe state under DBCs. Two states are defined as below:

The controlled state: The state of the plant, where stabilisation of any transient has been achieved, the reactor is sub-critical, adequate heat removal is ensured and radioactive releases are limited.

The safe shutdown state: State reached after the controlled state is achieved, where the core is subcritical, residual heat removal is established on a long-term basis, and radioactive discharges remain acceptable.

For each DBC, it must be demonstrated that the controlled state can be reached. The analysis of the transition from the controlled state to the safe shutdown state may be performed once per set of similar DBCs.

Initial Conditions

Initial conditions for each DBC analysis are defined as a particular steady state. In general the most conservative steady state uncertainties are added to nominal values. For some specific faults, the uncertainties are directly accounted for in the analysis methodology.

Rules for Operator Actions

Operator actions can significantly affect the course of a DBC event. The first manual action assumed from the main control room is not considered until at least 30 minutes after the first significant signal accepted by an operator, and the first local manual action outside the main control room 1 hour after the first significant signal accepted by the operator.

Safety Classification of Mechanical, Electrical and I&C Systems

Safety classification for each mechanical, electrical and Instrumentation and Control (I&C) system are assigned as discussed in Chapter 4. The analysis rules decide which system of which classification will be considered in the safety analysis. For the UK HPR1000 this information will be included in the Fault Schedule to be presented in the GDA PCSR.

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

The consequences of a single failure are considered in the DBA. In the DBA, a single failure, independent with the postulated initial event, and which may affect all or part of the equipment required is considered. The single failure considered in the reference plant DBA can be an active single failure (in the short term) or a passive single failure (considered 24 hrs after the initiating fault). It is recognised that these rules may have to be adapted in the frame of UK HPR1000 GDA.

Loss of Off-Site Power (LOOP)

In the DBC fault analysis, LOOP is assumed to occur during full power operation conditions for DBC-3 and DBC-4 events if that assumption is conservative.

12.5.2 Main Assumptions

Assumptions related to plant parameters

Parameters covering the plant initial conditions are basic inputs to the fault analysis. Plant initial conditions decide the initial state of plant when the fault occurs. Relevant parameters considered in defining the plant initial conditions include:

a) Core power;

b) Pressuriser pressure;

c) RCP[RCS] average temperature;

d) Pressuriser level;

e) SG level;

f) SG water mass.

The initial values of parameters are chosen by adding or subtracting the maximum uncertainties to the nominal value of each parameter in the most conservative way for each DBC event.

Assumptions related to core parameters

In assessing the core response, conservative values for reactivity coefficients, power distributions, fission power and decay heat after reactor trip are assumed.

I&C signals

I&C signals are classified into different safety classifications as discussed in Chapter 8. Safety Category 1 Function (FC1) and Safety Category 2 Function (FC2) I&C signals are considered in the deterministic fault analysis. Other I&C signals (which are not FC1 or FC2) are also considered if the assumption on their operation is conservative.

Conservative assumptions are made on uncertainties associated with the I&C setpoints and time delays for signals to be generated are modelled in a conservative manner.

Safety systems functions and characteristics

In identifying the performance of the modelled safety systems the following is assumed:

a) System performance which leads to the worst results covering all times in component life;

b) Most onerous single failure which reduces the level of system performance;

c) Only FC1 and FC2 safety systems are considered in the fault analysis. Other safety systems which are not FC1 or FC2 classified are also considered if the assumption of their operation is conservative.

12.6 Design Basis Analysis

During the HPR1000 (FCG3) design process, the design basis faults listed in sub-chapter 12.3 have been studied for the DBC analysis. The analysis has followed the rules and assumption described in sub-chapter 12.5. The results of the HPR1000 (FCG3) DBC analysis confirm that the acceptance criteria discussed in sub-chapter 12.5 are met.

12.6.1 Examples of Assessment

For the PSR, the results of three representative events are provided below to demonstrate the overall approach followed in the analysis and to show the robustness of the HPR1000 design. The GDA PCSR will discuss the detailed analysis procedure adopted, identify the codes used and present results for the UK HPR1000.

The assessments of the three representative events for the HPR1000 (FCG3) plant have been submitted to the NNSA of China as part of the HPR1000 (FCG3) Preliminary Safety Analysis Report (PSAR). The analysis results for these events are appropriate to reveal characteristics of the HPR1000 (FCG3). The examples events are as followings:

a) Uncontrolled RCCA bank Withdrawal at Power (URWP);

b) SLB;

c) LB-LOCA from full-power conditions.

