UK EPR	Title: PCSR – Sub-chapter 16.6 – Analysis of Extreme Beyond Design Basis Events Carried Out in Response to Fukushima **UKEPR-0002-168 Issue 00**

Total number of pages: 28	Page No.: I / III
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REVISION HISTORY

Issue	Description	Date
00	First issue	22-11-2012

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SUB-CHAPTER 16.6 – ANALYSIS OF EXTREME BEYOND DESIGN BASIS EVENTS CARRIED OUT IN RESPONSE TO FUKUSHIMA

0. INTRODUCTION AND SAFETY REQUIREMENTS

This sub-chapter outlines the analysis of the robustness of the UK EPR design against extreme events, carried out in response to the Fukushima earthquake and tsunami in March 2011. Specific design modifications are described herein, which are being implemented to enhance the robustness of UK EPR systems, structures and components against such extreme beyond design basis events.

0.1. DESCRIPTION OF FUKUSHIMA EVENT AND IMPACT ON UK EPR DESIGN

The Fukushima event, which occurred at the Fukushima Daiichi site on Japan's east coast on 11th March 2011, involved a magnitude 9 earthquake followed by impact of a large tsunami wave. The site contained six BWR units, three of which were operating and three of which were shut down and defuelled at the time of the event.

Inundation of the plant site by the tsunami wave resulted in loss of electrical supplies and cooling systems to four of the units, which caused severe core damage in the three operating reactors (units 1 to 3) resulting in a large radioactivity release to the environment. The event was classified at Level 7 on the International Nuclear and Radiological Event Scale (INES) scale.

In response to the Fukushima event, the European Nuclear Safety Regulators Group (ENSREG) [Ref-1], and Western European Nuclear Regulators' Association (WENRA) [Ref-2] requested that European NPPs that were in operation, under construction, or in decommissioning, were subjected to 'stress tests' in which sequential loss of the lines of defence was modelled deterministically, irrespective of the probability of the loss. The analysis was to confirm the validity of the design basis for certain extreme events like earthquakes and flooding, that could potentially lead to multiple losses of safety functions, requiring severe accident management. In the UK, the HM Chief Inspector of Nuclear Installations published a review report on the Fukushima event, containing recommendations for government, regulators and the nuclear industry to consider and adopt [Ref-3].

In response to stress test specifications, and the request of international regulators, EDF and AREVA have carried out a comprehensive analysis of the response of the UK EPR to extreme events. The aim has been to identify margins, analyse robustness of the plant to beyond design basis events, and identify (and if possible remove) cliff-edge effects to further improve robustness.

The analysis included:

- a robustness analysis of the reference design with respect to seismic, external flooding and other external hazards that are beyond the current design basis;

- a robustness analysis of plant behaviour in sequential loss of AC power sources and/or systems providing residual heat removal however caused, covering both the reactor building and the spent fuel pool;
- identification of additional measures that could be applied in the design and construction of the UK EPR to further mitigate the risk due to such beyond design basis events.

This PCSR sub-chapter summarises the results of the post-Fukushima analysis of the UK EPR design and identifies plant modifications that are being implemented as a result.

1. ROBUSTNESS ANALYSIS OF PROTECTION AGAINST SEVERE EXTERNAL EVENTS

1.1. SEISMIC EVENTS

The review of resilience of the UK EPR design against beyond design basis earthquakes carried out following the Fukushima event is described in [Ref-1]. The aim of the analysis was to identify the margin available in the design, and reasonably practicable improvements that might be required.

The resilience review [Ref-1] noted that the UK EPR is designed to resist an enveloping seismic event which is bounding for Nuclear Power Plants constructed in Western Europe, with a Design Basis Earthquake (DBE) corresponding to a 0.25 g peak ground acceleration in the horizontal direction (see Sub-chapter 13.1 of the PCSR). Structures, materials and systems which are required for controlling and mitigating accidents are designed so that they are able to fulfil their functions, maintain their integrity or remain stable under the conditions caused by the seismic motion.

The resilience review [Ref-1] further noted that the Seismic Margin Assessment (SMA) presented in Sub-chapter 15.6 of the PCSR had assessed the robustness of a plant against beyond design basis earthquakes. The SMA consists of defining sets of Structures, Systems and Components (SSCs) which form success paths, to provide a conservative measure of the plant's overall seismic capacity. The SMA shows that the UK EPR could tolerate a seismic event with 0.61g peak ground acceleration without significant risk of a severe accident and release of radioactivity from the plant: this seismic level is comparable to the seismic level experienced in the Fukushima event. The probability of occurrence of an earthquake of this magnitude in the UK is extremely small (of order one in one million years).

Modifications have been introduced during GDA to reduce the risk of leakages through penetrations in the walls and floors of connected compartments. These modifications will benefit the seismic robustness of the Spent Fuel Pool by reducing the risk of leakage in beyond design basis seismic events.

Considering the moderate seismicity of the UK EPR sites, and the existence of large safety margins beyond the design basis level, no further requirements have been identified to enhance the seismic design of the UK EPR to avoid 'cliff-edge effects'.

1.2. EXTERNAL FLOODING EVENTS

The robustness of the UK EPR design against external flooding has been reviewed in the light of experience from the Fukushima event [Ref-1]. The fundamental safety requirements related to protection against external flooding are stated as:

- to keep the buildings housing safety classified equipment dry, by setting the platform at a level at least equal to the Maximum Design Flood Level;
- to prevent as far as possible any water present on the platforms from entering buildings housing safety classified equipment.

The analysis aimed at identifying possible cliff-edge effects that could occur if the flood level exceeded the site platform level [Ref-1]. Three types of cliff edge effects potentially induced by a flood were identified:

- flood situations causing loss of heat sink, initiated by a rise in water level leading to the loss of filtration of main cooling water, and possible inundation of main and safety related cooling pumps;
- flood situations causing a loss of off-site power (LOOP), initiated by the presence of a significant depth of water on the platform of the main station and unit transformers;
- flood situations causing a total loss of external and internal electrical sources initiated by the presence of a significant depth of water on the platform of the Nuclear Island.

The Nuclear Island buildings that house safety classified equipment have thick reinforced concrete walls, with a limited number of access doors that are able to withstand significant water pressures with limited leakage rates. The buildings are judged to be able to withstand significant heights of flood water on the platform without sustaining water inflow that could impair the function of the equipment they house.

As a result of the resilience analysis, the following design modifications will be implemented to enhance the robustness of the UK EPR against beyond design basis external flooding events:

- leaktightness of security access doors to NI buildings housing safety functions to be checked and improvements implemented as necessary to prevent water entering the buildings;
- reinforced leaktightness of rooms containing the Ultimate Diesel Generators (UDGs) and severe accident batteries to be implemented;
- reinforced leaktightness of the pumping station main slab (openings and doors) to be implemented to help protect the Essential Service Water System (SEC [ESWS]) and Ultimate Cooling Water System (SRU [UCWS]) safety functions against beyond design flooding event of the pumping station.

Further details are given in section 4 below.

2. ANALYSIS OF ROBUSTNESS AGAINST LOSS OF AC POWER/LOSS OF ULTIMATE HEAT SINK

An analysis has been performed of the robustness of the UK EPR design against extreme beyond design basis events causing long term loss of AC power sources and/or total loss of long term heat sink [Ref-1]. The analysis followed the stress test approach of ENSREG described previously: hence, the loss of essential functions was modelled deterministically, without taking into account its probability of occurrence.

