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2020 TECHNICAL REPORT

BWRVIP-167, Revision 4: BWR Vessel and Internals Project

Boiling Water Reactor Issue Management Tables

PORTIONS TRANSLATED

BWRVIP-167, Revision 4: BWR Vessel and Internals Project

Boiling Water Reactor Issue Management Tables

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ABSTRACT

Nuclear utilities continue to face issues related to degradation of boiling water reactors (BWRs) pressure vessels, reactor internals, and American Society of Mechanical Engineers (ASME) Class 1 piping components. The BWR Issue Management Table (IMT) Report identifies key gaps in industry state-of-knowledge regarding materials degradation phenomena and associated management capabilities for BWR primary system components. The IMTs satisfy the BWRVIP commitment to develop and maintain a work plan that evaluates strategic materials management issues and ensures that materials degradation is proactively addressed through funding of appropriate research and development (R&D).

A comprehensive, integrated understanding of materials issues and management options is a fundamental consideration to ensure continued safe operation, as well as the development of overall plant business and operating strategies. The set of R&D gaps identified in this report were initially developed as a means of meeting the intent of the NEI 03-08 Materials Initiative and the results are updated periodically to address both utility asset management needs and the changes to the state-of-industry knowledge regarding BWR primary system materials degradation. The gap-assessment results are a critical element in ensuring that BWRVIP program R&D strategies and priorities continue to meet the needs of BWRVIP members.

Keywords

BWR Vessel and Internals Project
Issue management tables
NEI 03-08 materials initiative
R&D gaps

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EXECUTIVE SUMMARY

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PRIMARY AUDIENCE: Utility advisors either in BWR Vessel and Internals Project (BWRVIP) committees or responsible for providing input regarding BWRVIP research and development (R&D) priorities and strategic planning

KEY RESEARCH QUESTION

The BWR Issue Management Table (IMT) is a gap assessment intended to identify and prioritize key R&D gaps regarding proactive management of BWR materials degradation. In this assessment, an R&D *gap* is defined an area of research identified as important to achieving a reasonable standard of confidence that primary system component degradation can be managed such that components will remain serviceable and capable of performing their intended functions for the remainder of plant life.

Gaps from prior versions of the BWR IMTs were reviewed to assess progress toward gap closure, as well as general changes in knowledge resulting from completed R&D or new operating experience. In cases where completed R&D indicated that a gap can be closed, the bases and rationale for closing the gap are documented. Field experience that is potentially relevant to material degradation is reviewed to identify new gaps. The resulting set of R&D gaps are prioritized into high, medium, and low priority categories.

KEY FINDINGS

- Key R&D areas include assessment of stress corrosion cracking (SCC) crack growth, management of highly irradiated components, and effective implementation of water-chemistry-based SCC mitigation technologies.
- The effort undertaken to close R&D gaps primarily related to regulatory issues resulted in the closure of a significant number of gaps, with a majority of these gaps being assessment (AS) gaps.
- Although there are open RPV integrity-related regulatory issues, remaining commitments that must be fulfilled (i.e., Integrated Surveillance Program (ISP) capsule testing and evaluation), and opportunities for further optimization of RPV integrity management methods, the IMT Revision 4 results do not include any active/open technical R&D gaps related to RPV integrity.
- Given the significant time gap between Revisions 3 and 4, there were a number of gap closures resulting from completion of relevant R&D and changes in industry needs (i.e., issue obsolescence).
- Few new gaps were identified. This is consistent with the expectation that as IMT gap assessments are repeated over time, fewer issues not identified in initial assessments are captured. Additionally, it is observed that BWR RPV and internals performance has been relatively stable in recent years. There have been limited truly unique or new aging management issues.
- What are the strategic gaps in state-of-knowledge with regard to proactively managing boiling water reactor (BWR) materials degradation? How should these gaps be prioritized?

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EXECUTIVE SUMMARY

WHY THIS MATTERS

The gap assessment results are a critical element in ensuring that BWRVIP program R&D strategies and priorities continue to meet the needs of BWRVIP members. The gap assessment additionally ensures that U.S. BWRVIP member utilities continue to meet commitments to implement the NEI 03-08 Materials Initiative.

HOW TO APPLY RESULTS

BWRVIP program advisors should understand the R&D gaps resulting from this assessment and should consider R&D gap priorities when evaluating BWRVIP program R&D plans and providing input on prioritizing R&D projects.

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PROGRAM: Boiling Water Reactor Vessel and Internals Program (BWRVIP), P41.01.03

IMPLEMENTATION CATEGORY: CATEGORY 1

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ACRONYMS AND ABBREVIATIONS

Acronym/Abbreviation	Definition
ABWR	Advanced Boiling Water Reactor
AHC	Access Hole Cover
AS Gap	Assessment R&D Gap
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&PVC	Boiler & Pressure Vessel Code
BWR	Boiling Water Reactor
BWROG	BWR Owners' Group
BWRVIA	BWR Vessel and Internals Application
BWRVIP	Boiling Water Reactor Vessel and Internals Project
C&LAS	Carbon and Low Alloy Steel
CASS	Cast Austenitic Stainless Steel
CGR	Crack Growth Rate
CIR	Cooperative IASCC Research
CMTR	Certified Mill Test Report
CRGT	Control Rod Guide Tube
CUF	Cumulative Usage Factor
DM Gap	Degradation Mechanism Understanding R&D Gap
ECP	Electrochemical Corrosion Potential
EPRI	Electric Power Research Institute
ETC	Embrittlement Trend Curve
EVT	Enhanced Visual Test
FAC	Flow Accelerated Corrosion
FEA	Finite Element Analysis
FIV	Flow Induced Vibration
GE	General Electric Company
HAZ	Heat Affected Zone
HWC	Hydrogen Water Chemistry
I&E	Inspection & Evaluation

Acronym/Abbreviation	Definition
IASCC	Irradiation Assisted Stress Corrosion Cracking
IN Gap	Inspection R&D Gap
ISP	Integrated Surveillance Program
IMT	Issue Management Table
LAS	Low Alloy Steel
LPCI	Low Pressure Coolant Injection
LRA	License Renewal Application
LTCP	Low Temperature Crack Propagation
LTO	Long Term Operation
MDM	Materials Degradation Matrix
MT Gap	Mitigation R&D Gap
NDE	Nondestructive Examination (or Evaluation)
NEI	Nuclear Energy Institute
NMCA	Noble Metal Chemical Application
NPS	Nominal Pipe Size
NRC	Nuclear Regulatory Commission
NWC	Normal Water Chemistry
OLNC	On-line NobleChem™
PWR	Pressurized Water Reactor
R&D	Research and Development
RAMA	Radiation Analysis Modeling Application
R.G.	Regulatory Guide
RHR	Residual Heat Removal
RIC	Research integration Committee
RPV	Reactor Pressure Vessel
RQ	Program Requirement
RR Gap	Repair/Replacement R&D Gap
SBP	Small Bore Piping
SCC	Stress Corrosion Cracking
SLC	Standby Liquid Control
SSP	Supplemental Surveillance Program
USE	Upper Shelf Energy
UT	Ultrasonic Testing
UV	Ultra Violet
VT	Visual Test

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1
INTRODUCTION

1.1 Purpose

The BWR issue management tables (IMTs) proactively identify and prioritize BWR materials degradation knowledge state and management capability R&D gaps. The results are a critical element in ensuring that BWRVIP program R&D strategies and priorities continue to meet the needs of BWRVIP members. The gap assessment additionally ensures that U.S. BWRVIP member utilities continue to meet commitments to implement the NEI 03-08 Materials Initiative.

1.2 Background

Initial IMT development included comprehensive identification of all of the major BWR nuclear steam supply system subcomponents. For each subcomponent, the applicable degradation mechanisms, consequences of failure, mitigation techniques, repair / replacement approaches and inspection & evaluation guidance were summarized. Gaps in the current state of knowledge with regard to understanding relevant degradation mechanisms or managing materials degradation were identified in a set of "R&D gaps."

1.3 BWR IMT Content Management and Approach Changes

This Revision 4 of the IMTs includes several significant and important changes in approach that users familiar with prior versions of the IMTs will notice. These most significant changes are summarized below. See the Appendix A revision log for additional changes.

1.3.1 Component Table Maintenance

In Revision 3 and prior revisions of the BWR IMTs, Appendix A included a set of tables that were created as an aid for initial IMT development and early maintenance. These tables documented evaluation of BWR primary system components. However, maintenance of these tables in subsequent versions of the IMT has not been important to the R&D gap assessment results. Further, it is observed that at least some of the content within these tables becomes out of date quickly as new guidance documents or revisions to guidance documents are issued. Therefore, starting with Revision 4, these component tables will no longer be included in the IMT document. Abbreviated versions of these tables are available to BWRVIP members in the Library folder on the BWRVIP issue program cockpit.[1]

[1] www.membercenter.epri.com > Program Cockpits > P41.01.03 > Library > Issue Management Tables (IMTs)

1.3.2 Regulatory Issue Tracking

In Revision 3 and prior of the IMT document, a separate R&D gap category for regulatory issues was included. The content of this section was incomplete from the perspective that it primarily reflected a subset of U.S. licensing issues. Further, regulatory concerns were often interspersed within technical gaps. It was also observed that regulatory driven gaps represented a distraction from the IMT objective of proactive identification and resolution of materials degradation issues. Starting with Revision 4, an effort has been made to clearly separate technical gaps and regulatory issues. These issues are now documented as relevant to specific member countries and captured in a separate regulatory issues matrix document available to BWRVIP members in the Library folder on the BWRVIP issue program cockpit.[2]

1.3.3 Tracking Interim Changes to R&D Gaps

Instead of republishing the BWR IMT report, it has been found useful in the past to make minor periodic adjustments to gap content and priorities. Beginning in this Revision 4, the BWR IMT R&D gaps will be maintained in a document available to BWRVIP members in the Library folder on the BWRVIP issue program cockpit.[2] Interim changes to R&D gaps will be managed and tracked using this document on the BWRVIP cockpit. Periodically, any changes to gaps will be communicated to BWRVIP members.

1.3.4 Tracking Closed R&D Gaps

Prior revisions of the IMTs retained placeholders for closed gaps in the Section 3 results tables. In Revision 4 and going forward, placeholders for closed gaps will not be included. Appendix B, Gap Closure, provides the details associated with gaps closed in this revision of the BWR IMTs. Appendix A includes a listing of all BWR IMT gaps, including closed gaps and the revision in which the gap was closed. BWRVIP members can access the full content of closed gaps and associated closure bases for earlier IMT revisions in the Library folder on the BWRVIP issue program cockpit.[2]

1.3.5 R&D Gap Focus

The definition of an R&D gap was adjusted in Revision 4 to be more consistent with actual use. Additionally, although the NEI 03-08 materials initiative is binding only for U.S. utilities, it has been recognized that the proactive approach provided by NEI 03-08 provides significant value to the broader international community. As best possible, R&D gaps have been modified and broadened to capture issues that may be unique to international members and to ensure gap aspects important to international members are addressed.

1.4 Implementation Requirements

This report is provided for information only. Therefore, the implementation requirements of Nuclear Energy Institute (NEI) 03-08, Guideline for the Management of Materials Issues [1], are not applicable.

[2] www.membercenter.epri.com > Program Cockpits > P41.01.03 > Library > Issue Management Tables (IMTs)

2
SCOPE AND APPROACH

2.1 Evaluation Scope

2.1.1 Systems and Components

The scope of BWR systems and components evaluated includes the reactor pressure vessel (RPV), reactor internals, and primary pressure boundary piping systems. Consistent with the typical approach for defining the scope of materials management programs in the U.S., only passive components (i.e., components that perform their primary function without motion or a change in state) and long-lived components (i.e., components that are not periodically replaced with new or refurbished components) are included in the scope. Active components (e.g., control rod drives, valve internals and pump rotating assemblies) and short-lived components (e.g., fuel assemblies and in-core instruments) are excluded from the evaluation.

RPV components in the scope of the evaluation include vessel shells, top and bottom heads, flanges, nozzles, nozzle safe ends, bottom head penetrations, housings for control rod drives, instruments or reactor internal pumps, instrument penetrations, and ID and OD welded brackets and supports.

Depending on reactor design, reactor internals evaluated include the steam dryer, shroud head and steam separators, top guide, core shroud, core plate, shroud support, access hole cover, fuel supports, control rod guide tubes (CRGTs), core spray internals, jet pumps, low-pressure coolant injection (LPCI) couplings, feedwater spargers, surveillance capsule holders, and guide rods. Both safety related and non-safety related reactor internals are evaluated. Note that repair hardware is typically not addressed since details regarding repair designs and materials are not generally available. However, if a generic materials issue was identified to the BWRVIP related to materials used for repair hardware, that issue would be considered in the gap evaluation process.

Depending on reactor design basis, the ASME Class 1 pressure boundary typically includes piping, valves and fittings out to the second isolation valve. This typically results in all or portions of the following systems being included in this scope: Reactor Recirculation (BWR/2-6 only), Feedwater, Main Steam, RPV top head vent and bottom head drains, standby liquid control (SLC), and emergency core cooling systems (e.g., Core Spray, LPCI and ABWR core flooder piping systems).

2.1.2 Operating Conditions

All normal operating conditions were considered in the IMT gap assessment. These include normal power operations and startup / shutdown. Normal operating conditions are defined to include less controlled environmental conditions, such as significant oxygen ingress that may occur during maintenance periods and persist for some time during startup as well as water chemistry transients that have been historically known to occur (e.g., condenser tube leak, resin intrusion events).

All likely operational variations are considered, including power uprate conditions and flexible power operations.

The gap assessment is based on consideration of underlying engineering parameters, such as the impact of neutron fluence accumulation on material toughness or SCC crack growth rates (CGRs), that may potentially represent limiting factors with regard to materials degradation and aging management. As such, the assessment is not directly tied to any specific set of assumptions related to operating conditions, operating periods, or operational approaches.

2.1.3 Materials of Construction

Although BWR designs apply similar materials, it is acknowledged that a variety of alloy grades have been used in service. For example, stainless steel grades include normal and low carbon grades (e.g., 304 vs. 304L), low and high Molybdenum content grades (e.g., 304L vs. 316L), and stabilized grades (e.g., 316NG, 347). These differences can significantly affect material performance. The gap evaluations consider all known alloys & variants used in plants operated by BWRVIP members. For U.S. plants, the BWRVIP has a comprehensive understanding of the materials used. For international plants, the state of knowledge is not comprehensive, but the most commonly used materials have been considered.

2.2 Evaluation Approach

The approach used to update R&D gap assessment results and BWRVIP program requirements in the BWR IMT includes review of the existing set of R&D gaps and BWRVIP program requirements contained in the prior revision of the BWR IMT and an assessment to identify any new R&D gaps or program requirements that should be added to the BWR IMT:

1. Existing R&D gaps and program requirements are reviewed to determine if the gap or program requirement should be closed or, for R&D gaps, if the priority of the gap should be changed. If the gap is closed, a basis for closure is documented. If the gap is not closed, the gap content is reviewed and updated as appropriate. R&D gaps may be closed as a result of completed R&D, obsolescence of the issue, or other reasons. In some cases, the IMT gap may be closed and any remaining regulatory issues associated with the gap tracked in the regulatory issues matrix.
2. The assessment to identify new R&D gaps and program requirements includes review of updated MDM results (if applicable), new industry operating experience, and R&D results. In cases where new R&D gaps are added, the gaps are categorized and prioritized consistent with Section 2.3 below. The process applied to identify new gaps and requirements may identify issues that are more appropriately addressed outside the BWR IMT. Figure 2-1 generally illustrates the new issues assessment steps and potential disposition of new issues.

These evaluations are performed by EPRI BWRVIP staff with input from EPRI and industry subject matter experts and BWRVIP issue program leadership. Revisions to IMT R&D gaps are approved by the BWRVIP Research Integration Committee (RIC).

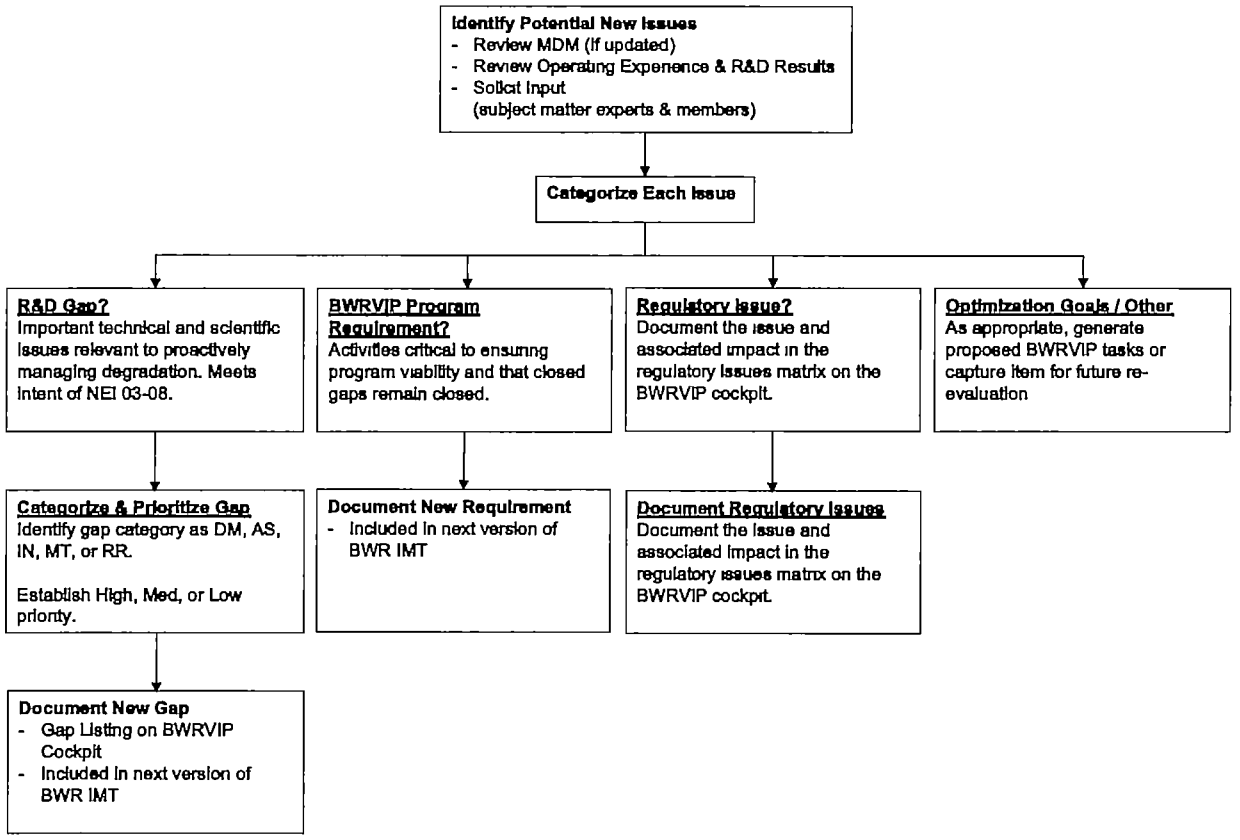

Figure 2-1
Overview of New Issue Identification and Evaluation Process

2.3 R&D Gaps

2.3.1 R&D Gap Definition

An R&D "Gap" is defined an area of research identified as important to achieving a reasonable standard of confidence that primary system component degradation can be managed such that components will remain serviceable and capable of performing their intended functions for the remainder of plant life. R&D gaps address issues that could represent direct challenges to nuclear safety, that have the potential to result in events that challenge the operability of safety-related systems, or that represent challenges to long-term asset management or enterprise risks. R&D gaps may encompass more than one technical area and solution paths should NOT be predefined. For example, a gap could guide R&D from multiple disciplines including structural and metallurgical engineering (analyses intended to quantify available structural margins), NDE (development of methods to better detect and characterize degradation).

R&D gaps are identified by EPRI BWRVIP staff and confirmed by the consensus opinion of the BWRVIP Research Integration Committee.

2.3.2 R&D Gap Categories

R&D gaps are categorized into the following groups based on the technical focus of the gap:

Degradation Mechanism Understanding (DM) Gaps: Degradation Mechanism understanding gaps are identified using data obtained from the EPRI Materials Degradation Matrix (MDM) [2]. These degradation mechanisms are shown as "?" and highlighted in blue in the MDM. An example of a degradation mechanism understanding gap is the development of an understanding of the applicability of low temperature crack propagation for BWR operating conditions.

Assessment (AS) Gaps: Assessment gaps are associated with R&D needs related to characterizing the potential impact of a degradation mechanism shown to be applicable to the BWR operating environment. Additionally, assessment may be needed to determine the proper approach for management of a degradation mode or to develop additional data to better characterize and manage a known degradation issue.

Mitigation (MT) Gaps: Mitigation gaps are associated with R&D needs in the area of new technology development or verification of technique effectiveness for preventing degradation mechanism initiation or limiting degradation mechanism progression. Gaps may involve mitigation of degradation via either water chemistry or mechanical means (e.g., surface stress modification).

Inspection (IN) Gaps (formerly I&E Gaps[3]): Inspection gaps are associated with component inspection guidance, NDE qualification, or development of new NDE technology to effectively detect and size indications in degraded components.

Repair/Replacement (RR) Gaps: Repair/Replacement gaps are associated with needs for further development or verification of the effectiveness of repair techniques.

2.3.3 R&D Gap Identification Numbers

Each gap ID number includes three aspects:

Design Type: BWR IMT gaps are prefixed with "B-" to associate them with the BWR IMTs and prevent confusion with PWR IMT gaps in documentation linking IMT gaps to ongoing research projects, planned research projects, and research proposals. This may occur for EPRI issue programs with responsibilities bridging both BWR and PWR designs, such as the EPRI NDE Center or Water Chemistry Program.

Gap Category: Each gap includes a category identifier consistent with the R&D gap categories presented in Section 2.2.2.

Sequential Numbering: Within each gap category, gaps are sequentially numbered. Note that these sequential numbers are never reused, even if the gap is closed. This approach reduces confusion when referring to gaps by ID number only. Figure 2-2 illustrates the elements of the R&D Gap identification number.

[3] Inspection (IN) gaps used I&E as the gap category identifier in prior revisions of the BWR IMT.

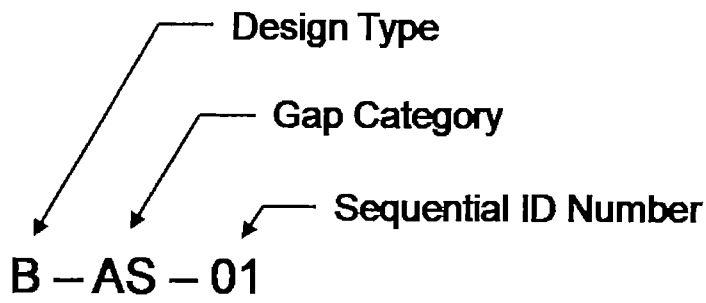

Figure 2-2
Gap ID Nomenclature

2.3.4 R&D Gap Priority and Status

Section 3 of this report contains the current R&D gaps prioritized into High, Medium, and Low priority categories. These categories are proposed by EPRI BWRVIP staff and confirmed by approval through the BWRVIP Research Integration Committee. R&D gaps that have been resolved/closed since the last revision of this report are described in Appendix B. Those R&D gaps that are new in this revision are labeled in Section 3.

2.4 Program Requirements

Program requirements relate to activities that are necessary to ensure that materials issues continue to be adequately managed in the future and to ensure that previously closed R&D gaps remain closed. Examples of program requirements items include maintenance of the BWR Integrated Surveillance Program (ISP) and maintenance of the BWRVIP NDE program. Requirements are not generally statused as open vs. closed, but rather as active vs. inactive. Active denotes a program requirement for which resources are currently needed to address the requirement. Inactive denotes program requirements for which resources are not currently needed.

Program requirements listed in Section 4 include only those associated with the BWRVIP, the primary issue program supporting the BWR IMTs. Program requirements are confirmed by the consensus opinion of the BWRVIP Research Integration Committee but are not prioritized.

Similar to the way R&D gaps are identified, each program requirement is assigned an ID number:

1. Issue Program: BWRVIP program requirements gaps are prefixed with "B-".
2. Requirement Identifier: Each program requirement includes the "RQ" identifier to differentiate the number from gaps and to enable fast searching for all references to specific program requirements.
3. Sequential Numbering.

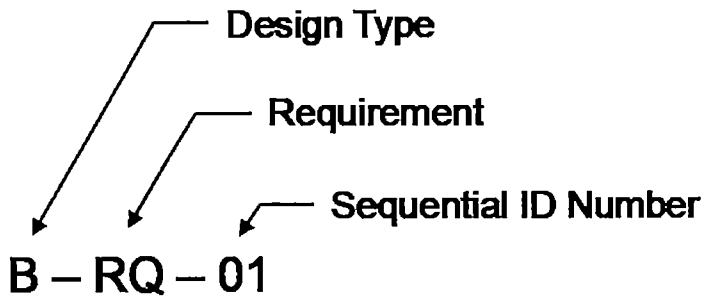

Figure 2-3
Program Requirement ID Nomenclature

2.5 Regulatory Issues

Based on the R&D gap definition planned for Revision 4 (see Section 2.1 above), all R&D gaps should have merit from a fundamental research perspective. That is, work to close the gap should be pursued as resources are available regardless of regulatory factors. Using this approach, all R&D gaps are "technical" in nature and it should be possible to define these gaps using only technical rationales and without reference to regulatory authorities, regulations, or regulatory policies. However, it is recognized that resources are often expended on technical work focused solely on addressing regulatory issues. This is work that the BWRVIP would otherwise not undertake but does so because of the significant benefit to members to have comprehensive, technically sound bases to address regulatory concerns.

Since these issues do not constitute gaps consistent with the objective of proactively addressing materials degradation issues, they do not belong within the IMT gap assessment. However, it is reasonable to track these issues since regulatory issues can be of critical importance for some utilities and there are a substantial number of these issues being tracked. BWRVIP members can access a matrix of regulatory issues in the Library folder on the BWRVIP cockpit.[4]

2.6 Program Optimization Goals/Other Issues

Program optimization is critical to facilitate continued improvements in program efficiency and program implementation. However, since program implementation is known to be highly dependent on regulatory environment, it is difficult to define optimization goals as gaps that are broadly applicable across plants in multiple regulatory environments. As such, program optimization is addressed outside of the IMT gap assessment process through normal R&D prioritization activities or through the regulatory issue matrix where issues are dependent on addressing regulatory requirements or regulator concerns. In addition, it is observed that allocation of R&D resources to address optimization receives broad support and need not be a focus of proactive aging management.

