EPRI | ELECTRIC POWER RESEARCH INSTITUTE

2019-085 BWR Vessel & Internals Project (BWRVIP)

September 9, 2019

Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Attention: Joseph Holonich

Subject: Docket No. 99902016 – September 11, 2019 Closed Meeting with the NRC to discuss BWRVIP Plans for submittal of "BWRVIP-315: BWR Vessel and Internals Project, Reactor Internals Aging Management Evaluation for Extended Operations", EPRI Technical Report 3002012535 (Published July 2019)

Enclosed is a copy of the BWRVIP's presentation materials for the closed meeting to discuss BWRVIP plans for submittal of "BWRVIP-315: BWR Vessel and Internals Project, Reactor Internals Aging Management Evaluation for Extended Operations" scheduled for September 11, 2019.

Please note that the enclosed presentation materials contain proprietary information. A letter requesting that the presentation materials be withheld from public disclosure and an affidavit describing the basis for withholding this information are provided as Attachment 1.

A non-proprietary version of the BWRVIP's presentation materials is also enclosed. This nonproprietary response is identical to enclosed proprietary presentation materials except that the proprietary information has been deleted.

Also enclosed are the Open Session presentation materials.

Please notify Debbie Rouse (EPRI) at email drouse@epri.com or by telephone at 704-595-2520 if you have any receipt difficulties.

Sincerely,

Nathan Palm, EPRI, BWRVIP Program Manager

G004
NRR

Together . . . Shaping the Future of Electricity

ELECTRIC POWER RESEARCH INSTITUTE

Ref. Docket No. 99902016

September 9, 2019

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding of the following Proprietary Information Included in:

EPRI Presentation Materials for September 11, 2019 Closed Meeting with the NRC to Discuss BWRVIP Plans for submittal of "BWRVIP-315: BWR Vessel and Internals Project, Reactor Internals Aging Management Evaluation for Extended Operations", EPRI Technical Report 3002012535. (Published July 2019)

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the report identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified in the referenced report. Proprietary and non-proprietary versions of the presentation material and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Proprietary Information in confidence to assist the NRC review of the enclosed submittal to the NRC by EPRI. The Proprietary Information is not to be divulged to anyone outside of the NRC or to any of its contractors, nor shall any copies be made of the Proprietary Infomation provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (704) 595-2732. Questions on the content of the Report should be directed to Nathan Palm of EPRI at (724) 288-4043.

Sincerely,

Attachment(s)

Together . . . Shaping the Future of Electricity

1300 West W.T. Harris Boulevard, Charlotte, NC 28262-8550 USA • 704.595.2732 • Mobile 704.490.2653 • nwilmshurst@epri.com

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Information Included In:

EPRI Presentation Materials for September 11, 2019 Closed Meeting with the NRC to Discuss BWRVIP Plans for submittal of "BWRVIP-315: BWR Vessel and Internals Project, Reactor Internals Aging Management Evaluation for Extended Operations", EPRI Technical Report 3002012535. (Published July 2019)

I, Neil Wilmshurst, being duly sworn, depose and state as follows:

I am the Vice President and Chief Nuclear Officer at Electric Power Research Institute, Inc. whose principal office is located at 3420 Hillview Avenue, Palo Alto, California ("EPRI") and I have been specifically delegated responsibility for the above-listed Report that contains EPRI Proprietary Information that is sought under this Affidavit to be withheld "Proprietary Information". I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Proprietary Information on behalf of EPRI.

EPRI Proprietary Information is identified in the above referenced presentation materials by double brackets. Example of such identification is as follows:

[[This sentence is an example]]

Tables, figures, or graphics containing EPRI Proprietary Information are identified with double brackets before and after the object. In each case this affidavit is the basis for the proprietary determination.

EPRI requests that the Proprietary Information be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information (see e.g. 10 C.F.R. §2.390(a)(4)):

a. The Proprietary Information is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Proprietary Information do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Proprietary Information. The Proprietary Information is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Proprietary information contained therein to constitute trade secrets of EPRI. As such, EPRI holds the Information in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Information.

c. The information sought to be withheld is considered to be proprietary for the following reasons. EPRI made a substantial economic investment to develop the Proprietary Information, and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Proprietary Information. If the Proprietary Information

were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Proprietary Information for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Proprietary Information.

d. EPRI's classification of the Proprietary Information as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

> "'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and
>
> (2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

e. The Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Information only after making a determination that the Proprietary Information was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Proprietary Information. EPRI was required to devote these resources and effort to derive the Proprietary Information. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Proprietary Information is highly valuable to EPRI.

f. A public disclosure of the Proprietary Information would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Proprietary Information both domestically and internationally. The Proprietary Information can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of North Carolina.

Executed at 1300 W WT Harris Blvd., Charlotte, North Carolina being the premises and place of business of Electric Power Research Institute, Inc.

Date: 9-9-2019.

[signature]

Neil Wilmshurst

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 9th day of September, 2019 by Neil Wilmshurst, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rouse (Seal)

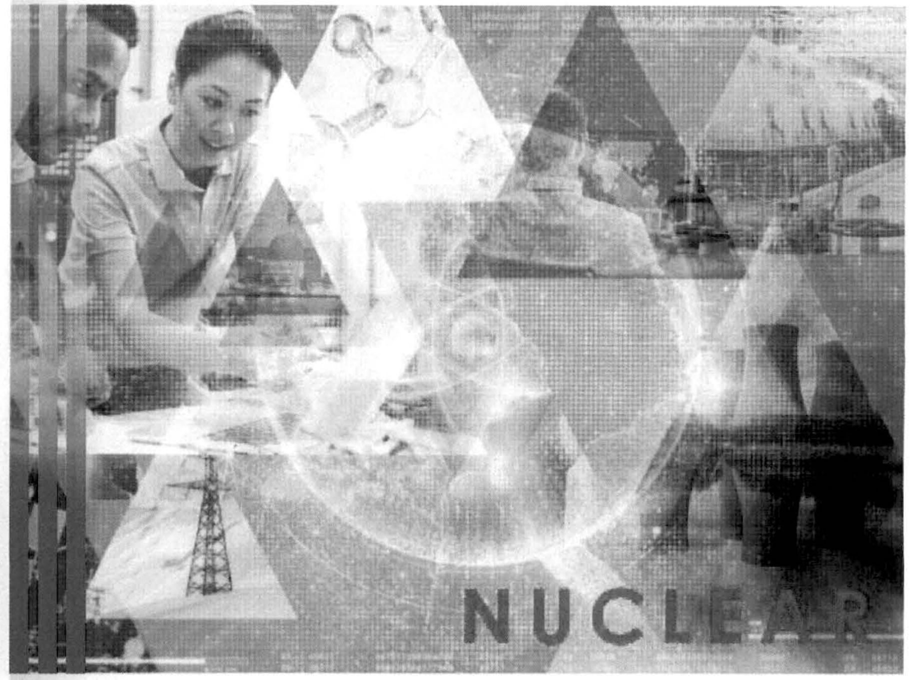
DEBORAH H. ROUSE
NOTARY PUBLIC
CABARRUS
COUNTY
NORTH CAROLINA

My Commission Expires 2nd day of April, 2021

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

BWRVIP-315: BWR Reactor Internals Aging Management for Extended Operations

(OPEN SESSION)



Wayne Lunceford, P.E.
Technical Executive, EPRI BWRVIP

NRC Public Meeting
September 11, 2019

BWRVIP History

- Program created in 1994
- Initial objective to address IGSCC identified in BWR core shrouds
- Expanded to encompass all materials issues associated with BWR RPVs and internals
- Large body of knowledge - currently, there are well over 300 BWRVIP reports, many of which have been revised one or more times
- Inspection and Flaw Evaluation (I&E) Guidance exists for all safety-related BWR RPV & internals components
- Consistent tracking, trending and evaluation of field inspection data for over 20 years
- Key involvement in development and deployment of IGSCC mitigation technologies
- History of proactive engagement with NRC to address aging management concerns associated with BWRs

