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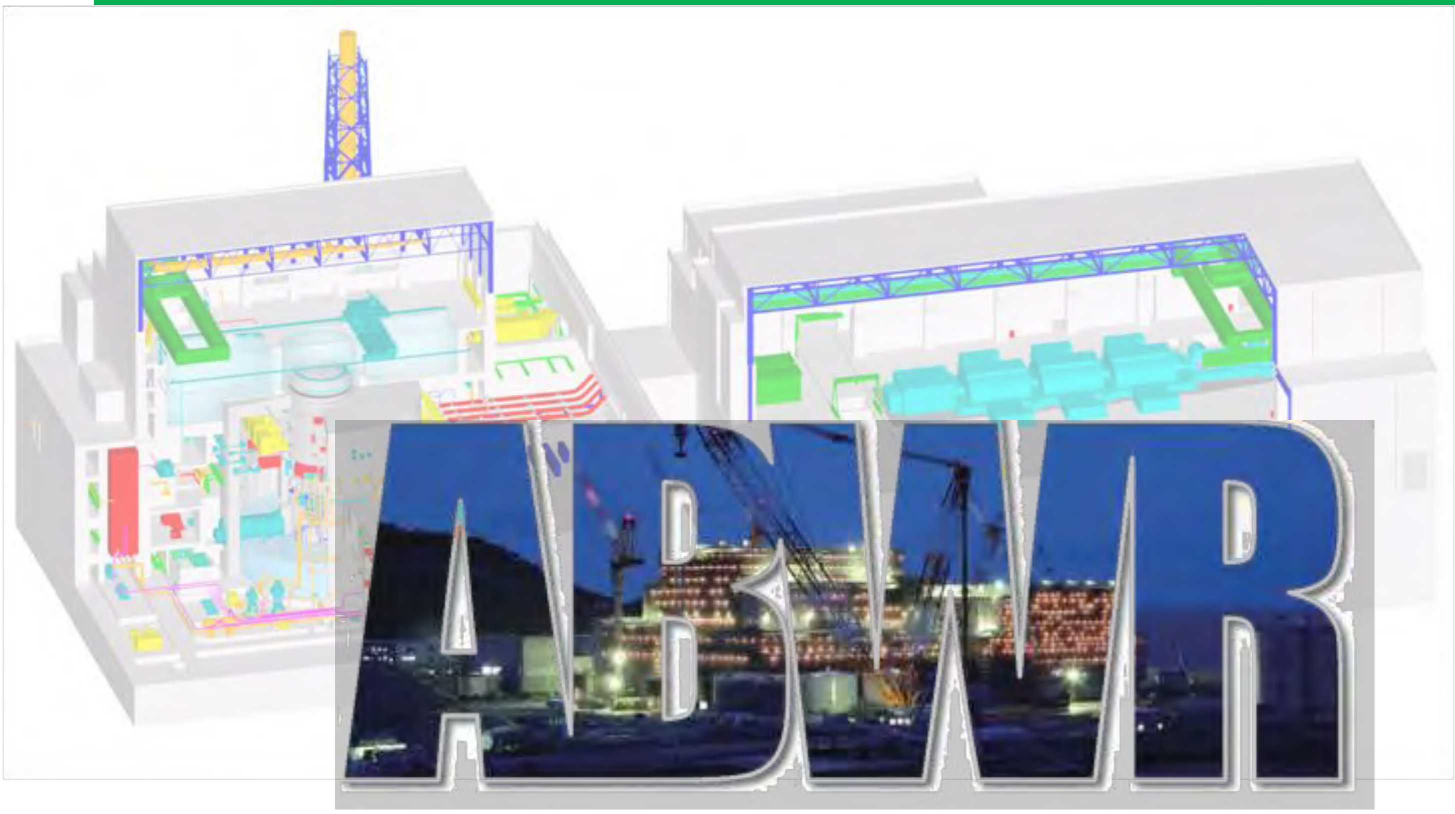
UK ABWR Generic Design Assessment

Generic PCSR Chapter 26 : Beyond Design Basis and Severe Accident Analysis



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26.1 Introduction

Beyond Design Basis Analysis (BDBA) and Severe Accident Analysis (SAA) are carried out as a part of fault assessment for UK ABWR. This chapter shows descriptions of BDBA and SAA.

26.2 Beyond Design Basis Analysis

26.2.1 Introduction

Beyond Design Basis Analysis (BDBA) demonstrates that beyond design basis faults do not lead to melting or considerable damage of the core to prevent significant environmental release of any radioactive material by the adequate safety provisions based on defense in depth concept. Design basis fault in combination with multiple failure of safety system, common cause failure of safety system, or safety related system is treated as a beyond design basis fault in the case that its frequency is estimated to be between 10^{-5} and 10^{-7} pa. In this section, identification and grouping of beyond design basis faults and acceptance criteria for beyond design basis faults are discussed. Thermal hydraulic analysis and dose evaluation for BDBA will be carried out under more realistic and best estimate assumptions differently from DBA in Step3. And then, analysis results will be described in the next updated PCSR.

26.2.2 Identification and Grouping of Beyond Design Basis Faults

26.2.2.1 Identification and Grouping Approach

Beyond design basis faults are identified on the basis of the following points of view.

(1) The core is significantly damaged by multi-systems failure caused by common cause failures (CCF) or functional dependency.
(2) The time margin to implement countermeasure for core damage prevention is small.
(3) The fault sequence is representative among the fault sequence group in terms of severity.

For UK ABWR, the following systematic approach is taken to identify beyond design basis faults and group these faults based on similar fault sequence and demands on safety functions.

(1) Identification of beyond design basis faults based on Event Tree Analysis
(2) Identification of beyond design basis faults based on key safety system CCF analysis
(3) Consideration of very low frequency event such as interfacing system LOCA (ISLOCA)

26.2.2.2 Result of Identification and Grouping Fault Sequence

This subsection summarizes a result of identification and grouping fault sequence of beyond design basis faults. More detailed information about this evaluation is described in Topic Report for Fault Assessment [Ref-1].

(1) Identification of Beyond Design Basis Faults Based on Event Tree Analysis

For the purpose of identification of beyond design basis fault, the Event Tree Analysis is carried out for reactor internal faults. UK ABWR has safety systems such as Reactivity Control function, Reactor Core Cooling function, Long-term Heat Removal function as a protective measure to prevent leading the significant core damage from initial event and to achieve safe and controllable shutdown state.

In the Event Tree Analysis, by estimating roughly a combination of initiating fault frequency in case these safety systems fail randomly and random failure probability of single or multiple safety system, the fault sequence of beyond design basis fault whose value of frequency estimation is between 10^{-5} and 10^{-7} pa is identified.

In light of initiating fault frequency and key safety function, the Event Tree Analysis is carried out for frequent design basis faults except LOOP, LOOP and LOCA. With regard to LOOP, the loss of the emergency AC power supply and emergency DC power supply will lead to loss of multiple safety function, thus, these losses are considered in the event tree.

Figure 26.2-1-3 shows Event Tree Analysis for each fault. The following faults are identified as Beyond Design Basis Faults from simple Event Tree Analysis.

- Loss of all ECCS
  - Frequent Fault with multiple failure of all ECCS pumps by mechanical failure
  - Frequent Fault with CCF of ECCS initiation logic component and failure of ECCS manual start
  - Short term LOOP with CCF of EDGs or AC-Bus + Failure of RCIC circuit or pump
  - Short/Medium term LOOP with CCF of DC-bus

- Station Blackout (SBO)
  - Long term LOOP with CCF of EDGs or AC-bus

- Loss of decay heat removal
  - Small break LOCA with CCF of RHR control circuit or multiple failure of RHR pumps/heat exchangers

– Coolant injection failure at LOCA
  • Small break LOCA with loss of high pressure coolant injection (CCF of RCIC/HPCF control circuit or multiple failure of RCIC/HPCF pumps)

– ATWS(ATWT)
  • Small break LOCA with CCF of RPS

(2) Identification of Beyond Design Basis Faults Based on Key Safety System CCF Analysis

The frequency of the combination of design basis faults and a Common Cause Failure (CCF) of essential service or support system might become between 10^{-5} and 10^{-7} pa. In order to identify beyond design basis fault, CCF Analysis is carried out for the key safety systems on UK ABWR. Table26.2-2 shows provisional basic safety system configuration for frequent design basis faults in UK ABWR. This table summarizes dependencies between front line safety systems including diverse protective measures and their essential services and support systems such as controllers, power distribution systems, cooling chain systems (RCW/RSW), heating, ventilation and air conditioning (HVAC) systems and so on.

