Status of Nuclear Power Plants in Japan and Future Research Strategy for Nuclear Safety and Materials

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Contents

1. Current Status of Nuclear Power Plants in Japan


6. The Fukushima Daiichi Accident and the Effect of Ageing on the Accident
1. Current Status of NPPs in Japan
On outage or stopped: 48 units
PWR In operation: 0 units
BWR In operation: 0 units
Under construction: 2 units
Planned: 10 units
decommissioning: 9 units

Under application for restarting the operation: 20 Units
The NRA has received 20 Applications for Restarting Reactors

<table>
<thead>
<tr>
<th>Applicants (Electric Utility Company)</th>
<th>Nuclear Power Plant and Unit #</th>
<th>Start of Operation (Age)</th>
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</tr>
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<tbody>
<tr>
<td></td>
<td>Tomari #3</td>
<td>2009 (4)</td>
<td>July 8, 2013</td>
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</tr>
<tr>
<td>Shikoku</td>
<td>Ikata #3</td>
<td>1994 (19)</td>
<td>July 8, 2013</td>
<td></td>
</tr>
<tr>
<td>Kyushu</td>
<td>Sendai #1,#2</td>
<td>1984/1985 (30/28)</td>
<td>July 8, 2013</td>
<td><strong>July 16, 2014</strong></td>
</tr>
<tr>
<td>Tohoku</td>
<td>Onagawa #2</td>
<td>1995 (19)</td>
<td>Dec. 27, 2013</td>
<td></td>
</tr>
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<td>Tohoku</td>
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</tr>
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<td>Hokuriku</td>
<td>Shika #2</td>
<td>2006 (8)</td>
<td>Aug. 12,2014</td>
<td></td>
</tr>
<tr>
<td>total</td>
<td>20 Units</td>
<td></td>
<td>-</td>
<td>2 Units</td>
</tr>
</tbody>
</table>

https://www.nsr.go.jp/activity/regulation/tekigousei/power_plants.html
Restarting of Sendai Unit 1 & Unit 2

PWRs, 890 MWe each, Start of Commercial Operation; July 4, 1984 (Unit 1)
Nov. 28, 1985 (Unit 2)

http://www.reuters.com/article/2014/07/16/us-japan-nuclear-restart-idUSKBN0FL02R20140716

“An employee of Kyushu Electric Power Co walks in front of reactor buildings at the company’s Sendai NPS on April 3, 2014”
Establishment of Nuclear Regulation Authority (NRA)

NRA was established on September 19, 2012, and merged with JNES on March 1, 2014.

**Prime Minister’s Cabinet Office**
- Nuclear Safety Commission (NSC)

**Ministry of Economy, Trade and Industry**
- Nuclear and Industrial Safety Agency (NISA)

**Ministry of Education, Science and Culture**
- Research reactors
- Radiation monitoring etc.

**Ministry of the Environment**
- Nuclear Regulation Authority (NRA)
  - Chairman + 4 Commissioners
  - Secretariat
  - *former JNES*

NIRS

Number of Staff members: about 1,000

JNES*

JAEA**

NIRS***

JAEA, NIRS

*JAPAN Nuclear Energy Safety Organization (merged to the NRA on March 1, 2014)

** JAPAN Atomic Energy Agency

*** National Institute of Radiological Sciences
New Safety Standards in Japan

New Safety Standards
Effective from July 8, 2013

< Previous Safety Standards>

Design Basis Standard
to prevent Severe Core Damage
(Only assuming single failure etc.)

- Consideration for Natural Events
- Consideration for Fire
- Consideration for Reliability
- Reliability of power source
- Performance of cooling equipment
- Performance of other equipment
- Tolerability for Earthquake & Tsunami

New
Countermeasures against Severe Accident

Enhancement

Preventing Large Scale Release
Intentional Airplane Crash
Preventing CV Failure
Preventing Severe Core Damage (Assuming multiple malfunction)
Consideration for Natural Events
Consideration for Fire
Consideration for Reliability
Reliability of Power Source
Performance of Cooling Equipment
Performance of Other Equipment
Tolerability for Earthquake & Tsunami
New Nuclear Regulation Systems in Japan
Became Effective from July 8, 2013

The NRA (Nuclear Regulatory Authority) implement new regulatory systems stipulated in the amended Nuclear Regulation Act;

- Regulation taking severe accidents into consideration
  - Legally request severe accident measures to the licensees

- Introduction of back-fitting system: Regulation applying latest scientific/technical knowledge on safety issues to existing facilities
  - Apply new technical standards to existing licensed nuclear facilities as a legal obligation

- An operation limit of 40 years to deal with aged reactors
  - As an exception, extension (<20 years) will be approved, only when compliance with the regulatory standards by the Government Order is confirmed.

- Special safety regulation in Fukushima Daiichi
Regulatory Systems for Inspection, Periodic Safety Review and Ageing Management Technical Evaluation

- **Start of Commercial Operation**
- **Regulatory Inspection** (every 13 ~ 24 Months)
- **Utility’s Periodic Safety Review (PSR)** (every 10 Years)
  - 10 Years
  - 10 Years
  - 10 Years
  - 10 Years
  - 10 Years
  - 10 Years
- **Ageing Management Technical Evaluation (AMTE)**
  - before 30 years for Continued Operation
  - Re-evaluation, every 10 Years
  - 10 Years
  - 10 Years
  - 10 Years
- **Approval of Operating Period Extension**
  - 30 Years
Nuclear Power Plants in Japan

Electrical Output (MWe)

Start of Commercial Operation

BWR : 24 units  BWR (Permanent Shut-down) : 8 units
PWR : 24 units  GCR, ATR (Permanent Shut-down)

1F : Fukushima Daiichi
Ageing Management Technical Evaluation before 30 years of Operation

1. Review Operational Experience and Possible Ageing Degradation in All the Safety-related SSCs
2. Reflect Latest Knowledge and Engineering Database
3. Evaluate Integrity of SSCs considering Ageing Degradation Assuming the Service for 60 years
4. Evaluate Seismic Safety Analysis considering Ageing Degradation Assuming the Service for 60 years
5. Evaluate Validity of Current Maintenance Program
6. Establish Long-Term Maintenance Program in the next 10 years, including R&D Plans
Ageing Management Technical Evaluation
Before 30 years of operation and following every 10 years

Regulatory Review
○ Results of the evaluation
○ Adequacy of the Long-Term Maintenance Program (LTMP)

All Components with Safety Function

Select Important Ageing Effects (low-cycle fatigue, neutron irradiation embrittlement, IASCC, etc.)

