Overview of the Research Needs of the BWRs

October 25, 2016

Kumiaki Moriya
Hitachi-GE Nuclear Energy, Ltd.
Contents

1. System and Technical Features of BWR

2. Important Research Fields and Needs in BWR

3. Conclusion
BWR adopts a direct heat cycle.

- Steam generated in reactor pressure vessel (RPV) directly transfers to turbine through main steam system.
1.1 Main Components of Nuclear Reactor System

- Core shroud
- RPV
- Steam Dryer
- Steam separator
- CR Drive
- RIP
- Fuel assembly
- Control rod
Fuel Bundle and Control Rod

- Reactor Pressure Vessel (RPV)
- Core Shroud
- 10 Reactor Internal Pumps

- Control Rod
- Fuel Assembly

- Fuel Assembly: 872 units
- Control Rod: 205 units
- Local Power Range Monitor: 52 x 4 units
- Startup Range Neutron Monitor: 10 units
- Neutron Source for Startup: 5 units

Equivalent diameter about 5.2m

About 310

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1.2 Technical Feature in Core

- Bulk boiling in the core
  - Two-phase Flow Tech.
  - Coupling of Neutron Physics and Thermal-hydraulics

Reactor Pressure Vessel (RPV)

- Core
- Internal Pump
- Dryer
- Separator
Backbit Application of Advanced Fuel

Increase of bundle discharge exposure by optimizing void distribution inside fuel bundle
- Lattice Configuration
- Water Rods
- Arrangement
- Fuel Rod Length

- Improvement of fuel economy
- Spent fuel reduction

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Higher Flexibility for Cycle Length and Power Uprate

- BWR has high flexibility for long cycle operation and power up-rate by replacing to advanced fuel.
- ABWR can achieve 24 months cycle operation and uprating to about 1600 MWe by replacing to advanced fuel.

Comparison of Longer Cycle Operation

Comparison of Power Up rate

For 1356MWe
- 1380MWe
- 1450MWe
- 1630MWe
Flexibility for Pu utilization

- ABWR has full MOX capability due to enough shutdown margin by uniform, dense CR insertion.

**Full MOX Loading Pattern in ABWR**

- Red: Fresh MOX fuel
- Grey: Once burnt fuel
- Clear square: Twice burnt fuel
- Filled square: Thrice burnt fuel

**Representative MOX Loading Pattern in PWR**
(1/3 MOX Loading)

- Red: Uranium Fuel
- Green: MOX Fuel
- Circle: Control Rod

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## 1.3 Development and Evolution of BWRs

BWR was always evolving by incorporating advanced concept and technology.

<table>
<thead>
<tr>
<th>Year</th>
<th>Domestic Production</th>
<th>Improvement &amp; Standardization</th>
<th>2000</th>
<th>2010</th>
</tr>
</thead>
<tbody>
<tr>
<td>1970</td>
<td></td>
<td></td>
<td></td>
<td>ABWR</td>
</tr>
<tr>
<td>1980</td>
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<td></td>
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</tr>
<tr>
<td>1990</td>
<td></td>
<td></td>
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</tr>
</tbody>
</table>

**Facilities:**
- SHIMANE-1
- FUKUSHIMA1-4
- TOKAI-2
- SHIMANE-2
- SHIKA-1
- FUKUSHIMA2-2
- FUKUSHIMA2-4
- KASHIWAZAKI-KARIWA-5
- KASHIWAZAKI-KARIWA-4
- KASHIWAZAKI-KARIWA-7
- SHIKA-2
- SHIMANE-3
- OHMA-1
- HIGASHIDORI-1
Evolution of ABWR

ABWR is also always evolving by incorporating various customer requirements, site conditions and improvements from earlier plant experience and technological advancements.

