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UK ABWR

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UK ABWR Generic Design Assessment

Generic PCSR Chapter 28 : ALARP Evaluation

Hitachi-GE Nuclear Energy, Ltd.

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Executive Summary

This chapter describes the high level arguments and evidence that demonstrate that the risks from the full lifecycle of the UK ABWR are as low as reasonably practicable (ALARP), as required by UK Law. This covers construction, commissioning, operation, maintenance and decommissioning. The level of argument related to construction, commissioning, operation, maintenance and decommissioning is commensurate with the design progress within GDA.

This chapter traces the development of the UK ABWR design from the earliest commercial BWRs, and shows how the best practice at the time was incorporated at each stage of development. It also shows how different options were considered at each stage and how the safety of the design improved with each new generation.

During GDA, the design has also developed, partly as UK Relevant Good Practice (RGP) has been adopted, and also through a process of systematically identifying options with potential risk benefits. Any of these risk reduction measures which have been deemed to be reasonably practicable have been incorporated into the design. This chapter describes the process used for the analysis of reducing risks to a level ALARP and describes some of the more important outcomes. Further details of the outcomes are presented in the ALARP sections of individual PCSR chapters.

Finally, the chapter gives a brief description and rationale for significant design changes that have arisen in the GDA process to reduce risks to levels that are ALARP.

Thus, by tracing the evolution of the design and showing the continual improvement in safety, and by describing the GDA process of identifying further reasonably practicable improvements, the chapter concludes that there are no further reasonably practicable changes that could be made to reduce risks within the context of GDA. However, this does not preclude a future licensee from considering any further reasonably practicable measures that may reduce risk.

Thus it is concluded that the UK ABWR design, as presented in GDA, has been developed to a point where there is high confidence that a future licensee would be able to operate the UK ABWR on a UK site achieving levels of risk which are both tolerable and ALARP..

28.1 Introduction

This chapter summarises the arguments and evidence that demonstrate that risks posed to the workforce and public from the construction, commissioning, operation, maintenance and decommissioning of the UK ABWR are As Low As Reasonably Practicable (ALARP).

28.1.1 Background

Nuclear Safety in the UK is based on the legal requirement that the risks to the workforce and public from a Nuclear Power Plant (or indeed any other industrial facility) should be reduced So Far As Is Reasonably Practicable (SFAIRP) or, equivalently, that risks should be demonstrated to be As Low As Reasonably Practicable. This chapter summarises the demonstration that the design of the UK ABWR reduces risks SFAIRP during construction, commissioning, operation, maintenance and decommissioning, at a level proportionate to the design maturity in GDA. Generally, and this applies throughout all Chapters of this PCSR for the UK ABWR, the acronym ALARP is used in preference to SFAIRP.

28.1.2 Document Structure

Most of the other chapters of the PCSR contain a section that summarises the ALARP justification relevant to that chapter with references to the Topic Reports that provide the detailed ALARP justification.

While individual chapters contain a section on ALARP, these are necessarily limited in their scope. Chapter 28 seeks to provide a high level summary of these individual chapters and also presents an integrated view of nuclear risk that is only possible by addressing together the totality of the structures, systems and components which make up the UK ABWR. An understanding of how the nuclear risk is distributed between all aspects of UK ABWR operation enables a designer, future licensee, and also a regulator to apply a proportionate approach to nuclear safety which has implications for assessing whether potential design changes may, or may not be reasonably practicable.

Section 28.2 outlines the purpose and scope of the chapter.

Section 28.3 traces the development of the ABWR design from the earliest BWR designs showing how safety has been continuously improved as each generation has been developed, culminating in the standard Japanese ABWR Kashiwazaki-Kariwa Units 6 and 7.

Section 28.4 outlines the development of later ABWR plants after Kashiwazaki-Kariwa Units 6 and 7.

The earthquake and tsunami that struck eastern Japan in March 2011 and the severe accident at the Fukushima Dai-ichi BWR plant led to major reviews of nuclear safety in plants around the world. These reviews led to a number of lessons being learned in Japan, the EU and the UK. How these lessons learned have been implemented in the design of the UK ABWR is the subject of Section 28.5.

Section 28.6 outlines the strategy used to demonstrate that the risks associated with the ABWR design in the UK are ALARP. This is based on the implementation of the standard ALARP process generally used in the UK industry and insights from the Probabilistic Safety Assessment (PSA) developed for UK ABWR.

Section 28.7 then describes the development of the UK ABWR design from the most recent designs in Japan, listing the design changes that have been identified from the need to implement UK and International good practice, from the application of a systematic ALARP analysis and from the insights from the PSA.

Section 28.8 provides the overall conclusions.

28.2 Purpose and Scope

28.2.1 Purpose

The purpose of Chapter 28 is to provide, at a high level, the arguments and evidence that the risks to the station workforce and the public from the construction, commissioning, operation, maintenance and decommissioning of the UK ABWR are ALARP. This chapter shows how the development of the design from the earliest BWRs has continually adopted international good practice and explored options to reduce risk, with options deemed reasonably practicable being implemented as the design has developed.

28.2.2 Scope

This chapter discusses the high level approach taken for demonstrating ALARP across all aspects of the design and operation but leaves the detailed safety justification to the relevant PCSR chapters and more usually their supporting documentation (BSCs and TRs) referenced in each chapter.

The approach to demonstrating that risks from the construction, commissioning, operation, maintenance and decommissioning of the UK ABWR are ALARP is based on a well-defined process that begins with the adoption of RGP and then continually looks for and identifies options to reduce risks. Any options that are determined to be reasonably practicable, that is, whose costs are not grossly disproportionate to the value of the risk reduction potentially achieved by their implementation, are then implemented.

This process of continuous improvement through analysing design options and then selecting reasonably practicable solutions has been followed throughout the development of the UK ABWR. The process of using options analysis started from the earliest BWR designs and continued during the development of the standard ABWR in the 1980s and 1990s and has continued in subsequent developments to the present day.

The standard design of ABWR has developed through an evolutionary process where each design change has sought to simplify and improve earlier BWR designs and to reduce safety risks by using the best practice available at the time. This chapter traces the development of the standard design from earlier BWR designs to Kashiwazaki-Kariwa Units 6 and 7, the first ABWRs to operate in Japan and subsequent developments of the Japanese ABWR - Shika Unit 2, Shimane Unit 3 and Ohma Unit 1. This chapter also discusses development of a wide range of specific options adopted in response to lessons learned from the accident in Fukushima in March 2011.

During the GDA process, further options have been considered across all technical areas to reflect UK Relevant Good Practice and from the ALARP process and have resulted in a number of important design changes. Decisions taken to implement these design changes have contributed to the development of a design for the UK ABWR whose risks are judged to be ALARP. Significant design changes agreed during GDA are discussed later in this chapter. It is also important to note that this chapter does not provide detailed information on how to conduct an ALARP analysis. It focuses instead on the evolution of the ABWR and the use of international good practice at different stages of its evolution. It is also important to note that the claims, arguments and evidence to support the GDA design described in this chapter are included in other relevant chapters of the PCSR and their supporting evidence in documents such as BSCs and TRs.

With regard to the environmental and security aspects of the UK ABWR design, please refer to Chapter 1: Introduction, for links to GEP and CSA documentation. Where required, specific GEP references are included in specific sections of PCSR chapters such as in the Radioactive Waste Management, Radiation Protection and Decommissioning chapters.

28.3 Development of the Standard ABWR

The BWR is based on two fundamental principles that distinguish it from Pressurised Water Reactor (PWR) nuclear power plants:

(1) Bulk boiling of water occurs in the reactor core, and
(2) Steam produced from boiling in the reactor core is sent directly to the turbine that is used to turn a generator to produce electricity.

The principles of a BWR were first confirmed in the 1950s with the construction and operation of prototype plants at Argonne National Laboratory in Illinois (USA) - the Experimental Boiling Water Reactor (EBWR) and at GE's Vallecitos facility in California (USA) - the Vallecitos Boiling Water Reactor (VBWR).

From these early prototypes, commercial BWRs were developed and evolved continuously through several improved product lines with greater levels of simplification, leading to the ABWR, as summarized in Table 28.3-1.

Table 28.3-1 : BWR Product Line Evolution

Product Line	First Commercial Operation Date	Representative Plant/Characteristics
BWR/1	1960	Dresden 1 • Initial commercial-size BWR
BWR/2	1969	Oyster Creek • Plants purchased solely on economics • Large direct cycle plant
BWR/3	1971	Dresden 2 • First jet pump application • Improved Emergency Core Cooling System (ECCS): spray and flood capability
BWR/4	1972	Vermont Yankee • Increased power density (20 percent)
BWR/5	1977	Tokai 2 • Improved ECCS • Valve flow control
BWR/6	1978	Clinton (1987) • Compact control room • Solid-state Reactor Protection System

The development of the Advanced Boiling Water Reactor (ABWR) began in 1978 and was first adopted for the construction of the Kashiwazaki-Kariwa Nuclear Power Station Units 6 and 7 (KK-6 and KK-7) owned and operated by the Tokyo Electric Power Company (TEPCO). The two reactors, KK-6 and KK-7, commenced commercial operation on 7th November 1996 and on 2nd July 1997, respectively. This marked the first commercial operation of the ABWR in the world. The majority of equipment for both the KK-6 turbine island and the KK-7 nuclear island were supplied by Hitachi Ltd.

The ABWR was jointly developed by Japanese electric utilities and plant technology suppliers, including Hitachi Ltd., (Japan) and General Electric Company (GE) (USA). The ABWR design was based on the evolution of and improvements to the conventional BWR technology and therefore represents a significant improvement over conventional BWRs. In addition, the ABWR has attained a favourable status under Japan's BWR Improvement and Standardization Program. Several improvements to ABWR components and facilities were introduced relative to those of conventional BWRs. Extensive factory tests, pre-operation tests and start-up tests have demonstrated that the components and facilities meet their required functionality and corresponding standards.

This section provides generic information about the technical evolution of the ABWR to illustrate how the ABWR design has been developed, starting from conventional BWRs, by the thorough consideration of options to improve its nuclear safety and environmental compatibility.

28.3.1 Genesis of ABWR

This section provides a brief history of the development of the ABWR. It is provided to demonstrate the evolution of the BWR technology through a process of continually reviewing options and then implementing, at the time, the best reasonably practicable solutions to improve the safety of the technology. It also shows that the technical safety and other improvements introduced in the evolution of the BWR through to the ABWR are similar to many of the developments in the fleet of UK reactors covering the same period on topics such as enhancing the emergency cooling systems, improving the design of the main control rooms, introducing digital protection systems, strengthening the on-site power supply systems against station blackout (SBO) events and improvements to the design and safety justification of major components such as the reactor pressure vessel.

The ABWR design was based on BWR construction and operating experience in Japan, U.S. and Europe and was jointly developed by the BWR suppliers as a next generation BWR. Originally, an Advanced Engineering Team (AET) was formed in the late 1970s to explore key design options and to build on best experiences and lessons learnt.

Participation by the Tokyo Electric Power Company (TEPCO) and five other Japanese utilities (Chubu Electric Power Company, Chugoku Electric Power Company, Hokuriku Electric Power Company, Tohoku Electric Power Company, and the Japan Atomic Power Company) during the 1980s brought a large amount of operating experience with conventional BWRs to bear on the design process, particularly leading to the identification of options to improve operability and reduce risk.

In 1987, TEPCO announced its decision to construct two ABWR units, Unit 6 (KK-6) and Unit 7 (KK-7), at the Kashiwazaki-Kariwa Nuclear Power Station in Japan. In this project, the two units were supplied by a joint venture of GE, Hitachi Ltd. and Toshiba Co. Ltd.

Also in 1987, the U.S. Nuclear Regulatory Commission (US NRC) began its technical review of the ABWR under the Standard Design Certification programme defined in 10CFR Part 52. The US NRC issued a final decision certifying the design to the General Electric Company on 12th May 1997. GE-Hitachi Nuclear Energy and Toshiba, Co. submitted ABWR design certification renewal applications to the US NRC in 2010.

28.3.2 BWR Safety System Evolution Leading up to Standard ABWR

Historically, the Emergency Core Cooling Systems (ECCSs) provided in a particular product line were dependent on the extant regulations during that period of time.

BWRs of the first generation were designed without an ECCS, but did make use of highly reliable feedwater systems. With the BWR/2 product line, concerns were raised with respect to providing adequate core cooling during a Loss-of-Coolant-Accident (LOCA) and preventing a core meltdown that would threaten containment integrity. It was required that all nuclear plants must have an ECCS. A separate ECCS was included in all BWR/2s, and ECCSs were also retrofitted to the BWR/1s.

For the BWR/3 and BWR/4 product lines, two divisions of ECCS and an Automatic Depressurization System (ADS) were carried forward from the BWR/2. The two divisions contain two ECCS pumps, each consisting of Residual Heat Removal (RHR) pumps that are also used for Low Pressure Coolant Injection (LPCI). Low Pressure Core Spray (LPCS) and steam turbine driven High Pressure Core Injection (HPCI) systems were also included in the ECCS network. Jet pump designs introduced in the BWR/3 eliminated the large recirculation pipe connections to the bottom head of the Reactor Pressure Vessel, and it became possible to re-flood the core region of the vessel up to the top of the jet pumps.

At the time the BWR/2, BWR/3 and BWR/4 ECCSs were designed, there were no regulations defining the ECCSs performance acceptance criteria. In 1973, US 10CFR50.46 was issued, which defined the ECCSs performance acceptance criteria, e.g., the peak cladding temperature was limited to approximately 1,200 °C for postulated loss of coolant accidents.

Improvement of the ECCS technology continued in the BWR/5 and BWR/6 product lines. The most significant improvement is the use of three electrical divisions to support ECCSs installed on the BWR/5 and BWR/6 product lines (Figure 28.3-1). Two divisions contain two ECCS pumps each, an RHR pump, that is also used for LPCI, and either a LPCS or a second LPCI pump dedicated to reactor vessel injection. The third electrical division has a High Pressure Core Spray (HPCS) system powered by a dedicated diesel generator. The HPCS system replaces a turbine-driven HPCI used in the earlier product lines.

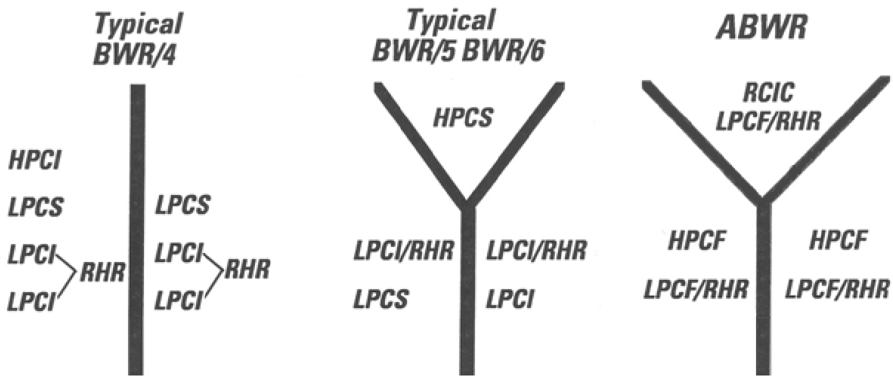


Figure 28.3-1: Evolution of BWR ECCS

For the ABWR, the introduction of internal recirculation pumps eliminated the need for large external recirculation loop piping. By eliminating this piping and the associated large vessel nozzles below the top of the active fuel, it became possible to reduce the capacity of the ABWR ECCS while achieving a major safety improvement of keeping the core covered with water during a design basis LOCA.