BWRVIP Guidelines to Manage Degradation

Component	Assessment (I&E) Guidelines	Inspection Guidelines	Repair/Replace Design Criteria	Mitigation Recommendations
Core shroud	BWRVIP-76	BWRVIP-03	BWRVIP-02/-04	BWRVIP-62/-190
Core spray	BWRVIP-18	BWRVIP-03	BWRVIP-16/-19/-34	N/A
Shroud support	BWRVIP-38	BWRVIP-03	BWRVIP-52	BWRVIP-62/-190
Top Guide	BWRVIP-26	BWRVIP-03	BWRVIP-50	N/A
Core Plate	BWRVIP-25	BWRVIP-03	BWRVIP-50	BWRVIP-62/-190
SLC	BWRVIP-27	BWRVIP-03	BWRVIP-53	BWRVIP-62/-190
Jet pump assembly	BWRVIP-41	BWRVIP-03	BWRVIP-51	BWRVIP-62/-190
CRD guide/stub tube	BWRVIP-47	BWRVIP-03	BWRVIP-17/-55/-58	BWRVIP-62/-190
In-core housing/dry tube	BWRVIP-47	BWRVIP-03	BWRVIP-17/-55	BWRVIP-62/-190
Instrument penetrations	BWRVIP-49	BWRVIP-03	BWRVIP-57	BWRVIP-62/-190
LPCI coupling	BWRVIP-42	BWRVIP-03	BWRVIP-56	N/A
Vessel ID brackets	BWRVIP-48	BWRVIP-03	BWRVIP-52	BWRVIP-62/-190
Reactor pressure vessel	BWRVIP-74	N/A	N/A	N/A
Primary system piping	BWRVIP-75	N/A	N/A	BWRVIP-62/-190
Steam dryer	BWRVIP-139	BWRVIP-03	BWRVIP-181	N/A
Access hole cover	BWRVIP-180	BWRVIP-03	TBD	BWRVIP-62/-190
Top guide grid beam	BWRVIP-183	BWRVIP-03	BWRVIP-50	N/A
Bottom head drain line	BWRVIP-205	N/A	BWRVIP-208	N/A

Utility Participation & Commitment

- On May 30, 1997 BWRVIP issued a letter to NRC, committing all BWR U.S. utilities to full support in the development, maintenance and implementation of the BWRVIP program
- This commitment stayed in place until superseded by NEI 03-08, Guideline for the Management of Materials Issues
  - NEI 03-08 Materials Initiative Policy Statement:

    "… the industry will ensure that its management of materials degradation and aging is ***forward-looking and coordinated*** to the maximum extent practical. Additionally, the industry will ***continue to*** rapidly identify, react and ***effectively respond to emerging issues***. The associated work will be managed to emphasize safety and operational risk significance as the first priority, appropriately balancing long term aging management and cost as additional considerations. To that end, as issues are identified and as work is planned, the groups involved in funding, managing and providing program oversight will ensure that the ***safety and operational risk significance of each issue is fully established prior to final disposition***."

BWRVIP Approach to SLR

BWRVIP Approach to SLR (1 of 2)

- Existing aging management guidance is robust and, unless impacted by time-dependent factors, remains adequate to manage the effects of aging
  - Although unanticipated adverse trends are possible, the BWRVIP program already includes activities to assess operating experience and new R&D results.
  - Field performance data are continually evaluated and aging management guidance updated as appropriate
- Time-Dependent Factors:
  - Factors associated with age-related degradation that directly or indirectly correlate with total accumulated operating time
  - Examples – neutron fluence, fatigue cycles
- Many of the elements of an effective AMP are clearly independent of operating time (e.g., program scope, administrative controls, corrective actions, operating experience evaluation)
  - No need to revisit these program elements for SLR

BWRVIP Approach to SLR (2 of 2)

- Aging management guidance implementation should be linked to engineering-based parameters
  - Contained in the underlying analytical work forming the technical basis for AMP implementation
  - Limitations on guidance applicability tied to parameters related to onset / progression of age-related degradation (*not any specific operating period*)
- Applicability confirmed on a plant-specific basis (*results need not be "bounding" for the entire fleet*)
  - Each owner / licensee confirms that their plant satisfies the conditions for use of the guidance
  - BWRVIP focus placed on technical aspects of aging management
  - Decisions regarding plant operating period and licensing approach are the responsibility of the owner / licensee

BWRVIP Position on Use of Analyses Containing Fluence-Based Limitations

- Regardless of the decision-making associated with the fluence inputs used for an analysis, plants may rely on the analysis conclusions so long as the plant verifies that fluence remains less than that used as a basis for the analysis
  - Content describing input selection bases as being associated with any licensed operating period (*e.g., 60 years, 80 years*) is informational in nature
- BWRVIP role is to ensure that fluence-based limitations on guidance applicability are clearly identified
  - Licensees / applicants are responsible to ensure that limitations are considered in AMP implementation

BWRVIP-315, Reactor Internals Aging Management Evaluation for Extended Operations

BWRVIP-315, Reactor Internals Aging Management Evaluation for Extended Operations

- Provides a comprehensive technical basis for conclusions regarding the impact of aging effects and associated degradation mechanisms on aging management of BWR reactor internals for extended operations
- Identifies and addresses time-dependent limitations in guidance documents
  - Proposes specific content revisions to impacted guidance documents
  - Addresses "clarifications" appropriate to ensure the applicability of guidance is clear (*not required, but considered reasonable and appropriate*)
- Addresses further evaluation items associated with BWR reactor internals identified in the Standard Review Plan for SLR (NUREG-2192)

BWRVIP-315 - Contents

1. Introduction
2. BWRVIP reactor internals program scope
3. Degradation mechanism screening for extended operations
4. Component-specific aging management evaluations
   - Each reactor internal evaluated to identify applicability of degradation mechanisms impacted by consideration of extended operations
   - Aging management guidance reviewed to identify clarifications or enhancements needed to address extended operation
5. AMP elements evaluation for extended operations
   - Demonstrates that most AMP elements are independent of operating time and remain adequate for use in any licensed operating period)
6. Summary of Results

BWRVIP-315 – Contents (continued)

Appendix A - Documents BWRVIP guidance for reactor internals that includes "Needed" elements under NEI 03-08

Appendix B – Identifies clarifications, enhancements, and revisions proposed for BWRVIP I&E guidance to address extended operations

Appendix C - BWRVIP assessment of "Further Evaluation" items related to BWR internals in NUREG-2192 (SRP-SLR)

Appendix D - Disposition of EAF for BWR Reactor Internals

Appendix E - Fleet Neutron Fluence Evaluation

Appendix F - CASS Internals Structural Evaluation

Appendix G - Jet Pump Holddown Beam Stress Relaxation Evaluation

Appendix H - Summary information regarding degradation mechanisms applicable to BWR reactor internals

BWRVIP-315 – Summary of Conclusions (1 of 2)

- Neutron fluence is the primary time-dependent factor affecting aging management of BWR reactor internals
  - Neutron flux associated with BWR reactor internals can vary significantly depending on reactor design, power uprate status, and fuel management strategies
  - Clearly supports the use of fluence-based limitations rather than bounding fleet assessments
- Degradation mechanisms relevant to extended operations
  - IASCC
  - Irradiation Embrittlement
  - CASS Embrittlement
  - Irradiation-Enhanced Stress Relaxation
  - IGSCC Initiation (in precipitation hardened Ni-base alloy components)
  - Low-Cycle Fatigue / Environmentally-Assisted Fatigue

BWRVIP-315 – Summary of Conclusions (2 of 2)

- Identified limitations on BWRVIP guidance applicability
  - Core plate holddown bolt fluence
  - CASS internals fluence
  - Reinspection requirements for lower plenum components addressed by BWRVIP-47-A
  - Jet pump diffuser / adapter scope expansion exemption allowances
  - Jet pump holddown beam fluence
  - Evaluation of core shroud tie rod repairs
- None of these limitations represent a significant barrier for SLR
  - Limitations are conservatively identified - many plants will not exceed any of these limitations during an 80-year operating life
  - Plant-specific evaluation options are available to address limitations

BWRVIP-315 – Status & Future Plans

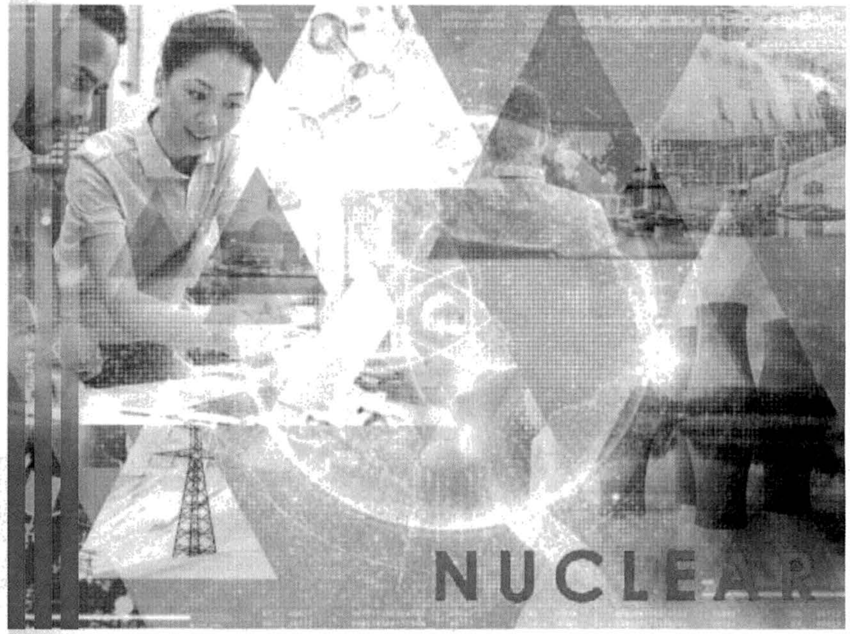
- *EPRI Report 3002012535, BWRVIP-315: BWR Vessel and Internals Project - Reactor Internals Aging Management Evaluation for Extended Operations*
  - Published July 2019
  - Preparation of Proprietary and Non-Proprietary versions in progress
  - BWRVIP intends to submit BWRVIP-315 to NRC for review and approval in October 2019