12.6.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

12.6.2.1 Description

An URWP results in the insertion of positive reactivity leading to an increase in the core heat flux. There is a net increase in the reactor coolant temperature and pressure as the heat transfer from the primary loops to the steam generators lags behind the core power generation.

In the event of a slow reactivity insertion transient, there is a small increase in nuclear power while the temperature in the core rises substantially, which can lead to reactor trip by reactor protection signal.

In the event of a very fast insertion transient, the nuclear power increases very rapidly in contrast to the coolant temperature, the transfer of the heat to the coolant being relatively slower. In that case, there would be a high neutron flux reactor trip.

The URWP event has been analysed as part of the HPR1000 (FCG3) PSAR. The results of a typical sequence are presented below.

12.6.2.2 Typical transient sequence

The following sequence represents the sequence of events most likely to occur during a URWP transient, including the actions of the safety systems.

The reactivity insertion causes an increase of the nuclear power and consequently the heat flux and coolant temperature. During this phase, a reactor trip could be initiated by a number of protection channels, e.g. power range high neutron flux protection. For any of the rates of withdrawal of the RCCA that can occur protection channels are available to provide protection for the core and maintain the core subcritical following shutdown.

Turbine trip is actuated following the reactor trip via a check-back signal and the residual heat is removed through the SGs by the main steam bypass or by the main steam relief train. The feedwater supply is provided by the Emergency Feedwater System (ASG[EFS]) if Main Feedwater Flow Control System (ARE[MFFCS]) is isolated during the transient. The RBS[EBS] system is used for boration during the cooldown to take the core to the long term safe state.

The normal or auxiliary pressuriser spray or the pressuriser safety valves are used to depressurise the RCP[RCS].

12.6.2.3 Acceptance criteria

For this accident the following acceptance criteria are defined:

The minimum DNBR during the transient must remain above the limit value discussed in sub-chapter 12.5.1, i.e. no DNB occurs during the transient.

12.6.2.4 Methods and assumptions

The transient simulation and minimum DNBR calculations have been performed using codes approved by the NNSA of China, as discussed in 12.5.1 above.

To be conservative, different power levels have been considered, bounding neutronic data have been used, and conservative uncertainty assumption have been applied to the protection system and mitigation actions.

The single failure is applied to one reactor protection signal channel (depending on which signal is actuated during the transient).

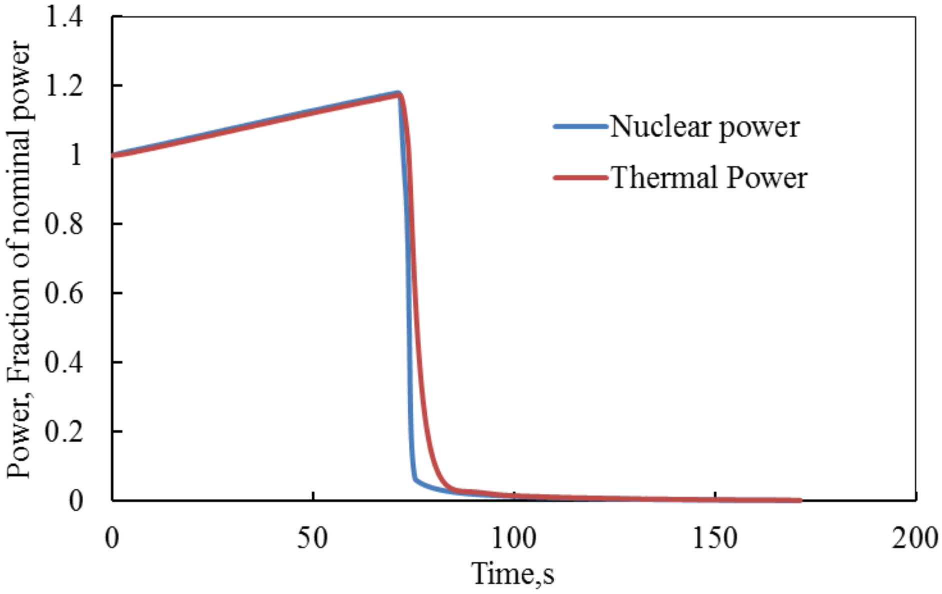
12.6.2.5 Results and conclusion

The results of the analysis show that for all possible RCCA withdrawal rates the minimum DNBR during the transient remains above the limit discussed in sub-chapter 12.5.1 and thus DNB does not occur during the transient. Therefore the acceptance criteria for this event are met.

