UK ABWR Generic Design Assessment

Genesis of ABWR design
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**Abbreviations and Acronyms**

<table>
<thead>
<tr>
<th>Abbreviations and Acronyms</th>
<th>Description</th>
</tr>
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<tbody>
<tr>
<td>3D-CAD</td>
<td>3D-Computer Aided Design</td>
</tr>
<tr>
<td>3D-CAE</td>
<td>3D Computer Aided Engineering</td>
</tr>
<tr>
<td>ABWR</td>
<td>Advanced Boiling Water Reactor</td>
</tr>
<tr>
<td>AC</td>
<td>Alternating Current</td>
</tr>
<tr>
<td>ADS</td>
<td>Automatic Depressurization System</td>
</tr>
<tr>
<td>AET</td>
<td>Advanced Engineering Team</td>
</tr>
<tr>
<td>AOO</td>
<td>Anticipated Operational Occurrences</td>
</tr>
<tr>
<td>BOP</td>
<td>Balance of Plant</td>
</tr>
<tr>
<td>BWR</td>
<td>Boiling Water Reactor</td>
</tr>
<tr>
<td>CDF</td>
<td>Core Damage Frequency</td>
</tr>
<tr>
<td>CFR</td>
<td>Code of Federal Regulations</td>
</tr>
<tr>
<td>CR</td>
<td>Control Rod</td>
</tr>
<tr>
<td>CRD</td>
<td>Control Rod Drive</td>
</tr>
<tr>
<td>CRGT</td>
<td>Control Rod Guide Tube</td>
</tr>
<tr>
<td>CST</td>
<td>Condensate Storage Tank</td>
</tr>
<tr>
<td>DC</td>
<td>Design Certification</td>
</tr>
<tr>
<td>DCD</td>
<td>Design Control Document</td>
</tr>
<tr>
<td>DCIS</td>
<td>Distributed Control Information System</td>
</tr>
<tr>
<td>DCR</td>
<td>Design Certification Rule</td>
</tr>
<tr>
<td>DG</td>
<td>Diesel Generator</td>
</tr>
<tr>
<td>DiD</td>
<td>Defence in Depth</td>
</tr>
<tr>
<td>DZO</td>
<td>Depleted Zinc Oxide</td>
</tr>
<tr>
<td>EBWR</td>
<td>Experimental Boiling Water Reactor</td>
</tr>
<tr>
<td>ECCS</td>
<td>Emergency Core Cooling System</td>
</tr>
<tr>
<td>ECP</td>
<td>Electrochemical Corrosion Potential</td>
</tr>
<tr>
<td>EDG</td>
<td>Emergency Diesel Generator</td>
</tr>
<tr>
<td>EPC</td>
<td>Engineering Procurement and Construction</td>
</tr>
<tr>
<td>EPRI</td>
<td>Electric Power Research Institute</td>
</tr>
<tr>
<td>ESBWR</td>
<td>Economic Simplified Water Reactor</td>
</tr>
<tr>
<td>EVESR</td>
<td>ESADA Vallecitos Experimental Superheat Reactor</td>
</tr>
<tr>
<td>FCC</td>
<td>Fuel Cycle Costs</td>
</tr>
<tr>
<td>FDA</td>
<td>Final Design Approval</td>
</tr>
<tr>
<td>FMCRD</td>
<td>Fine Motion Control Rod Drive</td>
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### Abbreviations and Acronyms (Cont’d)

<table>
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<th>Abbreviations and Acronyms</th>
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</tr>
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<tr>
<td>GDA</td>
<td>Generic Design Assessment</td>
</tr>
<tr>
<td>GEH</td>
<td>GE Hitachi Nuclear Energy</td>
</tr>
<tr>
<td>Hitachi-GE</td>
<td>Hitachi-GE Nuclear Energy, Ltd.</td>
</tr>
<tr>
<td>HWC</td>
<td>Hydrogen Water Chemistry</td>
</tr>
<tr>
<td>HPCI</td>
<td>High Pressure Core Injection System</td>
</tr>
<tr>
<td>HPCS</td>
<td>High Pressure Core Spray System</td>
</tr>
<tr>
<td>I&amp;C</td>
<td>Instrumentation and Control</td>
</tr>
<tr>
<td>IASCC</td>
<td>Irradiation Assisted Stress Corrosion Cracking</td>
</tr>
<tr>
<td>IGSCC</td>
<td>Intergranular Stress Corrosion Cracking</td>
</tr>
<tr>
<td>IHSI</td>
<td>Induction Heating Stress Improvement</td>
</tr>
<tr>
<td>KK-6</td>
<td>Kashiwazaki-Kariwa Nuclear Power Station Unit 6</td>
</tr>
<tr>
<td>KK-7</td>
<td>Kashiwazaki-Kariwa Nuclear Power Station Unit 7</td>
</tr>
<tr>
<td>LLC</td>
<td>Limited Liability Company</td>
</tr>
<tr>
<td>LOCA</td>
<td>Loss-of-Coolant-Accident</td>
</tr>
<tr>
<td>LPCRD</td>
<td>Locking Piston Control Rod Drive</td>
</tr>
<tr>
<td>LPCF</td>
<td>Low Pressure Core Flooder</td>
</tr>
<tr>
<td>LPCI</td>
<td>Low Pressure Coolant Injection</td>
</tr>
<tr>
<td>LPCS</td>
<td>Low Pressure Core Spray</td>
</tr>
<tr>
<td>LPFL</td>
<td>Low Pressure Flooder</td>
</tr>
<tr>
<td>LUHS</td>
<td>Loss of Ultimate Heat Sink</td>
</tr>
<tr>
<td>MOX</td>
<td>Mixed Oxide</td>
</tr>
<tr>
<td>MSR</td>
<td>Moisture Separator Re-heater</td>
</tr>
<tr>
<td>MSIV</td>
<td>Main Steam Isolation Valve</td>
</tr>
<tr>
<td>NI</td>
<td>Nuclear Island</td>
</tr>
<tr>
<td>NMCA</td>
<td>Noble Metal Chemical Addition</td>
</tr>
<tr>
<td>NPP</td>
<td>Nuclear Power Plant</td>
</tr>
<tr>
<td>NSSS</td>
<td>Nuclear Steam Supply System</td>
</tr>
<tr>
<td>NUCAMM90</td>
<td>Nuclear Power Plant Control Complex with Advanced Man-Machine Interfaces 90</td>
</tr>
<tr>
<td>NWC</td>
<td>Normal Water Chemistry</td>
</tr>
<tr>
<td>NZO</td>
<td>Natural Zinc Oxide</td>
</tr>
<tr>
<td>O&amp;M</td>
<td>Operation and Maintenance</td>
</tr>
<tr>
<td>OG</td>
<td>Off-Gas System</td>
</tr>
<tr>
<td>OLNC</td>
<td>On-Line NobleChem™</td>
</tr>
<tr>
<td>PAE</td>
<td>Project Application Engineering</td>
</tr>
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</table>
## Abbreviations and Acronyms (Cont’d)

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<tr>
<th>Abbreviations and Acronyms</th>
<th>Description</th>
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<tr>
<td>PCC</td>
<td>Plant Capital Cost</td>
</tr>
<tr>
<td>PGC</td>
<td>Power Generation Cost</td>
</tr>
<tr>
<td>PPE</td>
<td>Pre-Project Engineering</td>
</tr>
<tr>
<td>PRA</td>
<td>Probabilistic Risk Assessment</td>
</tr>
<tr>
<td>PSAR</td>
<td>Preliminary Safety Analyses Report</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurised Water Reactor</td>
</tr>
<tr>
<td>QA</td>
<td>Quality Assurance</td>
</tr>
<tr>
<td>QC</td>
<td>Quality Control</td>
</tr>
<tr>
<td>RCCV</td>
<td>Reinforced Concrete Containment Vessel</td>
</tr>
<tr>
<td>RCIC</td>
<td>Reactor Core Isolation Cooling</td>
</tr>
<tr>
<td>RHR</td>
<td>Residual Heat Removal</td>
</tr>
<tr>
<td>RIP</td>
<td>Reactor Internal Pump</td>
</tr>
<tr>
<td>ROCAEC</td>
<td>Taiwan's Atomic Energy Agency</td>
</tr>
<tr>
<td>RPV</td>
<td>Reactor Pressure Vessel</td>
</tr>
<tr>
<td>SAMG</td>
<td>Severe Accident Management Guideline</td>
</tr>
<tr>
<td>SBO</td>
<td>Station Blackout</td>
</tr>
<tr>
<td>SBWR</td>
<td>Simplified Boiling Water Reactor</td>
</tr>
<tr>
<td>SCC</td>
<td>Stress Corrosion Cracking</td>
</tr>
<tr>
<td>SFP</td>
<td>Spent Fuel Pool</td>
</tr>
<tr>
<td>SHE</td>
<td>Standard Hydrogen Electrode</td>
</tr>
<tr>
<td>SRV</td>
<td>Safety Relief Valve</td>
</tr>
<tr>
<td>SUMIT</td>
<td>Spectral Unit Module Initial</td>
</tr>
<tr>
<td>TAF</td>
<td>Top of Active Fuel</td>
</tr>
<tr>
<td>TEPCO</td>
<td>Tokyo Electric Power Company</td>
</tr>
<tr>
<td>TGSCC</td>
<td>Transgranular Stress Corrosion Cracking</td>
</tr>
<tr>
<td>TI</td>
<td>Turbine Island</td>
</tr>
<tr>
<td>TPC</td>
<td>Taiwan Power Company</td>
</tr>
<tr>
<td>URD</td>
<td>Utility Requirements Document</td>
</tr>
<tr>
<td>US NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>VBWR</td>
<td>Vallecitos Boiling Water Reactor</td>
</tr>
<tr>
<td>WJP</td>
<td>Water Jet Peening</td>
</tr>
</tbody>
</table>
Table of Contents