Together...Shaping the Future of Electricity

ELECTRIC POWER
RESEARCH INSTITUTE

BWRVIP-315: BWR Reactor Internals Aging Management for Extended Operations

(CLOSED SESSION)



Wayne Lunceford, P.E.
Technical Executive, EPRI BWRVIP

NRC Public Meeting
September 11, 2019

www.epri.com

Agenda (Closed Session)

- Report Contents
- BWRVIP Reactor Internals Program Scope (Section 2)
- Generic Degradation Assessment for Extended Operations (Section 3)
- Aging Management Review for Extended Operations (Section 4)
  - Generic Aging Management Elements Associated with Extended Operations
    - Applicability and Use of BWRVIP Safety Assessments
    - Periodic Inspection Philosophy
    - Management of IASCC
  - Component Aging Management Evaluation Approach and Conclusions
  - BWRVIP Aging Management Guidance Limitations & Clarifications (Sect. 4.5)
- AMP Attribute Assessment for Extended Operations (Section 5)
- Status & Plans for Submittal
- Additional Topics
  - Management of Cracking due to LCF / EAF
  - SRP-SLR and GALL-SLR Further Evaluation Items
  - BWRVIP Position on License Renewal Appendices

BWRVIP-315 - Contents

1. Introduction
2. BWRVIP reactor internals program scope
3. Degradation mechanism screening for extended operations
4. Component-specific aging management evaluations
   - Each reactor internal evaluated to identify applicability of degradation mechanisms impacted by consideration of extended operations
   - Aging management guidance reviewed to identify clarifications or enhancements needed to address extended operation
5. AMP elements evaluation for extended operations
   - Demonstrates that most AMP elements are independent of operating time and remain adequate for use in any licensed operating period)
6. Summary of Results

BWRVIP-315 – Contents (continued)

Appendix A - Documents BWRVIP guidance for reactor internals that includes "Needed" elements under NEI 03-08

Appendix B – Identifies clarifications, enhancements, and revisions proposed for BWRVIP I&E guidance to address extended operations

Appendix C - BWRVIP assessment of "Further Evaluation" items related to BWR internals in NUREG-2192 (SRP-SLR)

Appendix D - Disposition of EAF for BWR Reactor Internals

Appendix E - Fleet Neutron Fluence Evaluation

Appendix F - CASS Internals Structural Evaluation

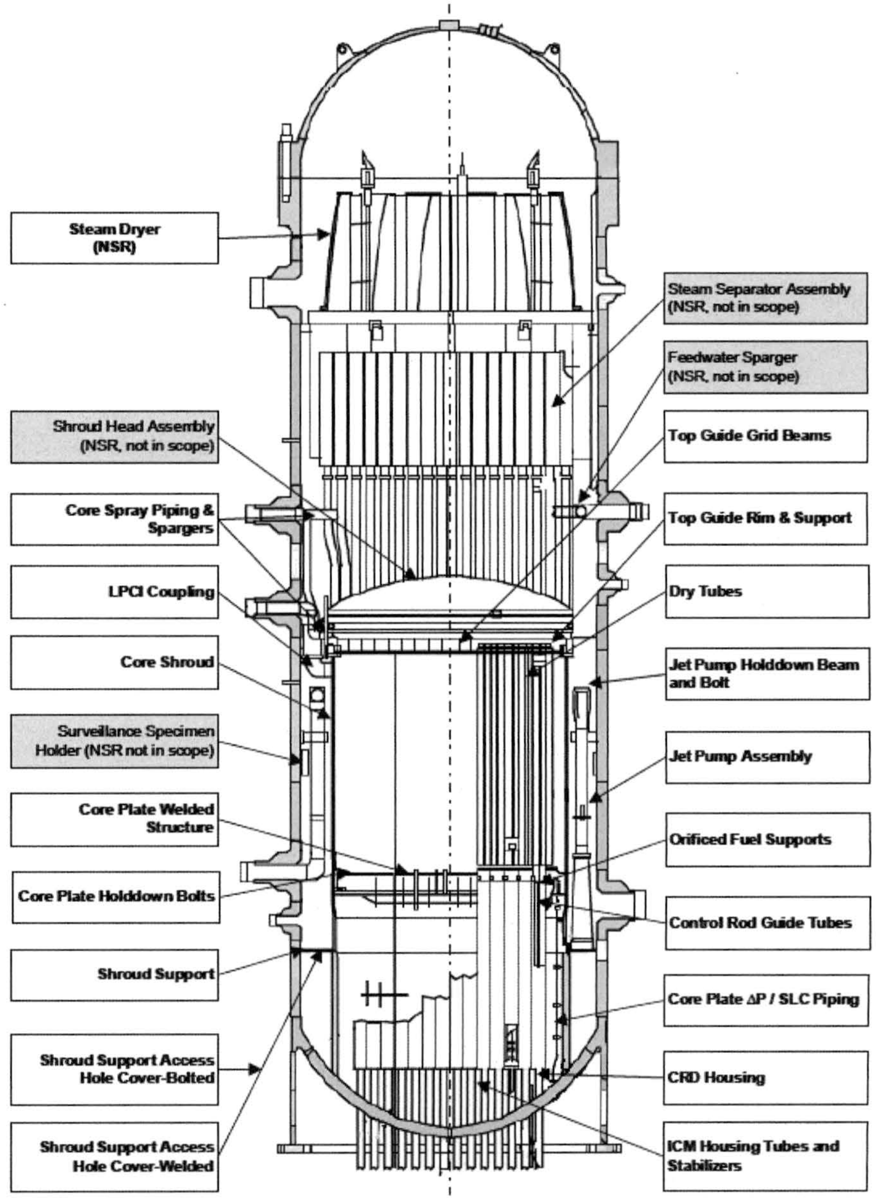
Appendix G - Jet Pump Holddown Beam Stress Relaxation Evaluation

Appendix H - Summary information regarding degradation mechanisms applicable to BWR reactor internals

BWRVIP Reactor Internals Program Scope (BWRVIP-315, Section 2)

BWRVIP Reactor Internals Program Scope

- Identifies reactor internals components addressed by BWRVIP aging management guidance, including:
  - Safety-related reactor internals
  - Original steam dryers
  - Replacement access hole cover designs, core shroud tie rod hardware
- Not intended to replace plant-specific scoping and screening required by 10CFR54



EPRI ELECTRIC POWER RESEARCH INSTITUTE

BWRVIP Reactor Internals Program Scope (continued)

- Program scope does NOT address:
  - Non-safety related components – with the exception of original steam dryers (*shroud head assembly, steam separator assembly, feedwater spargers, surveillance holders*)
  - Steam dryers having configurations significantly different than the original equipment dryers (*e.g., Nordic style replacement steam dryers*)
  - Jet pump repair and FIV mitigation hardware (*e.g., auxiliary wedges, slip joint clamps, labyrinth seals*)
  - Core spray internals piping section replacements
- In these cases, there are limitations which prevent development of comprehensive aging management guidance
  - Limited access to configuration information
  - Vendor design details not available – and may include TLAAs associated with neutron fluence, fatigue usage or fastener preload
  - BWRVIP-315, Section 4.4 provides general principles that can be applied by licensees to develop plant-specific aging management guidance

Generic Degradation Assessment for Extended Operations (BWRVIP-315, Section 3)

Generic Degradation Assessment Approach

Identify reactor internals materials of construction

↓

Identify relevant aging effects and associated age-related degradation mechanisms

↓

Extended operations aging management impact evaluation

Aging management guidance potentially impacted	Aging management guidance NOT impacted

↓

Establish conservative screening values for application to component-specific AMRs

BWR Reactor Internals Materials of Construction

Material	Applicable Material Grades
Structural Alloys	
Stainless Steel (and weld filler materials)	Types 304, 304L, 316, and 316L (weld materials: 308, 308L, 316, 316L)
Cast Austenitic Stainless Steel (CASS)	CF3, CF8 (CF3M, CF8M allowed as options)
Ni-base Alloys (and weld filler materials)	Alloy 600 (weld materials: Alloy 82, 182)
Fasteners & Hardware	
Solution Annealed Stainless Steel (SA SS)	Type 304 and 316
Nitrogen Strengthened Austenitic SS (XM-19)	Type XM-19
Precipitation Hardened (PH) Ni-base Alloys	Alloy X-750, 718

