PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG-15035-NP
Revision 0

WESTINGHOUSE NON-PROPRIETARY CLASS 3

Responses to NRC Requests for Additional Information (ADAMS ML15005A052) Related to WCAP-17096-NP Revision 2

Materials Committee

PA-MSC-0473

April 2015

Westinghouse

PWROG-15035-NP
Revision 0

Responses to NRC Requests for Additional Information (ADAMS ML15005A052) Related to WCAP-17096-NP Revision 2

PA-MSC-0473

Daniel B. Denis*, RAI 3
Materials Center of Excellence
Eric M. Benacquista*, RAI 4
Reactor Internals Design & Analysis I
Bradley T. Carpenter*
Reactor Internals Aging Management
Ernest W. Deemer*, RAI 5
Reactor Internals Aging Management
Joshua K. McKinley*, RAI 6
Materials Center of Excellence

April 2015

Verifier, RAI 3: John F. Kielb*
Reactor Internals Design & Analysis II
Verifier, RAI 4: Ibrahim M. Ezzi*
Reactor Internals Design & Analysis I
Verifier, RAI 5: Thomas E. Demers*
Major Reactor Component Design & Analysis I
Verifier, RAI 6: Michael R. Ickes*
Reactor Internals Aging Management
Reviewer, All: Randy G. Lott*, Consulting Engineer
Primary Systems Design and Repair

Approved: Patricia C. Paesano*, Manager
Reactor Internals Aging Management
Approved: James P. Molkenthin*, Program Director
PWR Owners Group PMO

*Electronically approved records are authenticated in the enterprise document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

ACKNOWLEDGEMENTS

This report was developed and funded by the Pressurized Water Reactor Owners Group (PWROG) under the leadership of the participating utility representatives of the Materials Committee. It is a project deliverable under PWROG Project Authorization PA-MSC-0473, Revision 5.

LEGAL NOTICE

COPYRIGHT NOTICE

DISTRIBUTION NOTICE

PWR Owners Group
United States Member Participation* for PA-MSC-0473, Revision 5

Utility Member	Plant Site(s)	Participant	
		Yes	No
Ameren Missouri	Callaway (W)	X	
American Electric Power	D.C. Cook 1 & 2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)	X	
Dominion Connecticut	Millstone 2 (CE)	X	
	Millstone 3 (W)	X	
Dominion VA	North Anna 1 & 2 (W)	X	
	Surry 1 & 2 (W)	X	
Duke Energy Carolinas	Catawba 1 & 2 (W)	X	
	McGuire 1 & 2 (W)	X	
	Oconee 1, 2, & 3 (B&W)	X	
Duke Energy Progress	Robinson 2 (W)	X	
	Shearon Harris (W)	X	
Entergy Palisades	Palisades (CE)	X	
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)	X	
Entergy Operations South	Arkansas 1 (B&W)	X	
	Arkansas 2 (CE)	X	
	Waterford 3 (CE)	X	
Exelon Generation Co. LLC	Braidwood 1 & 2 (W)	X	
	Byron 1 & 2 (W)	X	
	TMI 1 (B&W)	X	
	Calvert Cliffs 1 & 2 (CE)	X	
	Ginna (W)	X	
FirstEnergy Nuclear Operating Co.	Beaver Valley 1 & 2 (W)	X	
	Davis-Besse (B&W)	X	
Florida Power & Light \ NextEra	St. Lucie 1 & 2 (CE)	X	
	Turkey Point 3 & 4 (W)	X	
	Seabrook (W)	X	
	Pt. Beach 1 & 2 (W)	X	
Luminant Power	Comanche Peak 1 & 2 (W)	X	

PWR Owners Group
United States Member Participation* for PA-MSC-0473, Revision 5

Utility Member	Plant Site(s)	Participant Yes	Participant No
Omaha Public Power District	Fort Calhoun (CE)	X	
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	X	
PSEG – Nuclear	Salem 1 & 2 (W)	X	
South Carolina Electric & Gas	V.C. Summer (W)	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)	X	
Southern Nuclear Operating Co.	Farley 1 & 2 (W)	X	
	Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2 (W)	X	
	Watts Bar 1 & 2 (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	
Xcel Energy	Prairie Island 1 & 2 (W)	X	

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group
International Member Participation* for PA-MSC-0473, Revision 5

Utility Member	Plant Site(s)	Participant Yes	Participant No
Asociación Nuclear Ascó-Vandellòs	Asco 1 & 2 (W)	X	
	Vandellos 2 (W)	X	
Axpo AG	Beznau 1 & 2 (W)	X	
Centrales Nucleares Almaraz-Trillo	Almaraz 1 & 2 (W)	X	
EDF Energy	Sizewell B (W)	X	
Electrabel	Doel 1, 2 & 4 (W)	X	
	Tihange 1 & 3 (W)	X	
Electricite de France	58 Units	X	
Eletronuclear-Eletrobras	Angra 1 (W)	X	
Eskom	Koeberg 1 & 2 (W)	X	
Hokkaido	Tomari 1, 2 & 3 (MHI)	X	
Japan Atomic Power Company	Tsuruga 2 (MHI)	X	
Kansai Electric Co., LTD	Mihama 1, 2 & 3 (W)	X	
	Ohi 1, 2, 3 & 4 (W & MHI)	X	
	Takahama 1, 2, 3 & 4 (W & MHI)	X	
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 (W)	X	
	Hanbit 1 & 2 (W)	X	
	Hanbit 3, 4, 5 & 6 (CE)	X	
	Hanul 3, 4 , 5 & 6 (CE)	X	
Kyushu	Genkai 1, 2, 3 & 4 (MHI)	X	
	Sendai 1 & 2 (MHI)	X	
Nuklearna Electrarna KRSKO	Krsko (W)	X	
Ringhals AB	Ringhals 2, 3 & 4 (W)	X	
Shikoku	Ikata 1, 2 & 3 (MHI)	X	
Taiwan Power Co.	Maanshan 1 & 2 (W)	X	

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

TABLE OF CONTENTS

1 INTRODUCTION AND BACKGROUND

This report provides responses to U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAIs) [1] related to their review of WCAP-17096-NP [2] and previously submitted text updates to the original Westinghouse Commercial Atomic Power topical report (WCAP). The current NRC RAIs [1] provide direction to the specific source for the text requiring clarification.