[4] www.membercenter.epri.com > Program Cockpits > P41.01.03 > Library > Issue Management Tables (IMTs)

2.7 Component Tables

Through Revision 3, the IMTs have contained a set of tables that list all of the relevant BWR components within the scope of the IMT evaluation. However, these tables are not essential to performance of a comprehensive assessment of R&D gaps nor to meeting the intent of NEI 03-08. As a result, these tables are no longer included in the published BWR IMT report. However, BWRVIP members can access a streamlined version of these tables in the Library folder on the BWRVIP cockpit.[5]

[5] www.membercenter.epri.com > Program Cockpits > P41.01.03 > Library > Issue Management Tables (IMTs)

3
R&D GAPS

The gap assessment results are provided in Tables 3-1 through 3-6. Table 3-1 provides a summary listing of open R&D gaps. Tables 3-2 through 3-6 provide a description of each gap and full priority history. These tables are organized by R&D gap category as indicated below. These tables include only R&D gaps that are open as of the current IMT revision. For information on gaps closed in this IMT revision see Appendix B, Gap Closure. This appendix includes a listing of gaps closed in Revision 4, the full text of each closed gap, and a basis for gap closure.[6]

Table 3-1: Summary Listing of R&D Gaps and Gap Priorities

Table 3-2: Degradation Mechanism Understanding (DM) R&D Gaps

Table 3-3: Assessment (AS) R&D Gaps

Table 3-4: Mitigation (MT) R&D Gaps

Table 3-5: Inspection (IN) R&D Gaps

Table 3-6: Repair/Replacement (RR) R&D Gaps

Notes regarding gap priority histories included in Tables 3-2 through 3-6:

The priority column includes the current gap priority (top of column). Beneath the current priority, a listing of gap priorities from prior revisions of the BWR IMT is provided. This information is useful to visualize the change in priority occurring over time for each gap. The revision history includes several "interim" revisions to gaps. These are cases where new gaps were added, gaps were closed, or changes were made to gap priority between formal BWR IMT revisions. Revision history noted in the priority column includes:

- **Revision 0 (2007):** Initial version of BWR IMT, BWRVIP-167 [3].
- **Revision 1 (2008):** Formal published revision, BWRVIP-167NP, Rev. 1 [4].
- **Revision 1.1 (2009):** Interim revision, BWRVIP letter 2009-216 [5].
- **Revision 2 (2010):** Formal published revision, BWRVIP-167NP, Rev. 2 [6].
- **Revision 3 (2013):** Formal published revision, BWRVIP-167NP, Rev. 3 [7].

[6] For gaps closed in earlier IMT revisions, R&D gap content and closure basis details can be obtained from prior revisions of the BWR IMTs. In addition, a complete listing of closed R&D gaps is maintained on the BWRVIP cockpit: www.membercenter epri,com > Program Cockpits > P41.01.03 > Library > Issue Management Tables (IMTs).

- **Revision 3.1 (2013, 2014, or 2015):** Interim changes made in BWRVIP research integration committee (RIC) meetings in Dec 2013, Dec. 2014, and Dec. 2015. These changes were documented in the BWRVIP strategic plans issued in 2014 and following years. When Revision 3.1 is relevant to gap history priority change, the year of the revision may be listed as 2013, 2014, or 2015, depending on the year when the change was approved by the BWRVIP RIC.
- **Revision 4 (2020):** Formal published revision/current revision, BWRVIP-167, Rev. 4.

For additional information, see Appendix A, Revision Log. This appendix provides a full listing of BWR IMT changes by revision as well as a comprehensive gap status table that tracks changes to gap status and priority by revision.

Table 3-1
Summary Listing of R&D Gaps and Gap Priorities

Gap ID	Gap Description	Priority
Degradation Mechanism Understanding (DM) R&D Gaps (Table 3-2)		
B-DM-03	Low Temperature Crack Propagation	Low
B-DM-06	Environmental Effects on Fracture Resistance	Low
B-DM-08	SCC Initiation / Propagation in Low-Alloy Steel Materials	Low
B-DM-09	SCC Initiation in Austenitic Materials	Low
B-DM-11	SCC Modeling / Prediction Capabilities/Asset Management Tools for Highly Irradiated Components **[NEW]**	Med
Assessment (AS) R&D Gaps (Table 3-3)		
B-AS-07	Environmental Effects on Fatigue Resistance: Pressure Boundary Components	Med
B-AS-09	Assess the Impact of High Fluence on Stainless Steel Fracture Toughness	High
B-AS-10	Stainless Steel SCC Crack Growth	High
B-AS-15	FIV and High Cycle Fatigue Assessment: Reactor Internals	Low
B-AS-18	Jet Pump Degradation Management	Med
B-AS-20	Assess Non-Safety Locations	Low
B-AS-26	High Strength Alloy SCC	High
B-AS-27	Alloy 82 / 182 SCC Crack Growth	High
B-AS-31	BWRVIP-47-A (CRGT) Re-Inspection Requirements	Low
B-AS-37	Atypical Core Shroud Cracking	Med
B-AS-38	Reactor Internals Leakage Assessment **[NEW]**	Med

Table 3-1 (continued)
Summary Listing of R&D Gaps and Gap Priorities

Gap ID	Gap Description	Priority
Mitigation (MT) R&D Gaps (Table 3-4)		
B-MT-01	Upper Vessel Internal Region SCC Mitigation	Low
B-MT-02	ECP Measurement, Estimation, and Validation	Med
B-MT-04a	On-Line NMCA Effectiveness	High
B-MT-04b	On-Line NMCA Implementation	High
B-MT-05a	Water Chemistry Optimization for Startup and Shutdown	Med
B-MT-05b	Water Chemistry Transient Management	Med
Inspection (IN) R&D Gaps (Table 3-5)		
B-IN-01	Inspection of Core Plate Rim Hold Down Bolts	Low
B-IN-02	Inspection of Hidden Weld Locations	Low
B-IN-03a	Inspection of Core Shroud Weld Locations	Med
B-IN-03b	Inspection of Shroud Support Weld Locations	Med
B-IN-11	UT of BWR/6 Top Guide Grid Beams	Low
B-IN-12	Inspection of Access Hole Cover Welds	Med
Repair/Replacement (RR) R&D Gaps (Table 3-6)		
B-RR-02a	Analytical Tools for Weld Repair of Irradiated Components	Med
B-RR-02b	Irradiated Materials Weldability Data	High
B-RR-05	Alternate High Strength Materials	Low
B-RR-08	Availability of Advanced Welding Processes for Weld Repair of Highly Irradiated Components	Med

Table 3-2
Degradation Mechanism Understanding (DM) R&D Gaps

R&D Gap Description	Priority
B-DM-03 - Low Temperature Crack Propagation **Issue:** Low Temperature Crack Propagation (LTCP) is a hydrogen embrittlement phenomenon that is observed for some primary system alloy when tested at low temperatures. Although there has been no direct evidence of LTCP in the field, this issue remains a possible concern based on laboratory testing. **Description:** Although LTCP has now been validated through laboratory studies, it is not yet clear if actual service conditions are sufficient to induce this degradation mechanism. Primary pressure stresses are significantly reduced at the low temperatures where LTCP is observed and only secondary, deflection-controlled stresses remain. As a result, there remains insufficient data to disposition LTCP as insignificant operational concern for all BWR primary system materials and operating conditions. Higher strength alloys (i.e., X-750 and XM-19), heavily cold worked or highly irradiated stainless steels, irradiated low-alloy steels, and CASS components are at increased risk for LTCP. Resolution of this gap involves further characterization of the parameters influencing LTCP and, if needed, defining "safe" operating regions and operational guidance for eliminating or minimizing LTCP risk. **NOTE:** This gap is focused only on specific LTCP concerns. Broader concern regarding reduced fracture resistance in materials exposed to reactor coolant across the range of normal operating temperatures is described in gap B-DM-06. **References:** MDM-R4 [2], MRP-108 [8], MRP-209 [9], MRP-247 [10], MRP-293 [11], EPRI 1020957 [12]	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) Rev. 2: Low (2010) Rev. 1.1: Low (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)

Table 3-2 (continued)
Degradation Mechanism Understanding (DM) R&D Gaps

R&D Gap Description	Priority
B-DM-06 - Environmental Effects on Fracture Resistance **Issue:** Recent testing indicates that primary systems materials can have lower fracture resistance (J-R tearing resistance) when tested in coolant than when tested in air. Although there is an increasing body of data that can be used to quantify the effect, the factors influencing this phenomenon and the potential operational significance of this effect for BWR primary systems materials remain incompletely understood. **Description:** Although there is consensus that the observed effect is a hydrogen-induced phenomenon, there are insufficient data to quantitatively predict the effects for all of the relevant materials and service conditions. Hydrogen fugacity in the environment, the diffusivity of hydrogen within the metal, and the interaction of hydrogen with other parameters (e.g., temperature and mechanical loading rate) all remain poorly understood. Since any effect of hydrogen will increase with increasing yield strength, a more significant effect likely exists for irradiated materials and higher strength materials (e.g., cold-worked or precipitation hardened stainless steels). Synergisms with SCC, and corrosion fatigue must also be considered. Closure of this gap involves further characterization of this environmental fracture issue to fully disposition this effect as non-relevant or to fully evaluate any consequences on reactor operation. **NOTE:** With regard to observations of rapid fracture occurring under rising K conditions, similar testing in air concluded that plastic instability resulting from mechanical overload conditions was the primary factor in the observations. No evidence of an effect of environment was observed. This new data eliminates "rapid fracture" as a significant concern and reduces the overall level of concern associated with this gap. **References** MDM-R4 [2], MRP-209 [9], MRP-293 [11], MRP-431 [13], EPRI 1020957 [12]	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Med (2013–2015) Rev. 3: Med (2013) Rev. 2: High (2010) *Rev. 1.1: High (2009)* Rev. 1: Med (2008) *[New in Rev. 1]*

Table 3-2 (continued)
Degradation Mechanism Understanding (DM) R&D Gaps

R&D Gap Description	Priority
B-DM-08 – SCC Initiation / Propagation in Low-Alloy Steel Materials **Issue:** The envelope of material and environmental conditions that could result in either SCC initiation or SCC crack propagation in BWR low-alloy steel (LAS) components is not defined. **Description:** Although laboratory studies show SCC is possible in LAS materials and SCC CGR correlations have been developed for LAS materials in BWR service, there has been no field evidence of SCC occurring in any BWR RPV LAS component to date, even when stress intensity associated with cracks in adjacent Alloy 182 materials would suggest significant SCC propagation into the LAS material. Given these two apparently contradictory pieces of information, evaluations of SCC identified in Alloy 182 as well as generic assessments of RPV integrity have often taken the conservative approach of postulating SCC of LAS material. At times, such postulated cracking has a significant impact on the evaluation results. Factors that must be considered are the impact of water chemistry, methods used to predict crack tip stress intensity, and the possibility of increased propensity for crack occurrence and crack growth rates occurring as the yield strength of the pressure vessel material increases with fluence. Closure of this gap would involve developing a reasonable basis for predicting where the potential for SCC in LAS RPV components should be considered in evaluations, either in plant-specific evaluations of cracks identified by NDE or in generic RPV integrity evaluations involving postulated flaws. **References:** BWRVIP-233R2 [14], BWRVIP-306 [15], MDM-R4 [2] (Note b1-7a,b)	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) *Rev. 2: Low (2010)* *[New in Rev. 2]*

Table 3-2 (continued)
Degradation Mechanism Understanding (DM) R&D Gaps

R&D Gap Description	Priority
B-DM-09 – SCC Initiation In Austenitic Materials **Issue:** The envelope of operating conditions that potentially lead to additional incidents of SCC initiation in austenitic materials is not well defined. As a result, decisions regarding aging management often rely on assumption of upper end CGRs, even though such evaluations are often highly conservative. **Description:** Aging management approaches for reactor internals often rely on lab-based CGR data as a basis for establishing reinspection intervals. However, field data suggest significant resistance to SCC initiation. Further, much of the SCC documented in BWR OE is associated with early life events and there are no data suggesting an increasing trend of cracking due to SCC associated with longer service times. An understanding of the conditional probability of SCC initiation (or re-initiation, i.e., restarting crack growth) associated with various combinations of materials, fabrication factors, and environmental conditions could support development of improved probabilistic aging management assessments that consider initiation. Such an understanding would also support greater operational flexibility with regard to responding to chemistry transients. In addition to the effect of transient conditions, the potential for initiation due to long-term exposure must also be considered (i.e., that over time initiation becomes more probable for a given combination of material, environment, and fabrication factors). For stainless steels, there is concern that longer times for oxide formation could lead to an increased potential for cracking of less severely cold worked layers (i.e., cold worked layers that were previously not sufficiently damaged so as to result in susceptibility to SCC initiation). For nickel-base alloys, although surface cold work has not specifically been seen to promote SCC initiation in nickel-base alloys to date, there is concern that it could become a factor at long exposure times. Additionally, there is concern that long-term exposure to BWR water environments could promote SCC initiation at progressively lower levels of stress intensity, leading to cracking in locations not previously experiencing SCC. Finally, the transition of a majority of the BWR fleet worldwide to noble metal catalyzed HWC places surface ECP in a range closer to that of PWRs, where significant SCC has occurred after long service times. As a result, there is also some concern that a similar trend of SCC in nickel-base alloys could potentially occur late in service life. Resolution of this gap would involve significant advances in modeling and predictive capabilities regarding critical factors influencing SCC initiation in austenitic materials under various environmental conditions and service times. **NOTE:** Also see gap B-MT-05b which addresses the impact of water chemistry transients on SCC. **References:** MDM-R4 [2] (Note b1-6c, b1-6f)	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Low (2013–2015) Rev. 3: Low (2013) *Rev. 2: Low (2010)* *[New in Rev. 2]*

Table 3-2 (continued)
Degradation Mechanism Understanding (DM) R&D Gaps

R&D Gap Description	Priority
B-DM-11 – SCC Modeling / Prediction Capabilities/Asset Management Tools for Highly Irradiated Components **Issue:** An increase in SCC propensity is an important concern related to the long-term viability of BWR near core welded structures (i.e., core shrouds and top guides). The current state of knowledge remains limited with regard to understanding the effect of neutron fluence on SCC occurrence and propagation in near core components. **Description:** Although it is clear from laboratory testing that the material property changes resulting from neutron fluence increase CGRs, at least in the case of core shrouds, there remain too few data to separate the effect of neutron fluence from other factors, particularly fabrication-related factors. It is possible that the true effect of fluence on CGR so far has been minimal. However, these data are associated with plants using HWC. It is possible that under NWC, a more significant effect could be observed. A broader concern is the potential for new crack initiations or re-initiation of cracking in dormant flaws to occur in high fluence regions. The capability to predict future degradation with any degree of confidence or to confidently distinguish between higher and lower risk plants remains very limited and is primarily based on anecdotal observations. At present, there is no evidence of such an adverse trend. However, it is acknowledged that prediction capabilities remain limited. Development of improved statistical bases for CGRs associated with NWC conditions and predictive models for identification of parameters associated with increased risk of cracking as a result of neutron fluence would be valuable for the purpose of improving aging management guidance and confidence in long-term asset management plans. **References** MDM-R4 [2] (b-2-7a,b), BWRVIP-302 [16]	**Rev. 4 Priority:** **Medium** **Priority History:** *[New in Rev. 4]*

Table 3-3
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-07 – Environmental Effects on Fatigue Resistance: Pressure Boundary Components **Issue:** Accurately accounting for the effects of environment on fatigue usage associated with primary system pressure boundary components remains a significant knowledge gap. **Description:** Laboratory tests show that environmental effects are real; i.e., that fatigue resistance in reactor water environments is lower than in an air environment. This conclusion is supported by a database of laboratory tests on small specimens in simulated reactor coolant environments. It is also recognized that the fatigue design curves that provide the basis for fatigue life assessments of nuclear plant components are based on testing of small specimens in air that do not explicitly account for the effects of LWR environments on fatigue life. Over time, fatigue usage estimates have been refined through the application of more complex analytical methods to accommodate longer term operation, oftentimes at a significant cost burden to plant owners. This approach, while a pragmatic solution for meeting regulatory requirements, does not directly address the impact of water environments. A discrepancy remains between high calculated usage factors and the absence of fatigue cracking caused by EAF in actual components. A fundamental knowledge gap exists regarding the amount of conservatism in the current practice of using correlations based on testing of small-scale laboratory specimens and environmental sensitivity factors to estimate the fatigue lives of plant components that have complex geometries and are subject to complex pressure/temperature transients and multi-axial stresses. In addition, there are some areas where additional data or evaluations are needed to generate appropriate correlations of the influence of dissolved O_2 / electrochemical potential (ECP) and material chemistry on environmental multipliers and fatigue CGRs. Although management based on flaw tolerance represents a viable alternative when it is not possible to demonstrate that the accumulated fatigue life will not exceed the allowed value prior to the end of plant life, such assessments require periodic inspection of the relevant locations and availability of fatigue crack growth rate correlations appropriate to the material and environment being evaluated. Final resolution of this gap involves development of a fundamental understanding of environmental effects on fatigue life or, as a minimum, development of management techniques that allow operational flexibility while also ensuring adequate management of fatigue cracking. **NOTE:** Additional discussion regarding R&D needs for different primary system pressure boundary materials is contained in the MDM. **References:** EPRI 1026724 [17], EPRI 3002010554 [18], EPRI 3002014122 [19], EPRI 3002007973 [20], MRP-415 [21], MDM-R4 [2] (Note b1-9a)	**Rev. 4 Priority:** **Medium** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: Medium (2009) Rev. 1: Medium (2008) Rev. 0: Medium (2007)

Table 3-3 (continued)
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-09 – Assess the Impact of High Fluence on Stainless Steel Fracture Toughness **Issue:** There is a need for additional data to fully characterize the effect of high neutron fluence on the fracture toughness properties of austenitic stainless steel materials. **Description:** The serviceability of BWR core structures and components is verified periodically through a prescribed program of inspection and evaluation. Evaluations of current serviceability and projections of future serviceability rely on accurate fracture toughness data. Results of these evaluations support run/ repair decisions. Resolution of this gap includes development of comprehensive guidance for evaluating fracture toughness through development of sufficient irradiated materials data to address the fluence ranges of interest. While the data sets for austenitic stainless steel base metal are considered to be generally sufficient, the high fluence response of weld and HAZ materials, including assessment of high fluence saturation values, is not well established. Additionally, there are some data showing substantially lower than anticipated fracture toughness for which some further investigation may be warranted. As a result, there is incentive to fill this gap with supplemental data. When the potential for extended operations is considered, there could be a need to develop methods for further refinement of the fracture toughness guidance in BWRVIP-100 so that actual component condition is more accurately characterized. This need could become important for core shroud welds having both significant cracking and significant accumulated neutron fluence. Closure of this gap involves generation of data to resolve gaps in current datasets, further investigation into data that indicate lower than anticipated toughness for irradiated weld metals, and guidance refinements where needed to address extended operations. **References** BWRVIP-100R1-A [22], BWRVIP-140 [23], BWRVIP-154R2 [24], BWRVIP-294R2 [25] S. Fyfitch, et al, "Fracture Toughness of Irradiated Stainless Steel in Nuclear Power Systems," 14th International Conference on Environmental Degradation of Materials in Nuclear Power System, Virginia Beach, VA, August 23-27, 2009. [26]	**Rev. 4 Priority:** **High** **Priority History:** Rev. 3.1: High (2013-2015) Rev 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: High (2007)

Table 3-3 (continued)
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-10 – Stainless Steel SCC Crack Growth **Issue:** Although SCC crack growth in austenitic stainless steels in BWR environments has been studied extensively, there are remaining areas warranting further attention. **Description:** SCC crack growth correlations are a key input to flaw tolerance calculations that form the basis for inspection intervals for flawed components. Further, some utilities use a flaw tolerance-based approach to assess inspection intervals for components not known to contain flaws. As a result, it is important to have a comprehensive understanding of crack growth rates (CGRs). There are several remaining open issues related to crack growth assessment: Assessment of Field Data / Comparison with Laboratory Test Data & Models Evaluation of core shroud repeat inspection data continues to be used to assess stainless steel CGRs. However, there are a number of factors potentially influencing CGRs that are not easily separated, including NDE biases and measurement uncertainties, changes in stress intensity factor that occur as crack depth increases, and neutron fluence. Changes in the growth rate of cracks in thin-walled components over time is also not fully understood. Effect of Material Grade Most austenitic stainless steel components in U.S. plants are fabricated from Types 304 and 304L. However, worldwide other material grades have been used (e.g., 316LSS in Japan, Ti-stabilized grades in Sweden). It is not clear that potential differences in performance are fully understood. At a minimum, there is a need for additional field data associated with components fabricated from non-304/304L grades. High Purity NWC CGRs For locations not protected by HWC technologies (e.g., core spray piping) and for all stainless steel internals in plants that do not use HWC technologies, NWC the impact of operation under well controlled water chemistry conditions (i.e., essentially pure water with very low conductivity and concentrations of anionic impurities) on crack growth rate has not been evaluated in a systematic manner. **NOTE:** Consideration of the impact of water chemistry transients is addressed in mitigation gaps B-MT-05a and B-MT-05b. **References:** BWRVIP-14-A [27], BWRVIP-80-A [28], BWRVIP-99-A [29], BWRVIP-174R2 [30], BWRVIP-221 [31], BWRVIP-232 [32], BWRVIP-253 [33], BWRVIP-265 [34], EPRI 1019028 [35], EPRI 1019029 [36], EPRI 3002003103 [37], ASME CC N-889 [38]	**Rev. 4 Priority:** **High** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: High (2007)

Table 3-3 (continued)
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-15 – FIV and High Cycle Fatigue Assessment: Reactor Internals **Issue:** High-cycle fatigue, FIV, and associated wear have been an operational concern for a number of reactor internals locations. This gap addresses the need to comprehensively assess the reactor internals for potential high-cycle fatigue, FIV, and associated wear concerns. **Description:** Major high-cycle fatigue issues occurring in BWR reactor internals to date include jet pump and steam dryer. This gap addresses the potential for FIV and associated wear to occur in other locations, potentially resulting in unanticipated impacts. Wear is also believed to be occurring on the feedwater sparger end pins. Some shroud head bolts have experienced wear that is potentially associated with high-cycle vibration. There may be additional locations where FIV and wear eventually occur due to power uprates and additional service time. A full review of the internals for potential vibration related issues is potentially needed to resolve flow induced high-cycle fatigue concerns. Resolution of this gap includes development of an adequate understanding of the factors affecting susceptibility to FIV damage so that all potential risks related to anticipated operating conditions are identified and characterized. **References** BWRVIP-322 [39]	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: Medium (2008) Rev. 0: Medium (2007)

Table 3-3 (continued)
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-18 – Jet Pump Degradation Management **Issue:** Many BWRs experience jet pump performance and reliability issues resulting from flow instability and FIV. In a few instances, the resulting jet pump degradation has been significant. Although the state of knowledge has been significantly improved in recent years, there is still a substantial gap in understanding how jet pump installation conditions as well as plant operating regimes and associated flow conditions influence jet pump degradation. **Description:** Jet pump degradation has been observed in operating plants since early in plant operation. Not all plants have experienced degradation and when a plant has experienced degradation, not all jet pumps within a given reactor pressure vessel experience the same degradation. Degradation resulting from FIV includes: • Jet pump sensing line (JPSL) standoff cracking • Riser and riser brace cracking • Restrainer bracket set screw tack weld cracking and wear • Restrainer bracket wedge and slip joint wear These forms of degradation are believed to be a consequence of various jet pump flow induced vibration (FIV) mechanisms, including turbulence induced vibration, recirculation pump vane passing frequency induced resonance, and slip joint instability. In recent years, a number of units have installed hardware designed to prevent or mitigate the damage resulting from jet pump FIV, including labyrinth seals and auxiliary restraints (e.g., auxiliary wedges, slip joint clamps, riser brace clamps, JPSL clamps). Operating experience has shown that due to gaps in understanding the casual factors, these hardware installations are not always successful in mitigating degradation, and in some cases, repairs are removed and replaced with other repairs. The BWRVIP has completed an R&D program to gain fundamental knowledge pertaining to the slip joint instability. There remains a need to identify and understand the parameters that control the observed variability in degradation observed within jet pumps installed in a unit and between units. Identification and improved understanding of these parameters would aid utility decision-making regarding installation of stabilizing hardware, the need for augmented inspections, and appropriate restrictions on operating conditions. A detailed understanding of the parameters that would lead to degradation consequent to a change in operation is necessary for optimal jet pump management and elimination of unnecessary modifications and inspections. Resolution of this gap includes development of an adequate understanding of the factors affecting susceptibility to FIV damage and development of proven tools to prevent future FIV and subsequent component damage. **References:** BWRVIP-41R4-A [40], BWRVIP-252 [41], BWRVIP-263 [42], BWRVIP-266 [43], BWRVIP-322 [39]	**Rev. 4 Priority:** **Medium** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: High (2007)

Table 3-3 (continued)
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-20 – Assess Non-Safety Locations **Issue:** A qualitative evaluation is needed to assess the potential consequences of non-safety related component failures. **Description:** BWRVIP-06-A was initially developed to determine the short-term and long- term actions appropriate to assure continuing safe operation of the RPV. The evaluation focused primarily on systems critical to safety. However, significant degradation of non-safety related internals has occurred since completion of BWRVIP-06-A. Failure of non-safety RPV internal components can potentially cause generation of loose parts that can adversely impact safety related component function or cause forced outages that result in significant loss of production. Results from an assessment would include recommendations for managing degradation of non-safety related components. Guidance may include monitoring, augmented inspections, analyses, etc. Future tasks, such as development of component-specific I&E Guidelines, could be defined pending results of this work. **References:** BWRVIP-06R1-A [44]	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) Rev. 2: Low (2010) Rev. 1.1: Medium (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)
B-AS-26 – High Strength Alloy SCC **Issue:** There are remaining gaps in the material performance databases for high strength materials being used in BWR reactor internals and reactor internals hardware. Additional data are needed to fully characterize SCC susceptibility and neutron effects for these materials. **Description:** Testing completed in recent years has added new information regarding CGRs associated with X-750. Since reinspection requirements for jet pump holddown beams are based on flaw tolerance evaluations, new data associated with SCC CGRs should be considered to determine if there are impacts to reinspection intervals. In addition, the degradation in material properties due to neutron fluence has not been sufficiently quantified. Both X-750 and XM-19 are applied as core shroud repair hardware in the annulus between the core shroud and reactor vessel and are potentially subject to significant neutron fluence, especially when considering extended operations. Generation of additional data to characterize irradiated material mechanical property data, SCC growth rates, and stresses required to initiate cracks in these alloys is necessary to better understand the long-term performance of currently installed materials. Such information can also be extremely valuable to hardware designers to reduce the potential for inservice failures. **References:** MDM-R4 [2] (Notes b2-1h, b2-1i), BWRVIP-218 [45], BWRVIP-240 [46], BWRVIP-291R1 [47], BWRVIP-262NP [48]	**Rev. 4 Priority:** **High** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) *[New in Rev. 1]*