The robustness analysis calculated the grace periods that would be available before there was a significant increase in off-site radiological consequences following an irrecoverable deterioration in the plant state. Possible counter-measures to extend grace periods and avoid cliff edge effects were identified.

Results of the analysis are summarised below.

2.1. TOTAL LOSS OF ULTIMATE HEAT SINK (TOTAL LUHS)

2.1.1. Event sequence with Steam Generators available (States A to C)

If there is long term loss of both the main heat sink SEC [ESWS], and the back-up heat sink SRU [UCWS], in a plant state with the Steam Generators (SGs) available for reactor cooling, post accident management of the UK EPR is based primarily on the use of the steam generators for heat removal. In case of failure of the reactor coolant pump seals, a change in the cooling strategy would be required to one which managed the loss of primary coolant inventory. In analysing bounding scenarios for the total LUHS event, it was conservatively assumed that the loss of heat sink event would be combined with station blackout (SBO) i.e. loss of off-site power (LOOP) combined with loss of all four Emergency Diesel Generators (EDGs). The scenario analysis is presented below [Ref-1].

2.1.1.1. Event sequence in State A-C – no leakage of reactor coolant pump seals

For total LUHS in State A (assumed to be combined with SBO), automatic reactor trip is initiated: the control rods are automatically inserted into the nuclear core and the primary pumps are automatically stopped as the thermal barrier cooling and/or the seal injection are lost. The Stand Still Seal System (SSSS) is automatically activated and is assumed to successfully prevent reactor coolant pump seal leakage.

Core decay heat is removed via the Steam Generators by natural circulation. Feedwater is supplied to the Steam Generators by ASG [EFWS] pumps of Divisions 1 and 4 which are electrically supplied by the UDGs, with water and steam discharged to atmosphere by the atmospheric steam dump valves (VDA [MSRT]).

Sub-chapter 16.6 – Figure 1 shows the timeline for the event, which is the same as in the Station Black-Out (SBO) case. As long as electric power is available from the UDGs and feed water stocks are available to supply the SGs, there is no loss of primary circuit water inventory and core temperatures would be stable. To extend the grace period available before main heat sink recovery is required it would be necessary to refuel the UDGs after 24 hours, and to refill the ASG [EFWS] tanks after 48 hours using the JAC system or other on-site water sources.

To increase grace periods, design modifications are being implemented to:

- allow fuel to be transferred to the UDGs from the day-tanks of the unused EDGs (see section 4.1.3);
- enable ASG [EFWS] tanks to be resupplied with feedwater from on-site raw water storage sources (see section 4.1.7).

In State C, if the Reactor Coolant System (RCP [RCS]) is closed or if it can be re-pressurised, the loss of the Residual Heat Removal Mode (RRA [RHRS]) due to the initiating event causes a primary circuit temperature and pressure increase. The operating approach consists of allowing the temperature and pressure in the RCP [RCS] to increase again in order to reinstate cooling by the steam generators.

2.1.1.2. Event sequence in State A-C – failure of reactor coolant pump seals

The case of total LUHS combined with SBO with failure of the reactor coolant pump seals in State A has been analysed [Ref-1]. The results are described below.

Leakage from the reactor coolant pump seal shaft system is postulated after 24 hours, which is the duration of pressure and temperature qualification profiles for the pump seals. The mass flow rate through the leaking seals would be reduced due to the reduction of the primary side temperature and pressure resulting from the manual secondary side cooldown. Significant degradation of the primary circuit water inventory would not be expected until about five days after the initiating event.

Following seal leakage, the primary circuit water inventory would begin to decrease and after several days, conditions would be reached in the primary circuit corresponding to the criteria for primary side feed and bleed operation. The operator would then be required to open a Primary Depressurisation System (PDS) line and to start one of the Low Head Safety Injection (LHSI) pumps belonging to divisions 1 and 4, which are cooled by the diverse trains of the Safety Chilled Water System (DEL [SCWS]), which are air cooled. Due to the primary coolant depressurisation, the accumulators would inject at this time into the primary circuit.

This phase of LHSI water injection without any water cooling would end when the In-Containment Refuelling Water Storage Tank (IRWST) temperature reaches 120°C. Above this temperature, the LHSI pumps are assumed not to be operable, if the heat sink is not recovered. The injection into the RCP [RCS] would thus be stopped.

Once the LHSI pumps were no longer operable, core heat up would occur, leading to entry into Operating Strategy for Severe Accident (OSSA) conditions. Core meltdown would then occur, resulting in Reactor Pressure Vessel (RPV) failure and corium transfer into the dedicated spreading area in the containment building. If the ultimate heat sink was not restored a gradual build up of pressure in the containment would then occur. The pressure would reach 9 bara, at which it is conservatively assumed that containment leaktightness may be lost, at around nine days. The time line is shown in Sub-chapter 16.6 - Figure 2.

2.1.2. Event sequence with Steam Generators unavailable (State D)

In total LUHS in State D the SGs would be unavailable for reactor cooling. In the initial reactor state, cooling is by LHSI trains operating in RRA [RHRS] mode. The bounding case is considered to be State D with the water level at the ¾ loop level, as this corresponds to a minimum reactor coolant inventory and a high decay power. Following LUHS, the LHSI pumps would be tripped automatically on high Component Cooling Water System (RRI [CCWS]) temperature. The operating strategy consists of restarting an LHSI train (belonging to division 1 or 4) to inject water into the RCP [RCS], to compensate for the water lost due to steaming, taking suction from the IRWST. The LHSI pumps of division 1 and 4 are cooled by the air-cooled trains of the DEL [SCWS] and electrically supplied by the UDGs.

The LHSI injection phase is assumed to end when the IRWST temperature reaches 120°C, above which the LHSI pumps are assumed to be no longer operable. After stopping of the LHSI pumps, core heat up would occur after a few hours, leading to entry into Operating Strategy for Severe Accident (OSSA) conditions. Core meltdown would then occur, resulting in RPV failure and the corium transfer into the dedicated spreading area in the containment building. If the ultimate heat sink was not restored a gradual build up of pressure in the containment would then occur. The pressure would reach 9 bara, at which it is conservatively assumed that containment leaktightness may be lost, at around four days. The timeline is shown in Sub-chapter 16.6 - Figure 3.

2.1.3. Cooling of the Spent Fuel Pool

In the event of total LUHS combined with SBO, the Spent Fuel Pool (SFP) cooling system (PTR [FPCS]) would be unavailable due to the loss of heat sink.

In the robustness analysis [Ref-1], this case is bounded by the case of SBO + loss of two Ultimate Diesel Generators in state F when all fuel assemblies are unloaded from the core into the SFP. Due to the significant heat released by the fuel assemblies in state F, total loss of fuel pool cooling would lead to a relatively rapid heat up and vaporisation of the water stored in the fuel pool. Without any cooling or water make-up, the water level would decrease to the top of the fuel storage racks in approximately one day, based on conservative assumptions.