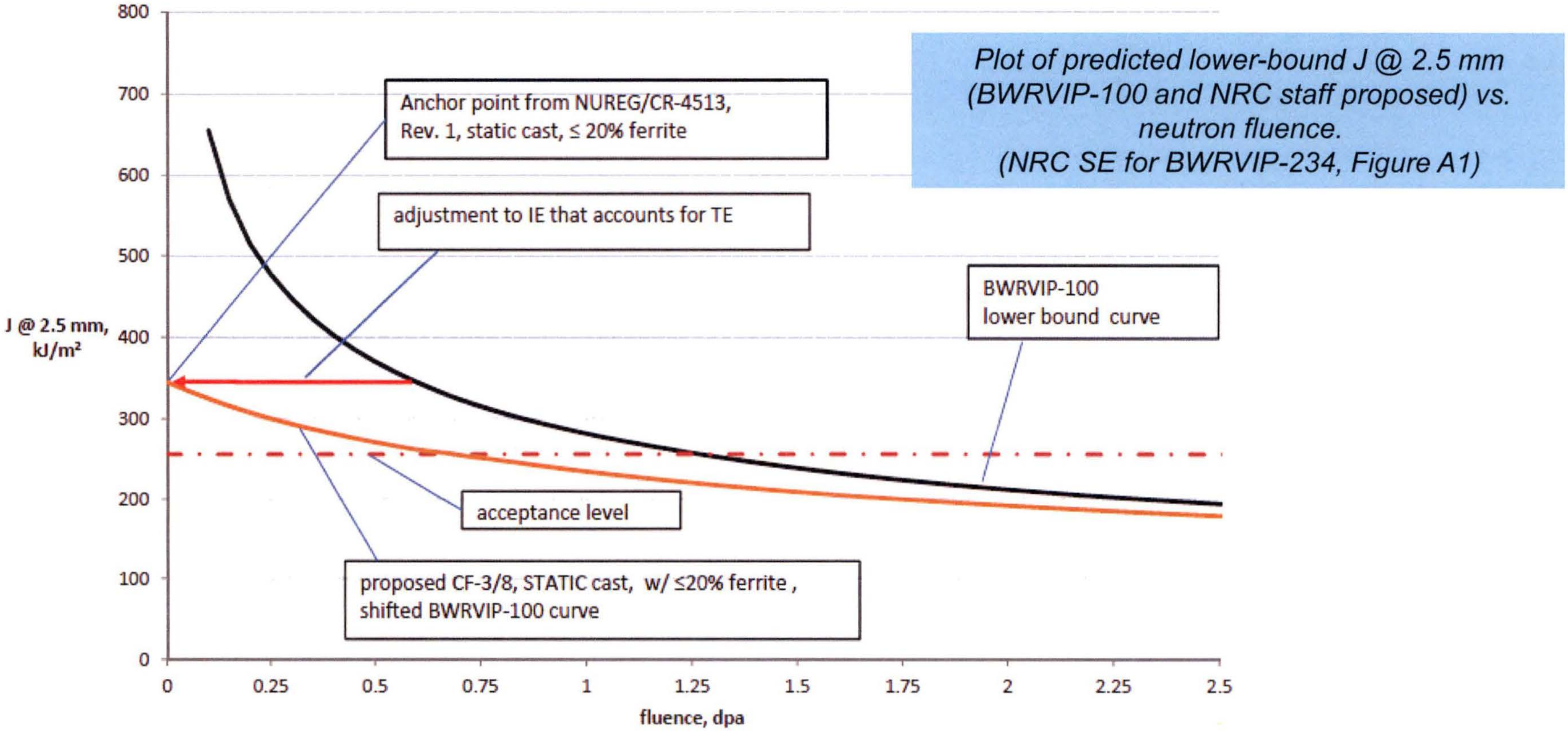
Applicable Aging Effects and Associated Degradation Mechanisms

Aging Effect	Age-Related Degradation Mechanisms
Loss of Material	General Corrosion
	Crevice Corrosion
	Wear
Cracking	Intergranular Stress Corrosion Cracking (IGSCC)
	Irradiation-Assisted Stress Corrosion Cracking (IASCC)
	Low-Cycle Fatigue (LCF) / Environmentally-Assisted Fatigue (EAF)
	High-Cycle Fatigue (HCF)
Loss of Fracture Toughness	Thermal Embrittlement (TE)
	Irradiation Embrittlement (IE)
Loss of Preload	Thermal Stress Relaxation
	Irradiation-Assisted Stress Relaxation (IA SR)

Treatment of Embrittlement Mechanisms

- In terms of aging management assessment, irradiation embrittlement (IE) and thermal embrittlement (TE) are treated as contributing elements associated with:
  - CASS Management
  - IASCC Management
- These simplifying actions are reasonable in the context of how BWR reactor internals components are managed by plants and with presentation in existing BWRVIP I&E guidelines

Basis for Managing CASS Embrittlement (BWRVIP-234-A)



- Thermal embrittlement (TE) applicable only to CASS components
- Treated synergistically with irradiation embrittlement (IE) in establishing aging management criteria
- Defined as "CASS Embrittlement" in BWRVIP-315

IASCC / Irradiation Embrittlement

- For irradiated structures, the primary detrimental effect of irradiation embrittlement on component function is a reduction in the capability to tolerate flaws
- In the absence of existing flaws, there is no detrimental impact on the component's capability to perform its intended function
- The onset of significant irradiation embrittlement in austenitic materials and the conservative threshold established for IASCC are similar
- Consequently, a simplifying assumption has been made to combine these mechanisms.

Aging Effects and Associated Degradation Mechanisms (as simplified in BWRVIP-315)

Aging Effect	Age-Related Degradation Mechanisms
Loss of Material	General Corrosion
	Crevice Corrosion
	Wear
Cracking	Intergranular Stress Corrosion Cracking (IGSCC)
	IASCC / Irradiation Embrittlement (IE)
	Low-Cycle Fatigue (LCF) / Environmentally-Assisted Fatigue (EAF)
	High-Cycle Fatigue (HCF)
Loss of Fracture Resistance	CASS Embrittlement
Loss of Preload	Thermal Stress Relaxation
	Irradiation-Assisted Stress Relaxation (IA SR)

Extended Operations Aging Management Impact Evaluation

- Identify those aging effects and associated age-related degradation mechanisms that require additional review to assess the impact of operation beyond 60 years (extended operations) on the adequacy of existing BWRVIP aging management guidance
- Where the potential for new occurrences of age-related degradation (*e.g., initiation of new cracks*) or for progression of degradation (*e.g., crack growth rates*) does NOT correlate with operating time, there is no reason to revisit prior conclusions regarding the adequacy of existing BWRVIP aging management guidance
- For degradation mechanisms not impacted by consideration of extended operations, continued implementation of existing aging management guidance is adequate, regardless of accumulated service time

Aging Management Extended Operations Impact Results

Aging Effect	Age-Related Degradation Mechanisms
Loss of Material	General Corrosion
	Crevice Corrosion
	Wear
Cracking	**Intergranular Stress Corrosion Cracking (IGSCC) ***
	IASCC / Irradiation Embrittlement (IE)
	Low-Cycle Fatigue (LCF) / Environmentally-Assisted Fatigue (EAF)
	High-Cycle Fatigue (HCF)
Loss of Fracture Resistance	**CASS Embrittlement**
Loss of Preload	Thermal Stress Relaxation
	Irradiation-Assisted Stress Relaxation (IA SR)

Potential Impact	No Impact

* Precipitation-Hardened Ni-Base Alloys (X-750, 718)

Degradation Mechanisms not Impacted by Extended Operations (1 of 2)

- Loss of Material
  - General corrosion rates are insignificant for austenitic reactor internals components with respect to the ability of components to perform their function, irrespective of exposure time
    - Corrosion allowances far exceed anticipated service lives
    - Excessive crevice corrosion of stainless steels and nickel alloys will not occur irrespective of exposure time provided that water chemistry controls are maintained
    - Managed by water chemistry program
  - Wear
    - No basis for postulating time-dependent acceleration or significant adverse trends with extended operating periods
    - Although an adverse trend could potentially occur if a plant undergoes significant changes in operation (*such as a power uprate or flexible operations*), changes in trends are not truly time-dependent, but are a result of the change in operating conditions

Degradation Mechanisms not Impacted by Extended Operations (2 of 2)

- High-Cycle Fatigue (HCF)
  - Failures generally occur within a few operating cycles after initial plant startup or after a change in operating conditions
  - Changes in plant conditions could occur at any time and are not linked with extended operation
  - No reason to anticipate an acceleration of HCF events late in plant life
- Thermal stress relaxation
  - Accounted for in joint design
  - Effect saturates after a relatively short time of operation

