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Sent: Friday, September 17, 2010 1:59 PM
To: KUEMIN, JAMES L; ERICKSON, JEFFREY S; Anderson, Paula
Cc: Mitchell, Matthew; Widrevitz, Dan; Cheruvenki, Ganesh; Pascarelli, Robert; Orlikowski, Robert; Ellegood, John; Taylor, Thomas
Subject: Palisades - Program Plan for Aging Management of Reactor Vessel Internals - ME4084

In a letter dated March 10, 2010, Entergy Nuclear Operations, Inc. (ENO) submitted an aging management program (AMP) for the Palisades Nuclear Plant (Palisades) reactor vessel internals (RVI). The NRC staff is in the process of reviewing the Palisades AMP report and based on the review conducted thus far, the staff has developed a first request for additional information (RAI) as addressed below. The MRP-227 report, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," and its supporting reports were used as technical bases for developing the subject AMP. The staff is currently reviewing the MRP-227 report and its associated technical basis and based on its review the staff may, however, issue another RAI at a later date. Please arrange a teleconference with the staff to discuss the information requested.

REQUEST FOR ADDITIONAL INFORMATION ON AGING MANAGEMENT PROGRAM FOR REACTOR VESSEL INTERNALS AT PALISADES NUCLEAR PLANT (TAC NO. ME4084)

RAI-1 Table IV of NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Volume 2, Revision 1, identifies aging effects for some of the RVI components that were designed by Combustion Engineering. Aging effects that are pertinent to some of the RVI components which are considered part of RVI AMP are addressed in GALL Table IV B3. The following table provides information related to the aging effects which were not included in the AMP for the RVI components listed in GALL Table IV B3.

Aging Effect	RVI Component	GALL Report-Table ID number
Loss of preload/stress relaxation, cracking/stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), loss of fracture toughness, changes in dimension, void swelling	Core shroud tie rods	IV.B3-7, 10 through 12
Loss of fracture toughness/ neutron irradiation embrittlement, void swelling, loss of material	Core support barrel upper flange	IV.B3-16, 17
Loss of fracture toughness/thermal aging and neutron irradiation embrittlement	Core support column	IV.B3-18

Provide a supplement to your AMP which addresses these aging effects for these components.

RAI-2 Many components are placed on a standard 10-year ISI interval coincident with typical American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) inspection requirements. It's not clear, however, whether this 10-year interval is technically acceptable for PWR RVI components. 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 has also not been established. Please provide a technical justification of the intervals chosen relative to the mechanisms being managed in the proposed AMP.

RAI-3 In Appendix B and C of the AMP, the licensee intends to implement visual testing (VT-3) examinations to identify cracking in some PWR RVI components. Historically, enhanced visual testing (EVT-1) or ultrasonic testing (UT) methods are used to effectively identify cracks. Explain why the use of a VT-3 inspection method should be considered acceptable for identifying cracking in some PWR RVI components.

RAI-4 The accessibility of the primary inspection RVI components is not typically addressed. It is therefore not clear how much inspection coverage is necessary to ensure timely detection of aging effects in the primary inspection RVI components. Discuss whether guidance should be provided in the proposed AMP regarding minimum inspection volumes/areas which must be achieved to take credit for having effectively inspected a particular RVI component.

RAI-5 During the extended period of operation, some PWR RVI components are subject to high levels neutron radiation which may lead to irradiation embrittlement and a loss of fracture toughness and the potential for IASCC. In combination, these effects may lead to the potential for component failure under some design basis loading conditions. Explain how the Palisades AMP will account for potential reduction in fracture toughness when evaluating cracks that are detected during the required inspections, in particular when establishing the frequency of subsequent inspections after cracking is identified.

RAI-6 Loose parts could be generated due to deterioration of some PWR RVI components during the extended period of operation. Provide information which addresses how the following consequences of loose parts generation were considered in development of the inspection program given in the proposed AMP.

(a) potential for fuel bundle flow blockage and consequential fuel damage,

(b) potential for interference with control rod operation, and

(c) potential for impact damage on reactor internals.

RAI-7 Alloy 600 RVI components and their associated welds are susceptible to primary water stress corrosion cracking (PWSCC) when exposed to PWR reactor coolant water. Please state which, if any, RVI components are made of Alloy 600. For each component so identified, the staff requests that the licensee provide the following information.

