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UK HPR1000 GDA Project

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Preliminary Safety Report
Chapter 13
Design Extension Conditions and Severe Accident Analysis

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13.1 List of Abbreviations and Acronyms

ALARP	As Low As Reasonably Practicable
ASG	Emergency Feedwater System [EFWS]
ASP	Secondary Passive Heat Removal System [SPHRS]
ATWS	Anticipated Transient Without Scram
CDF	Core Damage Frequency
DBC	Design Basis Condition
DEC	Design Extension Condition
DEL	Safety Chilled Water System [SCWS]
ECS	Extra Cooling System [ECS]
EHR	Containment Heat Removal System [CHRS]
EUF	Containment Filtration and Exhaust System [CFES]
EUH	Containment Combustible Gas Control System [CCGCS]
GCT	Turbine Bypass System [TBS]
GDA	Generic Design Assessment
HPR1000 (FCG3)	Hua-long Pressurized Reactor under Construction at Fangchenggang nuclear power plant unit 3
IRWST	In-containment Refuelling Water Storage Tank
IVR	In-Vessel Retention
KDS	Diversity Actuation System [DAS]
LB-LOCA	Large Break (Loss of Coolant Accident)
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MCD	Medium Pressure Rapid Cooldown
MCR	Main Control Room
MHSI	Medium Head Safety Injection
NNSA	National Nuclear Safety Administration

PCSR	Pre-Construction Safety Report
PSA	Probabilistic Safety Assessment
PSV	Pressurizer Safety Valve
PWR	Pressurized Water Reactor
RBS	Emergency Boration System [EBS]
RCCA	Rod Cluster Control Assembly
RCV	Chemical and Volume Control System [CVCS]
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SA	Severe Accident
SADV	Severe Accident Dedicated Valve
SAPs	Safety Assessment Principles
SBO	Station Black Out
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
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]).

13.2 Introduction

The analysis of Design Basis Condition (DBC) events is described in Chapter 12. It is possible, although unlikely, that the safeguards systems claimed in the DBC analysis fail to deliver the required safeguards, or the event is more severe than those considered in the DBC analysis. These circumstances are known as Design Extension Conditions (DEC). These are low frequency sequences where the conditions may be more onerous than those identified in DBC analysis and are assessed to identify the margins present in the design.

Postulated DEC events that occur are not considered in the same way as the DBC, but are considered in the design process for the facility utilising a best estimate methodology. In these circumstances, the releases of radioactive material are still kept within acceptable limits.

Design extension conditions are classified into two classes, which are assessed using different methodologies due to the different phenomena encountered as discussed below:

DEC-A: Complex sequences which involve failures beyond those considered in the deterministic design basis but do not involve fuel melt.

DEC-B: Severe accidents considered in the design, both to prevent early and delayed containment failure and to minimise releases for the remaining conditions with core melting.

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: Design Extension Conditions that have the potential to lead to a severe accident have been systematically analysed, and the analysis is used to identify further appropriate preventative or mitigating measures.

13.3 DEC-A analysis methodology

13.3.1 Identification of DEC-A Events

13.3.1.1 Introduction

The DEC-A events include complex accidents with multiple failures which are not covered by the analysis of DBC events, but must be considered to meet the probabilistic safety targets and to assess the radiological consequences. Consequently the probability of multiple failures and system unavailability will be evaluated, in a systematic manner, to determine the accident sequences which contribute significantly to the overall probabilistic safety target.

13.3.1.2 Methodology of DEC-A Identification

The DEC-A events are identified by engineering judgement, deterministic analysis and probabilistic analysis. Any additional safety features identified will be designed to enhance the capability of the plant to withstand these more severe accident situations.

The approach adopted for Hua-long Pressurized Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)) is to utilise the results of the Probabilistic Safety Assessment (PSA), combined with engineering judgment (generally following existing safety standards and guidelines, or operational experience) to establish the HPR1000 (FCG3) DEC-A events list.

For the purpose of consolidating the DEC-A list with the PSA, each DEC-A sequence Core Damage Frequency (CDF) is checked against the DEC-A probabilistic criteria. The process of DEC-A event identification is mainly performed as follows:

a) Selecting DEC-A events with the PSA;

1) Building a PSA model after the safety systems designed for DBC events have been included in the agreed design;

2) Screening sequences with high frequency for CDF;

3) Grouping the sequences into different events with the same general plant response or mitigation measure.

b) Selecting DEC-A events from the existing safety standards and guidelines, or operational experience if not identified via step a).