1. Introduction .......................................................................................................................... 1

2. Development of BWRs ........................................................................................................ 2
   2.1. Principle of BWR ........................................................................................................... 2
   2.1.1. BWR Evolution ....................................................................................................... 2
   2.1.2. BWR System .......................................................................................................... 5
   2.2. ABWR Development ..................................................................................................... 9
         2.2.1. Genesis of the ABWR ....................................................................................... 9
         2.2.2. BWR Safety System Evolution Leading Up to the ABWR ......................... 10
         2.2.3. BWR Reactor Pressure Vessel Evolution Leading Up to the ABWR ........ 13
         2.2.4. BWR Reactor Internal Assembly Evolution Leading Up to the ABWR .... 18
         2.2.5. BWR Materials and Reactor Chemistry Evolution Leading Up to the ABWR .. 24
         2.2.6.1 Materials and Stress Corrosion Cracking ...................................................... 26
         2.2.6.2 Reactor Water Chemistry .............................................................................. 30
         2.2.6. BWR Containment Evolution Leading Up to the ABWR ........................... 34
         2.2.7. Electrical and I&C Evolution Leading Up to the ABWR ............................ 37
         2.2.8. Design for the Environment ............................................................................ 43
   2.3. Construction Experience .............................................................................................. 45

3. Features of ABWR ............................................................................................................. 47
   3.1. Design Concept ............................................................................................................ 47
   3.2. Main Features of ABWR .......................................................................................... 48
         3.2.1. Improved Safety with Diversity ....................................................................... 48
         3.2.2. Improved Operation and Maintenance ............................................................ 49
         3.2.3. Advanced Technology Employment ................................................................. 51
         3.2.4. Construction Time Reduction ........................................................................ 51
         3.2.5. Power Generation Costs Reduction ................................................................. 52
   3.3. Design Comparison with BWRs ................................................................................ 54

4. ABWR Practice ................................................................................................................... 55
   4.1. International Practice .................................................................................................. 55
         4.1.1. General ............................................................................................................... 55
         4.1.2. Design Certification in the U.S. ....................................................................... 55
         4.1.3. Construction Project in Taiwan ....................................................................... 55
   4.2. Hitachi-GE Nuclear Energy’s ABWR Achievements in Japan ................................ 56
         4.2.1. General ............................................................................................................... 56
         4.2.2. Kashiwazaki-Kariwa Unit 6 and 7 ................................................................. 56
         4.2.3. Hamaoka Unit 5 ............................................................................................... 56
         4.2.4. Shika Unit 2 .................................................................................................... 57
         4.2.5. Present ABWR Construction Projects in Japan ............................................ 57
   4.3. Continuous Development ............................................................................................ 59
         4.3.1. Further Optimisation ....................................................................................... 59
         4.3.2. Safety Enhancement ....................................................................................... 59

5. Conclusion .......................................................................................................................... 61

6. References ........................................................................................................................... 62
1. Introduction

The development of the Advanced Boiling Water Reactor (ABWR) began in 1978, and was first adopted in the construction of the Kashiwazaki-Kariwa Nuclear Power Station Unit 6 and Unit 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 evolution of conventional Boiling Water Reactor (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 of ABWR components and facilities were introduced relative to those of conventional BWRs; and shop tests, pre-operation tests and startup tests have demonstrated that the components and facilities meet the required functionality and corresponding standards.

Hitachi-GE Nuclear Energy, Ltd. (Hitachi-GE) has already constructed four ABWRs and has continuously invested in optimization and standardization of the ABWR design. Therefore Hitachi-GE is confident that with the experience it has accumulated in all these projects, it can successfully construct ABWR power plants.

This document provides generic information about the technical evolution of BWRs, as one of the Step-1 submissions for the UK ABWR GDA. The document will illustrate how the ABWR design has been developed based on the conventional BWR technology. The document will also describe how safety and environmental features of the ABWR have been improved and implemented.
2. Development of BWRs

2.1. Principle of BWR

The BWR is based on two fundamental principles that distinguish it from other nuclear power plant reactors: (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 1950's with the construction and operation of prototype plants at Argonne National Laboratory in Illinois (USA), the Experimental Boiling Water Reactor (EBWR) (Figure 2.1-1), and at GE's Vallecitos facility in California (USA), the Vallecitos Boiling Water Reactor (VBWR) (Figure 2.1-2).

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 2.1-1. Figure 2.1-3 shows the example of the Reactor Pressure Vessel (RPV) evolution in BWRs.

Figure 2.1-1 Argonne EBWR (Ref 2-1)  Figure 2.1-2 GE VBWR (Ref 2-1)
Figure 2.1-3 Evolution of the RPV
### Table 2.1-1 BWR Product Line Evolution (Ref 2-3)

<table>
<thead>
<tr>
<th>Product Line</th>
<th>First Commercial Operation Date</th>
<th>Representative Plant/Characteristics</th>
</tr>
</thead>
<tbody>
<tr>
<td>BWR/1</td>
<td>1960</td>
<td>Dresden 1 • Initial commercial-size BWR</td>
</tr>
<tr>
<td>BWR/2</td>
<td>1969</td>
<td>Oyster Creek • Plants purchased solely on economics • Large direct cycle plant</td>
</tr>
<tr>
<td>BWR/3</td>
<td>1971</td>
<td>Dresden 2 • First jet pump application • Improved ECCS: spray and flood capability</td>
</tr>
<tr>
<td>BWR/4</td>
<td>1972</td>
<td>Vermont Yankee • Increased power density (20%)</td>
</tr>
<tr>
<td>BWR/5</td>
<td>1977</td>
<td>Tokai 2 • Improved ECCS • Valve flow control</td>
</tr>
<tr>
<td>BWR/6</td>
<td>1978</td>
<td>Clinton (1987) • Compact control room • Solid-state Reactor Protection System</td>
</tr>
<tr>
<td>ABWR</td>
<td>1996</td>
<td>Kashiwazaki-Kariwa 6 • Reactor internal pumps • Fine-motion control rod drives • Advanced control room, digital solid-state microprocessors • Fiber optic data transmission / multiplexing Increased number of fuel bundles • Titanium condenser • Improved ECCS; high/low pressure flooders</td>
</tr>
</tbody>
</table>
2.1.1. BWR Evolution

The original BWRs incorporated an external steam drum as well as steam generators, and utilized recirculation pumps external to the RPV, although a few early BWRs utilized natural circulation. High pressure steam from the steam drum was directed to high pressure stages of the turbine, while lower pressure steam from the steam generators was directed to lower pressure stages of the turbine. Figure 2.1-4 illustrates this concept.

![Figure 2.1-4 Early Commercial BWR (Ref 2-2)](https://example.com/figure214.png)

The steam drum was replaced with steam separators and dryers within the RPV with the evolution to BWR/2, and steam generators were completely eliminated in the BWR/3 product line.

Although various improvements were incorporated with each evolution of the product line from BWR/4 to BWR/6, one key change for these products was the incorporation of jet pumps (within the RPV) and external recirculation pumps for forced recirculation of reactor core flow.

The ABWR replaced the external recirculation pumps and the jet pumps with Reactor Internal Pumps (RIPs), as well as incorporating many other improvements. Incorporating RIPs in the design...
resulted in several benefits, including containment space reduction, improved maintenance and lower personnel doses, lower recirculation flow pumping power and expanded core flow range. Incorporation of RIPs also eliminated large RPV pipe penetrations below the top of active fuel (TAF) reducing Emergency Core Cooling System (ECCS) requirements and facilitating elimination of core uncovering and fuel clad heat up for design basis Loss of Coolant Accidents (LOCA).

Figure 2.1-5 illustrates the ABWR principles of a direct steam power cycle with a simplified recirculation system.

![ABWR Direct Steam Power Cycle](image)

**Figure 2.1-5 ABWR Direct Steam Power Cycle (Example) (Ref 2-3)**

### 2.1.2. BWR System

The BWR is a type of light water nuclear reactor used for the generation of electrical power. It is the second most common type of electricity-generating nuclear reactor after the Pressurised Water Reactor (PWR). The main difference between a BWR and PWR is that in a BWR, the reactor core heats water, which turns to steam and then drives a steam turbine. In a PWR, the reactor core heats water which does not boil. This hot water then leaves the Reactor Pressure Vessel and enters a heat exchanger vessel (a Steam Generator) where the hot water from the reactor exchanges heat with a lower pressure water system, which turns to steam and drives the turbine.
The BWR uses demineralised water as a coolant and neutron moderator. Heat is produced by nuclear fission in the reactor core, and this causes the cooling water to boil, producing steam. The steam is directly used to drive a turbine, after which the steam is cooled in a condenser and converted back to water. This water is then returned to the reactor core, completing the loop. The cooling water is maintained at about 7.5 MPa, or slightly lower, so that it boils in the core at about 285°C. In comparison, there is no significant boiling allowed in a PWR because of the high pressure maintained in its primary loop - approximately 16 MPa.