Integrity Evaluation assuming 60 years of Operation

Formulate the LTMP including necessary measures in addition to the current maintenance plan

Additional requirements before 40 years of operation

1. Verification of the results of the evaluation before 30 years of operation
2. Evaluate the effectiveness of the 30th-year LTMP
Maintenance Program and Ageing Management

**Maintenance Program**

- Policies and Goals of Maintenance Management
- Establishment of the Scope
- Establishment of the Importance
- Establishment of Indicators for Management of Maintenance Activities
- Implementation of Maintenance and Data Acquisition
- Monitoring Indicators for Management of Maintenance Activities
- Confirmation & Assessment
  - OK
  - NG
  - Correction of Incompatibility
- Validation Assessment of Maintenance Activities

**Ageing Management**

- Periodic Safety Review
- Technical Evaluation
- Establishment of Long-Term Maintenance Programs
- Implementation of Long-Term Maintenance Programs
The NRA (Nuclear Regulatory Authority) implement new regulatory systems stipulated in the amended Nuclear Regulation Act:

- **Regulation taking severe accidents into consideration**
  - Legally request severe accident measures to the licensees

- **Introduction of back-fitting system**: Regulation applying latest scientific/technical knowledge on safety issues to existing facilities
  - Apply new technical standards to existing licensed nuclear facilities as a legal obligation

- **An operation limit of 40 years** to deal with aged reactors
  - As an exception, extension (<20 years) will be approved, only when compliance with the regulatory standards by the Government Order is confirmed.

- **Special safety regulation in Fukushima Daiichi**
Approval of Operational Period Extension and Ageing Management Technical Evaluation

Extension of Operational Period beyond 40 years

<Requirements>
(1) Implementation of special safety inspection of the NPPs
(2) Technical evaluation on the degradation for the extension period
(3) Formulation of maintenance management policy for the extension period

Ageing Management Evaluation before 30 years and beyond

<Requirements>
(a) Technical evaluation on ageing degradation
(b) Formulation of the long-term maintenance management policy

<Requirements>
The appropriate maintenance is implemented according to the standard technical specifications.
**Special Inspections to Extend Operational Period**

Regulatory Requirements of Special Additional Inspections for Long Term Operation beyond 40 years up to 60 years

<table>
<thead>
<tr>
<th>Components</th>
<th>Current Inspection</th>
<th>Additional Inspection</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Pressure Vessel</td>
<td>Ultrasonic Tests of Welded Zone</td>
<td>100% UT Examination of Base Metal in addition to Welded Zone</td>
</tr>
<tr>
<td>Primary Containment Vessel (Steel)</td>
<td>Leak Rate Tests</td>
<td>Visual Test (Appearance of coating film)</td>
</tr>
<tr>
<td>Civil Concrete Structure</td>
<td>Visual Tests &amp; NDT</td>
<td>Core Sampling (Strength, Neutralization, Salt intrusion, etc.)</td>
</tr>
</tbody>
</table>

- Special Inspections are requirements to extend the operational period beyond 40 years by up to 20 years (60 years maximum) in addition to:
  - Ageing Management Technical Evaluation (AMTE) for every 10 years
  - Maintenance Program
## Ageing Management Technical Evaluation after the Fukushima Accident

<table>
<thead>
<tr>
<th>Unit</th>
<th>Ageing Management Technical Evaluation</th>
<th>Date (30 or 40 years)</th>
<th>Submitted Date of the Report</th>
<th>Date of Approval by the NRA</th>
</tr>
</thead>
</table>

( Establishment of NRA on Sep. 19, 2012 )

| Onagawa-1         | 30 years (cold shutdown)               | June 1, 2014           | Nov. 6, 2013                 | May 21, 2014                |
| Sendai-1          | 30 years                               | July 4, 2014           | Dec. 18, 2013                | ○                           |
| Takahama-1        | 40 years (cold shutdown)               | Nov. 14, 2014          | Nov. 12, 2013                | Nov. 12, 2014               |
| Takahama-4        | 30 years                               | June 5, 2015           | June 3, 2014                 | ○                           |
| Fukushima Daini-3 | 30 years (cold shutdown)               | June 21, 2015          | June 20, 2014                | ○                           |
| Kashiwazaki-1     | 30 years                               | Sep. 18, 2015          | Sep. 18, 2014                | ○                           |
| Genkai-1          | 40 years (cold shutdown)               | Oct. 15, 2015          | Oct. 15, 2014                | ○                           |
| Takahama-2        | 40 years (cold shutdown)               | Nov. 14, 2015          | Nov. 14, 2014                | ○                           |
| Sendai-2          | 30 years                               | Nov. 28, 2015          | Nov. 28, 2014                | ○                           |

○: Application to the NRA for restarting the reactors
The NRA has received 20 Applications for Restarting Reactors

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https://www.nsr.go.jp/activity/regulation/tekigousei/power_plants.html

English version is available
**Cross Table for Ageing Mechanisms and (Location and Material in) Components**

**Summary Sheets of Ageing Mechanisms (Attachment A of the AESJ Code)**

- Based on the experience in Ageing Management Technical Assessment in 14 Plants (7 PWRs and 7 BWRs).
- Cross tables of ageing mechanisms for 300 components in each PWR and BWR have been successfully summarized and published in the AESJ Code as a mandatory requirement.