13.3.2 DEC-A Events List

For HPR1000 (FCG3) the above approach has been followed to produce the DEC-A list provided in T-13.3-1 below, which will be the basis of the UK HPR1000 DEC-A fault list to be assessed in the GDA Pre-Construction Safety Report (PCSR).

13.3.3 Identification of DEC-A Features

DEC-A features are measures designed to mitigate the identified DEC-A events and reduce the CDF.

The identified DEC-A events with similar functional characteristics can be mitigated by the same DEC-A features. The features available to mitigate the effects of DEC-A events in the HPR1000 (FCG3) design are listed below:

a) Secondary Passive Heat Removal System (ASP [SPHRS]);

b) Extra Cooling System (ECS [ECS]);

c) Containment Heat Removal System (EHR [CHRS]);

d) Station Black Out (SBO) diesel generator;

T-13.3-1 DEC-A sequences considered in the HPR1000 (FCG3)

No.	Sequences Description
1	Loss of secondary cool down function accident; (State A)
2	Small break Loss of Coolant Accident (LOCA) with failure of Medium Pressure Rapid Cooldown (MCD); (State A)
3	Small break LOCA with total loss of Low Head Safety Injection (LHSI) (during power operation); (State A)
4	Small break LOCA with total loss of LHSI (during shutdown); (State C, D)
5	Loss of Residual Heat Removal (RHR) or failure of recovery of RHR after Loss Of Offsite Power (LOOP) accident (during shutdown); (State C,D)
6	Station black out (during power operation and during shutdown state); (State A to F)
7	Station black out to the cool down of spent fuel pool; (State A to F)
8	Anticipated transient without scram (ATWS) due to signal failure; (State A)
9	ATWS due to failure of Rod Cluster Control Assembly (RCCA) to insert; (State A)
10	Small break LOCA or Steam Generator Tube Rupture (SGTR) with total loss of Medium Head Safety Injection (MHSI) (during power operation); (State A)
11	Reactor coolant sealing leakage caused by total loss of cooling chain (in power state); (State A)
12	Total loss of cooling chain (during shutdown); (State D)
13	Rupture of multiple Steam Generator (SG) tubes (SGTR); (State A)
14	Rupture of Main Steam Lines combined with rupture of tubes in affected SGs; (State A)
15	SGTR combined with atmospheric relief valve opening failure in the SG affected (during power operation). (State A)

e) Diverse Actuation System (KDS [DAS]);

f) Emergency Boration System (RBS [EBS]);

g) Safety Chilled Water System (DEL [SCWS]);

h) Manual feed and bleed operation;

i) Manual low pressure full-speed cooldown operation.

During the design process for the UK HPR1000, the list of features available in the HPR1000 (FCG3) design will be reviewed and any additional provisions that could be implemented in the UK HPR1000 design to reduce the radiological consequences as low as reasonably practicable (ALARP) will be assessed for inclusion in the plant design.

13.3.4 DEC-A Analysis Rules

13.3.4.1 Main Assumptions

Deterministic safety analysis for DEC-A events is based on realistic assumptions.

a) Initial Condition

The DEC-A analysis initial conditions are the nominal best-estimate values for the steady-state operating condition of the plant state under consideration. For example, the initial thermal-hydraulic parameters of the reactor coolant system are nominal values, without uncertainty.

b) Boundary Conditions

The DEC-A analysis is undertaken on the basis of realistic assumptions. Therefore all safety and non-safety systems can be claimed in the analysis unless they are assumed to fail as a direct consequence of the DEC-A event sequence.

c) Manual Actions

In the DEC-A analysis, manual actions initiated from the Main Control Room (MCR) are considered 30 minutes after the first important information is provided to the operator while local manual actions are considered 60 minutes after the first important information is provided to the operator.

13.3.4.2 DEC-A Analysis Final State

DEC-A analysis is performed until the safe state called “DEC-A final state” is reached. The DEC-A final state is defined as follows:

a) The core is in the long-term sub-critical state;

b) Decay heat is being continuously removed;

c) Activity discharged meets the acceptance criteria;

d) All plant parameters are significantly below the design limits for components and structures and the associated margins are not decreasing.