Responses were developed under PWROG Project Authorization PA-MSC-0473 [3].

REFERENCES

1. U.S. Nuclear Regulatory Commission Letter, "Request for Additional Information Related to WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements' (TAC NO. ME4200)," February 11, 2015 (ADAMS Accession No. ML15005A052).

2. Westinghouse Report, WCAP-17096-NP, Rev. 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 2009.

3. PWROG Project Authorization, PA-MSC-0473, Rev. 5, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," October 2013. (Available from PWROG website.)

4. EPRI Letter, "Proposed Edits to WCAP-17096-NP, Revision 2," April 10, 2014. (ADAMS Accession No. ML14104B579)

5. EPRI Letter, "Proposed Edits to WCAP-17096-A Draft," August 5, 2013. (ADAMS Accession No. ML13219A183)

6. PWROG Letter, "Submittal of WCAP-17451-P, Revision 1, 'Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections' to the NRC for Information Only (PA-MSC-0688)," February 2015. (ADAMS Accession No. ML15041A106)

2 RAI RESPONSES

Responses to individual NRC RAIs provided in [1] are provided in this section. Each section contains the RAI exactly as transmitted by the NRC, followed by the proposed response and proposed WCAP revisions where applicable. If the RAI response does not require a revision to the WCAP, it is indicated as such at the end of the response. Where applicable, references cited in the response to a specific RAI are sourced in Section 1 of this document. RAIs in general are on the revised text provided in a previous RAI response to the original WCAP text submitted for NRC review. The current individual NRC RAIs [1] provide direction to the specific source for the text requiring clarification.

2.1 RESPONSE TO NRC RAI 3

NRC RAI 3

For the core support barrel assembly – lower core barrel flange welds (CE-ID: 9), Lower Support Structure – Core Support Plate (CE-ID: 10), Upper Internals Assembly – Fuel Alignment Plate (CE-ID: 11) and lower support structure - deep beams (CE-ID: 13), MRP-227-A allows performing a time-limited aging analysis (TLAA) to demonstrate acceptable fatigue life in lieu of inspection. In RAI 41, the staff asked if acceptance criteria based on ensuring structural integrity until the next inspection could be added to the evaluation methodologies in WCAP-17096-NP. In response to RAI 41, Electric Power Research Institute (EPRI) indicated such acceptance criteria would be added to the evaluation procedures for these components.

EPRI provided revised evaluation procedures for Lower Support Structure – Core Support Plate (CE-ID: 10), Upper Internals Assembly – Fuel Alignment Plate (CE-ID: 11) in its January 16, 2014, letter (Ref. 1) and provided revised evaluation procedures for the core support barrel assembly – lower core barrel flange welds (CE-ID: 9) and lower support structure - deep beams (CE-ID: 13) in its April 10, 2014, letter (Ref. 2). The revised evaluation procedures all included acceptance criteria for the evaluation of cracks found if the inspection is performed. However, the revised evaluation procedures for core support barrel assembly – lower core barrel flange welds (CE-ID: 9) and lower support structure - deep beams (CE-ID: 13) also provided guidance for performing the TLAA.

It is not clear to the staff that guidance for performing TLAAs in lieu of inspection for certain components is within the scope of WCAP-17096-NP as it is stated in Section 1 of the report.

Requested Information

1. If guidance for performing TLAAs for the four components listed above is within scope of WCAP-17096-NP,
    a. Modify the objective section of WCAP-17096-NP to clarify that guidance for TLAAs is within the scope of the document.
    b. Modify the evaluation procedures for the Core Support Plate (CE-ID: 10) and Upper Internals Assembly – Fuel Alignment Plate (CE-ID: 11) to include guidance for performing the TLAA similar to the other two components.
2. If guidance for performing TLAAs for the four components listed above is not within the scope of WCAP-17096-NP, remove this guidance from the evaluation procedures for the core support barrel assembly – lower core barrel flange welds (CE-ID: 9) and lower support structure - deep beams (CE-ID:13).

RAI 3 Response

As stated in "Option 2" of the NRC RAI 3, TLAA information will be removed. It is not the intent of MRP-227-A or WCAP-17096-NP to provide guidance on performing TLAAs. Markups for the CE-ID: 9 and CE-ID: 13 sections indicating revisions to remove the TLAA methodology

discussion have been provided. Revisions to the CE-ID: 10 and CE-ID: 11 sections are not required.

RAI 3 Revisions to WCAP-17096-NP Appendix C

The proposed changes will be implemented in the text in the following sections of WCAP-17096-NP.

Appendix C components as proposed in [4]:

CE-ID: 9 Core Support Barrel Assembly – Lower Flange Flexure Weld

CE-ID: 13 Lower Support Structure – Deep Beams

CE-ID: 9 **Core Support Barrel Assembly**

Lower Flange Flexure Weld

Category:	Primary	Applicability:	All plants where flexures exist
Degradation Effect:	Cracking (fatigue)		
Expansion Link:	None		
Function:	Primary core support structure		

Inspection

Method:	If fatigue life cannot be demonstrated by ~~time-limited aging analysis (TLAA)~~evaluation, enhanced visual (EVT-1) examination is required no later than two refueling outages from the beginning of the license renewal period. Subsequent examination is required on a 10-year interval.
Coverage:	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See MRP-227-A Figure 4-16 (flange and flexure).
Observable Effect:	The specific relevant condition is a detectable crack-like indication.

Inputs and Assumptions

There are several inputs and assumptions that are critical to the development of acceptance criteria for the RV Internals Core Support Barrel weld locations. These items are stated below:

- The inspections identified in MRP-227-A are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is intended to provide an adequate sampling for monitoring of fatigue.
- The change in resistance to fracture of the RV core barrel welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.
- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to

CE-ID: 9 **Core Support Barrel Assembly**

Lower Flange Flexure Weld

accumulate over the licensed operating lifetime. Since there has been no experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in BWRs are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.