Table 3-3 (continued)
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-27 – Alloy 82 / 182 SCC Crack Growth **Issue:** Although SCC crack growth in Alloys 600, 82, and 182 in BWR environments has been studied extensively, there are remaining areas warranting further attention. **Description:** Although most SCC associated with nickel-base alloys occurs in Alloy 182 or in creviced Alloy 600, there are cases where SCC has been detected in Alloy 82 weld material, primarily associated with highly restrained and creviced geometries. Since management of nickel-base alloys is often supported by CGR evaluations, it is important to have a robust understanding of SCC CGRs. There are several remaining open issues related to crack growth assessment: High K CGR Data For some welds, K value predictions exceed the K values addressed by the BWRVIP-59-A CGR correlations. Alloy 82 CGR Correlations BWRVIP-59-A currently provides a single disposition curve for both Alloy 182 and Alloy 82. This results in the potential for flaw evaluations associated with only Alloy 82 material to be overly conservative by use of a disposition curve fit to bound all Alloys 82 and 182 CGR data. HWC Data There are relatively few data associated with HWC conditions provided in BWRVIP-59-A and essentially no data at high or low K values. High Purity NWC CGRs BWRVIP-59-A includes presentation of high purity NWC CGRs that are generally a factor of 2 lower that the overall NWC curves. However, these curves are not applied by plants in part as a result of regulator concerns. Updates to these curves would be potentially beneficial to plants operating under NWC. **NOTE:** Current fluence estimates show that Alloy 600/82/182 locations will not be subject to end of life fluence that is high enough to cause concerns regarding irradiation impacts on fracture toughness or SCC CGRs. The gap description has been adjusted accordingly. Issues related to probability of initiation as a result of extended operations / changes in SCC susceptibility DM-09. **References** MDM-R4 [2] (Note b1-6g & h, b2-6e & f), BWRVIP-59-A [49]	**Rev. 4 Priority:** **High** **Priority History:** Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: High (2008) *[New in Rev. 1]*

Table 3-3 (continued)
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-31 – BWRVIP-47-A (CRGT) Re-Inspection Requirements **Issue:** There is a need to develop appropriate re-inspection criteria for CRGTs. Only a baseline inspection program is addressed by BWRVIP-47-A. **Description:** Currently, most units have completed baseline exams of CRGT components as specified in BWRVIP-47-A and remaining units will complete their baseline exams in the near future. There are currently no re-inspection requirements. Although results from baseline inspections did not identify significant degradation, additional review and evaluation is needed to make an informed decision regarding the need for additional CRGT inspections. **NOTE:** There is also a regulatory component to this issue that is tracked here for completeness. BWRVIP-47-A includes the following statement: *"Baseline inspection results will be reviewed by the BWRVIP and, if deemed necessary, reinspection recommendations will be developed at a later date and provided to the NRC."* **References** BWRVIP-47-A [50]	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) Rev. 2: Medium (2010) *[New in Rev. 2]*
B-AS-37 – Atypical Core Shroud Cracking **Issue:** There is a need to ensure that BWRVIP-76 provides appropriate guidance to address atypical (off-axis) cracking. **Description:** The existing inspection and evaluation guidance for core shrouds in BWRVIP-76 is based on the premise that cracking will be oriented parallel to the welds and contained in the heat affected zone (HAZ). The guidance does not address atypical cracking. BWRVIP interim guidance contained in letter 2016-030 provides for a one-time inspection to characterize fleet condition with regard to off-axis cracking. Resolution of this gap involves evaluation of the resulting dataset and development of appropriate revisions to BWRVIP-76 that address shroud weld off-axis cracking. **References** BWRVIP-302 [16], BWRVIP-311 [51], BWRVIP letter 2016-030 [52]	**Rev. 4 Priority:** **Med** **Priority History:** Rev. 3.1: High (2013-2015) *[Gap added in 2015 RIC Meeting.]*

Table 3-3 (continued)
Assessment (AS) R&D Gaps

R&D Gap Description	Priority
B-AS-38 – Reactor Internals Leakage Assessment **Issue:** Assessment of leakage from thru-wall cracks and reactor internal repairs has become a more important issue recently with the acknowledgement that even relatively short off-axis cracks can be thru-wall. In the process of addressing additional thru-wall cracks, plant leakage margins may be challenged. **Description:** Maintaining 2/3 core height coverage is a design basis requirement for management of a large break loss of coolant accident in BWR/3-6s and ABWRs. Through-wall cracks in the core shroud, jet pumps, shroud support or access hole covers can impact the capability of maintaining 2/3 core height coverage. Key issues include: Crack Morphology: Currently, BWRVIP guidance for evaluation of through-wall cracks takes no credit for friction losses associated with tortuous path along the crack front. As a result, predicted leakage rates using current guidance are very conservative. By applying the concept of "number of turns", leak rate predictions can be reduced by approximately an order of magnitude. However, high quality optical data are needed to develop appropriate factors addressing numbers of turns for SCC cracks. While there are some data characterizing wrought stainless steels, additional data would be beneficial. Additionally, there are almost no data characterizing SCC cracks in nickel-base alloy and stainless steel weld metals. Crack Opening: Where crack morphology has been used to remove conservatisms from the leak rate estimate, care must be taken not to apply this approach unless rigidity can be demonstrated. If loading causes larger than anticipated crack opening areas, leakage estimates crediting number of turns may be nonconservative. Statistical Evaluations: In cases where relatively high confidence regarding the number and nature of through-wall cracks cannot be demonstrated, there is a need to identify statistical methods to address the uncertainty. Previously, this was not an issue since conservative methods were being applied to address leakage through known cracks. However, when refined methods are used, this approach may no longer be appropriate. Guidance in 2016-030 provides a method, but this method should be revisited as new inspection data are generated. Guidance Consistency. Leakage assessment guidance in BWRVIP I&E guidelines is presented differently in different guidelines. An effort to make this guidance as clear and consistent as possible is deemed appropriate. **References:** BWRVIP-311 [51], BWRVIP-76R2 [53], BWRVIP-41R4-A [40], BWRVIP-180 [54], BWRVIP Letter 2016-030 [52]	Rev. 4 Priority: Medium Priority History: ***[New in Rev. 4]***

Table 3-4
Mitigation (MT) R&D Gaps

R&D Gap Description	Priority
B-MT-01 – Upper Vessel Internal Region SCC Mitigation **Issue:** Noble metal catalyzed hydrogen injection mitigation technologies are not effective in mitigating SCC for components located in the upper vessel internal region. **Description:** An effective solution to mitigate SCC in the upper vessel could be very beneficial with regard to long term aging management. However, techniques studied to date have significant limitations that have prevented adoption. Examples of alternative mitigation approaches previously studied include Zr / Zr alloy protective insulated coatings and TiO_2 application. Zr / Zr alloy protective coatings are applied by thermal spray methods and result in an insulating layer that mitigates SCC initiation. However, the need to apply these coating via thermal spray transfer methods is a significant barrier to implementation. TiO_2 acts as a photo-catalyst to promote a H_2O oxidation reaction and suppress the metal oxidation reaction, effectively lowering the ECP at the component surface. However, this process is limited to locations relatively close to the core because it relies on the UV irradiation provided by Cherenkov radiation from the reactor. As such, its overall effectiveness is unproven. Resolution of this gap involves development of viable methods and subsequent demonstration of the technologies (effectiveness, durability, cost) for specific locations, along with practical implementation guidance. **References** BWRVIP-62-A (2018 Version) [55], BWRVIP-109 [56], BWRVIP-163 [57], BWRVIP-190R1 [58], BWRVIP-195 [59], BWRVIP-201 [60]	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: Medium (2008) Rev. 0: Medium (2007)

Table 3-4 (continued)
Mitigation (MT) R&D Gaps

R&D Gap Description	Priority
B-MT-02 – ECP Measurement, Estimation, and Validation **Issue:** Over the years, ECP has become an integral parameter that plant operators track in order to assure integrity of plant internal components and validate the effectiveness of hydrogen water chemistry-based mitigation technologies. However, there remain technical barriers to obtaining validated ECP data for all locations of interest. There is a need for a comprehensive set of capabilities for electrochemical corrosion potential (ECP) measurement, estimation / modeling, and validation ECP is used as a key metric to assess SCC mitigation by noble metal catalyzed hydrogen injection chemistries. **Description:** Direct measurement of ECP in many cases is not practical due to access limitations. There are two primary challenges. The first is development of reference electrodes that can withstand the high temperature (288°C) and high radiation environment. The second challenge is to position the electrodes in areas of highest ECP interest since most operating BWRs were not designed to accommodate ECP electrodes. Additionally, there are known issues associated with plant mitigation monitoring system (MMS) implementation and application of the resulting ECP data. The two most notable are 1) understanding the effect of sample line length and dissolved oxygen injection on ECP measurements and 2) understanding the cause of high ECP measurements occurring after startup at some units. Resolution of this gap may involve development of improved monitoring equipment as well as improvements in modeling techniques or development of novel methods of accurately estimating ECP based on available data. **References** BWRVIP-268 [61], BWRVIP-296R2 [62], BWRVIP-62-A (2018 Version) [55]	**Rev. 4 Priority:** **Medium** **Priority History:** Rev. 3.1: Medium (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: High (2007)

Table 3-4 (continued)
Mitigation (MT) R&D Gaps

R&D Gap Description	Priority
B-MT-04a – On-Line NMCA Effectiveness **Issue:** Although OLNC is considered mature, there are remaining questions about the parameters critical to comprehensive mitigation and verification that plant conditions resulting from typical OLNC implementation can be represented by results obtained in controlled laboratory studies. **Description:** OLNC has been in use by plants for more than 10 years and there is a growing set of data from core shroud UT examinations that provide a basis for its effectiveness. In addition, the science regarding necessary conditions for effective mitigation (e.g., particle size, interparticle spacing) is also relatively well characterized through laboratory studies. However, there remain questions regarding the lower bound conditions that are sufficient to characterize a region of interest associated with a plant component as mitigated, as well as gaps in knowledge regarding how plant applications may be different than results predicted by laboratory simulations. Relevant research areas include: Flow Conditions and Surface Structure: Flow turbulence and boundary layer conditions potentially affect the deposition of Pt on surfaces. However, the overall impact of flow conditions and surface structure on Pt deposition and related conclusions regarding mitigation effectiveness are not well characterized. Pt Particle Incorporation into Oxide Layers: As Pt particles are incorporated into surface oxide layers, some portion of the particle surface area is no longer in direct contact with coolant and may no longer be effective to catalyze the recombination reaction. The overall magnitude and significance on Pt loading adequacy assessments is not well characterized. Pt Particle Deposition Within Cracks and Tight Crevices: It has been often speculated that mitigation of SCC requires deposition of Pt to some depth within a crack or tight crevice. However, there are few data that can be used to characterize the diffusion of Pt into cracks. There are also few data that can be used to characterize the potential for oxygenated conditions to exist at crack tips or deep within crevices due to a lack of catalyst. In summary, closure of this gap involves resolution of some of the fundamental questions related to assuring adequate mitigation in field applications. These data are valuable for the purpose of specifying appropriate controls and limitations on implementation of OLNC. **References** BWRVIP-62-A (2018 Version) [55], BWRVIP-174R2 [30], BWRVIP-304 [63], BWRVIP-219 [64], BWRVIP-320 [65]	**Rev. 4 Priority:** **High** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) *[Note: Gap focus revised significantly in Rev. 2, priorities in earlier IMT revisions not relevant.]*

Table 3-4 (continued)
Mitigation (MT) R&D Gaps

R&D Gap Description	Priority
B-MT-04b – On-Line NMCA Implementation **Issue:** Although OLNC is considered to be mature, there are remaining technical concerns related to ensuring effective OLNC implementation at plants for the purpose of ensuring mitigated conditions and providing a technical basis for inspection relief. **Description:** Pt Deposition Data from Plant Components: If noble metals applied via the OLNC process preferentially deposit in some locations, there is a possibility that the catalyst loading for other areas could be overestimated. Data available to characterize Pt loading for OLNC plants remain relatively limited. In addition, alternative methods to measure platinum on surfaces e.g. non-destructive analysis of platinum loading and examination of surface oxide replicas at high magnification are not yet mature enough to be applied in the field. Pt Loading Modeling Capabilities: Improved capabilities to model Pt loading on surfaces throughout the vessel, internals and primary system would be beneficial to units as an additional tool to validate Pt loading. Improved Pt Loading Demonstration Capabilities: The current approach to Pt loading demonstration involves laboratory analysis of MMS coupons or plant artifacts. Neither sample type is optimal. Data from MMS coupons are known not to be representative of in-vessel Pt deposition. Artifact removal and analysis involves significant resources and cost. An improved demonstration method that can be performed without removal of plant artifacts or MMS coupons, is non-destructive, and is low cost would be of significant benefit to the industry. Noble Metal Catalyst Deposit Durability: Possible challenges to durability include removal of deposits in areas of highly turbulent flow or as a result of cleaning (typically by hydrolasing) and flushing to support outage activities and in-vessel inspections. In summary, closure of this gap involves development of implementation methods and guidance as well as verification standards that provide reasonable confidence in SCC mitigation without the imposition of burdensome requirements on operators. **References** BWRVIP-62-A (2018 Version) [55], BWRVIP-174R2 [30], BWRVIP-219 [64], BWRVIP-304 [63], BWRVIP-320 [65]	**Rev. 4 Priority:** **High** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) *[Note: Gap focus revised significantly in Rev. 2, priorities in earlier IMT revisions not relevant.]*

Table 3-4 (continued)
Mitigation (MT) R&D Gaps

R&D Gap Description	Priority
B-MT-05a – Water Chemistry Optimization for Startup and Shutdown **Issue:** There is a need to better understand the effect of startups and shutdowns on SCC crack growth and the potential for new SCC initiations as well as a need for development of appropriate operational responses and mitigation solutions. **Description:** For plants operating under HWC, the greatest risk of SCC initiation or crack growth in a typical fuel cycle occurs during startup and shutdown periods where operating temperatures are high enough to promote SCC (> 200 °F), but excess hydrogen is not yet available to establish / maintain low ECP conditions. In particular, dissolved oxygen and hydrogen peroxide concentrations in the reactor coolant will be elevated during startup and early power ascension, but hydrogen injection cannot typically be initiated until significant power levels are achieved. For plants using HWC, initiation of H_2 injection earlier in startup represents one approach for attaining ECP reductions earlier in the startup phase. However, there remains a need to optimize this approach and to develop appropriate guidance in the EPRI Water Chemistry Guidelines. Resolution of this gap includes development and maturation of chemistry control technologies and associated implementation guidance supporting BWR startups and shutdowns that minimize or eliminate the detrimental impact on plant components. **NOTE:** Although transients that result in SCC initiation or growth may occur during startup, Gap B-MT-05b addresses transient management. **References** BWRVIP-190R1 [58], BWRVIP-225R1 [66], BWRVIP-226 [67], BWRVIP-249 [68], BWRVIP-261 [69]	**Rev. 4 Priority:** **Medium** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: High (2007)

Table 3-4 (continued)
Mitigation (MT) R&D Gaps

R&D Gap Description	Priority
B-MT-05b – Water Chemistry Transient Management **Issue:** Water chemistry transients can increase crack growth rates and may potentially also promote new SCC initiations in nickel-base alloys and austenitic stainless steels. An improved capability to characterize the impact of water chemistry transients would be highly beneficial in providing operators with optimal guidance for managing these transients. **Description:** Although the most significant water chemistry transients tend to occur at power and are associated with resin intrusions, condenser or RHR heat exchanger leakage, there are many potential sources of contaminants that can cause a transient. For example, chemistry transients can occur during startup periods following refueling outages as a result of system flow changes and residual chemical impurities from outage-related work activities. Regardless of the source, when water chemistry action levels are exceeded there are a number of interrelated factors that must be evaluated to determine the best course of action. Subsequent to identifying and isolating the transient source, a key question is whether to continue operating and clean up at temperature or to shutdown. Decision making related to materials management is currently based on conservative and simplistic application of CGR data. However, these data are known to be incomplete and to have large scatter and there are often significant differences in crack growth predicted by lab data in comparison with field observations. Data characterizing the effects ionic impurity transients (chloride, fluoride, sulfate and phosphate ions) on SCC of stainless steels and nickel-base alloys at low and high potentials are needed. Characterizations should be developed for both crack growth response and SCC initiation for relevant alloys. Data should also accommodate the needs of plants operating under NWC conditions where the response to a transient likely should be quite different than a plant operating under HWC. **NOTE:** This gap is intended to address significant transients, including transients occurring at power and abnormal / significant transients occurring during startup and shutdown for which decisions regarding operational actions must be made in an expedited manner to prevent the possibility of significant degradation of components fabricated from SCC susceptible materials. **References** BWRVIP-190R1 [58], BWRVIP-225R1 [66], BWRVIP-226 [67], BWRVIP-249 [68], BWRVIP-261 [69]	**Rev. 4 Priority:** **Medium** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: High (2007)

Table 3-5
Inspection (IN) R&D Gaps

R&D Gap Description	Priority
B-IN-01 - Inspection of Core Plate Rim Hold Down Bolts **Issue:** Inspection of core plate bolts, currently required by BWRVIP-25, cannot be effectively performed using currently available technologies. EVT-1 does not have access to the threaded areas where cracking is most likely to occur. Although UT is a possible alternative, access to either the top or bottom end of the bolt is needed. However, the lower end of the bolt is inaccessible and the top end of the bolt is covered by an anti-rotation keeper. **Description:** The core plate assembly provides lateral support for the fuel bundles, control rod guide tubes, and in-core instrumentation during seismic events, as well as vertical support for the peripheral fuel assemblies. The typical core plate assembly consists of a perforated stainless steel plate reinforced by stiffener beams and supported on the perimeter by a circular rim. The core plate rim is bolted to a ledge on the core shroud by stainless steel studs that prevent vertical movement. Recommended bolting inspection methods and frequencies are contained in BWRVIP-25. Recent focus has been on development of an engineering basis for elimination of core plate hold down bolt inspection requirements. BWRVIP-25, Rev. 1 provides an NRC-approved alternative to bolt inspection. However, plants must meet the set of load requirements specified in BWRVIP-25 Rev. 1 for the engineering basis to apply and it is not clear at present that all plants desiring to eliminate core plate hold down bolt examinations can demonstrate the load requirements are met. This inspection gap will remain open until either a novel method for demonstrating bolt integrity is developed or it is determined that the engineering basis in BWRVIP-25, Rev. 1 can be applied by the majority of plants desiring to eliminate core plate hold down bolt inspections. A low priority is assigned given that BWRVIP-25, Rev. 1 provides a method that is thought to apply to most plants and other options are potentially available to owners (e.g., plant-specific evaluation). **References** BWRVIP-25, Rev. 1	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: Medium (2008) Rev. 0: Medium (2007)

Table 3-5 (continued)
Inspection (IN) R&D Gaps

R&D Gap Description	Priority
B-IN-02 - Inspection of Hidden Weld Locations **Issue:** BWRVIP I&E guidelines recommend inspection of hidden welds when the technology becomes available. To date, efforts to develop effective NDE solutions for all hidden (i.e., inaccessible) weld locations have not been successful. **Description:** BWRVIP I&E guidelines require inspection of several hidden welds. Hidden welds include: Jet Pump Thermal Sleeves TS-1,-2,-3,-4 Core Spray Thermal Sleeves, and piping welds P1 and P9. LPCI Thermal Sleeve 45-12 and Elbow to Thermal Sleeve 6-1a. DF-3, AD-3a, b jet-pump diffuser welds containing backing ring configurations. Investigations into alternatives for ultrasonic examination of weld P9 by the EPRI NDE Center suggest that no ultrasonic solution will be feasible using current technologies. Future efforts will likely be focused on other NDE techniques. The BWRVIP has developed methodologies to assess core spray, jet pump, and LPCI coupling hidden welds. These methods provide a basis for not examining hidden welds until such time as a significant percentage of similar welds that are accessible for examination are found are found to be cracked. NRC has approved this approach for U.S. plants. Although the near-term need for inspection capability is mitigated by this situation, this gap remains open to address long-term needs. At some point in the future, units may not be able to meet the assessment criteria due to cracking in similar welds and would be required to inspect hidden weld locations. Finally, regardless of any regulatory need, the capability to assess the integrity of these hidden welds would be valuable. Jet-pump diffuser fillet weld locations contain differing inspection regions depending on which examination technique is used (i.e. outside surface EVT-1 versus inside surface UT). This has created scenarios where no solution exists to monitor some EVT-1 indications other than conservatively monitoring/re-evaluating previously identified indications and significantly increased EVT-1 examination frequencies. Resolution of this gap includes development of an NDE system capable of accurately detecting and characterizing relevant flaws in hidden welds. **References** BWRVIP-18R2-A [70], BWRVIP-41R4-A [40], BWRVIP-42R1-A [71], BWRVIP-152 [72], BWRVIP-168 [73]	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: Medium (2008) Rev. 0: High (2007)

Table 3-5 (continued)
Inspection (IN) R&D Gaps

R&D Gap Description	Priority
B-IN-03a - Inspection of Core Shroud Weld Locations **Issue:** Current UT techniques are not capable of being demonstrated to detect all expected cracking orientations and locations for Welds H3 or H6b for some plants. **Description:** Weld H3 (BWR/6 & ABWR): There are two BWR/6 H3 core shroud weld configurations and a slightly different ABWR H3 weld configuration that all differ significantly from BWR/2 through BWR/5 designs. The BWR/6 and ABWR shroud designs contain a machined top-guide plate that is bolted in between the H2 and H3 welds. Alignment pin and threaded bolt holes are contained in the ~7.25-inch thick ring above the H3 weld. The lower portion of the ring has a machined chamfer transitioning to the weld. To effectively examine the upper side of some BWR/6 H3 welds, a combination of inspection tooling as well as a combination of scan surfaces is needed. In some instances, this requires that access be provided to both the inside and outside surface of the core shroud. Since the H3 weld is located below the top guide, this has created scenarios where a significant volume of periphery fuel cells must be evacuated. New examination technology is needed to more efficiently examine some of these bolted H3 weld configurations. Weld H6b (BWR/6): OD access and scanning capability is limited because of the proximity of jet pump diffusers and due to scallops that are machined into the shroud to accommodate the jet pump diffusers. Scanning cannot be performed from the shroud inside surface due to numerous pockets that are machined into the shroud to provide access for core plate hole down bolts and nuts. Welds H3 and H6b (BWR/2 through BWR/5): Recent enhancement to examination tooling and technologies have increased the types of circumferential flaws that can be detected in H3 and H6b weld locations. However, unique flaw orientations, with relation to the scan surface for H3 and H6b welds, significantly challenge the ability to consistently and accurately measure the through-wall extent of certain flaw scenarios (i.e., shallow inside surface initiating flaws or significant outside surface initiating flaws). This could potentially result in challenges to performing flaw evaluations or attempting to trend flaw growth over successive examinations. Weld H7: Due to interferences (e.g., shroud gussets, jet pump instrumentation lines), obtaining significant volumes of coverage from both sides of H7 is difficult at some plants. **References** BWRVIP-03 (latest version) [74], BWRVIP-76, Rev. 2 [53]	**Rev. 4 Priority:** **Medium** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: High (2007)

Table 3-5 (continued)
Inspection (IN) R&D Gaps

R&D Gap Description	Priority
B-IN-03b – Inspection of Shroud Support Weld Locations **Issue:** Current UT techniques are not capable of detecting all expected cracking orientations and locations for Welds H8 and H9. Additionally, two-sided visual examination is not practical for H8 and H9 since access below the shroud support plate is not normally available. **Description:** Weld H8: From a structural perspective, updated BWRVIP analyses indicate that a single- sided exam of H8 is sufficient, as there would have to be significant through-wall cracking in the weld to challenge integrity of the component. If cracking is observed in the annulus-side exam of H8, scope expansion is required by BWRVIP-38. However, such scope expansion may in some cases include, an examination of the core-side of H8 This scope expansion may include supplemental examinations from the top and bottom if and where access is available. Therefore, identification of NDE techniques capable of detecting and sizing indications in H8, regardless of location or orientation, remains a gap. Weld H9: Several BWR units do not have access to the outside surface of the RPV at the elevation of weld H9. Vendors have demonstrated capabilities for detecting and sizing circumferential cracking by scanning on the horizontal surface of the shroud support plate, however, attempts to demonstrate the capability to detect and characterize transverse cracking of weld H9 have not been successful. Additionally, it is not currently possible to determine if an H9 flaw has propagated into the low-alloy RPV material when scanning is limited to only the inside surface of the RPV. Closure of this gap involves development of effective NDE techniques for that can be demonstrated to reliably detect relevant flaws in shroud support welds. Although engineering analyses confirm large flaw tolerance from the viewpoint of shroud support load carrying capability and RPV structural integrity, this gap remains open to address the potential need for scope expansion should cracking in Weld H9 be detected. **References** BWRVIP-03 (latest revision) [74], BWRVIP-38 [75], BWRVIP-269 [76], BWRVIP letter 2006-334 [77]	**Rev. 4 Priority:** **Medium** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: High (2007)

Table 3-5 (continued)
Inspection (IN) R&D Gaps

R&D Gap Description	Priority
B-IN-11 – BWR/6 Top Guide Grid Beam UT **Issue:** Ultrasonic examination techniques have not been demonstrated for the examination of BWR/6 top guide configurations. The material grain structure and chamfer surface of some BWR/6 configurations will require additional development of UT techniques.	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Low (2013-2015) *[Gap added in 2014 RIC Meeting.]*
B-IN-12 – Access Hole Cover Weld UT **Issue:** Current UT techniques cannot reliably detect and characterize all expected cracking orientations and locations for some access hole cover (AHC) designs. Potential undetected cracking could represent a structural integrity concern. **Description:** Significant cracking has been identified during EVT-1 and UT examinations of some AHC designs. SCC is a potential concern for AHC designs that are highly restrained or include the use of Alloy 182 weld metal. The proximity of the core shroud and RPV limit the amount of dual-side coverage that can be obtained during ultrasonic examinations. The dendritic structure of the Alloy 82/182 AHC welds negatively impact the ability to detect and reliably characterize flaws in regions where the examination can only be performed from a single side of the weld. **References** BWRVIP-03 (latest revision) [74], BWRVIP-180 [54]	**Rev. 4 Priority:** **Medium** **Priority History:** *[New in Rev. 4]*