13.3.4.3 Acceptance Criteria

In DEC-A analysis, it must be demonstrated that the criteria defined for the analysis of DBC-4 accidents are met. If these criteria are met, the radiological consequences of DEC-A accidents are no more severe than those of DBC-4 accidents.

13.3.5 Objectives on DEC-A Analysis

The assessments of DEC-A analysis of the HPR1000 (FCG3) plant have been submitted to the National Nuclear Safety Administration (NNSA) of China. The analysis results for these events are appropriate to reveal characteristics of HPR1000 (FCG3) technology as implemented in the HPR1000 (FCG3) design. As an example of the DEC-A events listed in 13.3.2, the results of the small break LOCA with failure of MCD (during power operation) are presented.

13.3.5.1 Description of DEC-A Events

MCD failure may result from the loss of the Emergency Feedwater System (ASG [EFWS]) or the loss of the Atmospheric Steam Dump System (VDA [ASDS]) and the Turbine Bypass System (GCT [TBS]) due to a common cause failure. Consequently, the secondary cooldown function fails.

The small break ($20cm^2$) in the primary loop results in a reduction of the coolant inventory which is beyond the capability of the Chemical and Volume Control System (RCV [CVCS]) to restore. The continuous coolant loss leads to the decrease of primary side pressure and pressuriser level.

The reactor trips following a “low pressuriser pressure (<Min 2) signal” and the turbine is tripped following the reactor trip signal.

As the primary side pressure continues to decrease, a safety injection signal is generated following a “low pressuriser pressure (<Min 3)” signal.

The MCD failure means that the secondary pressure can only be controlled by the main steam safety valve. The setpoint for these valves cannot be reduced by the operator to initiate a plant cooldown and hence decay heat cannot be removed through the SGs whilst maintaining sufficient inventory in the primary circuit. The break flow is itself not sufficient to remove the decay heat. Therefore the primary side pressure increases and stabilises above the MHSI maximum delivery pressure preventing delivery of injection to maintain the primary side inventory.

In this DEC-A event, manual feed and bleed operation, using manual opening of the pressuriser safety valves (PSV) to reduce primary pressure to below the MHSI delivery pressure, is used to prevent core degradation.

13.3.5.2 Main Assumptions

a) Initial Conditions

The unit is operating at full power steady state conditions. The thermal-hydraulic parameters for the reactor coolant system are at their nominal values, without uncertainty.

b) Boundary Conditions

1) Reactor trip occurs following a "low pressuriser pressure (<Min 2)" signal;
2) Turbine trips following the reactor trip signal;
3) Safety injection signal occurs following a "low pressuriser pressure (<Min 3)" signal;
4) Main steam safety valve opening set point is at the "high SG pressure (>Max 2)";
5) The operator manually opens the pressuriser safety valves 30 minutes after the reactor trip signal.

13.3.5.3 Results

The analysis for HPR1000 (FCG3) of the small break LOCA with MCD failure presented in the HPR1000 (FCG3) deterministic safety analysis is presented to demonstrate the response of the HPR1000 (FCG3) plant to a DEC-A event. The results of the analysis of HPR1000 (FCG3) show that core is not damaged and the fuel integrity criteria for DBC-4 events are not exceeded.

After the manual feed and bleed operation, the medium and low head injection pumps start to inject borated water into the reactor coolant system. This is sufficient to provide, long term core sub-criticality. Once PSV opening and the MHSI delivery to the primary circuit occur, a rapid cooldown and further depressurisation of the primary circuit result. Subsequently, decay heat can be removed by the LHSI heat exchangers. Typical results for this sequence are shown in F-13.3-1 to F-13.3-3.