(a) Evaluating Condition

For each major essential service or support system, the following six cases of CCF analysis are carried out. In addition, in carrying out CCF analysis, rough estimation of each CCF frequency is assumed as indicated in Table26.2-1. CCF frequencies of class 1 systems (EDGs, Digital C&I and RCW/RSW) are assumed to be about 10^{-4}/demand. However, CCF frequency of DC-bus is assumed to be about 10^{-5}/demand because DC-bus is usually considered to be more reliable than EDGs. In addition, CCF frequency of HVAC (Main Control Room (MCR)) is assumed to be about 10^{-3}/demand because it has 2 trains.

Table 26.2-1: Assumption of CCF frequency

Case	CCF mode	Assumption of Frequency
1	CCF in EDGs	to 10^{-4}/demand
2	Digital CCF	to 10^{-4}/demand
3	CCF in 6.9kV Switch Board	to 10^{-4}/demand
4	CCF in DC-bus	to 10^{-5}/demand
5	CCF in RCW/RSW	to 10^{-4}/demand
6	CCF in HVAC(MCR)	to 10^{-3}/demand

(b) Result of CCF analysis

Key safety system CCF analysis evaluates availability of front line safety systems in the case of CCF of each essential service or support system based on dependencies between front line systems and their essential services and support systems shown in Table26.2-2. In evaluation of consequential failure of front line systems due to CCF of essential service or support system, failures of support systems are considered that depend on essential service or support system initially failed and consequential failure of front line system due to failure of subordinative support system is considered. In addition, due to postulated failure of essential service or support system, it is identified not only the system that is not available immediately but also the ones are not available after delay.

(c) Grouping of Fault Sequence

CCFs of 6.9kV Switch Board (Case3), RCW/RSW (Case5) and HVAC (Case6) are bounded by SBO case (Case1) in terms of the similarity of availability of safety systems, which is already identified by Event Tree Analysis. CCF of digital C&I (Case2) is considered as a new fault sequence. In addition, CCF of DC-bus on Non-LOOP (Case4-1) is bounded by CCF in DC-bus on LOOP (Case4-2), which is already identified by Event Tree Analysis and equivalent to loss of all ECCS. As mentioned above, the following three bounding fault sequence are identified based on key safety system CCF analysis.

- Loss of all ECCS
- Station Blackout (SBO)
- Loss of digital C&I

(d) Identification of Beyond Design Basis Fault Based on Safety System CCF Analysis

Based on results of fault sequence grouping and simple estimation of fault sequence frequency, the following faults are identified as Beyond Design Basis Faults. In addition, small break LOCA with SBO bounds small break LOCA with CCF of RHR control circuit etc. identified by Event Tree Analysis because not only RHR but also HPCF are not available in the SBO fault sequence. And also, small break LOCA with loss of digital C&I bounds small break LOCA with CCF in RPS identified by Event Tree Analysis.

- Loss of all ECCS
  - FF with loss of all ECCS
  - Short/Medium term LOOP with loss of all ECCS

- Station Blackout (SBO)
  - Long term LOOP with CCF in EDGs

- Coolant injection failure at LOCA
  - Small break LOCA with loss of digital C&I
  - Small/Medium break LOCA with SBO

- Loss of digital C&I
  - Short/Medium term LOOP with loss of digital C&I

(3) Consideration of very low frequency event

As a very low frequency event, an Interfacing System LOCA (ISLOCA) is identified. It is a type of LOCA in which the reactor pressure boundary is breached and isolation valve is miss-opened by operator at the interface with a low pressure system. And then, the reactor coolant is released to the low pressure system. If the interfacing system can not withstand the resulting pressure, it will also be breached leading to a loss of coolant from its boundary. If this happens outside the containment, the coolant inventory will be depleted and a direct leak path to the environment will be created.

26.2.2.3 Summary of Identification of Beyond Design Basis Faults

As mentioned in subsection 26.2.1.2, the following faults are identified as bounding fault sequences of beyond design basis faults from Event Tree Analysis, safety system CCF analysis and consideration of very low frequency event. In addition, Short term LOOP with loss of all ECCS and short term LOOP with loss of digital C&I are bounded by Medium term LOOP ones in the light of severity. In the same way, Small break LOCA with SBO is bounded by Medium break LOCA with SBO. It is also note that this evaluation might be re-assessed in Steps 3 according to design progress and PSA feedback.

- Loss of all ECCS
  - FF with loss of all ECCS
    (Multiple failure of all ECCS by mechanical failure, CCF of ECCS initiation logic component and failure of ECCS manual start)
  - Medium term LOOP with loss of all ECCS
    (CCF of EDGs or AC-Bus + Failure of RCIC circuit or pump, CCF of DC-bus.)

- Station Blackout (SBO)
  - Long term SBO
    (CCF in EDGs or AC-bus)

- Coolant injection failure at LOCA
  - Small break LOCA with loss of high pressure coolant injection

(Failure of RCIC/HPCF)
- Small break LOCA with loss of digital C&I
- Medium break LOCA with SBO

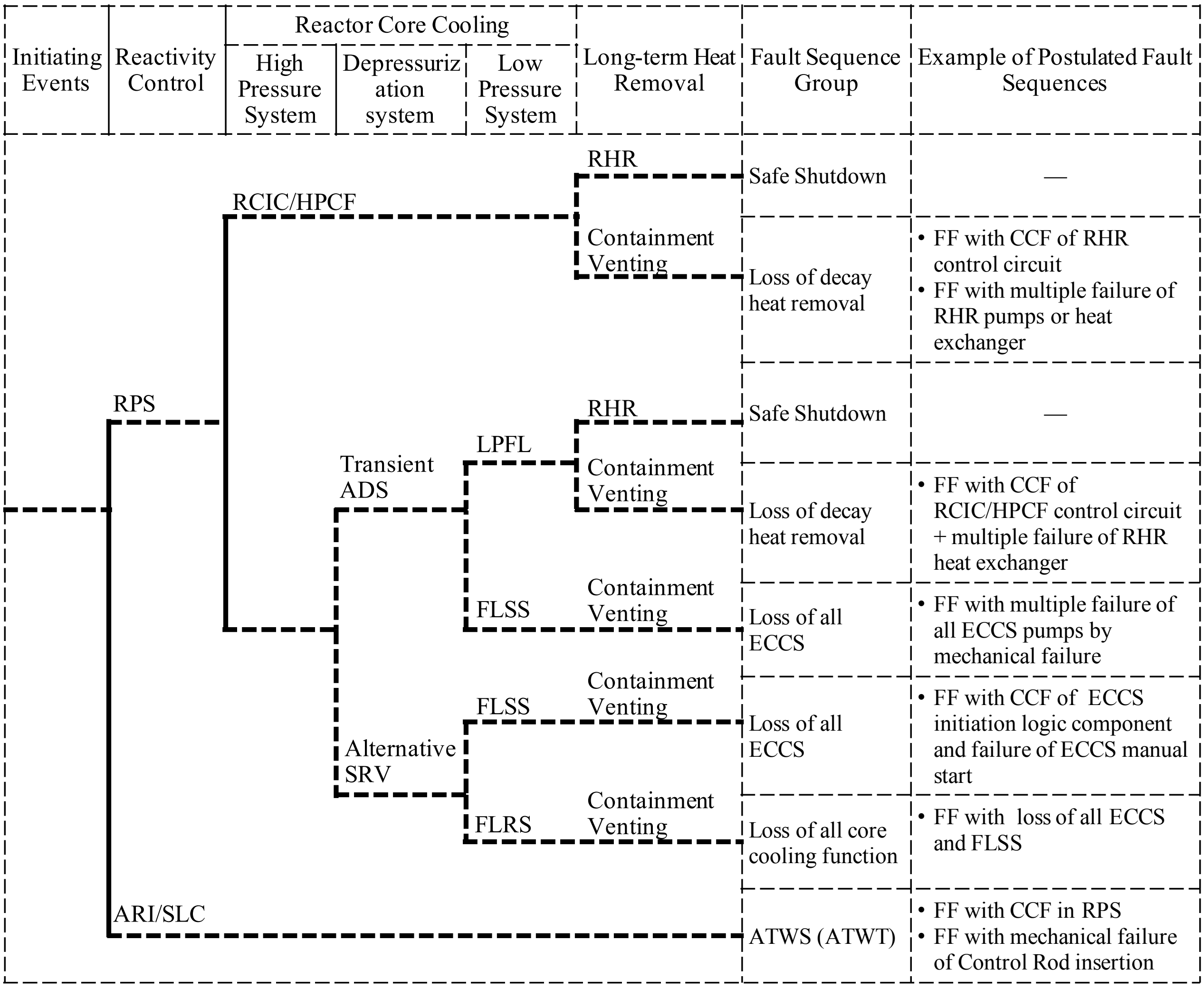
– Loss of digital C&I
  - Medium term LOOP with loss of digital C&I