A three divisional ECCS was adopted for the ABWR as shown in Figure 28.3-1. Each division provides both low pressure and high pressure makeup capability and incorporates its own emergency diesel generator power supply. One division includes Reactor Core Isolation Cooling (RCIC) as a high pressure makeup system, and a Low Pressure Flooder/Residual Heat Removal (LPFL/RHR) as low pressure system, while the other two divisions include HPCF and LPFL/RHR trains. RCIC employs a turbine-driven pump fed by reactor steam to provide high pressure makeup when the reactor is isolated from the normal power cycle heat sink, drawing water from either the containment Suppression Pool or the Condensate Storage Tank (CST). RCIC is also a feature of earlier BWR product lines to provide isolation cooling and cooling capability during a Station Blackout event, but it was not considered part of the ECCSs prior to the ABWR design. As with previous BWR product lines, the ABWR also incorporates an Automatic Depressurization System (ADS) to depressurise the RPV to allow effective makeup from the low pressure pumps over the full range of potential LOCA break sizes. These systems are described in detail in Chapter 13: Engineered Safety Features, of this PCSR.

28.3.3 BWR Reactor Pressure Vessel Evolution Leading up to the Standard ABWR

Although a key principle of the BWR is that bulk boiling and steam production occurs in the reactor core, historically this steam was employed in either a direct power cycle, that is, steam passed directly to the turbine, or an indirect cycle using primary/secondary steam generator heat exchangers similar to Pressurised Water Reactors (PWRs), or a dual cycle mode that included features of both direct and indirect cycles. Only one early plant used the indirect cycle exclusively, but a few early plants employed the dual cycle approach. However, the indirect cycle did not prove either commercially or operationally attractive for the BWR because of the cost and maintenance associated with large steam generators and the associated need for large containments to enclose them, as well as the consideration of high operating pressure on the primary side in order to achieve acceptable power cycle efficiency and economic competitiveness. Therefore, the BWR product line was simplified to a direct cycle plant, employing an operating pressure at about 7 MPa that basically matches the secondary pressure (approximately 5 – 7 MPa) and power cycle thermal efficiency of the PWR.

Elimination of steam generators resulted in considerable simplification of the BWR Nuclear Steam Supply System (NSSS). There was a concern about the potential for high radiation levels in the Turbine Island (TI) portion of BWR plants, but early prototypes and commercial BWRs demonstrated that the radiation levels were manageable. More shielding is required for the BWR TI, and the Reactor Pressure Vessel (RPV) is larger than that for a PWR, but these disadvantages were offset by eliminating steam generators and construction of a smaller volume containment.

The BWR RPV is larger than the PWR RPV for two reasons:

(1) The BWR core operates at a lower power density, approximately one-half the power density that a PWR core operates at.

(2) The BWR has incorporated steam separation (separators) and drying (dryers) within the RPV.

Lower power density coupled with lower operating pressure provides operating thermal margin for the BWR. Optimisation of power density, core flow and operating pressure also provides nuclear-thermal-hydraulic stability during operation of the BWR. Because the power density is lower, there is less build-up of Xenon and Samarium isotopic species that absorb neutrons and reduces core reactivity, facilitating lower enrichment requirements and improved fuel utilization than would be the case if the power density were higher. Void reactivity feedback in the BWR also provides damping of Xenon transients. Therefore, the lower power density in the larger RPV diameter of the BWR is a benefit in terms of core performance and safety.

A compact NSSS is a result of performing the functions of steam separation and drying within the BWR RPV. This compact design results in reduced radiation dose rates in the containment when compared to the employment of external steam generators and steam drum components as in early BWRs.

Manufacturing limitations on the height and diameter for the RPV have been an imposed constraint on the RPV size since the earliest BWRs. Therefore, an original strategy for the development of the BWR was to provide product lines consisting of natural circulation BWRs, forced circulation BWRs and superheat BWRs, with each product line supporting an increased output level, respectively. The use of natural circulation and forced circulation were both introduced commercially for the early BWRs. Furthermore, GE constructed a prototype superheat reactor, the ESADA Vallecitos Experimental Superheat Reactor (EVESR), to demonstrate increased efficiency and reduced plant size for this product line. This development programme was discontinued due to challenges to the fuel integrity and to the marginal economic advantages to be gained through the use of nuclear superheat.

Prior to the time that the superheat development was terminated, improvements in manufacturing capability resulted in the introduction of larger size RPVs. GE therefore focused on forced circulation technology along with increasing the core size, power density and RPV diameter as a path toward higher thermal power output for the BWR.

The ABWR has adopted improvements in design and fabrication of the RPV relative to earlier BWR product lines. Earlier RPVs were constructed from welded low alloy carbon steel rolled plate. The ABWR uses low alloy carbon steel forged shell rings at and below the core elevation, thereby avoiding welds in high neutron fluence locations in the core beltline. Shell rings above the core beltline region may be made of low alloy carbon steel forged rings or welded plate. The top head is made of low alloy carbon steel welded formed plates.

An annular region between the vessel inner wall and the outer surface of a cylindrical shroud encompassing the core region provides a flow path for feed and recirculation water. RIP diffusers and impellers are installed in this flow path, which results in a large separation between the reactor core and RPV wall relative to previous BWRs. This wide separation between the vessel wall and shroud, along with moderate power density adopted for the ABWR, results in reduced neutron fluence on the RPV wall. The reduced fluence, together with reduced content of impurities such as copper, phosphorus and sulphur in the RPV material, results in low neutron irradiation embrittlement of the RPV.

Other ABWR improvements and differences relative to previous BWRs include:

- Relatively flat bottom head with RIP penetrations,
- Conical vessel support skirt,
- Inward vessel flange design,
- Steam nozzle with flow restrictor, and
- Double feedwater nozzle thermal sleeve.

The bottom head consists of a spherical bottom cap, made from a single forging, extending to encompass the Control Rod Drive (CRD) penetrations and a conical transition section to a toroidal knuckle between the bottom head and vessel shell, also made from a single forging, named ‘Petal’. This eliminates the bottom head welds within the CRD pattern. Penetrations are included for the RIPs, and the RIP motor casings are welded to the vessel bottom head.

The vessel support skirt has a conical geometry and is attached to the lower vessel cylindrical shell course. The support skirt attachment is an integral part of the vessel shell ring. Steel anchor bolts extend from the RPV pedestal through the flange of the skirt to secure the support skirt with the pedestal.

To minimise the number of main closure bolts, the ABWR RPV has an inside type flange. This flange allows a hemispherical vessel head with a radius less than the vessel radius, which helps minimise the weight of the RPV closure. The vessel closure seal consists of two concentric O-rings.

The steam outlet nozzles incorporate a flow restricting venturi. This provides for steamline flow and steamline break detection, and serves as a flow-choking device to limit blowdown and associated loads on the RPV and internals in the event of a postulated main steamline break.

The feedwater nozzles utilize double thermal sleeves welded to the nozzles. The double thermal sleeves protect the vessel nozzle inner blend radius from the effects of high frequency thermal cycling.

The RPV is described in detail in Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems.

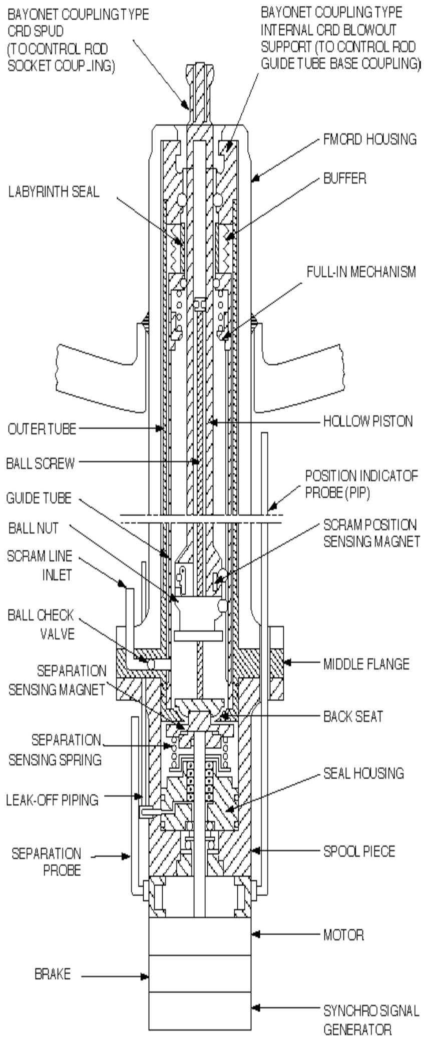
28.3.4 BWR RPV Internals Evolution Leading up to the Standard ABWR

The ABWR reactor vessel internals are described in Chapter 11: Reactor Core and Chapter 12: Reactor Coolant Systems and Associated Systems. The main features are that the CRDs and RIPs are mounted on the bottom of the vessel.

Bottom mounted, hydraulic CRDs were established during the design of Dresden-1, the first commercial BWR plant. Prior to Dresden-1, the EBWR prototype used bottom mounted CRDs, but the Control Rods (CRs) were actually withdrawn into a control rod guide structure above the core. Withdrawal of CRs was by an electric drive motor; fast shutdown ("scram") was accomplished by gravity and reactor pressure, supplemented with a spring assist to overcome inertia when the reactor was at low pressure. The VBWR, on the other hand, adopted CRDs mounted on a missile shield above the RPV top head, requiring an extension from the shield to and through the RPV top head. The CRDs were normally operated by electricity from storage batteries, but compressed air was used to assist scram of CRs during an emergency.

An obvious advantage of applying bottom mounted control rod drives was to avoid the need to remove the drive assembly in order to remove and replace fuel assemblies. Also, as described earlier, the BWR evolved to incorporate steam separation and drying in the flow path above the core within the RPV, which complicates application of a top mounted CRD and top withdrawn CR assembly. For these reasons, the commercial BWR opted for bottom mounted CRDs, and further simplified the drive to use hydraulic pressure to withdraw and insert the CRs for both normal positioning and fast shutdown ("scram") of the reactor.

The ABWR diversified the CRD to incorporate fine motion of CRs with an electric motor for operations, while retaining and simplifying the hydraulic scram function (Figure 28.3-2). Because the FMCRD has the additional electrical motor, it drives the control blades into the core even if the primary hydraulic system fails to do so, providing additional capability to electrically insert the CRs thereby reducing the risk of anticipated transient without scram (ATWS) events. Choosing the option to use clean purge flow and elimination of the water pipes associated with the locking piston type CRD used for drawing out CRs reduced radiation exposure in the ABWR design. Automated equipment is also applied to ease maintenance burden and reduce radiation dose.



Key Design Features
- Electro-hydraulic design
- Clean water purge flow
- Capability to detect drive/blade separation
- Electro-mechanical brake to prevent rod runout on pressure boundary failure
- Internal restraint to prevent blowout (no external restraints required)

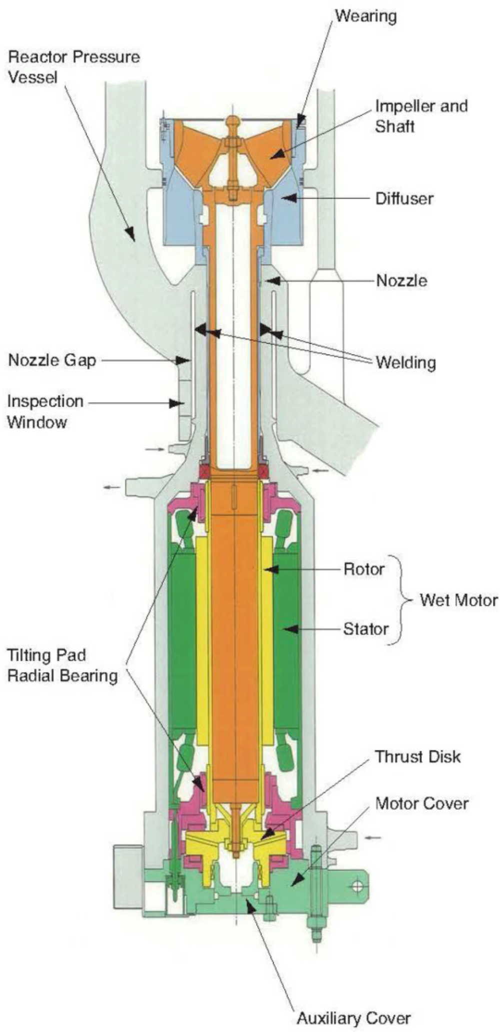
Key Benefits:
- Diverse shutdown capability
  - Hydraulic with electrical back-up
- Provides fine rod motion during normal operation
  - Small power changes
- Improved startup time & power manoeuvring
  - Gang rod movement of up to 26 rods possible
  - Increased protection against Rod drop and Rod ejection faults

Figure 28.3-2 : ABWR Fine Motion Control Rod Drive [Ref-1]

A significant development for the ABWR is using the option of replacing the external recirculation pumps and the jet pumps with RIPs (Figure 28.3-3). There are ten RIPs whose diffusers and impellers are arranged circumferentially between the shroud and the RPV near the RPV bottom head and whose motors are installed in the motor casing at the bottom of the RPV. The RIPs function collectively to force the reactor coolant through the lower plenum of the reactor and upward through openings in the fuel support, upward through the fuel assemblies and steam separators, and down into the annulus to be mixed with feedwater and then recirculated through the core. The RIPs are variable speed to adjust core flow. A change in RIP speed varies the core flow in the reactor, which, in turn, changes the reactor power during normal power range operation.

By adopting RIPs and opting to eliminate external recirculation pumps and piping in the containment associated with previous external and jet pump recirculation systems, a significant reduction in radiation exposure for inspection and maintenance work inside the containment is achieved. Elimination of external recirculation pumps and piping reduces worker radiation exposure associated with in-service inspection of the recirculation system itself, and also reduces worker radiation exposure as a result of reduction of a significant radiation source from the recirculation system while working on other equipment in the containment.

The introduction of the RIPs option also has the effect of significantly reducing the risk from LOCAs by removing the largest diameter pipework entirely from the design. By eliminating this piping and the associated large vessel nozzles below the top of the active fuel, it became possible to keep the core covered with water during a design basis LOCA and to significantly reduce heat-up and peak clad temperature during a LOCA event.



Key RIP Design Features

- Ten pumps can provide up to 111 percent flow
  - 100 percent flow with one pump out of service.
- Wet induction motor without shaft seal.
- Continuous purge with clean water.
- Impellers & motors removable without reactor draining.
- Solid-state, adjustable frequency speed control.
- Multiple power supplier reduces probability of significant flow loss.

RIP Benefits:

- Eliminates external recirculation loops:
  - Compact containment design.
  - No large nozzle for pipes below core.
    - ✧ Safer/ECCS optimised
  - Reduced In-service inspections.
    - ✧ Less occupational exposure
- Less pumping power required.

Figure 28.3-3 : ABWR Reactor Internal Pump [Ref-2]

In addition to the improvements in the RPV, FMCRD and RIP, several other safety improvements have been introduced for the Reactor Internal Components of the ABWR:

- The top guide has been changed from an assembly of welded plates to a single forged component with machined openings to support the fuel assemblies. This eliminates welds reducing the risk of crack initiation and simplifies in-service inspection which provides a benefit of reduction in occupational radiation exposure.

- The top guide and upper shroud are now integral to simplifying construction and increasing strength.
- The core plate structure with reinforcing beams and perpendicular reinforcing rods was changed to all reinforcing beams to increase its strength.
- Adoption of internal pumps eliminates large diameter pipe breaks below the top of active fuel. This has allowed the cooling system to have less capacity.

Improved design, materials selection, and fabrication practices adopted for the ABWR are important to address potential issues such as Intergranular Stress Corrosion Cracking (IGSCC), Irradiation Assisted Stress Corrosion Cracking (IASCC), and radiation exposure. The next section discusses the evolution of BWR material choices and reactor chemistry practices that have reduced and mitigated these issues in BWRs.

28.3.5 BWR Materials and Chemistry Evolution Leading up to the Standard ABWR

This section provides a brief high level introduction to materials selection and reactor chemistry background for the BWR with respect to component design and fabrication, coolant quality and control, radiation exposure, and fuel and water chemistry interaction. These aspects have a broad impact throughout the entire plant; however, this section focuses on impacts and evolution related to piping, reactor internals, and fuel. More detailed information can be found in Chapter 23: Reactor Chemistry.