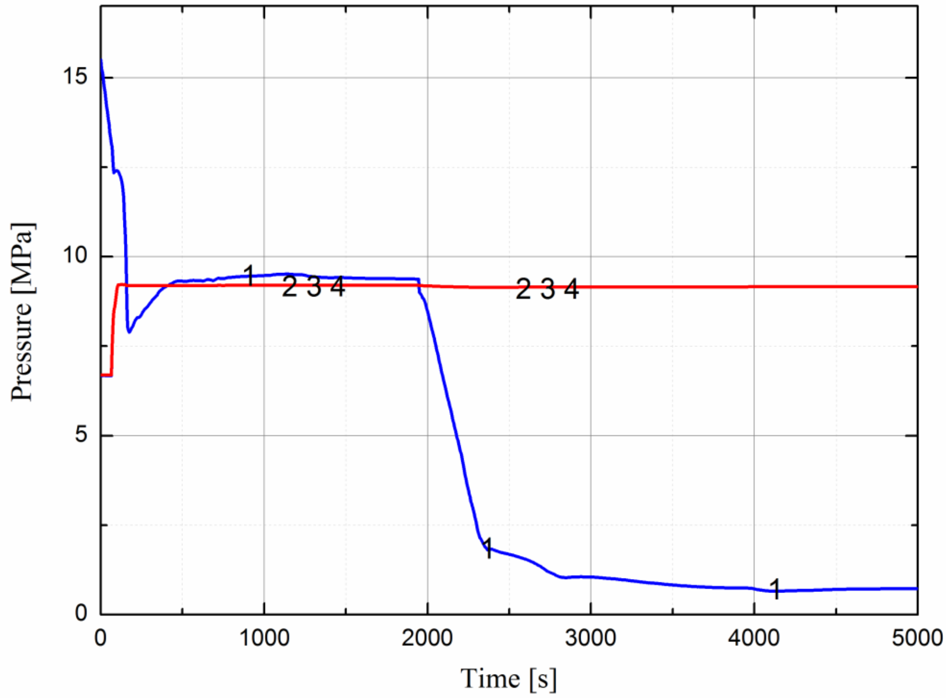
13.3.5.4 Conclusions

The conditions following a DEC-A event for the HPR1000 (FCG3) are such that the fuel integrity criteria for DBC-4 events are not exceeded. Thus the radiological consequences of a DEC-A event for HPR1000 (FCG3) are within the criteria set for DBC-4 events and are therefore acceptable.

13.4 DEC-B Analysis Methodology

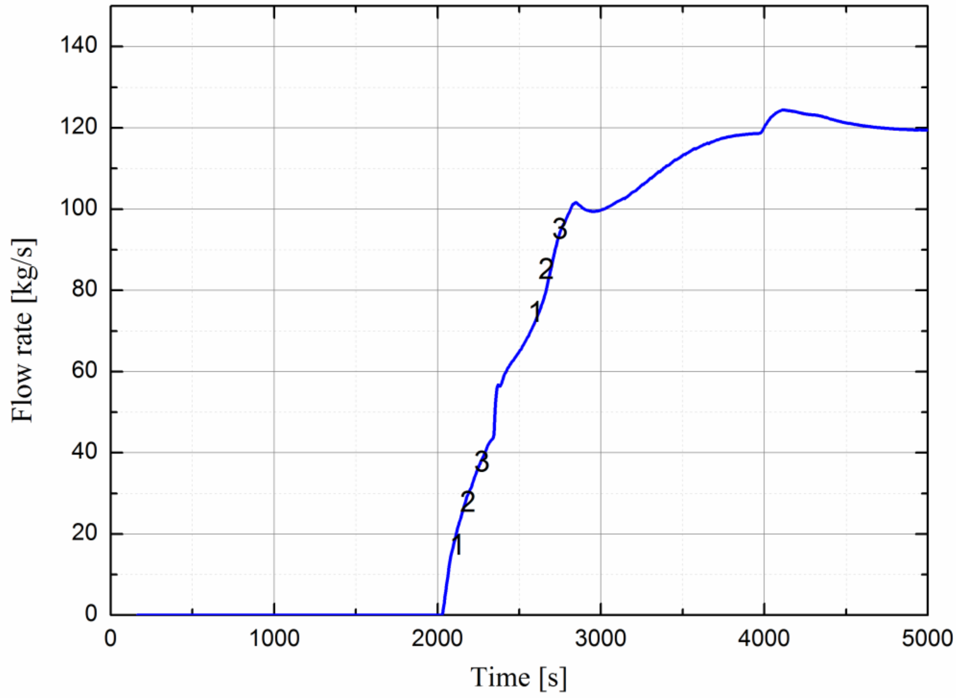
13.4.1 Introduction

If the continuous loss of coolant or secondary heat sink leads to core uncovering and consequently melting of the core, the resultant event is termed a Severe Accident (SA) or DEC-B event.