The heating from the core creates a thermal head that assists the recirculation pumps in recirculating the water inside of the RPV. However, the forced recirculation head from the recirculation pumps is very useful in controlling power. The thermal power level is easily varied by simply increasing or decreasing the forced recirculation flow through the recirculation pumps.

As illustrated in Figure 2.1-6, two phase fluid (water and steam) above the core enters the riser area, which is the upper region contained inside of the shroud. The height of this region may be increased to increase the thermal natural recirculation pumping head. At the top of the riser area is the water separator. By swirling the two phase flow in cyclone separators, the steam is separated and rises upwards towards the steam dryer while the water remains behind and flows horizontally out into the downcomer region. In the downcomer region, it combines with the feedwater flow and the cycle repeats. The saturated steam that rises above the separator is dried by a chevron dryer structure. The steam then exits the RPV through four main steam lines and goes to the turbine, as shown in Figure 2.1-5.
Figure 2.1-6 ABWR Coolant Recirculation Flow Path (Ref 2-3)

(In righthand figure, Shaded arrows: water, Unshaded arrows: steam)
2.2. ABWR Development

2.2.1. Genesis of the ABWR

The ABWR design was based on construction and operating experience in Japan, U.S. and Europe and was jointly developed by the BWR suppliers as a next generation BWR (See Figure 2.1-3). Originally, an Advanced Engineering Team (AET) was formed in the late 1970’s to explore key features and build on best experiences and lessons learnt.

AET consisted of a 25-member-team with representatives from a world-wide range of technology suppliers such as General Electric (U.S.), Hitachi and Toshiba (Japan), Asea-Atom (Sweden) and Ansaldo (Italy). Architect engineering support was provided by NUCON of the Netherlands and Bechtel and Ebasco from the U.S. The AET effort was completed in 1979.

Sponsorship from the Tokyo Electric Power Company (TEPCO) and five other Japanese utilities continued the design effort in the 1980s. The utilities participating in the phased efforts of the 1980s under TEPCO’s leadership, and who provided support for the test and development program, were Chubu Electric Power Company, Chugoku Electric Power Company, Hokuriku Electric Power Company, Tohoku Electric Power Company, and the Japan Atomic Power Company.

Phase-1 was a conceptual design study that determined the feasibility of the ABWR concept. Phase-2 was development work that included more detailed engineering and testing of new technologies and design features. Phase-3 was aimed at optimising the design and systematically reducing capital costs.

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. Through two phases of Project Application Engineering (PAE) and Pre-Project Engineering (PPE), top level common engineering design documents were developed and reviewed by all three suppliers. Common engineering design documents provided consistent system design specifications and configuration details applicable to ABWR projects. Suppliers developed component designs, such as Main Steam Isolation Valve (MSIV), Safety Relief Valve (SRV), Fine Motion Control Rod Drive (FMCRD), RIP and so on, based on the common engineering design specifications and requirements. The
construction project of KK-6 and KK-7 completed and started their commercial operation on 7 November 1996 and 2 July 1997, respectively as planned.

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 on 12th May, 1997 to the General Electric Company. GE-Hitachi Nuclear Energy, LLC, and Toshiba, Co. submitted ABWR design certification renewal applications to the US NRC in 2010.

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. The current two projects (both of which are Hitachi-GE’s projects), represent the fifth and sixth ABWRs in Japan. Through all these ABWR projects, Hitachi-GE has been able to continuously improve the ABWR design by incorporating various customer requirements, site conditions and improvements based on earlier plant experience as well as technological advancements.

On the other hand, the Economic Simplified Boiling Water Reactor (ESBWR) is derived from its predecessors, the Simplified Boiling Water Reactor (SBWR) and the ABWR. Both the ESBWR and SBWR are designs by GE Hitachi Nuclear Energy (GEH). The standard design of the ESBWR is under review by the US NRC toward design certification in the USA.

2.2.2. BWR Safety System Evolution Leading Up to the ABWR

The Emergency Core Cooling Systems (ECCSs) provided by a particular product line were dependent on the regulations during that period of time.

The first generation of BWRs was designed without ECCS, but did make use of highly reliable feedwater systems. With the BWR/2, 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 ECCSs. Separate ECCSs was included in the BWR/2, and some forms of ECCSs were also retrofitted for BWR/1s.
For the BWR/3 and BWR/4 product lines, two electrical 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) (Figure 2.2-1). Low Pressure Core Spray (LPCS) and steam turbine driven High Pressure Core Injection (HPCI) systems are 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 reflood 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 1200°C.

Improvement of ECCSs continued in the BWR/5 and BWR/6 product lines. Three electrical divisions are applied to support ECCSs (Figure 2-2-1). 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 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.

The internal recirculation pumps in the ABWR eliminated the 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 ECCSs and to keep the core covered with water during a design basis LOCA.

A three divisional ECCS was adopted for the ABWR as shown in Figure 2.2-1. Each division provides both low pressure and high pressure makeup capability and incorporates its own emergency diesel generator power supply. One division includes a Reactor Core Isolation Cooling (RCIC) as high pressure makeup system, and a Low Pressure Core Flooder/Residual Heat Removal (LPCF/RHR) as low pressure system, while the other two divisions include HPCF and LPCF/RHR trains (Figure 2.2-2). 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, 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 coping capability for 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 provide effective makeup from the low pressure pumps over the full range of potential LOCA break sizes.
From a systems point of view, the ABWR ECCS design is almost N+2. The exception to N+2 occurs if Trains B & C (Divisions 2 & 3) are not operational (i.e., unavailable because of failure or maintenance), leaving only Train A (Division 1). For a large or medium break, the steam turbine driven RCIC cannot be credited for break makeup, leaving LPCF/RHR(A). If the break occurs in LPCF/RHR(A) (i.e., the feedwater line that RHR(A) interfaces with for vessel injection), another system is needed from Trains B or C (Divisions 2 or 3). Hence, for this break, the system is N+1.

2.2.3. BWR Reactor Pressure Vessel Evolution Leading Up to the 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, an indirect cycle using a 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 (Figure 2.1-4). 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 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 economical 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 relative to 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 increased costs 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, than the PWR core; and (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
reduce core reactivity, facilitating lower enrichment requirements and improved fuel utilization than
would be the case if power density were higher. Void reactivity feedback in the BWR also provides
damping of Xenon transients. However, the lower power density of the BWR does result in larger
RPV diameter.

Steam separation and drying within the BWR RPV results in overall compactness of the NSSS
and reduced radiation dose rates in the containment relative to employment of external steam
generators and steam drum components as in early BWRs, although the RPV needs sufficient height
to accommodate the internal steam separators and dryer components.

Manufacturing limitations on the height and diameter for the RPV has 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. Natural circulation and forced circulation BWRs 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. The superheat program was discontinued, however, due to challenges to fuel integrity and marginal economic advantages to be gained through 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 along with increasing the core size, power density and RPV diameter as a path toward higher 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 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. The 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, along 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
- 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 head closure with a radius less than the vessel radius, which helps minimise the weight of the head closure. The vessel closure seal consists of two concentric O-rings.

The steam outlet nozzles incorporate a flow restricting venturi. This provides for streamline flow and streamline 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.

Figure 2.2-3 provides views of the RPV during fabrication and placement at a construction site.
Figure 2.2-3 Views of an ABWR Reactor Pressure Vessel during fabrication and placement at a construction site (Ref 2-5)
2.2.4. BWR Reactor Internal Assembly Evolution Leading Up to the ABWR

A summary of the ABWR reactor internal assembly stackup and reactor recirculation flow path is provided in Figure 2-1-6. Figure 2.2-4 shows additional details of the ABWR reactor internal assembly. The CRDs and RIPs are mounted on the bottom of the vessel. Control Rod Guide Tubes (CRGTs) are located in the lower plenum of the vessel below a core plate. The nuclear fuel assemblies are located above the core plate, extending to the top guide. A plenum is provided above the top guide to promote mixing before a steam and water mixture enters the steam separators. Within the steam separators, water is separated and returned to a bulk water region where it is mixed with feedwater, to be returned to the lower plenum via the annular space between the vessel wall and the core shroud (Figure 2.2-5). Steam exiting from the steam separators is further dried in a steam dryer assembly before entering the steamlines (Figure 2.2-6).

Figure 2.2-4 ABWR Reactor Assembly (Ref 2-3)
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 free up space to remove and replace fuel assemblies. It has further been described in Section 2.2.3 above that 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 selected 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 (Figure 2.2-7). 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 hydraulic insertion of the CRs.