For the early BWRs, reactor chemistry focused on keeping the reactor cooling water as pure as possible. At that time, BWR water chemistry specifications were primarily used to control conductivity and chloride levels to prevent the occurrence of transgranular stress corrosion cracking (TGSCC) and pitting corrosion. This seemingly fundamental approach did not recognize some of the issues dealt with today, particularly intergranular stress corrosion cracking (IGSCC), because early testing did not fully reveal the severity of the oxidizing nature of the BWR environment or the long incubation period for IGSCC initiation and subsequent growth in welds or thermal sensitized stainless steel. In addition, the high tensile residual stresses that could be produced by welding and grinding, which contributed to IGSCC for earlier BWRs, were not well understood.

It is now understood that material factors, stress factors and chemistry factors all contribute to IGSCC as shown in Figure 28.3-4 and a number of remedial and preventative strategies have been developed since IGSCC was first observed in BWRs in the late 1950s and early 1960s.

Some previous IGSCC concerns in BWRs were resolved by replacing the affected materials with more IGSCC-resistant materials or by performing repairs. One primary mitigation strategy was to focus on water chemistry practices to protect components and hence reduce the likelihood and/or extent of SCC growth. This grew out of an observation that oxidizing species in high purity coolant (oxygen and hydrogen peroxide), as well as anionic species that contribute to coolant conductivity, were correlated with the incidence of IGSCC cracking as well as the crack growth rate of any initiated cracks.

Laboratory tests indicated the beneficial effects of the addition of hydrogen to control IGSCC. Hydrogen additions to the reactor water that reduced Electrochemical Corrosion Potential (ECP) to less than -0.23V (SHE) were found to be effective in suppressing stress corrosion cracking. The addition of hydrogen is known as Hydrogen Water Chemistry (HWC).

Unfortunately, a side effect of adding hydrogen to the reactor water is an increase in gamma radiation emission in the main steam line from an increased fraction of nitrogen-16 in steam under HWC conditions.

Another important adverse effect of the HWC technology is the increase of piping dose rate due to the build-up of Co-60 in the oxide film on piping surfaces, which strongly affects the occupational worker radiation dose during a plant outage.

Both laboratory and in-plant crack growth rate data confirm the benefit of using HWC in mitigating existing cracks. Therefore, numerous methods have been explored to decrease the ECP of nuclear reactor structural materials exposed to high temperature water below -0.23V (SHE), while also reducing the amount of hydrogen addition in order to reduce gamma shine levels and suppressing the piping dose rate.

Noble Metal Chemical Addition (NMCA) achieves lowering of the ECP by injecting platinum and rhodium along with hydrogen levels that maintain gamma radiation levels in the turbine building close to those for HWC.

Zinc injection is currently used in BWRs which also apply HWC and noble metal technology, as zinc has been found to suppress the deposition of Co-60 in metal oxide layers. However, some zinc isotopes become activated in the reactor so depleted zinc is used, that is zinc with these isotopes removed. Zinc is injected as depleted zinc oxide.

The use of the combination of controls – hydrogen, noble metals and depleted zinc – is now considered best practice internationally for BWR water chemistry [Ref-3].

For the ABWR design, a group of measures corresponding to international best practice to reduce the likelihood of SCC have been adopted:

(1) Improvement of the design to avoid crevices, eliminate or minimise welds (e.g., one piece forged and machined top guide, simplified core plate design)
(2) Selection of materials that resist IGSCC [e.g., use of low carbon Type 316NG (Nuclear Grade) austenitic stainless steel, Alloy 600M with niobium stabilization]
(3) Reduction of residual stresses from fabrication (e.g., controls on welding, peening, polishing, solution annealing)
(4) Adoption of hydrogen injection, NMCA (as Online Noble Metal Chemical Addition OLNC) and depleted zinc injection.

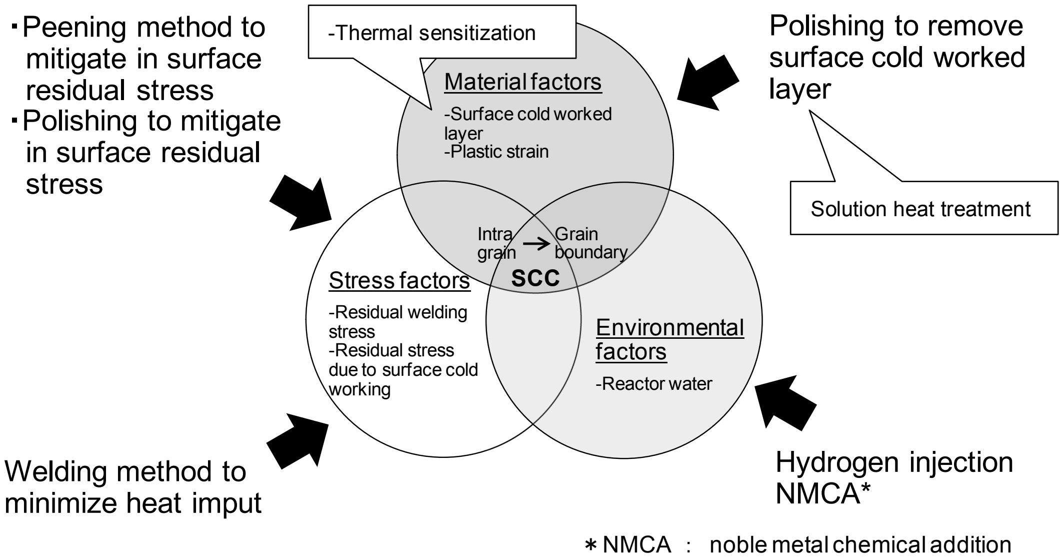


Figure 28.3-4 : Contributing Factors to SCC Mitigation

Finally, the choice of water chemistry affects the amount of radioactive waste generated during operation. In principle, the more complicated the chemistry regime the greater the amount of waste generated from the systems in place to implement the regime and therefore it is important to have close control of the levels of various additives in the coolant.

In conclusion, reactor chemistry and material selection have played an important role in the existing BWR fleet to minimise the incidence and growth of IGSCC. It has also played an important role in minimising plant radiation fields. This has been achieved while maintaining fuel integrity by minimising cladding corrosion. The chemistry regime to be adopted for the UK ABWR has been the subject of a study informed by more than three decades of options analyses to demonstrate that the chosen options make operating risks ALARP. This study is summarised in Chapter 8: Structural Integrity and Chapter 23: Reactor Chemistry.

28.3.6 BWR Containment Evolution Leading up to the Standard ABWR

The initial BWR containment vessels were constructed from steel in a spherical shape, and were of a dry type. However, this was soon superseded by the Mark I, Mark II and Mark III pressure suppression type containments, each incorporating a suppression chamber containing a large water pool (Figure 28.3-5).

The advantages of pressure suppression type containment include:

- A heat sink for LOCA blowdown steam, and reactor steam discharged from turbine driven pumped water makeup systems,
- A secure source of water for ECCS and reactor isolation makeup and cooling pumps,
- Suppression of safety/relief valve discharge steam and filtration of any activation or fission products contained in this steam, and
- A capability to filter and retain fission products that may be released during an accident or transient event.

Because the BWR may discharge steam when the reactor is isolated during frequent faults that pressurise the direct cycle, it is a natural fit for the BWR to adopt the pressure suppression system, and thereby adopt a smaller containment than a dry containment would allow by taking advantage of the pressure reduction that the suppression pool provides.

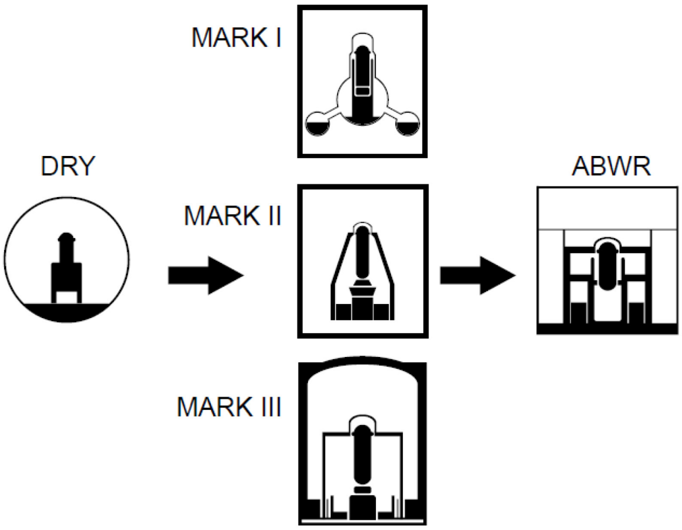


Figure 28.3-5 : Evolution of BWR Containment

The Mark I containment vessel consists of a "light bulb" shaped drywell surrounding the Reactor Pressure Vessel (RPV), and a suppression chamber consisting of a steel pressure vessel with a toroidal shape ("torus") and a large body of water inside the suppression chamber ("suppression pool"). The torus design provided a large surface area for venting steam.

The Mark II containment vessel consists of a steel dome head and either a post tensioned concrete wall or reinforced concrete wall standing on a base mat of reinforced concrete. The inner surface of the containment is lined with a steel plate which acts as a leak tight membrane. The Mark II design provides better access to piping and equipment in the drywell, a simpler vent configuration using straight pipes, the potential to use different construction materials, and a smaller reactor building.

The Mark III design has a strong Reactor Building providing containment similar to a standard PWR design. This has a simpler geometry and as a result is easier to construct than the Mark I or Mark II designs and also provides better access to equipment for maintenance. It can be built as either a free standing steel containment surrounded by a concrete shield building or as a concrete pressure vessel with a liner. The choice of steel or reinforced concrete for a containment vessel, as well as volume and design pressure rating, is the result of trade-offs related to construction and costs to achieve desired performance. This design does not easily lend itself to having the SFP inside the building as in the earlier designs, so the spent fuel pool is housed in a separate fuel building outside of the reactor building with a slanting fuel transfer tube to move fuel from the refuelling cavity alongside the reactor to the SFP in the fuel building – see Section 28.3.7.

However, the Mark III design also has some disadvantages. Firstly, the suppression pool is open and uncovered leading to steam being released to the upper containment drywell in the case of safety valve actuation. Also refuelling operations are performed inside containment – a departure from normal BWR practice. Finally, the overall design of the Mark III containment does not have the seismic stability of the Mark II or ABWR designs.

Therefore, the ABWR design did not develop the Mark III concept although it did adopt its simpler geometry of the right circular cylinder constructed using reinforced concrete. It otherwise retained characteristics closer to those of the Mark II containment design.

The ABWR Reinforced Concrete Containment Vessel (RCCV) is described in Chapter 10: Civil Works and Structures. The RCCV is cylindrical and consists of a top slab, a shell and a foundation. The RCCV is divided into a drywell and a suppression chamber by the diaphragm floor and the RPV pedestal which avoids the open, uncovered suppression pool of the Mark III containment. Its compact structure and reactor building integration improves seismic stability and capability to carry dynamic and shear loads. It also allows the Reactor Building to serve as a Shield Building, and provides easier modular construction and cost effectiveness. For these reasons, ABWR opted to retain the suppression pool while combining good safety design features of Mark II and Mark III.

28.3.7 Refuelling and Spent Fuel Storage

Most BWRs worldwide have used a similar approach to refuelling and spent fuel storage with the SFP separated from the refuelling cavity (the area immediately around the RPV) by a gate which can be removed or otherwise opened to allow spent fuel to be moved in one operation from the reactor to spent fuel racks, and to allow new fuel to be moved in one operation from storage racks to the reactor. Making provision for a number of years' fuel storage is part of this concept. When necessary, fuel is then removed from the SFP for processing or longer term storage using a transfer cask. Evolution of BWR SFP location is shown in Figure 28.3-7.

The BWR6 design with Mark III containment departed from this norm by having the SFP in an adjacent building with a slanting fuel transfer tube from the refuelling cavity to the SFP. Spent fuel from the reactor is moved directly to the transfer system and, two elements at a time, to the SFP in the adjacent building. New fuel is brought in using the reverse process. This system has experienced some operational issues, with outages having to be extended because of slow transfer rates between the reactor and the SFP. There are also safety issues relating to failure of the valves at the end of the transfer tube leading to loss of water in the refuelling cavity, the possibility of fuel getting stuck in the transfer tube, and the number of fuel movements outside the Reactor Building needed during each outage.

In choosing to develop the ABWR containment from the Mark II design rather than continuing the development of the Mark III design, it was also decided to go back to the more usual BWR design option of having the SFP at the operating floor level – that is level with the refuelling cavity – inside the Reactor Building. Apart from simplicity and the operational issues discussed above, aircraft crash and seismic protection of the SFP is provided by the Reactor Building rather than having to have a separate seismically qualified and aircraft crash resistant building. The detailed layout of this option, particularly in regard to the height of the SFP above ground level is fixed by two considerations, namely, the need for a certain depth of water above the reactor to provide shielding during refuelling and the position of the main steam lines with respect to the Turbine Building.

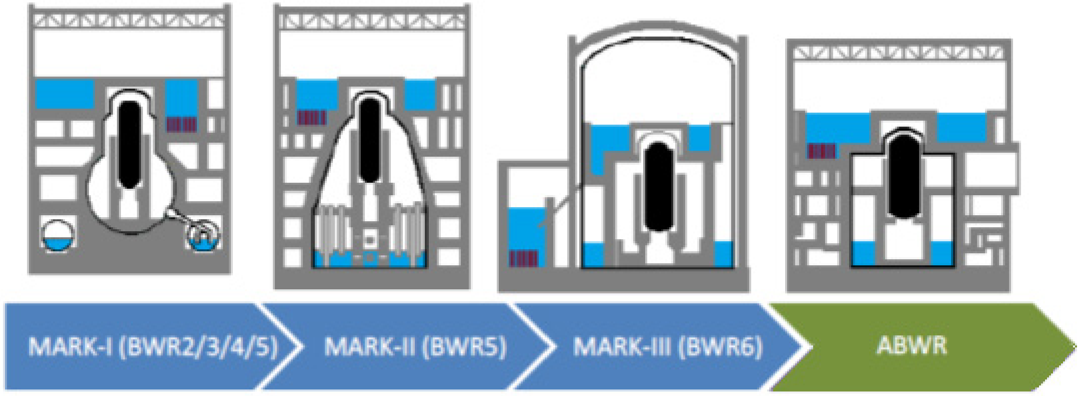


Figure 28.3-6 : Evolution of BWR SFP location

28.3.8 Electrical and C&I Evolution Leading up to the Standard ABWR

The BWR electrical distribution system and Control and Instrumentation (C&I) system, especially the primary safety system, which includes the reactor protection system, have developed over time as regulatory requirements and front line safety systems have evolved. In this document, the electrical and C&I evolution history for US BWRs is described below as an example; other non-US BWRs may differ in design depending on the regulatory requirements of their respective countries or regions. For the UK ABWR, additional design changes have been proposed as a part of the ALARP process and these are described in Chapter 15: Electrical Power Supplies and Chapter 14: Control and Instrumentation.

The early BWR product lines applied a two division safety system, and the electrical distribution system and the reactor protection system were two divisional as well (Figure 28.3-7 and Figure 28.3-8).