– Interfacing system LOCA
  - Interfacing system LOCA

Table 26.2-2: Basic Safety System Configuration for Frequent Design Basis Faults in UK ABWR (Provisional)

Essential Functions	Safety Class	Safety system	Platform	Power Source	Support System
Reactivity Control	Class 1	RPS Scram	FPGA	UPS	IA (Fail safe)
	Class 2	ARI	H/W	B/B-DC	
		SLC+RPT	H/W	B/B-AC, DC	Air-cooled DG(B/B)
Reactor Core Cooling	Class 1	RCIC	FPGA	Emergency-DC	HVAC(MCR, RCIC Room)
			H/W (Manual)	Emergency-DC Emergency-AC	EDG HVAC(RCIC Room)
		HPCF	FPGA	Emergency-AC Emergency-DC	EDG RCW/RSW HVAC(MCR, HPCF Room)
			H/W (Manual)	Emergency-AC	EDG RCW/RSW, HVAC(HPCF Room)
	Class 2	Alternative SRV	B/B-H/W	B/B-DC	
		FLSS	B/B-H/W	B/B-AC	Air-cooled DG(B/B)
Long-term Heat Removal	Class 1	RHR	FPGA	Emergency-AC Emergency-DC	EDG RCW/RSW HVAC(MCR, RHR Room)
			H/W (Manual) (Only RHR(A)(B))	Emergency-AC	EDG RCW/RSW, HVAC(RHR Room)
	Class 2	Containment Venting	H/W	B/B-AC	N_2 Accumulator(Local)

Dependencies :
- EDG(H/W) · · · E-DC, RCW/RSW(Cooling)(Note: Loss of cooling does not lead to EDG failure immediately), E-HVAC(EDG Room)
- RCW/RSW(FPGA, H/W) · · · E-AC(Pump, Valve, Platform(H/W)), E-DC(Platform(FPGA)), E-HVAC(RCW/RSW Room)
- Emergency HVAC(FPGA) · · · E-AC(Fan, Valve), E-DC(Platform), HECW(MCR) / RCW(ECCS)(Cooling)
  (Note: Loss of HVAC for each ECCS pump room does not lead to each ECCS failure because ECCS pumps are cooled by RCW as needed.)



FF: Frequent Fault

Figure 26.2-1: Event Tree Analysis for Frequent Design Basis Faults except LOOP

LOOP	Emergency-DC	Emergency-AC	Core cooling	Fault Sequence Group	Example of Postulated Fault Sequences
		EDG	RCIC/HPCF	Safe Shutdown	—
			RCIC/FLSS	Station Blackout (SBO)	• LOOP with CCF of Emergency Diesel Generators(EDGs) • LOOP with CCF of emergency AC-bus
			FLSS	SBO with RCIC failure (Loss of all ECCS)	• Above faults + failure of RCIC control circuit • Above faults + failure of RCIC pump/turbine
				SBO with loss of all DC power (Loss of all ECCS)	• LOOP with CCF of DC-bus

Figure 26.2-2: Event Tree Analysis for LOOP

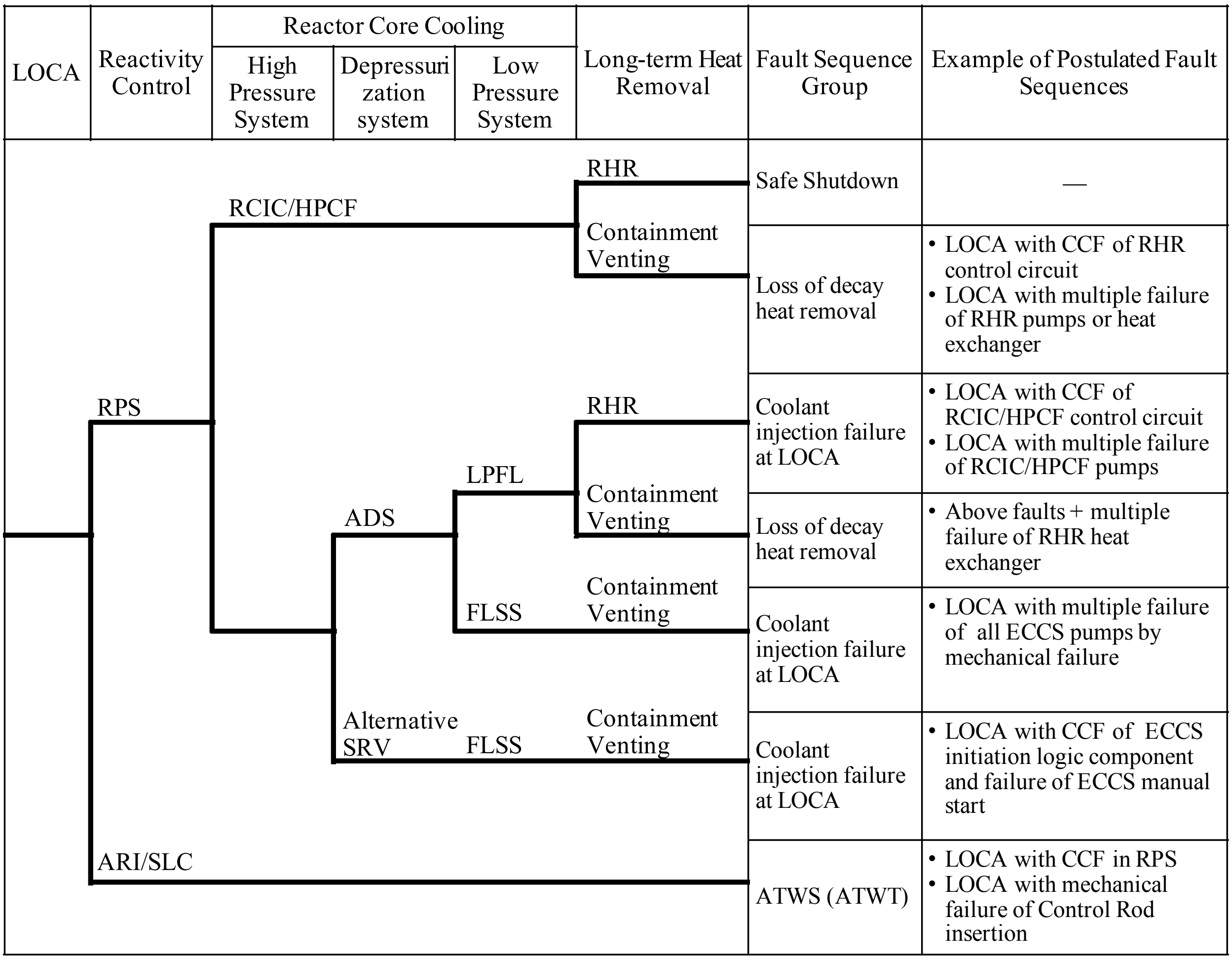


Figure 26.2-3: Event Tree Analysis for LOCA

26.2.3 Acceptance Criteria for Beyond Design Basis Faults

The acceptance criteria for beyond design basis faults are shown below.