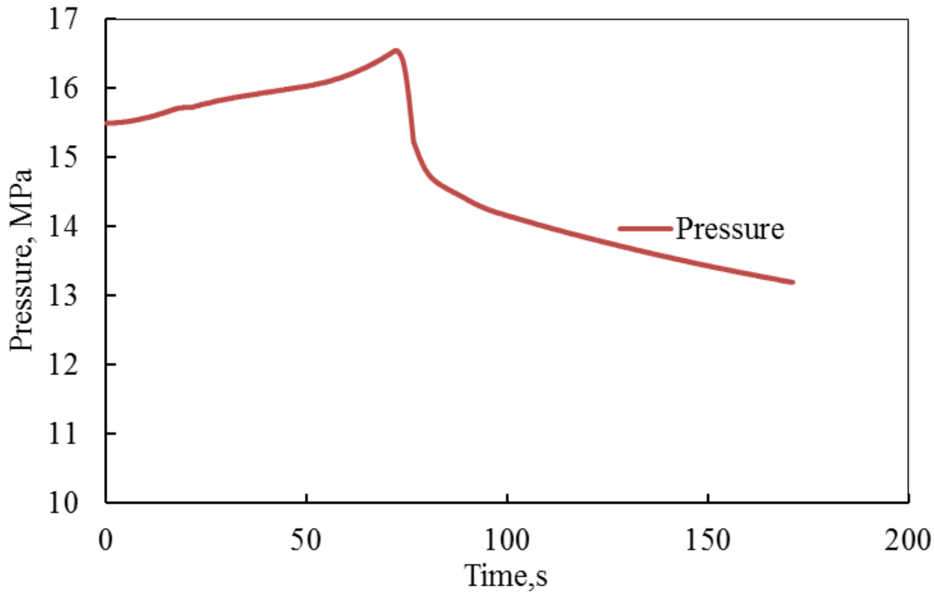
T-12.6-1 gives an example of the sequence of events following an URWP for the HPR1000 (FCG3). F-12.6-1 to F-12.6-3 present the transient conditions for the main parameters for the case resulting in the lowest value of minimum DNBR.

T-12.6-1 Time sequence of URWP (the most onerous case)

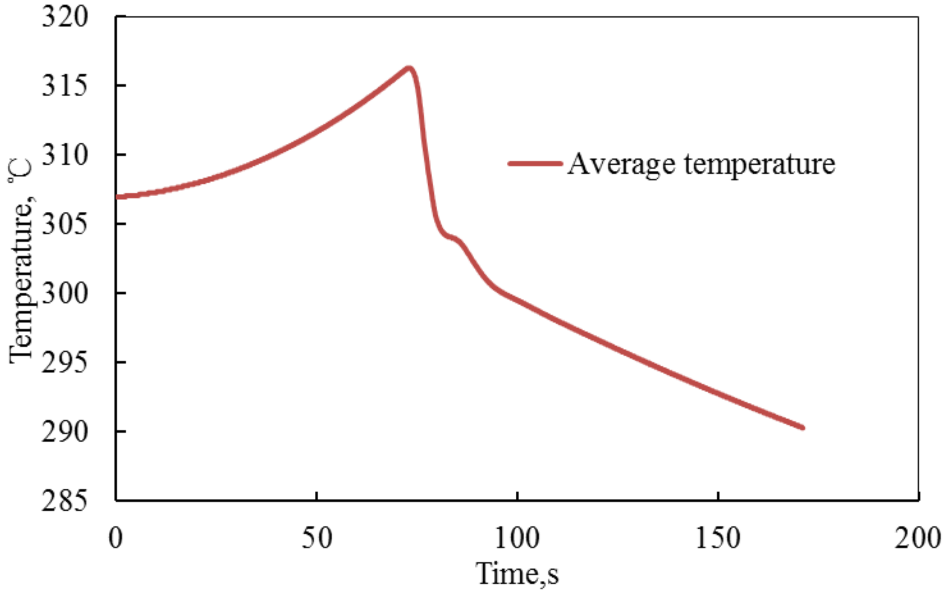
Event	Time（s）
Uncontrolled RCCA banks withdrawal	0.0
Reactor trip	69.0
RCCA banks start to insert	71.0
The minimum DNBR occurs	71.4



F-12.6-1 Thermal power and nuclear power-URWP



F-12.6-2 Pressurizer Pressure-URWP



F-12.6-3 Reactor Coolant Average Temperature -URWP

12.6.3 Steam Line Break (SLB)

12.6.3.1 Description

The main SLB break is defined as the rupture of a main steam line. The following description considers the event initiated from hot shutdown conditions as this minimises the energy stored in the core and results in the largest potential return to power during the transient.

