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Category 2 Public Meeting To Discuss the Nuclear Energy Institute (NEI) and Electric Power Research Institute (EPRI) Topical Report 1006596, "Materials Reliability Program (MRP): PWR Internals Inspection Guidelines (MRP-227, Rev. 0)

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Meeting Agenda

- Opening Remarks/Introductions
- NRC Discussion Topics (morning session)
  - Opportunity for Public Comment
- NEI/EPRI Topics (afternoon session)
  - Opportunity for Public Comment
- Closing Remarks

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Additional References

- EPRI to resubmit the following MRP reports, as referenced in the June 10, 2009 affidavit:
  - MRP-189, Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B& W-Designed PWR Internals."
  - MRP-210, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components."
  - MRP-231"Materials Reliability Program: Aging Management Strategies for B&W PWR Internals."
  - MRP-232, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals."

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MRP-227 Key Milestones

- 09/17/09: Issue Request for Additional Information (RAI).
- 10/17/09: RAI response provided to the NRC.
- 02/12/10: Issue Draft Safety Evaluation (SE).
- 03/12/10: Draft SE Comments (if any) from NEI/EPRI.
- 05/12/10: Issue Final SE.

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NRC Discussion Topics

- Issue 1: Generic Aging Lessons Learned (GALL)
- Issue 2: Failure Mode, Effects and Criticality Analysis (FMECA) and Functionality Analysis (FA)
- Issue 3: Time-Limited Aging Analysis (TLAA)
- Issue 4: Inspection

(Refer to handout for details)

Discussion Topics For The August 4, 2009, NRC Public Meeting

Issue 1 Generic Aging Lessens Learned (GALL)

(a) NRC Review of MRP-227-Rev. 0 under a fee waiver was based on the fact that the MRP would develop the new AMRs for the W, CE, and B&W RVI components. Consistent with aging management review (AMR) line items addressed in GALL Tables IV B2 (Westinghouse), IV.B3 (Combustion Engineering) and IV.B4 (Babcock & Wilcox), the MRP will be requested to provide AMR line items as part of aging management program (AMP) for the reactor vessel internal (RVI) components.

(b) Some of the aspects of the MRP-227-Rev. 0 methodology may need to be addressed by license renewal applicant action items for applications currently under review or those that have yet to be submitted to the NRC. The staff requests the MRP's assistance in identifying potential action items which are: (1) necessary to provide plant-specific information to complete the AMP; (2) necessary to confirm applicant compliance with important assumptions underlying the MRP-227 methodology; or (3) other considerations.

Issue 2 Failure Mode, Effects and Criticality Analysis (FMECA) and Functionality Analysis (FA)

(a) Provide information regarding the critical assumptions that were made in developing the FMECA and FA.

(b) The FMECA uses a probabilistic approach with regard to structural stability of any given RVI component. What methodology was used to establish the failure probability factor of any RVI component.

(c) Please clarify the conditions under which design basis event (DBE) effects on component performance were considered. How does this approach provide reasonable assurance that the margins against failure are adequately maintained during the license renewal period.

(d) Component failure due to the same degradation mechanism is not considered to be a common cause failure because of the expectation that damage initiation and growth occurs at different times. However, if the extent of damage does not affect (or be indicated during) the normal operations, any transient or DBA can potentially lead to a plant condition that would not occur unless multiple components were degraded.

(e) It's not clear how plant-specific differences were considered within the FMECA. Any needed modifications to the FMECA or additional evaluations to address plant-specific issues are not discussed.

(f) Irradiation induced stress relaxation: Please identify which of the bolted joints, springs and other components require preloads to maintain their functions. In a given RVI component system, identify: (1) minimum number (percentage) of bolts/springs in each component that are required to maintain preload without

compromising the structural integrity of the component and (2) the extent of inspections of these bolts/springs.

Issue 3 Time-Limited Aging Analysis (TLAA)

(a) Confirm if there are any TLAA issues associated with RVI components as part of AMP. TLAA may be required for PWR units that will be applying for more than one period of extended operation.

(b) Cumulative usage factor (CUF) values for several B&W components need to be confirmed by MRP.

Issue 4 Inspection

(a) The MRP intends to implement visual testing (VT-3) examinations to identify cracking in some RVI components. Historically, enhanced visual testing (EVT-1) or ultrasonic testing (UT) method is used effectively to identify cracks. Provide an explanation how VT-3 examination can detect IASCC/SCC cracks

(b) Highly susceptible Category D (FMECA) non-safety components may require inspection which could be a leading indicator in identifying aging degradation. Clarify how many of the components identified as Category D ultimately were binned into the primary inspection group.

(c) The accessibility of the primary inspection components is not typically addressed. It's therefore not clear how much area of inspection coverage is necessary to ensure timely detection of aging effects in the RVI components. Discuss whether guidance should be provided in MRP-227 regarding minimum inspection volumes/areas must be achieved to take credit for having effectively inspected a particular RVI component.

(d) Many components are placed on a standard 10-year ISI interval coincident with typical ASME Code inspection requirements. It's not clear, however, if this 10-year interval is technically acceptable. No justification in light of the specific degradation mechanisms being managed has been provided. Other inspection intervals and requirements are based on a certain number of operating cycles. The acceptability of these intervals have also not been established.

(e) Certain degradation mechanisms (i.e., void swelling in B&W components) are not inspected for a particular reactor type. Why isn't the most susceptible location for each mechanism in each reactor-type (i.e., B&W, CE, or West.) inspected as a primary component to insure that each degradation mechanism is not occurring within the reactor?

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NEI/EPRI Topics

- Refer to NEI/EPRI handouts