• Target
The effective dose received by any person arising from a beyond design basis fault sequence shall not exceed 100 mSv (Off-site) or 500mSv (On-site) according to Target 4 of HSE SAPs.

The following four intermediate targets are applied to meet the target above. These intermediate targets indicate that the excess embrittlement of fuel cladding is prevented, and the reactor coolant pressure boundary and reactor containment boundary are maintained.

(1) The calculated maximum fuel cladding temperature shall not exceed l200 deg. C.
(2) The calculated total oxidation of the fuel cladding shall not exceed 15% of the total cladding thickness before oxidation.
(3) Pressure on the reactor coolant pressure boundary shall be maintained below 120% of the maximum allowable working pressure.
(4) Pressure on the reactor containment boundary shall be maintained below the limiting pressure.
(5) Temperature on the reactor containment boundary shall be maintained below the limiting temperature.

26.2.4 Analysis Code

SAFER, MAAP and TRACG are planned for use in BDBA depending on fault sequences. Description of the computer code SAFER, TRACG and MAAP are described in the reference documents [Ref-2, Ref-3, Ref-4].

26.2.5 Analysis Results

The thermal hydraulic analysis and dose evaluation for beyond design basis faults will be performed in Step3. Analysis results will be described in the next updated PCSR.

26.2.6 Conclusion

The beyond design basis faults to be assessed are identified from Event Tree Analysis and key safety system CCF analysis. In addition, ISLOCA is considered as a beyond design basis fault. Thermal hydraulic analysis and dose evaluation for beyond design basis faults will be carried out to demonstrate the robustness of safety provisions on UK ABWR in Step 3.

26.2.7 References

[Ref-1] Hitachi-GE Nuclear Energy, Ltd., *"Topic Report on Fault Assessment*", GA91-9201-0001-00022 (UE-GD-0071) Rev.1, August 2014.

[Ref-2] Hitachi-GE Nuclear Energy, Ltd., "*Description of SAFER Code*", GA91-9201-0003-00143 (AE-GD-0183) Rev.0, August 2014.

[Ref-3] Hitachi-GE Nuclear Energy, Ltd., "*Description of the Computer Code TRACG*", GA91-9201-0003-00005 (UE-GD-0156) Rev.0, August 2014.

[Ref-4] Hitachi-GE Nuclear Energy, Ltd., "*Topic Report on physics models and benchmarking of MAAP code*", GA91-9201-0001-00035 (AE-GD-0144) Rev.A, May 2014.

26.3 Severe Accident Analysis

26.3.1 Introduction

The UK ABWR has a variety of engineered features, strategies and procedures for responding to design-basis accidents and for beyond design-basis accidents. In order to confirm the performance of the UK ABWR in severe accidents, evaluations using severe accident analysis code, the MAAP code, are carried out. The aims of the severe accident analyses are classified into the following 4 categories (The detail about PSA is described in chapter 25).

・Effectiveness evaluations of engineered features, strategies and procedures
・Success criteria evaluations using Level 1PSA and Level 2PSA
・Quantitative evaluations of the containment failure probability using Level 2PSA
・Quantitative evaluations of the source term using Level 3PSA

The objective of this section is to show the basic concept of the UK ABWR severe accident analysis, the review of SA analysis code, and the analysis method and results with respect to the 4 aims mentioned above.

26.3.2 Severe Accident Progression

Severe accident progression is categorized into three phases. The first one is called the in-core phase, which is defined as the period until core support plate failure. The second one is called the lower plenum phase, which is defined as the period until reactor pressure vessel failure. And the third one is called the ex-vessel phase. The following is the outline of severe accident progression for each phase [Ref-1].

(1) In-core phase

If a transient event occurs during rated power operations, the control rods are automatically inserted into the core and the reactor will scram. In the very unlikely event that the scram fails (ATWS event), the control rods and the standby liquid control system (SLC), which possesses diverse means for ensuring sub criticality of the core, are used to lead the core to sub-criticality. If the scram is successful, the reactor becomes sub critical, and reactor power promptly reduces to decay heat level which is below several percent of the rated power. After reactor power decreases to the decay heat level, the core is cooled by the main condenser of the balance of the plant (BOP) system or the emergency core cooling system (ECCS). However, if the both systems fail, water in the core is boiled off by the decay heat and the reactor water level gradually decreases. The steam generated in the core is lead to the suppression pool in the primary containment vessel (PCV) via the safety relief valves (SRV), and is condensed there.

If an operator recognizes a drop in reactor water level, depressurization of RPV and low pressure injection water from alternative water injection system are expected to keep reactor water level high enough to cool the core. However, if these fail (although it has a low probability because the UK ABWR has many safety relief valves to depress the RPV), reactor water level falls below the top of active fuel and fuel is exposed above the water surface. When the reactor water level falls below the top of active fuel, the fuel above the water level is cooled by the steam flowing. The forced convection heat transfer of steam generally decreases with the decreasing of the water level. Therefore it doesn't completely remove the decay heat generated from the fuel, and temperatures of the fuel and cladding above the water level gradually rise. If fuel temperature rises, the channel box and control rod temperatures rise due to the radiation from the fuel. If fuel and cladding temperatures exceed about 900 degrees Celsius, hydrogen is generated by the steam oxidation reaction of the stainless steel in the control rod structure and Zr in the channel box and cladding. The high temperature of cladding causes cladding collapse by creep or melting, and high pressure in the cladding causes cladding collapse by ballooning due to pressure difference between inner pressure and outer pressure of the cladding. The high temperature of the fuel pellet causes fission product gas release from the pellet to the pellet-cladding gap.

Due to eutectic reactions, neutron absorber (B_4C) in the control rod and fuel pellet (UO_2) might melt before the melting temperature of those materials is reached. Damaged fuel melts and migrates to the lower regions of the core due to gravity (this is called relocation). Mixture of molten fuel and core structure materials are called corium, and solidified corium is called crust. In relocation, blockage of the steam flow path due to the crust formation has to be considered. If the core support plate fails due to core melt that migrated to the lower regions during relocation, the reactor core melt falls down to the lower plenum.

(2) Lower plenum phase

If the core support plate is damaged, corium influxes to the lower plenum as a jet via the damaged opening of the core support plate. Since there is no heat source in the lower plenum before the corium flows into it, the lower plenum is generally filled with water when the corium flows into. If there is a water pool in the lower plenum, reactor core melt particles are generated due to jet break-up effect and fuel coolant interaction (In-Vessel FCI) occurs. At the same time hydrogen is produced due to the steam oxidation reaction at the surface of the corium with water. However, with regard to the PCV damage due to FCI within the pressure vessel, most experts acknowledge that the possibility of the PCV damage is extremely low, and that, from a risk perspective, they have been resolved [Ref-2].

The corium accumulated in the lower plenum forms the layer of particulate corium (particulate debris bed), metal layer with the low density, and corium which is surrounded by the solidified corium (crust). In this condition, heat transfer between the RPV lower head and the adjacent crust is impeded by the gap between them. The gap is expanded by creep deformation of the RPV lower head by over-heating from the crust. As a result, corium cooling is maintained due to the surrounding water and temperature rise in the RPV lower head is suppressed by the influx of cooling water into the gap between the deformed RPV lower head and the crust (gap cooling phenomena). Therefore, the lower plenum cooling water can effectively remove heat from the particulate debris bed, metal layer, and the corium. After the water in the lower plenum has been depleted, the corium is cooled by the radiation heat transfer, and the temperature of the structural materials of the upper region of the lower plenum rises due to the radiation heat transfer. However, as radiation heat transfer rate is too small to cool the corium, the structures in the lower plenum, such as the RPV lower head, CRD guide tube, CRD housing tube, instrumentation tube and housing, are over-heated and their temperature increases.

After the water in the lower plenum has been depleted, the structures in the lower plenum may be damaged. After the RPV lower head has failed, the corium discharges into the lower plenum through the failure opening. The failure opening is gradually expanded by the high temperature outflow of the corium (ablation effect), and corium flow rate may increase.