In order to extend the grace period, the following plant modifications are being implemented in UK EPR:

- provision of an ultimate make-up system for the SFP, which would use additional connections and a mobile pump to provide make-up to the spent fuel pool from on-site water reserves (see section 4.1.6);

- provision of a seismically qualified passive or automatic venting device between the Fuel Building outlet vent and the Nuclear Auxiliary Building stack to allow steam to be vented and thus prevent risk of pressure build-up in the Fuel Building. This venting device will supplement the outlet vent in the current UK EPR design, which includes an isolation damper. It will avoid the need for operator action inside the Fuel Building to de-isolate the existing vent (see section 4.1.8).

- provision of a means of ensuring that a fuel assembly being handled could be safely deposited in the fuel storage rack without off-site power or availability of EDGs and UDGs. This modification is being implemented because, if an extreme initiating event occurred during fuel handling operations, the grace period before fuel uncovering could be reduced due to the reduced water coverage above a fuel assembly that was being lifted above its set down position (see section 4.1.13).

- study of the risk of hydrogen formation in the Fuel Pool Hall by radiolysis in the SFP, in the absence of ventilation: implementation of equipment necessary to prevent formation of explosive hydrogen concentrations in the Fuel Pool Hall (see section 4.1.10);

- implementation of essential instrumentation required for fuel pool monitoring in the Severe Accident I&C system (backed by severe accident batteries), to ensure its continued availability in long term loss of AC power conditions. The essential instrumentation will include pool water temperature, pool water level and pool area dose rate measurements (see section 4.1.11).

2.2. TOTAL LOSS OF AC POWER SOURCES[1]

The robustness analysis was performed following the ENSREG approach, by analysing sequences in which the power supply functions supporting the essential systems ensuring heat removal, reactivity control and containment are progressively defeated due to unspecified causes [Ref-1].

Results for the bounding case of loss of all AC power sources (“Long term LOOP + Loss of all four Emergency Diesels + Loss of two Ultimate Diesel Generators”) are described below.

2.2.1. Systems supplied by batteries

Total loss of AC power sources involves loss of off-site power and loss of the EDG and UDGs. In this situation, the 2-hour batteries and the severe accident batteries are the only electrical power sources available in the Nuclear Island.

The four sets of 2-hour batteries, installed in the Safeguard Buildings, are each connected to a different electrical train. They have a 2-hour autonomy (at the end of life), ensuring their ability to provide a defined current at a minimum voltage for at least two hours. The 2-hour batteries supply the following functions:

- the containment isolation function: the 2-hour batteries can supply the containment inner isolation valves;

- the Steam Generator discharge function using the Main Steam Relief Train (VDA [MSRT]);

- the operability of the key operational and safeguard systems from the Main Control Room, in particular the Operational I&C (PICS) and safety-related I&C (SICS). The 2-hour batteries can supply all the I&C installed in the Safeguard Buildings;

[1] Loss of AC power supply denotes loss of 400 kV external grid and houseload connections, plus loss of the four emergency diesel generators (10 kV) and the two ultimate diesel generators (690 V). It does not imply loss of AC supplies derived from the DC batteries.

- Main Control Room (MCR) lighting. However, the MCR ventilation is not supplied with power. The MCR is normally cooled by the Control Room Air Conditioning System (DCL [CRACS]) system, which is cooled by the DEL [SCWS] system, which is not available due to loss of AC power sources. The MCR temperature would thus progressively increase until the I&C failed due to depletion of the 2-hour batteries. Ambient conditions are currently being assessed to confirm their acceptability (maximum acceptable temperature for human intervention). In the event that temperature limits are exceeded, design modifications will be implemented e.g. measures for de-loading of unneeded consumers to reduce local heat sources, or implementation of battery backed devices to redistribute heat to other rooms.

The two severe accident battery sets supply electrical trains 1 and 4. They have a 12-hour autonomy (at the end of life) ensuring their ability to provide a defined current for a minimum voltage for at least twelve hours. These batteries supply the following functions:

- the containment isolation function – the severe accident batteries can supply the containment outer isolation valves of the four divisions (after cross-connection from divisions 1 and 4);
- RCP [RCS] depressurisation using the Primary Depressurisation System (PDS);
- the I&C used by the operators in the Main Control Room for Severe Accident management. The severe accident instrumentation comprises:
  - in-containment pressure measurement,
  - in-containment dose measurement,
  - Main Control Room dose measurement.
- Main Control Room habitability during Severe Accident management:
  - the severe accident batteries can supply the Severe Accident I&C, the Severe Accident panel, the containment Annulus Ventilation System (EDE [AVS]) including iodine filtration, the crisis equipment room and the remote shutdown station.

The continued availability of the severe accident battery backed power supplies following loss of AC power sources requires the survival of the associated switchboards. The switchboards and the supported I&C systems are located in the Electrical Rooms within the four Safeguard Buildings (above the +0.0 m level). Operation of the essential switchboards and I&C systems requires that the temperatures in the rooms housing the equipment are maintained within the equipment qualification limits. Loss of AC power would cause loss of the Electrical Building Heating, Ventilation and Air Condition (HVAC) systems, leading to increasing environmental temperatures that could threaten equipment operability. Room by room thermal analysis will be provided to confirm that the temperature rise due to residual heating by consumers powered by the 2-hour and severe accident batteries will not threaten the continued availability of the severe accident battery-backed power supplies and the equipment they support for severe accident management. This analysis will be performed in the site-specific phase of UK EPR reactor licensing when details of heat loads and equipment layout are finalised. In the event that temperature limits are found to be exceeded, design modifications will be implemented, which could include one or more of the following options:

- measures for de-loading of unneeded consumers to reduce local heat sources
- reallocation of consumers to different rooms
- implementation of battery backed devices to redistribute heat away from critical rooms

2.2.2. Event sequence with SGs available (States A-C)

The key events and phenomena in the severe accident sequence that follows total loss of AC power are described in PCSR Sub-chapter 16.2. The PCSR sequence assumes recovery of the AC power supplies 12 hours after the event.

The event sequence in the bounding case (long-term loss of AC power in State A) is shown in Sub-chapter 16.6 - Figure 4. In that case, where ASG [EFWS] and Safety Injection System (RIS [SIS]) are not available, core heat-up occurs after a few hours, after steam generator emptying, leading to entry into Operating Strategy for Severe Accident (OSSA) conditions. Operators would act to depressurise the reactor using the PDS valves powered by the severe accident batteries to prevent RPV failure at high pressure that would result in a risk of containment failure due to Direct Containment Heating.

After occurrence of core meltdown and subsequent RPV failure, the corium would be dispersed into the dedicated spreading area of the core catcher. If AC power was not restored a gradual build up of pressure in the containment would then occur. The pressure would reach 9 bara, at which it is conservatively assumed that containment leaktightness may be lost, at around two days.

2.2.3. Event sequence with SGs unavailable (State D)

The bounding case for the grace period delay when there is no possibility of restoring core heat removal through the SGs, is total loss of AC power supply with the RCP [RCS] depressurised with the RPV upper head removed. The bounding case is State D with the water level at the ¾ loop level, as this corresponds to the minimum reactor coolant inventory.

A total loss of AC power implies that RCP [RCS] inventory make-up is not possible as all diesel generators supplying the two LHSI/RRA [RHRS] trains are unavailable. Core residual heat will thus cause vaporisation of the water inventory in the RCP [RCS] and the reactor cavity if filled. After core uncovery and meltdown RPV failure occurs within several hours. The containment pressurisation rate will be lower than that in the at-power case due to the reduced decay heat, so the event sequence up to loss of containment leaktightness is expected to be bounded by that shown in Sub-chapter 16.6 - Figure 4.