The steam released from the break depressurises and cools the secondary side of the SG and leads to a rapid decrease in the temperature and pressure of the RCP[RCS]. The cooldown of the primary coolant results in an insertion of positive reactivity due to the positive moderator density coefficient. If the RCCA contributing the most negative reactivity is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical during the cooldown and return to power. In these circumstances it must be shown that DNB does not occur, as discussed in sub-chapter 12.5.1.

The following discussion of the event and its consequences are based on the analysis presented in the PSAR for HPR1000 (FCG3).

12.6.3.2 Typical transient sequence

This accident is analysed from hot shutdown initial conditions.

Following the break occurring, the pressure of the SG rapidly decreases. When the fall in the SG pressure reaches the ‘high SG pressure drop (MAX1)’ setpoint or the ‘low SG pressure’ setpoint is reached, the Main Steam Isolation Valves (MSIVs) are closed. If the break is located upstream of the MSIV it cannot be isolated by the MSIV on the affected SG. The affected SG will continue to blow down through the break to the containment.

The low load ARE[MFFCS] line will be isolated if the affected SG pressure reaches the ‘SG pressure low 2’ setpoint. If the ARE[MFFCS] low load lines are isolated, the ASG[EFS] will be actuated following a ‘SG level low2 (WR, Wide Range)’ signal.

With the steam flow continuing, the temperature and pressure in the RCS continues to fall and results in an insertion of positive reactivity. When the ‘pressure of pressuriser low 3’ setpoint is reached, the RIS[SIS] is actuated to protect the reactor.

The controlled state is reached with the affected SG empty of water with the decay heat being removed by the Main Steam Relief Trains (VDA[MSRT]) of the unaffected SGs.

12.6.3.3 Acceptance criteria

The limiting criterion for this study is demonstrating that DNB does not occur. This is demonstrated by showing that the minimum DNBR remains above the limit value specified as the criterion for the HPR1000 (FCG3) analysis condition II events, as discussed in sub-chapter 12.5.1 above.

12.6.3.4 Methods and assumptions

The transient simulation and minimum DNBR calculations have been performed using codes approved by the NNSA of China as discussed in sub-chapter 12.5.1 above.

The key assumptions made for the analysis are listed below:

a) A maximum heat transfer coefficient across the SG tubes is used in the study. In addition, no credit is taken for the heat transferred to the reactor coolant through reverse heat transfer from the two unaffected SGs to maximise the cooldown of the RCP[RCS];

b) Perfect moisture separation in the SG is assumed. This assumption is conservative as a substantial quantity of water is likely to be entrained in the steam flow, resulting in lower heat removal from the secondary system via the break and consequently in a less severe cooldown transient;

c) Minimum safety injection capability is assumed. The single failure of 1 Medium Head Safety Injection (MHSI) pump is assumed;

d) Core-related assumptions

1) The moderator temperature coefficient corresponding to End of Cycle (EOC) with all the RCCAs inserted, except for the RCCA having the highest worth stuck in its fully withdrawn position;

2) The Doppler defect versus power level considers two cases:

- case 1: a Doppler power defect of 1197pcm at 20% Full Power (FP), which covers all fuel cycles and stuck rod combinations resulting in a Doppler power defect between 1197pcm and 1724pcm at 20%FP;

- case 2: a Doppler power defect of 1724pcm at 20% FP, which covers all fuel cycles and stuck rod combinations resulting in a Doppler power defect higher than 1724pcm at 20%FP.

3) The Doppler temperature coefficient of -4.4 pcm/℃;

4) Shutdown margin is 3300pcm;

5) No credit has been taken for core residual heat to slow the RCP[RCS] cooldown;

e) The single failure of 1 Medium Head Safety Injection (MHSI) pump is assumed.