- **KK-6&7**
  - First Standardized Design.
  - 1990

- **Hamaoka-5**
  - First 60Hz Turbine-Generator
  - 1995

- **Shika-2**
  - High Seismic Requirement
  - High Performance Initial Core Design
  - 2000

- **Shimane-3**
  - Rationalization of NSSS
  - 2005

- **Ohma-1**
  - Full MOX ABWR
  - 2010

- **UK-ABWR**
  - Fukushima Countermeasure
  - 2015

- **Full MOX Core**
  - High Performance Initial Core
  - Conventional design
  - SUMIT

- **Radial power peaking**

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Development Strategy of Future BWR

Not only “Economy of scale”, but also “Simplification of System” to meet a variety of customer needs.
Development of HP-ABWR (High-Performance ABWR)

- Stronger reactor building - withstands aircraft crashes
- Heat sink diversity and hybrid safety system - increase safety margin under severe accidents
- SC structure containment vessel - enhances safety margin and shortens construction period
- Advanced materials for major components - realize 80-year plant life
- Longer operation cycle, higher burn-up and innovative spectrum shift rods (SSR) - realize lower power generation costs
- Seismic isolation technology - ensures safety against earthquakes

1800MWe ~ 1000MWe

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Small & Medium BWR (DMS)

- **DMS reactor** -
  Double MS: Modular Simplified & Medium Small Reactor

<table>
<thead>
<tr>
<th>Specification</th>
<th></th>
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</thead>
<tbody>
<tr>
<td>Thermal Power</td>
<td>840 MW</td>
</tr>
<tr>
<td>Electric Power</td>
<td>300 MW</td>
</tr>
<tr>
<td>Reactor Type</td>
<td>BWR</td>
</tr>
<tr>
<td>Cooling Method</td>
<td>Natural Circulation</td>
</tr>
<tr>
<td>Coolant Pressure</td>
<td>7.2 MPa</td>
</tr>
<tr>
<td>Power Density</td>
<td>44 kW/l</td>
</tr>
<tr>
<td>No. Of fuel bundle</td>
<td>400</td>
</tr>
<tr>
<td>Fuel Enrichment</td>
<td>4.3 wt%</td>
</tr>
<tr>
<td>Active Fuel Length</td>
<td>2 m</td>
</tr>
<tr>
<td>Refuel Interval</td>
<td>&gt;2 years</td>
</tr>
<tr>
<td>Construction Period</td>
<td>~2 years</td>
</tr>
<tr>
<td>Safety System</td>
<td>Hybrid (Passive + Active)</td>
</tr>
</tbody>
</table>

- **Simplified reactor internals and system**
- **Enhanced hybrid (passive & active) safety system**
- **Multi-purpose use of energy (non-electric use of energy: Co-generation)**

**Free Surface Separation (FSS)**
Steam is separated from water by gravity force, hence, no “Separator” is required.

**Natural Circulation**
No reactor internal pumps (RIPs)

**Primary containment vessel**

**Reactor pressure vessel**

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Control Rod Free Core for Easy Operation in DMS

- Core power of DMS can be controlled just by the fuel design without using control rod operation by optimized Gd distribution.

**Optimized Gd distribution**

<table>
<thead>
<tr>
<th>NU</th>
<th>NU</th>
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<tbody>
<tr>
<td>4%</td>
<td>4%</td>
</tr>
<tr>
<td>8%</td>
<td>10%</td>
</tr>
<tr>
<td>5%</td>
<td>6%</td>
</tr>
</tbody>
</table>

Gd burns out at EOC

Gd burns out at MOC

NU: Natural uranium

- Power increases because of low concentration Gd at axially lower part
- Power increases because of power shape change

**Middle of cycle (MOC)**
- Uniform Gd
- Optimized Gd

**End of cycle (EOC)**
- Uniform Gd
- Optimized Gd

Core power of DMS can be controlled just by the fuel design without using control rod operation by optimized Gd distribution.
DMS can be used as co-generative plant, and heat may be used in various kinds of thermal utilization applications, such as desalination, district heating, etc.

Hitachi-GE, Hitachi, and the University of Saskatchewan in Canada have collaborated on a joint R&D initiative.

**Multi-purpose Thermal Utilization System in DMS**

- DMS can be used as co-generation plant, and heat may be used in various kinds of thermal utilization applications, such as desalination, district heating, etc.
- Hitachi-GE, Hitachi, and the University of Saskatchewan in Canada have collaborated on a joint R&D initiative.
RBWR is developed for Pu Breeding or for TRU burning by optimizing fuel configuration.