1: Pressuriser pressure 2: SG1 pressure 3: SG2 pressure 4: SG3 pressure

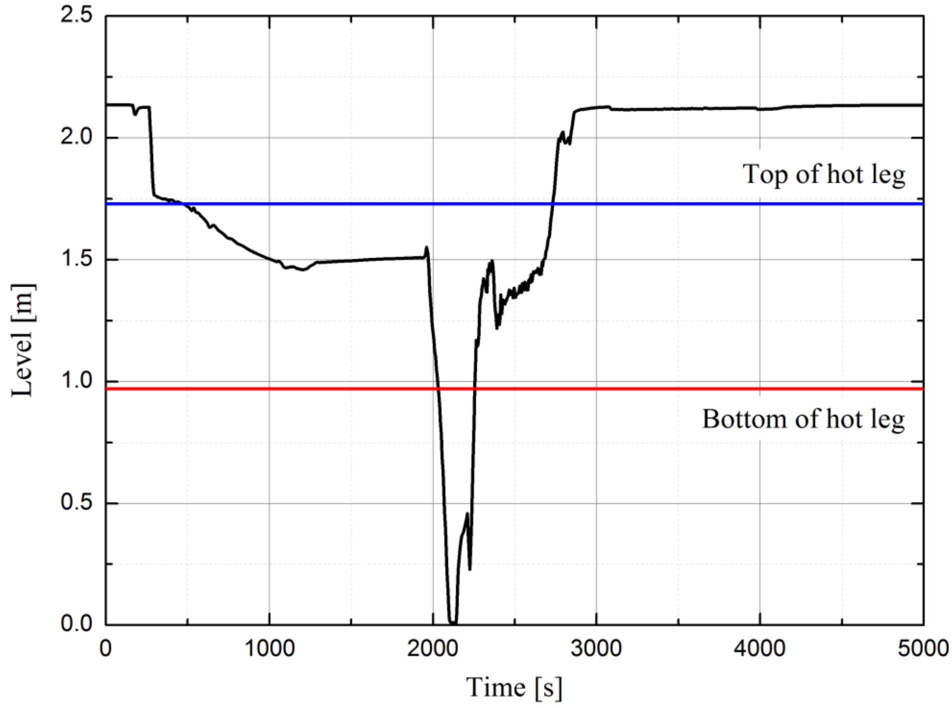
F-13.3-1 Primary side pressure and secondary side pressure



1: MHSI and LHSI train 1 flow rate; 2: MHSI and LHSI train 2 flow rate;

3: MHSI and LHSI train 3 flow rate

F-13.3-2 MHSI and LHSI flow rate



0 m corresponds to the top of the active core

F-13.3-3 Reactor Pressure Vessel (RPV) upper plenum water level

The overall objective in DEC-B events is to practically eliminate core melting sequences which could lead to large early release and prevent radioactive release (including containment bypass and core melting sequences) which exceeds safety objectives.

The safety objectives of radioactive release shall meet the following requirements that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation beyond the immediate vicinity of the plant, limited sheltering, and no long-term restrictions on food consumption) and that sufficient time is available to implement these measures.

To achieve the safety objectives above, the basic strategy is to maintain the integrity of the containment in both short and long term as far as possible.

However, the integrity of the containment can be challenged by various phenomena and threats occurring during a DEC-B event. To provide cooling to the melted core material (“corium”) and maintain the integrity of the containment, severe accident mitigation measures have been developed for HPR1000 (FCG3) and will form the basis of the provisions to be provided for severe accident mitigation in the UK HPR1000 design.

13.4.2 SA Phenomena and Threats

Although many different initiating events may lead to severe accidents in pressurized water reactor (PWR) nuclear power plants, the core melting processes and behaviours of molten corium and radioactive substances are almost the same, with only different sequence progression time-histories.

Primary system breaks or safety valves discharge due to insufficient core cooling lead to continuous loss of primary coolant. If it fails to supplement the coolant inventory, the core will be uncovered. The fuel temperature rises continuously due to the residual heat and core uncovering. As the temperature increases, control rods, fuel claddings and support structures in RPV begin to melt first. In the process of claddings oxidation by steam, a large amount of hydrogen is produced and released into containment through breaks or valves. With the accumulation of hydrogen in containment atmosphere, hydrogen combustion or explosion may occur.

The oxidation of claddings also generates chemical heat and the fuel temperature further rises. The fuel begins to melt and collapse. Molten corium falls into the lower plenum after the lower grid and lower support plate fail. After that, the residual water in lower head dries out and RPV soon fails because the molten pool keeps heating lower head. If primary pressure is high when RPV fails, the corium may be ejected into the reactor pit or the containment atmosphere, then direct containment heating may occur, which can seriously threaten the integrity of containment.