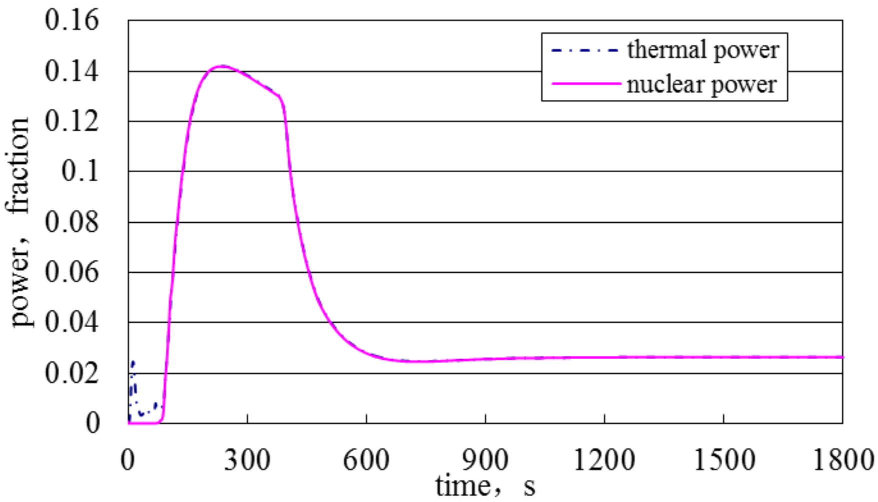
12.6.3.5 Results and conclusion

The analysis of the limiting case in the HPR1000 (FCG3) PSAR has shown that the calculated minimum DNBR is greater than the criterion limit defined as discussed in sub-chapter 12.5.1. Therefore DNB does not occur following a SLB at HPR1000 (FCG3). Therefore no fuel damage will occur following a SLB accident and all the criteria for this event are satisfied.

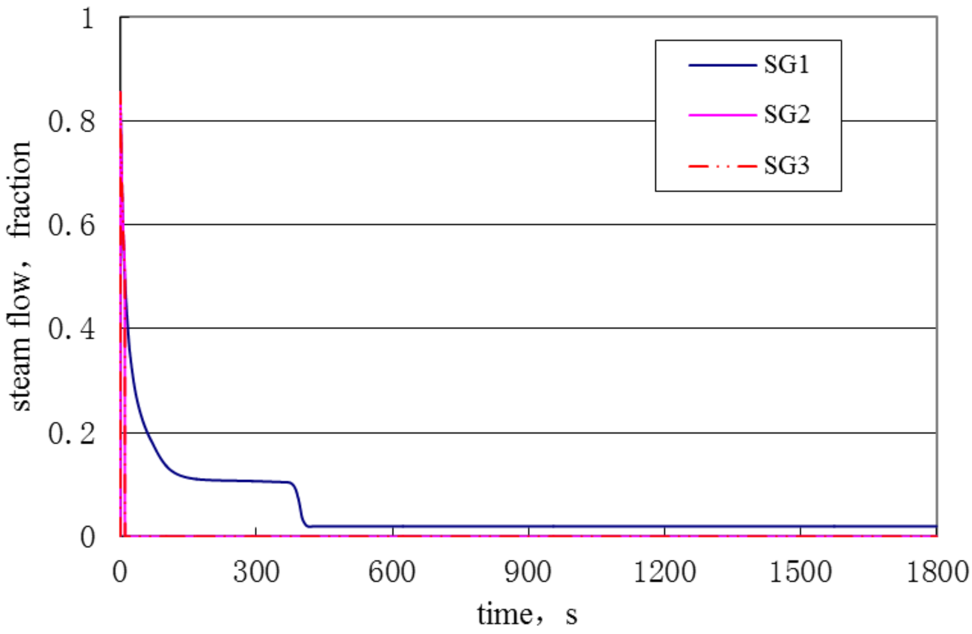
T-12.6-2 gives the sequence of events following a SLB accident from hot shutdown conditions for HPR1000 (FCG3). F-12.6-4 and F-12.6-5 present the transient conditions for nuclear power, thermal power and steam flow.

T-12.6-2 Time sequence of SLB for hot shutdown conditions

event	Time/s	
	case1	case2
Break occur	0	0
Steam line and ARE[MFFCS] isolation signal	5	5
PZR empty	14	14
SI signal	17.9	17.9
Thermal power peak	230.7	276



F-12.6-4 Thermal power and nuclear power



F-12.6-5 Steam flow

12.6.4 LB-LOCA at Full-power Conditions

12.6.4.1 Description

The LB-LOCA is defined as a large break of an RCP[RCS] pipe, or on any pipe connecting to RCP[RCS] before the first isolation valve forming part of the RCP[RCS] Boundary (RCPB). The event is analysed from full power conditions as this plant state has the largest stored energy in both the RCP[RCS] and the fuel.