![Figure 2.2-7 ABWR Fine Motion Control Rod Drive (Ref 2-3)](image)

**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 backup
- Provides fine rod motion during normal operation
  - Small power changes
- Improved startup time & power maneuvering
  - Gang rod movement of up to 26 rods possible
- Reactivity accidents eliminated
  - Rod drop & Rod ejection

As noted in Section 2.1.1, the ABWR replaced the external recirculation pumps and the jet pumps with RIPs (Figure 2.2-8). 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 recirculated through the core. The RIPs are variable speed to adjust core flow. A change in RIP speed conditions will vary the core flow in the reactor, which, in turn, will change the reactor power during normal power range operation.

Key design features
- Ten pumps can provide up to 111% flow
  - 100% 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

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

Figure 2.2-8 ABWR Reactor Internal Pump (Ref 2-5)

In addition to the improvements in the RPV, FMCRD and RIP, several other improvements have been introduced for the Reactor Internal Components of the ABWR, as summarized in Figure 2.2-9.

As an example, 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 and simplifies in-service inspection.

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 of this report discusses the evolution of BWR material choices and reactor chemistry practices that have reduced and mitigated these issues in BWRs.
Figure 2.2-9 ABWR Reactor Internals Improvements (Ref 2-5)

<table>
<thead>
<tr>
<th>No</th>
<th>Changes</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Steam separator changed from two stages to low pressure-loss three stage type (AS-2B)</td>
</tr>
<tr>
<td></td>
<td>Adoption of external pumps eliminates large diameter pipes breaking at bottom of reactor. This enabled core to be nonuniformly moved in water, allowing cooling system to be changed from spray to simple sprayer type</td>
</tr>
<tr>
<td>2</td>
<td>Cold Plate</td>
</tr>
<tr>
<td></td>
<td>Top guide structure was changed from an assembly of individual matching plates to one-piece high-strength and integral one-piece machined base</td>
</tr>
<tr>
<td>3</td>
<td>Upper Shroud</td>
</tr>
<tr>
<td></td>
<td>Inner diameter of core shroud has been increased to accommodate a larger number of fuel assemblies</td>
</tr>
<tr>
<td></td>
<td>Top guide and upper shroud are now integral</td>
</tr>
<tr>
<td>4</td>
<td>Core Plate</td>
</tr>
<tr>
<td></td>
<td>A structure with reinforcing beams and perpendicular reinforcing rods was changed to all reinforcing beams to increase strength</td>
</tr>
<tr>
<td>5</td>
<td>Low Pressure coolant injection pipe</td>
</tr>
<tr>
<td></td>
<td>Adaptation of external pumps eliminates large diameter pipes breaking at bottom of reactor, enabling a longer core coverage under water. This eliminates the necessity of supplying water to the inside of the shroud, allowing cooling system to be changed to sprayer type of outside of shroud</td>
</tr>
</tbody>
</table>

2. Development of BWRs
Ver. 0

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2.2.5. BWR Materials and Reactor Chemistry Evolution Leading Up to the ABWR

This section provides an introduction to materials and reactor chemistry background for the BWR with respect to materials selection, component design and fabrication, coolant quality and control, radiation exposure, and fuel and water chemistry interaction. These aspects have broad impact and interfaces throughout the entire plant; however, this section focuses on impacts and evolution related to piping, reactor internals, and fuel.

Some of the ABWR plant interfaces with respect to reactor chemistry are shown in Figure 2.2-10. The purpose of this figure is to emphasize that Reactor Chemistry broadly interfaces with the overall ABWR plant components. Some key examples of interfaces are shown on this figure. Multiple additional interfaces such as the Suppression Cleanup System, RIP & FMCRD purge flow, NI & TI/BOP piping and heat exchangers also exist.

For the early BWRs, reactor chemistry focused on simpler aspects, for example, keeping the reactor cooling water as pure as possible. At that time, BWR water chemistry specifications were primarily to control conductivity and chloride levels to prevent the occurrence of transgranular stress corrosion cracking (TSCC). This seemingly fundamental approach did not recognize some of the issues dealt with today, such as IGSCC. This in part was 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 furnace 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 either.

Since the recognition of IGSCC in BWR structural materials, a number of remedies have been qualified to address materials, tensile stress and environmental aspects of this phenomenon. While this is not the only issue that has captured attention in the discipline of reactor chemistry, it has been one of the major issues requiring considerable attention.

This section begins by describing background and mitigation strategies to address stress corrosion cracking. This is followed by a broader discussion on other issues such as reactor water chemistry.
Figure 2.2-10 Examples of ABWR Interfaces and Systems with the Discipline of Reactor Chemistry
2.2.6.1 Materials and Stress Corrosion Cracking

IGSCC and IASCC are not unique to the BWR plants, but have to be considered in PWR plants too (although the mechanisms and remedies, in BWRs and PWRs, have some differences).

Historically, instances of SCC that have occurred in BWR plants are summarized in Table 2.2-1.

Table 2.2-1 Evolution of SCC in the BWR (Ref 2-7)

<table>
<thead>
<tr>
<th>Event</th>
<th>Timeframe</th>
</tr>
</thead>
<tbody>
<tr>
<td>Stainless Steel Fuel Cladding IGSCC</td>
<td>Late ‘50s/Early 60’s</td>
</tr>
<tr>
<td>IGSCC of Furnace Sensitized Type 304 During Operation</td>
<td>Late ‘60s</td>
</tr>
<tr>
<td>IGSCC of Welded Small Diameter Stainless Steel Piping</td>
<td>Mid ‘70s</td>
</tr>
<tr>
<td>IGSCC of Large Diameter 304 Piping</td>
<td>Late ‘70s</td>
</tr>
<tr>
<td>IGSCC of Alloy X750 Jet Pump Beam</td>
<td>Late ‘70s</td>
</tr>
<tr>
<td>IGSCC of Alloy 182/600 in Nozzles</td>
<td>Late ‘70s</td>
</tr>
<tr>
<td>Occurrence of IGSCC of BWR Internals</td>
<td>Late ‘70s</td>
</tr>
<tr>
<td>Crevice-induced Cracking of Type 304L/316L</td>
<td>80’s</td>
</tr>
<tr>
<td>Localized Cold Work Initiates IGSCC in Resistant Material</td>
<td>80’s</td>
</tr>
</tbody>
</table>

Some previous IGSCC concerns in BWRs were resolved by replacing the affected materials with more IGSCC-resistant materials or by performing repairs. However, because such repairs can be expensive, mitigation strategies were adopted to reduce probability of SCC and escalating repair costs. One primary mitigation strategy was to focus on water chemistry practices to protect components hence mitigate SCC. This grew out of an observation that oxidizing species in high purity coolant (i.e., Oxygen, Hydrogen Peroxide), as well as anionic species that contribute to coolant conductivity, were correlated with the incidence of IGSCC cracking as well as the rate of progression of any initiated cracks.

For the ABWR design, additional measures to improve SCC mitigation have been adopted. Examples of these measures include:

1. Improvement of design and fabrication techniques 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 in low and high fluence (IASCC) locations (e.g., use of low carbon Type 316NG austenitic stainless steel, Alloy 600M with niobium stabilization)
(3) Improvement of fabrication processes (e.g., controls on welding, cold work, solution annealing)

Since the recognition of IGSCC in BWR structural materials, a number of remedies have been qualified that address the three key contributing aspects of this phenomena: materials (selection and fabrication processes), tensile stress and environmental factors. Examples of this multi-faceted technology approach to SCC, which will be optionally applied to actual design, are provided in Figure 2.2-11. Figure 2.2-12 shows the time frame for evolution of mitigating strategies in the three contributing aspects.

In the area of material selection, reactor internal structures resistance to sensitization for the ABWR is achieved by using a special Type 316 or austenite stainless steel. This alloy has carbon restricted to a maximum of 0.020% by weight to prevent sensitization and has specific fabrication and processing controls to increase the resistance to any crack initiation.

To increase resistance against IGSCC for nickel based alloys in ABWR, Alloy 600 and their weld metals have been modified with stabilizing additions of niobium to reduce the potential for chromium depletion. For wrought structures, the ABWR employs a modified Alloy 600 (designated as 600M) which has a niobium content on the order of 1 to 3% and low carbon. For weld metal, a modified Alloy 82 with high stabilizing ratios is used, leading to high IGSCC resistance as well.
Figure 2.2-11 Example of Multi-Technology Approach to SCC Mitigation Considering Contributing Factors [Ref?]

- Peening method to mitigate in surface residual stress
- Polishing to mitigate in surface residual stress
- Welding method to minimize heat imput
- Polishing to remove surface cold worked layer
- Hydrogen injection

**Material factors**
- Surface cold worked layer
- Plastic strain

**Stress factors**
- Residual welding stress
- Residual stress due to surface cold working

**Environmental factors**
- Reactor water

* NMCA*: noble metal chemical addition
### Figure 2.2-12 Time Frame of SCC Mitigation Considering Contributing Factors

<table>
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<th></th>
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<tr>
<td>BWR type</td>
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<td>BWR-5</td>
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<td>Material and manufacturing process improvements</td>
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<td>304</td>
<td>304L</td>
<td>316L/316 (NG) Low-carbon stainless steel</td>
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<td>182M Modified Ni-based alloy with niobium</td>
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<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>82/600M Modified Ni-based alloy with niobium</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Welding heat input control</td>
<td>Optimum manufacturing process management</td>
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<td></td>
<td></td>
<td>Cold working control</td>
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<td>Cold-worked layer removal Laser welding</td>
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<td>NMCA</td>
<td>Online-NMCA</td>
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<td>WJP/Polishing</td>
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<tr>
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<td>Improved testing method</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Maintenance standards, inspection and evaluation guideline</td>
<td></td>
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</tr>
</tbody>
</table>

**Note:** The designation 182M/82/600M indicates Modification (M) with addition of niobium to reduce potential for chromium depletion at grain boundaries.