When molten corium relocates into the lower head, it falls into the residual water filled in the lower head. Molten fuel may rapidly fragment and transfer its energy to the coolant resulting in steam generation, shock waves and possible mechanical damage. This phenomenon is called in-vessel steam explosion. Similar phenomena may occur when RPV fails and the corium falls into the cavity, interacting with the water in cavity. Steam explosion happened outside RPV is called ex-vessel steam explosion.

After the corium is ejected or falls into the reactor pit, it interacts with the concrete basement. The basement corrosion begins and the basement can be melted through in several days, and radioactive substances can be released to the environment underground. The corrosion releases substantial non-condensable and combustible gas. As the combustible gas accumulates, rising concentration increases the risk of combustion or explosion which can seriously threaten the integrity of containment. On the other hand, the continuous accumulation of non-condensable gas can finally cause containment overpressure failure.

During the whole severe accident progression, decay heat is accumulated in containment with more and more steam and non-condensable gas. Containment overpressure may occur in the long term.

Based on the description above, the following phenomena can threaten the integrity of containment and shall be practically eliminated to maintain the integrity of the containment in both short and long term as far as possible:

a) Containment overpressure;

b) Fuel-coolant interaction (steam explosion);

c) Direct containment heating / high-pressure melt ejection;

d) Hydrogen combustion and explosion;

e) Elevated temperature (equipment survivability);

f) Molten core-concrete interaction.

13.4.3 SA Mitigation Measures

To avoid large release of radioactive substances in DEC-B conditions, the mitigation measures are designed to prevent the early failure of the containment and ensure the long-term integrity of the containment by practically eliminating SA phenomena listed in 13.4.2. The mitigation measures include:

a) Severe Accident Dedicated Valve (SADV) to prevent containment over-heating and overpressure caused by high-pressure melt ejection;

b) Containment Combustible Gas Control System (EUH [CCGCS]) to reduce hydrogen concentrations and reduce the risk from combustion and explosion;

c) Containment Heat Removal System (EHR [CHRS]) to prevent containment overpressure failure;

d) In-Vessel Retention (IVR) measure (It is the sub-system of EHR [CHRS] that realises in-vessel retention, called reactor pit injection and passive reactor pit injection sub-system in chapter 7.8) using external reactor pressure vessel cooling to maintain the corium within the second barrier and consequently preserve the integrity of the third barrier, the containment, against ex-vessel steam explosion and corium attack;

e) Containment Filtration and Exhaust System (EUF [CFES]) to prevent containment overpressure and limit the release of airborne radioactivity at the site boundary within acceptable levels. After EUF [CFES] activation, containment under-pressure may occur. Appropriate operation strategies of EUF [CFES] are considered in severe accident management to avoid containment under-pressure.

Detailed descriptions of these measures can be found in chapter 6 and chapter 7.

Post-Fukushima improvements are thoroughly considered in HPR1000 (FCG3) design, which will be described in PCSR.

Severe accident management strategies to instruct the activation of SA mitigation measures will be described in PCSR.

UK HPR1000 has considered the emergency preparedness which will be discussed in details in PCSR.

13.4.4 Selection of DEC-B Events

Both probabilistic and deterministic approaches, along with engineering practice experience, are used to select DEC-B events to analyse.

DEC-B events analysed to assess against numerical targets in Safety Assessment Principles (SAPs) will be described in PCSR stage in PSA chapters.

13.4.5 Objectives on DEC-B Analysis

An integral systems severe accident analysis code for assessing off-normal transients that can progress to and include severe accidents is used in DEC-B analysis in HPR1000 (FCG3).

It integrates physical processes that could occur during severe accidents into a single plant simulation. These physical processes include steam formation, core heat-up, cladding oxidation and hydrogen evolution, vessel failure, core debris-concrete interactions, ignition of combustible gases, fluid (water and core debris) entrainment by high velocity gases, and fission product release, transport, and deposition. Models are included for engineered safeguard system logic and performance. Operator actions are also simulated by specification of intervention conditions and responses.

Behaviours of fission products are considered in the calculation of DEC-B source term. For different purposes of source term analysis, the integral systems severe accident analysis code and methodology introduced by NUREG-1465, Reference [1] are used respectively.

The DEC-B sequences are analysed based on realistic and best estimate assumptions to demonstrate that the mitigation measures provided for HPR1000 (FCG3) are effective.