Table 3-6
Repair/Replacement (RR) R&D Gaps

R&D Gap Description	Priority
B-RR-02a – Analytical Tools for Weld Repair of Irradiated Components **Issue:** The analytical models that would be applied to support development of acceptable welding process parameters for weld repair of irradiated components have limitations that must be resolved before being applied in the field. **Description:** While welding is often a preferred method for repairing degraded reactor internal components, welding on irradiated materials is problematic due to helium that develops within the steel as fluence accumulates. If the material to be welded is highly irradiated, welding can result in cracking due to the generation and coalescence of helium bubbles at grain boundaries (i.e., helium-induced cracking). The BWRVIP has developed both analytical and empirical methods to define acceptable welding parameters for qualification of weld repair procedures. The analytical model remains limited to application of weld overlays using a TIG welding process. However, actual repair applications would likely involve application to fillet and groove configurations and potentially involve the application of laser beam welding processes. Although empirical methods can be applied estimate weldability, there are significant gaps in the data used to develop the empirical correlations. Additionally, due to the inherent uncertainty in test data supporting empirical models, significant conservatisms must be included in the selection of welding process parameters. Overly conservative parameters extend the time required to complete a repair and could result in unnecessary rejection of welding as a repair option. Closure of this gap includes extension of analytical models to include fillet and groove configurations and laser welding processes and further reductions in the level of conservatism included in the empirical models to support qualification of multi-pass welding processes for highly irradiated materials. **References** BWRVIP-97R1 [78], BWRVIP-98 [79], BWRVIP-151 [80], BWRVIP-228 [81]	**Rev. 4 Priority:** **Medium** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: Medium (2008) Rev. 0: Medium (2007)

Table 3-6 (continued)
Repair/Replacement (RR) R&D Gaps

R&D Gap Description	Priority
B-RR-02b – Irradiated Materials Weldability Data **Issue:** There remain gaps in weldability datasets for irradiated materials potentially relevant to welded repairs in BWR RPVs and internals. **Description:** In some cases, welding potentially represents a cost-effective alternative to mechanical repair in terms of time saved during repair design and implementation. In a small number of cases, a welded repair could be the only viable option. Helium-induced cracking (HeIC) can occur due to the generation and coalescence of helium bubbles at grain boundaries. Components subject to higher fluence are progressively more difficult to repair by welding since helium is generated in the material through Boron and Nickel neutron transmutation reactions. The weldability limit for irradiated materials is commonly characterized as the value of weld heat input beyond which HeIC occurs for a given atomic He concentration in a given alloy. Understanding the heat input threshold value that must not be exceeded to ensure HeIC does not occur is critical to assessment of repair feasibility. The following gaps in the existing weldability datasets: Stainless Steels: Although the weldability of irradiated 304SS is relatively well characterized, it would be highly beneficial to have additional data to both confirm datasets obtained from Japanese test programs and to assess weldability limits at higher material helium concentrations (i.e., 10 appm and higher). For molybdenum containing 300 series stainless steels (i.e., 316SS and 316LSS), there are significantly fewer data than for 304SS. At helium concentrations greater than 1 appm, data associated with successful welding are extremely limited. Although a weldability boundary is provided in BWRVIP-97, Rev. 1, the boundary is drawn conservatively and extends only up to 1 appm He. There are too few data to support weldability at helium concentrations greater than 1 appm. Nickel-base Alloys: Finally, there are essentially no data that can be used to support a weldability boundary of any kind for relevant nickel-base alloys (i.e., Alloy 600, Alloy 82, & Alloy 182). The potential for successful welded repair of irradiated nickel-base alloys would involve either a substantial amount of uncertainty. Low-Alloy Steels: As plants continue to operate and fluence is increasing on RPV low-alloy steel materials, questions have been raised regarding weldability limits for low-alloy steels. Although there are anecdotal data suggesting that no issue exists, a comprehensive assessment of this issue has not been performed. **References** BWRVIP-97R1 [78], 3002015849 [82]	**Rev. 4 Priority:** **High** **Priority History:** Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: Medium (2008) Rev. 0: Medium (2007)

Table 3-6 (continued)
Repair/Replacement (RR) R&D Gaps

R&D Gap Description	Priority
B-RR-05 – Alternate High Strength Materials **Issue:** Due to SCC concerns associated with X-750, it is desirable to identify suitable alternatives. **Description:** Alloy X-750 is a nickel-base alloy used for BWR internal components where high strength, corrosion resistance, and in some cases, thermal expansion coefficient are considerations. In BWR reactor internals, high strength alloys are primarily used for shroud repair hardware and jet pump beams. In these applications, the 300-series stainless steels typically utilized for reactor internals are not capable of meeting the minimum strength requirements. Unfortunately, Alloy X-750 is now known to be susceptible to SCC at sustained stress levels less than the yield strength of the material. While the heat treatment for Alloy X-750 has been improved to optimize the SCC resistance, concerns regarding SCC susceptibility remain. Although Alloy 718 has now been qualified as an alternative to X-750, qualification of additional alternatives to X-750 would be beneficial. Testing indicates that Alloy 725 and Hicoroy 11 appear to be excellent alternatives, with SCC growth rates that are ~2 - 3 orders of magnitude lower than that for Alloy X-750. Additionally, depending on the intended application(s), there may be a need to study the irradiation behavior of these newer alternative alloys. Resolution of this gap involves qualification of additional X-750 alternatives for use in BWR environments and inclusion of these alloys in updates to BWRVIP-84. **References** BWRVIP-84R3 [83], BWRVIP-229 [84], EPRI 1026482 [85], EPRI 3002005470 (EPRI Materials Handbook) [86]	**Rev. 4 Priority:** **Low** **Priority History:** Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: Medium (2008) Rev. 0: Medium (2007)

Table 3-6 (continued)
Repair/Replacement (RR) R&D Gaps

R&D Gap Description	Priority
B-RR-08 – Availability of Advanced Welding Processes for Weld Repair of Highly Irradiated Components **Issue:** In order to perform successful weld repairs on highly irradiated reactor internal components, it is necessary to utilize low-heat-input welding techniques. However, additional development and qualification are needed before these processes can be applied for use in repairing BWR RPV and internals components. **Description:** Development of qualified and approved low heat input weld processes, such as laser beam welding (LBW), allow for application of welding as a repair technique in cases where welding was previously not an option. In some cases, welding represents a cost-effective alternative to mechanical repair in terms of time saved during repair design and implementation. In other cases, it represents the only viable option for repair. Additional advancements would be beneficial. For example, a new LBW process, auxiliary beam stress improved (ABSI) laser welding, has recently been developed and tested on irradiated material using proof-of-concept equipment. This low heat input LBW technique alters the in-process stress state during welding to provide further resistance to helium-induced cracking. Additional process design, development and qualification are required to refine the equipment for field-deployment and to ensure the process is properly tailored for low-profile, underwater BWR RPV internal repair applications. Other advanced processes, e.g., friction stir welding, could also potentially be applied. However, similar to ABSI LBW, development and qualification would be needed before these techniques could be applied to BWR RPV and internals components. **References** BWRVIP-151 [80], 1019167 [87], 1021174 [88], 3002005553 [89], 3002003146 [90]	**Rev. 4 Priority:** **Medium** Priority History: Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: Low (2008) *[New in Rev. 1]*

4
PROGRAM REQUIREMENTS

Table 4-1 contains a summary list of BWRVIP program requirements. Table 4-2 describes each of the BWRVIP program requirements. Program requirements are ongoing program activities that are necessary to ensure that materials issues continue to be adequately managed in the future and to ensure that previously closed R&D gaps remain closed.

Table 4-1
Summary of BWRVIP Program Requirements and Status (Active / Inactive)

RQMT ID	Requirement Description
B-RQ-01	Regulatory Interface and Support
B-RQ-02	Integrated Surveillance Program (ISP) Maintenance
B-RQ-03	BWR Fluence Modeling (RAMA Code)
B-RQ-04	Inspection Database Maintenance
B-RQ-05	BWR Water Chemistry Guideline Maintenance
B-RQ-06	BWRVIP Guidance Maintenance

Table 4-2
BWRVIP Program Requirements (RQ)

Program Requirement Description
B-RQ-01 – Regulatory Interface and Support The BWRVIP provides regulatory and aging management program implementation support for both U.S. and international members. Within the U.S., the BWRVIP frequently interacts with the NRC to ensure that the NRC understands BWRVIP guidance and that members are aware of regulatory commitments and regulator concerns. Where beneficial to the overall membership, new and revised BWRVIP reports may be transmitted to NRC and the BWRVIP supports the interactions needed for NRC review and approval.[a] Internationally, the BWRVIP engages regulators for the benefit of members where possible through sharing of information and advocating for aging management approaches that are supported by unbiased research. BWRVIP staff also support program implementation needs through engagement in international forums and workshops as appropriate. **References:** BWRVIP-94NP (current version) [91] [a] Where possible, revised guidance is released for implementation using the document screening process in Appendix C of NEI 03-08, Revision 3 [1] instead of NRC review and approval. This process is managed by the BWRVIP and results are communicated to the NRC.
B-RQ-02 – Integrated Surveillance Program (ISP) Maintenance 10CFR50, Appendix H requires that each BWR have a vessel materials surveillance program. The purpose of the surveillance program is to monitor the changes in vessel material properties resulting from neutron irradiation damage Appendix H specifies the design requirements of the surveillance program and for the withdrawal schedule for each capsule The BWR ISP is an integrated surveillance capsule testing program that utilizes data from individual plant specimens and the Supplemental Surveillance Program (SSP) previously initiated by the BWR Owners' Group (BWROG). The ISP proactively addresses vessel surveillance requirements for the U.S. fleet and eliminates testing of low value capsules. For many U.S. plants, the ISP is the only option to demonstrate compliance with U.S. regulations. Testing of these capsules in future years is a BWRVIP commitment that remains essential for many U.S. BWRs. This requirement will remain open until such time as all needed capsule testing has been completed to address fleet needs for both 60-year operations and extended operations (operation beyond 60 years). **References:** BWRVIP-86R1-A [92], BWRVIP-135R3 [93], 10CFR50 Appendix H [94], BWRVIP-321 [95]
B-RQ-03 – BWR Fluence Modeling (RAMA Code) Methods for neutron fluence determinations are needed to 1) determine neutron fluence in the RPV shells and welds and at surveillance capsule locations and 2) determine fracture toughness and crack growth rates for flaw evaluations, structural assessments, and for evaluating repair technologies. The RAMA Fluence Methodology provides BWRVIP members with state-of-the-art software for fluence calculations. Maintenance and improvement of the RAMA code as needed is most accurately described as a program requirement primarily because the code is widely used by BWRVIP members. In addition, the code is used to support maintenance of the ISP (B-RQ-02) and conformance with NRC R.G. 1.190. **References:** BWRVIP-114-A [96], BWRVIP-115-A [97]. BWRVIP-126R2 [98], BWRVIP-157 [99], BWRVIP-189 [100], BWRVIP-209 [101], NRC Reg. Guide 1.190 [102]

Table 4-2 (continued)
BWRVIP Program Requirements (RQ)

Program Requirement Description
B-RQ-04 – Inspection Database Maintenance The BWRVIP has committed to its member utilities as well as to the NRC to collect and evaluate the results of component inspections as a means of ensuring proactive identification of adverse trends. The BWRVIP inspection database is the primary tool used to ensure this requirement is met. These inspection data are also important to supporting revised guidance that provides appropriate credit for hydrogen water chemistry (HWC) technologies. This issue program requirement is expected to continue indefinitely, i.e., for the life of the current BWR fleet. **References:** BWRVIP-94 (current version) [91]
B-RQ-05 – BWR Water Chemistry Guideline Maintenance The EPRI BWR Water Chemistry Guidelines (BWRVIP-190) provide guidance that is an essential part of the BWRVIP strategy to manage degradation of BWR components. This guideline document is reviewed periodically and revised as appropriate to reflect new data and research results. BWRVIP-62 provides technical bases for inspection relief for BWR internal components with hydrogen injection. Maintenance of BWRVIP-62 represents a significant value to the BWRVIP membership in terms of reduced inspection costs for locations protected by hydrogen water chemistry technologies. Since BWRVIP guidance related to inspection and evaluation of BWR components is predicated on adherence to the requirements in these guidance documents, continued maintenance of these documents is a BWRVIP program requirement. **References:** BWRVIP-94 (current version) [91], BWRVIP-190R1 [58]
B-RQ-06 – BWRVIP Guidance Maintenance The BWRVIP program maintains important guidance for members including inspection and flaw evaluation (I&E) guidelines, NDE program implementation guidance, and repair design criteria (RDC): BWRVIP I&E guidelines provide a structure for plant inspection programs for BWR reactor internals that includes inspection locations, inspection techniques, sample sets, scope expansion, and re-inspection intervals. These documents also provide guidance for evaluation of inspection results and disposition of relevant indications. Maintenance of these documents is necessary to ensure that the guidance remains adequate and up to date. In addition to I&E Guidelines, the BWRVIP maintains a number of supporting guidance documents, addressing topics such as CGR correlations, the effect of neutron fluence on material properties, and loads and load combinations to be included in flaw evaluations. Maintenance of the BWRVIP NDE program as implemented by BWRVIP-03 represents one of the central functions of the Inspection Committee. This activity includes support for demonstration of new NDE techniques. New techniques, usually UT applications, must be demonstrated on realistic mockups and documented prior to implementation. If the BWRVIP does not support demonstrations, the adequacy of UT examinations performed in the field could be questioned. NDE program maintenance includes design and fabrication of new demonstration mockups and support for implementation of NDE techniques for which the existing mockups are not applicable. BWRVIP Repair Design Criteria and supporting reports provide guidance regarding BWR internal component repairs. In addition, BWRVIP-84 provides guidance on procurement, design, welding requirements, and fabrication limitations for BWR reactor internals repair hardware.

5
SUMMARY OF RESULTS

Section 5.1 provides a distilled summary of the current high priority R&D areas related to proactively managing materials degradation in BWRs. Section 5.2 provides a summary discussion of the R&D gap trends observed in Revision 4.

5.1 High Priority R&D Areas

Most of the high priority R&D needs identified by the gap assessment results can be grouped into four broad categories; SCC CGR correlations, management of irradiated reactor internals, ensuring effective SCC mitigation using HWC with online addition of noble metal catalysts (OLNC), and effective collection and analysis of field inspection data. These areas are described in Sections 5.1.1 through 5.1.4.

5.1.1 SCC CGR Correlations

Component inspections represent a significant resource requirement for BWR owners. In many cases, inspection requirements for unflawed components and inspection intervals for flawed components rely on flaw tolerance evaluation as a primary basis. Because SCC growth is presumed to occur, the fixed CGR or K-dependent CGR correlation used in the evaluation often controls the result obtained. Although SCC crack growth in austenitic materials in BWR environments has been studied extensively, there are remaining gaps. For austenitic stainless steel and nickel-base alloy welded components these gaps include ensuring that differences in material grade are sufficiently well understood, improving the state of knowledge regarding the effect of high purity NWC on CGR, and better quantifying the effect of water chemistry transients on CGRs.

A related topic is evaluation and disposition of field inspection data results which often show the CGR correlations and other predictive tools developed based on laboratory data to be overly conservative. Although the likely contributors are known (e.g., residual stress state), it is difficult to separate these factors in such a way that supports meaningful refinement of current methods based primarily on results obtained from laboratory testing. A second related topic involves improving SCC initiation prediction capabilities. Existing CGR correlations are at times established very conservatively. These conservative decisions were made in part to accommodate the possibility of new crack initiations. However, there is little evidence indicating that initiation of new cracks is occurring. As a result, there is a need to better understand the potential for new crack initiations so that conservative assumptions included in CGR correlations can be removed without a significant reduction in overall confidence in component integrity.

5.1.2 Management of Irradiated Reactor Internals

It is clear from laboratory testing that neutron fluence increases CGRs, negatively impacts fracture properties, and causes elemental segregation at grain boundaries that make irradiated materials potentially more susceptible to SCC. However, evaluation of field data remains inconclusive with regard to the relative importance of neutron fluence on SCC susceptibility. There are also remaining gaps in material property datasets such as the lack of test data to characterize fracture properties of highly irradiated weld and HAZ materials, including assessment of high fluence saturation values. Additionally, there are some anomalous data that are not easily dispositioned (i.e., weld metal test data showing substantially lower than anticipated fracture toughness).

Addressing these uncertainties is considered an important need since any unexpected adverse degradation trend in the highly irradiated core support structures could potentially represent a significant challenge. As a result, the capability to predict future degradation with a higher degree of confidence or to be able to confidently distinguish between higher and lower risk plants is a significant unaddressed R&D need.

5.1.3 Effective Implementation of OLNC

Within the subset of BWRs that have historically used HWC technologies, most plants have transitioned to use of the OLNC process. Mitigation of SCC through HWC technologies is deemed important to overall asset management and essential in cases where mitigation-based inspection relief is desired. Although OLNC is considered a mature technology, there are remaining questions about the parameters critical to comprehensive mitigation and verification that plant conditions resulting from typical OLNC implementation can be represented by results obtained in controlled laboratory studies. Fundamental questions involve the true impact of particle size, spacing and density on mitigation effectiveness, the effectiveness of catalyst particle partially or totally incorporated into oxide layers, penetration of catalyst particles into existing cracks, and assessment of the impact of fluid flow conditions and surface structure on catalyst deposition. Questions related to plant implementation of OLNC relate to the adequacy of various catalyst loading monitoring methods, modeling capabilities, and catalyst deposit durability.

5.1.4 Analysis of Plant Data

Although not traditionally included in the IMTs as specific gap, the importance of data obtained from plants directly in the form of inspection results, sample specimen analyses, or monitoring data is noted in the connection with many high and medium priority R&D gaps. These data are likely important to resolving a number of significant questions regarding the appropriate use of lab data as inputs to aging management program requirements. The data may also be important inputs to development of models for predicting degradation trends. Given that optimization efforts will continue to place pressure on inspection reductions, it is important that plant data critical to decision making are identified and curated in a way that maximizes their usefulness and application to the entire BWR fleet.

5.2 R&D Gap Trends

There are 30 open R&D gaps in this revision of the BWR IMTs (Revision 4). For comparison, there were 37 open gaps in Revision 3. Trends observed with regard to the overall number and priority of R&D gaps are listed below.

1. The effort undertaken to close R&D gaps primarily related to regulatory issues (with capture of these issues in a separate regulatory issue matrix) resulted in the closure of a significant number of gaps, with a majority of these gaps being assessment (AS) gaps.
2. Although a number of RPV integrity related regulatory issues remain open (and will be tracked in the regulatory issue matrix), there are remaining commitments that must be fulfilled (i.e., ISP capsule testing and evaluation), and there are opportunities for further optimization of RPV integrity management methods, the IMT Revision 4 results do not include any active / open R&D gaps related to RPV integrity. While it is acknowledged that RPV integrity is critical to safety, BWR RPV integrity issues have been studied extensively and due to the relatively low neutron flux, there are also many data from PWRs that can be applied to characterize BWR RPV integrity.
4. Given the significant time gap between Revision 3 and Revision 4, there were a number of additional gap closures resulting from completion of relevant R&D and changes in industry needs (i.e., issue obsolescence).
5. Relatively few new gaps were identified. This is consistent with the expectation that as gap assessments are repeated over time, issues overlooked in initial assessments are captured. Additionally, it is observed that BWR RPV and internals performance has been relatively stable in recent years. There have been few truly unique or new aging management issues.
6. Throughout the gaps, a consistent effort was made to ensure that issues important to international members are identified so that R&D proposed for future years can be more effective in addressing international member needs.
7. A number of degradation mechanism understanding (DM) gaps were extensively rewritten so that fundamental issues important to managing aging of plant components are better communicated.

6
REFERENCES

1. NEI 03-08 Rev. 3, "Guideline for the Management of Materials Issues", February 2017.
2. *EPRI Materials Degradation Matrix, Revision 4*. EPRI, Palo Alto, CA: 2018. 3002013781.
3. *BWRVIP-167: BWR Vessel and Internals Project, Boiling Water Reactor Issue Management Tables*. EPRI, Palo Alto, CA: 2007. 1014753.
4. *BWRVIP-167NP, Revision 1: BWR Vessel and Internals Project, Boiling Water Reactor Issue Management Tables*. EPRI, Palo Alto, CA: 2008. 1018111.
5. BWRVIP Letter 2009-216, Interim Revision to BWR Issue Management Tables to Include New, Revised and Re-Ranked R&D Gaps, July 16, 2009.
6. *BWRVIP-167NP, Revision 2, BWR Vessel and Internals Project: Boiling Water Reactor Issue Management Tables*. EPRI, Palo Alto, CA: 2010. 1020995.
7. *BWRVIP-167NP, Revision 3: BWR Vessel and Internals Project: Boiling Water Reactor Issue Management Tables*. EPRI, Palo Alto, CA: 2013. 3002000690.
8. *Materials Reliability Program: Low-Temperature Cracking of Nickel-Based Alloys and Weld Metals (MRP-108)*. EPRI, Palo Alto, CA: 2004. 1009400.
9. *Materials Reliability Program: Effects of Dissolved Hydrogen, Temperature, and Hydrogen Peroxide on Low Temperature Crack Propagation (LTCP) Fracture Resistance of Weld Metals 182, 52, and 152 (MRP-209)*. EPRI, Palo Alto, CA: 2007. 1015005.
10. *Materials Reliability Program: Evaluation of Potential for Low Temperature Crack Propagation in Reactor Pressure Vessel Outlet Nozzle Dissimilar Metal Butt Welds by Stress and Fracture Mechanics Analyses (MRP-247)*. EPRI, Palo Alto, CA: 2008. 1016606.
11. *Materials Reliability Program: Low-Temperature Crack Propagation in Pressurized Water Reactor Materials (MRP-293)*. EPRI, Palo Alto, CA: 2011. 1021018.
12. *Program on Technology Innovation: Scoping Study of Low Temperature Crack Propagation for 182 Weld Metal in BWR Environments and for Cast Austenitic Stainless Steel in PWR Environments (Revision 1)*. EPRI, Palo Alto, CA: 2010. 1020957.
13. *Potential Effects of Hydrogen-Related Degradation on Reactor Pressure Vessel Materials (MRP-431): Review of Current State of Understanding*. EPRI, Palo Alto, CA: 2018. 3002013225.
14. *BWRVIP-233, Rev. 2: Updated Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment*. EPRI, Palo Alto, CA: 2018. 3002013026.

15. *BWRVIP-306: BWR Vessel and Internals Project, Stress Corrosion Crack Growth Rates of Low Alloy Steels During Chloride Transients: Crack Activation and Arrest Times*. EPRI, Palo Alto, CA: 2017. 3002010544.

16. *BWRVIP-302: BWR Vessel and Internals Project, Examination of a Boat Sample Removed from the Core Shroud at Hatch Unit 1*. EPRI, Palo Alto, CA: 2016. 3002008370.

17. *Environmentally Assisted Fatigue Gap Analysis and Roadmap for Future Research: Roadmap*. EPRI, Palo Alto, CA: 2012. 1026724.

18. *Update on the Development of Fatigue Usage Life and Gradient Factors*. EPRI, Palo Alto, CA: 2017. 3002010554.

19. *Development of an Alternative Approach to ASME Code Simplified Elastic-Plastic Analysis*. EPRI, Palo Alto, CA: 2018. 3002014122.

20. *Mechanistic Understanding of Environmentally Assisted Fatigue Crack Growth of Austenitic Stainless Steels in PWR Environments*. EPRI, Palo Alto, CA: 2016. 3002007973.

21. *Materials Reliability Program: Environmentally Assisted Fatigue of Austenitic Stainless Steel Under Plant-Like Conditions (MRP-415)*. EPRI, Palo Alto, CA: 2016. 3002009472.

22. *BWRVIP-100, Rev. 1-A: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds*, EPRI, Palo Alto, CA: 2016. 3002008388.

23. *BWRVIP-140: BWR Vessel and Internals Project, Fracture Toughness and Crack Growth Program on Irradiated Austenitic Stainless Steel*. EPRI, Palo Alto, CA: 2005. 1008189.

24. *BWRVIP-154, Revision 2: BWR Vessel and Internals Project, Fracture Toughness in High Fluence BWR Materials*. EPRI, Palo Alto, CA: 2009. 1019077.

25. *BWRVIP-294, Rev. 2: Fracture Toughness of Zorita RPV Core Internals Applicable to BWRs: Final Report 2019*. EPRI, Palo Alto, CA: 2019. 3002015929.

26. S. Fyfitch, et al, "Fracture Toughness of Irradiated Stainless Steel in Nuclear Power Systems," 14th International Conference on Environmental Degradation of Materials in Nuclear Power System, Virginia Beach, VA, August 23-27, 2009.

27. *BWRVIP-14-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals*, EPRI, Palo Alto, CA: 2008. 1016569.

28. *BWRVIP-80-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Shroud Vertical Welds*. EPRI, Palo Alto, CA: 2007. 1015457.

29. *BWRVIP-99-A: BWR Vessel and Internals Project, Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components*, EPRI, Palo Alto, CA: 2008. 1016566.

30. *BWRVIP-174, Revision 2: BWR Vessel and Internals Project: Review of BWR Core Shroud UT Re-Inspection Results for Plants Mitigated with NMCA, OLNC, and HWC*. EPRI, Palo Alto, CA: 2019. 3002015904.

31. *BWRVIP-221: BWR Vessel and Internals Project, Crack Growth in High Fluence BWR Materials*. EPRI, Palo Alto, CA: 2009. 1019079.

32. *BWRVIP-232: BWR Vessels and Internals Project, Highly Irradiated Stainless Steel Crack Growth Program on General Electric Control Rod Blade Materials*. EPRI, Palo Alto, CA: 2009. 1019078.

33. *BWRVIP-253: BWR Vessel and Internals Project, Evaluation of Shroud Cracking at Plant Hatch, Unit 1*. EPRI, Palo Alto, CA: 2011. 1022839.

34. *BWRVIP-265: BWR Vessel and Internals Project, Crack Growth in High Fluence BWR Materials-Phase 2*. EPRI, Palo Alto, CA: 2012. 1026508.

35. *Crack Growth Rates of Irradiated Commercial Stainless Steels in BWR and PWR Environments*. EPRI, Palo Alto, CA: 2009. 1019028.

36. *Crack Growth Rates of Fast Reactor Irradiated Solute Addition Alloys in BWR and PWR Environments*. EPRI, Palo Alto, CA: 2009. 1019029.