- For weld locations subjected to fluence less than or equal to $5x10^{20}$ n/cm^2 (E>1MeV), the boiling water reactor (BWR) hydrogen water chemistry (HWC) crack growth equation specified in paragraph C-8520 of Appendix C of Section XI of the 2010 edition of the ASME Boiler and Pressure Vessel Code is appropriate. This crack growth rate model is consistent with the model in BWRVIP-14-A, which was previously reviewed and approved by the NRC.
- For fluence levels at or above $5x10^{20}$ n/cm^2 (E>1MeV), the BWR HWC crack growth equation specified in equation 6-5 of MRP-227-A is appropriate.

- ~~Depending on the magnitude~~Additional crack growth as a result of cyclic loads, such as thermal transients, ~~additional crack growth as a result of these loads~~ may need to be considered.
- Acceptance criteria for indications detected using a visual exam can be developed based on an assumed through-wall flaw with a length that is uniform through the wall thickness. If it can be shown that the critical crack length for a through wall flaw is less than that for a part through wall flaw, then this assumption is reasonable and conservative, because no information is available on the flaw depth from such a visual examination.
- Acceptance criteria can be developed for the entire 60-year license of a given plant by using predicted end-of-license fluence values. However, if there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable until the end of the 60-year license.

Failure

Failure Mechanism: Fatigue

Failure Effect: Loss of core support

Failure Criteria: ~~TLAA cannot demonstrate the fatigue usage factor is less than 1.0 at the next inspection and EVT-1 inspection determines that a~~An existing flaw is present that is projected to exceed an allowable length which would cause violation of the functional requirements of the lower support structure prior to the next inspection.

Methodology

Goal:

1. Demonstrate that observed flaws will not exceed the maximum allowable crack length for limiting loads that would occur under the current licensing basis (CLB) before the next inspection interval is reached.~~TLAA – demonstrate weld fatigue usage factor is less than 1.0 for all normal and upset conditions while considering environmental effects~~.
2. For pre-inspection analysis, assume a flaw is present at the most limiting

CE-ID: 9 **Core Support Barrel Assembly**

Lower Flange Flexure Weld

location in the weld. For evaluation of a discovered flaw, evaluate a flaw at the location and orientation of the existing flaw.~~If TLAA determines fatigue usage factor may exceed 1.0 or utility chooses to perform inspection in lieu of performing TLAA:~~

~~a. Pre Inspection: Perform a fracture mechanics evaluation to calculate the maximum allowable crack length that can be tolerated during the current inspection.~~

~~b. Post Inspection: Perform a fracture mechanics evaluation to demonstrate that observed flaws will not exceed the maximum allowable crack length for current licensing basis (CLB) before the next inspection interval is reached.~~

~~**TLAA Methodology:**~~

~~Data Requirements:~~

~~1. Operating loads (e.g. dead weight, pressure, flow, thermal)~~
~~2. Functional requirements of the lower support structure~~
~~3. ASME Design Fatigue Curves~~
~~4. Potential fatigue loading and cycles.~~
~~5. Operating temperatures~~
~~6. NUREG-CR-6909~~

~~Analysis:~~

~~1. Develop a model of the lower internals, including the core support barrel, core support barrel lower flange, and the lower support structure in sufficient detail to determine the stresses in the CSB lower flange flexure weld.~~
~~2. Determine the stresses due to the mechanical and thermal loads for normal, upset, and faulted conditions.~~
~~3. Determine the cumulative usage factor for 60 years operating life using the most recent ASME fatigue curves, and incorporate environmental effects using the fatigue evaluation procedure of Section A3 of NUREG-CR-6909.~~
~~4. If the cumulative usage factor is less than 1.0 no inspection is required.~~

Fracture Mechanics Evaluation Methodology for EVT-1 Inspection Acceptance Criteria:

~~1. For pre-inspection analysis assume a flaw is present at the most limiting location in the weld. For evaluation of a discovered flaw, evaluate a flaw at the location and orientation of the existing flaw.~~
~~2. Perform a fracture mechanics evaluation to determine the maximum permissible flaw size as described below.~~

Data Requirements:

1. Surface crack length determined by visual inspection
   a. The 2007 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the

CE-ID: 9 **Core Support Barrel Assembly**

Lower Flange Flexure Weld

material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length.

2. Flaw Depth
   a. For one-sided visual inspections, the flaw is assumed to be through-wall.
   b. Supplemental examinations may be used to determine flaw depth.
3. Fast neutron fluence at crack location, or confirmation that fast neutron fluence at lower core barrel flange flexure weld is below $3x10^{20}$ n/cm^2 (E> 1 MeV).
4. Steady-state and applicable operating transient stresses to be used to calculate SCC and fatigue crack growth rates.
   a. Axial stresses based on normal mechanical loads are required for circumferential flaws.
   ~~a.~~b. Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws.
   ~~b.~~c. Secondary weld residual and thermal stresses need to be considered in determination of circumferential and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental UT examinations.

5.Limiting externally applied transient stresses to be used to calculate allowable flaw lengths.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations.

Analysis:

All analyses require an assumption of the crack growth expected over the upcoming period of service. The methodology is based on analysis of a through-wall flaw with weld residual and thermal stresses relieved. The crack growth rate models will be based on K dependent crack growth under hydrogen water chemistry conditions. In order to apply the acceptance criteria to a full 10-year inspection interval, follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement. For detailed crack growth models used, see "Inputs and Assumptions" above.

~~Depending~~ Additional ~~on the magnitude~~fatigue crack growth as a result of cyclic loads, such as thermal transients, ~~additional crack growth as a result of these loads may need to~~should be considered.

Failure of the weld is assumed to occur when unstable circumferential crack growth is initiated from the analyzed flaw. Two options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection or generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

CE-ID: 9 **Core Support Barrel Assembly**

Lower Flange Flexure Weld

Fluence Range (n/cm^2 E>1MeV)	**Dose (dpa)**	**MRP-227-A Requirement**	**Suggested Pre-Inspection Analysis**	**Suggested Flaw Specific Analysis**
$\leq 3 \times 10^{20}$	≤ 0.5	Limit Load	LEFM using 150 ksi$\sqrt{in}$ Fracture Toughness Value or Limit Load	Limit Load

Different evaluation options may be used depending upon the plant- specific fluence levels at the location of the weld being evaluated. Option 1, though conservative, can be used for all fluence levels.

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow- induced vibration (FIV) is below the threshold for fatigue crack growth.
d. For normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77.
e. For the governing emergency or faulted loading condition, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39.