**Table:**
- **ABWR:** advanced BWR
- **NWC:** normal water chemistry
- **HWC:** hydrogen water chemistry
- **IHSI:** induction heating stress improvement
- **WJP/Polishing:** welding and polishing

2. Development of BWRs
   Ver. 0

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Figure 2.2-12 Time Frame of SCC Mitigation Considering Contributing Factors
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2.2.6.2 Reactor Water Chemistry

As noted above, during the early years of the nuclear industry, the common practice was to use Normal Water Chemistry (NWC) by controlling impurities, e.g., Cl, SO$_4^{2-}$, i.e., plant operation in highly purified water chemistry. However, as IGSCC issues arose (Table 2.2-1), laboratory tests indicated the beneficial effects of addition of hydrogen, Hydrogen Water Chemistry (HWC), to control IGSCC. Hydrogen additions to the reactor water that reduced Electrochemical Corrosion Potential (ECP) less than -0.23V (SHE) were found to be effective in suppressing stress corrosion cracking.

Unfortunately, a side effect of adding hydrogen to the reactor water is an increase in gamma radiation emission in the main steam line. The primary source of the radiation increase is from increased volatilization of nitrogen-16 under HWC conditions produced from neutron activation of oxygen-16:

\[
\text{O-16} \rightarrow (n, p) \rightarrow \text{N-16}
\]

\[
\text{(N-16} \rightarrow (\beta, \gamma) \rightarrow \text{O-16} \sim 7.1 \text{ sec half life; 6.1/7.1 MeV gammas})
\]

Although the half life is very short and does not cause substantial radiation exposure for maintenance work during outages, gamma shine in the turbine building is increased during normal operation.

Another important adverse effect of the HWC technology is the increase of piping dose rate which strongly affects the worker’s occupational dose during plant outage. This phenomenon is caused by the buildup of radioactivities such as Co-60 inside the oxide film formed on the piping surface in the strongly reducing environment due to the HWC application.

Both laboratory and in-plant crack growth rate data confirm the benefit of 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 as well.

The reduction in ECP and increase in main steam radiation is shown in Figure 2.2-13. (Note: the upper curve of In-Vessel ECP in Figure 2.2-13 shows the effect of HWC without considering other chemical injections, i.e., without Noble Metal injection, which is discussed below.)
The zinc effect to suppress the buildup of Co-60 inside oxide film was first recognized by the observation that plants with high zinc concentration in reactor water had less radiation dose rate of piping. Through the experimental and field experience studies, it was found that amounts of soluble zinc in reactor water reduces Co-60 buildup in the corrosion films on piping and components by promoting formation of a protective oxide film of spinel structure. Radiation dose rates are lowered since Zn is favored for incorporation in the oxide film relative to Co-60. While Natural Zinc Oxide (NZO) is effective in reducing radiation fields, activation of the Zn-64 (48% of NZO isotopic composition) to form Zn-65 contributes to radiation fields and low level wastes. Consequently, depleted Zinc Oxide (DZO), that is, zinc depleted in Zn-64, is commonly used for zinc injection application. When DZO is coupled with low iron input and cobalt alloy control, low shutdown dose rates may be obtained. Zinc injection is recommended to reduce dose rates when plants implement Hydrogen Water Chemistry (HWC) with or without noble metal chemical addition. (Ref 2-8)
A qualification program of zinc addition to reduce shutdown dose levels additionally indicated a positive effect of zinc ion on IGSCC susceptibility. Research showed low levels of zinc ion reduced crack propagation rates in Type 304 SS and Alloy 182 under reducing conditions, and high levels reduced crack growth rates even under oxidizing conditions. Beneficial effects of zinc ion on crack initiation and growth of Alloy 600 was also observed in PWR environments. Therefore, work was initiated directed at finding a specification to optimise the level of zinc ion concentration and hydrogen injection rate that would minimise operating dose rates while providing IGSCC crack mitigation for the bottom region of the reactor vessel (Refs 2-9 through 2-11). However, in the present days, zinc is used in BWRs, which apply HWC and noble metal technology, mainly for reduction of piping dose rate.

Noble Metal Chemical Addition (NMCA) is the technology introduced to control IGSCC with lower levels of hydrogen and thereby lower levels of gamma radiation. The effect of NMCA can be seen in the lower curve of In-Vessel ECP in Figure 2.2-13. ECP less than -0.23V (SHE) at hydrogen levels that maintain gamma radiation levels in the turbine building close to those for NWC.

NMCA achieves lowering of the ECP by injecting Platinum (Pt) and Rhodium (Rh) along with low hydrogen addition. Typical NobleChem™ applications were performed just prior to an outage at a temperature of 116 to 143°C. All added chemical species are cleaned-up during application and during outage, before the plant resumes its start-up operation. More recently, NobleChem™ has been applied as On-Line NobleChem™ (OLNC) while the BWRs are in power operation at approximately 280°C. OLNC application eliminates the need to have the 60- hour critical path time required for conventional NobleChem™ applications, thus saving the utilities an enormous expense. In addition, OLNC deposits noble metal inside cracks more efficiently because of their more open nature during plant operation. The advantage of NobleChem™ or OLNC is that they require very little hydrogen addition in to the feedwater (0.15 to 0.35 ppm) to achieve low ECPs, (Ref 2-13). As is mentioned above, zinc injection technique is normally applied in parallel with NobleChem™ or OLNC to suppress the radiation dose rate increase of piping.

It is clear that hydrogen, noble metal chemical additives, protect the plant structural materials. Investigations have also been conducted to assure there are no effects on fuel cladding integrity. For example, zinc deposits can form a tenacious, insoluble spinel form of zinc ferrite on the fuel cladding, and hence there has been concern with respect to spalling and corrosion from these crud deposits. Therefore, monitoring the effects of chemical additions has been a focus of recent fuel surveillance programs. However, no gross adverse effects of hydrogen, zinc and noble metal
chemical additives on corrosion and crud deposition have been observed. No gross hydriding has been identified either, although detection methods are not capable of detecting minor hydriding. (Refs 2-14 through 2-16)

In conclusion, reactor chemistry has played an important role in the existing BWR fleet to minimise the incidence and growth of intergranular stress corrosion cracking (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.
2.2.6. BWR Containment Evolution Leading Up to the ABWR

The initial BWR containment vessels were constructed from steel in a spherical shape, and were dry. However, this was soon superseded by the Mark I, Mark II and Mark III pressure suppression containments, each incorporating a suppression chamber containing a large water pool (Figure 2.2-14).

The advantages of a pressure suppression containment include:

- a heat sink for LOCA blowdown steam, safety/relief valve discharge steam, and reactor steam discharged from turbine driven pumped water make up systems;
- a secure source of water for ECCS and reactor isolation make up and cooling pumps;
- 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 Anticipated Operational Occurrences (AOOs) 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 2.2-14 Evolution of BWR Containment (Ref 2-3)](image-url)
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 containment vessel adopted the simpler geometry of a right circular cylinder, which was easier to construct and 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 tradeoffs related to construction and costs to achieve desired performance.

The ABWR adopted the simpler geometry of the right circular cylinder constructed using reinforced concrete, but otherwise retained characteristics closer to those of the Mark II containment design.

The ABWR Reinforced Concrete Containment Vessel (RCCV) is illustrated in Figure 2.2-15. 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. Compact structure and reactor building integration improves seismic stability and resistance and provides construction and cost effectiveness.
Figure 2.2-15 ABWR Reinforced Concrete Containment Vessel (Ref 2-5)
2.2.7. Electrical and I&C Evolution Leading Up to the ABWR

The BWR electrical distribution system and Instrumentation and Control (I&C) system, especially the safety system including the reactor protection system, have evolved over time as regulatory requirements and front line safety systems have evolved. In this document, the electrical and I&C evolution history in US BWRs is shown below as an example; details experienced by other non-US BWRs may differ in design depending on regulatory requirement of each country or region.

The early BWR product lines applied a two division safety system (Figure 2.2-1), and the electrical distribution system and the reactor protection system were two divisional as well (Figures 2.2-16 & 2.2-17).

![Figure 2.2-16 Early Vintage BWR Electrical Distribution System (Ref 2-17)](image-url)
Figure 2.2-17 Early Vintage BWR Reactor Protection System (Scram Function) (Ref 2-18)

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 independent on-site standby power source has been incorporated, as examples, a Combustion Turbine Generator or Air-Cooled Diesel Generator. In general, non-safety loads not connected to safety loads. (Note: This requirement does not apply to preferred non-safety power supply circuits that are connected to the input terminals of the safety distribution systems.)