2.2.4. Event sequence for the Spent Fuel Pool

The bounding case for the grace period delay and cooling water consumption is state F, when all fuel assemblies are unloaded from the core into the spent fuel pool. The bounding sequence is the same as that described in section 2.1.3.

2.2.5. Modifications to prevent or mitigate loss of AC power sequences

Design changes have been identified to improve UK EPR resilience against loss of AC power sequences, including preventive measures and measures to extend the time before the loss of key safety systems that could lead to possible cliff-edge effects. The design changes proposed are described below:

- Preventive modifications will be implemented to seal all openings in the walls between the main EDG rooms and the UDG and severe accident battery rooms in the Diesel Building (see section 4.1.2). The aim will be to preserve the UDGs and severe accident batteries from severe external flooding events, which might cause water to enter the Diesel Buildings.

- A modification will be implemented to extend the life autonomy of the 'severe accident' batteries from 12 to 24 hours, and to provide an electrical connection to an internal fixed or mobile diesel generator set (see section 4.1.4). Even though an extension of the lifetime of the severe accident batteries will not have any significant impact in the prevention of core meltdown or uncovering of fuel assemblies in the SFP, this extension should significantly improve the overall plant robustness against such severe events prior to eventual recovery of AC power supplies. In particular the following would be improved:

    - severe accident management robustness, particularly through continued availability of severe accident I&C;

    - habitability of the main control room, crisis equipment room and remote shutdown station for human intervention (lighting, dose assessment and dose control by maintaining iodine filtration).

- A modification will be implemented to provide an external connection to the EVU [CHRS] to enable water injection into the containment spray system using a mobile pump (see section 4.1.9). This enables the containment pressure rise to be temporarily controlled. In that case, the amount of water which can be injected into the containment is limited by the need to maintain containment Severe Accident instrumentation integrity. It is estimated that the grace period extension before loss of containment leaktightness can be extended by three days by this means, leading to a total grace period of around five days before there is a need for recovery of an external electrical supply.

- Provisions will be made for the reconnection of a high power mobile diesel generator set (about 3 MVA) after three days following the complete loss of AC power supply (see section 4.1.5). The final power requirements for this diesel generator set and the fixed electrical connections remain to be fully defined, pending a decision on the consumers it will be required to support.

2.3. COMBINED LOSS OF POWER SUPPLIES AND LOSS OF ULTIMATE HEAT SINK

The most severe event involving total loss of AC electrical supplies also bounds the case of total loss of AC power supplies combined with total loss of ultimate heat sink. Therefore, the analyses of grace periods and potential countermeasures for grace period extension for the combined loss case are the same as those described in section 2.2 above.

3. ROBUSTNESS ANALYSIS OF SEVERE ACCIDENT MITIGATION MEASURES

The UK EPR reactor contains design features to limit the radiological consequences of severe accidents (core melt accidents), such that only very limited off-site countermeasures would be needed (no need for emergency evacuation beyond the immediate vicinity of the plant (i.e. no permanent relocation or long-term restrictions on the consumption of foodstuffs). The main severe accident mitigation features are as follows:

- to avoid rupture of the RPV at high pressure, which could lead to containment over-pressure failure due to Direct Containment Heating, dedicated valves (PDS valves) are provided to depressurise the primary circuit;
- to avoid a hydrogen explosion within the containment, the UK EPR is equipped with passive hydrogen recombiners;
- a corium spreading area is provided to produce a stable configuration for the molten core materials without endangering the integrity of the containment;
- a dedicated system is provided for heat removal from the containment (EVU [CHRS]) system. This safeguard system uses the SRU [UCWS] as a heat sink. Power is supplied to the EVU [CHRS] by the UDGs: one UDG is sufficient to carry out this function. The connection of the EVU [CHRS] to remove the decay power from the containment can be delayed by about two days without any risk of loss of containment leaktightness;
- cooling of the corium in the early phase of the accident is carried out passively by flooding the core spreading area with water from the In-Containment Refuelling Water Storage Tank (IRWST); this enables the connection of the EVU [CHRS] to be delayed by about two days;
- an Annulus Ventilation System maintains the inter-space between the containment and the adjacent structures at sub-atmospheric pressure so that leakages can be captured and filtered before discharge to atmosphere, helping to further limit off-site radiological consequences.

Operating principles for severe accident conditions are introduced in Sub-chapter 18.3 of the PCSR. Emergency operating procedures, and emergency planning measures to be carried out in the event of a severe accident, will have a major role in mitigating the consequences of a radioactivity release.

In the context of the analysis of the events that took place at Fukushima an additional robustness analysis was performed to identify areas where modifications could improve the robustness of the UK EPR design in severe accident situations [Ref-1]. The following additional design provisions were identified for implementation in the UK EPR to enhance the robustness of the plant against severe accidents (in addition to the modifications identified in section 2):

- addition of an activation command to the SA I&C cabinets containing severe accident processing units to ensure that the outputs from the SA I&C system severe accident processing units can always be reactivated following a loss of electrical power supplies (see section 4.1.12);

- implementation of a sound-powered telephone system that can be used in situations involving loss of all AC power supplies situations, to enable bi-directional communication to take place between field operators and the control room (see section 4.1.1).

More detail on these proposed modifications is provided in section 4 below.

Depending on each site situation, additional specific provisions will be considered on a case by case basis: mitigation measures will be described in the relevant site-specific PCSRs.

4. DESCRIPTION OF PLANNED MODIFICATIONS

As described in sections 2 and 3, the safety evaluation of the robustness of the UK EPR design against extreme events carried out in the light of the Fukushima events has identified a number of proposed design modifications, which will be incorporated into the UK EPR. These modifications are considered generic and are described in greater detail in section 4.1 below.

In addition to the generic design modifications, other post-Fukushima modifications were identified for implementation in the FA3 EPR, which is the reference design for the UK EPR [Ref-1]. These modifications apply to structures excluded from the GDA process as their design is site-specific (e.g. modifications to the pumping station), or are related to flood protection measures (e.g. improvements to leaktightness to doors of buildings to withstand depths of water on the site platform, which can only be defined when the flooding characteristics of a particular site are known). They are described in section 4.2 below.

Finally, a number of other modifications introduced during GDA are considered beneficial in improving resilience against beyond design basis events or preventing their occurrence. These ‘other relevant’ modifications are described in section 4.3.

The impact of the Fukushima event on the design requirements for new Nuclear Power Plants is still under evaluation in several countries. New requirements are emerging from these reviews. When a consensus view emerges, expected to be in the next 12 to 18 months, these requirements will be implemented in the UK EPR. A final identification of UK EPR design changes arising from the Fukushima event will only be possible in the site licensing phase of individual UK EPR projects.

4.1. POST-FUKUSHIMA MODIFICATIONS WITHIN GDA SCOPE

The following modifications introduced as a result of the post-Fukushima review are considered generic and will be included in GDA via the GDA design change management process.

4.1.1. Addition of a sound powered telephone network to the site communication system

A sound-powered telephone system will be implemented that can be used in loss of all electrical power supplies situations, to provide a local telecommunication solution for every state of the unit power supplies. The modification will enable bi-directional communication to take place between field operators and the control room.