Option 2: Limit Load Analysis (For neutron fluence < 3×10^{20} n/cm^2 at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.
b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.
c. Ensure that the ΔK resulting from flow- induced vibration (FIV) is below the threshold for fatigue crack growth.
d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.
e. The applied moment, increased by a factor of 1.39 (for emergency and faulted conditions) or 2.77 (for normal and upset conditions) must be less than the limit moment (from step C) for the flaw length determined in (b), for the flaw to be acceptable.

CE-ID: 9 **Core Support Barrel Assembly**

Lower Flange Flexure Weld

Acceptance Criteria: ~~TLAA demonstrates that fatigue usage factor remains less than 1.0 for all normal and upset conditions throughout the license extension period.~~

~~If inspection is required based on TLAA results, p~~Projected crack growth does not violate lower support structure functional requirements.

Approach: ~~TLAA (plant specific) to assess the need for inspection.~~

~~If inspection is required or selected, determine~~ Determine allowable flaw length using fracture mechanics evaluation.

CE-ID: 13 **Lower Support Structure**

Deep Beams

Category: Primary Applicability: All plants with core shrouds assembled with full-height shroud plates

Degradation Effect: Cracking (fatigue) that results in a detectable surface-breaking indication in the welds or beams. Aging Management (IE)

Expansion Link: None

Function: Primary core support structure

Inspection

Method: Enhanced visual (EVT-1) examination, no later than two refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval, if adequacy of remaining fatigue life cannot be demonstrated.

Coverage: Examine beam-to-beam welds in the axial elevation from the beam top surface to four inches below. See MRP-227 Figure 4-19.

Observable Effect: Fatigue crack growth along ~~welds at beams~~beams at welds. Check for a detectable surface-breaking indication in the welds or beams.

Inputs and Assumptions

There are several inputs and assumptions that are critical to the development of acceptance criteria for the deep beam locations. These items are stated below:

- The inspections identified in MRP-227-A are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is intended to provide an adequate sampling for monitoring of fatigue.
- The change in resistance to fracture of the deep beam welds can be correlated to the accumulated fluence at each weld location. Welds that are subject to low fluence are considered to have a high degree of resistance to fracture. Correspondingly, those welds subject to high fluence have lower resistance to fracture.

CE-ID: 13 **Lower Support Structure**

Deep Beams

- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been no experience of SCC initiated cracks in operating PWRs to date, growth rates developed for the prediction of SCC in BWRs are assumed to be appropriate for prediction of crack growth due to SCC in PWR reactor internals.
  - For weld locations subjected to fluence less than or equal to $5x10^{20}$ n/cm^2 (E>1MeV), the boiling water reactor (BWR) hydrogen water chemistry (HWC) crack growth equation specified in paragraph C-8520 of Appendix C of Section XI of the 2010 edition of the ASME Boiler and Pressure Vessel Code is appropriate. This crack growth rate model is consistent with the model in BWRVIP-14a, which was previously reviewed and approved by the NRC.
  - For fluence levels at or above $5x10^{20}$ n/cm^2 (E>1MeV), the BWR HWC crack growth equation specified in equation 6-5 of MRP-227-A is appropriate.
- Since the degradation effect is fatigue, low and high cycle fatigue crack growth should be evaluated. Inputs to consider include the magnitude of cyclic loads, such as thermal transients.
- Acceptance criteria can be developed for the entire 60-year license of a given plant by using predicted end-of-license fluence values. However, if there are changes to these fluence projections, such as in the event of a power uprate or change in core loading pattern, it would be necessary to confirm that the inputs selected based on fluence, such as SCC growth rate and fracture toughness, remain applicable until the end of the 60-year license.

Failure

Failure Mechanism: Cracking (fatigue)

Failure Effect: Loss of fuel assembly alignment

Failure Criteria: ~~TLAA cannot demonstrate the fatigue usage factor is less than 1.0 at the next inspection and EVT-1 inspection determines that a~~An existing flaw is present that is projected to exceed an allowable length, which would cause violation of the functional requirements of the lower support structure prior to the next inspection.

Methodology

Goal:

1. Demonstrate that observed flaws will not exceed the maximum allowable crack length for limiting loads that would occur under the current licensing basis (CLB) before the next inspection interval is reached.~~TLAA – demonstrate weld fatigue usage factor is less than 1.0 for all normal and upset conditions while considering environmental effects.~~
2. For pre-inspection analysis, assume a flaw is present at the most limiting location in the weld. For evaluation of a discovered flaw, evaluate a flaw at the location and orientation of the existing flaw.

~~1. ,If TLAA determines fatigue usage factor may exceed 1.0 or utility chooses to perform inspection in lieu of performing TLAA:~~
   ~~a. Pre Inspection: Perform a fracture mechanics evaluation to calculate the~~

CE-ID: 13 **Lower Support Structure**

Deep Beams

~~maximum allowable crack length that can be tolerated during the current inspection.~~

~~b.~~a. ~~Post Inspection: Perform a fracture mechanics evaluation to demonstrate that observed flaws will not exceed the maximum allowable crack length for current licensing basis (CLB) before the next inspection interval is reached.~~

~~**TLAA Methodology:**~~

~~Data Requirements:~~

~~1. Operating loads (e.g. dead weight, pressure, flow, thermal)~~
~~2. Functional requirements of the lower support structure~~
~~3. ASME Design Fatigue Curves~~
~~4. Potential fatigue loading and cycles.~~
~~5. Operating temperatures~~
~~6. NUREG-CR-6909~~

~~Analysis:~~

~~1. Develop a model of the lower internals, including the core support barrel, core support barrel lower flange, and the lower support structure in sufficient detail to determine the stresses in the deep beams.~~
~~2. Determine the stresses due to the mechanical and thermal loads for normal, upset, and faulted conditions.~~
~~3. Determine the cumulative usage factor for 60 years operating life using the most recent ASME fatigue curves, and incorporate environmental effects using the fatigue evaluation procedure of Section A3 of NUREG-CR-6909~~
~~4. If the cumulative usage factor is less than 1.0 no inspection is required.~~

Fracture Mechanics Evaluation Methodology for EVT-1 Inspection Acceptance Criteria:

~~1. For pre-inspection analysis assume a flaw is present at the most limiting location in the weld. For evaluation of a discovered flaw evaluate a flaw at the location and orientation of the existing flaw.~~
~~2. Perform a fracture mechanics evaluation to determine the maximum permissible flaw size as described below.~~

Data Requirements:

1. Surface crack length determined by visual inspection
    a. The 2007 edition of the ASME Code Section XI: IWA-3330(a) and Figure IWA-3330-1 provide requirements for the minimum allowable separation distance that will serve as the basis for guidance in this report. The requirement in Section XI is that the ligament between adjacent cracks must be greater than half of the thickness of the material. If this criterion is not met, the individual crack lengths and the length of the ligament between the cracks must be summed. This total length is then compared to the allowable length.
2. Flaw Depth
    a. For one-sided visual inspections, the flaw is assumed to be through-wall.
    b. Supplemental examinations may be used to determine flaw depth.