These trends are consistent with Electrical Distribution System Requirements documented in the Electric Power Research Institute (EPRI) Utility Requirements Document (URD). The URD notes that a single medium voltage level preferred, but a dual voltage system acceptable when determined necessary. For ABWR, both single voltage and dual voltage systems have been designed. Figure 2.2-18 shows a single voltage system one line diagram.
Figure 2.2-18 ABWR Single Voltage System One Line Diagram (Ref 2-4)
The I&C system also evolved with evolution of the safety system including the reactor protection system. Additionally, I&C system has evolved as technology has evolved, most notably, the transition from analog electrical circuits to digital electrical circuits. Firstly, the main control systems such as the recirculation flow control system etc. and the BOP local control systems had adopted the digital electrical circuits step by step. Finally, for ABWR, a four division digital I&C with two-out-of-four trip setpoint voting and divisional logic unit voting has been adopted in the reactor protection system and the safety system (Figure 2.2-19). After that, several existing BWRs have retrofitted digital and microprocessor based I&C equipment, including for safety system applications, to replace obsolete components, improve system performance and provide enhanced self diagnostics and maintenance capabilities.

Additionally, ABWR has implemented Human Factors Evaluation as a discipline in the design of the Human-Machine Interface and Advanced Main Control Room. Figure 2.2-20 shows the overview of the Main Control Room.
Figure 2.2-19 ABWR Reactor Protection System (Scram and Main Steam Isolation Valve Functions) (Ref 2-19)
Figure 2.2-20 Overview of Main Control Room (Ref 2-5)
2.2.8. Design for the 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:

- The preventing and minimising (in terms of radioactivity) the creation of radioactive waste whilst 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 size of material that would ultimately become a radioactive waste.
- Allowing for flexibility in the design to have site specific waste management systems for treatment and managing of solid wastes and 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 System was designed to recombine oxygen and hydrogen in the gaseous waste from the main condenser, then to store in the pressurised tank for approximately one day for decay before discharging via the stack. In order to reduce the environmental release of emission activity of the discharged gas, Hitachi-GE had started R&D about the noble gas hold up technology that only the KRB power plant in (West) Germany had adapted at that time. This R&D had been carried out from 1968 to 1971, and this technology was adopted as the activated carbon noble gas hold up system for the Tsuruga Unit 1, known as the Off-Gas System (OG). The Noble Gas Hold-up System with activated carbon is designed to have the hold-up performance for approximately 30 days against a Xenon. The radioactivity release is reduced to one-tenth of the past design.

(2) Liquid Waste Treatment System

The liquid waste was 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 discharge. In terms of the radioactivity release, the release of the floor drain was dominant, and in order to reduce the activity release, the discharge of floor drain was changed to re-use within the plant with the treatment of evaporation and
demineralization as well as chemical drain. Due to this change, the development of a new evaporation system was required for durability and maintainability because the load to the evaporation system drastically increased.

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

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

In order to reduce the corrosion potential, a mitigation of operation conditions such as pH adjustment were studied and corrosion inhibitor and/or corrosion resistant material was adopted.

For the countermeasures of blocking scale, 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 the recent plant, the amount of the floor drain is quite small comparing the amount of the equipment drain. From this point, we have a plan to treat the floor drain together with the equipment drain (filtration and demineralization). This new process achieves the reduction of the concentrated liquid waste generation and consequently radioactivity release. And also contribute to reduce the 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 ozone layer protection. Then, a water washing laundry became a mainstream, and a reduction of release radioactivity has been attempted by adopting a evaporation + demineralization treatment and/or an activated charcoal absorption + microfiltration treatment system.

In addition to the improvements over recent years, further work is underway within Hitachi-GE to align the UK ABWR to the UK regulatory requirements. For example, a UK ABWR specific Radioactive Waste Management Arrangements document will demonstrate how solid radioactive wastes can be managed within the UK and evidence will be provided to show that the UK ABWR has been optimised to minimise its impact upon the environment.
2. Development of BWRs

2.3. Construction Experience

Hitachi-GE has established proven advanced construction methodology for building nuclear power plants that enhances construction safety and quality and reduces construction schedule and costs.

Hitachi-GE has continuously improved its construction technologies since the first BWR Nuclear Power Plant construction in 1970’s to improve the construction period, safety and quality. This methodology is based on the knowledge acquired with more than 30 years of construction experience. It consists of four major technologies: (1) Modularization with a Very Heavy Lift Crane, (2) Open-top and Parallel Construction Floor Packaging, (3) Front-Loaded Construction Engineering Detailed Schedule Management, and (4) Integrated Construction Management System.

(1) Modularization

Hitachi-GE initiated the use of very heavy lift crane in 1985 and a module dedicated factory in 2000 to increase modularization benefits. This methodology has been used to construct an ABWR and has led to the successful construction of four ABWRs.

(2) Open-top and Parallel Construction Floor Packaging

In this method, major components to be installed in the area are carried in prior to the ceiling work. Then, after the curing of concrete in the ceilings, mechanical/electrical installation work can proceed in parallel with the upper level of building construction. Therefore, it enables not only a reduction of the manpower peak but shortening of the construction schedule. In addition, a new concept, named “Floor Packaging method”, allows progressive hydro-static pressure testing of piping floor-by-floor, which reduces the maximum workload, further.

(3) Front-Loaded Construction Engineering Detailed Schedule Management

To introduce and implement the above strategies as planned, construction oriented engineering is critical. For this purpose, Hitachi-GE conducts front-loaded construction engineering with fully integrated 3D-Computer Aided Engineering (CAE) system and a detailed schedule management system, which improves the quality of plant engineering to achieve on-schedule construction.

(4) Integrated Construction Management System

The design stage of a nuclear power plant requires the overall coordination of a broad range of engineering tasks, including conceptual design, layout design, an equipment carry in/out plan,
shielding plan, as well as the plant construction, operation and maintenance plan. Schedule management, workforce management and QA/QC management are also important during each task phase. In order to perform these tasks efficiently, Hitachi has developed an "Advanced Integrated CAE System" to actualize high-quality and efficient works. This system works based on not only the plant engineering database but also the accumulated experiences and management know-how of the previous projects. Also, it is enhanced day by day through the actual projects as our core in-house engineering system.
3. Features of ABWR

3.1. Design Concept

The ABWR has been developed in collaboration with various international partners and support from power companies with experience in operating BWR plants. The primary design objectives were:

(1) Improved Safety with Diversity
(2) Improved Operation and Maintenance
(3) Advanced Technology Employment
(4) Construction Time Reduction
(5) Power Generation Costs Reduction
(6) Environmental Impact Minimisation

While these objectives have been achieved through the following product achievements:

(1) Ten Reactor Internal Pumps (RIPs)
(2) New Fine Motion Control Rod Drive design (FMCRD)
(3) Segregated Three Divisional ECCSs
(4) Digital Instrumentation and Controls (I&Cs)
(5) Low Operation and Maintenance (O&M) burden
(6) Reduced personnel radiation exposure
(7) Reinforced Concrete Containment Vessel (RCCV)
(8) Large scale, highly efficient power cycle
(9) Highly economical reactor core
(10) Noble Gas Hold-up System for Off-Gas System
(11) Improved Liquid Waste Treatment System

Quantitative criteria to implement these objectives were also established such as:

(1) Capacity factor of 87 - 90 %, depending upon the length of the operating cycle
(2) Less than one unplanned scram per year
(3) Personnel radiation exposure of less than 1 man-mSv per year
(4) Increase of power output to 1,350 MWe class
While these criteria were appropriate during the time of the development activities, it is noted that the quantitative criteria have been achieved and even exceeded for the intended projects. The features described above constitute highly functional, enhanced safety nuclear reactor systems, with a compact, easy-to-operate, and efficient turbine that offers excellent performance.

With regard to the UK ABWR design, “Hitachi-GE UK ABWR Concept design” (Ref 2-20) is provided for more information.

3.2. Main Features of ABWR

3.2.1. Improved Safety with Diversity

In the ABWR, pump impellers to recirculate the reactor coolant are internal to the reactor vessel with the driving motors mounted directly to the bottom of the ABWR reactor pressure vessel. This RIP system is simple because there is no external recirculation pumps and piping and no jet pumps internal to the vessel are required. The elimination of the external recirculation pumps and piping is particularly significant, as it provides a wide space inside the primary containment vessel and removes a significant radiation source from recirculation system piping that has contributed to radiation exposure of personnel doing maintenance work in the containment. As a result, work efficiency is enhanced and radiation exposure in maintenance work is reduced.

Furthermore, because there is no large-diameter nozzle below the top of the core region of the reactor pressure vessel when employing the RIP recirculation system, the water level can be maintained above the top of the active fuel with flooding systems even in the event of a design basis LOCA. All three segregated trains of the ECCS have a respective high pressure injection system. For ensuring the diversity on the power sources, one of the high-pressure ECCS (e.g. RCIC) is designed to apply the turbine-driven pump system with the main steam from the RPV and the DC battery. Other ECCS divisions employ HPCF designed to apply motor-driven pumps using AC power by divisional Emergency Diesel Generators (EDGs). This will allow HPCF to establish water injection at high or low RPV pressure. As a result, in the event of a loss of coolant accident, the core flooding will be maintained and safety will be preserved.
In case of the Common Cause Failure of the high-pressure ECCS, the ADS will reduce RPV pressure to permit water injection from Low Pressure Flooder (LPFL). The LPFL also consists of three divisions which are segregated, and are empowered by the EDGs in a respective division.