Safety system, BOP, etc. are almost same as conventional BWR.
## 2. Important Technical Areas and Research Needs in BWR

<table>
<thead>
<tr>
<th>Technical Area</th>
<th>Research Needs</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Core &amp; Fuel</strong></td>
<td>• Precise core simulator, transient and accident analytical codes, considering the coupling of neutron physics and thermal-hydraulics</td>
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<tr>
<td></td>
<td>• Precise modeling of two-phase flow</td>
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<td></td>
<td>• Advanced fuel cladding material and pellet</td>
</tr>
<tr>
<td><strong>Safety</strong></td>
<td>• Severe accident evaluation and accident management procedure</td>
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<td></td>
<td>• Probabilistic safety assessment</td>
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<td></td>
<td>• Human factor</td>
</tr>
<tr>
<td><strong>Equipment Design</strong></td>
<td>• Thermal-hydraulics tests and analysis</td>
</tr>
<tr>
<td></td>
<td>• Seismic Tests and analysis</td>
</tr>
<tr>
<td><strong>Material</strong></td>
<td>• SCC/IA SSC of austenitic alloys</td>
</tr>
<tr>
<td></td>
<td>• Thermal aging of duplex stainless steel</td>
</tr>
<tr>
<td></td>
<td>• Co-base hard facing material</td>
</tr>
<tr>
<td><strong>Water Chemistry</strong></td>
<td>• Structural material integrity (Improvement of corrosion environment)</td>
</tr>
<tr>
<td></td>
<td>• Dose rate reduction</td>
</tr>
<tr>
<td><strong>Inspection</strong></td>
<td>• High sensitive and environment-tolerant sensor</td>
</tr>
<tr>
<td></td>
<td>• Robot technology for remote inspection</td>
</tr>
<tr>
<td><strong>Others</strong></td>
<td>• I&amp;C</td>
</tr>
<tr>
<td></td>
<td>• Manufacturing/Construction Technology</td>
</tr>
</tbody>
</table>
BWR has sufficient safety capability with inherent safety features and large thermal margin.

Many research activities should be continued in severe accident for furthermore enhancement of safety, since SA includes various complicated and uncertain phenomena.

PSA Study Results

Hydrogen Burning Test

Since the PCV of BWRs has no oxygen, no combustion reaction occurs in the PCV.

Direct contact of the corium

Direct contact hardly occurs on ABWRs because the corium in the lower D/W hardly move into the upper D/W due to their configuration.

Heat Flux of Ex-vessel Debris
Research Needs for Equipment Design

Thermal-hydraulic Test and Analysis are indispensable for the development of advanced equipment.

Steam-water test facility (HUSTLE)
Void fraction prediction in chimney is important to reliable natural circulation flow in ESBWR and DMS.

Precise two-phase flow CFD analytical method is developed and verified by HUSTLE test.

Analyzed three-dimensional void fraction distribution by test analysis.

Ref.: ICONE20POWER2012-55117
Example: Test and Analysis of Flow Distribution in the Reactor Vessel

- Confirm the effect of pump on flow distribution by tests and CFD analysis

360 degree Scale Test Facility (1/5Scale)

Operating pump

Tripped pump

Vertical Flow velocity around pump and shroud support leg

Flow velocity along RPV bottom with tripped pump

Examples of flow distribution by STAR-CD calculation
The performance of control rod insertion has demonstrated by test and analysis in seismic condition.

Test Facility
- Seismic scramability test can be performed.
- Maximum deflection is 60mm.

Material science is very important for the reliability and worker dose rate reduction.

1. SCC/IASCC
   • Detailed mechanism of SCC/IASCC
   • Quantitative characteristics of SCC/IASCC
   • Confirmation of performance of existing SCC/IASCC countermeasures
   • Improvement of SCC/IASCC resistance

2. Irradiation effect on thermal aging
   • Mechanism of neutron irradiation effect on thermal aging of duplex stainless steel

3. Co dissolution free hard facing material
   • Alternate material for conventional Co-base alloy
Possible R&D themes

1. SCC/IASCC
   • Elucidation of SCC/IASCC mechanism
     Role of microstructure, stress/strain and environment for crack initiation and growth
   • Quantification of SCC/IASCC behavior
     Crack initiation condition
     Crack growth rate disposition curve
     Threshold stress intensity factor or crack size for steady crack growth (if exists)
     Threshold neutron fluence
     Fracture toughness of irradiated stainless steel
   • Development of new SCC/IASCC countermeasures
     IASCC resistant stainless steel, SCC resistant Ni-base alloy
     Refined fabrication process to improve microstructure or stress/strain