Connection sockets will be installed throughout the plant at appropriate locations which will be identified in the detailed design phase.

4.1.2. Sealing penetrations in structures between the Emergency Diesel Generators and Ultimate Diesel Generators

Feedback from the events at Fukushima has underlined the critical role of the 12-hour batteries and the Ultimate Diesel Generators (UDG) in the prevention and mitigation of severe accidents in the UK EPR. This modification will involve sealing of all openings in the wall located between the main diesel generator rooms and the UDG and 12-hour battery rooms between the foundation raft and 1.00 m above the platform level. The seals must be leaktight against a water head between the foundation raft and 1.00 m above the platform level.

4.1.3. Extension of Ultimate Diesel Generator (UDG) running time

The modification is to extend the running time of the Ultimate Diesel Generators (UDG) to more than 24 hours, by providing mobile pumping equipment to extract fuel oil from the main diesel generator tanks and supply it to the ultimate diesel generators. It is assumed that the room containing the UDG fuel oil tank is not flooded, but that the room containing the Emergency Diesel Generator (EDG) fuel oil tank may be flooded.

The modification involves providing a leaktight pump to deliver a fuel flow which meets the requirement of the UDG motor. A control panel for the pump will be provided in the UDG "I&C cabinet" room, supplied with power from the UDGs.

Connecting pipework will be provided between the day tank feed systems for the emergency diesel generators and the day tanks for the ultimate diesel generators. The connecting pipework will contain a removable section, which will be manually connected by the operators in an emergency situation. The pipework will be disconnected in normal operation of the plant to eliminate any risk of Common Cause Failure of EDGs and UDGs due to fuel supply faults.

The sharing of fuel between the EDGs and UDGs in emergency conditions is preferred to the enlargement of fuel tanks, as enlargement of the tanks would cause problems of plant congestion and layout, which are avoided by the fuel sharing option. In addition, storage of an increased volume of fuel increases the potential consequences of fire, which is a safety disbenefit. As no safety disadvantages are introduced in the fuel sharing option, this option is therefore considered ALARP.

4.1.4. Modification of severe accident batteries capacity (from 12 hours to 24 hours) and implementation of fast connection between the severe accident battery chargers and a mobile diesel generator set

The severe accident 12-hour batteries will be modified in order to increase the autonomy of the associated equipment. The storage capacity of the two severe accident batteries will be increased from 12 hours to 24 hours. A fast connection between a mobile diesel generator set and the severe accident battery chargers will be provided to enable the batteries to be recharged as necessary.

4.1.5. Implementation of devices and equipment used to provide a high-power supply from day three after a severe accident

Devices and equipment will be implemented to allow the connection of a high-power mobile diesel generator that could restore power to the PTR [FPCS] and EVU [CHRS] systems in one electrical division in fault situations lasting longer than three days. The high-power mobile diesel generator would supply essential safety functions for controlling a severe accident, including particularly ensuring the habitability of the control room. In the event of road access to site not being possible, the high-power diesel would be brought to site by a transport helicopter.

4.1.6. Addition of motor-driven pump for re-supply of the Spent Fuel Pool from the raw water storage tank

An ultimate water make-up supply for the spent fuel pool will be installed via the external connection to the dry-risers.

The ultimate water make-up for the spent fuel pool will be installed as hard piping and will be designed to be operational after an earthquake. The external connectors will be protected against external hazards, in particular natural events such as flooding, and should be accessible to allow connection to a mobile make-up device drawing water from the raw water storage reserves. A fixed connection between the dry-risers and the safeguard systems for water make-up of the spent fuel pool should be installed, with the adequate isolation points and the appropriate level of classification (mechanical for instance).

4.1.7. Addition of motor-driven pump for re-supply of ASG tanks from the raw water storage tank

The purpose is to increase the running time of the secondary side cooling by setting up a fresh water re-supply of the ASG [EFWS] tanks utilising the raw water storage reserves. The current design already has provisions for a final supply line used to replenish tank ASGi110BA via isolation valve ASGi102VD using a mobile device. The proposed change therefore consists of providing details in the ASG [EFWS] system design manual of how this make-up could be performed by using the raw water storage reserves.

4.1.8. Implementation of a seismically qualified facility, passive or automatic, for opening the outlet from the fuel pit to the Nuclear Auxiliary Building chimney in order to improve protection against overpressure in the Fuel Building fuel pit area

In the case of a total loss of all emergency power source supply and/or total loss of heat sink, the temperature of the spent fuel pool increases until it begins to boil. Water evaporation in the spent fuel pool may result in a pressure increase in the spent fuel pool area that could lead to a loss of integrity of the civil engineering structures. To prevent such pressure increases, the steam produced will be evacuated, via the outlet in the spent fuel pool area, to the stack in the Nuclear Auxiliary Building.

The outlet duct between the spent fuel pool area and the Nuclear Auxiliary Building stack contains an isolation damper that is opened manually from the control room or local to plant. Either a rupture disk will be installed in the duct or the current damper will be replaced by a damper that fails open on total loss of the electrical power supplies. In this way, in the event of total loss of power sources, the outlet from the spent fuel pool area will be opened passively, preventing over pressurisation of the Fuel Building.

The steam discharge through the stack in this situation will be unfiltered. It is confirmed in Sub-chapter 15.5 of the PCSR that the radiological consequences of an unfiltered discharge of steam due to boiling of the Spent Fuel Pool are acceptable for low frequency event sequences (release consequences below Dose Band 1).

Sub-chapter 3.3 of the PCSR confirms that the Nuclear Auxiliary Building stack will be seismically qualified at SC2, and will thus retain its geometry in seismic events.

4.1.9. Provision of mobile pump for make-up to Reactor Building from raw water storage tank via the EVU [CHRS]

In long term loss of electrical supplies, it is not possible to utilise the EVU [CHRS] system to limit pressure build-up in the reactor building.

This modification is to enable make-up to be supplied to the reactor building via the EVU [CHRS] system at 48 hours, at which point the pressure in the containment will be below a pressure of 9 bara, which is conservatively assumed as the pressure at which containment leaktightness would be lost. Make-up water will be provided by a mobile pump located on the platform at +0 m, which would be supplied with water from the raw water storage reserves.

4.1.10. Implementation of facility to prevent generation of an explosive hydrogen atmosphere (resulting from radiolysis of water) in the Fuel Building fuel pit area in the event of long-term loss of electrical supplies

During radiolysis of water in the spent fuel pool, hydrogen is released in the fuel pool area. In the event of total loss of electrical supplies, the ventilation system in the fuel pool area would not function. Hydrogen could therefore accumulate in the spent fuel pool area, which in the event of ignition may threaten the integrity of the building.

To prevent any risks related to hydrogen combustion, the hydrogen concentration in the spent fuel pool area should be reduced and the local formation of regions with higher hydrogen concentrations should be prevented.

The modification will consist of a study to quantify the hydrogen risk from radiolysis in the absence of ventilation and the implementation of mitigation measures if required.

4.1.11. Backup of essential data relating to changing conditions in the spent fuel pool in the event of loss of cooling

The safety assessment conducted in the light of the Fukushima accident revealed a potential loss of essential data required by the operators for ascertaining the physical condition of the spent fuel pool in the event of loss of cooling. The following data were identified as essential:

- water temperature;
- water level;
- dose rate in the fuel pool area.

