Boiling Water Reactor Vessel and Internals Project (QA)

Program Overview

Program Description

As boiling water reactors have aged, various forms of operation-limiting stress corrosion cracking have appeared, first in the recirculation piping, then in the reactor pressure vessel internals. Typically, poor materials performance has been addressed by focusing on the specific component or system. This near-term, reactive approach has resulted in costly unplanned outages and expensive weld-by-weld mitigation and repair methods. A longer-term, strategic approach can address a broader range of factors impacting pressure vessel internals.

The Boiling Water Reactor Vessel and Internals Project (BWRVIP) provides an integrated approach for managing materials-related degradation issues in reactor coolant system components in boiling water reactors. The program assesses all facets of operation, maintenance, and repair to develop reliable and cost-effective detection, inspection, and mitigation techniques.

Research Value

BWRVIP maintains alignment with current industry internals integrity concerns affecting boiling water reactors. Research results lead to cost-effective solutions to reduce damage related to stress corrosion cracking; cost savings due to reduced inspection scope, extended intervals between inspections, and improved operating characteristics; reduced personnel radiation exposure; and improved models to better characterize the mitigation of internals components. BWRVIP participants gain access to the following:

- Strategic roadmaps outlining research gaps confronting key issues—such as jet pump vibration and internals degradation—and the collaborative actions needed to address these gaps.
- Technologies and technical guidance that drive increased capacity factors.
- Cost-effective techniques to mitigate stress corrosion cracking of reactor internal components. Economic evaluations indicate that cost savings for implementing hydrogen water chemistry or noble metal chemical application exceed $40 million per plant.
- Cost-effective options for replacing or repairing reactor components.
- Technical solutions to internals inspection needs.
- Industry operating experience and technical insights to optimize inspection requirements, reduce outage critical path times, and inform regulatory decisions.

Approach

The BWRVIP Program takes an integrated approach to degradation management, encompassing assessment, mitigation, and inspection. Through improved inspection techniques, new results from materials research and development, and plant operating experiences, best practices can be deployed to make cost-effective decisions. BWRVIP closely collaborates with other Electric Power Research Institute (EPRI) programs, including Nondestructive Evaluation and Chemistry, to ensure appropriate technologies and technical guidance are effectively integrated into research activities.

There are both base and supplemental components to the BWRVIP research program. The base program focuses on improving the understanding of materials performance in areas such as fracture toughness of stainless steel exposed to high fluence levels, weldability of irradiated materials, and crack growth rates. Because these factors can all impact materials aging, enhanced understanding of their interactions and their impact on materials performance is essential. Research results are provided in the form of guidelines to ensure prompt detection of material degradation, technical reports to support materials performance assessments, and cost-effective tools to more effectively identify and manage degradation for current and extended operations.

To address strategic objectives established for each of its programs, EPRI has developed roadmaps to plan, coordinate, and execute needed research among multiple entities. For the BWRVIP Program, specific roadmaps
have been developed to address flow-induced jet pump vibration and the degradation of boiling water reactor (BWR) internals. A few roadmaps being led by other programs also will impact BWRVIP, including roadmaps on the welding of irradiated materials and irradiated materials testing. Additional roadmaps will be developed as conditions warrant.

The supplemental portion of the BWRVIP program encompasses research related to assessment, inspection, repair and mitigation.

- **Assessment:** BWRVIP develops inspection and evaluation guidelines that provide the scope for what needs to be inspected and a methodology for evaluating or repairing any indications.
- **Inspection:** BWRVIP develops advanced nondestructive evaluation techniques to improve detection of indications in internals components.
- **Repair:** BWRVIP develops technically based repair criteria for degraded components and equips nuclear plants with information needed to safely plan and implement repairs.
- **Mitigation:** BWRVIP provides guidance for implementing effective chemistry-based countermeasures for stress corrosion cracking of reactor internal components.

Also included in the supplemental portion is an integrated task that provides overall BWRVIP technical and administrative program management (including coordination and engagement with regulators, global entities, and other stakeholders), and a surveillance program for monitoring changes in reactor pressure vessel materials properties due to neutron irradiation.