Following the LB-LOCA RCP[RCS] water is rapidly discharged from the primary circuit to the containment via the break. This leads to a rapid decrease in RCP[RCS] pressure and the water level in the Pressuriser. This can result in a reduction of heat removal from the fuel with a consequent increase in fuel temperature.

The impact of LB-LOCA has been considered in the design of the HPR1000 (FCG3). The analysis of these events should show that the acceptance criteria discussed in sub-chapter 12.5.1 are satisfied.

12.6.4.2 Typical transient sequence

The condition most likely to happen during a LB-LOCA transient, including actions of the main safety systems are described below.

The break results in a rapid loss of RCP[RCS] inventory. The loss of fluid results in a decrease in the RCP[RCS] pressure and Pressuriser level. The rapid decrease of pressure causes a rapid decrease in moderator density in the core. The consequential decrease of core reactivity is sufficient to take the core subcritical causing a rapid reduction in core power. Subsequently, significant voidage of the fluid in the core combined with reactor trip, borated water injection from the Accumulator (ACC) and the RIS[SIS] pumps ensures the core remains shutdown.

A reactor trip is initiated following a low pressuriser pressure signal. The reactor trip signal automatically trips the turbine and closes the main feedwater high load lines. A 'Low pressuriser pressure signal 3' generates a safety injection signal, which automatically actuates the MHSI, Low Head Safety Injection (LHSI) and ASG[EFS] Systems. Borated water from the MHSI, ACC and LHSI flows into the core to cool the fuel clad. These stop any further increase in the temperature of fuel clad, place the reactor in the controlled state, and ensure sufficient heat removal from the fuel, the core remains subcritical and the water inventory in the core is steady or increasing.

The steam created in the reactor core does not contain a significant amount of boron. Therefore the concentration of boron in the RCP[RCS] will increase with time. To prevent blocking of coolant channels within the fuel by boron crystals, operator actions to maintain the boron concentration below the crystallization limit are necessary. Therefore, in the later stages of the event, the LHSI injection point must be switched from just the cold leg to a combination of cold leg and hot leg to control the boron concentration in the core.

Long term cooling of reactor core is provided by at least one train of RIS[SIS] in Safety Injection mode with water flows into both the cold and hot legs from the LHSI.

12.6.4.3 Acceptance criteria

The LOCA analyses should meet the following acceptance criteria:

a) The calculated maximum fuel element clad temperature shall not exceed 1204℃;

b) Maximum clad oxidation. The calculated total oxidation of the clad at the limiting point shall not exceed 17% of the total clad thickness before oxidation;

c) Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the clad with water or steam shall not exceed 1% of the amount that would be generated if all of the clad material in the active core, were to react;

d) Coolable geometry. Calculated changes in core geometry shall be such that the core remains capable of being cooled;

e) Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated 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.

12.6.4.4 Methods and assumptions in HPR1000 (FCG3)

The analysis of the LB-LOCA for HPR1000 (FCG3) has utilised the deterministic realistic method. The method introduces penalties into a thermal hydraulic analysis code to ensure that uncertainties on the parameters considered in the analysis are bounded.

Sensitivity studies were performed to ensure that the worst case was identified.

Conservative assumptions were adopted as discussed in subchapter 12.5 and confirmed via sensitivity analysis. These assumptions can be divided into 4 types:

a) Initial conditions associated with the steady state;

b) Modelling of the accident scenario;

c) Safety system performance;

d) Core parameters.

12.6.4.4.1 Initial conditions associated with the steady state

The Initial conditions for the steady state were selected to be conservative. The main assumptions are discussed below:

a) The initial core power was that for full-power operation increased by the maximum measurement error for steady state operation;

b) The temperature of the primary coolant was the nominal temperature for full-power operation, reduced by the maximum expected temperature fluctuation and measurement error. This combination was shown to be the most conservative by a sensitivity analysis;

c) The Pressuriser pressure was the nominal value for full power operation increased by the maximum operational pressure fluctuation and measurement error. This results in the maximum delay to the generation of the reactor trip signal and safety injection signal;

d) The primary coolant flow assumed was the thermal design flow, the minimum calculated reactor flow for design purposes;

e) Maximum core bypass flow to minimise the amount of flow through the fuelled region available for core cooling;

f) 10% of the SG tubes were assumed to be plugged for calculating the initial water inventory of the primary circuit. The calculation of reverse heat transfer between primary and secondary circuit assumes no plugging as this maximises the amount of energy transferred to the primary water from the secondary side of the SGs;

g) Minimum containment pressure and temperature and maximum humidity, as this delays core reflooding.