CE-ID: 13 Lower Support Structure

Deep Beams

3. Fast neutron fluence at crack location, or confirmation that fast neutron fluence at the deep beams is below $3x10^{20}$ n/cm^2 (E> 1 MeV).
4. Steady-state and applicable operating transient stresses to be used to calculate SCC and fatigue crack growth rates.
   a. Horizontal stresses based on normal mechanical loads are required for vertical flaws.
   b. Secondary weld residual and thermal stresses need to be considered in determination of ~~circumferential and~~ through-wall crack growth rates in partial through-wall flaws.
5. Limiting externally applied transient stresses to be used to calculate allowable flaw lengths.
   a. More detailed load-deformation histories may be required for elastic-plastic or limit load calculations.

Analysis:

All analyses require an assumption of the crack growth expected over the upcoming period of service. The methodology is based on analysis of a through-wall flaw with weld residual and thermal stresses relieved. The crack growth rate models will be based on K dependent crack growth under hydrogen water chemistry conditions. In order to apply the acceptance criteria to a full 10-year inspection interval, follow-up action is required to verify the assumptions used in the predicted crack-growth rate. A re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement.

Failure of the deep beam~~s~~ welds is assumed to occur when unstable crack growth is initiated from the analyzed flaw. Two options are outlined for determining the limiting allowable flaw length, based on neutron dose. Analysis methods are suggested for both pre-inspection and generic analysis (Suggested Pre-Inspection Analysis) and for flaws observed in-service (Suggested Flaw Specific Analysis), where more detailed characteristics of the flaw and its location are known. In all cases, a more detailed evaluation may be completed using a semi-elliptic surface flaw, but such an evaluation would require more detailed inspection by UT.

Fluence Range (n/cm^2 E>1MeV)	Dose (dpa)	MRP-227-A Requirement	Suggested Pre-Inspection Analysis	Suggested Flaw Specific Analysis
$\leq 3 \times 10^{20}$	≤ 0.5	Limit Load	LEFM using 150 ksi$\sqrt{in}$ for fracture toughness or Limit Load	Limit Load

Different evaluation options may be used depending upon the plant- specific fluence levels at the location of the weld being evaluated. Option 1, though conservative, can be used for all fluence levels.

CE-ID: 13 Lower Support Structure

Deep Beams

Option 1: LEFM Analysis

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow- induced vibration (FIV) is below the threshold for fatigue crack growth.

d. For normal and upset loading conditions, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 2.77.

e. For the governing emergency or faulted loading condition, the final stress intensity factor (K) for the flaw size from (b) must be lower than the fracture toughness for the material by a factor of at least 1.39.

Option 2: Limit Load Analysis (For neutron fluence $< 3 \times 10^{20}$ n/cm^2 at E>1 MeV or dose less than approximately 0.5 dpa)

a. Establish initial crack length (and depth if determined by supplemental examinations) based on inspection results.

b. The final crack dimensions are calculated by adding 10 years of crack growth under normal loading conditions.

c. Ensure that the ΔK resulting from flow- induced vibration (FIV) is below the threshold for fatigue crack growth.

d. Determine the bending moment (M) that can be tolerated as a function of the postulated flaw length.

e. The applied moment, increased by a factor of 1.39 (for emergency and faulted conditions) or 2.77 (for normal and upset conditions) must be less than the limit moment (from step d) for the flaw length determined in (b), for the flaw to be acceptable.

Acceptance Criteria: The deep beams continue to perform their functional requirements with the projected flaw length at the end of the inspection interval.

Approach: ~~Limit load or LEFM analysis~~Determine allowable flaw length using fracture mechanics evaluation.

2.2 RESPONSE TO NRC RAI 4

NRC RAI 4

In its April 10, 2014, letter, EPRI provided a revised evaluation methodology and acceptance criteria for W-ID: 1 Control Rod Guide Tube Assembly – Guide Plates (Cards) (guide cards). The April 10, 2014, letter indicates the revised acceptance criteria are derived from WCAP-17451-P, Rev. 1, and are based on ensuring wear is not severe enough to allow loss of guidance of the control rods that could prevent insertion or stepping of the rod cluster control assembly (RCCA). The revised methodology and acceptance criteria differ significantly from the methodology and acceptance criteria for the guide cards in WCAP-17096-NP, Rev. 2.

The proposed methodology states each utility shall perform an initial "baseline" examination measurement based on the generic inspection schedule provided in WCAP-17451-P, Rev. 1. Inspections shall be performed earlier than the generic schedule for plants as noted in Section 5.4 of WCAP- 17451-P, Rev. 1. No wear measurements prior to 2015 are required. Alternate wear measurement schedules may be developed based on the guidance provided in WCAP-17451-P, Rev. 1.

To complete its review of the revised methodology and acceptance criteria for the guide cards, the staff requires the following additional information (EPRI may choose to provide WCAP-17451-P for information and answer each question by referring to the appropriate section in the WCAP):

1. Provide the basis for the margins included in the acceptance criteria, for example, the selection of 0.8 x DR (where DR is the rodlet diameter), as the maximum allowable measured slot width for wear to be classified as “green,” in the case where similar wear is observed in all the guide cards in the same control rod guide tube assembly.

2. Provide the generic schedule and the basis for the generic schedule inspection dates. Explain why no wear measurements are required prior to 2015.