Nuclear plant owners also can participate separately in a project aimed at maintaining and improving the BWR Vessel and Internals Application (BWRVIA) software code, which performs radiolysis analysis and electrochemical corrosion potential calculations for boiling water reactors. Continued maintenance ensures the BWRVIA code reflects the latest industry operating experience and is equipped to address analytical needs.

**Accomplishments**

The BWRVIP Program supports nuclear power industry efforts to assess and implement effective countermeasures for stress corrosion cracking of reactor internal components. BWRVIP research provides utilities with the information necessary to make cost-effective decisions for managing degradation of boiling water reactor vessel and internal components.

- Issued Nuclear Regulatory Commission (NRC-)approved guidelines on steam dryer inspection and flaw evaluations that define visual inspection requirements for BWR steam dryer assemblies. Baseline inspection results can be compared to subsequent results to assess potential effects of time and power uprates. Developed repair design criteria for steam dryers.
- Published revision 13 of the examination guidelines for reactor pressure vessels and internals in 2010.
- Developed technical basis for inspection relief for reactors using hydrogen injection. This technical basis can be used to inform regulatory decisions regarding the frequency and extent of inspections.
- Issued guidelines for performing weld repairs on irradiated BWR internals, providing a mechanism for determining the weldability of reactor components.
- Compiled data to advance understanding of the relationship between fracture toughness and neutron fluence in highly irradiated stainless steel materials. Also conducted tests to collect crack growth rate data on irradiated stainless steels that can be used to extend EPRI's flaw evaluation methodology to higher neutron doses.
- Designed, developed, and demonstrated a radiographic testing system to facilitate corrosion detection in difficult-to-access BWR drain lines.
- Demonstrated online noble metal chemical addition as a mitigation technique for stress corrosion cracking. Field tests indicate that the technique is effective in reducing electrochemical corrosion potentials and has had no adverse plant impacts. Because the application is performed during operation, 60 critical path hours can be saved versus the classic noble metal application.
Current Year Activities

BWRVIP research and development for 2012 will continue to focus on the technical gaps defined in the BWR Issue Management Tables. Highest priority gaps include the impacts of fluence on the material properties of BWR materials, high-cycle fatigue in jet pump assemblies, and flow-assisted corrosion of the BWR bottom head drain line. Specific efforts will include the following:

- Continue jet pump mitigation hardware testing in full scale jet pump facility
- Support technical review of the steam dryer loads methodology as it works its way through Nuclear Regulatory Commission review
- Continue crack growth and fracture toughness evaluations of highly irradiated materials
- Develop advanced mitigation techniques for stress corrosion cracking
- Optimize inspection and flaw evaluation guidelines for selected BWR internals components

Selected reports will be developed in whole or in part under Title 10 of the Code of Federal Regulations Part 50 (10 CFR50) Appendix B, Quality Assurance, 10 CFR 21, and the EPRI Quality Assurance Program. Additional products may be developed under 10 CFR 50 Appendix B, and 10 CFR 21 at the discretion of the BWRVIP member utilities or EPRI, when such action is deemed appropriate.

Estimated 2012 Program Funding

$9.2 million

Program Manager

Randal Stark, 650-855-2122, rstark@epri.com

Summary of Projects

<table>
<thead>
<tr>
<th>Project Number</th>
<th>Project Title</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>P41.01.03.01</td>
<td>Management of Jet Pump Flow-Induced Vibration (Roadmap) (QA)</td>
<td></td>
</tr>
<tr>
<td>P41.01.03.02</td>
<td>Ensuring Reactor Pressure Vessel Integrity Through Eighty Years of Operation (Roadmap) (QA)</td>
<td></td>
</tr>
<tr>
<td>P41.01.03.03b</td>
<td>BWR Vessel and Internals Application User Group (supplemental)</td>
<td>The BWR Vessel &amp; Internals Application (BWRVIA) computer code, which performs radiolysis analysis and electrochemical corrosion potential calculations for BWRs, has progressed through several upgrades to enhance its value to the nuclear power industry. Continued maintenance through this project ensures the code reflects the latest industry operating experience and is equipped to address analytical needs. The BWRVIA User Group provides training and ongoing support to all BWR utilities using the model and participating in this program.</td>
</tr>
</tbody>
</table>
Management of Jet Pump Flow-Induced Vibration (Roadmap) (QA)