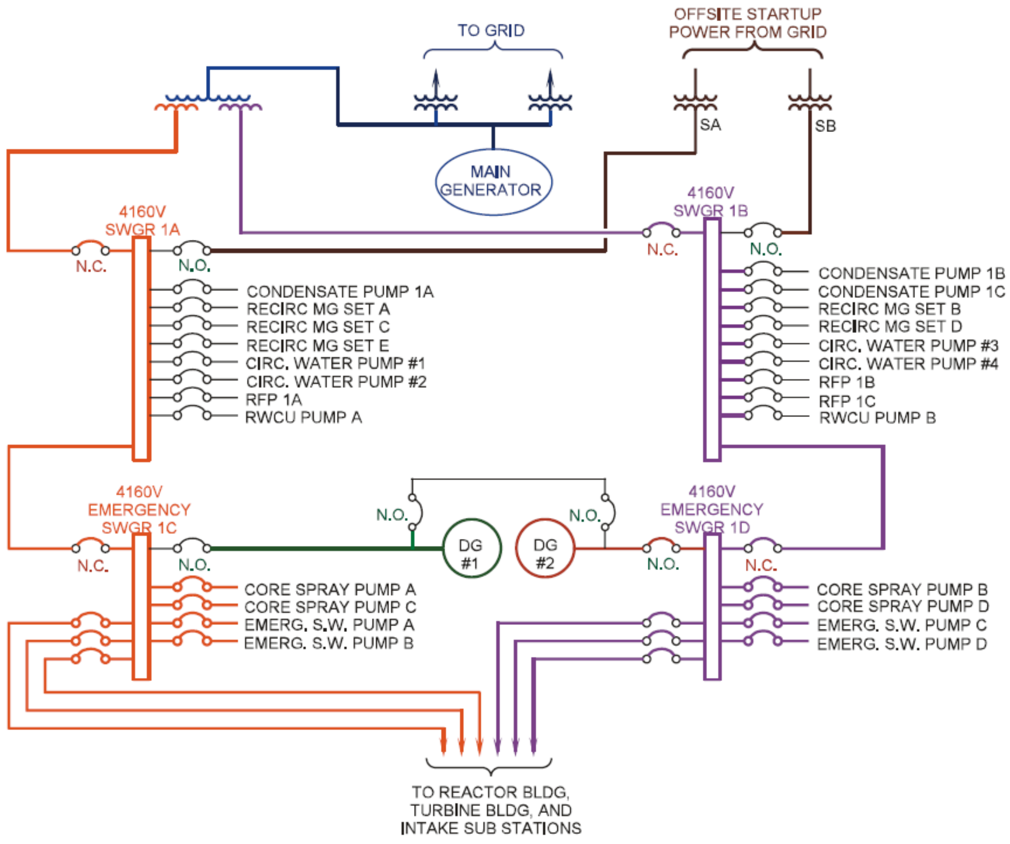


Figure 28.3-7 : Early BWR Electrical Distribution System [Ref-5]

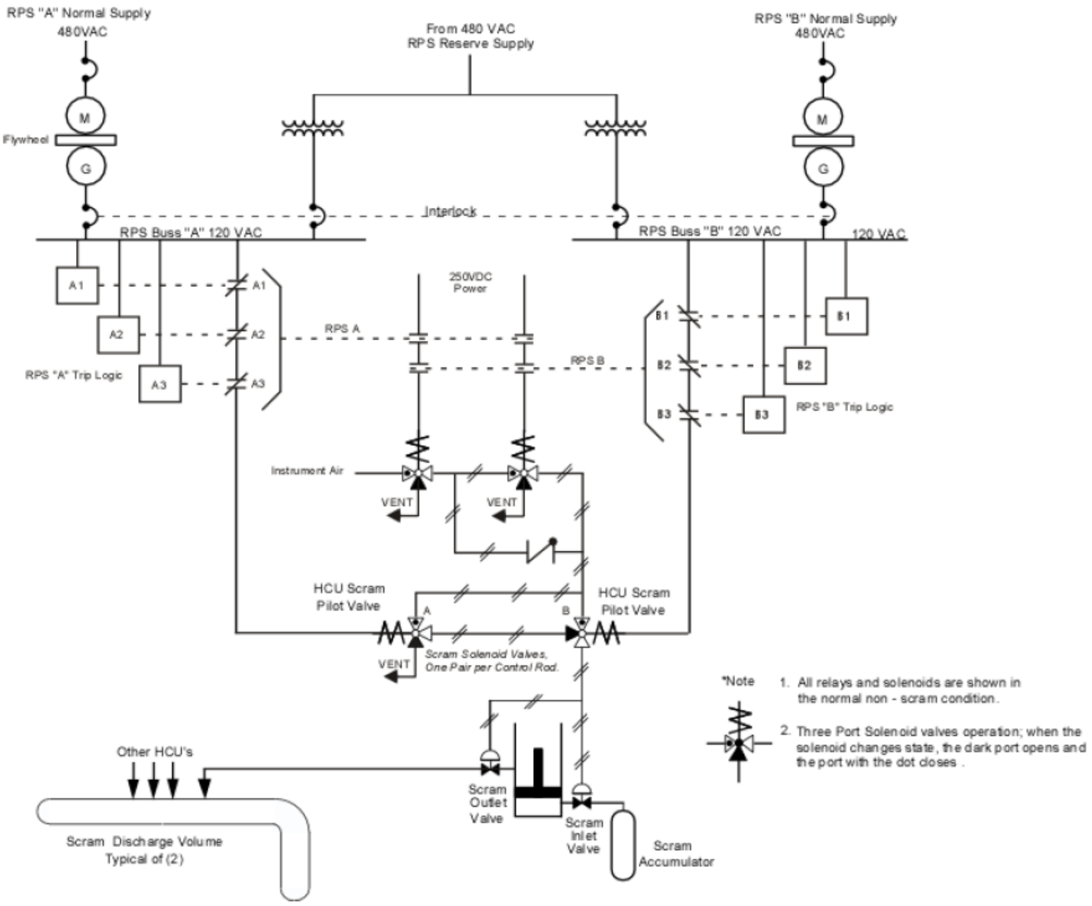


Figure 28.3-8 : Early BWR Reactor Protection System [Ref-6]

More recent trends have incorporated a three tier electrical distribution system concept, with separation of Power Generation, Plant Investment Protection, and Safety Loads. Also, an additional independent on-site standby power source has been incorporated based on either a Combustion Turbine Generator or Air-Cooled Diesel Generator. In the UK ABWR design, non-safety loads are not connected to safety-related electrical supplies.

Figure 28.3-9 shows a single line diagram for the reference design [Ref-9].

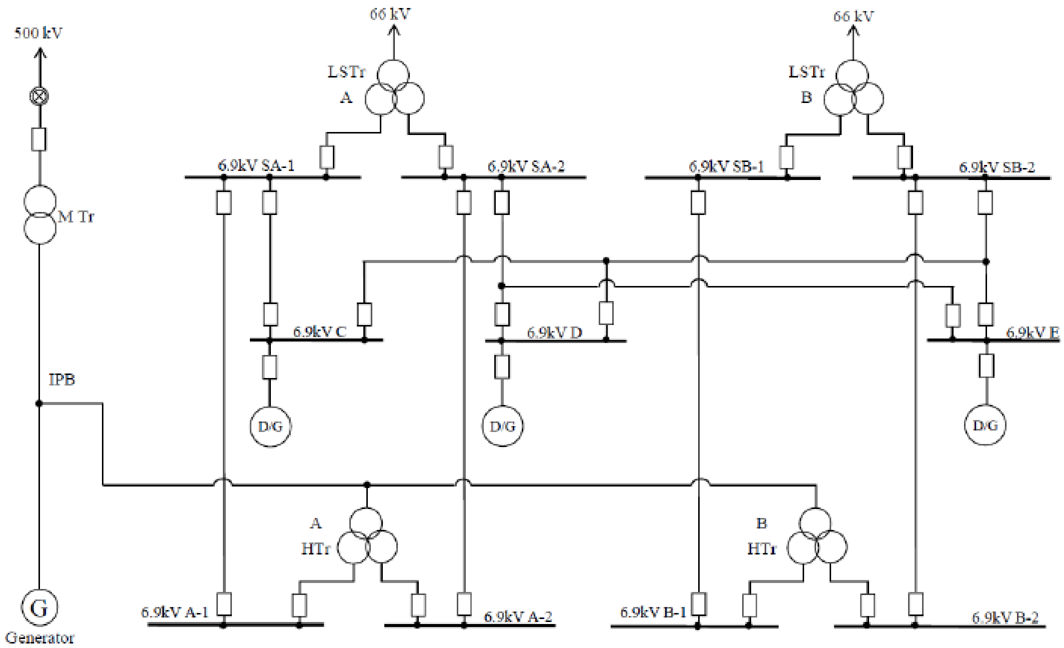


Figure 28.3-9 : ABWR Single Voltage System One Line Diagram [Ref-8]

The C&I, for example the reactor protection system, has also evolved alongside the safety systems such as the ECCS. Additionally, the C&I system has evolved as technology has evolved, most notably, the transition from analogue electrical circuits to digital electrical circuits. The main control systems such as the recirculation flow control system, feedwater control system and the Balance of plant (BOP) local control systems were first to adopt digital computer-based technology. For the ABWR, a four division microprocessor-based digital C&I system with two-out-of-four trip setpoint voting and divisional logic unit voting was adopted for the reactor protection system and the safety system (Figure 28.3-10). Following the example of the ABWR, several existing BWRs have retrofitted digital microprocessor based C&I equipment, including safety system applications, to replace obsolete components, improve system performance and provide enhanced self-diagnostics and maintenance capabilities.

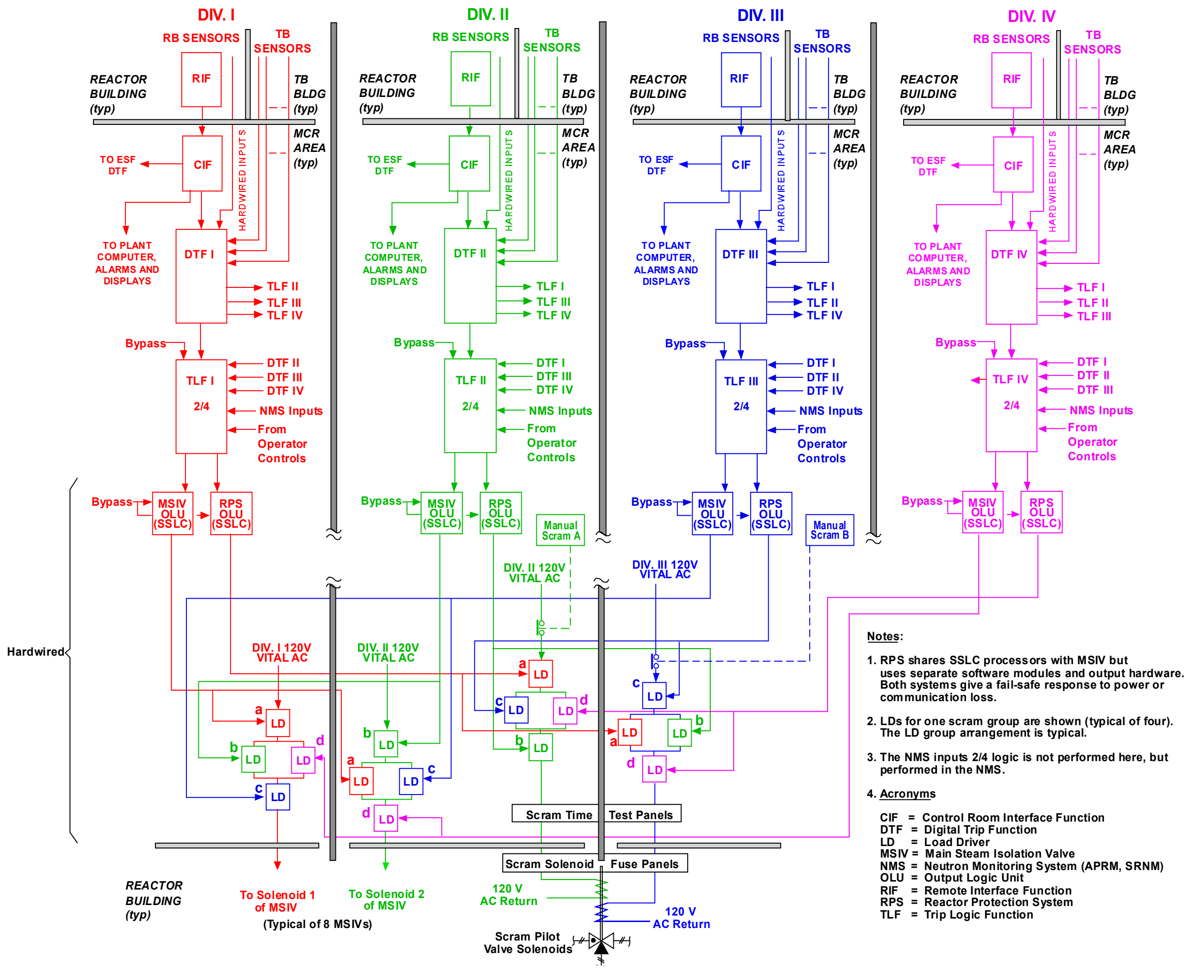


Figure 28.3-10 : ABWR Reactor Protection System (Scram and Main Steam Isolation Valve Functions)[Ref-7]

Additionally, the ABWR has made use of modern Human Factors Evaluation as a discipline in the design of the Human-Machine Interface and Advanced Main Control Room. This has led to a number of safety improvements in the operation of the reactor:

- Large scale display board facilitates to facilitate sharing of information between operators,
- Warnings are displayed using hierarchies for improved identification,
- Main monitoring operations are consolidated into a compact console for ease of operation,
- An expansion of automatic operations allows operators to concentrate on plant monitoring operations,
- Improved reliability and ease of maintenance as a result of integrated digitisation, and
- Electrical and physical separation between safety systems and non-safety systems.

The Human-Machine Interface for the UK ABWR is described in detail in Chapter 21: Human-Machine Interface.

28.3.9 Design for Environment

The evolution of the BWR technology has led to a number of improvements in the environmental performance of the ABWR. The range of improvements extends across all disciplines including:

- Preventing and minimising (in terms of radioactivity) the creation of radioactive waste while still supporting the operation of the reactor,
- Minimising (in terms of radioactivity) the discharges of gaseous and aqueous radioactive wastes through implementing systems which capture the activity, or allow for natural decay, prior to release,
- Minimising the number of components and the volume of material that would ultimately become radioactive waste, and
- Allowing for flexibility in the design to have site specific waste management systems for the treatment and management of solid wastes and for controlling discharges into the environment.

The two descriptions below provide an example of the improvements made to the ABWR design over recent years:

(1) Gaseous Waste Treatment System
Early Gaseous Treatment Systems were designed to recombine oxygen and hydrogen in the gaseous waste from the main condenser, then to store them in a pressurised tank for approximately one day for decay before discharging via the stack. In order to reduce the environmental release of the activated discharged gases, Hitachi-GE started a R&D programme into noble gas holdup technology that only the KRB power plant in (West) Germany had adapted at that time. This R&D was undertaken from 1968 to 1971, and this technology was adopted as the activated carbon noble gas holdup system for the Tsuruga Unit 1, known as the Off-Gas System (OG). The Noble Gas Holdup System with activated carbon is designed to have the holdup performance of approximately 30 days to allow for the radioactive decay of Xenon. The radioactivity release was reduced to one-tenth of the past design.

(2) Liquid Waste Treatment System
Liquid waste is separated into equipment drain (high purity, high activity), floor drain (low purity, low activity), chemical drain (low purity, high activity), and laundry drain (low purity, extremely low activity), and treated respectively. The floor drain and the laundry drain were designed to be directly discharged following treatment. In terms of the radioactivity released, the release of the floor drain liquid waste was dominant, and in order to reduce the activity release, the discharge of floor drain was changed to re-use within the plant after treatment with evaporation and demineralization together with the chemical drain waste. Due to this change, the development of a new evaporation system was required to enhance its durability and maintainability because the load on the evaporation system considerably increased.

The original evaporation system was a single barrel or natural circulation multi barrel type. Its durability and maintainability were insufficient for the following reasons:

- Occurrence of pitting corrosion and crevice corrosion, and
- Occurrence of blocking scale in the heat exchanger tube.