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### Summary Sheet of Ageing Mechanisms for Turbo Pumps

<table>
<thead>
<tr>
<th>N o.</th>
<th>Issues required to achieve intended functions</th>
<th>Part</th>
<th>Material</th>
<th>Ageing phenomena</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Assurance of pump capacity (head)</td>
<td>Main shaft</td>
<td>SS</td>
<td>Wear</td>
<td></td>
</tr>
<tr>
<td>2</td>
<td></td>
<td>Main shaft</td>
<td>SS</td>
<td>Corrosion (pitting, etc.)</td>
<td></td>
</tr>
<tr>
<td>13</td>
<td>Maintenance of boundary</td>
<td>Discharge elbow</td>
<td>Cast iron</td>
<td>Corrosion (pitting, etc.)</td>
<td></td>
</tr>
<tr>
<td>14</td>
<td></td>
<td>Discharge pipe</td>
<td>Cast iron</td>
<td>Corrosion (pitting, etc.)</td>
<td></td>
</tr>
<tr>
<td>23</td>
<td>Support of component</td>
<td>Support plate</td>
<td>SS</td>
<td>(N/A)</td>
<td></td>
</tr>
<tr>
<td>24</td>
<td></td>
<td>Support plate</td>
<td>SS</td>
<td>(N/A)</td>
<td></td>
</tr>
</tbody>
</table>
Sharing and Updating Knowledge for Ageing Management with Summary Sheets of Ageing Mechanisms

Operators of NPPs (Utilities)

- Maintenance program
  - Details of Inspection and Maintenance Plan and Activities
  - Evaluation of Maintenance Effectiveness
  - Operational Experience
    - AMTE Report

Expert Meeting from All the Utilities (JANSI)

- Ageing Mechanism Management Chart
  - Additional Information including consumables
  - Proposal for Updates

Academic Society (AESJ)

- Summary Sheets of Ageing Mechanisms
  - Research for Ageing Management
  - Subcommittee for Plant Life Management

Regulator (NRA)

- Endorsement of the AESJ Code for Regulatory Activities
  - Participation

(JANSI: Japan Nuclear Safety Institute)
<table>
<thead>
<tr>
<th>No.</th>
<th>Function</th>
<th>Location</th>
<th>Material</th>
<th>Ageing phenomenon</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Upper shell, intermediate shell, lower shell, Bottom head</td>
<td>Low alloy steel (overlaid with stainless steel)</td>
<td>Fatigue cracking</td>
<td>Neutron irradiation embrittlement (intermediate and lower shells)</td>
</tr>
<tr>
<td>2</td>
<td>Top head flange, upper shell flange</td>
<td>Low alloy steel (overlaid with stainless steel)</td>
<td>Fatigue cracking</td>
<td>Pinning</td>
</tr>
<tr>
<td>3</td>
<td>Coolant inlet nozzle</td>
<td>Low alloy steel (overlaid with stainless steel) [safe end is stainless steel; weld metal is Inconel 600 alloy]</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking (weld metal)</td>
</tr>
<tr>
<td>4</td>
<td>Coolant outlet nozzle</td>
<td>Low alloy steel (overlaid with stainless steel) [safe end is stainless steel; weld metal is Inconel 600 alloy]</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking (weld metal)</td>
</tr>
<tr>
<td>5</td>
<td>Safety injection nozzle</td>
<td>Low alloy steel (overlaid with stainless steel) [safe end is stainless steel; weld metal is Inconel 600 alloy]</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking (weld metal)</td>
</tr>
<tr>
<td>6</td>
<td>Top head nozzle</td>
<td>Inconel 600 alloy</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking (including weld metal)</td>
</tr>
<tr>
<td>7</td>
<td>Air vent nozzle</td>
<td>Inconel 600 alloy</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking (including weld metal)</td>
</tr>
<tr>
<td>8</td>
<td>Bottom mounted instrumentation nozzle</td>
<td>Inconel 600 alloy</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking (including weld metal)</td>
</tr>
<tr>
<td>9</td>
<td>Radial support</td>
<td>Inconel 600 alloy</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking (including weld metal)</td>
</tr>
<tr>
<td>10</td>
<td>Stud bolt</td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Corrosion</td>
</tr>
<tr>
<td>11</td>
<td>Vessel support metal</td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Fatigue cracking</td>
</tr>
<tr>
<td>12</td>
<td>O-ring</td>
<td>Low alloy steel (Consumables and periodic replacement parts)</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking (including weld metal)</td>
</tr>
</tbody>
</table>
### Example (2)