37. *Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments*. EPRI, Palo Alto, CA: 2014. 3002003103.

38. ASME Code Case N-889, Reference Stress Corrosion Crack Growth Rate Curves for Irradiated Austenitic Stainless Steels in Light Water Reactor Environments, Division 1, Section XI, 2018.

39. *BWRVIP-322: BWR Vessel and Internals Project, Revised Allowable High-Cycle Fatigue Curves for Austenitic and Ferritic Steels*, EPRI, Palo Alto, CA: 2019. 3002014859.

40. *BWRVIP-41, Rev. 4-A: BWR Vessel and Internals Project: BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines*, EPRI, Palo Alto, CA: 2018. 3002014254.

41. *BWRVIP-252: BWR Vessel and Internals Project, Evaluation of BWR Jet Pump Operational Data*, EPRI, Palo Alto, CA: 2011. 1022838.

42. *BWRVIP-263: BWR Vessel and Internals Project, Full Flow Testing to Investigate Flow Induced Vibration Phenomena in BWR/5 Jet Pumps*. EPRI, Palo Alto, CA: 2012. 1026577.

43. *BWRVIP-266: BWR Vessel and Internals Project, Technical Bases for Revision of the BWRVIP-41 Jet Pump Inspection Program*. EPRI, Palo Alto, CA: 2012. 1025140.

44. *BWRVIP-06 Rev. 1-A: BWR Vessel and Internals Project, Safety Assessment of BWR Reactor Internals*, EPRI, Palo Alto, CA: 2009. 1019058.

45. *BWRVIP-218: BWRVIP Vessel and Internals Project, Alloy X-750 Characterization Study*. EPRI, Palo Alto, CA: 2009. 1019070.

46. *BWRVIP-240: BWR Vessel and Internals Project, Metallurgical Analyses and Macro- and Microstructural Mapping of Alloy X-750 and AlloyXM-19 Plates*. EPRI, Palo Alto, CA: 2010. 1021003.

47. *BWRVIP-291, Revision 1: BWR Vessel and Internals Project: Microstructure, SCC, and J-R Fracture Resistance of Alloy X-750 and XM-19: Final Report 2017*. EPRI, Palo Alto, CA: 2017. 3002011583.

48. *BWRVIP-262NP: BWR Vessel and Internals Project, Baseline Fracture Toughness and Crack Growth Rate Testing of Alloys X-750 and XM-19 (Idaho National Laboratory Phase I)*. EPRI, Palo Alto, CA: 2012. 1025135.

49. *BWRVIP-59-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals*, EPRI, Palo Alto, CA: 2007. 1014874.

50. *BWRVIP-47-A: BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines*, EPRI, Palo Alto, CA: 2004. 1009947.

51. *BWRVIP-311: BWR Vessel and Internal Project, Core Shroud Weld Off-Axis Crack Growth Assessment*. EPRI, Palo Alto, CA: 2017. 3002010682.

52. BWRVIP 2016-030, *Core Shroud Off-Axis Cracking Interim Inspection & Flaw Evaluation Guidance*, March 4, 2016.

53. *BWRVIP-76*, Rev. 2: *BWR Vessel and Internals Project: BWR Core Shroud Inspection and Flaw Evaluation Guidelines*. EPRI, Palo Alto, CA: 2014. 3002003095.

54. *BWRVIP-180: Access Hole Cover Inspection and Flaw Evaluation Guidelines*, EPRI, Palo Alto, CA: 2007. 1013402.

55. *BWRVIP-62-A (2018 Update): BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection.* EPRI, Palo Alto, CA: 2018. 3002014434.

56. *BWR Vessel and Internals Project: Optimization of Zirconium Oxide Coating Technology to Mitigate Intergranular Stress Corrosion Cracking in BWRs (BWRVIP-109)*. EPRI, Palo Alto, CA: 2002. 1002838.

57. *BWRVIP-163: BWR Vessel and Internals Project: Evaluation of Advanced Mitigation Techniques for BWR Internals*. EPRI, Palo Alto, CA: 2006. 1009878.

58. *BWRVIP-190, Rev. 1: BWR Vessel and Internals Project, Vol. 1: BWR Water Chemistry Guidelines – Mandatory, Needed and Good Practice Guidance and Vol 2: BWR Water Chemistry Guidelines – Technical Basis*, EPRI, Palo Alto, CA: 2014. 3002002623.

59. *BWRVIP-195: BWR Vessel and Internals Project, Effect of Methanol on Electrochemical Corrosion Potential of Prefilmed NMCA Treated and Untreated Stainless Steel During Simulated BWR Startup Conditions*. EPRI, Palo Alto, CA: 2008. 1018115.

60. *BWRVIP-201: BWR Vessel and Internals Project, Technical Basis for Plant Specific Safety Evaluation for Hydrazine Injection for Startup ECP Reduction in BWRs.* EPRI, Palo Alto, CA: 2008. 1016581.

61. *BWRVIP-268: BWR Vessel and Internals Project: Electrochemical Corrosion Potential Sourcebook.* EPRI, Palo Alto, CA: 2012. 1025138.

62. *BWRVIP-296, Revision 2: BWR Vessel and Internals Project, Brunswick Unit 1 Lower Plenum ECP Analysis after Implementation of On-Line NobleChem (OLNC)*. EPRI, Palo Alto, CA: 2018. 3002014156.

63. *BWRVIP-304: BWR Vessel and Internals Project, Boiling Water Reactor Crack Mitigation Monitoring Results with Noble Metal Treatment.* EPRI, Palo Alto, CA: 2016. 3002009291.

64. *BWRVIP-219: Technical Basis for On-Line NobleChem™ Mitigation and Effectiveness Criteria for Inspection Relief*, EPRI, Palo Alto, CA: 2009. 1019071.

65. *BWRVIP-320: BWR Vessel and Internals Project Continuous On-Line NobleChem™ Feasibility Study*. EPRI, Palo Alto, CA: 2018. 3002013104.

66. *BWRVIP-225, Revision 1: BWR Vessel and Internals Project: BWR Shutdown and Startup Chemistry Experience and Application Sourcebook.* EPRI, Palo Alto, CA: 2013. 3002001942.

67. *BWRVIP-226: BWRVIP Vessel and Internals Project, Feasibility Evaluation of BWR Hydrogen Injection for ECP Reduction During Startup.* EPRI, Palo Alto, CA: 2009. 1019074.

68. *BWRVIP-249: BWR Vessel and Internals Project, Technical Basis for Plant-Specific Safety Evaluation for Early Hydrogen Water Chemistry in BWRs.* EPRI, Palo Alto, CA: 2012. 1024895.

69. *BWRVIP-261: BWR Vessel and Internals Project, Early Hydrogen Water Chemistry Demonstration.* EPRI, Palo Alto, CA: 2012. 1025137.

70. *BWRVIP-18, Rev. 2-A: BWR Vessel and Internals Project, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines*, EPRI, Palo Alto, CA: 2016. 3002008089.

71. *BWRVIP-42, Revision 1-A: BWR Vessel and Internals Project, Low Pressure Coolant Injection (LPCI) Coupling Inspection and Flaw Evaluation Guidelines.* EPRI, Palo Alto, CA: 2017. 3002010548.

72. *BWRVIP-152: BWR Vessel and Internals Project: NDE Development 2005.* EPRI, Palo Alto, CA: 2006. 1011697.

73. BWRVIP-168: BWR Vessel and Internals Project, Guidelines for Disposition of Inaccessible Core Spray Piping Welds in BWR Internals. EPRI, Palo Alto, CA: 2007. 1013390.

74. *TR-105696-R19 (BWRVIP-03) Revision 19: BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines.* EPRI, Palo Alto, CA: 2016. 3002008095.

75. *BWR Vessel and Internals Project: BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38).* EPRI, Palo Alto, CA: 1997. TR-108823.

76. *BWRVIP-269: BWR Vessel and Internals Project, Nondestructive Evaluation Development 2012.* EPRI, Palo Alto, CA: 2012. 1025143.

77. BWRVIP 2006-334, "BWRVIP Response to NRC Request for Additional Information on BWRVIP-104", July 12, 2006.

78. *BWRVIP-97, Revision 1: BWR Vessel and Internals Project, Guidelines for Performing Weld Repairs to Irradiated BWR Internals.* EPRI, Palo Alto, CA: 2015. 3002005568.

79. *BWRVIP-98: BWR Vessel and Internals Project, Technical Basis for Guidelines for Performing Weld Repairs to Irradiated BWR Internals.* EPRI, Palo Alto, CA: 2002. 1006385.

80. *BWRVIP-151: BWR Vessel and Internals Project, Technical Basis for Revision to BWRVIP- 97 Welding Guidelines.* EPRI, Palo Alto, CA: 2006. 1011692.

81. *BWRVIP-228: BWR Vessel and Internals Project, A Computational Modeling Tool for Welding Repair of Irradiated Materials.* EPRI, Palo Alto, CA: 2009. 1019066.

82. *Welding and Repair Technology Center: Status Report for the Light Water Reactor Sustainability Collaborative Research Program on Irradiated Materials Weldability*. EPRI, Palo Alto, CA: 2019. 3002015849.

83. *BWRVIP-84, Revision 3: BWRVIP Vessel and Internals Project, Guidelines for Selection and Use of Materials for Repairs to BWR Internal Components*, EPRI, Palo Alto, CA: 2017. 3002010552.

84. *BWRVIP-229: BWR Vessel and Internals Project, Alloy X-750 Replacement Roadmap*. EPRI, Palo Alto, CA: 2009. 1019064.

85. *Critical Issues Report and Roadmap for the Advanced Radiation-Resistant Materials Program*. EPRI, Palo Alto, CA: 2012. 1026482.

86. *Materials Handbook for Nuclear Plant Pressure Boundary Applications (2015)*. EPRI, Palo Alto, CA: 2015. 3002005470.

87. *Welding and Repair Technology Center: Underwater Laser Welding Studies*. EPRI, Palo Alto, CA: 2009. 1019167.

88. *Welding Repair Technology Center: Welding Irradiated Material*. EPRI, Palo Alto, CA: 2010. 1021174.

89. *Welding and Repair Technology Center: Advanced Underwater Laser Welding Technology for Irradiated Reactor Internals*. EPRI, Palo Alto, CA: 2016. 3002005553.

90. *Welding and Repair Technology Center: Development of Low Heat Input Laser Beam Welding for Repair of Irradiated Reactor Components*. EPRI, Palo Alto, CA: 2014. 3002003146.

91. *BWRVIP-94NP, Revision 3: BWR Vessel and Internals Project, Program Implementation Guide*. EPRI, Palo Alto, CA: 2018. 30021013101.

92. *BWRVIP-86, Revision 1-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan*. EPRI, Palo Alto, CA: 2012. 1025144.

93. *BWRVIP-135, Revision 3: BWR Vessel and Internals Project: Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations*. EPRI, Palo Alto, CA: 2014. 3002003144.

94. Title 10, United States Code of Federal Regulations, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." U.S. Nuclear Regulatory Commission: January 2008.

95. *BWRVIP-321: Boiling Water Reactor Vessel and Internals Project: Plan for Extension of the BWR Integrated Surveillance Program (ISP) Through the Second License Renewal (SLR)*. EPRI, Palo Alto, CA: 2018. 3002013097.

96. *BWRVIP-114-A: BWR Vessel and Internals Project RAMA Fluence Methodology Theory Manual*. EPRI, Palo Alto, CA: 2009. 1019049.

97. *BWRVIP-115-A: BWR Vessel and Internals Project RAMA Fluence Methodology Benchmark Manual – Evaluation of Regulatory Guide 1.190 Benchmark Problems*. EPRI, Palo Alto, CA: 2009. 1019050.

98. *BWRVIP-126, Revision 2: BWR Vessel and Internals Project, RAMA Fluence Methodology Version 1.20 Linux-Intel.* EPRI, Palo Alto, CA: 2010. Software 1020240.

99. *BWRVIP-157: BWR Vessel and Internals Project, Evaluation of RAMA Thermal Neutron Fluence Predictions.* EPRI, Palo Alto, CA: 2006. 1013388.

100. *BWRVIP-189: BWR Vessel and Internals Projects, Evaluation of RAMA Fluence Methodology Calculation Uncertainty.* EPRI, Palo Alto, CA: 2008. 1016938.

101. *BWRVIP-209: BWR Vessel and Internals Project, Data Requirements Specification for Constructing RAMA Fluence Models.* EPRI, Palo Alto, CA: 2009. 1018607.

102. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission: March 2001.

A
REVISION LOG

Table A-1 provides a summary of revisions for each version of the BWR IMT.

Table A-1
Revision Log

Revision	Summary of Changes
Revision 0 (2007)	Initial issue.
Revision 1 (2008)	Important changes to R&D Gaps in Revision 1 are summarized. 1. The following gaps were closed: B-DM-04, B-DM-05, B-AS-04, B-AS-06, B-AS-08, B-AS-21, B-AS-23, B-AS-24, B-AS-25, B-I&E-04, B-RR-01, B-RR-04, B-RG-02, B-RG-03 2. The following new gaps were added: B-DM-06, B-AS-26, B-AS-27, B-AS-28, B-I&E-08, B-I&E-09, B-RR-05, B-RR-06, B-RR-07, B-RR-08, B-RR-09, B-RG-05, B-RG-06, B-RG-07 3. A category for program requirements was added. The following gaps were effectively closed and moved into the requirements category: B-AS-01, B-RG-04 4. Gap B-MT-03 was merged into B-AS-09. A number of minor changes to the IMTs and formatting changes are also included in Revision 1. See BWRVIP-167, Rev. 1 (EPRI 1018111) for details.
Revision 1.1 (2009)	Based on completed work and updated prioritization by the BWRVIP Integration Committee, an interim revision to IMT gaps was made in mid-2009 (referred to as Revision 1.1). Revision 1.1 of the BWR IMTs refers to BWRVIP Correspondence 2009-216, "Interim Revision to BWR Issue Management Tables to Include New, Revised, and Re-Ranked R&D Gaps." This correspondence added new R&D Gaps, re-ranked some R&D Gaps, and revised the content of some R&D Gaps. The BWR IMT Report was not re-published in 2009 to reflect these changes. The important changes to R&D Gaps from Revision 1 of the IMT Report are summarized in items 1 through 6 below. 1. The following gaps were closed: B-AS-02, B-AS-03, B-AS-16, B-RR-03, B-RR-07, B-RG-07 2. Gap B-AS-15 was amended to remove Steam Dryer High Cycle Fatigue Evaluation Methodology issues. These steam dryer issues are now addressed separately in a new gap (B-AS-29). This change separates steam dryer high-cycle fatigue issues from other FIV concerns for reactor internals. 3. The following gaps changed from medium priority to high priority: B-DM-06, B-RR-02 4. The following gaps changed from high priority to medium priority: B-RG-05, B-AS-27, B-AS-28 *(Continued on next page.)*

Table A-1 (continued)
Revision Log

Revision	Summary of Changes
Revision 1.1 (2009) (continued)	5. The following gaps changed from low priority to medium priority: B-RR-08, B-AS-20 6. The following gap changed from medium priority to low priority: B-I&E-05
Revision 2 (2010)	Revision 2 of the BWR IMTs addresses consideration of long-term operations (LTO). A majority of long-term operation issues are associated with the 2010 MDM panel results. Consideration of LTO resulted in the identification of several new gaps and significantly affected a number of other gaps. Gaps created by or affected by LTO are indicated as such in Table 3-1 of BWRVIP-167, Revision 2. The most significant changes to R&D Gaps in Revision 2 are summarized below. 1. The following gaps were closed: B-DM-01, B-DM-02, B-AS-13, B-MT-06, B-RR-09, B-RG-01, B-RG-06 2. The following new gaps were added: B-DM-07, B-DM-08, B-DM-09, B-DM-10, B-AS-30, B-AS-31, B-AS-32, B-RG-08, B-RG-09 3. Gaps B-AS-09 and B-AS-10 have been restructured such that all crack growth rate data needs are addressed in B-AS-09 and fracture toughness data needs are addressed in B-AS-10. 4. Gap B-AS-26 has been modified to address all high strength alloy issues, for both unirradiated and irradiated applications. Outstanding issues associated with irradiated X-750 previously captured in gap B-AS-13 have been incorporated into this gap. 5. Gap B-AS-27 has been modified to address all Alloy 182 / creviced Alloy 600 SCC issues, for both unirradiated and irradiated applications. Outstanding issues associated with irradiated Alloy 182 weld metal previously captured in gap B-AS-13 have been incorporated into this gap.
Revision 3 (2013)	The most significant changes to R&D Gaps from Revision 2 are summarized below. 1. The following gaps were closed: B-DM-10, Long-Term Stress Stability B-AS-17, Evaluate Hidden Weld Locations B-AS-19, Assess In-Vessel Fastener Loosening B-I&E-05, Appendix VIII Compliance B-I&E-07, NDE Capability: RPV Bottom Head Drain Line B-RR-06, Repair Solutions for Bottom Head Drain Line Locations 2. The following new gaps were added: B-AS-33, Equivalent Margins Analyses for BWR Nozzles (Medium priority), B-RG-10, R.G. 1.161 and ASME Section XI Appendix K Stress Intensity Factor Equation Non-conservatisms (Low priority) 3. The following changes were made to gap priority: B-DM-06 was changed from high to medium B-AS-11 was changed from low to medium B-AS-22 was changed from low to medium B-AS-31 was changed from medium to low *(Continued on next page.)*

Table A-1 (continued)
Revision Log

Revision	Summary of Changes
Revision 3 (2013) (continued)	4. Many of the R&D gaps have been significantly revised to reflect the 2012 Materials Degradation Matrix (MDM) work, recent OE, and research results published since 2010. 5. Gap categorization related to LTO impact was improved based on the results of recent LTO-focused R&D assessments. The new categorization process includes groups for direct, indirect, and no LTO impact. The Appendix A Issue Management Tables were updated to reflect new and revised aging management guidance documents, to remove reference to closed gaps, and to add reference to new gaps. The Appendix A IMT for the Reactor Pressure Vessel was modified to include a line item addressing the use of Alloys 690 and 52 for repair of a leaking water level instrument penetration. A number of other changes to the BWR IMTs and formatting changes are also included in Revision 3.
Revision 3.1 (2013, 2014, and 2015)	Interim changes to the BWRVIP-167, Rev. 3 gaps were made in the December 2013, December 2014, and December 2015 BWRVIP Research Integration Committee (RIC) Meetings as follows: Note: These changes were documented within revisions of the BWRVIP strategic plan issued after the 2013, 2014, and 2015 BWRVIP RIC meetings. **Dec. 2013 RIC Meeting:** The priority of gap B-AS-29 was changed from High to Low **Dec 2014 RIC Meeting:** 1. The following new gaps were added: B-AS-34, Assess impact of shallow surface-breaking flaws on reactor vessel integrity during leak test (BWR Leak Test Issue), High Priority B-AS-35, Estimation of Initial Fracture Toughness Properties for Low Alloy Pressure Vessel Steels (NRC BTP 5-3 Issue), High Priority B-AS-35, Estimation of Initial Fracture Toughness Properties for Low Alloy Pressure Vessel Steels (NRC BTP 5-3 Issue), High Priority B-AS-36, Fluence Attenuation and Cavity Streaming Effects Outside the RV Beltline, Medium Priority B-I&E-10, UT of Group 3 Jet Pump Beams, Medium Priority B-I&E-11, UT of BWR/6 Top Guide Grid Beams, Low Priority 2. B-RG-08, Reactor Pressure Vessel Material Surveillance Program Implementation for 80-Year Service Lives, was set to closed since the approach at the time did not include separation of regulatory issues and technical gaps and this issue was already being addressed by gap B-AS-30. However, as of NOTE: For Revision 4, the related issue of regulatory acceptance of an ISP for SLR is now tracked as a regulatory issue in the regulatory issue matrix. See item 8 in the list of significant changes occurring in Revision 4 (Table A-1, row for Revision 4) which shows B-RG-08 as moved into the new regulatory issue matrix. 3. Changed gap B-AS-30, Material Surveillance Program Implementation for 80-Year Service Lives, from Medium to High priority. *(Continued on next page.)*

Table A-1 (continued)
Revision Log

Revision	Summary of Changes
Revision 3.1 (2013, 2014, and 2015) (continued)	**Dec 2015 RIC Meeting:** 1. The following new gaps were added· B-AS-37, Atypical Core Shroud Cracking, High Priority B-RQ-09, Support of International Members, Requirement 2. B-DM-10, Chloride Transient Effects on Stainless Steel and Nickel Alloy Crack Growth Rates was added as gap. However, the gap ID DM-10 had been previously used for a gap that was closed in Revision 2 (Long-Term Stress Stability). Instead of correcting this error through addition and gap ID renumbering in Revision 4, the intent of this gap has been captured within gap B-MT-05b, Water Chemistry Transient Management and this interim gap is treated as if it had never been added
Revision 4 (2020)	Revision 4 includes several significant changes to the BWR IMT structure and approach that are intended to improve the usability of IMT R&D gaps and the BWR IMT report overall. These changes include: 1. The IMT component tables included in Appendix A of prior revisions to the BWR IMT report have been removed in Revision 4. BWRVIP members can access a streamlined version of these tables on the BWRVIP cockpit website. A historical version of the IMT component tables is available to the public in BWRVIP-167, Revision 3, which can be downloaded from www.epri.com. 2. The definition of an R&D gap was revised to place focus on accurately capturing technical needs that potentially could have an impact on safe plant operation. 3. The range of operating approaches potentially envisioned were incorporated into gap content without the use of separate identifiers for these approaches (e.g., LTO, flexible operations). Gap resolutions should consider and address all of the possible variations. As a result, LTO impact identifiers were removed. 4. R&D gaps exclusively focused on resolving a regulatory issue for which the BWRVIP expends resources only to support member needs associated with regulation have been removed from the set of R&D gaps and relocated to a new regulatory issue matrix that is available to BWRVIP members on the BWRVIP cockpit website. 5. The I&E gap type was changed to inspection (IN) to better reflect that the gaps in this category focus on inspection technology / capability. 6. Previously, placeholders were included in the R&D gap tables to note closed gaps and to provide reference to the revision of the BWR IMT in which the gap was closed. These placeholders have been removed from the R&D gap results tables in Section 3. This function is now provided by a gap status tracking table within the Appendix A Revision Log. *(Continued on next page.)*

Table A-1 (continued)
Revision Log

Revision	Summary of Changes
Revision 4 (2020) (continued)	The most significant changes to R&D Gaps in Revision 4 are summarized below. 1. The following gaps were closed: B-DM-07, B-AS-05, B-AS-11, B-AS-12, B-AS-14, B-AS-22, B-AS-28, B-AS-29, B-AS-30, B-AS-32, B-AS-33, B-AS-34, B-AS-35, B-AS-36, B-IN-06, B-IN-08, B-IN-09, B-IN-10 [*Note: Some of these gap closures reflect transfer of the issue to the regulatory issues matrix, not that the issue(s) described in the gap are resolved. Bases for gap closure are documented in Appendix B. In addition, Revision 4 eliminates the RG gap category. See item 8 below for additional information regarding disposition of RG gaps.*] 2. The following new gaps were added: B-DM-11 – SCC Modeling / Prediction Capabilities / Asset Management Tools for Highly Irradiated Components, Medium Priority B-AS-38 – Reactor Internals Leakage Management, Medium Priority) B-IN-12 – Inspection of Access Hole Cover Welds, Low Priority 3. Gap B-MT-04 was separated into two gaps (B-MT-04a and B-MT-04b) to address OLNC effectiveness and OLNC implementation issues in separate gaps. 4. Gap B-MT-05 was separated into two gaps (B-MT-05a and B-MT-05b) to address water chemistry optimization for startup and shutdown separate from water chemistry transient management. 5. Gap B-IN-03 was separated into two gaps (B-IN-03a and B-IN-03b) to address NDE needs for core shroud welds separate from NDE needs for the shroud support. 6. Gap B-RR-02 was separated into two gaps (B-RR-02a and B-RR-02b) to address irradiated materials weldability data separate from development of analytical tools for weld repair of irradiated materials. The resulting gap for analytical tool development (B-RR-02a) was assigned a priority of medium (vs. high). 7. The following changes were made to gap priority: B-DM-06 – Priority reduced from Medium to Low B-AS-07 – Priority reduced from High to Medium B-AS-09 – Priority increased from Medium to High B-AS-15 – Priority reduced from Medium to Low B-AS-18 – Priority reduced from High to Medium B-AS-27 – Priority increased from Medium to High B-AS-37 – Priority reduced from High to Medium B-IN-01 – Priority reduced from Medium to Low B-IN-02 – Priority reduced from Medium to Low B-IN-10 – Priority reduced from Medium to Low B-MT-01 – Priority reduced from Medium to Low B-MT-02 – Priority reduced from High to Medium B-MT-05a / B-MT-05b – Priority reduced from High to Medium B-IN-03a / B-IN-03b – Priority reduced from High to Medium B-RR-05 – Priority reduced from High to Low *(Continued on next page.)*

Table A-1 (continued)
Revision Log

Revision	Summary of Changes
Revision 4 (2020) (continued)	8. Revision 4 eliminates the regulatory gap category. Open regulatory issues are now being tracked within the BWRVIP regulatory issue matrix. Regulatory gaps from IMT Revision 3 are dispositioned as follows: <u>Closed and Moved to Regulatory Issues Matrix (to track remaining open issues):</u> B-RG-05, Evaluation of Remote EVT-1 B-RG-08, RPV Material Surveillance Prog. Implementation for 80-Year Service Lives <u>Closed (no outstanding issues identified to be tracked in the Regulatory Issues Matrix)</u>: B-RG-09, Management of License Renewal Issues B-RG-10, R.G. 1.161 and ASME Section XI Appendix K Stress Intensity Factor Equation Non-conservatisms (See gap closure basis in Appendix B.) 9. BWRVIP program requirements for I&E guidance maintenance (B-RQ-06), NDE maintenance (B-RQ-07), and repair design criteria maintenance (B-RQ-08) were consolidated into a single requirement.

B
GAP CLOSURE

Table B-1 provides a summary list of the R&D gaps closed in Revision 4. Table B-2 provides the content of each closed R&D gap from Revision 3, the priority history for the gap, and a basis for gap.