With this type of system, according to the results of a probabilistic safety assessment, core melt frequency will be less than that for the deployed conventional BWR, and thus safety is enhanced. Typical Probabilistic Risk Assessment (PRA) results indicate Core Damage Frequency (CDF) for ABWR less than $10^{-6}$ per Reactor-Year (Ref 2-3). An example of PRA result is shown in “Hitachi-GE UK ABWR Concept design” (Ref 2-20).

Additional diversity and safety enhancement has been achieved with the introduction of the FMCRD for nuclear reactivity control. The FMCRD uses two different power sources: an electric motor drive for normal operation and a conventional hydraulic drive for emergency insertion (scram). If a scram is required, it is actuated by the hydraulic control unit. If the hydraulic control unit fails, the control rod insertion will be actuated by the electric motor control unit. Thus enhanced reliable emergency insertion system is achieved.

Several other features of the FMCRD also enhance safety. The drive uses a bayonet-type coupling to a mating coupling on the control rod blade, which precludes separation of the blade and the FMCRD hollow piston. The piston is surrounded by an outer tube whose top end is a bayonet connection that couples to the control rod guide tube, which is supported in turn by the core plate and hence provides a positive means of preventing ejection of the FMCRD and control rod blade for any postulated CRD vessel housing (stub tube) break. The piston head contains latches that latch into notches in the drive guide tube to assure the control rod blade remains in place after scram, and an electro-mechanical brake is incorporated to prevent rod runout on pressure boundary failure. Redundant and diverse instrumentation is used to detect the control rod blade position, and a weighing table and magnetic reed switches detect if the hollow piston and control rod blade have separated from a ball nut used for fine positioning by the motor and ball screw.

3.2.2. Improved Operation and Maintenance

In the ABWR, a new digital control complex system was adopted (See Figure.2.2-20) to make the human-machine interface more useful and to expand the scope of automatic control. So, the operator’s workload during startup and shut-down of the plant is reduced to a large extent by automatic power control, which includes FMCRD and control of the RIP etc. Especially FMCRD is adopted in the automatic rod operation system so that as many as 26 rods can be withdrawn.
simultaneously, requiring no manual selection and operation. On the other side, in conventional plants, operators spend a long time operating control rods because the rods are manually selected one by one and operated step by step. Additionally, in the ABWR, several routine operations such as feedwater pump change, turbine trip, and other operations after the reactor scram are automated to reduce operators’ workload.

The enhanced safety features of the FMCRD also reduce maintenance obstructions and burden associated with the Locking Piston Control Rod Drive (LPCRD) used in previous BWRs. For example, internal shootout support structure is provided to prevent the inadvertent ejection of the drive and control rod in the event of the CRD housing failure. This replaces the external beam support structures used in previous BWRs.

The FMCRD also requires less periodic maintenance compared with the LPCRD used in previous BWRs. For example, the FMRD scram water is discharged directly into the reactor vessel. The scram discharge valve, hydraulic discharge lines and the scram discharge volume for previous BWRs are eliminated. Purge flow during normal operation precludes entrance of reactor water and crud into the drive. These features have simplified maintenance activities and effectively reduced personnel exposure.

Regarding the reduction of personal radiation exposure, the application of the RIP system contributes to achieve the reduction of personal radiation exposures due to the elimination of external recirculation pumps and piping. As noted in the Section above, the elimination of the external recirculation pumps and piping removes a significant radiation source from recirculation system piping that has contributed to radiation exposure of personnel doing maintenance work in the containment. As a result, work efficiency is enhanced and radiation exposure in maintenance work is reduced. The trend of radiation exposure in Japan indicates lower record for ABWRs than other light water reactors, which is described in “Hitachi-GE UK ABWR Concept design” (Ref 2-20).

Hitachi-GE also has developed other technical treatments to reduce the radioactive materials out of the nuclear power plant. Such technologies include application of low cobalt material to the primary coolant systems and the following technical aspects of the radwaste facilities contribute to the radwaste material reduction:

(1) Gaseous Waste Treatment System
3. Features of ABWR

- Charcoal adsorber to hold up noble gases (Xenon and Krypton) in order to decay the radioactivity.

(2) Liquid Waste Treatment System
- Hollow Fiber Filter to reduce secondary waste generation
- Integration of drain systems to achieve system size reduction for minimising of decommissioning waste.

(3) Waste Solidification System
- Waste generation reduction for solid waste has been able to adopt simplified solidification system.

3.2.3. Advanced Technology Employment

Hitachi-GE has been committed to continuous improvements in the ease of operation and reliability of its nuclear power plant monitoring and control systems. Hitachi-GE has pursued increased reliability in the form of standardization of digital control panels, the increased utilization of multiplexing technologies and fault tolerance improvement technologies, as well as the use of optical multiple transmission technology in the creation of hierarchical information networks. The Distributed Control Information System (DCIS), Nuclear Power Plant Control Complex with Advanced Man-Machine Interface 90 (NUCAMM-90), incorporates a background of digital technological development and expanded calculation capacity with a high level human engineered interface and increased scope of automation.

3.2.4. Construction Time Reduction

The use of reinforced concrete provides sufficient strength to withstand the high internal pressure postulated during accidental condition, and the use of a steel liner ensures the required air seal is maintained. Because the steel containment vessel in conventional BWRs must perform both functions, the steel must be very thick. In the ABWR RCCV, however, the steel mass is reduced since only a thin steel liner is required.

During the early phases of ABWR Development, several advantages as below were identified for the RCCV relative to Mark III Containment (Section 2.2.6).

(1) RCCV integrated with Reactor Building
3. Features of ABWR

- Better seismic stability and resistance
- Better to carry dynamic and shear loads
- Reduced overall size and thickness of walls

(2) Simple configuration / Compact
- Smaller Modules
- Lower Costs

In addition to the reduction in construction time achieved from using the RCCV, further improvement has been achieved from construction experience using proven practices such as Modularization, Open-Top and Parallel Construction Floor Packaging, Front-Loaded Construction Engineering Detailed Schedule Management, and Integrated Construction Management System, as described in Section 2.3

3.2.5. Power Generation Costs Reduction

In the ABWR, a high efficiency tandem compound turbine system is adopted, featuring the use of a 52-inch long blade for the last stage of the turbine, a two-stage Moisture Separator Re-heater (MSR), and a high pressure heater drain forward pump-up system connected to the condensate system. Thermal efficiency is enhanced through the use of this system.

The output was increased from approximately 1100 MWe for the BWR-5 plants in Japan to 1356 MWe for the ABWR. This was achieved by increasing the number of fuel assemblies in the core and increasing the diameter of the RPV. Other equipment required uprating as well, but the plant footprint was not dramatically impacted and overall improvements in economy of scale provided a cost reduction per MW of electric generation capacity in terms of Plant Capital Costs (PCCs).

The larger core size reduces radial neutron leakage, enhancing reactivity and reducing fuel enrichment requirements. Other core and fuel improvements, such as a wider bypass water gap between fuel assemblies and a modest volumetric core power density provided further efficiency of core neutronics and fuel utilization. Furthermore, the increased core flow range provided by the RIP system facilitates the use of spectral shift using core flow, that is operation with lower core flow and higher average steam void fraction in the core during most of the fuel cycle, resulting in transmutation from a "hardened" higher neutron energy spectrum of more U-238 to Pu-239 during the cycle, followed by higher core flow and lower average steam void fraction in the core at end of
the cycle, providing enhanced thermalisation of a "softer" lower emery spectrum to promote increased fissions of remaining fissionable species, including Pu-239, to extend the cycle and lower core enrichment requirements. The result of these features and operating strategies is lower Fuel Cycle Costs (FCCs).

Finally, improvements in O&M such as those described in Section 3.2.2 lead to lower maintenance costs and higher plant availability, thereby reducing O&M costs. Since Power Generation Costs (PGCs) is the summation of PCCs, FCCs and O&M costs on a MWe-hr, the reduction in these individual components results in an overall reduction in PGCs.
3.3. Design Comparison with BWRs

High level comparison table about standard BWR-5 vs. ABWR is shown in Table 5-1.

Table 5-1 Design Comparison with BWRs
4. ABWR Practice

4.1. International Practice

4.1.1. General

This section describes ABWR licensing and deployment activities outside Japan. Corresponding achievements in Japan are described in Section 4.2.

4.1.2. Design Certification in the U.S.

In 1987, the U.S. Nuclear Regulatory Commission (US NRC) began its technical review of the ABWR under the Standard Design Certification (DC) programme defined in 10CFR Part 52. On March 31, 1989, the US NRC initiated formal review of ABWR when GE completed submission for the Final Design Approval (FDA) and the standard DC for the U.S. ABWR design. Subsequently, GE submitted changes to the ABWR design information in Revision 4 to the design control document (DCD). The NRC evaluated those changes and on December 6, 1996, the Commission approved the final Design Certification Rule (DCR) for the U.S. ABWR standard plant design. On May 12, 1997, the NRC issued the U.S. ABWR final DCR in the Federal Register. Applicants or licensees intending to construct and operate a plant based on the U.S. ABWR design may do so by referencing its DCR, as set forth in 10 CFR Part 52, Appendix A.