**PWR Pressurizer**

<table>
<thead>
<tr>
<th>No.</th>
<th>Function</th>
<th>Location</th>
<th>Material</th>
<th>Ageing phenomenon</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Upper head, upper shell,</td>
<td>Low alloy steel (overlaid with stainless steel)</td>
<td>Under clad cracking (UCC)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>intermediate shell, lower shell,</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Bottom head</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>Manhole seat</td>
<td>Low alloy steel (overlaid with stainless steel)</td>
<td>Pitting</td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>Manhole cover</td>
<td>Low alloy steel</td>
<td>Pitting</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>Manhole bolt</td>
<td>Low alloy steel</td>
<td>Corrosion</td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>Instrumentation nozzle</td>
<td>Stainless steel</td>
<td>Stress corrosion cracking</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>Heater sleeve</td>
<td>Stainless steel</td>
<td>Stress corrosion cracking</td>
<td></td>
</tr>
<tr>
<td>7</td>
<td>Spray line nozzle</td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking</td>
</tr>
<tr>
<td>8</td>
<td>Spray line nozzle</td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking</td>
</tr>
<tr>
<td>9</td>
<td></td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking</td>
</tr>
<tr>
<td>10</td>
<td>Spray line nozzle</td>
<td>Low alloy steel</td>
<td>Stress corrosion cracking</td>
<td>(weld metal)</td>
</tr>
<tr>
<td>11</td>
<td>Surge nozzle</td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking</td>
</tr>
<tr>
<td>12</td>
<td></td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking</td>
</tr>
<tr>
<td>13</td>
<td></td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking</td>
</tr>
<tr>
<td>14</td>
<td></td>
<td>Low alloy steel</td>
<td>Stress corrosion cracking</td>
<td>(weld metal)</td>
</tr>
<tr>
<td>15</td>
<td></td>
<td>Low alloy steel</td>
<td>Stress corrosion cracking</td>
<td>(weld metal)</td>
</tr>
<tr>
<td>16</td>
<td></td>
<td>Low alloy steel</td>
<td>Fatigue cracking</td>
<td>Stress corrosion cracking</td>
</tr>
<tr>
<td>17</td>
<td></td>
<td>Low alloy steel</td>
<td>Stress corrosion cracking</td>
<td>(weld metal)</td>
</tr>
<tr>
<td>18</td>
<td>Safety valve nozzle</td>
<td>Low alloy steel</td>
<td>Stress corrosion cracking</td>
<td></td>
</tr>
</tbody>
</table>
Knowledge Transfer between IGALL and the AESJ Code

- The combinations of intended and required safety functions, portion of SSCs and the ageing mechanism/effect are summarized in the summary sheet in the Attachment A of the AESJ code, based on the experience from the 17 AMTEs in Japan.

- Knowledge-base from IAEA IGALL is currently under review by AESJ members be reflected in the next major revision, including the lessons learned from the Fukushima accident.
Knowledge Transfer from IGALL to the AESJ code

In the AESJ code, specific methods for implementing a technical evaluation of the ageing phenomena are defined in Attachment C.

Knowledge from these evaluation methods in AMPs and TLAAs of the IGALL is reviewed and taken into Attachment C if necessary.

Attachment C of the AESJ code

- Low-cycle fatigue
- Neutron irradiation embrittlement
- Irradiation assisted stress corrosion cracking
- High-cycle thermal fatigue
- Thermal ageing of duplex stainless steel
- Frettling fatigue
- Insulation degradation of electric and instrumentation equipment
- Reduced strength and shielding performance of concrete
Domestic Steering Team Members for IGALL
Reflection of IGALL Knowledge in the NRA

- The Regulatory Standards and Research Department of Secretariat of Nuclear Regulatory Authority (NRA) has participated in the IAEA IGALL (International Generic Ageing Lessons Learned program) project to investigate technical Knowledge on ageing management of SSCs used in nuclear power plants.

- NRA will issue a Technical Report, which summarizes the results of surveys on the issues of electrical and I&C equipment discussed in the IAEA IGALL, identifying the importance of technical basis for environmental qualification (EQ) program for electrical and I&C equipment.
Phase 2 Project

**JAMPSS**

Japan Ageing Management Program on Systems Safety
(October 2011-March 2016)
**funded by Nuclear Regulatory Authority (NRA)**

- Management for Systems Safety (Defense-in-Depth)
- Knowledge-base for Plant Life Management of LWRs

- Research for Ageing Degradation
  - Radiation Embrittlement
  - Irradiation Assisted Stress Corrosion Cracking
  - Cable Degradation
  - Concrete Structure Degradation
  - Other Issues
## Four Major Categories in the Strategy Roadmap for Ageing Management and Safe Long Term Operation

<table>
<thead>
<tr>
<th>(1) Safety Research on Ageing Degradation</th>
<th>(2) Establishment of Codes and Standards</th>
<th>(3) Establishment of Technical Information Basis</th>
<th>(4) International Collaboration</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Ageing Phenomena</strong></td>
<td><strong>Current Status</strong></td>
<td><strong>Mid Term Issues</strong></td>
<td><strong>Long Term Issues, Next Generation Reactors</strong></td>
</tr>
<tr>
<td>Radiation Embrittlement</td>
<td>Countermeasures are being taken by rule of thumb based on plant data.</td>
<td>Predication methods and monitoring technology will be upgraded.</td>
<td>Advanced designs will be developed based on past operating experience.</td>
</tr>
<tr>
<td>Stress Corrosion Cracking</td>
<td>Countermeasures suited to materials in use are being taken. Database is being constructed.</td>
<td>Use of SCC-resistant materials will be ensured. Database will be constructed.</td>
<td>Simulation methods will be established. ISI technology will be upgraded.</td>
</tr>
<tr>
<td>Fatigue</td>
<td>Countermeasures suited to materials in use or environment are being taken, and database is being constructed.</td>
<td>Countermeasures suited to materials in use or environment will be taken, and database will be constructed.</td>
<td>Countermeasures suited to materials in use or environment will be taken, and database will be constructed.</td>
</tr>
<tr>
<td>Wall Thinning</td>
<td>Countermeasures suited to materials in use are being taken, and database is being constructed.</td>
<td>Mechanism-based predication methods will be established. Risk-based maintenance methods will be established.</td>
<td>Monitoring technology will be upgraded.</td>
</tr>
<tr>
<td>Cable Insulation</td>
<td>Countermeasures suited to materials in use are being taken. Database is being constructed.</td>
<td>Deterioration diagnosis technology will be upgraded.</td>
<td>Monitoring technology will be upgraded.</td>
</tr>
<tr>
<td>Strength of Concrete</td>
<td>Intensive study is being made on scarcely known fields.</td>
<td>The reliability of integrity evaluation methods will be improved or enhanced. CCV integrity evaluation methods will be established.</td>
<td>Past records of performance will be reflected in the maintenance technology and durability designs for new plants. Recycling methods will be established for replaced structures and materials.</td>
</tr>
</tbody>
</table>
International Collaboration