Table B-1
List of Gaps Closed in IMT Revision 4

Gap ID	R&D Gap Title
B-DM-07	Chloride Transient Effects on Low Alloy Steel Crack Growth Rates
B-AS-05	Assess Neutron Dose Rate Effects on Embrittlement of C&LAS
B-AS-11	Assess Non BWR Reactor Irradiated Materials Data Applicability to the BWR Environment
B-AS-12	Thermal & Irradiation Embrittlement: Synergistic Effects (on CASS BWR Reactor Internals)
B-AS-14	Environmental Effects on Fatigue Resistance: Reactor Internals
B-AS-22	High-Cycle Thermal Fatigue: Piping Locations
B-AS-28	Impact of BWR Nozzle Penetrations on Pressure-Temperature Limit Curves
B-AS-29	Steam Dryer Evaluation Methodology
B-AS-30	Material Surveillance Program Implementation for 80-Year Service Lives
B-AS-32	Assessment of Core Plate Rim Hold Down Bolts
B-AS-33	Equivalent Margins Analysis for BWR Nozzles
B-AS-34	Assess Impact of Shallow Surface Breaking Flaws on Reactor Vessel Integrity
B-AS-35	Estimation of Initial Fracture Toughness of RPV Steels (BTP 5-3)
B-AS-36	Fluence Attenuation and Cavity Streaming Outside RPV Beltline
B-IN-06 [a]	NDE Capability: CASS Components
B-IN-08 [a]	Inspection and Evaluation Guidance for Repairs
B-IN-09 [a]	Examination Techniques for Detection of Loss of Preload in Reactor Internals Components
B-IN-10 [a]	Jet Pump Holddown Beam UT

Table B-1 (continued)
List of Gaps Closed in IMT Revision 4

Gap ID	R&D Gap Title
B-RG-05 [b]	Reactor Pressure Vessel Material Surveillance Program Implementation for 80-Year Service Lives
B-RG-08 [b], [c]	B-RG-08 - Reactor Pressure Vessel Material Surveillance Program Implementation for 80-Year Service Lives
B-RG-09 [d]	Management of License Renewal Issues
B-RG-10 [d]	R.G. 1.161 and ASME Section XI Appendix K Stress Intensity Factor Equation Non-conservatisms

[a] Previously Inspection (IN) gaps were designated as I&E gaps. Although the identifier has been changed to IN, sequential numbering was not reset

[b] The Regulatory Issue (RG) gap category was eliminated in Revision 4. RG gaps B-RG-05 and B-RG-08 have remaining open regulatory issues that are now tracked in a matrix of regulatory issues in the Library folder on the BWRVIP cockpit:
www.membercenter.epri.com > Program Cockpits > P41.01.03 > Library > Issue Management Tables (IMTs)

[c] Gap B-RG-08 was documented as closed as an outcome of the Dec. 2014 BWRVIP RIC Meeting. Since this Revision 4 is the first formal revision of the BWR IMTs since that decision, B-RG-08 is included in the list of closed gaps for Revision 4. However, the Intent of B-RG-08 is now tracked in the matrix of open regulatory issues indicated in note [b] above.

[d] Gaps B-RG-09 and B-RG-10 are considered to be closed and need not be tracked in the BWRVIP regulatory Issue matrix at the time Revision 4 was completed.

Table B-2
Gaps Closed In IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-DM-07 - Chloride Transient Effects on Low Alloy Steel Crack Growth Rates **GAP DESCRIPTION:** **Issue:** There is a need to better characterize the effect of chloride on low alloy steel crack growth rates. New data suggests that increased crack growth occurs at chloride concentrations below the EPRI BWR Water Chemistry Guideline action level 1 value of 5 ppb. **Description:** Recent testing performed at the GE Global Research Center indicates the potential for increased crack growth rates at chloride concentrations lower than 5 ppb. Previously, chloride concentrations below 5 ppb were not thought to be sufficient to significantly affect crack growth rates, even at high ECP. The EPRI BWR Water Chemistry Guidelines Action Level 1 for chlorides is 5 ppb. Additional research is needed to fully assess the effect of low chloride concentrations on crack growth. Resolution of this gap should include sufficient study of low chloride conditions < 5 ppb to determine if a change to the BWR Water Chemistry Guideline is warranted. **CLOSURE BASIS:** Available data are now considered sufficient to disposition the effect of chloride on LAS CGRs. A reduction in the EPRI BWR water chemistry action level 1 value for chloride from 5 ppb to 3 ppb has been implemented to address chloride effects. BWRVIP-233, Rev. 2 is the document providing resolution for this gap. **References** BWRVIP-306 [B-1], BWRVIP-233R2 [B-2], BWRVIP letter 2019-025 [B-3]	Rev. 4: CLOSED Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) [New in Rev. 2]

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-05 - Assess Neutron Dose Rate Effects on Embrittlement of C&LAS **GAP DESCRIPTION:** **Issue:** Additional evaluation is needed to assess the magnitude and significance of data indicating higher than predicted low alloy steel RPV material embrittlement rates in BWR vessels. **Description:** Current models appear to be under-predicting the amount of irradiation embrittlement occurring in BWR low-alloy steel reactor vessel materials. Data obtained from irradiation in BWR vessels show reductions in toughness than are predicted by data obtained from PWR or test reactor irradiations. It is hypothesized that there may be an effect of neutron flux that is not adequately addressed by existing embrittlement prediction models. Alternatively, lower BWR operating temperatures could be contributing to the increased loss of toughness. Regardless of the specific cause, there is a need to better characterize the magnitude of the effect for fluence values through plant end of life. Life extension to 80 years increases the need for embrittlement correlations that accurately account for the flux that occurs in BWRs, as prediction errors are magnified with increasing neutron fluence. **CLOSURE BASIS:** More recently developed embrittlement trend curves (ETCs), such as ASTM E900 and the ETC in 10CFR50.61a, have shown a much-improved ability to predict embrittlement behavior versus the ETC of Regulatory Guide 1,99, Revision 2. The prediction abilities of the recent ETCs are reliable enough that procedures are not provided for adjustment of the ETC predictions based on plant-specific surveillance data. Furthermore, in the derivation of the E900 ETC, the effect of neutron dose rate (flux) was found to be statistically insignificant. As such, the E900 ETC does not include flux as an input term. A regulatory issue has been created to address E900 not being an accepted ETC by regulatory authorities. **References** NRC R.G. 1.99 Rev. 2 [B-4] ASTM E900-15e1, Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, ASTM International, West Conshohocken, PA, 2015, www.astm.org, [B-5] 10CFR5061a, Alternate fracture toughness requirements for protection against pressurized thermal shock events. [B-6]	Rev. 4: CLOSED Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: Medium (2008) Rev. 0: Medium (2007)

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-11 - Assess Non BWR Reactor Irradiated Materials Data Applicability to the BWR Environment **GAP DESCRIPTION:** **Issue:** The usefulness of irradiated materials data obtained from accelerated tests in fast neutron reactors is limited in that direct applicability of the data to BWR conditions has not been definitively established. **Description:** A large body of radiation damage data exists from the fast and test reactor irradiation programs worldwide. To date, available data do not show any flux effect for austenitic stainless steel irradiated material properties. Recent Cooperative IASCC Research (CIR) Program test results indicate that irradiation in either a fast reactor or a BWR resulted in similar crack growth rate response. However, a possible exception was noted for the effect of corrosion potential where HWC reduced the CGR in the specimen irradiated in a fast reactor, but not in the specimen irradiated in a BWR. The difference may relate to K validity effects, but this effect needs to be confirmed before this gap can be closed. **CLOSURE BASIS:** Over the last several years, a significant amount of progress has been made with regard to generation of irradiated materials data. Many of these data have been generated through testing of materials harvested from decommissioned PWRs. Since PWRs have significantly higher neutron flux, testing using these materials provides data that easily envelope anticipated BWR end-of-life fluences. Current investigations into the use of test reactors to generate new data are focused on PWR applications. Given this current status, the use of irradiated materials data generated in test reactors to fill in data gaps relevant to BWRs is no longer considered needed. **References** MRP-211R1 [B-7]	Rev. 4: CLOSED Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Low (2010) Rev. 1.1: Low (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-12 - Thermal & Irradiation Embrittlement: Synergistic Effects (on CASS BWR Reactor Internals) **GAP DESCRIPTION:** **Issue:** The NRC has raised a potential concern related to thermal aging of CASS reactor vessel internals with focus on the potential synergistic effects of thermal aging and irradiation embrittlement. Thermal aging concerns are limited to castings meeting specific screening criteria in NUREG-1801 for casting method, molybdenum content, and ferrite content (as calculated by Hull's equivalent factor method specified in NUREG/CR-4513 Revision 1). Currently, this is primarily a regulatory issue associated with license renewal. Augmented inspection requirements, such as EVT-1, may be recommended for castings exceeding the NRC specified fluence threshold or meeting thermal embrittlement screening criteria contained in a May 19, 2000 letter from NRC to NEI. **Description:** The fluence threshold set by NUREG-1801 Section XI.M9 is $1x10^{17}$ n/cm^2 (E > 1.0 MeV). This threshold is much lower than fluence thresholds for onset of irradiation effects that have been identified in laboratory studies, which are on the order of $3x10^{20}$ n/cm^2 (E > 1.0 MeV). Concerns for BWR reactor internals are mitigated by relatively low operating temperatures and very limited use of high molybdenum castings (CF8M). BWRVIP-234 documents the results of a comprehensive assessment to address embrittlement concerns for 60-year operation using the criteria contained in NUREG-1801. Based on this study, the end-of-life fluence levels for the orificed fuel support, the jet pump assembly castings and LPCI couplings exceed the assumed fluence threshold, but the toughness data for irradiated austenitic stainless steel show that these components will have sufficient fracture toughness at the end of license renewal period. Potential reductions in toughness (due to higher fluence) will be somewhat greater for 80-year operating lives. However, since thermal aging is a diffusion dependent process, further reductions in fracture toughness between 60 and 80 years are not likely to be significant at BWR operating temperatures. Additional thermal aging beyond 60 years is not considered to be a research need for BWRs. Regardless, this gap is being held open to highlight the need to revisit the BWRVIP-234 study with 80-year service life criteria.	Rev. 4: CLOSED Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) Rev. 2: Low (2010) Rev. 1.1: Low (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-12 - Thermal & Irradiation Embrittlement: Synergistic Effects (on CASS BWR Reactor Internals) *(Continued)* **CLOSURE BASIS:** BWRVIP-234-A provides a basis for resolution of this concern for accumulated neutron fluences up to $6x10^{20}$ n/cm^2 (E > 1.0 MeV). BWRVIP-315 documents that most BWRs will not have castings with fluence exceeding this threshold value through at least 80 years of operation, with the exception of the fuel support castings. Additionally, BWRVIP-315 includes generic evaluations of some CASS jet pump and LPCI coupling components to demonstrate that significant structural margins exist. These generic methods can be applied by plants as needed to disposition castings with fluence exceeding $6x10^{20}$ n/cm^2. Finally, although fuel support castings are known to have end of life fluences far in excess of $6x10^{20}$ n/cm^2, BWRVIP-315 includes an assessment of these castings that provides a reasonable technical basis for the adequacy of current activities to manage these castings. This topic is also addressed by BWRVIP Letter 2017-030. As a result, the technical issues associated with this topic are considered to be resolved. Remaining issues are primarily regulatory in nature and are documented in a regulatory issue. **References:** BWRVIP-234-A [B-8], BWRVIP-315 [B-9], BWRVIP Letter 2017-030 [B-10]	See previous page.

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-14 - Environmental Effects on Fatigue Resistance: Reactor Internals **GAP DESCRIPTION:** **Issue:** There is a need to assess reactor vessel internals fatigue with consideration of environmental effects on fatigue life. **Description:** For high fluence reactor internals, there are no data that can be used to assess the effect of neutron fluence on fatigue resistance. It is presently not clear if there is any effect of neutron irradiation beyond reductions in the endurance limit resulting from increased yield strength. However, a mitigating factor is that fatigue usage values for near core reactor internals experiencing high neutron fluence are quite low, generally below 0.1. However, the higher strength alloys currently in use in BWR shroud repairs (i.e., X-750 and XM-19) may have higher fatigue usage, especially when considering 80-year plant service lives. And, even without consideration of neutron effects, available data are all associated with low-alloy steel, 300-series austenitic stainless steels, and ordinary austenitic nickel alloys (Alloys 600, 82, and 182). There are no environmental fatigue test data for these higher strength alloys. Resolution of this gap may include assessment of the design criteria applied to the reactor internals, with consideration of appropriate reactor environmental factors and neutron fluence effects to ensure that sufficient fatigue margins exist to support an 80-year plant operating life. **CLOSURE BASIS:** Management of environmentally-assisted fatigue for BWR reactor internals is addressed through the BWRVIP program since SCC is the limiting mechanism for the internals and this mechanism is being adequately managed through periodic inspection and flaw evaluation. See BWRVIP-315, Appendix D. Further, in its Safety Evaluation on MRP-227, Rev. 1, the U.S. NRC staff make it clear that environmental fatigue need not be considered for reactor internals by clarifying that within NUREG-1800, Revision 2, "the guidance is clear that the effects of the reactor water environment only need to be addressed for ASME Code, Section III, Class 1 reactor coolant pressure boundary components". **References** BWRVIP-315 [B-9], U.S. NRC Final Safety Evaluation for MRP-227 Revision 1 [B-11]	Rev. 4: CLOSED Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) Rev. 2: Low (2010) Rev. 1.1: Low (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-22 - High-Cycle Thermal Fatigue: Piping Locations **GAP DESCRIPTION:** The PWR piping system has been found to be susceptible to high-cycle fatigue, depending on piping geometry and operating conditions. The two high-cycle fatigue types that have been identified are: Interaction of swirl flow with cold water in-leakage past normally closed valves in normally stagnant lines. This high cycle fatigue process has resulted in the fatigue cracking of PWR Residual Heat Removal branch connections. Cyclic penetration and retreat of swirl flow in the branch line, combined with heat transfer to the environment. No valve in-leakage is required. This high-cycle fatigue process has resulted in fatigue cracking of PWR reactor coolant loop drain lines. For the BWR design, studies have shown that while in-leakage events are not plausible, cyclic penetration and retreat of swirl flow in branch lines could potentially impact BWR branch lines. The primary location of concern would be stagnant down-horizontal oriented drain lines. The results of investigation into this issue are documented in BWRVIP-155. This report provides guidance for evaluation of stagnant down-horizontal oriented lines. BWRVIP-155 is adapted for BWRs from PWR guidance provided in MRP-146R1. Recent updates to the model equations contained in these MRP reports introduced a swirl factor. Efforts were limited to assessment of swirl factor for branch to main pipe diameter ratios < 0.35. This is adequate for PWR applications, but BWRs have branch to main pipe diameter ratios up to 0.80. Consequently, there is a need to address swirl factors exceeding 0.35 for BWRs Closure of this gap should include resolution of the outstanding modeling issues and development of a comprehensive tool for use by BWR utilities to assess thermal fatigue vulnerabilities. **CLOSURE BASIS:** Revision 1 of BWRVIP-155 provides a basis for extension of the applicability of the swirl factor from a branch to main pipe diameter ratio of 0.35 to a ratio of 0.8. This resolved the previously identified limitations of the MRP-146 model with respect to application to BWRs. Additionally, BWRVIP-155R1 includes NEI-03-08 "good practice" recommendations for identification and inspection of stagnant branch lines that may be susceptible to thermal fatigue. BWRVIP-196 Revision 1 provides "good practice" recommendations for identification and inspection of mixing tees that may be susceptible to thermal fatigue. **References:** BWRVIP-155R1 [B-12], BWRVIP-196R1 [B-13], BWRVIP letter 2018-143 [B-14], MRP-146R2 [B-15]	Rev. 4: CLOSED Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Low (2010) Rev. 1.1: Low (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-28 - Impact of BWR Nozzle Penetrations on Pressure-Temperature Limit Curves **GAP DESCRIPTION:** **Issue:** The methodology for addressing stress concentrations occurring at sharp corner instrument nozzles is not well defined. Methodologies have been approved for individual plants, but an industry-wide solution or revision of ASME B&PVC to address the issue does not exist. **Description:** In most BWR designs, there are 2 instrument nozzles located approximately 6 inches above the top of active fuel. Heretofore, the stress concentration effect occurring at these penetrations have not been specifically included in the determination of pressure-temperature curves required by 10CFR50 Appendix G using the methodology in ASME Section XI Appendix G because the predicted 40-year design life accumulated fluence at the instrument penetration was below the $1x10^{17}$ n/cm^2 (E > 1.0 MeV) fluence threshold specified in 10CFR50 Appendix H (above which materials must be evaluated for the effects of radiation damage). As plants continue to age and accumulate more fluence, and as methods have improved for calculating fluence, these penetrations are now predicted to exceed the $1x10^{17}$ n/cm2 (E > 1.0 MeV) fluence threshold. Furthermore, there are no explicit rules in the ASME Code for evaluating this geometry. A stress concentration occurs at these locations which could become limiting as compared to the peak fluence location. **CLOSURE BASIS:** The BWROG has developed methods for determining the pressure-temperature limits of sharp cornered instrument penetrations (nozzles). These methods are contained in a publicly available topical report that was approved by the NRC in 2013. Furthermore, methods for addressing nozzles have been incorporated into ASME Section XI, Appendix G starting with the 2013 Edition. These methods are applicable to BWR instrument penetrations. **References** BWROG-TP-11-022-A Rev. I-A [B-16], ASME Section XI, Appendix G [B-17]	Rev. 4: CLOSED Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) Rev. 1.1: Medium (2009) Rev. 1: High (2008) [New in Rev. 1]

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-29 - Steam Dryer Evaluation Methodology **GAP DESCRIPTION:** **Issue:** Industry operating experience has shown that high frequency acoustic pressure oscillations in the main steam header can cause increased loading on not only the steam dryer, but also on the entire steam path. **Description:** The BWRVIP has published various documents addressing steam dryer inspection & evaluation (BWRVIP-139-A) and integrity assessment to address power uprate (BWRVIP-182, and BWRVIP-194). Although there has been substantial progress in modeling the flow conditions, acoustic loadings, and resultant steam dryer assembly loadings causing the degradation, there are still some remaining items needing resolution. In particular, there are ongoing efforts to address NRC RAIs on BWRVIP-194. Open NRC questions include minimum stress margin, FEA sub-modeling approach, overall benchmarking of the acoustic circuit model, and applicability of the methodology to Nordic steam dryers (i.e., Westinghouse design). [1] There is also a need to review the power uprate integrity assessment data to determine if this information may be useful in evaluating FIV concerns in non- uprated units. [1] Discussion with NRC in 2012 indicates an unwillingness to issue a generic safety evaluation for BWRVIP-194 because BWRVIP-194 does not include analysis or benchmarking for nordic design dryers. **CLOSURE BASIS:** The BWRVIP has completed all planned work related to steam dryer integrity assessments. In 2015, the BWRVIP withdrew BWRVIP-194 from NRC review since economic conditions that favored power uprates no longer exist. Accordingly, no new Extended Power Uprates (EPUs), that would require use of the methods documented in BWRVIP-194, are being planned within the BWR fleet for the foreseeable future. If the issue of an approved evaluation method becomes relevant at some point in the future, it would be tracked as a regulatory issue, not an open technical gap. **References:** BWRVIP Letter 2015-041 [B-18]	Rev. 4: CLOSED Rev. 3.1: Low (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) [New in Rev. 1.1]

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-30 - Material Surveillance Program Implementation for 80-Year Service Lives **GAP DESCRIPTION:** **Issue:** Maintenance of the ISP is necessary to optimize testing requirements and prevent unnecessary capsule tests. This program adequately addresses operation through 60 years of operation. There is a need to revisit this program to in the context of 80-year operating lifetimes. **Description:** Program requirement B-RQ-02 addresses ongoing maintenance of the ISP. The ISP combines surveillance materials from across the BWR fleet, including the BWROG supplemental surveillance program (SSP), to provide a better representation of the limiting beltline materials for each plant, while simultaneously reducing the total number of capsules to be tested by the BWR fleet. The program was developed to address BWR fleet operation through 40- year operations and was approved by NRC in 2002. As soon as the NRC approved the BWRVIP ISP for the original BWR operating license period, the BWRVIP began development of a plan to extend the ISP into the license renewal period. The updated ISP to address 60-year operations was approved by NRC in 2006. Specific assessment of the material surveillance data needed to meet BWR fleet needs for 80-year operations has not yet been performed. It may be that additional specimen materials based on capsule reconstitution and capsule re-insertion may be necessary. In contrast with some other long-term operations gaps, this issue needs attention relatively soon. Because BWRs lack substantial lead factors, if surveillance capsule reconstitution and re-insertion is needed, then near term action is likely needed to ensure relevant end of life data. **CLOSURE BASIS:** The feasibility of options for extending or replacing the ISP to address 80-year operations was evaluated in BWRVIP-295. A recommendation was made to extend the ISP to provide 80-year data by using a combination of already existing data and the reconstitution, irradiation, and testing of previously tested specimens. Data investigations to support this approach were performed and documented in BWRVIP-313. The program plan for extending the ISP was published in BWRVIP-321. NRC approval of the program plan, which is required before the program can be implemented, is addressed by an open regulatory issue in the BWRVIP regulatory issue matrix. **References** BWRVIP-295 [B-19], BWRVIP-313 [B-20], BWRVIP-321 [B-21]	Rev. 4: CLOSED Rev. 3.1: High (2013-2015) Rev. 3: Medium (2013) Rev. 2: Medium (2010) [New in Rev. 2]

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-32 - Assessment of Core Plate Rim Hold Down Bolts **GAP DESCRIPTION:** **Issue:** BWRVIP-25 requires inspection of core plate bolts when lateral support wedges are not installed. Efforts to develop effective NDE have not been successful to date (see gap B-I&E-01). Assessment to demonstrate inspection of core plate rim hold down bolts is not required is an alternative solution. **Description:** Although development of advanced NDE techniques may eventually make some meaningful inspection of core plate bolts possible, recent NDE efforts have not appeared promising. As a result, current BWRVIP efforts are focused on assessment to support elimination or relaxation of core plate bolt inspection requirements. In the past, the BWRVIP has undertaken generic analyses in an attempt to justify that no inspections are required. These analyses did not provide satisfactory results for several of the affected plants. This new BWRVIP study is taking a different approach by focusing on determining the minimum number of core plate bolts (with no preload assumed) in combination with the core plate aligner pins that are necessary to prevent unacceptable lateral displacement of the core plate and fuel assemblies. If the minimum number of core plate bolts is sufficiently low, this result can justify relaxation or deletion of the core plate bolt inspection requirement in BWRVIP-25. **CLOSURE BASIS:** BWRVIP-25, Rev. 1 includes the results of an engineering-based evaluation of core plate integrity. The results provide a technical basis for the elimination of core plate holddown bolt inspection requirements. The generic approach documented in BWRVIP-25, Rev. 1 represents a generic solution that is anticipated to be usable by most BWRs for up to an 80-yr licensed operating life. As such, this technical gap is considered to be closed. **References** BWRVIP-25R1 [B-22], BWRVIP-276 [B-23], BWRVIP-315 [B-9]	Rev. 4: CLOSED Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) [New in Rev. 2]

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-33 - Equivalent Margins Analysis for BWR Nozzles **GAP DESCRIPTION:** **Issue:** Operation beyond 40 years in some cases results in a need to evaluate additional materials, particularly nozzle forgings to demonstrate that the materials will retain adequate fracture toughness for 60-year operations. This occurs because the nozzles will exceed a fluence of $1x10^{17}$ n/cm^2 (E ≥ 1.0 MeV). and some may be limiting with regard to radiation damage since the material chemistry of these forgings was not tightly controlled and may not be well documented. **Description:** For a 60-year operating period, nozzles affected include water level instrumentation and some LPCI nozzles. To address the NRC concerns, fracture toughness evaluations to demonstrate the integrity of these "extended beltline" materials are necessary. These components must meet the requirements of 10CFR50 Appendix G and must maintain a Charpy USE of at least 50 ft-lb (68 J) throughout the life of the vessel. USE evaluations are performed using RG 1.99 methods; these methods require initial USE and chemistry values (%Cu). However, nozzle forging materials were not planned to be within the beltline region at the time of design and fabrication. As such, these nozzle forgings typically did not include rigorous chemistry controls nor detailed CMTR information. The result is that, for some units, these nozzles may become a controlling material for RPV pressure-temperature limits or pressure test temperatures. This issue does not constitute a significant challenge to safety and current expert opinion is that sufficient data exist to demonstrate integrity for 60 year operations. However, fleet-wide evaluation is needed to confirm this result. For LTO, the analysis would need to be extended to bound the increased fluence. Also, it is not presently clear if additional nozzle forgings, such as the recirculation inlet nozzles, could also exceed $1x10^{17}$ n/cm^2. If so, then some evaluation would also be needed to address these nozzles. **CLOSURE BASIS:** Databases of initial upper shelf energy (USE) were reviewed for both SA 508, Cl. 1 and Cl. 2 forgings. Because there were no requirements to consider USE drop due to neutron fluence for nozzle forgings, many of the initial USE values reported, while satisfying Code and regulatory minimums, are not indicative of upper shelf properties. Screening of the data to eliminate low shear values and statistical evaluation of the remaining high shear initial USE data indicates that there is significant margin to Code and regulatory minimums, such that no generic gap exists.	Rev. 4: CLOSED Rev. 3.1: Medium (2013-2015) Rev. 3: Medium (2013) [New in Rev. 3]

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-34 – Assess Impact of Shallow Surface Breaking Flaws on Reactor Vessel Integrity (Leak Test) **GAP DESCRIPTION:** The frequency of reactor pressure vessel failure must be evaluated under leak test conditions and the presence of shallow surface-breaking flaws. If this frequency cannot be shown to be less than 1E-6 events per year, changes to the methodology for determination of leak test temperature, as identified in ASME Section XI Appendix G, may be recommended. **CLOSURE BASIS:** Work completed by EPRI demonstrates that there is no safety concern related to this issue for BWR RPVs. Remaining work is considered regulatory in nature and as such, this issue is tracked in the regulatory issues matrix.	Rev. 4: CLOSED Rev. 3.1: High (2013-2015) [New in Rev. 3.1]
B-AS-35 – Estimation of Initial Fracture Toughness of RPV Steels (BTP 5-3) **GAP DESCRIPTION:** Branch Technical Position 5-3 (BTP 5-3) provides estimation methods for determining a materials initial RT_{NDT} value in the absence of data required to define RT_{NDT} per the ASME Code. The BTP 5-3 methods, along with an additional method developed by GE, have been used extensively by the fleet and have been determined to be potentially non-conservative. The uncertainty associated with the use of these methods needs to be determined so it can be factored into plant specific reactor vessel integrity evaluations. **CLOSURE BASIS:** Based on work documented in BWRVIP-287, it was concluded that values of initial RT_{NDT} and USE based on the methods documented in BTP 5-3 (Revision 2, March 2007) (or the equivalent in MTEB 5-2) are acceptable for continued use in demonstrating compliance with the requirements of 10 CFR 50, Appendix G. Furthermore, the NRC issued a closure memorandum on this subject (ADAMS Accession Number ML16364A285). **References:** BWRVIP-287: [B-24], NRC Memorandum Dated April 11, 2017 [B-25]	Rev. 4: CLOSED Rev. 3.1: High (2013-2015) [New in Rev. 3.1]