Key Research Question
Worldwide, a number of boiling water reactor plants (primarily BWR/4 and /5 designs) are experiencing jet pump degradation associated with flow-induced vibration. The primary causes for jet pump degradation from flow-induced vibration are thought to be turbulent flow through the jet pumps, slip joint leakage flow instability, and pressure pulsations from pump vane passing. Jet pump degradation is generally characterized by wedge and rod wear, but one plant recently experienced substantial cracking in a jet pump riser extending 240 degrees around the circumference. The rate and severity of observed jet pump degradation is increasing. Further, the power uprates being pursued by many plants can lead to increased flow-induced vibration loads on the jet pump assembly.

Approach
The jet pump degradation management research program includes compilation of plant data on jet pump degradation; plant operation and repair histories to understand relationships between plant operation, configuration and degradation; sub-scale phenomenological testing to study the slip joint leakage flow instability phenomenon; and full-scale testing of prototypical jet pump assemblies to assess the effectiveness of vendor-proposed flow-induced vibration mitigation solutions. In addition, lessons learned from testing, and insights from the jet pump degradation and operational history review, will be factored into revisions to jet pump inspection and repair guidelines.

Accomplishments to date include compilation of jet pump degradation and operational history information, sub-scale testing at 1/5th and 1/2 scale to better understand the slip joint instability phenomena, testing using a full scale mock-up of a typical slip joint configuration to assess the relationship between slip joint flow and differential pressure, and fabrication of a full-scale test facility that includes a BWR/4 jet pump assembly and a system to provide slip joint leakage flow up to 180 F. Tests completed to date have reproduced the SJLFI instability phenomenon and highlighted the need for on-line measurement of the slip joint gap and well as remote control of the alignment between the mixer and diffuser to provide repeatable test results.

In addition, two vendor proprietary conceptual designs for improved lateral support of the jet pump mixer/diffuser have been developed and are documented in BWRVIP-207 and 210. These designs hold promise to significantly reduce the potential for flow-induced vibration (FIV) damage to jet pump components.

Going forward, the full-scale test facility will undergo modifications to allow full flow testing of both BWR/4 and BWR/5 full-scale jet pump assemblies up to 600 psia and 320 F. Under these conditions, the facility will be capable of reproducing prototypical turbulent pressure loading, slip joint instability excitation, and pressure pulsations from recirculation pump vane passing. The facility will be used to define operating conditions and jet pump configurations that result in “worst case” flow-induced vibration loading. Based on the results, a worst case “demonstration test protocol” will be developed for use in evaluating vendor-developed flow-induced vibration mitigation solutions. At the completion of the EPRI research, the test facility will be made available to participating vendors at their cost to assess the effectiveness of their solutions in mitigating flow-induced vibration loading.

Impact
There are a number of factors that drive the need for timely resolution of the jet pump flow-induced vibration issue, including the following:

Asset Management—Asset management is the primary driver associated with the management of jet pump flow-induced vibration. Jet pump degradation has become one of the most costly BWR material issues, with industry costs currently exceeding $20 million. Several plants have experienced wedge and rod wear even after installing clamps, auxiliary wedges, or labyrinth seals. In 2008, a BWR/5 plant experienced a significant fatigue crack in a jet pump riser that resulted in several months of downtime and required a temporary repair. Currently, the plant is only allowed to operate for a single cycle at reduced power.
Anticipated Growth in the Extent and Severity of Degradation—As more plants pursue power uprates, the potential for more widespread and severe jet pump degradation increases. Power uprates result in higher back-pressure on the jet pumps, which can exacerbate jet pump degradation if mitigation techniques are not available and implemented.