As an example, the analysis addressing the following phenomenon chosen from the list above is given:

-Containment Overpressure

Representative accident sequences will be analysed to demonstrate that containment failure can be prevented with the following mitigation measure available from the list above:

-EHR [CHRS]

The assessments of EHR [CHRS] in HPR1000 (FCG3) have been submitted to the NNSA of China. The analysis results for these events are appropriate to reveal the characteristics of the HPR1000 (FCG3) technology as implemented in the HPR1000 (FCG3) design during a DEC-B event. As an example of the response to the DEC-B phenomena discussed in 13.4.2, the results of a large break LOCA (LB-LOCA) scenario are presented in this section.

13.4.5.1 DEC-B Event Description

A LB-LOCA without safety injection which leads to the fastest pressurisation of the containment could threaten the integrity of the containment in the short term. The primary coolant is discharged into the containment at the start of the accident. The core melts rapidly if no safety injection is available to reflood the core and re-establish decay heat removal. The corium is maintained within the RPV by external reactor vessel cooling initiated by flooding the reactor pit using the passive IVR. The passive IVR delivers water from the tank located at a position higher than the reactor pit, to the reactor pit which then boils to remove the necessary heat from the RPV. After the tank is depleted, the active IVR will continue to inject water into the reactor pit. This maintains the corium inside the RPV. The IVR system is discussed in detail in sub-chapter 7.8. The containment pressure increases with the steam release through the rupture and the steam generated by decay heat and the non-condensable gas generated by the interaction between water and hot fuel cladding. Containment spray (sub-system of EHR [CHRS] described in sub-chapter 7.8) is put into operation by the operator which provides sufficient heat removal to result in a continuous decrease of containment pressure.

13.4.5.2 Main Assumptions

The initial conditions and the assumptions of the LB-LOCA scenario presented are shown below:

a) The unit is initially running at full power;

b) The event commences with a double ended guillotine break in one Cold Leg;

c) No safety injection operation occurs;

d) IVR is available;

e) EHR [CHRS] is available.

13.4.5.3 Results

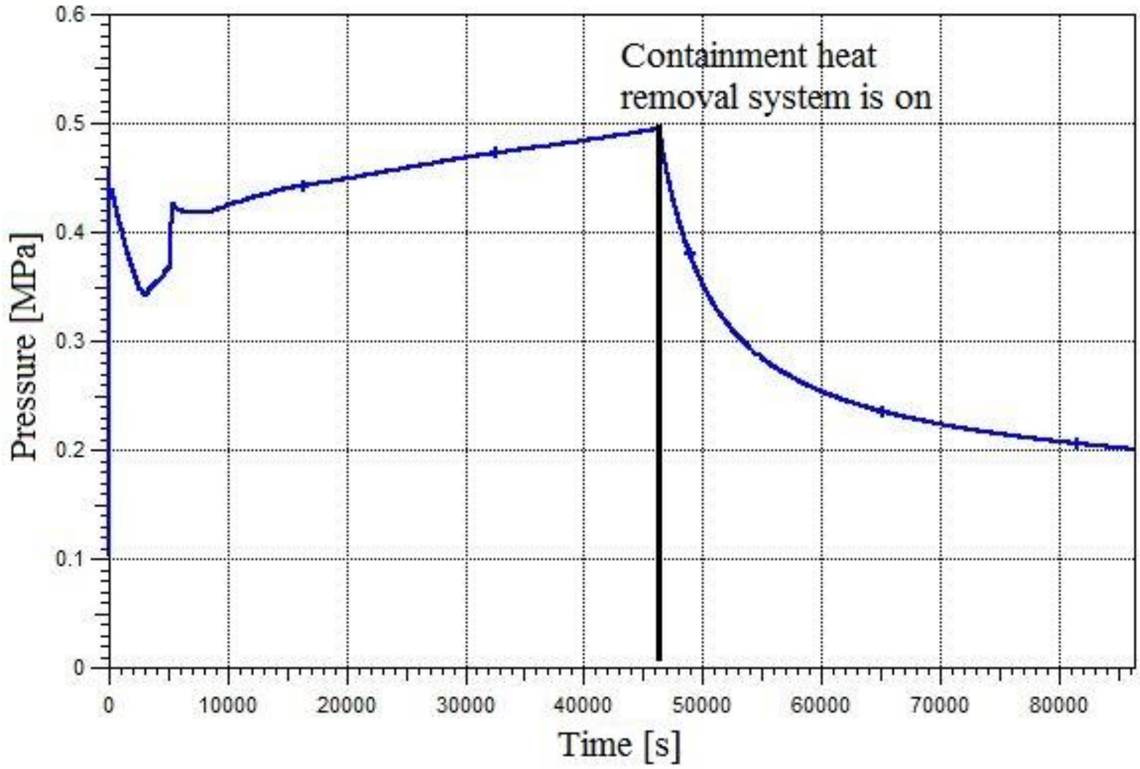
The assessments of EHR [CHRS] in HPR1000 (FCG3) demonstrate that the corium is cooled by the external RPV cooling from the IVR fluid delivered to the reactor pit. The decay heat is removed from the containment by EHR [CHRS]. The containment pressure is maintained below the design pressure by the operator initiation of the systems. Typical results are shown in F-13.4-1 and F-13.4-2.