(3) Ex-vessel phase

If depressurization of RPV fails and RPV failure occurs at high pressure condition, high-pressure melt ejection (HPME) could occur. At this time, the corium jet is torn by high velocity gas flow and fragmented corium droplets are generated. If the fragmented corium droplets are small enough to be able to ignore inertial impaction, the fragmented corium droplet may move from the lower D/W to the other PCV regions. Because the corium droplets have a large surface area per volume, instantaneous heat transfer from the corium to the gas increases. Steam oxidation reaction at the surface of the corium droplet may occur. As the result, PCV may be pressurized and fail rapidly. These phenomena are called Direct Containment Heating (DCH).

The probability of the DCH occurring is extremely small for the UK ABWR because of two reasons. The first reason is that UK ABWR has many diverse procedures to depressurize the RPV. The second reason is the PCV of UK ABWR is designed such that substantial debris dispersal hardly occurs in the geometry of UK ABWR containment. However, if DCH occurs, the possibility of an early large amount of fission product release would be high and there would be the possibility of high exposure to the public [Ref-4]. Therefore, it is important to evaluate the soundness of the PCV due to DCH because the phenomenon has some uncertainties. The evaluation is given in 26.3.6.

In the case of the UK ABWR, alternative water injection to the lower D/W before RPV fails is planned as an accident management and the corium fallen into the lower D/W can be cooled by the pre-injected water effectively. If the water is present in the lower drywell, surface of the corium jet is torn and the particulate debris bed is generated in the lower D/W. In this condition, fluid-corium interaction (FCI) could occur in the lower D/W.

However, steam explosion hardly occurs for two reasons. The first reason is the energy transfer from the corium to the water is limited by the small amount of water. The second reason is the high vapour film stability at the surface of the corium particle because saturation water can evaporate easily. Therefore, steam explosion with this condition doesn't need to be considered. Even if there is a large amount of water on the floor of the lower D/W, steam explosion by the contact of water and the corium does not occur easily. For ex-vessel FCI, a lot of large-scale experiments and studies have been conducted since the 1970's. In FARO test, KROTOS test and COTELS test, which are experiments using UO_2 corium, steam explosion does not occur except for some cases which send shock waves to the corium from outside as an external trigger [Ref-3] [Ref-4].

Even if steam explosion occurs, the possibility that discharged mechanical energy directly affects the PCV boundary is relatively small because the lower D/W of UK ABWR is surrounded by thick concrete structures. However, there is some possibility that the pedestal structure which supports the RPV may be damaged due to steam explosion in the lower D/W. If the RPV is shifted to the lower due to the pedestal wall break, piping which connect the RPV and the PCV penetration might tear the PCV penetration and the PCV may fail. Therefore, it is necessary to evaluate the soundness of the PCV due to an Ex-vessel steam explosion in consideration of some uncertainties for the steam explosion phenomena. The evaluation is described in 26.3.6.

If water injection to the lower D/W fails, the corium that falls to the floor of the lower D/W spreads across the floor area and forms continuous corium (debris bed). Because there is no cooling water, the top surface of the corium is cooled by natural convection heat transfer of gas or by radiation heat transfer. Because these heat transfer removals are generally smaller than decay heat from the corium, MCCI due to the high temperature corium may proceed until the water cooling is provided in the lower D/W. It is important to evaluate the soundness of the PCV due to MCCI because MCCI phenomenon has some uncertainties. The evaluation is described in 26.3.6.

The PCV gets pressurized due to the generated steam after the condensation function of the suppression pool is lost because the suppression pool water temperature reaches the saturated temperature. If depressurization of containment pressure by containment venting fails, the containment overpressure failure occurs. Also, if water injection including spray to PCV fails, the containment overtemperature failure occurs. In the unlikely event that containment leakage or failure due to high pressure and temperature, the fission products are released to the reactor building passing through the suppression pool.

26.3.3 Strategies, Measures and procedures for Severe Accident

26.3.3.1 Severe Accident Management Strategy

Severe Accident (SA) measures are designed to prevent or mitigate failure of nuclear reactor fuel in the core or in the spent fuel pool in a case where an accident which may lead to severe accident occurs because of failure of Design Basis Accident (DBA) measures. They are also designed to prevent containment failure and a large amount of fission product release to environment as an alternative of DBA measures in a severe accident. Furthermore, they are designed to prevent hydrogen burning in reactor building by controlling the accumulation of hydrogen in a case where severe fuel damage occurs. For UK ABWR, accident management strategy, measures and procedures to prevent or mitigate severe accident progression indicated in 26.3.2 are as follows.

(1) Core cooling strategy

When loss of water injection system or Loss Of Coolant Accident (LOCA) occur, severe accident progresses as described in 26.3.2 if DBA measures lose function. In the severe accident management strategy for UK ABWR, core cooling using various alternative low pressure injection system is prioritized in the initial stage of the accident and as a results occurrence of RPV failure is prevented. In the very unlikely event that RPV pressure maintains high and RPV depressurization function fails, alternative depressurization function is sure to be actuated and water injection into RPV by alternative low pressure injection system is accomplished. With these countermeasures core damage is prevented in most of accident scenarios excepting certain scenarios in which RPV water level decrease rapidly like large break LOCA. Even if core damage occurs, core is cooled with core reflooding and likely prevented to transfer lower plenum phase described in 26.3.2. On the basis of strategy described above, procedures are prepared so that operator can appropriately respond to accident based on information obtained from SA instrumentation that can measure, monitor, and record the plant data in severe accident.

(2) Containment control strategy

Containment control strategy has two parts. The first one is flooding strategy whose purpose is to maintain containment boundary integrity. The second one is heat removal strategy whose purpose is to remove the heat from containment. On the basis of strategy described below, procedures are prepared, therefore operator can appropriately respond to accident based on information obtained from SA instrumentation which measure, monitor, and record the plant data in severe accident.

(a) Flooding strategy for containment

In a case that core cooling successes, the PCV gets pressurized due to the generated steam and hydrogen in RPV and after the condensation function of the suppression pool is lost because the suppression pool water temperature reaches the saturated temperature. At the same time, steam and fission product released into PCV increase PCV temperature. If PCV spray with DBA measures fails, Drywell spray with alternative water injection system control temperature and pressure of PCV and prevent the occurrence of PCV overtemperature failure and overpressure failure.

In a case that core cooling fails or delays to start, the core condition transfers to lower plenum phase described in 26.3.2 and melting core heats RPV lower head. In severe accident management for UK ABWR alternative water injection system supply water into lower D/W before RPV fails. Furthermore, alternative depressurization function is actuated before RPV failure to prevent the occurrence of high-pressure melt ejection (HPME) described in 26.3.2.

After RPV failure, the corium ejected from RPV is spread on the floor of the lower D/W and cooled by the pool water. The amount of decay heat removal mainly depends on the surface area of the spreading corium. UK ABWR has enough wide the floor area of the lower D/W to remove the decay heat from the corium and so the depth of erosion on the floor and wall of the lower D/W is suppressed [Ref-5]. As described above, various alternative low pressure injection system can supply water into lower D/W and maintain flooding debris. In the very unlikely event that water injection into lower D/W fails or delays to start, passive equipment actuated by an atmosphere temperature of lower D/W supplies water into lower D/W. Containment control strategy after RPV failure is the same as the case that core cooling successes. Eventually DBA equipment is recovered and cool suppression pool water and core debris in RPV or PCV instead of alternative injection system.

(b) Heat removal strategy for containment

In a case that Residual Heat Removal system (RHR) fails, Containment venting removes decay heat by releasing non-condensable gases and steam from PCV into environment. And also containment venting prevents PCV overpressure failure. In a case that operator fails to manually actuate PCV venting, a rupture disk placed on venting line opens automatically at the rupture disk setpoint pressure. If RHR in available but reactor auxiliary cooling system is failed due to failure of water intake, alternate reactor auxiliary cooling remove decay heat from PCV using RHR.