IGSCC Extended Operations Impact Conclusions

Material	Extended Operations Impact?
Stainless Steel (and weld filler materials)	NO - Most occurrences associated with cold work, sensitization initiated early in plant life.
Cast Austenitic Stainless Steel (CASS)	NO - CASS components have not experienced IGSCC to date. For extended operations, no basis for anticipating any adverse trend.
Ni-base Alloys (and weld filler materials)	NO - Most occurrences associated with creviced conditions / Alloy 182. No reason to anticipate any adverse trend during extended operations.
Solution Annealed Stainless Steel (SA SS)	NO - No failures identified to date. No basis for postulating adverse trends. (see BWRVIP-25R1)
Nitrogen Strengthened Austenitic SS (XM-19)	NO - No failures identified to date. No basis for postulating adverse trends. (see BWRVIP-25R1)
Precipitation Hardened (PH) Ni-base Alloys	**YES - A time-dependency on initiation is conservatively assumed to exist based on available data**

- IGSCC has been, and will continue to be, a focus of BWRVIP aging management
- For most reactor internals materials, no evidence suggesting the potential for an adverse trend in performance – and therefore existing guidance is deemed adequate
- **Conservative decision made to consider initiation in precipitation hardened Ni-base alloys as relevant to extended operations**

IGSCC of Welded Structures (SS, Ni-Base Alloys)

- For welded internals structures, available OE indicates a declining trend
- Few new cracks are being identified and many IGSCC flaws in components remaining in service have not grown over time
  - Since implementation of BWRVIP-18 and BWRVIP-41, significant decline in numbers of cracks identified and in the size of cracks newly identified for both core spray internals piping and jet pump assemblies (see BWRVIP-251, BWRVIP-266)
  - Most reactor internals cracking now thought to have initiated / progressed relatively early in plant life, with detection not occurring until the application of high-resolution visual techniques and UT
  - Many of the most susceptible locations have been repaired or replaced – often pre-emptively
  - No reason to anticipate any significant change in IGSCC occurrence or growth trends
- CONCLUSION – AGING MANAGEMENT GUIDANCE ADDRESSING IGSCC IS ADEQUATE, BOTH NOW AND FOR EXTENDED OPERATIONS

Example: Core Spray Piping IGSCC

[[

Content Deleted
EPRI Proprietary Information

]]

Source: BWRVIP-251

Example: Core Spray Piping IGSCC (continued)

EPRI Proprietary Information

- NDE uncertainty a significant factor impacting CGRs estimated from 1 or 2 cycles of operation
- As the time between measurements increases, few flaws are found to be actively growing
- General observation of crack arrest in most cases
- No conclusive evidence of new crack initiations

[[

Content Deleted
EPRI Proprietary Information

]]

IGSCC of Solution Annealed SS / XM-19 Fasteners

- Fasteners and hardware fabricated from solution annealed stainless steel and XM-19 have not shown susceptibility to IGSCC in BWR service
  - No IGSCC identified
  - Bolts supplied in solution annealed condition
  - Strain introduced by manufacturing significantly below the values anticipated to introduce IGSCC susceptibility
  - BWR reactor internals applications do not involve high applied stresses nor weld residual stresses
  - BWRVIP-25, Rev. 1 provides technical bases supporting this conclusion
- For extended operations
  - No data indicating the potential for an adverse trend to occur (*i.e., no established correlation between IGSCC initiation and time in service*)

Degradation Mechanism Applicability Screening Values

- Used as a basis for component-specific AMR conclusions
- Generally identified using conservative values
- Ensures that component evaluations conservatively consider the potential impact of extended operations on aging management guidance

IGSCC Screening Value for PH Ni-Base Alloys (X-750, Alloy 718)

- Component-specific AMRs consider the potential for IGSCC to impact aging management guidance for any applications involving [[Content Deleted EPRI Proprietary Information]]
- Very conservative screening threshold value
  - Although laboratory studies suggest initiation is plausible at applied stresses as low as [[Content Deleted EPRI Proprietary Information]], X-750 treated in the HTA condition has not failed in service
  - VERY long initiation periods anticipated [[Content Deleted EPRI Proprietary Information]]

IASCC / IE Screening Values (1 of 3)

- Welded austenitic structures: [[EPRI Proprietary Information Content Deleted]]
  - Consistent with generally accepted threshold value
  - Corresponds with use of a different CGR correlation (BWRVIP-99-A vs. BWRVIP-14-A)
  - In practice, only applicable to SS structures, unlikely that any welded Ni-base alloy structures will be subject to fluence approaching this threshold value
- CASS: Not Applicable
  - No operating experience indicating that CASS components are susceptible to IGSCC in BWR environment
  - CASS embrittlement fluence threshold ensures that castings subject to significant neutron fluence are evaluated

IASCC / IE Screening Values (2 of 3)

- SA SS / XM-19 Fasteners

[[

Content Deleted
EPRI Proprietary Information

]]

IASCC / IE Screening Values (3 of 3)

EPRI Proprietary Information

- PH Ni-base materials (X-750, Alloy 718)

[[

Content Deleted
EPRI Proprietary Information

]]

Other Screening Values

- CASS Embrittlement:
  - Neutron fluence > $6x10^{20}$ n/cm^2 (E > 1.0 MeV)
  - Consistent with NRC SE on BWRVIP-234
- Irradiation-Enhanced Stress Relaxation:
  - [[

Content Deleted
EPRI Proprietary Information

  -]]
- LCF / EAF: Generic screening value not used
  - BWRVIP approach to managing cracking due to fatigue relies on management of SCC – a significantly more limiting cracking mechanism for BWR reactor internals
  - Addressed in component-specific evaluations in Section 4 and in Appendix D

Aging Management Review for Extended Operations (BWRVIP-315, Section 4.2) Approach & Generic Aging Management Elements

Generic Degradation Assessment Approach

Generic Aging Management Elements (applicable to multiple components)

BWRVIP Safety Assessment	Periodic Inspection Philosophy	IASCC Management

Component Evaluations

Description / Function	Degradation Assessment	Extended Operations Evaluation

Limitations on Use of BWRVIP Guidance

BWRVIP Guidance Revisions

Generic Aging Management Elements: Applicability and Use of BWRVIP Safety Assessments

- BWRVIP-06 (current version is Rev. 1-A) documents safety assessment bases for BWR reactor internals
- Not developed within the context of extended operations
- In assessing long-term aging management needs, conclusions in BWRVIP-06, Rev. 1-A were based on evaluation of detectability, redundancy, and inspection
- These elements generally do not change over time and as such, safety assessment conclusions in BWRVIP-06, Rev. 1-A remain valid
- No "new" component locations identified as requiring inspection to address extended operations

Generic Aging Management Elements: Periodic Inspection Philosophy

- BWRVIP reinspection requirements are interval-based and not tied to plant service life or other calendar-based metrics (*e.g., EFPYs*)
  - Reinspection intervals based on generic flaw tolerance evaluations, evaluation of inspection data, and engineering judgement
  - Components / locations found to be more susceptible to degradation and / or to have lower flaw tolerance are inspected more frequently
  - Components / locations found not to be subject to significant degradation or are highly tolerant of flaws are inspected less frequently
- The BWRVIP is a living program
  - New OE and R&D results are reviewed and, over time, may result in changes to inspection intervals or the introduction of new inspection requirements
- This approach remains unchanged regardless of service life. Periodic inspections, OE assessments, and appropriate adjustments to inspection requirements will continue to occur
- Therefore, there is no basis for restricting BWRVIP aging management to any specific plant service life

Generic Aging Management Elements: IASCC / Irradiation Embrittlement – SCC Initiation Trends

- No evidence that increasing neutron fluence has had a significant effect on initiation of new cracks
  - Jet Pump Risers – essentially no cracking detected in upper riser region (*BWRVIP-41, Rev. 4 and associated BWRVIP response to NRC RAI-4 in BWRVIP letter 2017-022*)
  - Top guides – No evidence of a correlation between cracking observed and accumulated neutron fluence
    - Small number of indications identified to date fleet-wide, with all of the locations having factors that would indicate IGSCC susceptibility
    - Highly oxidizing environment above the core, some IGSCC is anticipated to, regardless of fluence
    - Most locations not high fluence (*rim, pin locations on top guide periphery*)
- General conclusions
  - No evidence of any ongoing adverse trends associated with IASCC
  - Periodic inspections remain adequate to identify any future adverse trends

Example:
Jet Pump Weld Performance

[[

Content Deleted
EPRI Proprietary Information

]]

Source: BWRVIP-266

Generic Aging Management Elements: IASCC / Irradiation Embrittlement – Flaw Evaluation