The identified limiting condition is the loss of off-site power and of all installed emergency power sources (i.e. station blackout + loss of the UDG diesel generators). In such conditions, the instrumentation ceases to be powered when the 2-hour batteries are depleted.

Continued operation of essential instrumentation would be required in the event of long-term loss of cooling of the spent fuel pool, during which the pool water boils but the fuel assemblies must remain covered with a layer of water, irrespective of the unit's operating condition.

The proposed modification consists of including the pool water temperature, pool water level and pool area dose rate measurements in the dedicated severe accident I&C system and the severe accident control desk. Consideration will be given to the need to provide emergency power for the identified instrumentation from the severe accident batteries.

4.1.12. Activation command added to Severe Accident I&C cabinets

If the electrical power supply is lost in the Severe Accident I&C system, it is not currently possible to reactivate the outputs from the severe accident processing in some accident scenarios.

The proposed modification is to add a dedicated pushbutton in the Severe Accident I&C cabinets containing the Severe Accident processing units, in order to ensure that the outputs from the Severe Accident I&C system severe accident units can always be reactivated. In addition to the pushbutton, a lamp (also located inside the cabinets) will be added to indicate that the outputs are activated.

4.1.13. Study into achieving safe position for fuel assemblies during accident conditions

The post-Fukushima additional safety studies carried out for the FA3 EPR indicated the need to secure the ‘move fuel assembly to safe position’ operation during fuel handling, in the scenario: Earthquake + Loss of power supplies + Loss of spent fuel pool cooling system.

Following loss of the spent Fuel Pool Cooling System (PTR [FPCPS/FPPS]), the rapid increase in water temperature requires that any fuel assembly being handled must be moved to a safe position within a guaranteed timeframe. The modification consists of providing means to ensure that the Refuelling Machine and Spent Fuel Mast Bridge can be operated manually after an earthquake and to identify self-contained, portable devices that could be used to operate the Fuel Handling System to speed up the movement of a fuel assembly to a safe position.

4.2. POST-FUKUSHIMA MODIFICATIONS NOT INCLUDED IN GDA SCOPE

4.2.1. Modifications/studies identified for FA3

The following modifications, which are considered outside GDA scope due to their site/project specific nature or their preliminary state of development, have been identified for the FA3 EPR [Ref-1]. These modifications will be considered for implementation during the NSL phase of individual UK EPR projects. (Note that where studies are indicated, these are feasibility/assessment studies which, depending on their outcome, may result in physical modifications to plant.)

- modification to ensure seismic resistance of piping, valves and pumps involved in raw water reserve used by other post-Fukushima enhancement measures;
- modification to limit ingress of water to the pumping station slab (increase in flooding protection of the pumping station);

- modification to limit ingress of water to the outfall slab (reinforced leaktightness of access doors to the rooms housing the fire fighting system, JAC);

- study of the seismic resistance of sealing elements (flood protection) of structures that house equipment used to mitigate severe accident scenarios[2]. This will include the seismic verification of the seals that prevent water ingress into the core spreading area (core catcher): the core spreading area must remain initially dry under severe accident conditions;

- study of leaktightness of security access doors to Nuclear Island buildings housing safety functions to identify if improvements are necessary to prevent water entering the buildings;

- study of the performance of equipment needed in extreme accident scenarios beyond the current design basis to identify:

    - minimum instrumentation required in the reactor building (containment integrity);

    - leaktightness of containment penetrations.

4.3. OTHER RELEVANT MODIFICATIONS INTRODUCED IN GDA

A number of other design modifications have been introduced in GDA via Change Management Forms which are considered beneficial in helping the UK EPR to withstand beyond design basis events or avoid their occurrence. These modifications are listed below:

<u>Upgrade of classification of Ultimate Diesel Generator Safety Features</u>

This design modification involves upgrading the safety class of the Ultimate Diesel Generator safety features from Class 3 to Class 2. It will reduce the risk of severe accidents by increasing the reliability that may be claimed from the UDGs in events involving failure of off-site and on-site AC power supplies.

<u>Upgrade of classification of main fuel pool cooling train Safety Features (part of PTR [FPCS/FPPS] system)</u>

This design modification involves upgrading the safety class of the main fuel pool cooling train safety features from Class 2 to Class 1. It will be of benefit by increasing the reliability that may be claimed from the fuel pool cooling system in within and beyond design basis accidents.

<u>Modification of DVL / DEL HVAC systems</u>

This modification consists of design improvements to the DVL/DEL HVAC systems, which provide cooling to the Electrical, Control and Instrumentation (EC&I) rooms within the Safeguard Buildings. The modification consists of upgrading the Safety Features within the main HVAC trains from Class 2 to Class 1, and providing two new back-up trains (DVLnew/DELnew) at Class 1/Class 2 and of diverse design to the main trains, which will be automatically actuated on loss of a main HVAC train. The modifications will help reduce the risk of beyond design basis events arising from failures within the HVAC systems.

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[2] Modification being considered for the UK EPR only

Loss of Essential Support Systems – Post Accident Management

These modifications are to improve the robustness of the UK EPR against faults in the RRI [CCWS] or SEC [ESWS] essential cooling systems. They include upgrading to Class 1 the Safety Features which isolate an RRI [CCWS] train in case of leakage, and those which switch over the RRI [CCWS] cooling functions to the standby train in case of failure of the operating train. The I&C systems and sensors that automatically trip a Reactor Coolant Pump following loss of pump cooling systems will also be upgraded to Class 1. The modifications will help reduce the risk of beyond design basis events arising from failures within essential cooling water systems.

Internal Flooding Protection – Design Modification of the Firefighting System (JPI) in the Annulus

The modification involves replacing the valves that would be used for isolating the leak (currently manual valves operated local to plant) with motorised valves, which would be automatically closed on detection of flooding in the Reactor Building Annulus. Additional valves will be provided so that isolation can be achieved despite a random failure of a valve to close. The modification will allow automatic termination of the leakage before safety limits are exceeded and will help reduce the risk of beyond design basis events arising from flooding of the Reactor Annulus.

Internal Flooding Protection – Design Modification of the Essential Service Water System (SEC [ESWS]) in the Safeguard Auxiliary Building

To mitigate the flooding risk due to gross failure of SEC [ESWS] pipework in a Safeguard Auxiliary Building, a modification is being introduced to install new dedicated level sensors in each Safeguard Auxiliary Building (in rooms containing SEC [ESWS] pipework) linked to a new alarm in the MCR. The new alarm will warn the operator of the occurrence of a flooding event due to a large leakage flow rate. A new operational procedure linked to the alarm will require the operator to trip the relevant SEC [ESWS] pump and close a manual valve upstream of the pump to isolate the leak. The modifications will help reduce the risk of beyond design basis events arising from flooding of the Safeguard Buildings.

Internal Flooding Protection – Design Modification of the Nuclear Island Demineralised Water Distribution System (SED) in the Reactor Building Annulus

To mitigate the flooding risk due to gross failure of pipework in the SED system in the Reactor Building Annulus, a modification is being introduced to replace an isolation valve currently operated local to plant with a motorised valve operable from the MCR. The modification will allow more reliable actions to be carried out to terminate the leakage before safety limits are exceeded and will help reduce the risk of beyond design basis events arising from flooding of the Reactor Annulus.