(3) Fission product release control strategy

As described in “Core cooling strategy” and “Containment control strategy”, basic strategy for fission product release control is to prevent and mitigate SA progression and so maintain containment boundary integrity. In a case that RHR or alternate reactor auxiliary cooling fail, gases including fission product are eventually released to environment by containment venting. In severe accident management strategy for UK ABWR, the timing of PCV venting is delayed by containment pressure control as much as possible and PCV venting at suppression chamber is prioritized because the fission products are scrubbed as they pass through the suppression pool. In addition, there is containment venting system equipped with filter which can reduce further amount of the fission products released from PCV.

(4) Spent fuel pool cooling strategy

In a case where cooling function and injection function of Spent Fuel Pool (SFP) are failed or water leakage occurs in SFP, alternative water injection system inject water into SFP, cool the fuels in the SFP and keep the water level high. In the very unlikely event that is impossible to maintain water level of SFP due to large break of SFP, SFP spray cool spent fuel, prevent severe spent fuel damage and so suppress the amount of fission product release from SFP.

(5) Hydrogen control strategy

In UK ABWR, as hydrogen management facility for containment, nitrogen supply system is installed. Since containment atmosphere is filled with inert nitrogen using this system, hydrogen burning or explosion never occur in the containment even if severe accident occurs and hydrogen is generated due to water-metal reaction in the containment. Also, considering lessons learnt from the accident at Fukushima Daiichi Nuclear Power Station, alternative water injection system directly supply coolant water to reactor well to reduce risk of hydrogen leakage from the containment head flange and containment venting releases hydrogen gases from PCV, and so decrease hydrogen leakage from containment to reactor building. In a case that hydrogen leaks to reactor building, SGTS control increasing hydrogen in reactor building by discharging hydrogen gas from reactor building to environment during severe accident.

26.3.3.2 Basic requirements for Severe Accident measures

SA measures are designed based on the requests below.

(1) SA measures are designed to function as required with sufficient reliability under severe accident condition.

(2) SA measures are designed to be controlled for sure under severe accident condition.

(3) SA measures are designed to be capable of inspection and testing.

(4) SA measures are designed so as not to cause any detrimental impact on other equipment.

(5) SA measures are designed to have sufficient capacity for severe accidents.

(6) SA measures are designed that diversity is considered as much as possible for the Design Basis Accident Measures to be substituted.

(7) SA measures are designed to have a resistance against externals.

(8) SA measures that can be moved are designed to connect easily and surely to permanent equipment, and multiple connections are prepared with appropriate spatial dispersion to avoid disconnection due to common modes.

26.3.3.3 Basic Design of Severe Accident Measures

(1) Reactor Depressurization facility

Reactor Depressurization facility is designed to depressurize reactor coolant pressure boundary by opening safety relief valves with alternative driving source in the loss of DC power source condition. This facility is equipped with alternative power source as permanent equipment and is equipped with storage battery for safety relief valves and nitrogen supply equipment as transportable equipment. The detail about alternative depressurization function is described in chapter 16.5.

(2) Core Flooder system

Core flooder system is designed to inject water to reactor core from low pressure water supply system and prevent significant core damage and containment failure even if all emergency core cooling systems are failed. Low pressure water supply system is equipped with low pressure water supply pumps as permanent equipment and mobile pumps as transportable equipment. The detail design of the flooder system is described in chapter 16.5.

(3) Containment heat removal facilities

Containment heat removal facilities are equipped with alternate reactor auxiliary cooling system and containment venting system.

a. Alternate reactor auxiliary cooling system is designed to remove the heat from reactor residual heat removal system in a case where reactor auxiliary cooling system is failed due to failure of water intake. The detail design of the alternative heat exchange facility is described in PCSR chapter 16.5.

b. Containment venting system is designed to be actuated manually and release non-condensable gases and steam to prevent damage to the PCV for overpressure. It is also designed to filter the radioactive iodine and long-half life fission products generated during severe accident in the PCV through the filter vent device. The detail design of the containment venting system is described in PCSR chapter 16.5.

(4) Containment Depressurization facility

Containment depressurization facility is equipped with containment venting system. Containment venting system is designed to release non-condensable gases and steam through the main stack to prevent damage to the PCV for overpressure. It is also designed to filter the radioactive iodine and long-half life fission products (Cs) generated during severe accident in the PCV through the filter vent device. The detail design of the containment venting system is described in PCSR chapter 16.5.

(5) Containment Flooder system

Containment flooder system is equipped with alternate containment spray, lower drywell injection system, and lower drywell flooder system.

a. Alternate containment spray is designed to inject water into containment and control containment pressure and temperature to prevent containment failure in a case where containment cooling by residual heat removal system is failed. The detail design of the flooder system is described in PCSR chapter 16.5.

b. Lower drywell injection system is designed to inject water into lower drywell to prevent molten core concrete interaction in a case where reactor vessel is failed and molten core is discharged into lower drywell. The detail design of the flooder system is described in PCSR chapter 16.5.

c. Lower drywell flooder is installed to passively inject water into lower drywell to prevent molten core concrete interaction in a case where lower drywell injection system is failed. The detail design of the lower drywell flooder system is described in PCSR chapter 16.5.

(6) Hydrogen Control facilities

As hydrogen control facility for containment, nitrogen supply system is installed in UK ABWR. Since containment atmosphere is filled with inert nitrogen using this system, hydrogen burning or explosion never occur in the containment even if severe accident occurs and hydrogen is generated due to water-metal reaction in the containment. The detail design of the nitrogen supply system is described in PCSR chapter 16.2.

As hydrogen control facility for reactor building, reactor well injection system, containment venting system, and Stand-by Gas Treatment System (SGTS) are installed in UK ABWR.

a. Reactor well injection system is designed to inject water to reactor well, mitigate the hydrogen leakage from containment to reactor building by cooling the top head flange of containment, and prevent hydrogen burning in reactor building. The detail design of the flooder system is described in PCSR chapter 16.5.

b. Containment venting system is designed to release hydrogen gases through the main stack and prevent hydrogen burning in reactor building by decreasing hydrogen leakage from containment to reactor building. The detail design of the containment venting system is described in PCSR chapter 16.5.

c. Stand-by Gas Treatment System (SGTS) is designed to prevent hydrogen burning in reactor building by discharging hydrogen gas from reactor building to environment during severe accident. The detail design of the flooder system is described in PCSR chapter 16.3.

(7) Spent Fuel Pool Flooder System

Spent fuel pool flooder system is designed to inject water into spent fuel pool, cool the fuels in the spent fuel pool and keep the water level high in a case where cooling function and injection function of spent fuel pool are failed or water leakage occurs in spent fuel pool. The detail design of the flooder system is described in PCSR chapter 16.5.

(8) Alternative Generator facility

Alternative generator facility is designed to supply alternative power source to SA engineered features to prevent core damage, containment failure, and fuel damage of spent fuel pool in a case where severe accident occurs due to station blackout. The detail about alternative generator system is described in chapter 15.4.

(9) Severe Accident Instrumentation

a. SA instrumentation is designed to measure, monitor, and record the plant data, which is important to deal with severe accident, such as temperature, pressure, water level and radiation doze rate in containment. The detail design of severe accident instrumentation is described in chapter 14.6.5.

b. SA instrumentation is designed to measure, monitor, and record the concentration of fission product and radiation doze rate in nuclear power plant and the surrounding area at the time of severe accident. The detail design of severe accident instrumentation is described in chapter 14.6.5.

(10) Alternative Water Source

Alternative water source is designed to secure an adequate supply of water enough to deal with severe accident. The detail design of alternative water source is described in PCSR chapter 16.5.

26.3.3.4 Severe Accident Management

For UK ABWR, various levels of accident management procedures will be prepared as follows;

・Abnormal Operation Procedures (AOPs)
・Emergency Operation Procedures (EOPs)
・Severe Accident Management Guidelines (SAMGs)

AOPs are used for postulated events that have been analyzed and discussed in the design based analyses and limited to single initiating events follows by success operation of safety systems designed to respond to those events. EOPs are symptom-based procedures limited before core damage and respond to multiple failure induced severe accidents which are low frequency scenarios. SAMGs provide guidelines for preventing and mitigating accident scenarios in which severe core damage occurred, reactor pressure vessel fails and containment integrity is challenged by the accident progression. The results of the severe accident analyses presented in section 26.3.5 will be appropriately incorporated into the AMG.

a. Establish equipment for cooling the reactor and prevent severe core damage

b. Establish equipment for cooling the reactor and prevent reactor pressure vessel failure

c. After core damage or reactor pressure vessel failure, establish equipment for preventing containment vessel failure and minimize offsite releases

26.3.3.5 Response to the Accident at Fukushima

The design of the UK ABWR has made use of the lessons learnt from the accident at Fukushima Daiichi to make provision for the flexible management of any external event beyond the design basis. The detail about countermeasures for Fukushima accident is described in chapter 28.4.