Safety Impact—The integrity of the jet pump assembly is of paramount importance to safe plant operations. The jet pump ensures reflooding to at least 2/3 core height in the event of a concurrent recirculation pipe break, and for some BWRs the jet pump assembly provides a flow path for low-pressure injection.

Potential for Regulatory Action—Although the Nuclear Regulatory Commission (NRC) has not taken action to-date in response to observed jet pump degradation, increased regulatory scrutiny is possible if such degradation is not addressed by industry, particularly if another significant jet pump fatigue crack is identified.

How to Apply Results

Improved jet pump inspection guidance (BWRVIP-41R3 and NRC approved BWRVIP-41R3-A) and repair guidance (BWRVIP-51R1 and NRC-approved BWRVIP-51R1-A) will be developed applying lessons learned during the test program as well as insights from the jet pump degradation and operational history review. Nuclear plant owners will implement the revised jet pump inspection and repair guidance in accordance with NEI-08 implementation guidelines.

Results of BWRVIP testing to investigate jet pump phenomena will allow utilities to assess their relative susceptibility to jet pump degradation based on jet pump design and plant operating conditions. The results also will provide the basis for defining a “demonstration test protocol” to assess the effectiveness of proposed flow-induced vibration mitigation solutions.

Demonstration test results will be provided to BWRVIP utilities by each participating vendor and can be used to inform decisions on potential implementation of one or more jet pump mitigation solutions.

Selected reports and products may be prepared in whole or in part in accordance with the EPRI Quality Program Manual that fulfills the requirements of 10CFR50 Appendix B, 10CFR21 and ANSI N45.2-1977. Reports and products developed under the EPRI QA program will be marked and identified as such.

Ensuring Reactor Pressure Vessel Integrity Through Eighty Years of Operation (Roadmap) (QA)

Key Research Question

The ability to monitor and demonstrate the structural integrity of the reactor pressure vessel (RPV) through 80 years of operation is essential to ensure continued operation of the nuclear fleet. For an RPV subjected to years of neutron radiation, adequate demonstration of integrity will become increasingly difficult when applying existing analytical tools and correlations, which were established using the technology and analytical capabilities of the late 1970s.

Revisions to analytical tools, material property databases, and embrittlement correlations are necessary to accurately predict the service life of the RPV. Technical advances are needed for the following:

- Ensure that necessary research data is available to identify and assess fluence- and flux-based damage mechanisms at the levels anticipated through 80 years of operation.
- Establish a damage mechanism-based definition of the region of the reactor vessel that must be evaluated for maintenance of structural integrity. Improving the understanding of these damage mechanisms could help inform regulations.
- Ensure that design materials, end-of-license fluence, and flux are considered to optimize design and fabrication practices for advanced light water reactors.
Ongoing research and regulatory interactions are needed to ensure that the appropriate analytical tools and correlations are developed to analyze and model vessel integrity for safe and efficient operation through 80 years.

**Approach**

The project plan is comprised of the following elements:

**Operational Support through 80 years**
- PWR utilities will implement EPRI’s coordinated reactor vessel surveillance program beginning in 2011. This program will generate the high-fluence surveillance data and irradiated material samples needed to support embrittlement correlation databases and damage mechanism assessments at fluences representative through 80 years of operation.
- EPRI and the PWR Owners Group (PWROG) will conduct research and development (R&D) to determine whether the Code and regulatory requirements for RPV integrity could put plants at safety and economic risk in coming decades. This will be accomplished through technical support of American Society of Mechanical Engineers (ASME) Code activities and collaboration with NRC Research in several areas: Master Curve, ASME Section XI Risk-informed (RI) Appendix G, ASME Section XI Appendix E, fluence monitoring, and definition of RPV “extended beltline.”
- EPRI and the PWROG will evaluate the operational impacts related to embrittlement correlations and environmental damage projections for materials and components in the “extended beltline.”
- The BWR Integrated Surveillance Program will be re-assessed to ensure it supports 80 years of operation.