In order to reduce the corrosion potential, a number of options, such as pH adjustment, were studied and the use of a corrosion inhibitor and / or corrosion resistant material was adopted.
For the countermeasures against blocking scale, a circulation type (by pump) evaporation system was developed. This type can make the velocity in the heat exchanger tube faster than that of the natural circulation type. Higher velocity can lessen the occurrence of blocking scale.

In more recent plants, the quantity of water in the floor drain is quite small compared to the amount of water in the equipment drain, meaning that it is possible to treat the floor drain together with the equipment drain (filtration and demineralization). Most importantly this new process achieves the reduction of the concentrated liquid waste generation and consequently radioactivity release and it has the added benefit of reduced equipment cost.

Until the mid-1990s, the Dry-cleaning laundry system was widely used with solvent of chlorofluorocarbon in order to reduce the release radioactivity, but the Montreal Protocol Agreement International Conference decided to restrict the production and usage of chlorofluorocarbon for the protection of the ozone layer. To replace the dry-cleaning system water washing laundry system was the chosen option and a reduction of release of radioactivity has resulted by adopting an evaporation plus demineralization treatment and/or an activated charcoal absorption plus microfiltration treatment system.

In addition to the improvements over recent years, UK ABWR is aligned to UK RGP. For example, a UK ABWR specific Radioactive Waste Management Arrangements document demonstrates how solid radioactive wastes can be managed within the UK and evidence is provided to show that the UK ABWR has been optimised to minimise its impact upon the environment.

28.3.10 **Conclusion**

The development of the standard ABWR has its basis in the development of conventional BWRs in Japan, the USA and Europe. Throughout this development process many design options have been reviewed and those selected for the ABWR were based on good design and operating experience. From the description in this section it can be seen that the chosen options have led to a modern design which has significantly reduced design and operational complexity, and corresponding reductions in risks to workers and the public. At each stage of development technology options considered to be best practice at the time were adopted to minimise risks.

28.4 Development of ABWR after Kashiwazaki-Kariwa Units 6 and 7

After the KK-6/7 projects, there were several ABWR construction projects in Japan, including Hamaoka Unit 5 for Chubu Electric Power Company and Shika Unit 2 for Hokuriku Electric Power Company. Currently, Shimane Unit 3 for Chugoku Electric Power Co., Inc. and Ohma Nuclear Power Station for Electric Power Development Co., Ltd. are under construction by Hitachi-GE, representing the fifth and sixth ABWRs in Japan. Through all of these ABWR projects, Hitachi-GE has been able to continuously improve the ABWR design by incorporating various options based on customer requirements, site conditions and improvements based on earlier plant experience as well as technological advancements.

Major changes incorporated into the design of the Shika Unit 2, Shimane Unit 3 and Ohma Unit 1 are shown in Table 28.4-1.

Table 28.4-1 : Summary of major options incorporated into more recent ABWR designs

Reactor	Options	Discussion
Shika-2	Turbine Island-Configuration Human-Machine Interface (MCR) 60Hz optimization Thick-walled nozzle For RIPs (Enhanced Seismic Design)	Turbine axis is perpendicular to reactor building reducing risk to the reactor from turbine failures Changes to the MCR are based on operational experience from KK-6/7 and significantly reduce operator workload and minimise potential for human errors Not applicable to UK Reduced seismic risk – not a significant risk for the UK
Shimane-3	High seismic acceleration Thick-walled nozzle For RIPs (Enhanced Seismic Design) Human-Machine Interface (MCR) FMCRD with Induction motor and unsealed driving shaft Large-capacity small size of SRV	Reduced seismic risk – not a significant risk for the UK Reduced seismic risk – not a significant risk for the UK Adopt the same MCR concept as Shika-2 SM type of FMCRD has been adopted for UK ABWR Adopt the same capacity of the SRV as Ohma-1
Ohma-1	Full MOX core Human-Machine Interface (MCR) Large-capacity small size of SRV	Use of MOX is a commercial consideration and is outside the scope of GDA Adopt the same MCR concept as Shika-2 Increased capacity of the SRV to apply for the full MOX design (The number of SRVs was reduced to 16 from 18)

As in the case of the development of the BWR throughout its history, the design of each new ABWR in Japan has incorporated options considered to be best practice at the time in order to reduce risks as well as maximise performance.

28.5 Response to the Fukushima Accident

It is well known that the Great East Japan Earthquake and the subsequent Tsunami, both beyond the design basis levels for the plant which occurred in March 2011 caused severe damage to 3 of the reactors at the Fukushima Dai-ichi site. The severity of this event meant that not only was off-site electrical power lost for a period much longer than that analysed in the safety case, it also led to a failure of all of the on-site diesel generator based back-up supplies. The Tsunami also caused extensive damage to the electrical distribution system.

Responding to seismic acceleration, the Fukushima Dai-ichi Nuclear Power Station went into automatic shutdown at approximately 2:46 pm on March 11, 2011. With off-site power having been lost due to the earthquake, the back-up DGs started automatically to provide emergency power. Both the automatic shutdown and automatic initiation of the DGs were according to the design despite suffering from a beyond the design basis seismic event.

Subsequently, at approximately 3:35 pm, the power station was struck by the tsunami, resulting in the inundation of water supply equipment, including the emergency seawater system, and the flooding of the yards around buildings. Water also entered the buildings resulting in the failure of many power related safety equipment and led to the plant condition Station Blackout (SBO) and Loss of Ultimate Heat Sink (LUHS). The inundation of the building also left some switchboards out of action, including DC power resulting in the unavailability of safety related DC power infrastructure. This loss of control power and plant status monitoring functions greatly impeded effective management of the accident. The resulting station blackout meant that the reactors could not be adequately cooled and consequently severe damage to the reactor cores ensued.

28.5.1 **Lessons Learnt**

Hitachi-GE, in conjunction with other vendors and nuclear operators around the world has examined the events following the March 2011 Beyond-Design-Basis (BDB) event to ensure that all appropriate lessons have been learnt. This section describes seven lessons learnt from the accident. Sections 28.5.2 and 28.5.3 reflect how these lessons have been applied to the UK ABWR.

<u>Lesson 1: Arrangement, back-up and recovery of important equipment</u>
The BDB tsunami, which breached the sea defences, left the Fukushima Daiichi Nuclear Power Station submerged beneath almost 5 metres of water. The tsunami flooded the first floor of the reactor building (R/B) through air inlets and outlets and through doors broken by the impact force of the water. Emergency AC power, DC power supply systems and switchboards could not fulfil their safety functions because of the water flooding in the basement.

Relocation of important equipment and facilities to a higher places and/or waterproofing are effective protection countermeasures. It is also important to provide back-up power and cooling systems with the flexibility to function in unexpected and severe conditions and circumstances.

<u>Lesson 2: Configuration and deployment of isolation valves</u>
The Isolation Condenser (IC) or the RCIC provide core cooling functions during isolated conditions of the RPV. For a loss of this function, high pressure coolant injection systems and/or low pressure coolant injection systems with RPV depressurization are installed.

In Unit 1 of Fukushima Dai-ichi Nuclear Power Plant, the isolation valves of the IC were closed due to loss of instrumentation DC power following the tsunami. Therefore, measures for opening closed isolation valves, which can be operated outside the containment or remotely, should be added.

Lesson 3: Provision of back-up DC power supply
Loss of DC power caused the following situation during the Fukushima Dai-ichi accident:

- Considerable difficulties in understanding the plant status due to loss of all instrumentation functions, which caused misunderstanding of the plant status and the consequent failure to take appropriate recovery actions.
- Loss of the RPV depressurization function via the SRVs, which require DC power to operate.
- Loss of control signals for coolant injection systems such as RCIC.

Accordingly, it is essential to provide DC power back-up supplies for these important functions.

Lesson 4: Instrument reliability and credibility
Instrumentation for reactor water level, pressure and temperature, which are required for effective accident management, could not be relied on to provide credible readings as the accident progressed. Credibility of the instrumentation was lost due to the accident producing a hostile environment that exceeded their design range. The reliable performance of a key set of instrumentation during a severe accident is essential for the effective management of the event and although reliability requirements for these key instruments are less stringent than those required for design basis faults their environmental qualification to survive the severe accident is of critical importance.

In addition, accident management procedures and practices for loss of all instrumentation should be prepared.

For the same reason, additional temperature and level indication should be added for the Spent Fuel Pool (SFP).

Lesson 5: Provision for flexible coolant injection
For BDB flooding, flexible coolant injection into the RPV and containment are vital in addition to other protection measures such as waterproofing and relocation of important equipment. Water sources should also be flexible. In the event of a situation in which the loss of all permanently installed cooling injection systems is assumed to occur, mitigation may be provided by manually operated mobile equipment in conjunction with accessible pipework connection points.. Flexible and simple event management procedures for the effective deployment of these mobile safety systems must also be developed.

Lesson 6: Accessibility, operability and assurance of effectiveness
Problems of accessibility, operability and assurance considerably hindered the effectiveness of the deployment of accident management equipment occurred during the Fukushima Daiichi accident. These include:

- Difficulty of access to the Wet Well (W/W) venting valve because it is located near the suppression chamber, which has no radiation shielding,
- Difficulty of access to alternative coolant injection systems from outside the R/B,
- Ineffective coolant injection because of core bypass, and
- Possible ineffectiveness of W/W venting because it was not possible to be certain of the state of the rupture disk, which is required for maintaining boundary integrity in normal plant conditions.

To mitigate these problems in UK ABWR, the following countermeasures are recommended:

- Heavy machinery to have clear access to the plant,
- Installation of remote manual valves to reduce radiation exposure in a severe accident,
- Installation of several separate, readily accessible and robust connections to the R/B for alternative mobile coolant injection systems,
- Installation of an isolation valve to prepare an appropriate line and to avoid bypass,
- Installation of an alternative coolant injection line to the SFP, and

- Installation of a bypass line for the rupture disk or removal of the rupture disk.

Lesson 7: Provision for alternative means for protecting the RCCV
A major reason for a land contamination by the Fukushima Dai-ichi accident was leakage from the containment gasket because of the over-heating of containment to a level beyond the design basis temperature following a delay in establishing core debris cooling. Fission product release through W/W venting due to S/P scrubbing of fission product was assumed to be much less than release due to leakage from the gasket. Therefore, countermeasures to prevent over-heating of the containment are important. Ultimately, containment over-heating can be prevented by cooling any core debris. The following are examples of such provisions to prevent over-heating the containment:

- Avoiding containment over-heating by enhancing coolant injection into the RPV and containment,
- Avoiding containment over-pressurisation by W/W venting with measures to restore inerting to the RCCV, and
- Increasing the safety margin for containment over-heating by cooling the non-metallic part of the containment head flange with direct coolant injection into the reactor well.

As part of the GDA process, Hitachi-GE has considered the lessons learned from the Fukushima accident. In a separate Topic Report [Ref-15], the following have been reviewed:

- The recommendations from the report into the accident produced by the then Chief Inspector of NII (now ONR),
- ONR’s findings following the European Nuclear Safety regulators’ group ENSREG “Stress Tests”, and
- Recommendations and issues considered by UK licenses as a result of their responses to the ENSREG “Stress Tests”.

All of the recommendations and issues identified from this review have been compared to the current UK ABWR design, in particular, its provisions against Fukushima type accidents and external events in general outside the design basis.

28.5.2 Basic Strategy for Safety Measures

Lessons learnt from the accident at Fukushima Dai-ichi Nuclear Power Station strongly support the need to consider the potential for site-wide damage caused by BDB external hazards.
The most important approach for BDB external hazards is to provide diversified measures of coolant injection into RPV because the BWR uses direct-cycle operation at low pressure, and it is easy to inject water directly into the reactor.
The following is the basic strategy for safety measures based on the above considerations:

(1) Protection of important safety equipment from damage caused by external events through location of equipment or the provision of appropriate barriers,
(2) Use of portable equipment in the event of failure of installed safety equipment to perform their safety functions,
(3) Making accident management procedures as simple as possible to ensures coordination proceeds smoothly in the event of an external event leading to widespread damage across the entire site and a need for off-site assistance.

28.5.3 Overview of Countermeasures

Based on the strategy for safety measures described above, the following is an overview of the measures provided in the standard design and adopted in the UK ABWR design:

(1) Because of the complex situation during the initial response to a BDB external event leading to a loss of all installed safety systems, accident management should be simple. The initial response to such severe events is to initiate direct coolant injection into the RPV using mobile equipment. This simple strategy is also an effective way of cooperating with off-site support during times of confusion.
(2) Provide diverse connections to the R/B for on-site and off-site support (eg coolant injection, power supply) using mobile equipment.
(3) In addition to alternative ways to achieve coolant injection into the containment, coolant injection to the containment head is provided by the Specific Safety Facility Flooder System (FLSS) or the Reactor Building Flooder System (FLSR). These systems are described in Chapter 16: Auxiliary Systems of this PCSR are considered as countermeasures for Lesson 7 in Section 28.5.1.
(4) Connections are provided for the injection of coolant into the SFP.
(5) In addition to the above measures, a back-up building with an alternative power supply and coolant injection function is installed for use in the event of severe damage to the R/B. This facility is located in a separate part of the site from the R/B. The back-up building can be used for functions such as providing a frontline base during emergencies and as a storage facility for mobile equipment. The back-up building is considered as a countermeasure for Lesson 1 in the Section 28.5.1.
(6) Mobile equipment is provided for AC and DC power and coolant injection to RPV, RCCV and SFP (by FLSR, see Chapter 16,), and long term heat removal (by AHEF, see Chapter 16,) plus heavy machinery for gaining access to the plant in the case of wide-spread disruption to the site by any beyond design basis event. This is considered as a countermeasure for Lesson 3 described in Section 28.5.1.
(7) Installation of an extension handle for some isolation valves is being considered for manually opening closed isolation valves so that the valves can be opened even in the event of a loss of AC or DC power event. This is considered as a countermeasure for Lesson 2 described in Section 28.5.1.
(8) SevereAccident Instrumentation (see Chapter 14, Section 14.6.5) is an important part of the UK ABWR design. This is considered as a countermeasure for Lesson 4 described in Section 28.5.1.

(9) FLSS (fixed installation) and FLSR (mobile) are planned to be installed as flexible coolant injection systems to the RPV, PCV spray, reactor well, SFP and lower drywell (D/W). These are considered as countermeasures for Lesson 5, 6 and 7 described in Section 28.5.1.

28.5.4 Conclusion

The design of the UK ABWR has made use of the lessons learnt from the accident at Fukushima Dai-ichi to make provision for the flexible management of any external beyond design basis event. The options considered and implemented have the effect of significantly reducing the risk of Beyond Design Basis events for the UK ABWR. More detailed information can be found in Chapter 26: Beyond Design Basis and Severe Accident Analysis and its supporting references.

28.6 Strategy for the Demonstration of ALARP in GDA

28.6.1 ALARP Evaluation Methodology

As noted earlier, it is a legal requirement that the risks arising from the construction, commissioning, operation, maintenance and decommissioning of UK ABWR are ALARP. This section describes the process used in GDA to demonstrate this. The process has been applied in all areas of the design in a manner commensurate with the detail available in GDA. The ALARP process will necessarily continue through the detailed design phase following on from GDA.

The strategy for the demonstration of ALARP during GDA has a number of elements:

(1) Justification of the claim that relevant good practice has been used in the development of the design

The discussion that forms the bulk of this chapter and charts the development of UK ABWR from the earliest BWR designs provides a high level description of the evidence that good international practice has been incorporated in the design as that RGP has developed. Some of the options for design changes considered in the early part of GDA and described in more detail in other chapters of this Generic PCSR arose because of UK-specific practices and regulatory expectations that are considered RGP in the UK.