26.3.4 SA Analysis Code

The Modular Accident Analysis Program (MAAP) is a computer code that is widely used by nuclear utilities and research organizations to predict the progression of light water reactor accidents. The MAAP code is used as severe accident code for UK ABWR. The scope of MAAP code for UK ABWR is classified into the following categories.

・Effectiveness evaluations of engineered features, strategies and procedures
・Success criteria evaluations using Level 1PSA and Level 2PSA
・Quantitative evaluations of the containment failure probability using Level 2PSA
・Quantitative evaluations of the source term using Level 3PSA

Considering the applicable scope of MAAP code for UK ABWR, it is important to confirm that the MAAP code has physics models that could occur during an accident of level 1 and level 2 are incorporated. And it is also important that these models have been properly validated by experiments and plant accidents. Therefore, MAAP physics models and MAAP validation were reviewed.

(a) MAAP Physics model

Based on the scope of MAAP code, the incorporation of physics models into the MAAP code was reviewed. As the results, it is found that the MAAP code has a variety of physics models that should be considered during severe accident. For example, the MAAP code incorporates the physics models of gas and water flow, natural circulation, steam evaporation and condensation, boiling, critical flow, conduction, convection and radiation heat transfer for level 1 phenomena. And the MAAP code also incorporates the physics models of cladding oxidation and hydrogen evolution, core material eutectic formation, core relocation, lower head-core debris dynamics, failure of vessel penetrations and the lower head, debris entrainment, debris-concrete interactions, and fission product release, transport, and deposition for level 2 phenomena [Ref-6].

(b) MAAP validation

It was also reviewed that MAAP physics models have been validated properly especially for BWRs. As the benchmark of separate effect experiment or integral effect experiment, the MAAP models related to thermal hydraulic behavior of RPV, thermal hydraulic behavior of PCV, damaged fuel behavior, in-core fission product release, boiled-up liquid level, breaking-up of corium jet, and molten core concrete interaction has been benchmarked using a variety of experiments. Comparisons between the MAAP results and the test data show good agreement in all the benchmark [Ref-6] [Ref-10]. As the benchmark of overall plant behavior, the accident of Fukushima Dai-ichi nuclear power plant and Three Mile Island Unit 2 has been simulated by the MAAP code. As the results, the MAAP code provides a reasonable characterization of the system response not only the PWR plant [Ref-6] but also the BWR plant [Ref-7] [Ref-8] [Ref-9]. Therefore, it is apparent that the MAAP code has the ability to evaluate severe accident of UK ABWR, which has similar design concept and similar configurations of BWRs.

26.3.5 Severe Accident Analysis

In Level 1 PSA the various plant responses to an event that performs plant operation are modeled. The plant response paths are called accident sequences. There are numerous accident sequences for a given initiating event. The various accident sequences result from whether plant systems operate properly or fail and what actions operators take. Some accident sequences will result in a safe recovery and some will result in reactor core damage. In preliminary severe accident analysis, representative accident sequences are evaluated to discuss the success criteria in PSA. As the results, TUQV, TQUX, LOCA, TW, TC, and TB are selected as the representative accident sequences for UK ABWR. The detail about how to select the representative accident sequences is described in chapter 25.

Figure 26.3-1 and 26.3-2 shows the nodalization of UK ABWR. The Analysis results will be described in the next updated PCSR [Ref-11].