3. Summarize the operating experience supporting the statement that the largest amounts of wear are typically observed in lowest guide card levels. Provide a justification for only requiring inspection of the lowest six guide cards based on operating experience.

4. Describe the method, and provide the basis for the method, of the predictive wear calculations that will be used to calculate the time until the next inspection for guide cards with wear in the green zone or yellow zone.

5. Provide a diagram showing “ligament wear depth.”

RAI 4 Response

WCAP 17451-P has been provided to the NRC for information via [6]. The WCAP report contains the information requested in RAI 4.

1. Provide the basis for the margins included in the acceptance criteria, for example, the selection of 0.8 x DR (where DR is the rodlet diameter), as the maximum allowable measured slot width for wear to be classified as “green”, in the case where similar

wear is observed in all the guide cards in the same control rod guide tube assembly.

- Response contained in WCAP-17451-P, subsection 5.2.1. Original criteria used 0.9 x rodlet diameter (DR) as a "red" zone. One or two cycles' worth of material was added, along with additional conservatism. As the acceptance criteria process evolved, 0.8 x DR was settled on as the boundary for "green".

2. Provide the generic schedule and the basis for the generic schedule inspection dates. Explain why no wear measurements are required prior to 2015.

   - Response contained in WCAP-17451-P, Section 5.3.

3. Summarize the operating experience supporting the statement that the largest amounts of wear are typically observed in lowest guide card levels. Provide a justification for only requiring inspection of the lowest six guide cards based on operating experience.

   - Response contained in WCAP 17451-P, Section 2.3 and subsection 3.5.1. These sections give examples of Operational Experience (OE) wear data and wear prediction models.

4. Describe the method, and provide the basis for the method, of the predictive wear calculations that will be used to calculate the time until the next inspection for guide cards with wear in the green zone or yellow zone.

   - Response contained in WCAP 17451-P, Section 4.

5. Provide a diagram showing "ligament wear depth."

   - Response contained in WCAP-17451-P, Section 5, Figure 5-1.

RAI 4 WCAP-17096-NP Revisions

No revisions to WCAP-17096-NP required.

2.3 RESPONSE TO NRC RAI 5

NRC RAI 5

In its October 31, 2014, response to the NRC staff's August 18, 2014 RAI 2 related to core barrel weld evaluation procedures (Ref. 4), EPRI stated it would modify the evaluation procedures to state a default verification period of one fuel cycle. The response to RAI 2 also stated that a technical justification will be required for any verification period greater than one fuel cycle. The proposed revised wording is as follows:

"In order to apply the acceptance criteria to a full 10-year inspection interval, additional action is required. The flaw evaluation will address the verification process of the predicted crack growth rate. Depending on the flaw size and knowledge of the plant conditions, a re-inspection at the next refueling outage may be required to provide data needed to operate beyond one cycle. The verification plan shall be included in the evaluation which is submitted to the regulator for their information."

The inspection of the core barrel welds is only required to be performed visually from one side of the weld. Therefore, unless a supplementary volumetric or visual examination is performed to determine the through wall extent of the crack, an observed crack may not be through wall as assumed, thus the crack growth rate could be affected by weld residual stresses and thermal stresses. A re-inspection at a shorter interval to verify crack growth rate assumptions implies that a higher crack growth rate is assumed to set the initial re-inspection interval. However, the revised procedures do not indicate what model would be used to determine the higher crack growth rate.

Requested Information

What crack growth rate model is used to determine the timing of the re-inspection interval? Justify that the assumed crack growth rates are sufficiently conservative to account for uncertainties in crack growth rate, which may result from the action of residual or thermal stresses on a non-throughwall crack.

RAI 5 Response

The crack growth rate model used to determine the timing of the re-inspection interval is the same model used to determine the allowable crack length for 10 years of operation. Both growth rates (summarized in the table below) are Boiling Water Reactor (BWR) Hydrogen Water Chemistry (HWC) crack growth rates. The crack growth rate model is conservatively based on the fluence predicted at the end of the 10-year inspection cycle.

Fluence Range (n/cm^2, E > 1.0 MeV)	Crack Growth Rate (in/hour)
≤ 5E20	$da/dt = 5.31 \times 10^{-9} (K)^{2.181}$
> 5E20	$da/dt = 2.72 \times 10^{-8} (K)^{2.5}$

Per the accompanying response provided for RAI-6 (Section 2.4 of this document), an alternative crack growth rate model is required for cracks growing in material with neutron fluences greater than $3x10^{21}$ n/cm2 (E>1MeV). Such models are currently available from sources such as EPRI technical reports.

The proposed approach evaluates the flaw as a through-thickness flaw. This means that for a given set of loading conditions, mechanical stresses will generate a smaller critical flaw size than would be calculated using a part-through semi-elliptical flaw solution. The approach assumes BWR HWC crack growth rate models for application to PWRs. Fatigue crack growth (FCG) is also included in the overall growth calculations or must be determined to be negligible. The small critical flaw size, the K-dependent BWR HWC SCC crack growth rate model, and consideration of the FCG contribute to the conservatism in the determination of the allowable observed flaw size. If cracks are observed in the core barrel welds, verification of the predicted crack growth after a re-inspection interval provides additional assurance that the crack growth will not exceed the anticipated value.

The crack growth estimate is based on established stress intensity factor (K)-dependent growth rates as described in the methodology. The acceptance criteria contain an allowance for ten years of growth of a through-wall crack. Weld residual and thermal stresses are not included in the calculation of the K value because these secondary stresses are relieved as the crack grows through-wall. However, these secondary stresses can impact crack growth rates of part-through-wall cracks. The impact of the secondary stresses on the crack growth rate will depend on the local value of the stress at the crack tip. Tensile stresses will increase crack growth rates and compressive stresses will decrease crack growth rates. These stresses will impact both the rate at which the flaw propagates through the wall and the rate the crack propagates along the surface.

Secondary stresses are expected to contribute to increased flaw growth near the surface of the core barrel. These secondary stresses would be relieved before the flaw grew through-wall; therefore, secondary stresses would have the greatest effect on shallow flaws. While the part-through-wall flaw could experience growth faster than our predictive model due to the contribution of secondary stresses, the critical flaw length for a part-through-wall flaw would be longer than the critical length for a through-wall flaw for a given set of loads.