12.6.4.4.2 Accident condition

The following key assumptions have been made to define the accident scenario modelled in the HPR1000 (FCG3) analysis.

a) The break was assumed to be on the cold leg between the pump and the inlet to the reactor vessel;

b) Maximum delays were assumed in the generation of the safety injection signal and any flow from the RIS[SIS] or the ACC to the broken leg is assumed to be lost;

c) LOOP is assumed to occur at the same time as the break. This has the maximum impact on the performance of the safety systems.

12.6.4.4.3 Safety system

The performance of the safety systems can have a significant impact on the course of the transient. To ensure that the results of the transient are appropriately conservative the performance of the individual safety system has been pessimised as discussed below.

12.6.4.4.3.1 Safety injection

a) A Single failure of the Emergency Diesel Generator (EDG) in the safeguard train connected to an intact loop has been assumed. This leads to the loss of a train of the RIS[SIS] (including a MHSI pump and a LHSI pump) and a train of the ASG[EFS];

b) Minimum SI pump performance corresponding to the end of component life with a maximum resistance in the SI injection pipe;

c) Maximum temperature of the water in the In-containment Refuelling Water Storage Tank (IRWST).

12.5.4.4.3.2 Accumulator

Sensitivity studies were undertaken to identify conservative assumptions for the flowrate and water temperature from the accumulator.

12.6.4.4.4 Core parameters

Maximum values for the ratio of power in the peak channel to the average channel ($F_{\Delta H}$), the power at the hottest point in the core relative to the average core point power, (F_Q), the decay heat and the worst axial power shape were assumed.

12.6.4.5 Results and conclusion in HPR1000 (FCG3)

Sensitivity studies were undertaken using the above assumptions to identify the worst case. T-12.6-3 gives an example of the sequence of events following a LB-LOCA for HPR1000 (FCG3). F-12.6-6 to F-12.6-9 present the transient conditions for the main parameters for the case resulting in the highest value of Peak Clad Temperature (PCT). For this worst case, the PCT calculated in the HPR1000 (FCG3) analysis was 992.2℃ (see T-12.6-4).

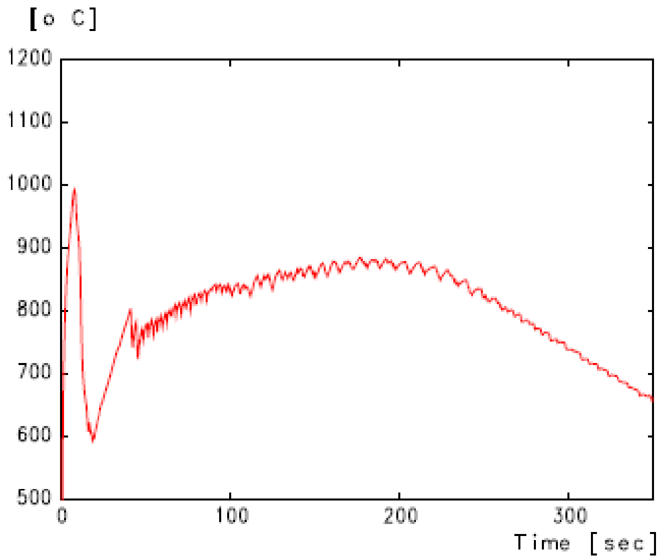
The PCT limit and the remaining acceptance criteria identified in sub-chapter 12.6.4.4 above were shown to be met for HPR1000 (FCG3).

T-12.6-3 Time sequence of LB LOCA

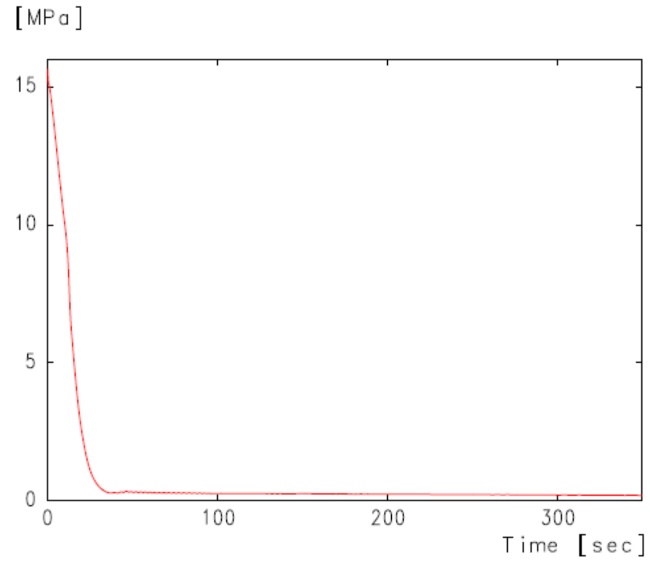
Event	Time(s)
RT signal	4.65
SI signal	8.00
Event	Time(s)
Start of accumulator injection	14.21
Start of SI injection	38.01
Start of core reflooding	38.58
End of accumulator injection	43.51
Start of ASG[EFWS]	63.01