**OECD/NEA**
- CNRA*
- LTO*
- CSNI*
- OPDE*
- SCAP*
- CODAP*
- IAGE*

**IAEA**

**ISaG** (International Symposium)

International Symposium on the Ageing Management and Maintenance of Nuclear Power Plants -

**Other International Collaboration**
- PARENT*
- RCOP-1 & 2*
- IGRDM*

*1 Committee on Nuclear Regulatory Activities (CNRA)
*2 Long Term Operation (LTO) Regulatory Guidance Green Booklet
*3 Committee on the Safety of Nuclear Installations (CSNI)
*4 OECD Piping Damage Data Exchange (OPDE)
*5 SCC and Cable Ageing Project (SCAP)
*6 Component Operation Experience, Damage, Aging Program (CODAP)
*7 Integrity of Components and Structures (IAGE)
*8 Program to Assess the Reliability of Emerging Nondestructive Techniques (PARENT)
*9 Feedback of Operational Experiences at NPPs in North east Asia Regional Cooperative Projects (RCOP-1)
*10 International Group on Radiation Damage Mechanisms in Pressure Vessel Steels (IGRDM)
5. New Coordinated Research Project in IAEA
IAEA New Coordinated Research Project


The first Meeting of the CRP on June 11 – 13, 2014 in Vienna
New IAEA Coordinated Research Project

**Title of the CRP**
Evaluation of structure’s and component’s material properties utilizing actual aged materials removed from decommissioned reactors for safe LTO

**Project Preparation**
- Initial proposal to IAEA prepared by Japan in 2012
- Approved in IAEA in November 2012 for 2014 -2015 biannual plan
- Coordination Meetings
  - February 2013 – Vienna
  - July 2013 – Tokyo
- Draft of “Proposal for a New Coordinated Research Project“ prepared

**Official Meetings**
- First Meeting of the CRP in Vienna in June, 2014 (participants from 12 countries and 2 international institutes)
- Second Meeting in Spain (Madrid and Zorita) in January 13-15, 2015
To establish international collaboration aimed to collecting, measuring, recording and analysis of properties of sample materials removed from SSCs of decommissioned NPPs or replaced components, which are subject to physical ageing,

- addressing synergetic effects of combination of different degradation mechanisms in real operational conditions,
- providing basis for comparison with results of laboratory tests and calculations; and
- providing possibility for removing of unnecessary conservatism.
New IAEA Coordinated Research Project

The specific objective of the CRP will address degradation mechanisms of mechanical, electrical and I&C components and also structures,

**Phase 1 (2014-2017)**
1. RPV
2. Core internals
3. Concrete structures, and other non-metallic materials

**Additional Topics in Phase 2 beyond 2017**
(can be initiated in 2015 or later)
1. Low-cycle fatigue including environmentally assisted fatigue
2. Degradation of cable insulation and electrical and I&C penetrations
Subgroup for Coordination of Harvesting and Joint Analysis of RPV Materials

Scope:
To share opportunities for harvesting of RPV materials, discuss and coordinate exchange of specimens for joint or shared analysis and coordinate with other international technical groups focused on aging of RPV materials (such as IGRDM, ASTM, and ASME). Joint technical research would be better accomplished by these international groups that already meet on an annual basis.

Possible candidate reactors for the CRP in 4 years:
Hamaoka and Zion
Subgroup for Coordination of Harvesting and Joint Analysis of In-core Materials

1. Develop the **Library of the Harvested Materials is the first priority**

2. Materials of highest priority and interest will depend on reactor technology but in general **austenitic SS** recognizing that there will be microstructure differences
   - a. PWRs/ VVER – SS 304 and SS 316, A 321
   - b. BWRs – SS 304, SS 304L, SS 316L, SS347
   - c. CANDU – 304L, 403m and pressure tube material Zr-2.5 Nb

3. **Short-term opportunities based on known harvested materials:**
   - a. Thimble tubes in Sweden – high dpa SS 316
   - b. Zorita baffle plates – 10, 30 and 50 dpa SS 304
   - c. UJV – has VVER materials, A 321 from German plant
   - d. SCK.CEN – has flux thimble tube materials of 40 to 80 dpa SS316 (Requires approval from Utilities)
   - e. Romania, Canada – SS 304L and 403 m, pressure tube material Zr-2.5Nb and Incoloy 800
   - f. DOE has baffle former bolts material
Topics for concrete: Irradiation effects, NDE methods
Schedule: Nothing is likely within the first 2 years. Possible results may be available in the next four years with materials from Hamaoka, Zion and Zorita.

Priority of actions:
1) Pursue Zorita concrete collaborative research;
2) BWR Hamaoka-1 reactor in Japan.

Concern: Testing will not be completed in two years. Results are restricted, available to participants only. For Japan reactors some of data will be available in 2 – 3 years. For Zorita some test coupons will be available in 1Q 2015.

This sub-group would coordinate and interface with other groups such as CSNI and ICIC in OECD/NEA which have similar activities.
Summary and Conclusion in the First Meeting

• The common item for the CRP is to have database or library for operational and decommissioned reactors with their plans for harvesting and testing materials with cost estimates.

• The library should be discussed in the Meetings of the CRP and updated for the future CRP topics.

• The participants are welcomed to provide IAEA with their plans for harvesting and testing materials. The data will be uploaded to the CRP web folder and will serve as the basis for discussion.