13.4.5.4 Conclusions

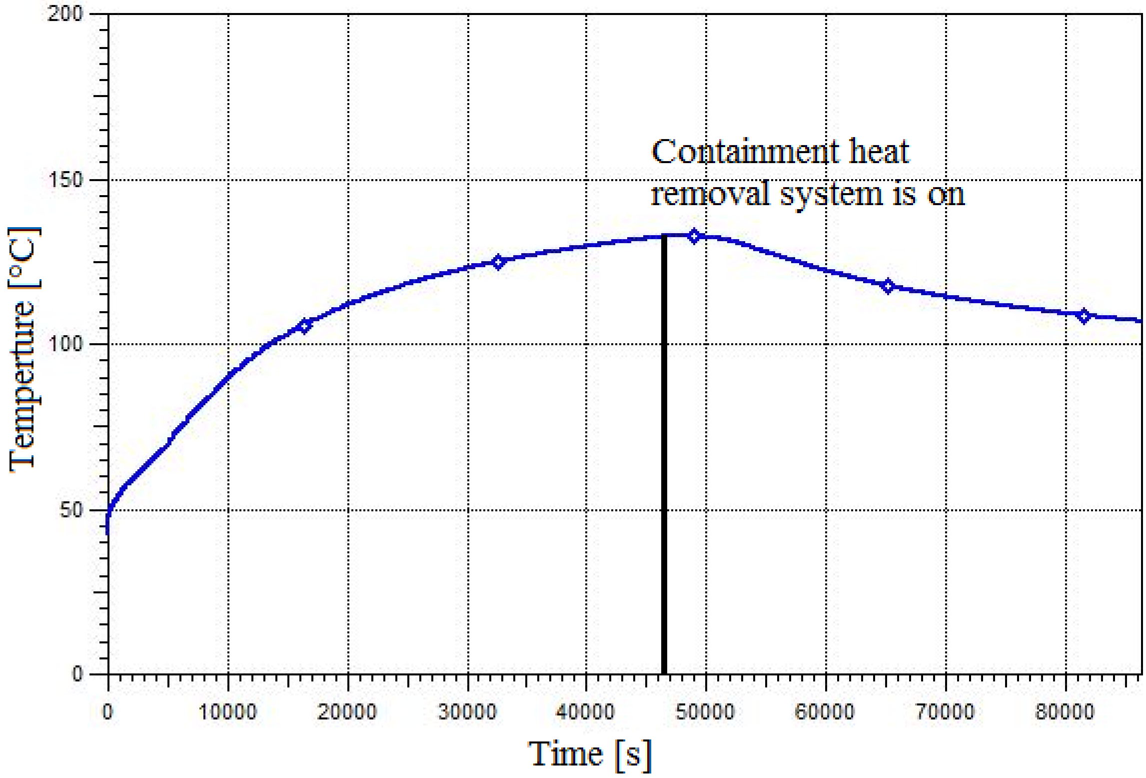
The analysis has shown that the containment pressure remains below the design pressure and therefore both early containment failure and a significant release of radioactive material are avoided. Therefore the objectives outlined in chapter 22 are met.

13.5 References

[1] NUREG, Accident Source Terms for Light Water Nuclear Power Plants, NUREG-1465, Final Report, February 1995.



F-13.4-1 Containment pressure



F-13.4-2 In-containment Refuelling Water Storage Tank (IRWST) surface water temperature