- Evaluation of flaws in irradiated components is often necessary, with the primary component affected being the core shroud
- However, evaluation methods and supporting data / models are available:
  - CGRs - BWRVIP-99-A, ASME Code Case N-889
  - Fracture Toughness (FT): BWRVIP-100, Rev.1-A, MRP-211, Rev.1
- Clarifications made to guidance (*not considered changes to guidance – just addition of words to clearly describe current guidance*):
  - Ensure that that applicability limits are clearly defined (*e.g., fluence limit on use of BWRVIP-99-A CGR correlation*)
  - Ensure guidance for components with late life fluence potentially > IASCC threshold clearly states the need to consider fluence in flaw tolerance evaluations (*e.g., jet pumps, LPCI couplings*)

Generic Aging Management Elements: IASCC / Irradiation Embrittlement – Flaw Evaluation

- References:
  - BWRVIP Letter 2012-074, Superseded "Needed" Guidance Regarding Crack Growth Assumptions
  - BWRVIP-99-A, Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components
  - BWRVIP-100, Rev. 1-A, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds
  - ASME Code Case (CC) N-889, Reference Stress Corrosion Crack Growth Rate Curves for Irradiated Austenitic Stainless Steels in Light Water Reactor Environments
  - MRP-211, Rev. 1, PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data—State of Knowledge

Flaw Tolerance Evaluation Options and Associated Fluence Limits

[[

Content Deleted
EPRI Proprietary Information

]]

Generic Aging Management Elements: IASCC / Irradiation Embrittlement – Flaw Evaluation

- General conclusions
  - Necessary inputs to flaw evaluations are available – there are no fundamental gaps in state of knowledge that prevent performance of flaw evaluations for even highly irradiated BWR reactor internals
  - Depending on the evaluation details and projected end of interval fluence, some evaluations could represent deviations from BWRVIP guidance potentially resulting in submittal to NRC
    - From BWRVIP-94NP, Rev. 3:

      "Flaw evaluations that deviate from the guidance in BWRVIP reports (e.g., assumptions, methods, acceptance criteria, etc.) shall be submitted to the NRC. If the flaw evaluation is later revised, the results shall be communicated to the NRC. The submittal schedule for the analyses shall be determined by the licensee and the NRC. Flaw evaluations submitted to the NRC shall also be submitted to the BWRVIP."

Aging Management Review for Extended Operations (BWRVIP-315, Section 4.3) Component Aging Management Evaluations

Component Aging Management Evaluation Elements

- Component Description / Function
  - Provides an overview of the component's function for general context
  - Review BWRVIP-06, Rev. 1-A and I&E guideline content regarding component safety function and safety assessment conclusions
- Degradation Assessment
  - Evaluate the applicability of each degradation mechanism relevant to extended operations (*from Section 3*)
  - Applies screening thresholds as appropriate
  - Reviews factors relevant to degradation potential – material of construction, fabrication factors, environment (neutron fluence, water chemistry), relevant OE & R&D results
- Extended Operations Aging Management Evaluation
  - Assesses the adequacy of existing guidance to manage the applicable degradation mechanisms
  - Reviews aging management guidance for time-dependent limitations that may limit applicability for extended operations
  - Identifies where enhancements or revisions of BWRVIP reactor internals aging management guidance are appropriate to address extended operations

Extended Operations Component Aging Management Evaluation Results (1 of 6)

[[

Content Deleted
EPRI Proprietary Information

]]

Extended Operations Component Aging Management Evaluation Results (2 of 6)

[[

Content Deleted
EPRI Proprietary Information

]]

Extended Operations Component Aging Management Evaluation Results (3 of 6)

[[

Content Deleted
EPRI Proprietary Information

]]

Extended Operations Component Aging Management Evaluation Results (4 of 6)

[[

Content Deleted
EPRI Proprietary Information

]]

Extended Operations Component Aging Management Evaluation Results (5 of 6)

[[

Content Deleted
EPRI Proprietary Information

]]

Extended Operations Component Aging Management Evaluation Results (6 of 6)

[[

Content Deleted
EPRI Proprietary Information

]]

BWRVIP Aging Management Guidance Limitations & Clarifications (BWRVIP-315, Section 4.5)

BWRVIP Reactor Internals Aging Management Guidance Limitation 1 – CP Holddown Bolting Stress Relaxation

- Core plate holddown bolts require adequate preload to perform their intended function (restrain core plate movement to ensure core orientation and capability to insert control rods)
- Loss of preload occurs over time due to the combined effects of thermal loosening (change in elastic modulus), thermal creep, and irradiation-induced stress relaxation
- BWRVIP-25, Revision 1 provides a technical basis for bolt integrity and functionality that includes:
  - A margin assessment to show how many bolts are required for various load levels to ensure that core plate horizontal displacement is maintained below an acceptable level and ASME allowable stress limits are met
  - An evaluation of irradiation-enhanced stress relaxation occurring due to neutron fluence

[[

Content Deleted
EPRI Proprietary Information

]]

Typical Core Plate Holddown Bolt (from BWRVIP-25, Rev. 1 Appendix I)

Core Plate Holddown Bolt Relaxation: Integrated Relaxation due to Fluence

- Amount of relaxation is directly related to the amount of irradiation (fluence) along the length of the bolt

[[

Content Deleted
EPRI Proprietary Information

]]

[[

Content Deleted
EPRI Proprietary Information

]]

Core Plate Holddown Bolts: BWRVIP Conclusions Related to Extended Operations

- Evaluation provides a fluence-based limitation
  - Plants must confirm that core plate bolt fluence remains less than the value assumed in BWRVIP-25, Rev. 1, Appendix I through the end of the time period for which the analysis is credited
- Review of available fluence calculations for a number of plants indicates that most plants will have 80-yr core plate holddown bolt peak fluence below the limiting value used in the generic analysis
- Core Plate Holddown Bolts irradiation-enhanced stress relaxation is identified as "further evaluation" item 3.1.2.2.14 in the SRP-SLR. BWRVIP-315, Appendix C includes a review of the further evaluation item that can be used to aid SLR applicants to address this item.

BWRVIP Reactor Internals Aging Management Guidance Limitation 2 – CRGT Aging Management

BWRVIP-47-A provides for a set of baseline examinations of CRGTs. Section 3.2.2 of BWRVIP-47-A states:

"Currently no additional inspections are recommended beyond the baseline inspections described in Section 3.2.2, and scope expansion and follow-on inspections deemed necessary in the event flaws are found as given in Section 3.2.3. 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."

- The BWRVIP is in the process of preparing a revision to BWRVIP-47
- Until such time as a new version of BWRVIP-47 is developed, owners should either
  - Commit to implementing a future version of BWRVIP-47 that addresses extended operations or
  - Propose a set of plant-specific activities

BWRVIP Reactor Internals Aging Management Guidance Limitation 3 - CASS Embrittlement

- Jet pump and LPCI coupling CASS components subjected to fluence exceeding $6x10^{20}$ n/cm^2 (E > 1.0 MeV) must be evaluated on a plant-specific basis or be included in a plant-specific aging management program
  - Basis provided in BWRVIP-234-A
- BWRVIP I&E guidelines affected include BWRVIP-41 and BWRVIP-42

CASS Embrittlement: BWRVIP Position for Extended Operations

- Evaluation consistent with fluence limit contained in BWRVIP-234-A remains an appropriate, regardless of operating period
- CASS internals exceeding this threshold fluence value and subject to embrittlement must be evaluated on a plant-specific basis
- Primary concern occurs for jet pump and LPCI coupling castings
  - Located in a region for which fluence may exceed $6x10^{20}$ n/cm^2
  - However, there are factors that can be applied to address embrittlement concerns
    - In cases where fluence exceeds the threshold, EOL values are only marginally above $6x10^{20}$ n/cm^2
    - Structural evaluations based on typical component geometries and loadings indicate that the J value associated with NRC SE on BWRVIP-234 is conservative (*documented in Appendix F of BWRVIP-315*)
    - Methods documented in Appendix F could be applied by plants as needed to develop plant-specific evaluations of CASS components
- CASS embrittlement identified as "further evaluation" item 3.1.2.2.13 in the SRP-SLR. BWRVIP-315, Appendix C includes a review of the further evaluation item that can be used to aid SLR applicants to address this item.