• Communication with international groups such as OECD/NEA CSNI IAGE, IGRDM and other community is necessary to define the future CRP topics.

Second Meeting

January 13-15, 2015 at IETcc (Instituto Eduardo Troja de Ciencias de la Construccion) in Madrid & Jose Cabrera NPP in Zorita, Spain
## Hamaoka Nuclear Power Station (Chubu)

<table>
<thead>
<tr>
<th></th>
<th>Unit 1</th>
<th>Unit 2</th>
<th>Unit 3</th>
<th>Unit 4</th>
<th>Unit 5</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Reactor type</strong></td>
<td>BWR-4</td>
<td>BWR-5</td>
<td>ABWR</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Thermal output (MWt)</strong></td>
<td>1593</td>
<td>2436</td>
<td>3293</td>
<td>3293</td>
<td>3926</td>
</tr>
<tr>
<td><strong>Containment type</strong></td>
<td>Mark-1</td>
<td>Advanced Mark-1</td>
<td>RCCV</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Power output (MWe)</strong></td>
<td>(540)</td>
<td>(840)</td>
<td>1100</td>
<td>1137</td>
<td>1380</td>
</tr>
<tr>
<td><strong>Total power output (MWe)</strong></td>
<td></td>
<td></td>
<td></td>
<td>3617</td>
<td></td>
</tr>
<tr>
<td><strong>Start of commercial operation</strong></td>
<td>March 1976</td>
<td>November 1978</td>
<td>August 1987</td>
<td>September 1999</td>
<td>January 2005</td>
</tr>
<tr>
<td><strong>Present status</strong></td>
<td>Under decommissioning (Nov. 2009~)</td>
<td>Operation Outage</td>
<td>Licensing Safety Review</td>
<td>Operation Outage</td>
<td></td>
</tr>
</tbody>
</table>

Countermeasures against new safety regulations are being implemented.
# Decommissioning Schedule in Hamaoka #1 and #2

<table>
<thead>
<tr>
<th>FY2008-</th>
<th>FY2013-</th>
<th>FY2018-</th>
<th>FY2023-</th>
<th>FY2028-</th>
<th>FY2033-</th>
</tr>
</thead>
<tbody>
<tr>
<td>Phase 1 Preparation</td>
<td>Phase 2 Dismantling Peripheral Equipment</td>
<td>Phase 3 Dismantling Reactor Zone</td>
<td>Phase 4 Dismantling Building Structures</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- Application for the Approval of the Decommissioning Plan (1. Jun. 2009)
- Approval of DP (18. Nov. 2009)

- Removal of Nuclear Fuel (SF removed from Unit-1/2, FF remain in the pool Unit-2)
- System Decontamination
- Dismantling outside RCA
- Radioactive Contamination Survey
- Dismantling Peripheral Equipment out of Reactor
- Installation of Waste Conditioning Facility
- Dismantling Reactor Zone
- Dismantling Building
- Radioactive Waste Disposal from Dismantling
Radioactive Characterization in RPV

Target Radionuclides: C-14, Co-60, Ni-63, I-129, Cs-137

C-14: Most important nuclide in safety assessment for disposal of long half life nuclide
Co-60: Easy to analyze representative corrosion product
Ni-63: For verification of the activation calculation (representative corrosion product)
I-129: For verification of the activation calculation (representative fission product)
Cs-137: Easy to analyze representative fission product

Parent elements are also subject to analyze for activation calculation.
Radiochemical laboratory will be ready in Hamaoka NPS.

Cumulative neutron exposure can be evaluated at the target points, as a result of the above investigation.
Sampling Point in and around the RPV (Unit 1)

For confirming calculation accuracy

For confirming calculation accuracy at locations with small neutron flux

Sampling with core boring (about 1m in depth)

Figure 31-1 Reactor internal structure schematic diagram

Target-1 (RPV)

Target-2 (Concrete)

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Sampling Device for RPV (Unit 1)

RPV (Sample)

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## Specification of the RPV in Hamaoka Unit-1

<table>
<thead>
<tr>
<th></th>
<th>Specification</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Reactor type</strong></td>
<td>BWR-5</td>
</tr>
<tr>
<td><strong>Thickness</strong></td>
<td>120mm</td>
</tr>
<tr>
<td><strong>Material of belt line</strong></td>
<td>SA533 Gr.B Cl.1</td>
</tr>
<tr>
<td><strong>Manufacturing method</strong></td>
<td>Welding of rolled plates</td>
</tr>
<tr>
<td><strong>Material of inner clad</strong></td>
<td>309 stainless steel (weld metal)</td>
</tr>
<tr>
<td><strong>Major impurities</strong></td>
<td>0.09wt%Cu, 0.56wt%Ni</td>
</tr>
<tr>
<td><strong>Operated time</strong></td>
<td>144,570 hours (approx. 16.5 years)</td>
</tr>
<tr>
<td><strong>Neutron fluence (max.)</strong></td>
<td>$1.5 \times 10^{18} \text{ n/cm}^2 (E&gt;1\text{MeV})$</td>
</tr>
</tbody>
</table>
Specimen for Task-1 for the Validation of Prediction Formula and Previous Embrittlement Management

1/3-Charpy specimen
(3.3 × 3.3 × 23.6mmL)
Charpy Test

APT needle specimen
(1mm)

TEM and PAS disk
(3mmΦ)

Microstructural Observation
Specimen for Task-2 to Confirm the Safety Margin

CT specimen (10 × 10 × 4mm)
Toughness tests

Tensile specimen (50 × 12 × 1mm)
Tensile tests
The Objective of Hamaoka Unit-1 Project on RPV by Chubu Electric Power Co.