(a) the type of inspections that were performed thus far on this component,

(b) identification of any PWSCC in this component and the appropriate corrective actions taken by the licensee and,

(c) justification for using 10 year inspection frequency.

RAI-8 With respect to the management of cast austenitic stainless steel (CASS) aging and embrittlement, the licensee does not appear to address the program's compliance with the requirements specified in the relevant GALL Report AMPs. Provide a discussion of how the AMP adequately addresses the requirements specified in GALL AMP, XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," and GALL AMP XI.M13, "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS)," for CASS materials used in PWR RVI components. Alternatively, if the management of CASS PWR RVI component aging is not treated within the scope of the licensee's submittal dated March 10, 2010, the licensee is requested to provide a proposed modification of the report which documents how it would manage this mechanism outside of its AMP of the RVI components.

RAI-9 According to Section A.1.4 in MRP-175, "Materials Reliability Program: PWR Internal Aging Degradation Mechanism Screening Threshold Values," susceptibility to SCC in nickel-based Alloy X-750 PWR RVI components depends on the type of heat treatment that is performed on the alloy. High temperature heat treatment (HTH) processes that are used on Alloy X-750 components offer better resistance to SCC than the other age hardened heat treatment processes. Licensee determination of the heat treatment applied to their Alloy X-750 PWR RVI components would appear to be a critical parameter in ensuring the licensee's AMP will adequately manage the potential effects of aging. Therefore, the staff requests the licensee provide information related to the type of heat treatment process that was used for any Alloy X-750 RVI components and a summary of the inspection history of Alloy X-750 RVI components at Palisades.

RAI-10 PWSCC has been identified as the primary degradation mechanism affecting PWR high nickel alloy nozzles and welds (e.g., Alloy 600 tubing, piping, or forging material, and Alloy 82/182 weld material) in the reactor coolant system. Bottom mounted instruments (BMI) components made of Alloy 600 are susceptible to PWSCC and therefore, the ASME Code, Section XI, developed a Code Case N-722, "Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials ASME Section XI, Division 1," which recommends that bare metal visual inspections be performed every other refueling outage on all BMI penetrations in the PWR RPV lower head. Therefore, the staff recommends that the inspection criteria specified in ASME Code, Section XI Code Case N-722 for the BMI components should be included in the licensee's inspection program.

RAI-11 When exposed to a light-water reactor temperature of approximately 500 ºF or higher, the 17-4 precipitation hardened (PH) martensitic stainless steel (MSS) embrittlement and an increase in hardness (i.e., a reduction in Charpy "V" notch fracture toughness value). Operating experience from Oconee Nuclear Station shows that thermally embrittled 17-4 PH martensitic SS is susceptible to failure when exposed to unexpected loading conditions. Subsequently, on March 7, 2007, the staff issued Information Notice (IN)-2007-02 (ADAMS Accession Number-ML 0701004590), in which the staff recommends that the licensees can prevent the deleterious effects of thermal embrittlement in the 17-4 PH martensitic SS components by identifying aging degradation (i.e., cracks), implementing early corrective actions, and monitoring and trending age-related degradation. Therefore, the staff requests that the subject AMP should include thermal embrittlement as an aging effect for any 17-4 PH martensitic SS RVI components at Palisades.

RAI-12 The staff requests that the licensee explain for each aging effect whether any variability in RVI design or operating conditions (i.e., any planned future uprates) could potentially cause a change in category of a component from A or B to C.

RAI-13 The licensee is requested to clarify whether the program requires the most susceptible location for each mechanism be inspected as a primary component to ensure that each degradation mechanism is not occurring within the reactor.

RAI-14 The staff requests that the licensee discuss how the RVI components in the Palisades reactor design considered to be the most susceptible to (or most likely to first demonstrate the effects of) a particular degradation mechanism did, or did not, get binned in the primary inspection component group for Palisades.

RAI-15 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?

RAI-16 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, certain DBEs could potentially lead to a plant condition (damage state) that would not occur unless multiple components were degraded. Discuss how the potential for multi-component failure due to a DBE was considered as part of the development of the AMP for the RVI components at Palisades.