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-AS-36 – Fluence Attenuation and Cavity Streaming Outside RPV Beltline **GAP DESCRIPTION:** The correlation in RG 1.99 R2 for fluence attenuation through the thickness of the reactor vessel is based upon attenuation studies for the RV adjacent to the core. Recent studies have shown that due to neutron streaming in the cavity or annulus between the vessel and the bioshield, the attenuation in regions above and below the core, particularly in areas of nozzles, is not accurately predicted by the RG 1.99R2 correlation. Methods for prediction of attenuation outside of the core region are needed along with recommendations for their use in RV integrity calculations. **CLOSURE BASIS:** Methods exist for plant-specific determination of attenuation in the regions above and below the reactor core. However, regulatory guidance is still lacking as to how or whether this should be addressed in RV integrity calculations. As such, this issue is regulatory in nature and will be tracked within the regulatory issue matrix.	Rev. 4: CLOSED Rev. 3.1: High (2013-2015) [New in Rev. 3.1]
B-IN-06 - NDE Capability: CASS Components **GAP DESCRIPTION:** **Issue:** Detection of flaws within cast austenitic stainless steel components continues to represent a significant challenge for the industry. **Description:** Field examination of CASS materials using ultrasonic examination techniques are characterized by high attenuation and scattering of the acoustic energy due to coarse grain structure, and variety of grain structures. As a result, inservice inspection techniques currently available for examination of CASS piping are not effective at identifying and sizing indications. There are inspection challenges specific to CASS BWR reactor internals that differ from the generic inspection issues related to examination of external CASS components. The BWR internals specific inspection issues relates to challenging configuration and access constraints that would require the use of small ultrasonic probes. Small ultrasonic probes are inherently ineffective at examining CASS. CASS NDE capability is a long-range objective of the EPRI NDE C. In addition to NDE development, other planned work includes assessment of casting microstructure using ultrasonic investigation to assess which castings have microstructures susceptible to more rapid crack growth.	Rev. 4: CLOSED Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) Rev. 2: Low (2010) Rev. 1.1: Low (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-IN-06 - NDE Capability: CASS Components *(continued)* The importance of this gap is reduced (as compared with the need in PWR designs) by a number of factors. The use of CASS is more limited in BWR designs than in PWR designs and does not include any pressure boundary piping and fittings. Due to lower operating temperatures and lower neutron dose, the expected thermal and irradiation embrittlement is substantially lower than for PWRs. Finally, BWRVIP-234-A documents an evaluation of CASS components that concludes no additional evaluation of CASS components is needed so long as accumulated fluence remains less than $6x10^{20}$ n/cm^2 (E > 1.0 MeV). **CLOSURE BASIS:** Assessment IMT gap B-AS-12 was closed based on completion of BWRVIP-234-A and BWRVIP-315. These reports conclude that no inspection or further evaluation is required for most BWR CASS internals. Further, should a need for augmented inspection arise, it is noted that visual examination (EVT-1) provides a suitable method for examining flaw tolerant reactor internals components. **References** BWRVIP-234-A [B-8], BWRVIP-315 [B-9]	See previous page.
B-IN-08 - Inspection and Evaluation Guidance for Repairs **GAP DESCRIPTION:** **Issue:** Current BWRVIP guidance regarding inspection of repair hardware is provided only for the core shroud tie rods (BWRVIP-76). For other repair hardware, the I&E guidelines state that the inspection criteria should be obtained from the hardware vendor. Generic criteria to guide the utilities in developing an appropriate inspection program for repaired components do not currently exist. Description: Some repairs use threaded connections that present unique challenges for inspection and overall material condition assessment. Other repairs are pre- loaded and verification of pre-load following some years of service might be required.	Rev. 4: CLOSED Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) Rev. 2: Low (2010) Rev. 1.1: Low (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)

Table B-2 (continued)
Gaps Closed In IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-IN-08 - Inspection and Evaluation Guidance for Repairs *(continued)* Resolution of this gap includes development of generic aging management and inspection guidance for repairs to ensure inspectability of repair hardware and consistent specification of inspection requirements. Resulting technical data should include the following areas: Summary of repairs installed 1. Review of inspections performed to date 2. Materials degradation assessment 3. General inspection guidance following installation of a repair 4. Recommended revisions to BWRVIP-84 The importance of guidance to address long term management of repair hardware is increased by consideration of 80-year service lives. Extended service lives for repair hardware increase the likelihood of some degradation mechanisms. SCC may be a concern for long term service of X-750 and XM-19 (also see gap B-AS-26). Irradiation enhanced stress relaxation is an increased concern for shroud tie rod repairs. **CLOSURE BASIS:** This gap is being closed on the basis that it has been assigned a low priority for 10 years and a critical need to develop generic criteria for inspection of repair hardware has not emerged. Furthermore, the potential for developing useful generic criteria for repair hardware is questionable considering that repair hardware design details are generally treated as proprietary by designers. Finally, BWRVIP-315 provides a set of generic principles that can be applied by owners to evaluate susceptibility of repair hardware to age-related degradation and to establish appropriate aging management requirements. **References** BWRVIP-315 [B-9]	See previous page.

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-IN-09 - Examination Techniques for Detection of Loss of Preload in Reactor Internals Components **GAP DESCRIPTION:** **Issue:** Currently, field ready technologies are not available to accurately assess the remaining joint pre-stress in BWR reactor internals bolted connections. **Description:** Maintenance of adequate preload can be an important parameter for some internals locations. Shroud repair hardware is a primary example of an installation associated with preload requirements for certain designs Additionally, assessment of bolt preload can be used as a parameter indicative of bolt condition (i.e., to ensure bolt integrity is maintained). The importance of ensuring adequate pre-load is maintained is increased by consideration of 80-year operations due to increased irradiation enhanced stress relaxation. EPRI 1019122 documents that detection of loss of bolted connection preload is possible. These tests, performed on Seabrook spare vessel internals, indicated that the vibration-acoustic response of the bolts that are free from creep damage or cracks is similar throughout the vessel, independent of the bolt position and that the vibration-acoustic technique was found to effectively characterize the no-load condition of the bolt when compared to bolts in the loaded stress state. In addition, engineering design of a submersible system for deployment during reactor in-service examinations was completed. However, this technology would need to be adapted to BWR applications. **CLOSURE BASIS:** Based on resolution of gap B-AS-32 for management of core plate holddown bolts and the lack of any clearly identified need for measurement of fastener preload using NDE, this gap is being closed. Although component preload is recognized to be important for some repair / stabilization hardware, loss of preload can be managed either through refined analyses or reestablishing fastener tension. **References** BWRVIP-25, Rev. 1 [B-22], BWRVIP-315 [B-9]	Rev. 4: CLOSED Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013) Rev. 2: Low (2010) Rev. 1.1: Low (2009) Rev. 1: Low (2008) Rev. 0: Low (2007)

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-IN-10 – Jet Pump Holddown Beam UT **Issue:** The BWRVIP has not been able to obtain a Group 2 Hybrid jet pump beam for use in UT demonstrations. This lack of availability has created the scenario where inspection vendors have had to supply their own ratcheting hardware for demonstrations. In cases, a licensee's preferred inspection vendor has not been able to support a scheduled examination because of this limitation. **Description:** As a result of in-vessel repair/maintenance activities, standard Group 2 jet-pump beams have been sporadically replaced with Group 2 "Hybrid", or "Ratchet Style", jet-pump beams (GEH Part Number 177D4665). The hybrid configuration differs from the standard Group 2 configuration in that the newer hybrid configurations contain ratcheting locking plate and keeper mechanisms. The ratcheting hardware influences ultrasonic examination data by affecting the inspection tool position on top of the jet-pump beam surface. Since jet-pump beam examinations employ immersion ultrasonic examination techniques, even slight adjustments in the inspection tools intended height or angle can have a significant impact on the validity of the examination data. It has been shown that ultrasonic techniques demonstrated on standard Group 2 configurations can be ineffective at examining Group 2 Hybrid configurations. The BWRVIP has been attempting to purchase a Group 2 Hybrid jet-pump beam from the manufacturer without success. The lack of availability of an industry ratcheting mechanism for demonstrations has created the scenario where inspection vendors have had to supply their own ratcheting hardware. Only one inspection vendor has obtained access to this hardware, so only this single vendor has a demonstrated technique applicable to the Group 2 Hybrid configuration. **References** BWRVIP-03 (latest revision) [74] **CLOSURE BASIS:** This issue is not a fundamental knowledge gap. Rather, the issue is a vendor business decision that should be addressed / resolved outside of the IMT process.	Rev. 4: CLOSED Rev. 3.1: Medium (2013-2015) [Gap added in 2014 RIC Meeting.]

Table B-2 (continued)
Gaps Closed In IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-RG-05 – Evaluation of Remote EVT-1 **GAP DESCRIPTION:** **Issue:** There are NRC concerns that remote EVT-1 as currently implemented by the BWRVIP may not reliably detect tight IGSCC cracking. The BWRVIP maintains that these visual inspections are capable of reliably detecting cracks of significance to reactor operations. **Description:** EVT-1 makes use of high-resolution camera systems to routinely examine nuclear plant components, including core shrouds and other pressure vessel internals. Techniques and methods for performing EVT-1 examinations were developed in the field at the time of the first core shroud examinations (1993-1995 time frame). Subsequently, use of EVT-1 was expanded to many additional BWR reactor internals components to address emerging degradation concerns. As a result, EVT-1 did not have a formal laboratory-derived technical basis. The methods and techniques evolved as lessons were learned about the necessary camera distance, lighting, and surface-condition requirements. Increased attention on remote EVT-1 has resulted from NRC questions and assessments concerning its effectiveness. The Research division of the NRC has been evaluating the suitability of remote visual methods, specifically EVT-1, for detection of tight cracking in nuclear safety-related components. NUREG-6860, prepared by Pacific Northwest National Laboratory, expresses reservations about the reliability of EVT-1 for detection of tight cracking and makes recommendations for modifications to the technique and for further study. This study additionally concluded that the inspection reliability of the various VT systems, calibration standards, and procedures is not well characterized and that a performance demonstration initiative may be needed. The NUREG cites VT experiments performed by PNNL; Swedish studies of VT reliability and qualification; various references regarding general aspects of measuring and specifying visual systems' performance; and various studies that have measured the crack opening dimensions of IGSCC and other types of cracking. In response, EPRI NDE Center member utilities initiated a multi-year effort to evaluate EVT-1 and investigate the findings presented in NUREG/CR-6860. This work resulted in some tightening of the EVT-1 resolution, angle, and scan speed requirements in BWRVIP-03. EPRI research also found that care should be taken when selecting the equipment used for data acquisition and review. The entire system should be viewed as a whole, and that every step in the process should be evaluated. Inferior components were found to effectively degrade the performance of the entire system.	Rev. 4: CLOSED Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) Rev. 1.1: High (2009) Rev. 1: High (2008) Rev. 0: Low (2007)

Table B-2 (continued)
Gaps Closed In IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-RG-05 – Evaluation of Remote EVT-1 *(continued)* Progress has been made toward resolution of this issue and research intended to resolve remaining concerns is proceeding. However, this gap will remain open until NRC concerns related to remote VT implementation are resolved. The impact of not proactively completing the research necessary to resolve this issue could be an NRC mandate for a NDE qualification program similar to PDI, resulting in significant, ongoing cost. **CLOSURE BASIS:** Within the U.S., concerns regarding the adequacy have largely been resolved by completion of research that demonstrates the capability of remote EVT-1 to reliably detect cracks of engineering significance in reactor components. Remaining concerns are regulatory in nature and have been added to the BWRVIP regulatory issues matrix.	See previous page.
B-RG-08 - Reactor Pressure Vessel Material Surveillance Program Implementation for 80-Year Service Lives **GAP DESCRIPTION:** **Issue:** Maintenance of the ISP is necessary to optimize testing requirements and prevent unnecessary capsule tests. This program adequately addresses operation through 60 years of operation. There is a need to revisit this program to in the context of 80-year operating lifetimes. **Description:** Program requirement B-RQ-02 addresses ongoing maintenance of the ISP. The ISP combines surveillance materials from across the BWR fleet, including the BWROG supplemental surveillance program (SSP), to provide a better representation of the limiting beltline materials for each plant, while simultaneously reducing the total number of capsules to be tested by the BWR fleet. The program was developed to address BWR fleet operation through 40-year operations and was approved by NRC in 2002. As soon as the NRC approved the BWRVIP ISP for the original BWR operating license period, the BWRVIP began development of a plan to extend the ISP into the license renewal period. The updated ISP to address 60-year operations was approved by NRC in 2006.	Rev. 3.1: CLOSED Rev. 3: Medium Rev. 2: Medium [New in Rev. 2]

Table B-2 (continued)
Gaps Closed In IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-RG-08 - Reactor Pressure Vessel Material Surveillance Program Implementation for 80-Year Service Lives *(continued)* Specific assessment of the material surveillance data needed to meet BWR fleet needs for 80-year operations has not yet been performed. It may be that additional specimen materials based on capsule reconstitution and capsule reinsertion may be necessary. In contrast with some other long-term operations gaps, this issue needs attention relatively soon. Because BWRs lack substantial lead factors, if surveillance capsule reconstitution and re-insertion is needed, then near term action is likely needed to ensure relevant end of life data. [Note: Regulatory issues associated with 80-year RPV surveillance programs are addressed by gap B-RG-08.] **CLOSURE BASIS:** From the December 2014 RIC disposition: Although use of the ISP for 80-years will be needed for subsequent license renewal (SLR), the issue is not really a regulatory issue and need only be covered by gap B-AS-30. As of Revision 4, technical work supporting B-AS-30 is complete and the issue of NRC acceptance is being tracked in the BWRVIP regulatory issue matrix.	See previous page

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-RG-09 – Management of License Renewal Issues **GAP DESCRIPTION:** **Issue:** There are instances where NRC license renewal guidance contained in NUREG-1801R2 is inconsistent with BWRVIP guidance. In some cases, this results in the unnecessary expenditure of resources on low value inspections. **Description:** Examples of inconsistencies between the GALL Report and BWRVIP guidance include: Management of ASME Class 1 small bore piping (< NPS 4) to address SCC and high-cycle thermal fatigue: There are technical bases for management of SCC and high-cycle thermal fatigue of small bore piping without implementation of volumetric examinations. Visual (VT-2) examinations combined with management of high-cycle fatigue concerns consistent with BWRVIP-155 and BWRVIP-196 represent a technically sound approach. The GALL Report (in section XI.M35) recommends volumetric examination. While volumetric examination of some small bore piping segments could result from implementation of risk-informed ISI programs, generic implementation of volumetric inspections is not warranted. Management of CASS Reactor Internals: The GALL report recommends augmented inspection for CASS components exceeding 1x1017 n/cm2 (E > 1.0 MeV). This position conflicts with existing BWRVIP (and MRP) guidance for CASS and recent BWRVIP studies addressing CASS vulnerability to thermal and irradiation embrittlement (BWRVIP-234). Reference Updates: The BWRVIP is an active management program with periodic updates to important management guidance such as inspection & evaluation guidelines and water chemistry guidelines. GALL references to BWRVIP reports quickly become out of date once GALL is published. As a result, there is a need to work with the NRC to improve the manner in which BWRVIP references are cited in GALL so that applicants need not justify differences in revision level from those specified in the GALL report. **References:** NUREG-1801R2, BWRVIP-155, BWRVIP-196, BWRVIP-234 NRC SE Related to the Fourth 10-Year Interval ISI Program, Surry Power Station, Unit 2 - Virginia Electric and Power Company - Docket No. 50-281, U.S. Nuclear Regulatory Commission: August 2005. Hope Creek Generating Station LRA, Duane Arnold Energy Center LRA NRC SE Related to the License Renewal of the Susquehanna Steam Electric Station, Units 1 and 2	Rev. 4: CLOSED Rev. 3.1: High (2013-2015) Rev. 3: High (2013) Rev. 2: High (2010) [New in Rev. 2]

Table B-2 (continued)
Gaps Closed in IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-RG-09 – Management of License Renewal Issues *(continued)* **CLOSURE BASIS:** There are instances where NRC license renewal guidance contained in NUREG-1801R2 is inconsistent with BWRVIP guidance. In some cases, this results in the unnecessary expenditure of resources on low value inspections. The issues addressed in this gap are no longer considered to be relevant in the context of U.S. license renewals. The following paragraphs provide information supporting this conclusion. Management of ASME Class 1 small bore piping (< NPS 4) to address SCC and high-cycle thermal fatigue: With regard to aging management of small bore piping, requests for input from personnel involved in license renewal have not resulted in identification of this issue as a significant concern. In addition, the BWRVIP has recently completed an updated assessment of thermal fatigue (BWRVIP-155, Rev. 1) and has provided good practice guidance appropriate to address thermal fatigue concerns (see closure basis for assessment gap B-AS-22). Finally, where utilities have made decisions to inspect small bore locations using volumetric techniques, no issues have been identified with the application of UT techniques developed for the purpose of interrogating small bore welds. Management of CASS Reactor Internals: The NRC final SE on BWRVIP-234 provides a resolution of this issue for initial license renewal. So long as neutron fluence remains less than 6E+20 n/cm2, no augmented inspection of CASS reactor internals is required. Reference Updates: Since most BWR LRAs have now been approved for plants intending to operate beyond an initial license period. For the remaining plants, precedents have been set that can be followed. Therefore, this issue is considered to be resolved for initial license renewal. **References:** BWRVIP-155, Rev. 1 [B-12], BWRVIP-234-A [B-8]	See previous page.

Table B-2 (continued)
Gaps Closed In IMT Revision 4 – Gap Descriptions and Closure Bases

R&D Gap Description and Closure Basis	Gap Priority History
B-RG-10 – R.G. 1.161 and ASME Section XI Appendix K Stress Intensity Factor Equation Non-Conservatism **GAP DESCRIPTION:** **Issue:** Stress intensity factor equations in R.G. 1.161 and ASME Section XI Appendix K may be non-conservative for BWRs. **Description:** Stress intensity factor equations in R.G. 1.161 and ASME Section XI Appendix K were developed with the intent of addressing PWRs. The stress intensity equation generally has a form of $K=(F)(P)(R/t)(a^{0.5})$, where the geometry factor F depends on a/t and R/t. Development of solutions for F generally assumed PWR vessel dimensions (R/t = 10) and not BWR vessel dimensions (R/t = 20). Using F values based on R/t = 10 may not be conservative for R/t > 10. A generic solution to this issue using a methodology similar to that developed by the BWRVIP for core shrouds is possible. Since there is not a significant aging management concern identified for this issue, a Low gap priority is appropriate. **References** NRC Regulatory Guide 1.161, ASME Section XI Appendix K **CLOSURE BASIS:** The source of this gap comes from FEA work performed to assess equivalent margins for BWR RPV plates and welds. It was observed that the R.G. 1.161 and ASME Section XI Appendix K solutions were different. However, there is no information directly associating differences in R/t with the different results obtained. In addition, the solutions in R.G. 1.161 and ASME Section XI Appendix K are not applied routinely for BWRs and the FEA solutions are available if a need arises to perform a BWR equivalent margins analysis. Therefore, this gap is categorized as closed.	Rev. 4: CLOSED Rev. 3.1: Low (2013-2015) Rev. 3: Low (2013-2015) [New in Rev. 3]

B.1 References

B-1. *BWRVIP-306: BWR Vessel and Internals Project, Stress Corrosion Crack Growth Rates of Low Alloy Steels During Chloride Transients: Crack Activation and Arrest Times.* EPRI, Palo Alto, CA: 2017. 3002010544.

B-2. *BWRVIP-233, Rev. 2: Updated Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment.* EPRI, Palo Alto, CA: 2018. 3002013026.

B-3. BWRVIP Letter 2019-025, Transmittal of 2019 BWR Water Chemistry Guidelines Interim Guidance, March 25, 2019.

B-4. Regulatory Guide 1.99, Revision 2, Radiation Embrittlement of Reactor Vessel Materials, May 1988.

B-5. ASTM E900, Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials, ASTM International, West Conshohocken, PA, 2015.

B-6. 10CFR50.61a, Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.

B-7. *Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data–State of Knowledge (MRP-211, Revision 1).* EPRI, Palo Alto, CA: 2017. 3002010270.

B-8. *BWRVIP-234-A: BWR Vessel and Internal Project, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals.* EPRI, Palo Alto, CA: 2017. 3002010550.

B-9. *BWRVIP-315: BWR Vessel and Internals Project, BWR Reactor Internals Aging Management Evaluation for Extended Operations.* EPRI, Palo Alto, CA: 2019. 3002012535.

B-10. BWRVIP Letter 2017-030, BWRVIP Assessment of BWR Orificed Fuel Support Aging Management, February 23, 2017.

B-11. US. NRC, FINAL SAFETY EVALUATION FOR ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT MRP 227, REVISION 1, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATIONS GUIDELINE", April 29, 2019.

B-12. *BWRVIP-155, Rev 1: BWR Vessel and Internals Project: Evaluation of Thermal Fatigue Susceptibility in BWR Stagnant Branch Lines.* EPRI, Palo Alto, CA: 2018. 3002013098.

B-13. *BWRVIP-196, Rev 1: BWR Vessel and Internals Project: Assessment of Mixing Tee Thermal Fatigue Susceptibility in BWR Plants.* EPRI, Palo Alto, CA: 2018. 3002013099.

B-14. BWRVIP Letter 2018-143, Transmittal of Recently Published BWRVIP Reports – BWRVIP-155, Rev. 1 and BWRVIP-196, Rev. 1, November 27, 2018.

B-15. *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 2)*. EPRI, Palo Alto, CA: 2016. 3002007853.

B-16. BWROG-TP-11-022-A, Rev. 1-A, Pressure-Temperature Limits Report Methodology for Boiling Water Reactors, June 2013.

B-17. ASME Boiler & Pressure Vessel Code, Section XI, Rules for Inspection of Nuclear Power Plant Components, Division 1, Rules for Inspection and Testing of Components of Light-Water-Cooled Plants, July 2019. (Appendix G, Fracture Toughness Criteria for Protection Against Failure)

B-18. BWRVIP Letter 2015-041, Withdrawal of BWRVIP-194, "BWRVIP-194: BWR Vessel and Internals Project, Methodologies for Demonstrating Steam Dryer Integrity for Power Uprate," from NRC Review, April 20, 2015.

B-19. *BWRVIP-295: Boiling Water Reactor Vessel and Internals Project, Reactor Vessel Material Surveillance for U.S. BWRs During the Second License Renewal: Feasibility Evaluation*. EPRI, Palo Alto, CA: 2016. 3002007041.

B-20. *BWRVIP-313: BWR Vessel and Internals Project, Data Investigations to Support the ISP for SLR*. EPRI, Palo Alto, CA: 2017. 3002010409.

B-21. *BWRVIP-321: Boiling Water Reactor Vessel and Internals Project: Plan for Extension of the BWR Integrated Surveillance Program (ISP) Through the Second License Renewal (SLR)*. EPRI, Palo Alto, CA: 2018. 3002013097.

B-22. *BWRVIP-25, Rev. 1: BWR Core Plate Inspection and Flaw Evaluation Guidelines*, EPRI, Palo Alto, CA: 2015. 3002005594.

B-23. BWRVIP-276. BWRVIP-276: BWR Vessel and Internals Project, Evaluation to Justify Core Plate Bolt Inspection Elimination. EPRI, Palo Alto, CA: 2013. 3002000688.

B-24. *Assessment of the Use of NUREG-0800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels (MRP-401, BWRVIP-287)*. EPRI, Palo Alto, CA: 2015. 3002005348.

B-25. Closure Memorandum Supporting the Limited Revision of NUREG-0800 Branch Technical Position 5-3, "Fracture Toughness Requirements, US. Nuclear Regulatory Commission, D. Rudland, April 11, 2017.

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TRANSLATED TABLE OF CONTENTS

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Siedewasserreaktor-Issue-Management-Tabellen

3002018319

Abschlussbericht, Juni 2020

EPRI-Projektmanager
W. Lunceford

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Leitender Prüfer
W. Lunceford

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Dieser Bericht basiert auf dem folgenden zuvor veröffentlichten Bericht:

BWRVIP-167NP, Revision 3: SWR-Behälter- und Einbauten-Projekt: Siedewasserreaktor-Issue-Management-Tabellen. EPRI, Palo Alto, CA: 2013. 3002000690.

Diese Publikation ist ein Unternehmensdokument, das in der Literatur folgendermaßen zitiert werden sollte:

BWRVIP-167, Revision 4: SWR-Behälter- und Einbauten-Projekt: Siedewasserreaktor-Issue-Management-Tabellen. EPRI, Palo Alto, CA: 2020. 3002018319.

ABSTRAKT

Nukleare Versorgungsunternehmen sind nach wie vor mit Problemen konfrontiert, die mit der Verschlechterung der Druckbehälter und Reaktoreinbauten von Siedewasserreaktoren (SWR) sowie der Rohrleitungskomponenten der Klasse 1 der American Society of Mechanical Engineers (ASME) zusammenhängen. Der Bericht zur SWR-Issue-Management-Tabelle (IMT) zeigt wichtige Lücken im Wissensstand der Industrie über Materialverschlechterungs-Phänomene und die damit verbundenen Behandlungsmöglichkeiten für primäre Systemkomponenten von SWR auf. Die IMTs erfüllen die BWRVIP-Verpflichtung, einen Arbeitsplan zu entwickeln und zu pflegen, der strategische Fragen des Materialmanagements bewertet und sicherstellt, dass die Materialverschlechterung durch die Finanzierung geeigneter Forschung und Entwicklung (FuE) proaktiv angegangen wird.

Ein umfassendes, integriertes Verständnis von Materialproblemen und Optionen für deren Management ist eine grundlegende Überlegung, um einen weiterhin sicheren Betrieb sowie die Entwicklung des gesamten Anlagengeschäfts und der Betriebsstrategien zu gewährleisten. Die in diesem Bericht aufgezeigten FuE-Lücken wurden ursprünglich als Mittel entwickelt, um die Absicht der NEI 03-08-Materialinitiative zu erfüllen, und die Ergebnisse werden regelmäßig aktualisiert, um sowohl die Anforderungen an die Verwaltung der Anlagen von Versorgungsunternehmen als auch die Änderungen des Wissensstandes der Industrie in Bezug auf die Materialverschlechterung in SWR-Primärsystemen zu berücksichtigen. Die Ergebnisse der Lückenbewertung sind ein entscheidendes Element, um sicherzustellen, dass die FuE-Strategien und -Prioritäten des BWRVIP-Programms weiterhin die Bedürfnisse der BWRVIP-Mitglieder erfüllen.