**Data Modeling**
- EPRI will support NRC development of an updated RPV database for generation of new embrittlement trend curves and surveillance data analysis.
- EPRI and the PWROG will assess appropriate stress intensity and flaw distribution models for use in assessing materials/components of the “extended beltline.”
- Data from the International Atomic Energy Association (IAEA) fluence attenuation program will be integrated into the Regulatory Guide 1.99 attenuation model. EPRI will work with NRC and the research community to appropriately identify the conservatism and accuracy of the model to be used.
- EPRI will develop tools and modeling applicable to advanced light water reactor for use in informing design and fabrication specifications.
- EPRI will develop a theoretically derived dislocation-based ferritic steel transition toughness model as input to the Department of Energy (DOE) Light Water Reactor Sustainability Program’s material analysis and modeling project.

**Regulatory and Communications**
- Since NRC is responsible for drafting and issuing regulations dealing with irradiation damage mechanisms, EPRI and the PWROG will regularly interface with the NRC Research and Regulation branches.
- EPRI will continue to integrate PWROG, BWRVIP, and MRP activities relative to RPV integrity and regularly communicate their activities with the NRC.

**Impact**

The primary driver for reactor pressure vessel integrity is to ensure that plants can safely and efficiently operate through 80 years without significant operational constraints or mitigation of RPV embrittlement. This encompasses operating restrictions associated with assuring RPV integrity during startup and shutdown activities and the assessment and evaluation of irradiation and other damage mechanisms of the RPV materials in and near the current beltline region.
How to Apply Results

The successful management of reactor vessel integrity issues helps ensure that regulations reflect known technical constraints, but also retain operational flexibility. The end product of this R&D will be EPRI reports that support the bases for ASME Code and NRC regulations affecting RPV asset management.

Additionally, within the next several years, EPRI will develop assessment tools for use in developing site-specific strategies for RPV management. Utilities will use these tools as part of integrated surveillance programs to address 80-year operation.

As data gathering and modeling proceed, the needs of the advanced light water reactor fleet will be considered to ensure that long-term plans envelope advanced designs. Moreover, where program results may affect design or fabrication practices, this guidance will be made available to vendors via the EPRI Utility Requirements Document.

Selected reports and products may be prepared in whole or in part in accordance with the EPRI Quality Program Manual that fulfills the requirements of 10CFR50 Appendix B, 10CFR21 and ANSI N45.2-1977. Reports and products developed under the EPRI QA program will be marked and identified as such.

BWR Vessel and Internals Application User Group (supplemental) (047065)

Key Research Question

Two technologies—moderate hydrogen injection, known as hydrogen water chemistry, and noble metal chemical addition—have been applied in boiling water reactors to mitigate intergranular stress corrosion cracking (IGSCC) by lowering primary water electrochemical corrosion potential. Analytical capabilities are needed to determine appropriate injection concentrations that can maintain electrochemical corrosion potential values at levels that mitigate corrosion. This user group provides information and training on the use of the radiolysis and electrochemical corrosion potential models used in EPRI's BWR Vessel and Internals Application (BWRVIA) software program. Ongoing development of the codes also is evaluated and reported at the annual member's meeting.

Approach

The BWRVIA User Group provides technology that operating BWRs can use to help mitigate IGSCC of reactor piping and internals. The technical project team performs comprehensive reviews of research and development in the areas of radiation chemistry and electrochemical corrosion potential modeling. Sensitivity analyses are performed to evaluate the model's response due to changes in input parameters such as chemical reaction rate constants and dose rate profiles. Adjustments are then made to these sensitive parameters to provide the best possible correlations. Finally, the results of the sensitivity analyses are compared to actual plant data to provide a technical basis for plant application of the calculated results.

Impact

By incorporating the current state of the art in radiation chemistry and electrochemical corrosion potential formulation into the BWRVIA code and benchmarking the revised code against all plant and laboratory data available, this user group ensures the availability of an accurate model for BWR plant owners. The model can then be used, for example, to predict the amount of hydrogen injection needed for IGSCC mitigation of susceptible reactor internals and piping in BWRs.

How to Apply Results

The BWRVIA User Group provides annual training workshops and ongoing support to run the software.