RAI-17 The licensee is requested to explain how the thermally and irradiation-enhanced stress relaxation criteria for the bolts would ensure that the applicable RVI components will be able to maintain their design function during design emergency and faulted conditions at the end of the period of extended operation. If the thermally and irradiation-enhanced stress relaxation criteria have not been evaluated for design emergency and faulted conditions, the licensee should identify plant-specific action items that must be performed to ensure that these RVI components will be able to maintain their design function during design emergency and faulted conditions at the end of the period of extended operation.

RAI-18 The licensee is requested to explain how the inspection method is capable of determining whether the bolting in some RVI components will be able to maintain their design function during design emergency and faulted conditions at the end of the period of extended operation. If the inspection method is not capable of determining whether the components will be able to maintain their design function during design emergency and faulted conditions at the end of the period of extended operation, the licensee should identify plant-specific action items that must be performed by licensees to ensure that these components will be able to maintain their design function during design emergency and faulted conditions at the end of the period of extended operation.

RAI-19 The licensee is requested to provide the following information regarding the upper and lower column support bolts, hold-down spring bolts, and clevis insert bolts as and if any exist in the Palisades RVI.

(a) Number of bolts in each RVI component that have access for an inspection,

(b) Number of bolts that were inspected thus far in each of these RVI components and the inspection results,

(c) Minimum number of bolts in each component that are required to maintain preload without compromising the structural integrity of the component,

(d) Combination of stress relaxation and changes in transient and high cycle vibration may increase fatigue susceptibility of these RVI components. Therefore, the staff requests that the licensee address fatigue evaluation for these components when the loss of preload can compromise their structural integrity.

RAI-20 During the extended period of operation, some RVI components are subject to irradiation embrittlement which results in loss of fracture toughness and IASCC due to exposure to neutron fluence. Lower fracture toughness values and IASCC crack growth rates relevant to irradiated components shall be used for evaluating the flaws in these components. The licensee is requested to address this issue in the AMP. If cracks are detected during the 10 year ISI inspection, the staff expects that the licensee revise the frequency of subsequent inspections based on its analysis using a lower fracture toughness value. Transient loads shall be considered as part of the flaw evaluation methodology for the cracked RVI components.

RAI-21 The staff requests that the licensee discuss whether an evaluation was performed for any specific high consequence of failure RVI components such that their inspection might be warranted even in the absence of a currently identifiable mechanism. Are there any RVI components that should be monitored through in-service inspection to protect against unforeseen failure due to the emergence of a potential future degradation mechanism?

RAI-22 It is not clear how much void swelling is needed before distortion is detectable via VT-3 examination in susceptible PWR RVI components and whether this threshold for detectability will also address the concern over potential loss of fracture toughness due to void swelling. The licensee is requested to provide a discussion of this topic.

RAI-23 The licensee is requested to address the following issues in the proposed AMP.

(1) Confirm if there are any time limited aging analysis (TLAA) issues associated with RVI components as part of AMP,

(2) Cumulative usage factor values for applicable RVI components need to be confirmed by the licensee.

RAI-24 The staff expects that the licensee would comply with any applicable plant-specific license renewal action item that would be recommended in the staff's final safety evaluation for the MRP-227 report. Therefore, the licensee shall make a commitment that it will comply with the staff's plant-specific license renewal action item for PBNP RVI components.

RAI-25 Only one significant area of operating experience was addressed in Section 5.10 of the AMP submittal, namely that of cracking in bolting. Please discuss any other significant areas of operating experience applicable to the Palisades RVI that would be related to aging management.

RAI-26 In Appendix B, for the Core Barrel Assembly, Upper Core Barrel flange weld, the licensee states that EVT-1 exams will be done on 100% of the accessible surfaces of the weld and adjacent base metal. The staff assumes that it is the inside diameter (ID) of the Core Barrel Assembly that is accessible for EVT-1 exams. Provide a description of the residual stresses and service loads expected to act across the section through the weld joint. If the tensile stresses are higher on the outside diameter (OD) of the Core Barrel Assembly at the upper flange weld, then justify why the EVT-1 inspection on the ID is recommended.

RAI-27 In Appendix C a number of items, specifically the Lower Support Structure core support plate, and lower support column welds have as the examination coverage requirements, "Examination coverage determined by plant specific analysis." Please discuss how the coverage was determined and what this coverage was determined to be.