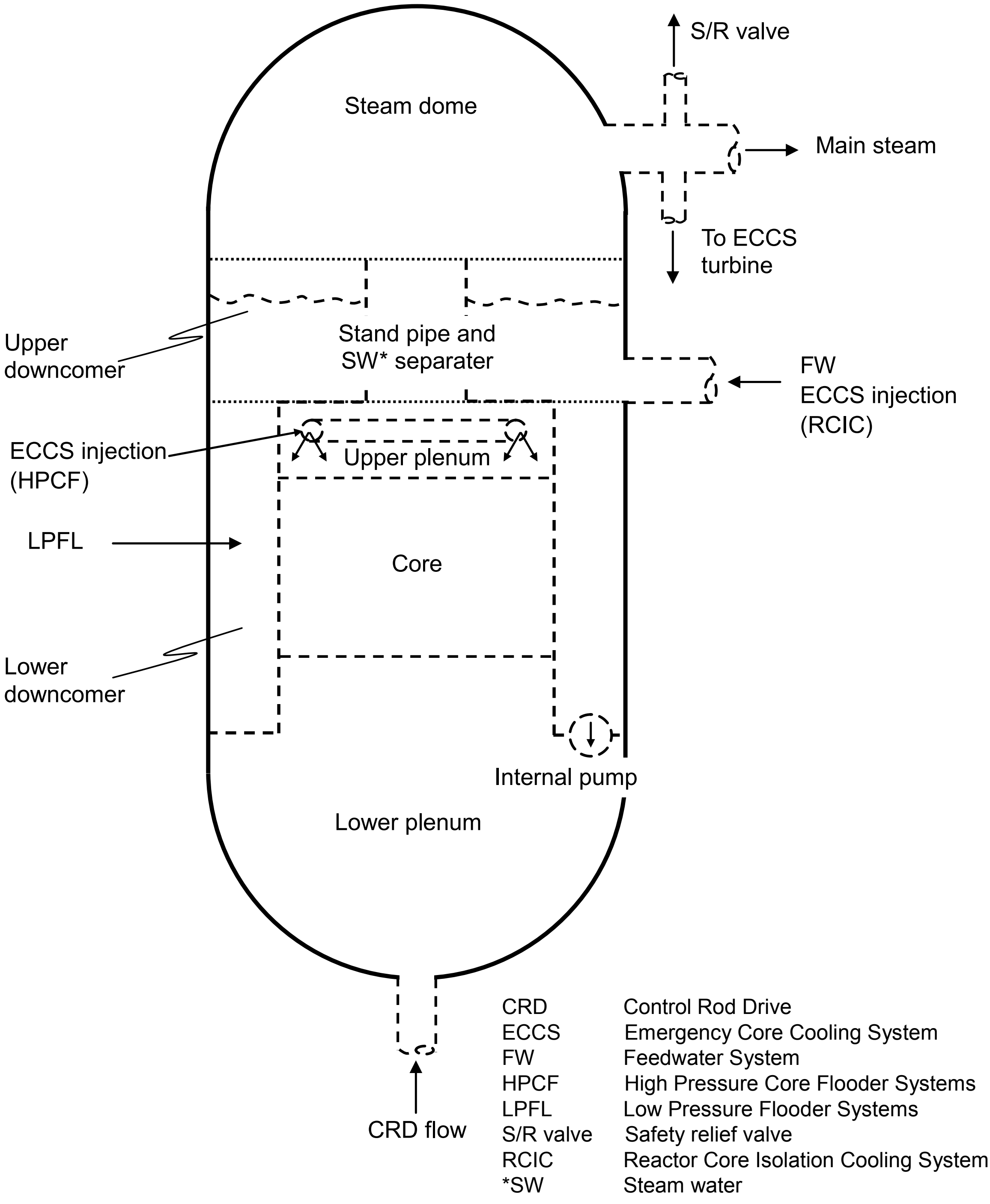


Figure 26.3-1: UK ABWR Primary System Model

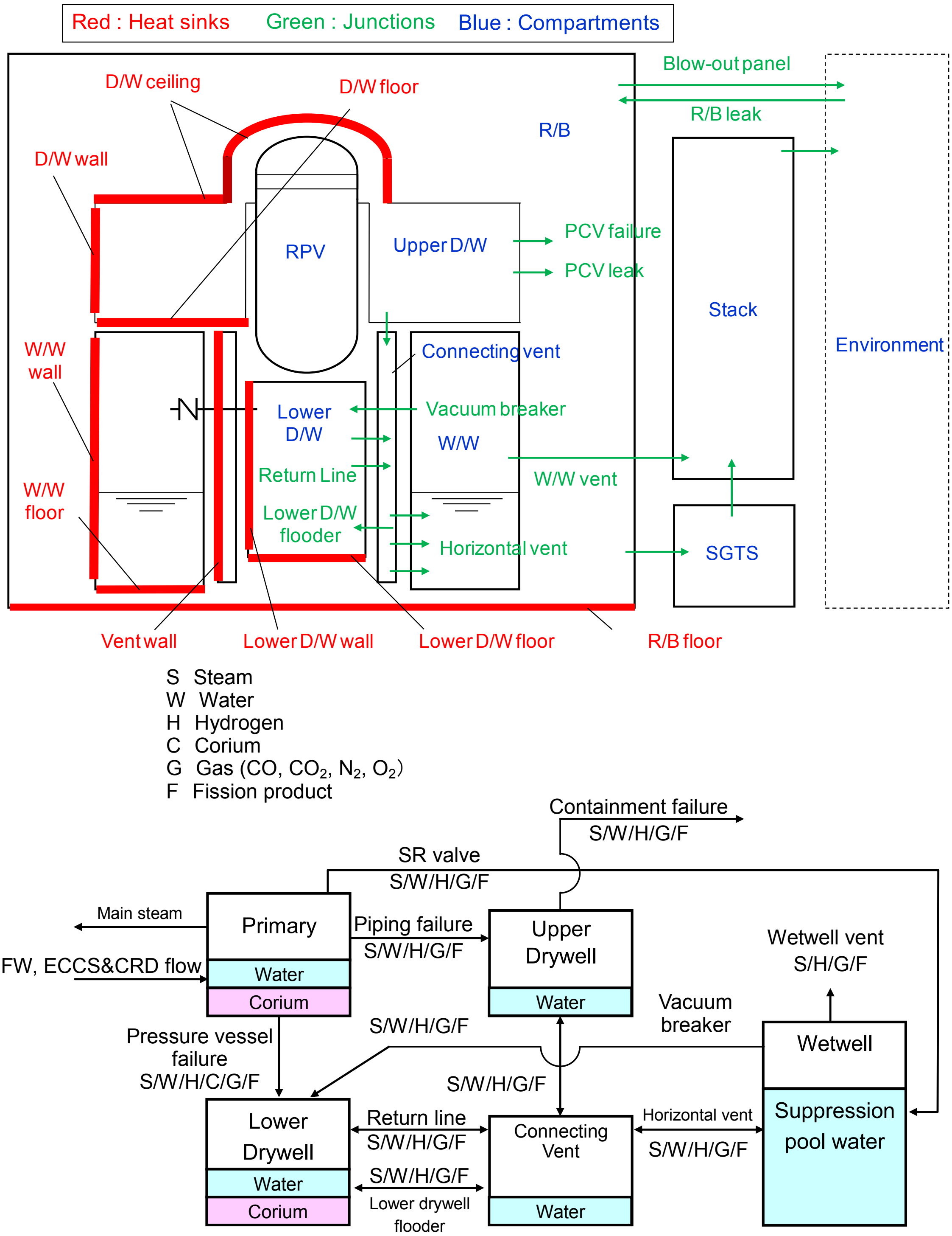


Figure 26.3-2: UK ABWR Containment Vessel and R/B Model

26.3.6 PCV Failure Probability Evaluation

The failure probabilities for three failure modes, molten core concrete interaction (MCCI), fuel coolant interaction (FCI) and direct containment heating (DCH), are evaluated considering probabilistic density of dominant parameters for each failure mode. They are not directly used as the branching probabilities of these failure modes for level 2 PSA, but they are used as reference values to determine the branching probabilities. The detail about level 2 PSA is described in chapter 25.

The Analysis results will be described in the next updated PCSR.

26.3.7 Source Term Analysis

The release categories are defined in consideration of containment integrity, release timing, release path, duration, and scrubbing effect, and etc. Containment failure sequences are categorized into 16 groups (tentative) as shown below:

(1) Containment Leakage from D/W (Intact RPV)
(2) Containment Leakage from W/W (Intact RPV)
(3) Containment Leakage from D/W after RPV breach
(4) Containment Leakage from W/W after RPV breach
(5) Containment Venting (without RPV breach)
(6) Containment Venting (with RPV breach)
(7) Early Containment Failure (D/W breach)
(8) Early Containment Failure (W/W breach)
(9) Late Containment Failure (D/W breach)
(10) Late Containment Failure (W/W breach)
(11) Over-temperature Failure
(12) In-vessel FCI
(13) Ex-vessel FCI
(14) Direct Containment Heating
(15) PCV Isolation Failure
(16) MCCI

Source term analysis is performed by MAAP considering a variety of mitigation systems for UK ABWR. Therefore public dose can be reduced effectively. The analysis will be performed in Step 3. Analysis results will be described in the next updated PCSR.

26.3.8 References

[Ref-1] Hitachi-GE Nuclear Energy, Ltd., “Topic Report on physics models and benchmarking of MAAP code”, GA91-9201-0001-00035 Rev.A, May 2014.

[Ref-2] O.Zuchuat, et al., “Steam explosions-induced containment failure studies for SWISS nuclear power plants”, OECD/CSNI specialist meeting, JAERI, Japan, May 1997.

[Ref-3] Yuri Vasilyev, Alexander Kolodeshinivkov and Vladimir Zhdanov, “COTELS Fuel Coolant Interaction Tests under Ex-Vessel Conditions”, JAERI-Conf., 2000-2015.

[Ref-4] Bal Raj Sehgal, “Nuclear Safety in Light Water Reactor -Severe Accident Phenomenology-“, Academic Press, 2012

[Ref-5] Electric Power Research Institute, “ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT”, March 1999.

[Ref-6] Electric Power Research Institute,”MAAP4 – Modular Accident Analysis Program for LWR Power Plants– Computer Code Manual”, December 2013.

[Ref-7] Hideaki Sadamatsu, et al., “The Accident Analysis for Unit1 at Fukushima Dai-ichi Nuclear Power Station”, N9P0279, The 9th International Topical Meeting on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-9), 2012.

[Ref-8] Shinya Mizokami, et al., “The Accident Analysis for Unit2 at Fukushima Dai-ichi Nuclear Power Sation”, N9P0272, The 9th International Topical Meeting on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-9), 2012.

[Ref-9] Hiromasa Yanagisawa, et al., “The Accident Analysis for Fukushima Dai-ichi Nuclear Power Station Unit 3”, N9P0326, The 9th International Topical Meeting on Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-9), 2012.

[Ref-10] Electric Power Research Institute, “Use of Modular Accident Analysis Program (MAAP) in Suppourt of Post-Fukushima Applications”, June 2013.

[Ref-11] Hitachi-GE Nuclear Energy, Ltd., “Topic Report on Severe Accident Phenomena and Severe Accident analysis”, GA91-9201-0001-00024 Rev.B, May 2014.

26.4 Conclusion

This chapter shows descriptions of BDBA and SAA.

The beyond design basis faults to be assessed are identified from Event Tree Analysis and key safety system CCF analysis. In addition, ISLOCA is considered as a beyond design basis fault. Thermal hydraulic analysis and dose evaluation for beyond design basis faults will be carried out to demonstrate the robustness of safety provisions on UK ABWR in Step 3.

In severe accident analyses, UK ABWR will be designed such that environmental release of any radioactive material from the plant during all modes of operation is acceptably minimized. The UK ABWR has a variety of engineered features, strategies and procedures for responding to design-basis accident and beyond design-basis accident. In order to confirm the performance of the UK ABWR in severe accidents, the following categories of severe accident will be carried out.

・Effectiveness evaluations of engineered features, strategies and procedures
・Success criteria evaluations using Level 1PSA and Level 2PSA
・Quantitative evaluations of the containment failure probability using Level 2PSA
・Quantitative evaluations of the source term using Level 3PSA