Schlüsselwörter

SWR-Behälter- und Einbauten-Projekt
Issue-Management-Tabellen
NEI 03-08-Materialinitiative
FuE-Lücken

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PRIMÄRE ZIELGRUPPE: Berater für Versorgungsunternehmen entweder in den Ausschüssen fürs SWR-Behälter- und Einbauten-Projekt (BWRVIP) oder Verantwortliche für die Bereitstellung von Input zu den Prioritäten und der strategischen Planung der Forschung und Entwicklung (FuE) des BWRVIP

WICHTIGE FORSCHUNGSFRAGE

Die SWR-Issue-Management-Tabelle (IMT) ist eine Bewertung, durch die die wichtigsten FuE-Lücken im Hinblick auf eine proaktive Behandlung der Verschlechterung von SWR-Material erkannt und priorisiert werden sollen. In dieser Bewertung ist eine F&E-*Lücke* als ein Forschungsbereich definiert, der für das Erreichen eines vernünftigen Vertrauensstandards als wichtig ermittelt wurde, um die Verschlechterung der primären Systemkomponenten so handhaben zu können, dass die Komponenten für die restliche Lebensdauer der Anlage betriebsfähig bleiben und ihre vorgesehenen Funktionen erfüllen können.

Die Lücken aus früheren Versionen der SWR-IMTs wurden überprüft, um den Fortschritt bei der Schließung der Lücken sowie allgemeine Wissensänderungen aufgrund abgeschlossener Forschung und Entwicklung bzw. neuer Betriebserfahrungen zu bewerten. In Fällen, in denen abgeschlossene FuE darauf hindeutete, dass eine Lücke geschlossen werden kann, werden die Grundlagen und Gründe für die Schließung der Lücke dokumentiert. Felderfahrungen, die für die Materialverschlechterung potenziell relevant sind, werden überprüft, um neue Lücken zu ermitteln. Die daraus resultierenden FuE-Lücken werden in Kategorien mit hoher, mittlerer und niedriger Priorität eingeteilt.

WICHTIGSTE ERKENNTNISSE

- Zu den wichtigsten FuE-Bereichen gehören die Bewertung des Spannungsrisskorrosions-(SCC-)Risswachstums, die Behandlung hoch bestrahlter Komponenten sowie die effektive Umsetzung von SCC-Minderungstechnologien auf Basis der Wasserchemie.
- Die unternommenen Bemühungen, um FuE-Lücken zu schließen, die sich in erster Linie auf regulatorische Fragen bezogen, führten zur Schließung einer beträchtlichen Anzahl von Lücken, wobei es sich bei den meisten dieser Lücken um Bewertungslücken (ASsessment) handelte.
- Obwohl es offene regulatorische Fragen im Zusammenhang mit der RDB-Integrität, verbleibende Verpflichtungen, die erfüllt werden müssen (d. h. Kapselprüfung und -bewertung im Rahmen des Integrated Surveillance Program (ISP)) und Möglichkeiten zur weiteren Optimierung der Methoden des RDB-Integritätsmanagements gibt, finden sich in den Ergebnissen der IMT-Revision 4 keine aktiven/offenen technischen FuE-Lücken im Zusammenhang mit der RDB-Integrität.
- Angesichts der beträchtlichen Zeitspanne zwischen den Revisionen 3 und 4 gab es eine Reihe von Lückenschließungen, die sich aus dem Abschluss einschlägiger FuE und Änderungen des Industriebedarfs (d. h. Veralterung des Problems) ergaben.
- Es wurden nur wenige neue Lücken erkannt. Dies steht im Einklang mit der Erwartung, dass mit der Wiederholung der IMT-Lückenbeurteilungen im Laufe der Zeit weniger Probleme erfasst werden, die in den anfänglichen Beurteilungen nicht erkannt wurden. Darüber hinaus ist zu beobachten, dass die Leistung von SWR-RDB und deren Einbauten in den letzten Jahren relativ stabil war. Es gab nur wenige wirklich einzigartige oder neue Probleme beim Alterungsmanagement.

- Welche sind nun die strategischen Lücken im Wissensstand in Bezug auf die proaktive Bewältigung der Materialverschlechterung in Siedewasserreaktoren (SWR)? Wie sollten diese Lücken priorisiert werden?

WARUM DAS WICHTIG IST

Die Ergebnisse der Lückenbewertung sind ein entscheidendes Element, um sicherzustellen, dass die FuE-Strategien und -Prioritäten des BWRVIP-Programms weiterhin die Bedürfnisse der BWRVIP-Mitglieder erfüllen. Die Lückenbewertung stellt außerdem sicher, dass die Versorgungsunternehmen der BWRVIP-Mitglieder in den USA weiterhin ihren Verpflichtungen zur Umsetzung der NEI 03-08-Materialinitiative nachkommen.

DIE ANWENDUNG DER ERGEBNISSE

Die Berater des BWRVIP-Programms sollten die FuE-Lücken verstehen, die sich aus dieser Bewertung ergeben, und bei der Bewertung der FuE-Pläne des BWRVIP-Programms und bei der Lieferung von Input für die Priorisierung von FuE-Projekten auch Prioritäten bei den FuE-Lücken berücksichtigen.

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PROGRAMM: Siedewasserreaktor-Behälter und -Einbauten (BWRVIP), P41.01.03

UMSETZUNGSKATEGORIE: KATEGORIE 1

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INHALT

ABBILDUNGSVERZEICHNIS

TABELLENVERZEICHNIS

BWRVIP-167、改訂 4 版：
BWR 容器と内部構造物プロジェクト

沸騰水型原子炉課題管理表

3002018319

最終 報告書　2020 年 6 月

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主任研究員
W. Lunceford

本報告書は、EPRI とその BWRVIP 参加メンバーが後援する研究について説明するものである。

本報告書は、以前に発表された以下の報告書に基づいている。

BWRVIP-167NP、改訂 3 版：BWR 容器と内部構造物プロジェクト：沸騰水型原子炉課題管理表。 EPRI, Palo Alto, CA: 2013. 3002000690.

本刊行物は、以下の形式によって引用されるべき企業文書である。

BWRVIP-167、改訂 4 版：BWR 容器と内部構造物プロジェクト：沸騰水型原子炉課題管理表。 EPRI, Palo Alto, CA: 2020. 3002018319.

要約

原子力事業者は、沸騰水型原子炉（BWR）の圧力容器、原子炉内部構造物、および米国機械学会（ASME）のクラス 1 配管構成部品の劣化に関連する問題に常に直面している。BWR 課題管理表（IMT）に関する報告書は、BWR 一次系構成部品の材料劣化現象や、それにまつわる管理機能に関連する業界知識の主なギャップを特定している。戦略的な材料管理の問題を評価し、適切な研究開発（R&D）への資金提供を通じて、材料の劣化を先に見越して確実に対処するための作業計画を開発および維持するという BWRVIP のコミットメントを満たす役割を IMT は果たしている。

材料問題および管理オプションを包括的かつ統合的に理解することは、継続的に安全な運転を保証するため、そして発電所全体の事業と運転戦略の開発において基本的な考慮事項である。本報告書で特定された一連の R&D ギャップは、NEI 03-08 材料イニシアチブの目的を満たす手段として当初開発された。その結果は、事業者資産管理のニーズと、BWR 一次系材料の劣化に関する業界知識の変化の両方に対応するために、定期的に更新されている。ギャップ評価の結果は、BWRVIP プログラムの R&D 戦略と優先事項が、BWRVIP メンバーのニーズを引き続き満たすために重要な要素であると言える。

キーワード

BWR 容器と内部構造物プロジェクト
課題管理表
NEI 03-08 材料イニシアチブ
R&D ギャップ

ELECTRIC POWER RESEARCH INSTITUTE

エグゼクティブサマリー

納品番号：3002018319

製品タイプ：技術報告書

製品タイトル：BWRVIP-167、改訂 4 版：BWR 容器と内部構造物プロジェクト：沸騰水型原子炉課題管理表

主な対象読者：BWR 容器と内部構造物プロジェクト（BWRVIP）委員会の事業者アドバイザー、または BWRVIP の研究開発（R&D）の優先順位と戦略的計画に関する情報提供を担当する事業者アドバイザー

主な研究課題

BWR 課題管理表（IMT）は、BWR 材料の劣化の予防的管理に関する主要な R&D ギャップを特定し、優先順位を付けることを目的としたギャップ評価となっている。この評価において、R&D ギャップは、合理的な信頼水準を達成するために重要であると認識された研究領域を明確に定めている。この信頼水準は、構成部品が実用可能であり、発電所の残りの寿命にわたって意図した機能を実行できるように、一次系構成部品の劣化を管理するためのものである。

以前のバージョンの BWR IMT ギャップは、ギャップの解消に向けた進捗状況、および完了した R&D または新しい運転経験から得た知識の全般的な変化を評価するために審査された。完了した R&D がギャップの解消を示した場合、ギャップ解消の基準と根拠が文書化されている。材料劣化に関連する可能性のある現場での経験を審査し、新しいギャップを特定する。結果として特定された一連の R&D ギャップは、高、中、低の優先度別に順位付けされている。

主な所見

- 主な R&D 領域には、応力腐食割れ（SCC）のクラック進展評価、高度に照射された構成部品の管理、および水化学に基づく SCC 緩和技術の効果的な実施が含まれる。
- 主に規制上の問題に関連する R&D ギャップを解消するための取り組みが行われた結果、相当数のギャップ（その大部分は評価（AS）ギャップ）が解消された。
- RPV の健全性に関連する規制上の問題でまだ未解決のもの、実行しなければならない残りのコミットメント（すなわち統合監視プログラム（ISP）カプセルの試験と評価）、および RPV の健全性管理方法をさらに最適化する方法などの課題は残ってはいるものの、RPV の健全性に関連するアクティブあるいは未解決の技術 R&D ギャップは、IMT 改訂 4 版の結果に含まれていない。
- 改訂 3 版と 4 版の間にはかなりの時間の開きがあったため、関連する R&D の完了と業界のニーズの変化（つまり、問題の陳腐化）により、多くのギャップが解消された。
- 新しいギャップはほとんど確認されなかった。これは、IMT ギャップ評価が時間の経過とともに繰り返されるため、最初の評価で特定されなかった問題が少なくなるという予測と合致している。さらに、BWR RPV および内部構造物の性能は近年比較的安定していることが観察されている。本当に独特な、または新しい経年管理の問題は限られている。
- 沸騰水型原子炉（BWR）の材料劣化の予防的管理に関する、知識の戦略的なギャップとは何か？これらのギャップはどのように優先順位を付けるべきか？

本文書の重要性

ギャップ評価の結果は、BWRVIP プログラムの R&D 戦略と優先事項が、BWRVIP メンバーのニーズを引き続き満たすために重要な要素である。ギャップ評価はさらに、米国の BWRVIP メンバーの事業者が NEI 03-08 材料イニシアチブを実施するというコミットメントの継続的な履行を確実にする。

研究結果の使い方

BWRVIP プログラムアドバイザーは、この評価の結果である R&D ギャップを理解し、BWRVIP プログラム R&D 計画を評価し、R&D プロジェクトの優先順位付けに関する情報を提供する際に、R&D ギャップの優先度を考慮しなければならない。

EPRI 連絡先： Wayne Lunceford (テクニカルエグゼクティブ)、walunceford@epri.com

プログラム：沸騰水型原子炉容器および内部構造物プログラム（BWRVIP）、P41.01.03

実施カテゴリー: カテゴリー1

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3002018319

Informe final, junio de 2020

Jefe de Proyecto de EPRI
W. Lunceford

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RECONOCIMIENTOS

Este documento ha sido preparado por el Instituto de Investigación de Energía Eléctrica (Electric Power Research Institute - EPRI).

Investigador principal
W. Lunceford

Este informe describe la investigación patrocinada por EPRI y sus miembros participantes en el proyecto de vasija y partes internas del reactor de agua en ebullición.

Este informe se basa en el siguiente informe previamente publicado:

BWRVIP-167NP, revisión 3: Tablas de gestión de problemas del reactor de agua en ebullición del proyecto de vasija y partes internas del reactor de agua en ebullición. EPRI, Palo Alto, CA (EE. UU.): 2013. 3002000690.

Esta publicación es un documento corporativo que debe ser citado en la documentación de la forma siguiente:

BWRVIP-167, revisión 4: Proyecto de vasija y partes internas del reactor de agua en ebullición: Tablas de gestión de problemas en el reactor de agua en ebullición. EPRI, Palo Alto, CA (EE. UU.): 2020. 3002018319.

RESUMEN

Las empresas de servicios del sector nuclear siguen enfrentándose a problemas relacionados con la degradación de las vasijas a presión y partes internas de los reactores de agua en ebullición (BWR) y los componentes de las tuberías clase 1 de la Sociedad Americana de Ingenieros Mecánicos (ASME). El informe sobre las tablas de gestión de problemas de los BWR identifica las principales deficiencias en los conocimientos actuales sobre el fenómeno de la degradación de los materiales y la capacidad de gestión correspondiente de los componentes del sistema del primario del BWR. La tabla de gestión de problemas cumple el compromiso del proyecto de vasija y partes internas del reactor de agua en ebullición de elaborar y mantener un plan de trabajo que evalúe los problemas de gestión de los materiales estratégicos y que garantice que la degradación de dichos materiales se afronte proactivamente mediante la financiación de una adecuada investigación y desarrollo.

El conocimiento exhaustivo e integrado de los problemas de los materiales y las opciones de gestión es una consideración fundamental para garantizar un funcionamiento seguro, así como el desarrollo del negocio general y las estrategias de funcionamiento de la central. El conjunto de deficiencias de la investigación y desarrollo identificado en este informe se elaboró inicialmente para satisfacer la intención de la iniciativa sobre materiales NEI 03-08 y los resultados se actualizan periódicamente para abordar las necesidades de gestión de activos de las empresas de servicios y los cambios en los conocimientos del sector en relación con la degradación de los materiales del sistema del primario del BWR. Los resultados de la evaluación de las deficiencias son un elemento crítico a la hora de garantizar que las estrategias y prioridades de la investigación y desarrollo del proyecto de vasija y partes internas del reactor de agua en ebullición siga satisfaciendo las necesidades de los miembros de dicho proyecto.

Palabras clave

Proyecto de vasija y partes internas del BWR
Tablas de gestión de problemas
Iniciativa de los materiales de NEI 03-08
Deficiencias en la investigación y desarrollo

ELECTRIC POWER RESEARCH INSTITUTE

RESUMEN EJECUTIVO

Número del documento para entregar: 3002018319

Tipo de producto: Informe técnico

Título del producto: BWRVIP-167, revisión 4: Proyecto de vasija y partes internas del reactor de agua en ebullición: Tablas de gestión de problemas en el reactor de agua en ebullición

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PÚBLICO PRINCIPAL: Los asesores de las empresas de servicios de los comités del proyecto de vasija y partes internas del reactor de agua en ebullición o responsables de proporcionar información en relación con la planificación estratégica y prioridades de investigación y desarrollo

FUNDAMENTO DE LA INVESTIGACIÓN

La tabla de gestión de problemas del BWR es una evaluación de deficiencias destinada a identificar y priorizar las principales deficiencias en la investigación y desarrollo sobre la gestión proactiva de la degradación de los materiales del BWR. En esta evaluación, se define la *deficiencia* en la investigación y desarrollo como un área de investigación considerada importante para lograr un estándar razonable de confianza en que la degradación de los componentes del sistema del primario pueda gestionarse de tal forma que dichos componentes sigan estando en buen estado y sean capaces de realizar las funciones previstas durante el resto de la vida útil de la central.

Las deficiencias de versiones anteriores de las tablas de gestión de problemas del BWR se revisaron para evaluar los progresos a la hora de subsanar dichas deficiencias, así como los cambios generales en los conocimientos derivados de una investigación y desarrollo finalizados o de las nuevas experiencias en el funcionamiento. En los casos en los que la investigación y desarrollo realizada indicó que podía darse por finalizada una deficiencia, se documentan las bases y fundamentos para dar dicha deficiencia por finalizada. Se revisa la experiencia en campo que puede ser importante para la degradación del material y así identificar nuevas deficiencias. El conjunto resultante de deficiencias en la investigación y desarrollo se prioriza en categorías de prioridad alta, media y baja.

CONCLUSIONES PRINCIPALES

- Entre las principales áreas de la investigación y desarrollo se incluye el crecimiento de las fisuras por corrosión bajo tensión, la gestión de componentes sometidos a alta irradiación y la aplicación eficaz de tecnologías de mitigación de la fisuración por corrosión bajo tensión basadas en la química del agua.
- El esfuerzo realizado para dar por concluidas las deficiencias en la investigación y desarrollo está relacionado, principalmente, con los problemas regulatorios derivados de la finalización de un número importante de deficiencias, donde una mayoría de ellas son deficiencias en la evaluación.
- Aunque hay problemas regulatorios actualmente existentes relacionados con la integridad de la vasija del reactor a presión, deben cumplirse los compromisos restantes (es decir, el ensayo y evaluación de la cápsula del programa de supervisión integrado) y las oportunidades para una mayor optimización de los métodos de gestión de la integridad de la vasija del reactor a presión, los resultados de la revisión 4 de las tablas de gestión de problemas no incluyen ninguna deficiencia en la investigación y desarrollo activa ni abierta relacionada con la integridad de la vasija del reactor a presión.

- Dada la importancia del margen de tiempo entre las revisiones 3 y 4, hay una serie de deficiencias finalizadas que se derivan de la finalización de la investigación y desarrollo correspondientes, así como los cambios en las necesidades del sector (como el problema de la obsolescencia).
- Se identificaron solo algunas nuevas deficiencias. Esto es coherente con la expectativa de que, a medida que las evaluaciones de las deficiencias realizadas por las tablas de gestión de problemas se repitan a lo largo del tiempo, se detecten menos problemas no identificados en las evaluaciones iniciales. Asimismo, se observó que el rendimiento de las partes internas y la vasija a presión del reactor de agua en ebullición ha sido relativamente estable en los últimos años. Ha habido una cantidad limitada de problemas de gestión del envejecimiento realmente únicos o nuevos.
- ¿Cuáles son las deficiencias estratégicas en el conocimiento actual en relación con la gestión proactiva de la degradación de los materiales del BWR? ¿Cómo deben priorizarse estas deficiencias?

POR QUÉ ESTO ES IMPORTANTE

Los resultados de la evaluación de las deficiencias son un elemento crítico a la hora de garantizar que las estrategias y prioridades de la investigación y desarrollo del proyecto de vasija y partes internas del reactor de agua en ebullición sigan satisfaciendo las necesidades de los miembros de dicho proyecto. La evaluación de las deficiencias garantiza además que las empresas de servicios miembros del proyecto de vasija y partes internas del reactor de agua en ebullición de EE. UU. sigan cumpliendo los compromisos para implementar la iniciativa de los materiales de NEI 03-08.

CÓMO USAR LOS RESULTADOS

Los asesores del programa del proyecto de vasija y partes internas del reactor de agua en ebullición deben conocer las deficiencias en la investigación y desarrollo derivadas de esta evaluación y deben tener en cuenta las prioridades de dichas deficiencias al evaluar los planes de investigación y desarrollo del programa del proyecto de vasija y partes internas del reactor de agua en ebullición, así como proporcionar información sobre la priorización de los proyectos de investigación y desarrollo.

CONTACTO DE EPRI: Wayne Lunceford, ejecutivo técnico, walunceford@epri.com

PROGRAMA: Programa de vasija y partes internas del reactor de agua en ebullición (BWRVIP), P41.01.03

CATEGORÍA DE APLICACIÓN: CATEGORÍA 1

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3002018319

Slutgiltig rapport, juni 2020

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W. Lunceford

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TACK

Den här rapporten har utfärdats av The Electric Power Research Institute (EPRI).

Huvudforskare
W. Lunceford

Den här rapporten beskriver forskning sponsrad av EPRI och dess deltagande BWRVIP-medlemmar.

Den här rapporten baseras på följande publicerade rapport:

BWRVIP-167NP, revision 3: Projekt om kärl och interna komponenter i kokvattenreaktorer: Problemhanteringstabeller för kokvattenreaktorer. EPRI, Palo Alto, CA: 2013. 3002000690.

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BWRVIP-167, revision 4: Projekt om kärl och interna komponenter i kokvattenreaktorer: Problemhanteringstabeller för kokvattenreaktorer. EPRI, Palo Alto, CA: 2020. 3002018319.

SAMMANDRAG

Kärnkraftsanläggningar har fortfarande problem relaterade till degradering av tryckkärl i kokvattenreaktorer (BWR:s), reaktorns interna komponenter och rörkomponenter med American Society of Mechanical Engineers (ASME) Klass 1. Rapporten med problemhanteringstabeller för kokvattenreaktorer identifierar viktiga luckor i branschens nuvarande kunskap om materialdegradering och associerade hanteringsförmågor för primära systemkomponenter i kokvattenreaktorer. Problemhanteringstabellerna tillfredsställer BWRVIP:s åtagande att utveckla och underhålla en arbetsplan som utvärderar strategiska materialhanteringsproblem och säkerställer att materialdegradering behandlas proaktivt genom finansiering av forskning och utveckling (FoU).

En omfattande och integrerad kunskap om materialproblem och hanteringsalternativ är ett fundamental beaktande för att säkerställa fortsatt säker drift samt utveckling av övergripande strategier för anläggningars verksamhet och drift. Uppsättningen med luckor i FoU i den här rapporten utvecklades initialt som ett sätt att möta syftet med NEI 03-08 Materials Initiative. Resultaten uppdateras periodvis för att behandla både hanteringsbehov för tillgångshantering och ändringar av branschens nuvarande kunskap om materialdegradering i primära system i kokvattenreaktorer. Resultaten från utvärderingen av luckor är en kritisk del för att säkerställa att BWRVIP-programmets strategier och prioriteter av FoU fortsätter att möta behoven hos BWRVIP-medlemmar.

Nyckelord

Projekt om kärl och interna komponenter i kokvattenreaktorer
Problemhanteringstabeller
NEI 03-08 Materials Initiative
FoU-luckor

ELECTRIC POWER RESEARCH INSTITUTE

SAMMANFATTNING

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PRIMÄR MÅLGRUPP: Anläggningsrådgivare antingen på kommittéer för projekt om BWR-kärl och interna komponenter (BWRVIP) eller ansvariga för att tillhandahålla underlag angående prioriteter av FoU för BWRVIP och strategisk planering

HUVUDSAKLIG FORSKNINGSFRÅGA

Problemhanteringstabellen för kokvattenreaktorer är en utvärdering av luckor som är avsedd att identifiera och prioritera viktiga luckor i FoU angående proaktiv hantering av degradering av material i kokvattenreaktorer. I den här utvärderingen är en *lucka* i FoU definierad som ett forskningsområdet. Detta identifieras som viktigt för att uppnå en rimlig förtroendestandard för att degraderingen av primära systemkomponenter ska kunna hanteras på ett sätt så att komponenter fortsätter att vara brukbara och kapabla att utföra sin avsedda funktion under resten av anläggningens livstids.

Luckor från tidigare versioner av problemhanteringstabellerna för kokvattenreaktorer granskades för att utvärdera om framsteg har gjorts för att täppa till luckor, samt att utvärdera allmänna ändringar i kunskap tack vare avslutad FoU eller ny driftserfarenhet. I fallen där avslutad FoU indikerade att en lucka hade täppts till dokumenterades underlagen och anledningarna till detta. Praktiskt erfarenhet som eventuellt är relevant för materialdegradering granskas för att identifiera nya luckor. Den resulterande uppsättningen med luckor i FoU prioriteras i kategorierna hög, medel och låg prioritet.

HUVUDSAKLIGA SLUTSATSER

- Viktiga FoU-områden inkluderar utvärdering av sprickbildning till följd av spänningskorrosion (SCC), hantering av spricktillväxt för högbestrålade komponenter och effektiv implementation av vattenkemibaserade åtgärdstekniker för SCC.
- Arbetet som vidtogs för att täppa till luckor i FoU primärt relaterade till regleringsfrågor, resulterade i att ett stort antal luckor täpptes till och en stor del av dessa luckor var luckor i utvärdering (AS).
- Resultaten för problemhanteringstabeller, revision 4, innehåller inga aktiva/öppna tekniska luckor i FoU relaterade till integriteten hos reaktortryckkärl, även om det finns öppna integritetsrelaterade regleringsfrågor angående reaktortryckkärl, utestående åtaganden som måste uppfyllas (dvs. Integrated Surveillance Program (ISP) test och utvärdering av hölje) och möjligheter till ytterligare optimering av hanteringsmetoder för integritet av reaktortryckkärl.
- På grund av den långa tiden mellan revision 3 och 4, täpptes ett antal luckor till tack vare avslutande av relevant FoU och ändringar i branschens behov (dvs. problemföråldring).
- Endast ett fåtal nya luckor identifierades. Det är förenligt med förhoppningen att färre problem fångas upp som inte differentierades initialt, när utvärderingar av luckor för problemhanteringstabeller upprepas med tiden. En ytterligare iakttagelse är att prestandan hos reaktortryckkärl och interna komponenter i kokvattenreaktorer har varit relativt stabila under de senaste åren. Det har funnits ett begränsat antal verkligt unika eller nya hanteringsproblem angående föråldring.
- Vilka är de strategiska luckorna i nuvarande kunskap angående proaktiv hantering av materialdegradering i kokvattenreaktorer (BWR)? Hur bör dessa luckor prioriteras?

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PÅ VILKET SÄTT HAR DETTA BETYDELSE

Resultaten från utvärderingen av luckor är en kritisk del för att säkerställa att BWRVIP-programmets strategier och prioriteter gällande FoU fortsätter att möta behoven hos BWRVIP-medlemmar. Utvärderingen av luckor säkerställer ytterligare att medlemsanläggningar i U.S. BWRVIP fortsätter att möta kraven att implementera NEI 03-08 Materials Initiative.

TILLÄMPA RESULTATEN

Rådgivare för BWRVIP-programmet bör förstå luckorna i FoU som har resulterat från den här utvärderingen och bör överväga prioriteter av luckor i FoU när planer för FoU från BWRVIP-programmet utvärderas och när underlag tillhandahålls för att prioritera FoU-projekt.

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PROGRAM: Program för kärl och interna komponenter i kokvattenreaktorer (BWRVIP), P41.01.03

GENOMFÖRANDEKATEGORI: Kategori 1

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