Modification to install cover plates and standpipes over floor drains in flooded compartments

The modification consists of installing cover plates and standpipes for temporary isolation of pool purification lines in the floors of flooded compartments, avoiding risk of draining compartments due to gross failure of the purification lines.

The modification contributes to the seismic robustness of the reactor design, and helps respond to the recommendation in the ONR review report on Fukushima [Ref-1] relating to avoidance of bottom penetrations in spent fuel pools.

Upgrade of classification of fuel pool make up Safety Feature (part of JAC/JPI)

This design change involves upgrading the fuel pool emergency make-up safety feature from Class 2 to Class 1. It will be of benefit by increasing the reliability that may be claimed from the fuel pool make-up safety function following within and beyond design basis accidents.

Modification to provide leaktight containment of the Fuel Transfer Tube

The modification consists of making the rooms enclosing the Fuel Transfer Tube watertight to a pressure corresponding to the maximum water level in the pools. It avoids the risks due to draining of compartments due to a seismically induced failure of the Fuel Transfer Tube, when the Fuel Transfer Tube is de-isolated to allow transfer of fuel assemblies between the Reactor Building and Fuel Building.

The modification contributes to the seismic robustness of the reactor design, and helps respond to the recommendation in the ONR review report on Fukushima [Ref-1] relating to avoidance of bottom penetrations in spent fuel pools.

Modification to remove personnel access doors to the reactor cavity and Fuel Transfer Comportments

The modification consists of removing the three personnel access doors located at the floor level of the Reactor Cavity and the Fuel Transfer Compartments.

The modification will ensure the avoidance of risks due to draining of compartments caused by a hypothetical seismically induced failure of the personnel access doors, when fuel assemblies are being transferred between the Reactor Building and Fuel Building.

The modification contributes to the seismic robustness of the reactor design, and helps respond to the recommendation in the ONR review report on Fukushima [Ref-1] relating to avoidance of bottom penetrations in spent fuel pools.

Modification to cask loading procedure

The modification consists of modifying the cask loading procedures so that the door between the Spent Fuel Pool and the Cask Loading Pit will be closed before the penetration upper cover is opened to allow a fuel assembly to be lowered into the fuel cask. This will create a second barrier to prevent draining of the Spent Fuel Pool following hypothetical failure of the bellows connecting the fuel cask to the bottom penetration in the Cask Loading Pit. The modification will allow operators to continue working in the Spent Fuel Pool Hall to recover a dewatered fuel assembly in the Cask Loading Pit in case of failure of the bellows.

The modification contributes to the seismic robustness of the reactor design, and helps respond to the recommendation in the ONR review report on Fukushima [Ref-1] relating to avoidance of bottom penetrations in spent fuel pools.

Connection of Reactor Coolant Pump thermal barrier cooling system

The design modification is to improve the robustness of the Reactor Coolant Pump thermal barrier cooling function by inter-connecting the thermal barrier cooling systems of the four Reactor Coolant Pumps so that the cooling function is not lost in the event of failure of two of the four trains of the RRI [CCWS] system. The Safety Features performing the thermal barrier cooling function will also be upgraded from Class 2 to Class 1. The modifications will help reduce the risk of beyond design basis events arising from failures within essential cooling water systems.

Modifications to address Common Cause Failure (CCFs) of Electrical Systems

The modifications consist of reallocating electrical consumers to different voltage levels so that the plant can be brought to a controlled state in beyond design basis events involving CCF of 690V (LJ) or 400V (LV) electrical systems. They involve reallocating the back-up ventilation system of the electrical buildings from the 690V (LJ) to the 400V supplies, and reallocating various valve actuators from the 400V (LV) to the 220V (LA) supplies. The modifications help reduce the risk of beyond design basis events involving multiple failures within redundant electrical systems operating at the same voltage level.

Modifications to manage the Total Loss of Cooling Chain (TLOCC)

The modifications involve design improvements to systems used for mitigation of total loss of the cooling function provided by the RRI [CCWS] / SEC [ESWS], which could result in a Loss of Coolant Accident due to a failure of the Reactor Coolant Pump seals if it occurred at power. The main modifications consist of upgrading to Class 1 the Safety Features providing the containment spray function or IRWST pool cooling function and the Safety Features in the DEL [SCWS], which provides diverse cooling of LHSI pumps of trains 1 and 4 in the event of failure of cooling by the RRI [CWCS]. The modifications will help reduce the risk of beyond design basis events arising from failures within essential cooling water systems.

5. SUMMARY OF ROBUSTNESS ANALYSIS FOLLOWING FUKUSHIMA EVENT

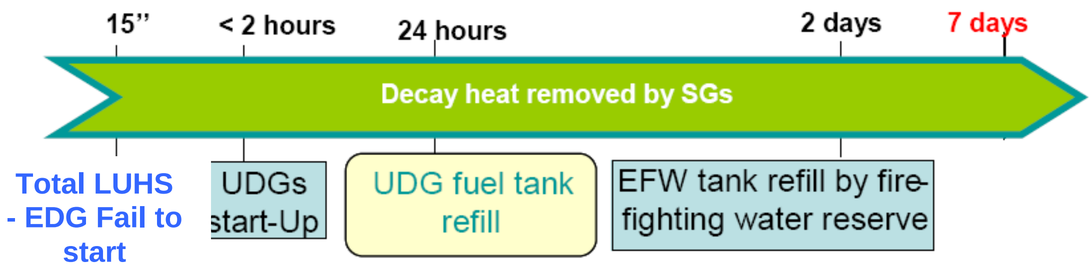
The UK EPR has been designed to meet safety objectives for Generation 3+ reactors, which include reduced core melt frequency, enhanced protection against external and internal hazards, and significant reduction in the radiological risk to the public if a core melt was to occur. The reduced risk of a severe accident is achieved by the implementation of four-fold redundancy in main safety systems such as the Emergency Feedwater and Safety Injection Systems, and by provision of diversified back-up systems which can be used in case of common cause failure of redundant safety trains. Severe accident scenarios have been taken into account at the design stage, the design objective being that only very limited off-site countermeasures would be needed in a core melt accident.

Following the events at the Fukushima Daiichi plant in Japan, EDF and AREVA initiated a safety evaluation of the UK EPR to review the robustness of the design against extreme events such as those which occurred at Fukushima (earthquake and flooding) and against other unspecified scenarios involving prolonged loss of AC power and the ultimate heat sink. The aim was to identify margins in the design to cliff-edge effects, and define additional reasonably practicable measures that could be applied to further improve robustness.

The safety analysis performed has shown that following extreme events such as long term loss of AC power or heat sink, or a combination of the two, more than two days would be available in the worst case to restore reactor cooling, before there was a risk of a significant radiological release. Modifications using fixed and mobile equipment have been identified which will allow this grace period to be significantly extended. Other design modifications introduced during GDA which are beneficial in reducing the risk due to beyond design basis accidents have been identified.