(2) Identification and consideration of risk-reducing options in all technical areas for design and operation of UK ABWR

This chapter identifies some of the major design and operational decisions that should be considered as part of the overall ALARP justification for UK ABWR. The areas to be considered come from assessment in a number of technical areas and also from a systematic assessment of risk based on the PSA. The overall ALARP justification also uses the strategy of identifying risk-reducing options in individual technical areas. These options are identified and discussed in the relevant chapters of this Generic PCSR. As stated earlier the information given in this chapter is not a detailed methodology for producing ALARP assessments. A standard methodology has been produced that it used for all aspects of the Generic PCSR and its supporting references [Ref-10]. However typical factors (this is not a complete list) that would justify a detailed ALARP analysis are as follows:

- Areas identified in the PSA that are higher risk
- Where fault studies indicate that DBA acceptance criteria are not fully met,
- Where hazards assessment indicate risk reduction is possible (e.g. changing the location of the EDGs),
- Where OPEX indicates that changes to the design or operation could reduce risk (e.g. WANO, EPRI, BWR UG, etc.),
- From discussions with the future licensee or regulatory bodies, and
- Where claims are made on the operator responding in less than 30 minutes.

(3) Determining which options (if any) are reasonably practicable

This hinges on determining whether the likely costs of implementing an option are grossly disproportionate when considered against the possible reduction in risk that might be achieved if the option were implemented. The [Ref-10] provides a methodology to determine if options identified in the ALARP assessments are reasonably practicable.

Options arising during the GDA process have been assessed in this way and are shown diagrammatically in Figure 28.6-1.

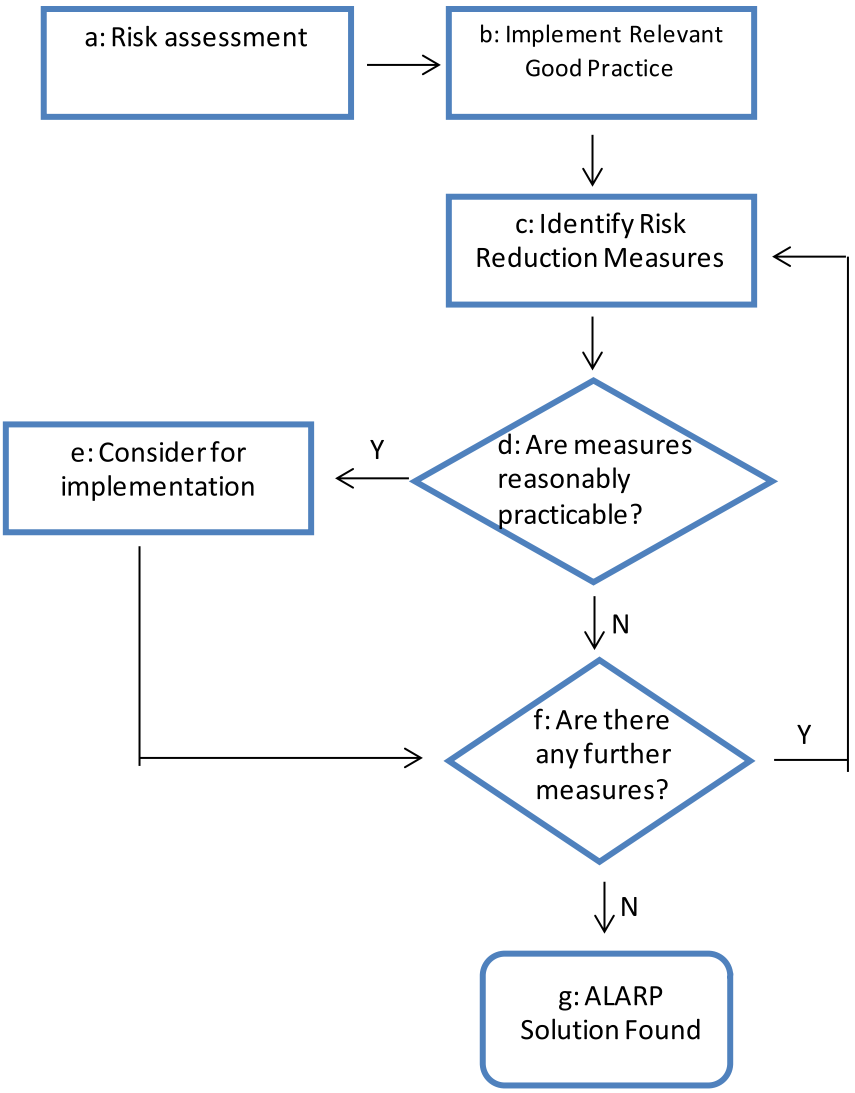


Figure 28.6-1 : ALARP process overview

(4) ALARP assessment with multiple risk indicators

The methodology described in the previous paragraphs assumes a single risk is being assessed and reduced. However, in many practical applications, the situation is more complex with multiple, often competing, risks, risk indicators and cost indicators. In these cases, a more complex methodology is required that allows competing risks and options to be compared.

The methodology adopted for UK ABWR is based on multi-attribute decision analysis (MADA). Risk indicators and cost indicators are identified and weights assigned to reflect the relative importance of each. Cost indicators are assigned lower weights to account for gross disproportion.

Each option is then scored according to how well it addresses each risk indicator or how much each cost indicator applies. Finally, for each option the total weighted score is calculated. This allows options to be compared taking into account the multiple risk and cost indicators. The option with the highest score is most likely to be the ALARP option.

Appendix A gives an example of the use of this methodology.

(5) ALARP assessment results

Most technical areas that form the basis of the contents of this PCSR have their own ALARP assessments, which are summarised in the corresponding PCSR chapter. Chapter 28 summarises the overall ALARP status of the UK ABWR and makes reference to the most important ALARP assessments from other chapters. However, details of specific ALARP assessments can be found in the references in the corresponding PCSR chapter.

Some major design decisions taken as a result of the ALARP studies undertaken in GDA are discussed in Section 28.7.1.

28.6.2 ALARP Insights from Probabilistic Safety Assessment

Probabilistic Safety Assessment (PSA) has been an important part of the safety justification of nuclear plants since the 1970s, starting from the Reactor Safety Study (also known as the Rasmussen Report or WASH-1400 (Ref-14) published by the US NRC in 1975. Originally, PSA concentrated on the risk measure provided by the Core Damage Frequency (CDF). Over the decades since the Reactor Safety Study, further measures have been added, principally, Large Release Frequency (LRF), which is a surrogate measure of societal risk and which can be mitigated by taking mitiagtive accident countermeasures such as evacuation and prohibition of foodstuffs into account, and Large Early Release Frequency, in which the much reduced timescales make such mitigative accident countermeasures impossible.

Although PSA originally concentrated on CDF, it was quickly realised that the strength of the methodology was that it allowed the contributions of initiating events, individual system failures and other features to be compared.

PSA thus provides a number of ways of gaining insight into nuclear risks with a view to demonstrating that overall nuclear risk is ALARP:

- PSA shows the balance of risk contributors so that the highest risk contributors can be addressed first, maximising the benefit of any proposed design changes.
- PSA allows the risk reduction potential of different options to be compared, giving insight into which option, or combination of options, is likely to give the biggest risk improvement.
- PSA allows the risk reduction potential of a proposed design change to be assessed as part of the assessment of gross disproportion, that is, it allows the risk reduction to be compared with the cost of the design change in terms of effort, time or money.

One of the most powerful ways that PSA can be used in the overall ALARP analysis is to use a list of contributors (basic events, system failures, human errors, etc.) ranked according to risk importance measures. There are a number of importance measures routinely used in PSA studies but the two that are most useful in ALARP assessments are Fussell-Vesely (FV) Importance and Risk Achievement Worth (RAW). FV gives a measure of the sensitivity of the PSA results to changes (improvements) to individual contributors in terms of their potential contribution to risk reduction, and the sensitivity of the PSA results to changes (degradation) to individual contributors in terms of their potential contribution to risk increase. RAW gives a measure of the increase in frequency (RAW $\geq$ 1) of a plant damage state if a basic event failure probability was 1 (e.g. taking a component out of service or it being in a failed state).

The FV importance used in this study is a normalised (0 $\leq$ FV$\leq$ 1) risk reduction importance measure which ranks contributors in decreasing order of their potential contribution to risk reduction. Changes in or improvements to the contributors with the highest FV leads to the largest decrease in risk. Therefore, addressing contributors in order of decreasing FV is most likely to lead to a design whose risks are ALARP, provided the costs of improving the contributor are not grossly disproportionate. Similarly, contributors with the highest RAW have the largest contribution to overall risk and addressing those with the highest RAW first identifies contributors with the greatest potential to achieving a design whose risks are ALARP.

The full scope PSA for UK ABWR is summarised in Chapter 25: Probabilistic Safety Assessment and examples of design decisions that have come about from insights from the use of PSA are summarised in Section 28.7.2 of this chapter.

28.6.3 ALARP Insights from Beyond-Design-Basis Accident and Severe Accident Assessments

Beyond-Design-Basis Accident Assessment (BDBA) and Severe Accident Assessment (SAA) complement the DBA and PSA performed as part of fault studies and potentially identify further reasonably practicable preventative or mitigating measures beyond those identified in the DBA or PSA.

BDBA assesses events that have potential unmitigated consequences above the NSEDP Target 4 BSL but whose initiating event frequency is lower than the DB cut-off of 10^{-5}/year or whose sequence frequency is lower than the cut-off of 10^{-7}/year – see Chapter, Section 5.5. The principal rationale for these studies is to show that there is not a sudden increase in risk as the frequency of events drops below the DB cut-off, that is, there are no “cliff-edge” effects. From an ALARP point of view, the main insight from BDBA is in highlighting events or event sequences where there is a potential discontinuity in the risk profile.

SAA assesses events that potentially result in large releases, that is, releases with unmitigated consequences above the highest NSEDP Target 4 BSL for public dose of 100 mSv or events with unintended relocation of radioactive material. The analysis of large releases provides a suitable basis for accident management strategies and emergency plans that contribute to the overall demonstration that risks from the plant are ALARP. Of particular importance in SAA are events with early releases, that is, releases with unmitigated consequences above 100 mSv but with insufficient time for effective countermeasures to be put in place.

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The BDBA and SAA for the UK ABWR are described in Chapter 26: Beyond Design Basis and Severe Accident Analysis and examples of design decisions that have come about from insights from the BDBA and SAA are given in Section 28.7.3 of this chapter.

28.7 The Development of the UK ABWR Design during GDA

In the previous sections, the development of the BWR, using internationally recognised good practice, into the standard ABWR, the Japanese ABWR fleet and, finally, to some aspects of the UK ABWR design has been described. It has been shown that each development has benefitted from insights from operational experience around the world, from lessons learned and from improvements in understanding of materials, chemistry, human performance and many other areas of technology. At each stage of the long development from the earliest BWRs, each successive evolutionary change has implemented options that were considered to be best practice at the time to reduce risks so far as is reasonably practicable.

Thus, the UK ABWR design described in this Generic PCSR can be justly claimed to be based on good international practice. This is the foundation for the demonstration that the risks from the construction, commissioning, operation and decommissioning of the UK ABWR are ALARP. This section summarises the development of the UK ABWR during the GDA process, particularly how the iterative ALARP process has contributed to the reduction of risks in various areas, and demonstrates that the overall risks from the construction, commissioning, operation, maintenance and decommissioning of UK ABWR are ALARP.

The remainder of this section provides a high level summary description of design options that have been considered during GDA and are being implemented through the formal design change process. There are four batches of design changes that have been submitted to the regulators throughout GDA. For ease of description, some of the design changes have been grouped as they cover a single topic.

A comprehensive description of the UK ABWR design incorporating these design changes is given throughout the PCSR and in the more detailed supporting evidence referenced in all of its chapters.

28.7.1 Design Decisions Arising from the Application of the ALARP process

The ALARP process described in the previous section has been applied throughout GDA and has led to a number of deign decisions or confirmations of design choices. Many design changes have come about as a result of the application of the ALARP process, either from the adoption of UK RGP or from options studies. A number of selected design changes are listed in Section 28.7.4. This section describes some of the major design changes or design confirmations that have arisen as a result of the application of the ALARP process in GDA.

New Class 1 Platform

UK RGP and therefore regulatory expectation is that the Class 1 safety system logic control (SSLC) is diverse from all safety Classes 2 and 3 control and instrumentation systems. The reference ABWR design is based on the use of similar technology for the SSLC and many of the classes 2 and 3 control and instrumentation systems. To meet UK RGP for the diversity of the safety Class 1 SSLC from all other control and instrumentation systems a number of different options were analysed, including continuing with microprocessor based technology. This analysis resulted in the outcome that the technology selected for the SSLC platform is to be designed using field programmable gate array (FPGA) technology. It is important to note that the FPGA integrated circuits selected donot include embedded microprocessor technology.

The Class 1 platform and related systems are described in Chapter 14 “Control and Instrumentation”..

Back-up Building

SAA assesses events that potentially result in large releases, that is, releases with unmitigated consequences above the highest NSEDP Target 4 BSL for public dose of 100 mSv or events with unintended relocation of radioactive material. The analysis of large releases provides a suitable basis for accident management strategies and emergency plans that contribute to the overall demonstration that risks from the plant are ALARP. Of particular importance in SAA are events with early releases, that is, releases with unmitigated consequences above 100 mSv but with insufficient time for effective countermeasures to be put in place.

Handling, storage and export of spent fuel

As a result of a number of ALARP studies, a number of changes have been made to the way that spent fuel is handled, stored and exported from the Reactor Building:

- As a result of an ALARP options study the FPC system has been redesigned (design change A-5) so that it is Class A1 so that its failure and the consequential SFP boiling fault are infrequent faults.

- The Fuel Handling Machine (FHM) and Reactor Building Crane (RBC) have been redesigned to provide Class 1 protection separate from the Class 3 control system in accordance with UK RGP. The design of both has adopted the ASME NOG-1 design code and both have a number of modifications to comply with the ACoP associated with the Lifting Operations Lifting Equipment Regulations (LOLER) to meet UK RGP.

- As a result of an ALARP options study impact limiters are provided to protect against potential cask drops in the reactor building hoist well and in the cask pit (see also Section 28.7.4.1 below).

- As a result of an ALARP options study a cask stand has been introduced into the Cask Pit so that the cask lid can be welded in that location to avoid lifting the cask with the lid not completely fixed, and to modify the access point from the SFP to the Cask Pit to avoid having to lift items over stored fuel.

- A major ALARP options study confirmed that both the location of the SFP and the export route for spent fuel from the Reactor Building constituted the ALARP arrangement.

- A design change has been introduced so that fuel handling above stored fuel is avoided.

Water chemistry and materials selection

Water chemistry is a dominant controlling mechanism affecting structural integrity (stress-corrosion cracking (SCC), flow-assisted corrosion (FAC), etc) and radiation doses via the effect on the redistribution and removal of radioactive species. These mechanisms often compete, for example, the introduction of hydrogen minimises SCC but leads to a substantial increase in doses through the partition of nitrogen-16 into the steam phase in the reactor. A major ALARP options study confirmed that the chemistry regime that leads to risks that are ALARP is the regime specified by EPRI as best practice, namely, hydrogen water chemistry with noble metal addition and depleted zinc addition. The chemistry regime is described in detail in Chapter 23 “Reactor Chemistry”..

A further options study developed the chemistry regime for commissioning, start-up and shut-down.

Following on from the decision regarding the chemistry regime, a number of options studies reviewed the selection of materials specified for the manufacture of different systems. Again, the competition was often between structural integrity issues and doses through the source term. The options studies led to the optimum choice of materials for each system that reduced risks ALARP. Details of material selection are given in Chapter 8 "Structural Integrity".

Off gas system

In the Japanese design of ABWR, isolation of the Off Gas system in the case of a fault is performed manually. Early in GDA, the assessment of DB faults for the Off Gas system led to the conclusion that automatic isolation would significantly reduce risk and would be reasonably practicable. It was therefore decided to implement this change.