A final DCR is valid for 15 years from the date of issuance under 10 CFR 52.55(a). A DCR may be renewed for a period of 10 to 15 years. By letter dated December 7, 2010, GEH submitted an application to renew the ABWR DC. The purpose of this renewal is to update the ABWR DC. The US NRC provided its acceptance to review the GEH DC Renewal Application for the U.S. ABWR on February 14, 2011.

4.1.3. Construction Project in Taiwan

Two ABWRs are being constructed by the Taiwan Power Company (TPC) at TPC's Lungmen site. Engineering and Procurement for the NI equipment of the two 1,350 MWe units were contracted to GE Hitachi Nuclear Energy Americas LLC, Hitachi and Toshiba were subcontracted to supply nuclear specialty equipment (e.g., RPV/large reactor internals, RIPS, FMC RDs) for the project.

The project was initiated in October 1996; a Preliminary Safety Analyses Report (PSAR) was submitted to Taiwan's Atomic Energy Agency (ROCAEC) in October, 1997. The Construction
Permit for both units was issued in March 1999. Unit 1’s RPV was installed on March 20, 2005, and Unit 2’s RPV was installed on October 5, 2006. Subsequently, Unit 1 was energized with 161 kV on July 15, 2007, and with 345 kV on October 28, 2010. Unit 1 is currently undergoing pre-operation testing.

4.2. Hitachi-GE Nuclear Energy’s ABWR Achievements in Japan

4.2.1. General

This section briefly describes Hitachi-GE’s technical achievements in the design and construction of past 4 ABWR projects in Japan.

4.2.2. Kashiwazaki-Kariwa Unit 6 and 7

Kashiwazaki-Kariwa nuclear power plant Units 6 and 7 of TEPCO were the first ABWR fleet in the world completed by the efforts of an international joint venture with Toshiba Corporation, GE and Hitachi. Hitachi provided the turbine island facilities for the Unit 6, and the reactor facilities for Unit 7.

In this project, Hitachi achieved their construction in a short term with the application of 3D-Computer Aided Design (3D-CAD) and a large mobile crane for construction, and expanded the scope of modularization to larger modules than previously used. In start-up tests, the performance of new equipment and facilities such as the RIP and FMCRD were confirmed and verified to achieve the technical requirements. The design conformance of the automation function for the CRD was also checked in the control performance test as well. In startup, transient tests and steady state operation runs, in which overall performance is verified, the design conformance was also demonstrated. This evidence indicates the technical excellence of the ABWR in terms of operability, safety and reliability.

4.2.3. Hamaoka Unit 5

Unit 5 of Hamaoka Nuclear Power Plant, owned by the Chubu Electric Power Co., Inc., was the first ABWR which was designed for a 60Hz electric frequency service area. The first Construction Permission was authorized in March 1999, and Unit 5’s commercial operation started in January 2005 with safety, reliability and economical improvement in its construction project realised.
Hitachi, Ltd. was in charge of supplying and installing the turbine and generator facilities as the Engineering Procurement and Construction (EPC) contractor in the TI with the latest design and construction technology. Above all, the high efficiency steam turbine and the large-capacity generator, which was designed based on technology previously applied in existing Unit 4, contributed to the achievement of Japan's largest electricity output of 1,380MW from a single reactor at the time.

In addition, various construction methods such as the large block modulisations and the field work supporting systems were applied with the use of 3D-CAD system, and contributed to improved field work efficiency as well as safety and quality.

4.2.4. Shika Unit 2

Unit 2 at Shika Nuclear Power Plant finished its start-up tests and started its commercial operation in March 2006. Shika Unit 2 is the world’s forth running ABWR nuclear power station, and is one of the largest power plants in Japan (with a capacity of 1,358MWe). Hitachi-GE independently designed, manufactured, constructed and commissioned the complete set of reactor and steam turbine generator facilities.

Shika Unit 2 obtained the construction permission in August 1999 and officially started its construction. Hitachi-GE completed the construction and commissioning program in 78 months from the start of excavation at the main power block buildings (September 1999) to commercial operation, and its construction in 57 months from rock inspection (June 2001) as planned.

4.2.5. Present ABWR Construction Projects in Japan

At present, HGNE is involved in the construction works of Shimane Unit 3 and Ohma nuclear power plants.

Shimane Unit 3 nuclear power plant (with a capacity of 1,373MWe), under construction at the Shimane Nuclear Power Station is a full-plant ABWR order received by Hitachi-GE following completion of Unit 2 at the Shika Nuclear Power Station of Hokuriku Electric Power Company. Hitachi-GE is responsible for the design and manufacture of the main reactor internal and other reactor components, turbine, generator, condenser, and related equipment; Babcock-Hitachi K.K. is responsible for manufacture of the RPV, and Hitachi Plant Technologies, Ltd. undertakes the construction (installation) works. In summary, this large-scale project comprehensively undertaken by Hitachi-GE and Hitachi’s group companies in all processes ranging from basic planning to
manufacturing of main components, installation of the reactor and TI main components and other equipment, and final delivery (turn key) to the customer.

The Ohma Nuclear Power Station, (with a capacity of 1,383MWe), is the world’s first ABWR constructed to use Mixed Oxide (MOX) fuel (an oxide fuel based on a mixture of Uranium and Plutonium) in the entire core. Hitachi-GE is also in charge of most of the main components scheduled to be installed in the future, and will continue to play a leading role in the construction project.
4.3. Continuous Development

4.3.1. Further Optimisation

In addition to an uninterrupted track record in construction since completion of the first ABWR, enhancements have been made to the following key technologies and products to improve plant economics, ease-of-operation, and performance.

(1) Increased capacity of main steam Safety Relief Valve (SRV)
Increased capacity of the valve by 16% with the valve throat diameter enlarged can reduce the number of valve units.

(2) Spectral Unit Module Initial (SUMIT) core
By decreasing the radial power peaking in the core more than the conventional BWRs, a SUMIT core achieves more than 40 GWd/t initial fuel discharge exposures and no refuelling in the first outage.

(3) Full MOX-ABWR core
Based on the belief that making effective use of plutonium is important in terms of resource efficiency, a full MOX-ABWR core (an improved type of BWR in which MOX fuel can be used throughout the entire core) has been developed and will be adopted for Ohma Unit 1.

4.3.2. Safety Enhancement

ABWR safety features are based on the Defence in Depth (DiD) concept wherein multiple layers of protection are provided with each layer designed to provide the safety function without reliance on the other layers. ABWR design is compliant with international criteria by well-designed Safety Systems to achieve a sufficiently low core damage frequency. Furthermore, to accomplish an enhanced level of nuclear safety, supplementary safety enhancements against severe conditions have been incorporated.

These enhancements, comprising a further layers in DiD, are designed to address the Fukushima-daiichi NPP accident caused by the huge earthquake and subsequent tsunamis on March 11, 2011. The major enhancements are further prevention of SBO and/or Loss of Ultimate Heat Sink (LUHS). Moreover, the enhanced functions ensure water supply into the reactor, PCV integrity, and Spent Fuel Pool (SFP) water level is maintained even in the event of SBO and/or LUHS. These enhancements, based on lessons learned from the Fukushima accident, along with provision and
maintenance of Severe Accident Management Guidelines (SAMGs), ensure that the integrity of inherent safety features of the ABWR is retained even in the event of a severe accident.

(1) Secure Power Source
   - Alternative DC Power Source.
   - Diversity of AC Power Sources (Water-cooled DG versus Air-cooled DG).
   - Sealed building structure to secure components and power panels in case of flooding.

(2) Secure water injection systems and ultimate heat sink
   - Diversity of alternate water injection capabilities.
   - Enhancement of mobility by applying portable pumps.
   - Diversity of heat sink through use of portable heat removal system.

(3) Prevention of PCV damage
   - Prevention of PCV damage caused by elevated temperatures by enhancing the PCV cooling system.

(4) Secure SFP Cooling function
   - Diversity of pool water injection method.
   - Accident Management operability enhancement by applying external water injection filler.
   - Incorporation of additional SFP temperature and water level monitoring systems in case of severe accident.
5. Conclusion

As shown in this document, the ABWR design can be recognized as substantially the best available technology from world’s BWR evolution experiences. ALARP consideration has been achieved in the ABWR design as indicated by: (1) lower Core Damage Frequency (CDF) compared with that of conventional BWRs by use of enhanced safety systems, (2) reduction of personnel radiation exposure by the eliminating piping surrounding RPV, (3) reduction of construction time by adopting RCCV and (4) reduction of power generation costs as shown in Section 3.2.5. In addition, the ABWR design has been developed to achieve further safety and reliability by incorporating technical lessons learnt from the Fukushima accident.

For the design of the UK ABWR, Hitachi-GE believes that the current ABWR design will be able to serve as a technical baseline, with modifications as required to satisfy UK safety requirements and criteria. In addition, Hitachi-GE’s ABWR construction experience will bring beneficial aspects in the actual UK construction project following substantial discussions throughout the UK’s Generic Design Assessment (GDA).
6. References


6. References


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