Examination of the samples from the RPV materials of Hamaoka Unit-1 of Chubu Electric Power Co. Inc,

1) To validate past irradiation embrittlement management based on surveillance specimens by confirming that $\Delta R_{NT}$ of the RPV in Hamaoka unit-1

2) To Assess and verify the correctness and representativeness of prediction curve formula by conducting microscopic observation

3) To confirm safety margin in present management for RPV irradiation embrittlement by fracture toughness measurement
Managing Radiation Embrittlement in Hamaoka Unit-1

Surveillance (Charpy impact) tests and Prediction curve has been used for embrittlement management in Japan (BWR)

- **①**: Embrittlement Prediction curve (Calc.)
- **②**: Calc.+Margin1
- **③**: Calc.+Margin2

1st : 1yr operated
2nd: 5yr operated
3rd : 24yr operated

1.5 × 10^{18} n/cm^2 (Pre-estimation)
## Schedule for RPV Examination in Hamaoka

<table>
<thead>
<tr>
<th>Task</th>
<th>Year 2014</th>
<th>Year 2015</th>
<th>Year 2016</th>
<th>Year 2017</th>
<th>Year 2018</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>IAEA Meetings</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
| Task #1-1 | Mechanical properties measurement  
<1/3 Charpy impact test> |  |  | Unirr. |  |
| Task #1-2 | Microstructure observation (APT etc.)  
followed by comparison with prediction formula |  |  |  | Irr. |
| Task #2 | Mechanical properties measurement  
<Fracture Toughness test> |  |  |  |  |

Unirr.: Test of un-irradiated specimens, Irr.: Test of irradiated specimens

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Materials Database for Concrete Structure

(A) Data Collection from Core Sampling and NDT

(A-1) Core Sampling
- Candidate Sampling Locations
  - Reactor Building
  - RPV Pedestal / Primary Shielding Wall
  - Spent Fuel Storage Pool / Walls etc.

(A-2) Nondestructive Testing
- Impact elastic wave method

Example of Materials Database

<table>
<thead>
<tr>
<th>Properties</th>
<th>RPV Pedestal</th>
<th>Primary Shield Wall</th>
<th>Seismic Wall</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>a1</td>
<td>a2</td>
<td>a3</td>
</tr>
<tr>
<td>Strength/Stiffness</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Carbonation Depth</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Salt Content</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Density</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Free Water Content</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Chemical Bound Water Content</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pore Size Distribution</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>X-Ray Diffraction</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Wave Propagation Rate</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>...</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Sampling Locations
- a1～a3
- b1～b3
- c1～c3

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**Comprehensive Soundness Evaluation Method on Compressive Strength of Concrete**

**A)** Data Collection from Core Sampling and NDT

(A-1) Core Sampling  
(A-2) Nondestructive Testing

**B)** Numerical Analysis Approach

Analysis Conditions of Reactor Building
- Material Properties (cement, aggregates etc.)
- Mix Proportion, Dimensions of Members
- Operation History (temp., humidity, flux, dose rates)

Effects of Operation History

**C)** Validation of Analysis Approach

![Graph showing variation of compressive strength over operation periods](image)

- Data from Lab Tests  
- Data from Aged Structures by Core Sampling and NDT

**D)** Comprehensive Soundness Evaluation

- Core Sampling  
- Nondestructive Testing  
- Numerical Analysis Approach  
- Assessment of Concrete Strength

Strength Evaluated >= Fc

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## Schedule for Concrete Examination

<table>
<thead>
<tr>
<th>IAEA Meetings</th>
<th>2014</th>
<th>2015</th>
<th>2016</th>
<th>2017</th>
<th>2018</th>
</tr>
</thead>
<tbody>
<tr>
<td>(A-1)Performing Core Sampling and Property Tests for Database</td>
<td>Planning</td>
<td>Core Property Data</td>
<td>Core Property Data</td>
<td>Core Property Data</td>
<td>Core Property Data</td>
</tr>
<tr>
<td>(A-2)Performing Nondestructive Test and Obtaining Property Data for NDT Validation</td>
<td>Planning</td>
<td>Property Data by NDT</td>
<td>Property Data by NDT</td>
<td>Property Data by NDT</td>
<td>Property Data by NDT</td>
</tr>
<tr>
<td>(B,C)Performing Numerical Analysis Approach for Validation and Upgrading</td>
<td>Planning</td>
<td></td>
<td>Outline of Approach</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(D)Proposal of Comprehensive Soundness Evaluation Methods</td>
<td>Planning</td>
<td></td>
<td>Outline of Method</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

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6. The Fukushima Daiichi Accident and the Effect of Ageing on the Accident

Seismic Effects on the Integrity of Components considering Ageing Degradation in Fukushima Dai-ichi
Estimated Level of Water
- Reached top of active fuel in 3 hours after the scram (around 18:00 on March 11)
- Reached bottom of active fuel in 4 and a half hours after the scram (around 19:30 on March 11)

Water Injection
- Fresh water injection started around 5:50 on March 12.
- Water injection stopped fresh around 14:50 on March 12.

Hydrogen Explosion on 15:36

Sea water injection started around 20:00 on March 12.
Primary System Modeling

Lower Plenum Modeling

Containment Modeling

Example of the MAAP code
Some measure values used for model justification are also shown.
MCCI: Molten Core Concrete Interaction in Unit 1

Estimated Ablation Depth

- MAAP/DECOMP code
  - 0.65 m
- COCO code
  - 0.45 m: Basement
  - 0.38 m: Side Wall

Estimated Ablation Depth

- MAAP/DECOMP code
  - 0.65 m
- COCO code
  - 0.45 m: Basement
  - 0.38 m: Side Wall

“Development of Thermodynamic Database for U-Zr-Fe-O-B-C- FPs System,”
**Generation of Hydrogen in Unit 1**

Zr-H$_2$O reaction at high temperature is main source of H$_2$.