Radwaste and other systems

Japanese design philosophy and UK design philosophy for process plant have a number of differences which have led to the redesign of part of the liquid Radwaste system to bring them in line with UK RGP. In particular, some processes have been changed to ensure that liquids flow back in to the process rather than flowing to drain or Radwaste.

A separate options study demonstrated that operational safety considerations outweighed considerations of dose during maintenance and that the provision of a bottom drain line in the RPV was ALARP, confirming the reference design.

28.7.2 Design Decisions Arising from Insights from the Use of PSA

PSA has been used in the development of BWRs starting with the Reactor Safety Study (also known as the Rasmussen Report or WASH-1400) [Ref-14] published by US NRC in 1975, which attempted to provide a comprehensive treatment for a representative BWR. The methodology has developed considerably since then so that these results are not directly comparable to modern PSAs. However, the work provided insights that informed subsequent BWR development.

28.7.3 Design Decisions Arising from Insights from the BDBA and SAA

The requirement to introduce a back-up building to the reference ABWR design was in response to the lessons learned from the Fukushima accident. This facility provides diverse and fully independent water injection systems for the nuclear power plant (NPP). It also provides other major safety support functions in the building such as severe accident countermeasures.

Assessment of BDB Accidental Aircraft Impact (AIA) have led to the enhancement of the design of the Reactor Building and requirements on the position of the Control Building and Back-up Building in relation to other buildings to mitigate the effects of AIA. See also paragraph 28.7.4.1 below.

A number of further evaluations have been performed in relation to severe accident mitigation to show that the UK ABWR is consistent with Relevant Good Practice and the ALARP principle.

There are eight topics considered:

- Methods / technologies for confining a molten core,
- Methods of core or containment cooling,
- Further increasing grace / response times,
- Further capturing / reducing fission products inside containment,
- Design of containment head flange and other systems to protect from containment leakage,
- Method for flammable gas control,
- Containment venting, and
- Additional severe accident management measures.

In each topic, options have been considered against international guidance (IAEA, EPRI, EC) on what can be regarded as RGP on severe accident measures/strategies and evaluated to determine any that are reasonably practicable. Details are given in Chapter 26 and summarised in the corresponding ALARP justification section 26.7.

The demand for continuous improvement in safety of nuclear power plants has led to an international requirement for new nuclear power plant designs to provide a demonstration of 'Practical Elimination of Early or Large Fission Product Release' as a result of severe accidents. For UK ABWR, the practically elimination of large or early releases has been demonstrated for internal initiators for reactor at-power, shutdown and the SFP. The basis of the demonstration has principally been based on a combination of consideration of the diverse design measures and defence in depth provisions in the design supported by results from extensive PSA and severe accident analysis to show that such releases are "extremely unlikely with a high degree of confidence".

A specific implication of the determination that a particular risk has been practically eliminated is that there is no requirement for a specific design solution to be implemented to address that risk. Thus, the demonstration that early and large releases from severe accidents are practically eliminated is a key indicator that the risks from severe accidents are ALARP.

28.7.4 Summary of Design Changes Considered during GDA

The remainder of this section provides a high level summary description of design options that have been considered during GDA Step 3 and Step 4 and are being implemented through the formal design change process. There are four batches of design changes that have been submitted to the regulators throughout GDA. For ease of description, some of the design changes have been grouped as they cover a single topic. Alongside each is given an indication of the reason the design option was include:

- NSEDP the option was included deterministically to satisfy one or more NSEDPs
- RGP the option was included deterministically to meet UK Relevant Good Practice
- ALARP the option was included as the result of a specific ALARP study during GDA
- PSA the option was included as a result of an insight gained from the PSA.

28.7.4.1 Design Changes Proposed and Agreed in GDA Steps 2 and 3

Proposals to automate some operator manual actions (ALARP, RGP)
Automatic logic that already forms part of the US ABWR design, for example automatic initiation of FLSS and realignment of RHR, is being implemented.

Design changes to the electrical systems architecture (PSA, RGP)
Three major changes electrical system architecture are:

- Introducing a Diverse Additional Generator (DAG) to provide additional support for the 3 ED/Gs for added protection in the case of LOOP or SBO,
- Diversifying the switchboard technology through having different voltage levels between the reactor building supplies and the back-up building supplies, and
- The addition of two motor generator sets to provide capability to meet national grid code fault ride through performance criterion.

The first two options protect against the loss of off-site power (LOOP) events and a number of on-site maintenance human errors and many other sources of common cause failures. They also have an effect on the DC power failure events by making the plant less reliant on batteries. The combined effect provides a significant contribution to reducing risk. The third design change avoids a reactor scram and shutdown should a major transient fault occur on the off-site national grid electrical power supply system.

Location of EDGs (RGP)
A fourth design change for the on-site electrical power supply is the relocation EDGs from the R/B to three separate buildings. The purpose of this is to reduce the risk of fire in the R/B as the diesel day tanks are also moved.

Proposals for changes to ECCS - "N+2" criteria compliance (NSEDP, RGP)
The safety system configuration for the standard ABWR doesn't fully meet the UK RGP when judged against the UK application of the single failure criterion. The interpretation of the single failure criterion is that one division is assumed to be in a test/maintenance state, while the other division is assumed to suffer a single failure occurrence; this is referred to as the 'N+2' criterion. For example on the emergency core cooling system (ECCS) if Division B and C are not operational (i.e., unavailable because of failure or maintenance of two of the three EDGs), leaving only Division A which has RCIC and LPFL/RHR(A) the steam turbine driven RCIC cannot be credited for long-term LOCA makeup when a LOCA break occurs in LPFL/RHR(A) (i.e. in the feedwater line that RHR(A) interfaces with for RPV injection). Therefore, two design changes have been implemented to enable the UK ABWR to accomplish full "N+2" criteria compliance:

- Introduction of a tie line between LPFL(A) and RCIC with corresponding valves and C&I to maintain the N+2 architecture in the case of a break of the FW line to which LPFL(A) is connected, and
- Replacement of the 50 percent capacity heat exchangers in the RHR system with 100 percent heat exchangers (see Figure 28.7-1, red circles indicate 100 percent capability). Subject to detailed design studies this may involve using a different type of heat exchanger because of layout requirements.

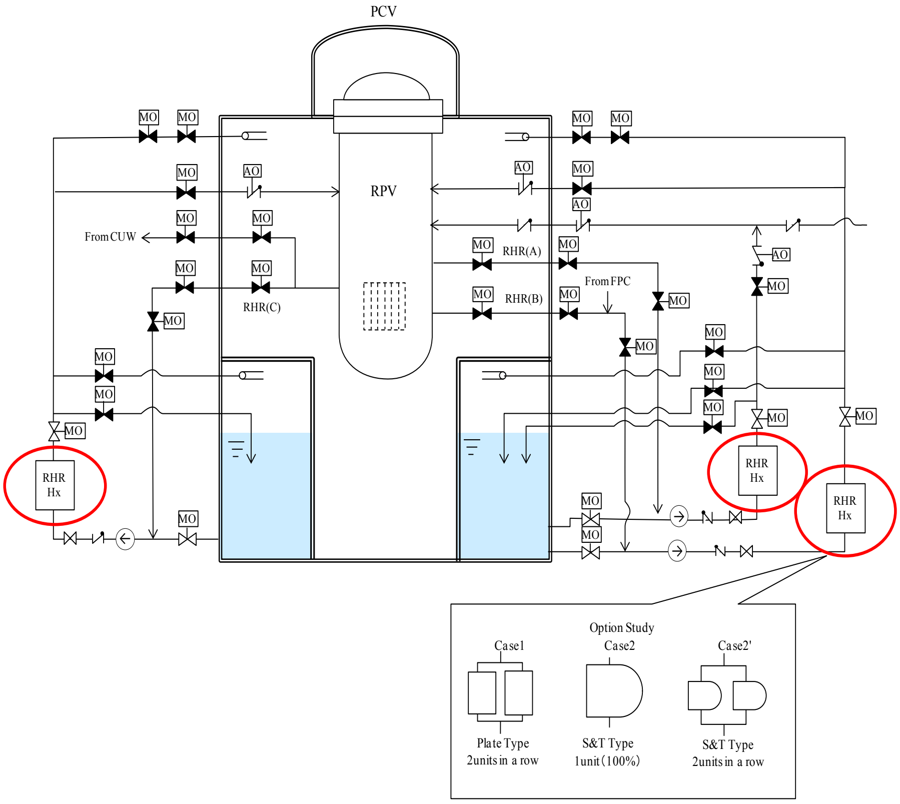

Figure 28.7-1 RHR Change from 50 percent to 100 percent (ALARP, RGP)

A number of design changes are proposed to increase protection against internal hazards. One of the identified hazards is the possibility of dropping a spent fuel cask in the cask pit alongside the SFP. It is proposed to increase the depth of the pit and to install impact limiting material to protect the cask and pit in the event of a cask drop. The pit is made deeper in order to maintain the depth of water above the cask for shielding during fuel transfer to the cask.

To comply with UK RGP, a design change has been implemented to minimise the number of doors in Class 1 safety barriers inside the R/B and to introduce double fire doors at the remaining locations where reasonably practicable. Where double doors are not reasonably practicable, single doors have alarms to annunciate in the MCR if they are open. Furthermore, for conservatism, hazard analyses in the corresponding areas assumes that the single doors are open.

ATWS countermeasure (NSEDP, ALARP)
To meet UK RGP anticipated transient without scram (ATWS) events have to be considered as infrequent design basis events. For all design basis postulated initiating events automation is preferred to relying on manual action. To meet this principle design changes have been implemented to automate the following systems:

- Standby liquid control (SLC) system,
- ADS inhibit function, and
- Feedwater injection rate.

Scope of VHI Components (ALARP, RGP)

Hitachi-GE classifies and justifies RPV main welds, MS piping in RCCV, MSIV, their RCCV penetrations and others (if necessary), as VHI components in the UK ABWR design in order to comply with RGP.

Countermeasure for beyond design basis event of aircraft impact accident (AIA) (ALARP)

The following countermeasures have been implemented for the UIK ABWR:

- The reactor building (R/B) external envelope has been enhanced to limit physical damage resulting from an aircraft impact. Shock and vibration, fire and physical damage evaluation has been undertaken for the whole building,
- The control building (C/B) is protected by the surrounding buildings to prevent direct impact by an aircraft,
- The back-up building (B/B) is physically separated from the R/B and C/B by sufficient distance to prevent simultaneous damage by the same aircraft,
- The turbine building, radwaste building and services building has been assessed using NEI7-13 methodology to confirm that their structures are sufficiently robust to protect the C/B, and
- Radiological assessments have been carried out to confirm that the dose arising from damage to the radwaste and turbine building is within acceptable limits.

C/B HVAC System Design (RGP)

Control building heating, ventilation and air-conditioning (HVAC) system design has changed to introduce the following improvements (see figure 28.7-2):

- Full implementation of the UK approach to the single failure criterion by implementing an N+2 architecture,
- Introduction of a fully diverse Class 2 HVAC system, and
- Improved segregation and separation between each independent division and between the A1 and A2 HVAC systems.

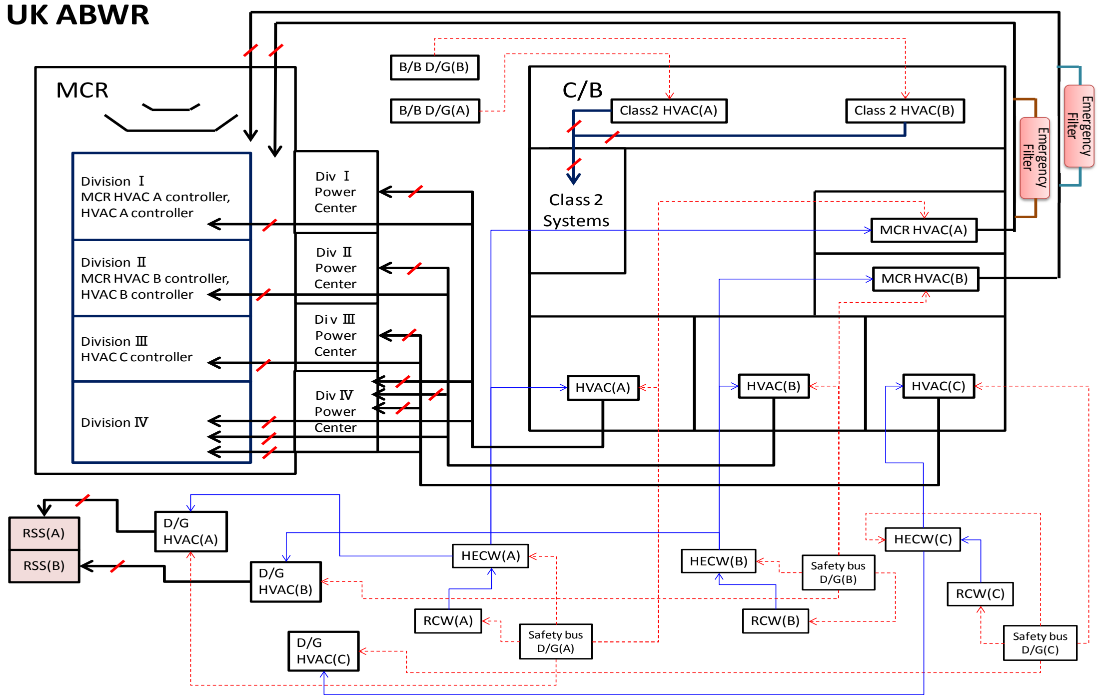

Figure 28.7-2 C/B HVAC System Design

MCR Size/layout (RGP)
To enhance segregation of safety system logic controller and to include additional and larger display panels in the main control room, the control building has been widened by 10.5m. It also allows for the provision of additional air-conditioning equipment (see S-16 under batch 3 design changes).

Safety Auxiliary Panels (NSEDP)
A hardwired safety auxiliary panel is provided to display key parameters covering the fundamental safety functions of control of reactivity, fuel cooling, long term heat removal and confinement/containment of radioactive materials. This has been provided as a defence against the common cause failure of the digital safety system and logic controller (SSLC).

Countermeasure for the removal of Hydrogen in PCV (ALARP)
This design change changes the design of the flammability control system (FCS) by introducing passive autocatalytic recombiners (PARs). An analysis of options showed that the PARs had significant advantages over other methods of removing hydrogen such as the FCS. It requires no power, no operator action and no fission products are released to the environment etc.

Countermeasure for Core instability (RGP)
To ensure that the reactor core does not become unstable under any foreseeable conditions, the UK ABWR installed the following countermeasures. These are the provision of:

- An automatic selected control rod run in (SCRRI) function,
- An automatic thermal power monitor (TPM) scram function,
- A manual alternative rod insert (ARI) function, and
- A manual standby liquid control (SLC) system function.

Removal of New Fuel Vault (ALARP)
A review of the loading and storing of new fuel has shown that the new fuel vault (NFV) is not required. This conclusion is also borne out by the fact that where NFVs are provided in Japanese BWRs these facilities are not used and have become redundant. This design change removes the NFV.

Changes to the RCIC Pump/Turbine Integrated Pump (TWL type) (ALARP)
Improvements to the design of the reactor core isolation cooling (RCIC) system steam driven turbine pump have been implemented since Kashiwazaki Kariwa Units 6 and 7 were commissioned. This design change introduces the turbine water lubricated (TWL) steam driven turbine as the prime mover for the RCIC. It has a number of advantages:

- It requires less DC power,
- It is more compact that the reference steam turbine,
- It conforms to major international codes, and
- Good international operational experience has accrued due to its use in PWRs.

SSLC Class 1 HMI (RGP)
This design change replaces the Japanese ABWR safety Class 2 Human Machine Interface (HMI) with a safety Class 1 HMI designed to the same standards as the safety Class 1 safety system logic control (SSLC) and uses the same field programmable gate array (FPGA) technology. The reason for this change is that it is in two-way communication with the SSLC and provides important support functions.