T-12.6- 4 PCT following LB LOCA

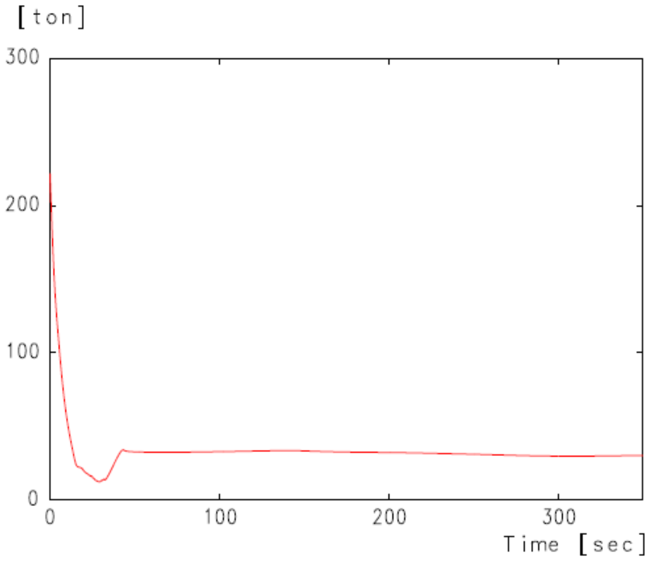
PCT	temperature(℃)	992.2
	time(s)	7.5



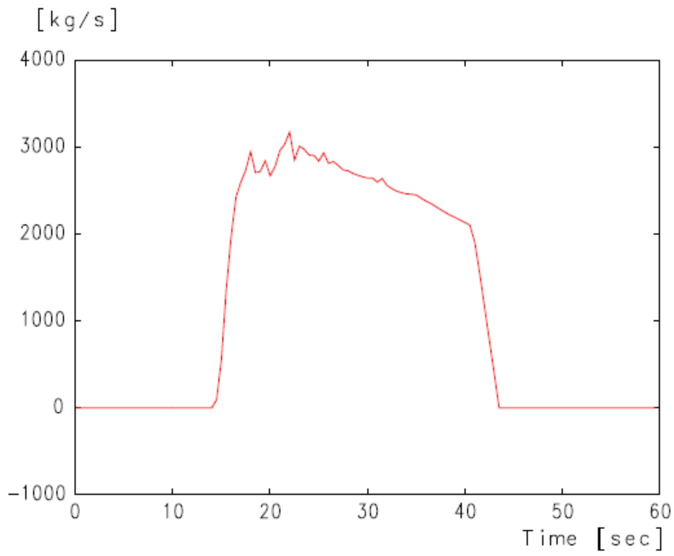
F-12.6-6 Peak clad temperature



F-12.6-7 Primary pressure



F-12.6-8 Primary water inventory



F-12.6-9 ACC flowrate

12.7 Radiological Consequences of Design Basis Events

Detailed analyses of the radiological consequences of DBC events at HPR1000 (FCG3) are presented in the HPR1000 (FCG3) PSAR. The analysis demonstrates that the effective dose received by any person arising from a design basis accident meets the requirements of NNSA in China. As the detailed assumptions made in the assessment for HPR1000 are Fangchenggang site specific, they are not valid for the GDA assessment. Therefore, further information on the dose assessment for the HPR1000 is not included in the PSR. Since the requirements in term of the Numerical Targets are different in China and in the UK, the PCSR of the UK HPR1000 will provide details of the specific approach to be followed for the assessment of the radiological consequences of DBC events for UK HPR1000.

12.8 References

[1] ONR, Safety Assessment Principles for Nuclear Facilities, Revision 0, November 2014

[2] ONR, Transient Analysis for DBAs in Nuclear Reactors，NS-TAST-GD-034, Revision 3, July 2016