Limiting SCC crack growth rates derived from operating experience with BWR HWC environments can be used to account for uncertainties in the SCC growth, pending verification of an acceptable crack growth rate model. These alternative models are beyond the scope of WCAP-17096.

In order to apply the predicted allowable through-wall crack length associated with 10 years of additional crack growth following the inspection, the methodology requires additional action to verify the predicted crack growth rate. If this action is, as anticipated, a follow-up visual examination at the next refueling outage, it will be based on the observed surface crack growth rate. The re-inspection after one additional fuel cycle interval is the shortest practical interval for the follow-up action. During the first refueling cycle (typically 18 months), the crack will only have about 15 percent (18 months / 120 months) of the time necessary to reach the allowable

10 year maximum. This provides considerable margin that the observed crack will not grow beyond the allowable maximum length during the one cycle interval.

The WCAP-17096 methodology contains several other sources of additional margin beyond the crack growth model that help assure that the observed crack will not grow beyond a critical flaw size during one fuel cycle interval:

- The conservative critical flaw size generated using the through-wall flaw assumption.
- The methodology employs conservative fracture toughness values.
- The methodology contains an allowance for ten years of growth in a through-wall flaw.
- The methodology does not credit time required for flaw to grow through-wall.
- The maximum allowable crack size predicted by LEFM (the suggested method) provides conservative results compared to the other available methods such as elastic-plastic fracture mechanics (EPFM) or limit load.

RAI 5 WCAP-17096-NP Revisions

No revisions to WCAP-17096-NP required.

2.4 RESPONSE TO NRC RAI 6

NRC RAI 6

The proposed revised evaluation procedures for the core barrel welds and core support barrel welds listed in the staff's August 18, 2014, RAI (Ref. 5) specify the crack growth rate from MRP-227-A, Equation 6-5 for welds with neutron fluence greater than or equal to $5x10^{20}$ n/cm^2. This crack growth rate equation is the same as that specified in BWRVIP-99-A, "BWR Vessel and Internals Project Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components," for the fluence range of $5x10^{20}$ n/cm^2 to $3x10^{21}$ n/cm^2 at E>1.0 MeV. Since the applicability of this equation is limited to $3x10^{21}$ n/cm^2 or less, but the fluence for some core barrel welds may exceed this fluence, what crack growth rates, or adjustments to the crack growth rates, will be applied for portions of the welds with fluence greater than $3x10^{21}$ n/cm^2? Modify the proposed evaluation procedures as necessary to include this information.

RAI 6 Response

The majority of the welds for which crack growth rate calculations would be used in developing reactor internals acceptance criteria do not receive enough neutron fluence to exceed the limit of $3x10^{21}$ n/cm^2 (E > 1.0 MeV). However, there are some welds that likely or potentially will exceed this threshold. The data available for irradiated austenitic stainless steel crack growth rates in environments relevant to PWR primary water conditions are limited, and none of the studies or compilations of data available have currently been given a safety evaluation report by the staff. Thus, the proposed resolution for creating acceptance criteria for welds or sections of welds that receive greater than $3x10^{21}$ n/cm^2 (E > 1.0 MeV) is to adjust the methodology to require the use and justification of a model applicable to these welds above $3x10^{21}$ n/cm^2 (E > 1.0 MeV). Such models for stainless steel exposed to fluence greater than $3x10^{21}$ n/cm^2 (E > 1.0 MeV) are currently available from sources like Electric Power Research Institute (EPRI) proprietary technical reports, which are available for use by the individual licensee in developing plant-specific acceptance criteria.

RAI 6 WCAP-17096-NP Revisions

The proposed changes will be implemented in the text in the following sections of WCAP-17096-NP:

Appendix C as proposed in [5]:
- CE-ID: 2 Core Shroud Plate-Former Plate Weld
- CE-ID: 2.1 Remaining Axial Welds
- CE-ID: 3 Shroud Plates
- CE-ID: 3.1 Remaining Axial Welds, Ribs and Rings
- CE-ID: 7 Lower Cylinder Girth Welds
- CE-ID: 7.1 Core Barrel Assembly Axial Welds

Appendix E as proposed in [4]:

W-ID: 4 Upper and Lower Core Barrel Girth Welds

W-ID: 4.1 Upper and Lower Core Barrel Cylinder Axial Welds

In all cases, the fourth bullet under "Inputs and Assumptions" will be revised to read:

- The prediction of crack growth is based on the stress intensity factor, K, calculated using linear elastic fracture mechanics. The rate of crack growth is dependent on the amount of neutron fluence that the weld is expected to accumulate over the licensed operating lifetime. Since there has been no experience of IASCC initiated cracks in this or comparable components or welds in operating PWRs to date, growth ~~rates~~ rate models must be obtained from other sources. One source is the models developed for ~~the prediction~~predicting ~~of~~ IASCC crack growth rate (CGR) in ~~BWRs~~ BWR components. The other source is the models developed based on laboratory data that may include test results for both PWR and BWR representative conditions. These models were ~~are assumed~~ considered to be appropriate for the prediction of crack growth due to IASCC in PWR ~~reactor~~ internals.
    - For weld locations subjected to fluence less than or equal to $5x10^{20}$ n/cm^2 (E>1MeV), the ~~boiling water reactor (~~BWR~~)~~ hydrogen water chemistry (HWC) crack growth equation specified in paragraph C-8520 of Appendix C of Section XI of the 2010 edition of the ASME Boiler and Pressure Vessel Code is appropriate. This crack growth rate model is consistent with the model in BWRVIP-~~14a~~14-A, which was previously reviewed and approved by the NRC.
    - For weld locations subjected to fluence levels above $5x10^{20}$ n/cm^2 (E>1MeV) up to and including $3x10^{21}$ n/cm^2 (E>1MeV), the BWR HWC crack growth equation specified in equation 6-5 of MRP-227-A is appropriate.
    - For weld locations subjected to fluence levels above $3x10^{21}$ n/cm^2 (E>1MeV), a CGR model appropriate for the fluence level and material must be used and a justification for its use provided. Such models are available from sources like proprietary EPRI technical reports.