Off-gas Automatic Isolation (ALARP)
This design changes introduces enhanced automatic isolation in the event of any failed (rupture) section of piping or charcoal absorber tanks within the off-gas treatment system. The following describes some of the key changes:

- The addition of an automatically actuated isolation valve on the turbine auxiliary steam system line,
- Enhanced radiation detection and actuation systems employing typically two-out-of-three logic to have low failure on demand probabilities and spurious failure frequencies, and
- Additional interlocks to prevent hydrogen combustion.

Turbine type 54inch Application (not safety related)
The reference Japanese ABWR design uses a 52inch turbine blade design (TC6F-52) in the low pressure turbine. Developments in turbine design have resulted in a 54inch turbine blade design (TC6F-54) that has resulted in improvements in vibration mitigation performance and enhanced component verification. Hence the design change for the UK ABWR is to replace the TC6F-52 with the TC6F-54.

Category and Class of FPC (ALARP)
Established UK RGP is that the fuel pool cooling function is safety category A, which as a minimum requires a safety Class 1 system to fulfil that function. This design change enhances the reference FPC design through making significant design changes to, for example, equipment segregation, provision of an additional cooling circuit and adding a third pump etc. making the system fully single failure tolerant. It also has the benefit that it does not need to call on support from the residual heat removal system for all modes of operation other than refuelling outages when a full core offload is in the fuel pool.

Main Steam Tunnel Room Design (ALARP, RGP)
This design change eliminates the possibility of a 'domino effect' from a single guillotine fracture of either one of the main steam lines or feedwater lines located in the main steam tunnel room (MSTR). This design change ensures that a single pipe failure will not result in consequential multiple domino failures through the process of pipe-whip. It achieves this by enhancing the existing piping support structures on the main steam lines and main feedwater lines located within the MSTR.

Installation of Safety Auxiliary Control Systems (SACS) (RGP, PSA)
In the Japanese ABWR the SSLC performs a wide range of safety functions. The UK ABWR fault analysis (see Generic PCSR Chapter 24 "Design Basis Analysis", GA91-9101-0101-24000, Generic PCSR Chapter 25 "Probabilistic Safety Assessment", GA91-9101-0101-25000, and Generic PCSR Chapter 26 "Beyond Design Basis and Severe Accident Analysis", GA91-9101-0101-26000) has shown that these functions whilst mainly Category A also include Category B and Category C functions. For the UK ABWR separation of each category of safety function is an important goal to be implemented, so far as is reasonably practicable. This design change removes the lower category safety functions from the SSLC and places them in a new system called the safety auxiliary system (SACS) which is designed to safety Class 2 standards using the same FPGA technology as the safety Class 1 SSLC.

Chloride Ingress Protection (RGP)
The aim of this design change is to keep the water conductivity in the reactor to acceptable levels even in the event of a guillotine failure of a condenser tube or multiple condenser tube leakages. This design change provides a high integrity interlock system which stops water flow to the reactor by tripping the high pressure condensate pumps which in turn, through an existing interlock, trips the reactor feedwater pumps. The interlock system will also switch the CRD water cooling source to the condensate storage tank by closing the condenser spill-over valve. A two-out-of-three voting interlock system is used to attain low probability of failure-on-demand and low spurious failure frequencies.

C&I Architecture (RGP)
To meet UK RGP on independence, full separation between safety Class 1 control systems and other control systems of lower safety classes is required. UK RGP covers not only physical and electrical separation but also separation of data communications. This design change (see Figure 28.7-3) enforces one way communication from the Class 1 systems to the lower safety class systems through the use of data diodes and these devices use inherent physics of the device to ensure that the lower class device cannot interfere with the operation of the higher safety Class 1 system regardless of failure mode. In addition data diodes are used to isolate the plant computer system from all external networks, once again meaning that information cannot flow into the control system network. This design change not only improves safety it is also one of a number of important measures to enhance cyber security.

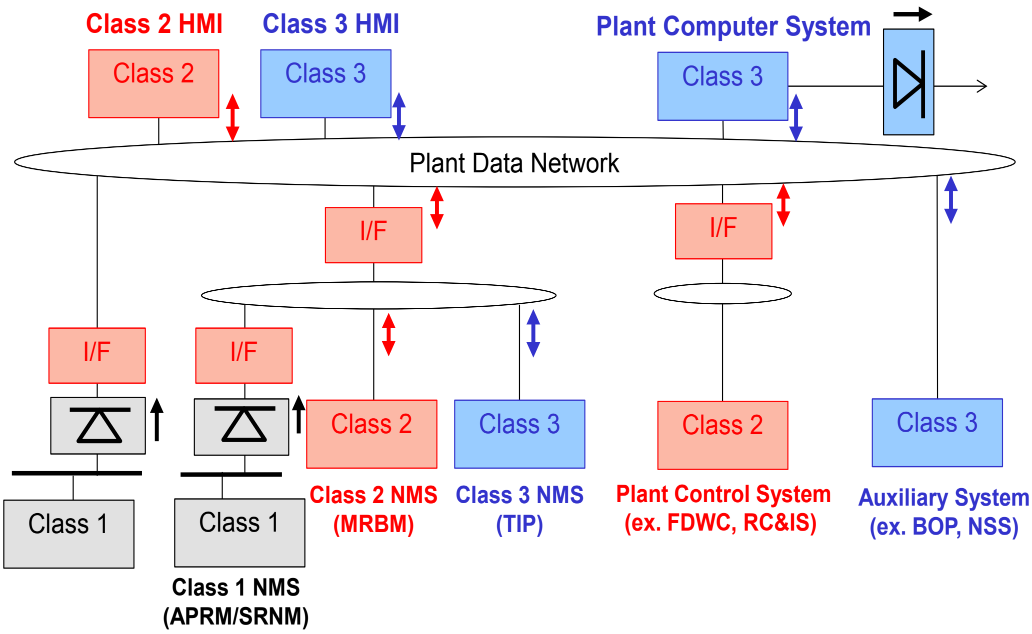

Figure 28.7-3 C&I Architecture

SSLC ADS (RGP)
The reference design uses two duplex safety logic units (SLUs) from the SSLC to actuate the automatic depressurisation system (ADS). A failure modes and effects analysis has shown that this design arrangement does not meet the N+2 interpretation of the single failure criterion. Therefore, to meet UK RGP, a third division of duplex SLUs has been added as well as adding another solenoid valve to each of the seven safety relief valves that constitute the ADS (see Figure 28.7-4).

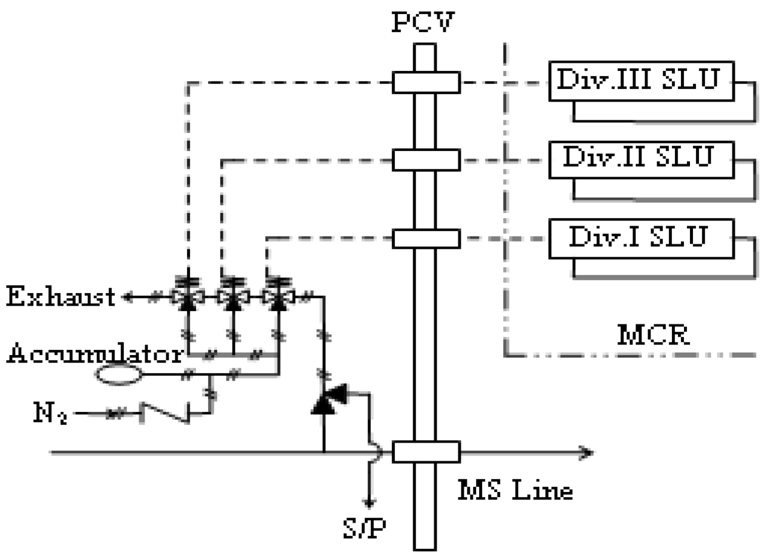

Figure 28.7-4 SSLC ADS

Safe Change HEPA Filter Units (RGP)
UK RGP has an expectation that safe change high efficiency particulate air (HEPA) filters are used for exhaust air treatment facilities. This design change introduces:

- Safe change type casing for exhaust air treatment facility to allow safe replacement of HEPA filters, and
- The extraction system for radiation controlled C2 areas use a single HEPA filtration stage whereas C3 areas use a double HEPA filtration stage.

Frost Coil and Coalescer Filter (RGP)
To meet UK RGP frost coils and coalescer filters are provided for the once-through ventilation systems proposed for the UK ABWR (see Figure 28.7-5).

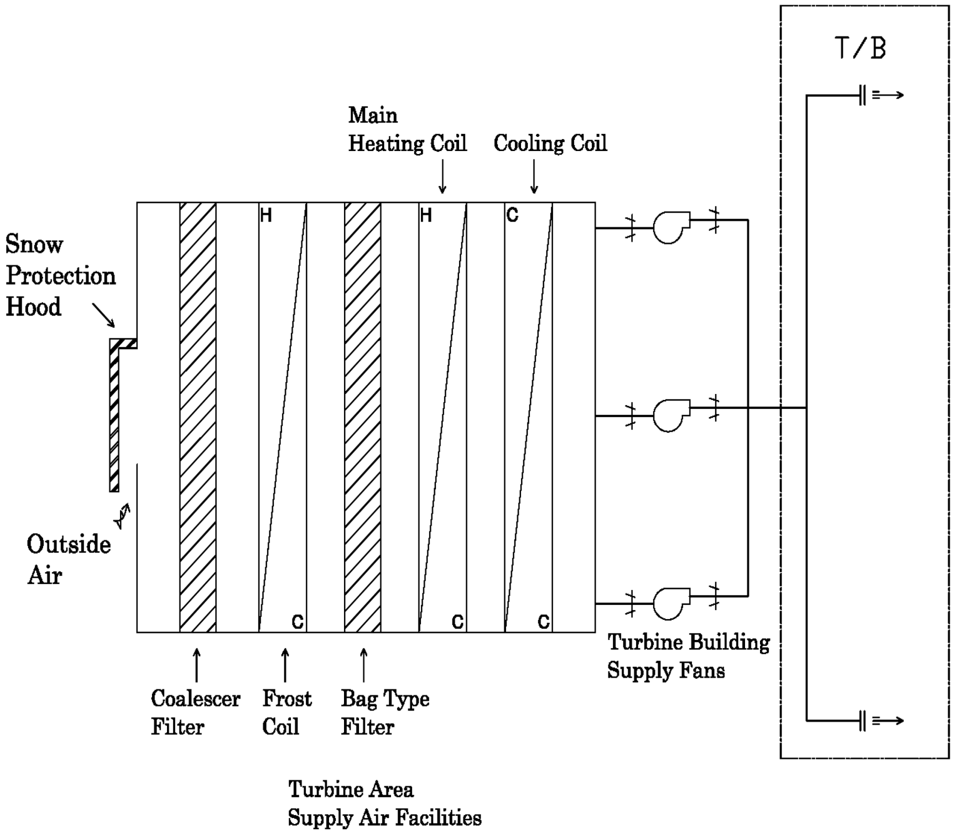

Figure 28.7-5 Frost Coil and Coalescer Filter

28.7.4.2 Design Changes Proposed in Step 4

There are a number of design changes proposed in early Step 4 and /or during the detailed assessments in Step 4. These design changes have been discussed with the regulators and implemented through the formal design change process. Below is a list of some of the relevant design changes.

28.8 Conclusions

This chapter has traced the development of the UK ABWR from the earliest BWRs and ABWR designs currently operating in Japan. It has shown how each successive design has adopted the good practice available at the time, not only to increase power output and economy but also to increase reliability and reduce risks as evidenced by the reduced Core Damage Frequency as compared with earlier designs and the operating JABWRs.

Lessons learned from the 2011 Fukushima accident have led to a number of safety improvements, particularly in the resilience of the design to external events by increasing margins before failure and providing safety support which is resilient and flexible.

As the UK ABWR design has developed during GDA, there have been a number of exercises that have led to changes to the design:

ALARP exercises have led to UK RGP being applied and possible risk-reduction options being identified and assessed to determine those that may be reasonably practicable.

Further, the full-scope Probabilistic Safety Assessment (PSA) developed during GDA has shown that the risks from UK ABWR are extremely low and meet the targets given in the NSEDPs. These low risks mean that cost threshold values for trouble, time and money associated with risk-reduction measures would be very low. Nevertheless, a systematic procedure, based on the PSA results, has been used to identify further possible risk-reduction measures.

Because of the low cost threshold, identified options have only been identified for further consideration if they satisfied one (or more) of the following criteria:

- The option significantly impacts an identified plant vulnerability,
- The option significantly reduces overall plant risk, and
- The option significantly reduces overall uncertainty.

This process has led to a few insights that are worth considering to prompt further design improvements although the analysis has given confidence that the defence-in-depth and diversity included in the GDA Design Reference is more than adequate.

A third exercise has used the results of Beyond-Design-Basis Assessment (BDBA) and Severe Accident Assessment (SAA) to identify measures that would reduce risks in the extremely unlikely event of an accident of this type occurring. Again, a small number of potential improvements have been identified for further consideration.

Consideration of the small number of potential design improvements arising from insights from the PSA, BDBA and SAA indicates that the cost of most of the identified changes would be grossly disproportionate to the potential risk reduction, in the case of SAA particularly since large and early releases from SAs are shown to be practically eliminated. These options are therefore not reasonably practicable. The very few remaining options are being progressed in more detail to determine if they are reasonably practicable to include in the design.

The above paragraphs show that every possible means has been systematically employed to identify reasonably practicable improvements to the design. All that have been identified as reasonably practicable have been included in the design or are being put through the process to assess them in more detail.

We can therefore state with confidence that, within the bounds of GDA, there are no further reasonably practicable improvements that could be implemented to reduce the residual risks and therefore that the final UK ABWR Design Reference [Ref.-13] has reduced risks SFAIRP.

28.9 References

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Appendix A. Example of the use of MADA in ALARP assessments

This Appendix gives an example of the use of the methodology based on multi-attribute decision analysis (MADA) in UK ABWR ALARP studies where there are multiple, sometimes competing, risks, risk indicators and cost indicators.

The example presented here is a summary of the MADA methodology applied in Reactor Chemistry (See Chapter 23 for more details). Using workshops involving representatives from all relevant technical areas, risk indicators (attributes or criteria) are identified and grouped into technical areas or themes. For Reactor Chemistry, the list of such indicators includes:

Risk Indicators
Source term theme
N-16 dose
Corrosion product deposition
Activation
Radioactive waste
Risk of "hot spots"
Materials theme
Stress corrosion cracking
Flow assisted corrosion
Environmental assisted corrosion
Oxide layer restructuring
CRUD generation and mobilisation
Fuel theme
…

In the workshop, identified options are scored against risk indicators, attributes or criteria according to how much the particular risk indicator, attribute or criterion applies to that option. The scoring scheme used is

+3 Very much better than base case: risk very much less or cost* extremely low for same risk reduction

+2 Much better than base case: risk much less or cost* very much less for same risk reduction

+1 Better than base case: risk less or cost* much less for same risk reduction

0 Same as base case: risk the same or cost* same or similar for same risk reduction

-1 Worse than base case: risk more or cost* much more for same risk reduction

-2 Much worse than base case: risk much more or cost* very much more for same risk reduction

-3 Very much worse than base case: risk very much more or cost* extremely high for same risk reduction

Risk indicators, Attributes or Criteria in a technical area or themes are then weighted according to their contribution to the technical area or theme (Total = 100% within the theme) and technical areas or themes are weighted according to their importance to the ALARP case (Total = 100%)

The methodology adopted for UK ABWR is based on multi-attribute decision analysis (MADA). Risk indicators and cost indicators are identified and weights assigned to reflect the relative importance of each. Cost indicators are assigned lower weights to account for gross disproportion. Each option is then scored according to how well it addresses each risk indicator or how much each cost indicator applies. Finally, for each option the total weighted score is calculated. This allows options to be compared taking into account the multiple risk and cost indicators. The option with the highest score is most likely to be the ALARP option.