2. Irradiation effect on thermal aging
   • Elucidation of thermal aging mechanism of duplex stainless steel and neutron irradiation effect

3. Co dissolution free hard facing material
   • Development of alternate material (Co-free alloy, etc.)
## Research Needs for Water Chemistry

<table>
<thead>
<tr>
<th>Item</th>
<th>1970s</th>
<th>1980s</th>
<th>1990s</th>
<th>2000s</th>
<th>2010s</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Integrity</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>Structural material Integrity</td>
<td></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>Dose rate reduction</td>
<td></td>
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<td></td>
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<tr>
<td>Others</td>
<td></td>
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</tbody>
</table>

### 1970s
- First SCC**
- Dose increase
- Low Co material
- Shroud replacement

### 1980s
- CILC* (USA)
- Activity increase
- Fe reduction
- Aerosol recovery

### 1990s
- * Crud Induced Localized Corrosion
- Second SCC
- HWC
- Chemical Decon.

### 2000s
- NMCA (GE)
- Evaluation of H$_2$O$_2$ effect
- Recontamination reduction
- S/P pH control

### 2010s
- OLNC (GEH)
- Spread of environmental relaxation
Following items are examples which are required to fully understand corrosion products behavior under HWC and to make effective measures to reduce dose rate and radiation exposure of workers;

- Solubility of metal oxides in high temperature water
- Measurement of ferrite formation rate on the fuel surface with zinc and/or platinum
- Deposition and detachment behavior of fuel crud under power change in case of load following operation
- Corrosion rate of metals (stainless steel, Ni based alloy, low alloy steel, Stellite and carbon steel) at the operating temperature in various water chemistry conditions
- Metal impurity release rate from structural material (same as above) at the operating temperature in various water chemistry conditions
Research Needs for Inspection

Advanced Inspection technology is developing for reliability and worker dose rate reduction.

1960

Eddy Current Inspection System developed by Hitachi Laboratory in 1961.

1980

Automatic in-service inspection (ISI) systems were lined up for weld line inspection of RPV and pipes.

2000

New robot and sensors were applying for in-vessel inspection.

2016

Remotely operated vehicle for shroud inspection

Flexible eddy current sensor for bottom head inspection

Next Research Needs

1) High sensitive and environment-tolerant sensor
2) Robot technology for remote inspection
3. Conclusion

- Hitachi is making an effort diligently to build ABWRs in the Wylfa site. ABWR is the latest proven BWR developed as highly safe, economic and reliable BWR, but the deep understanding of BWR technology by own research activity in UK is indispensable to actually secure safety and stable operation.

- Hitachi is tackling to develop various future BWRs like RBWR, SMR and so on. The advanced nuclear technology and research capability in UK are valuable for the development of such future BWRs.
<table>
<thead>
<tr>
<th>Purpose</th>
<th>Item</th>
<th>Main Research Theme</th>
</tr>
</thead>
<tbody>
<tr>
<td>Establishment of ABWR Tech.</td>
<td>Cross Assessment</td>
<td>• Development of Core and Safety Analysis Code and Cross Check Evaluation</td>
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<tr>
<td></td>
<td></td>
<td>• PSA Study</td>
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<td></td>
<td></td>
<td>• Tech. Review of Design</td>
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<tr>
<td>Adaptation to UK</td>
<td></td>
<td>• UK Standards</td>
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<tr>
<td></td>
<td></td>
<td>• Design, Manufacture, Operation, Decommissioning</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Site Environment</td>
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<tr>
<td>Trouble Response</td>
<td></td>
<td>• Cause Study and Countermeasure</td>
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<tr>
<td></td>
<td></td>
<td>• Technical Support in Emergency</td>
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<tr>
<td>Improvement of UK-ABWR</td>
<td>Enhancement of Safety</td>
<td>• Severe Accident Study</td>
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<td></td>
<td>• Human Factor Study</td>
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<tr>
<td></td>
<td>Improvement of Economy</td>
<td>• Core and Fuel</td>
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<tr>
<td></td>
<td></td>
<td>• System and Equipment, Plant Performance</td>
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<tr>
<td>Development of Future BWR</td>
<td>Plant Concept</td>
<td>• Small BWR, RBWR etc.</td>
</tr>
<tr>
<td></td>
<td>Innovative Tech.</td>
<td>• Innovative System and equipment</td>
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