Program Management Office
20 International Drive
Windsor, Connecticut 06095

May 1, 2015

WCAP-17096-NP, Rev. 2
Project Number 694

OG-15-178

Mr. Kyle Amberge, EPRI Project Manager
Electric Power Research Institute (EPRI)
3420 Hillview Avenue
Palo Alto, CA 94304

Subject: Pressurized Water Reactor Owners Group
Additional RAI Responses (PWROG-15035-NP, Revision 0) to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473R5

Dear Mr. Amberge:

In January 2010, the Pressurized Water Reactor Owners Group (PWROG), provided the Electric Power Research Institute (EPRI) WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Reference 1). EPRI submitted the document to the Staff for review and comment under the MRP-227 umbrella (Reference 2). In June 2010, the Staff accepted the topical report (Reference 3) and provided a Request for Additional Information (RAI) (Reference 4) on May 19, 2011. On July 15, 2011, the Staff provided a revision for Request for Additional Information (RAI) (Reference 5) and the PWROG provided responses. Based on the draft RAI input provided under Reference 6, teleconferences were held with the Staff to discuss both the AREVA and Westinghouse responses. The minutes from both calls are provided in Reference 7. Based on those minutes, the RAI responses were revised and provided to EPRI for submittal to the NRC (Reference 8). On October 11, 2012, the NRC provided a second Request for Additional Information (RAI) (Reference 9). The Planning Team has reviewed and provided comments on the second Request for Additional Information (RAI) (Reference 10) and submitted them to EPRI under Reference 11. On December 10, 2012, the NRC provided additional comments via email (Reference 12) on the responses to the RAIs that where submitted under Reference 11. The PWROG provided additional responses on December 20, 2012 (Reference 13). On August 18, 2014 the NRC provided two additional RAIs to the PWROG through the EPRI MRP (Reference 14). The PWROG provided additional responses on October 27, 2014 (Reference 15). On February 11, 2015, the NRC provided additional RAIs to the PWROG through the EPRI MRP (Reference 16). Enclosure 1 provides the PWROG response.

The PWROG would like to request that we are kept on distribution, via letter, once the additional response is submitted to the Staff. Updates to the WCAP will be made in parallel with the NRC review of the RAI response. The updated WCAP will be provided to the Staff at a later date.

References:

1. PWROG Letter from Dennis Buschbaum to Anne Demma, EPRI Transmittal of Final WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements", December 2009, OG-10-22, dated 1/13/10.
2. Report Transmittal: Westinghouse Non-Proprietary Class 3 Report, "Reactor Internals Acceptance Criteria Methodology and Data Requirements, WCAP-17096-NP, Revision 2, December 2009, MRP 10-034, dated 5/19/10.
3. Acceptance for Review of PWR Owners Group (PWROG) Topical Report WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200), letter from NRC to EPRI, dated 6/28/10.
4. Request of Additional Information on WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements (TAC NO. ME4200) PA-MSC-0473, dated 5/19/11 and posted to the PWROG website under OG-11-163, dated 5/23/11.
5. Revision 1 to the Request of Additional Information on WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473, dated 7/10/11 and posted under the PWROG website under OG-11-223, dated 7/ 15/11.
6. Responses to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473, OG-11-264, dated 9/6/11.
7. High Level Minutes (LTR-RIAM-11-50) from October 4th and 12th Teleconferences with the NRC to Discuss the Draft RAI Responses Related to PWROG WCAP-17096, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (PA-MSC-0473), OG-11-343, dated 11/2/11.
8. Revised Responses (LTR-RIAM-12-12) to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473, OG-12-202, dated May 25, 2012.
9. Request of Additional Information Related to WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" December 2009 (TAC NO. ME4200) PA-MSC-0473, dated 10/11/12 and posted to the PWROG website under OG-12-444, dated October 26, 2012.
10. Review and Comment of Additional RAI Responses (LTR-RIAM-12-134) to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473, OG-12-475, dated November 9, 2012.

References (Continued)

11. Additional RAI Responses (LTR-RIAM-12-138, Rev 0) to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473, OG-12-495, dated November 21, 2012.
12. Comments to Additional RAI Responses via email from the NRC. Joe Golla (NRC) to Kyle Amberge (EPRI), dated December 10, 2012.
13. Additional RAI Responses (LTR-RIAM-12-138, Rev 1) to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473, OG-12-519, dated December 20, 2012.
14. Letter from J. Holonich (U.S. NRC Senior Project Manager) to B.C. Rudell (EPRI Materials Reliability Program Chairman), "Request for Additional Information Related to WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements' (TAC No. ME4200)," dated August 18, 2014, ML14177A071 and posted to the PWROG website under OG-14-331, dated September, 18, 2014.
15. Additional RAI Response (LTR-RIAM-14-91, Rev 2) to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473R5, OG-14-349, dated October 27, 2014.
16. Request of Additional Information Related to WCAP-17096-NP, Revision 2 "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (TAC NO. ME4200) PA-MSC-0473R5, dated February 11, 2015 and posted to the PWROG website under OG-15-112, dated March 16, 2015.

If you have any questions feel free to contact Mr. Jim Molkenthin of the PWR Owners Group Project Management Office at (860) 731-6727.

Regards,

Norman J Stringfellow

Jack Stringfellow
Chief Operating Officer & Chairman
Pressurized Water Reactor Owners Group

NJS:JPM:las

Enclosures (1): PWROG-15035-NP, Revision 0

cc: PWROG Steering Committee
PWROG Licensing Subcommittee
PWROG Program Management Office
J. Rowley, USNRC
S. Stuchell, USNRC
J. Kielb, Westinghouse
I. Ezzi, Westinghouse
J. Andrachek, Westinghouse
P. Paesano, Westinghouse
C. Boggess, Westinghouse
E. Deemer, Westinghouse
M. Spak, Westinghouse
D. Denis, Westinghouse
PWROG Management Committee
PWROG Materials Subcommittee
R. Lott, Westinghouse
J. McKinley, Westinghouse
E. Benacquista, Westinghouse
T. Demers, Westinghouse
M. Ickes, Westinghouse
B. Carpenter, Westinghouse
E. Deemer, Westinghouse
B. Grambau, AREVA Inc
T. Natour, AREVA Inc
S. Fyfitch, AREVA Inc
A. Demma, EPRI