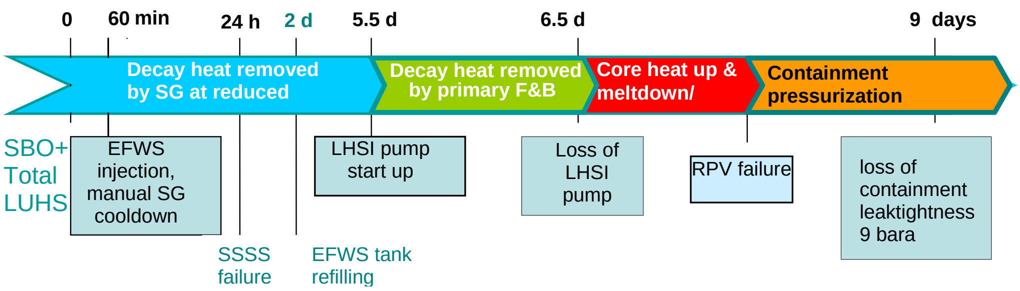
SUB-CHAPTER 16.6 - FIGURE 1

Time Line for Post Accident Management (SBO + Total LUHS) – SGs Available [Ref-1]



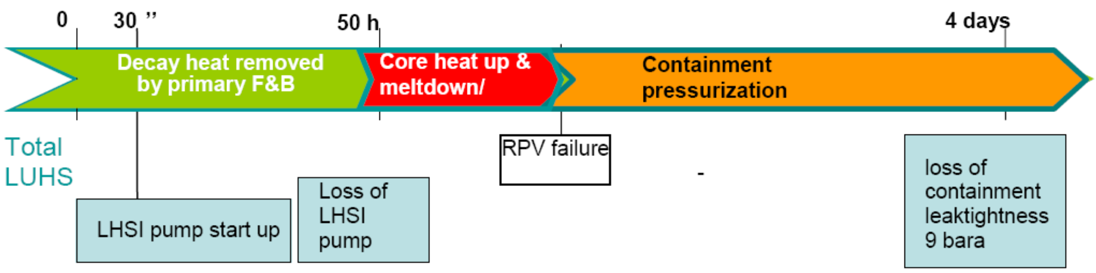
SUB-CHAPTER 16.6 - FIGURE 2

Time Line for Post Accident Management (SBO + Total LUHS + Failure of Reactor Coolant Pump Seals – SGs Available [Ref-1]



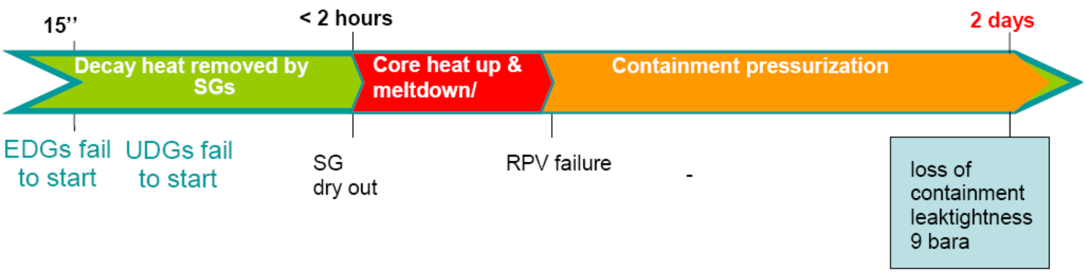
SUB-CHAPTER 16.6 - FIGURE 3

Time Line for Post Accident Management (SBO + Total LUHS) – State D [Ref-1]



SUB-CHAPTER 16.6 - FIGURE 4

Time Line for Post Accident Management (Total Loss of AC Power Sources) - Power States [Ref-1]



SUB-CHAPTER 16.6 – REFERENCES

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

0. INTRODUCTION AND SAFETY REQUIREMENTS

0.1. DESCRIPTION OF FUKUSHIMA EVENT AND IMPACT ON UK EPR DESIGN

[Ref-1] European Nuclear Safety Regulators Group (ENSREG) “Stress test” specifications. May2011. (E)

[Ref-2] Western European Regulators Association (WENRA) “Stress test” specifications proposed by the WENRA task force. April 2011. (E)

[Ref-3] Japanese earthquake and tsunami: Implications for the UK nuclear industry: Final Report. HM Chief Inspector of Nuclear Installations. September 2011. (E)

1. ROBUSTNESS ANALYSIS OF PROTECTION AGAINST SEVERE EXTERNAL EVENTS

1.1. SEISMIC EVENTS

[Ref-1] UK-EPR GDA Project - Summary of the Additional Safety Evaluation against Beyond Design Basis Earthquakes. PEPS-F DC 151 Revision A. AREVA. October 2012. (E)

1.2. EXTERNAL FLOODING EVENTS

[Ref-1] UK EPR GDA - Design against flooding events. E.T.DPNN/120048 Revision E. EDF. October 2012. (E)

2. ANALYSIS OF ROBUSTNESS AGAINST LOSS OF AC POWER/LOSS OF ULTIMATE HEAT SINK

[Ref-1] UK EPR GDA Project – Robustness of Power Sources and Long Term Cooling. PEPS-F DC 133 Revision C. AREVA. November 2012. (E)

2.1. TOTAL LOSS OF ULTIMATE HEAT SINK (TOTAL LUHS)

2.1.1. Event sequence with Steam Generators available (States A to C)

[Ref-1] UK EPR GDA Project – Robustness of Power Sources and Long Term Cooling. PEPS-F DC 133 Revision C. AREVA. November 2012. (E)

2.1.1.2. Event sequence in State A-C – failure of reactor coolant pump seals

[Ref-1] UK EPR GDA Project – Robustness of Power Sources and Long Term Cooling. PEPS-F DC 133 Revision C. AREVA. November 2012. (E)

2.1.3. Cooling of the Spent Fuel Pool

[Ref-1] UK EPR GDA Project – Robustness of Power Sources and Long Term Cooling. PEPS-F DC 133 Revision C. AREVA. November 2012. (E)

2.2. TOTAL LOSS OF AC POWER SOURCES

[Ref-1] UK EPR GDA Project – Robustness of Power Sources and Long Term Cooling. PEPS-F DC 133 Revision C. AREVA. November 2012. (E)

3. ROBUSTNESS ANALYSIS OF SEVERE ACCIDENT MITIGATION MEASURES

[Ref-1] Severe Accident Management. ECESN120395 Revision B. EDF. November 2012. (E)

4. DESCRIPTION OF PLANNED MODIFICATIONS

[Ref-1] Summary of the additional safety evaluation of the UK EPR design following Fukushima events. ECUK110959 Revision A. EDF. December 2011. (E)

4.2. POST-FUKUSHIMA MODIFICATIONS NOT INCLUDED IN GDA SCOPE

4.2.1. Modifications/studies identified for FA3

[Ref-1] Summary of the additional safety evaluation of the UK EPR design following Fukushima events. ECUK110959 Revision A. EDF. December 2011. (E)

4.3. OTHER RELEVANT MODIFICATIONS INTRODUCED IN GDA

[Ref-1] Japanese earthquake and tsunami: Implications for the UK nuclear industry: Final Report. HM Chief Inspector of Nuclear Installations. September 2011. (E)

SUB-CHAPTER 16.6 - FIGURES 1 TO 4

[Ref-1] UK EPR GDA Project – Robustness of Power Sources and Long Term Cooling. PEPS-F DC 133 Revision C. AREVA. November 2012. (E)