CASS Management - Example Structural Margin Assessment for Jet Pump Inlet Casting

[[

- Through-wall crack length corresponding to the allowable J value is equal to [[ ]] of the circumference

Content Deleted
EPRI Proprietary Information

Content Deleted
EPRI Proprietary Information

]]

J_{app} values for through-wall crack in IN-1 under faulted condition (BWRVIP-315, Appendix F, Figure F-11)

BWRVIP Reactor Internals Aging Management Guidance Limitation 4 - Jet Pump Large Diameter Weld Scope Expansion Exemption

- A scope expansion exemption is provided within BWRVIP-41, Rev. 4-A for large diameter jet pump diffuser, adapter, and lower ring welds (DF-1, DF-2, DF-3, AD-1, AD-2, and AD-3a,b) inspected by UT
- As currently included in BWRVIP-41, Rev. 4-A, the exemption is based on assumption of a 60-year service life
- Included in BWRVIP-41, Rev. 4 and considered in the NRC SE for Rev. 4
- Exemption to be revised to be [[Content Deleted EPRI Proprietary Information]]
  - Interval based on generic flaw tolerance evaluations and consideration of other jet pump weld inspection intervals

EPRI Proprietary Information

BWRVIP Reactor Internals Aging Management Guidance Limitation 5 – Jet Pump Holddown Beam IASCC

[[

Content Deleted
EPRI Proprietary Information

]]

- Unlikely that any JP beams that reach this fluence during an 80-yr service life
- If necessary, beam replacement is an option available to owners

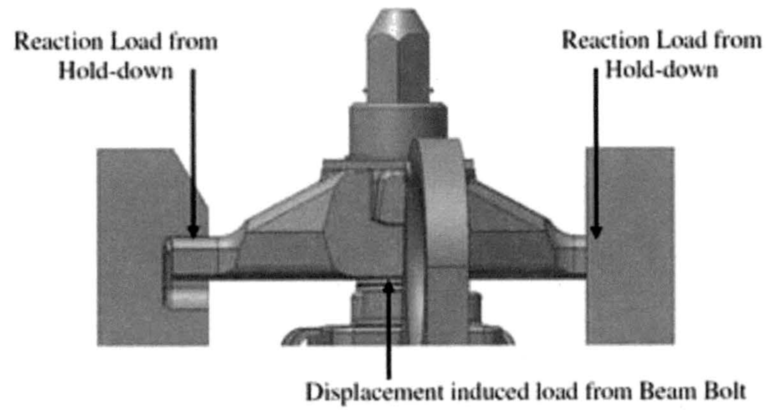
[[

Content Deleted
EPRI Proprietary Information

]]

BWRVIP Reactor Internals Aging Management Guidance Limitation 6 - Jet Pump Holddown Beam Stress Relaxation

- For some beams, irradiation-enhanced stress relaxation could be significant during extended operations
- Concern was that beam preload could be reduced to a value below the minimum allowable
- In a Sept. 2018 meeting, NRC expressed a strong desire that this issue be resolved in the initial version of BWRVIP-315 transmitted for review and approval



BWRVIP Reactor Internals Aging Management Guidance Limitation 6 - Jet Pump Holddown Beam Stress Relaxation

- Evaluation now complete

[[

Content Deleted
EPRI Proprietary Information

]]

  - Limitation ensures that sufficient preload to prevent jet pump disassembly and potential damage is maintained
  - Plant-specific disposition may include refined analysis to demonstrate adequate preload remains for operation at higher neutron fluences
  - Alternatively, plants may replace or re-tension beams with neutron fluence exceeding the threshold value
- Evaluation details provided in Appendix G of BWRVIP-315

BWRVIP Reactor Internals Aging Management Guidance Limitation 7 - Core Shroud Repair Hardware Aging Management

- Core shroud tie rod repairs require plant-specific evaluation
  - Inspections should, as a minimum, meet the requirements listed in BWRVIP-76, Rev. 1-A
  - Additional evaluations needed to address aging management associated with operation beyond the original repair hardware service life specified by the designer.

BWRVIP Reactor Internals Aging Management Guidance - Clarifications (A & B)

[[

Content Deleted
EPRI Proprietary Information

]]

Reactor Internals Aging Management Conclusions

- No changes to periodic inspection requirements within BWRVIP I&E guidelines are necessary to address extended operations
- Limited number of revisions and enhancements to I&E guidance identified to ensure the impact of increased neutron fluence is appropriately considered by BWRVIP program owners
- Revisions are proposed for the following I&E Guidelines:
  - BWRVIP-41, Jet Pump
  - BWRVIP-42, LPCI Coupling
  - BWRVIP-47, Lower Plenum / Control Rod Guide Tubes
  - BWRVIP-76, Core Shroud
  - BWRVIP-26 / BWRVIP-183, Top Guide
- In most cases, Appendix B of BWRVIP-315 contains the detailed set of guidance changes proposed
  - Guidance revisions to be made as guidelines are revised for other purposes, with all changes to be made well before any plant reaches 60 years of operation

AMP Attribute Assessment (BWRVIP-315, Section 5)

AMP Attribute Assessment

- Reviews the BWRVIP reactor internals aging management program against the 10 elements of an effective AMP
- Purpose is to identify those elements that have time-dependency and therefore could potentially be impacted by consideration of extended operations

AMP Element Assessment for Extended Operations

Element	Extended Operations Impact?	Summary Basis
1) Scope of Program	No	The scope of SCs managed by the AMP are fixed. Adjustments may occur, but are not likely to be the result of time-dependent factors.
2) Preventive Actions	N/A	The BWRVIP Reactor Internals AMP is not credited to provide preventive actions.
3) Parameters Monitored or Inspected	No	The nature of the program inspections performed to detect the effects of aging are equally applicable to any operating time period.
4) Detection of Aging Effects	No	The methods used to detect aging effects are equally applicable to any operating time period. Program guidance related to inspection frequency and sample populations are interval-based.
5) Monitoring and Trending	Yes	Elements associated with monitoring and trending may be based upon assessments of degradation risk which may be indirectly dependent on plant service time. Guidance clarifications and enhancements are specified within BWRVIP-315 to ensure program applicability for extended operations.

AMP Element Assessment for Extended Operations (continued)

Element	Extended Operations Impact?	Summary Basis
6) Acceptance Criteria	Yes	Guidance must allow for consideration of changes in material properties associated with accumulated neutron fluence or other time-dependent factors.
7) Corrective Actions	No	Processes for corrective actions are not time-dependent and are addressed by the plant owner's Quality Assurance Program (e.g., 10 CFR 50 Appendix B for U.S. owners).
8) Confirmation Process	No	Confirmation processes are not time-dependent and are addressed by the plant owner's Quality Assurance Program (e.g., 10 CFR 50 Appendix B for U.S. owners).
9) Administrative Controls	No	Each plant owner defines their own appropriate administrative controls. In addition, in implementing the BWRVIP program under the NEI 03-08 materials initiative, the plant owner is further subjected to the administrative controls provided under BWRVIP-94NP (currently Revision 3). Regardless, administrative controls are not time-dependent.
10) OE	No	Processes for managing operating experience are not time-dependent.

AMP Attribute Assessment Conclusions

- Only Elements 5 (monitoring and trending) and 6 (acceptance criteria) are affected by consideration of extended operations for the BWR reactor internals AMP
- For the other elements, previous evaluations concluding AMP adequacy remain applicable and need not be revisited
- With regard to Elements 5 and 6:
  - Evaluations provided in Sections 3 and 4 of BWRVIP-315 provide the bases that support continued adequacy of the BWRVIP reactor internals AMP for extended operations
  - Appropriate guidance clarifications and enhancements are identified in Section 4 and proposed guidance changes are detailed within Appendix B of BWRVIP-315
- Program elements related to review of operating experience and program administration are concluded to provide appropriate structure to address unanticipated adverse trends

BWRVIP-315 Status & Plans for Submittal

BWRVIP-315 Status & Plans for Submittal

- *EPRI Report 3002012535, BWRVIP-315: BWR Vessel and Internals Project - Reactor Internals Aging Management Evaluation for Extended Operations*
  - Published July 2019
  - Preparation of Proprietary and Non-Proprietary versions in progress
  - BWRVIP intends to submit BWRVIP-315 to NRC for review and approval in October 2019

- BWRVIP requests that review of BWRVIP-315 be assigned a lower priority than the reviews being requested for other SLR-related reports:
  - BWRVIP-321, *Plan for Extension of the BWR Integrated Surveillance Program (ISP) Through the Second License Renewal (SLR)*
  - BWRVIP-329, *Updated Probabilistic Fracture Mechanics Analyses for BWR RPV Welds to Address Extended Operations*

Additional Topics

Additional Topics

- Management of Cracking due to LCF / EAF
- SRP-SLR and GALL-SLR Further Evaluation Items
- BWRVIP Position on License Renewal Appendices

Management of Cracking due to Low-Cycle Fatigue / Environmentally-Assisted Fatigue

Background

- There are questions associated with evaluation of environmentally assisted fatigue (EAF) for BWR internals
  - Cumulative usage factor (CUF) values generally not available
    - Exceptions are:
      - Internals for newer (BWR/6) plants, where internals were evaluated to the requirements of ASME Section III Subsection NG, *Core Support Structures*
      - Internals repairs or replacements
      - Other internals where updated analyses were performed
- To address these questions, the BWRVIP has developed a basis for how the existing guidance provided by the BWRVIP inspection-based reactor internals program addresses EAF
  - The basis describes how the BWRVIP program adequately fulfills the guidance currently defined in NUREG-2191, *Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report*, for managing EAF
  - As a result, EAF CUF (CUF_{en}) calculations to disposition fatigue time limited aging analyses (TLAAs) are not necessary