Zr + 2H$_2$O $\rightarrow$ ZrO$_2$ + 2H$_2$

**Release Rates of Fission Products in Various Chemical Forms in Unit 1**

Noble Gas

- CsI
- TeO$_2$
- SrO

Graph showing release rates of fission products over time.
Distribution of CsI and CsOH in Various Location in Unit 1 (MAAP code)

Distribution of CsI

Outside of PCV

S/C of PCV

D/W of PCV

RPV

Distribution of CsOH

Outside of PCV

S/C of PCV

D/W of PCV

RPV

PCV: Primary Containment Vessel

RPV: Reactor Pressure Vessel

D/W: Drywell of PCV

S/C: Suppression Chamber of PCV
Fundamental Safety Functions for Light Water Reactors

1. High-pressure Coolant Injection
2. Depressurization
3. Low-pressure Coolant Injection
4. Heat Removal to Ultimate Heat Sink (Ocean)
Cesium Deposition

April 29, 2011
Monitoring of Radiation Dose Change in 2 Years

November 5, 2011 → November 19, 2013

Decrease of Dose Rate through
(1) Radioactive decay
(2) Reduction by natural dispersion
(3) Decontamination action

μ Sv/hr

- 19.0 <
- 9.5 - 19.0
- 3.8 - 9.5
- 1.9 - 3.8
- 1.0 - 1.9
- 0.5 - 1.0
- 0.2 - 0.5
- 0.1 - 0.2
- ≤ 0.1
The Fukushima Daiichi Nuclear Accident
Final Report of AESJ Investigation Committee

This report was completed by the AESJ, Japanese academia in nuclear science/engineering, with concerted effort by specialists. It covers all aspects of technical issues of the Accident.
**Maximum Acceleration Observed in Fukushima Daiichi**

<table>
<thead>
<tr>
<th>Location of Seismometer (bottom floors of the reactor buildings)</th>
<th>Records</th>
<th>Max. Response Acceleration to the Design Basis Ground Motion, Ss (Gal)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Max. Acceleration (Gal)</td>
<td>NS</td>
</tr>
<tr>
<td></td>
<td></td>
<td>NS</td>
</tr>
<tr>
<td>Fukushima Daiichi</td>
<td>Unit 1</td>
<td>460(^\times)(^1)</td>
</tr>
<tr>
<td></td>
<td>Unit 2</td>
<td>348(^\times)(^1)</td>
</tr>
<tr>
<td></td>
<td>Unit 3</td>
<td>322(^\times)(^1)</td>
</tr>
<tr>
<td></td>
<td>Unit 4</td>
<td>281(^\times)(^1)</td>
</tr>
<tr>
<td></td>
<td>Unit 5</td>
<td>311(^\times)(^1)</td>
</tr>
<tr>
<td></td>
<td>Unit 6</td>
<td>298(^\times)(^1)</td>
</tr>
</tbody>
</table>

\(^1\): Each recording was interrupted at around 130-150(s) from recording start time

\(^2\): 1 Gal=0.01m/s\(^2\) , 981 Gal=1G

Source: Added to “The impact of the 2011 off the Pacific coast of Tohoku Earthquake to Nuclear Reactor Facilities at Fukushima Dai-ichi Nuclear Power Station” (Sep. 9, 2011, revised Sept. 28, 2011, TEPCO)
Effects of Ageing in the Fukushima Accident?

• As the results of the evaluation based on the knowledge obtained so far, it is quite unlikely that there was an effect of ageing degradation on loss of functions in SSCs important to safety due to the ground motion by the earthquake.

• It is also unlikely that ageing degradation phenomena have caused the occurrence and enlargement of the Fukushima Daiichi accident.

• However, as it is difficult to confirm the status of equipment at this moment, additional investigation will be needed, when new knowledge is obtained in the future.

**Evaluation of Additional Effect of the Earthquake on Low Cycle Fatigue of Main Steam Line Piping of Unit 1**

<table>
<thead>
<tr>
<th>Cumulative Fatigue Coefficient for 60 years</th>
<th>Fatigue Coefficient considering Earthquake</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Due to Seismic Load</td>
</tr>
<tr>
<td>0.064</td>
<td>( S_2 )</td>
</tr>
<tr>
<td></td>
<td>( S_s )</td>
</tr>
<tr>
<td>Earthquake on March 11, 2011</td>
<td>0.264</td>
</tr>
</tbody>
</table>

Equivalent number of cycle due to seismic load is conservatively assumed to be 100. \( S_2, S_s \) : Design basis seismic ground motion.
### Evaluation of Effect of the Earthquake on March 11, 2011 on Structural Integrity of Important Pumps considering General Corrosion of Anchor Bolts

<table>
<thead>
<tr>
<th>Pumps Evaluated</th>
<th>Shear Stress [MPa]</th>
<th>Allowable Stress [MPa]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Without Corrosion</td>
<td>With Corrosion for 60 years</td>
</tr>
<tr>
<td>Unit 1 Reactor Shut Down Cooling System Cooling Pumps</td>
<td>8</td>
<td>9</td>
</tr>
<tr>
<td>Unit 2 Residual Heat Removal System Pumps</td>
<td>34</td>
<td>36</td>
</tr>
<tr>
<td>Unit 3 Residual Heat Removal System Pumps</td>
<td>23</td>
<td>24</td>
</tr>
</tbody>
</table>

- The corrosion of the anchor bolts was evaluated by multiplication of cross-section decrease rate (6.0%/y) in consideration of the corrosion amount for 60 years of 0.3 mm.
- The shear stress with consideration of the amount of corrosions for 60 years was confirmed that there was sufficient margin to the allowable stress.
Thank you very much for your attention