Objective of the EAF Internals Technical Basis

- The primary basis for this approach is that IGSCC and IASCC are significantly more limiting for reactor internals than cracking caused by EAF
- The BWRVIP guidance assumes the potential for SCC occurrence and subsequent growth
  - In a number of cases, there are significant structural redundancies (e.g., top guide, core plate)
  - In all cases, evaluation methods are provided to disposition crack growth due to SCC
- Monitoring CUF values for components known (or presumed) to contain cracking has limited value
- The BWRVIP performed sample crack growth rate calculations to document that any fatigue crack growth (FCG) occurring in BWR reactor internals is accommodated by the conservatisms contained within existing BWRVIP IGSCC and IASCC crack growth correlations

Background of the EAF Internals Technical Basis

- Table B1, Item IV.B1.R-53 of the GALL-SLR Report identifies that one applicable aging effect and mechanism for BWR RPV internal components fabricated from stainless steel and nickel alloys is cumulative fatigue damage and cracking due to fatigue cyclical loading
  - This effect/mechanism may be adequately managed as a time-limited aging analysis (TLAA) under NUREG-2192, *Standard Review Plan for SLR (SRP-SLR)*, Section 4.3, "Metal Fatigue"
- Disposition can be addressed by:
  - Demonstrating the analysis remains valid for the period of extended operation (10 CFR 54.21(c)(1)(i))
  - Projecting the analysis to the end of the period of extended operation (10 CFR 54.21(c)(ii))
    OR
  - **Demonstrating that the effects of aging will be adequately managed for the period of extended operation (10 CFR 54.21(c)(1)(iii))**
- **The BWRVIP Program is adequate to manage cracking of reactor internals due to either SCC or fatigue, with SCC being by far the more limiting mechanism**

Six Elements of the EAF Internals Technical Basis (1 of 2)

1. BWRVIP Inspection Requirements
   - BWRVIP I&E guidelines address management of BWR reactor internal components and provide confidence in continued structural integrity and reasonable assurance that BWR RPV internals will continue to perform their intended function(s)
2. BWRVIP Inspection Methods
   - The techniques described in BWRVIP-03 are primarily for the detection and characterization of service-induced cracking, such as IGSCC and IASCC
   - Volumetric methods must be performed using techniques that have been demonstrated to be adequate to detect SCC in the component / geometry
3. Application of BWRVIP Inspections to Detect and Characterize Fatigue Cracking
   - The methods applied by the BWRVIP reactor internals AMP to detect cracking have been shown capable of detecting even small cracks and are are suitable for detecting fatigue cracks

Six Elements of the EAF Internals Technical Basis (2/2)

4. Significance of Crack Growth
   - The controlling cracking mechanism evaluated in the determination of BWR RPV internal component inspection intervals is SCC
     - Effective EAF crack growth rates for BWR reactor internals are significantly less than SCC growth rates
     - In all cases (for all materials and for both NWC and HWC environments), BWRVIP SCC CGR correlations are at least two orders of magnitude larger than fatigue CGRs at similar K / ΔK values
5. Potential for Flaw Initiation
   - For welded core support structures, the BWRVIP program for reactor internals includes the presumption that crack initiation is plausible and flaw evaluation methods use conservative upper end CGRs that encompass the potential for cracking due to EAF
6. Operating Experience
   - The BWRVIP program monitors and evaluates relevant operating experience and, as necessary, enhances program requirements or develops new program requirements to manage the effects of aging
   - The combination of BWRVIP activities and licensee OE program procedures provides a basis for concluding that any unexpected cracking will be reviewed and aging management requirements will be enhanced as appropriate to the observed degradation

Comparison of SCC and FCG Rates

[[

Content Deleted
EPRI Proprietary Information

]]

Conclusions of the EAF Internals Technical Basis

- Inspection intervals and methods established by the BWRVIP for management of IGSCC are limiting and provide adequate assurance that cracking due to EAF in BWR reactor internals will be managed
- Additional margin against failure exists within the BWRVIP flaw evaluation procedures since the generic fracture mechanics evaluations apply safety factors against failure and use appropriately conservative SCC crack growth correlations
- Consideration of crack occurrence due to EAF or growth of cracks due to FCG is accommodated by the conservatisms in the BWRVIP reactor internals AMP

SRP-SLR and GALL-SLR Further Evaluation Items

SRP-SLR and GALL-SLR Further Evaluation Items

- The Standard Review Plan for SLR (NUREG-2192) includes three "further evaluation" (FE) items relevant to BWR internals that must be addressed in SLRAs:
  - IASCC (3.1.2.2.12)
  - CASS Embrittlement (3.1.2.2.13)
  - Irradiation-enhanced stress relaxation of core plate holddown bolts (3.1.2.2.14)
- This conclusion is consistent with the BWRVIP position regarding degradation mechanisms impacted by consideration of extended operations
- BWRVIP positions for each further evaluation item generally outlined in preceding slides
  - IASCC managed by continued implementation of I&E guidance with clarifications related to fluence limits for flaw evaluations
  - CASS Embrittlement addressed by fluence limits established consistent with BWRVIP-234-A
  - BWRVIP-25, Rev. 1 provides a generic method for addressing stress relaxation of holddown bolts

BWRVIP Position on License Renewal Appendices

Background

- License renewal appendices have been developed for a significant number of BWRVIP reports, including most I&E Guidelines
- Initially developed for the purpose of demonstrating that the report provided the information needed to meets the technical information requirements of 10 CFR 54.
- Concept originated and most LR appendices developed before availability of the GALL Report
- Provided an option for plants to incorporate BWRVIP reports by reference into plant-specific integrated plant assessments (IPAs), thereby simplifying application content and NRC review
- LR Appendices were developed for most BWRVIP I&E Guidelines, exceptions are:
  - BWRVIP-75-A, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (*not a typical I&E guideline*)
  - BWRVIP-205, Bottom Head Drain Line (*baseline exam requirements only, additional inspections to be addressed by FAC Program*)
- By the time of initial GALL Report issue, majority of LR Appendices were already prepared and approved by NRC

I&E Guideline LR Appendix Submittals and Approvals

Document	Component	Submittal	Approval
BWRVIP-49	RPV Instrument Penetrations	Mar 1998	Sep 1999
BWRVIP-27	Standby Liquid Control	Apr 1997	Dec 1999
BWRVIP-18	Core Spray	Dec 1996	Jan 2001
BWRVIP-25	Core Plate	July 1997	Jan 2001
BWRVIP-26	Top Guide	July 1997	Jan 2001
BWRVIP-41	Jet Pump	Oct 1997	Jun 2001
BWRVIP-42	LPCI Coupling	Dec 1997	Jan 2001
BWRVIP-47	Lower Plenum	Dec 1997	Jan 2001
BWRVIP-48	RPV ID Attachment Welds	Feb 1998	Jan 2001
BWRVIP-38	Shroud Support	Sep 1997	Mar 2001
BWRVIP-74	RPV License Renewal Guide	Sep 1999	Oct 2001
BWRVIP-76	Core Shroud	Nov 1999	Nov 2009
BWRVIP-139	Steam Dryer	Feb 2014	Jan 2017
BWRVIP-180	Access Hole Cover	Jul 2009	N/A (info only)
BWRVIP-183	Top Guide Grid Beams	Jul 2009	N/A (withdrawn)

- Relevant NUREG-1801 Dates:
  - Jul 2001 (Initial)
  - Sep 2005 (Rev. 1)
  - Dec 2010 (Rev. 2)
- BWR "Pre-GALL" LRA submittals:
  - Mar 2000 (Hatch)
  - July 2001 (Peach Bottom)
- Other LR Appendices:
  - BWRVIP-108 / 241 (Nozzle ISI Relief)

BWRVIP Perspective on LR Appendices

- Availability of the GALL Report effectively eliminated the need for LR Appendices
  - GALL Report recommends use of BWRVIP guidelines to manage BWR reactor vessels and reactor internals
  - Cited by applicants within plant-specific LRAs, evaluations based on consistency with GALL
  - Same approach taken regardless of LR Appendix submittal / approval status
- BWRVIP-315 provides a license renewal based aging management evaluation of BWR reactor internals
- Significant resources would be required to update and maintain the appendices and associated NRC SEs, without a significant change in the overall level of safety provided
- Current BWRVIP practice is to make LR Appendices historical whenever the corresponding I&E guideline is revised
  - "Time-dependent" limitations specifically affecting use of the guidance clearly documented in the main body of each affected guidance document (*as recommended in Appendix B of BWRVIP-315*)

Together...Shaping the Future of Electricity