OFFICE OF NUCLEAR REACTOR REGULATION

EVALUATION OF WCAP-17780-P, "REACTOR INTERNALS AGING MANAGEMENT MRP-227-A APPLICABILITY FOR COMBUSTION ENGINEERING [(CE)] AND WESTINGHOUSE [ELECTRIC COMPANY (WESTINGHOUSE)] PRESSURIZED WATER REACTOR DESIGNS," AND MRP-227-A, APPLICABILITY GUIDELINES FOR CE AND WESTINGHOUSE PRESSURIZED WATER REACTOR DESIGNS

1.0 BACKGROUND

Applicant/Licensee Action Item (A/LAI) 1 from the U.S. Nuclear Regulatory Commission (NRC) staff's final safety evaluation (SE) of MRP-227-A, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (Ref. 1), states that:

> … each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227.

As stated in Section 2.4 of MRP-227-A, the assumptions related to core loading (e.g., switch to a low-leakage core by year 30 of plant operation) and operational aspects (e.g., base loaded plant) of design and operation are a conservative representation of U.S PWR operating plants, which implemented low-leakage core loading patterns early in operating life. Section 2.4 of MRP-227-A also states that the recommendations in the report are applicable to all U.S PWR operating plants as of May 2007.

MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs" (Ref. 2), documents the screening for susceptibility to aging effects, the failure modes, effects and criticality analysis (FMECA) results, and the categorization and ranking of the reactor vessel internals (RVI) components. In addition to the assumptions listed in Section 2.4 of MRP-227-A, MRP-191 documents additional assumptions that were used. In particular, the neutron fluence range, operating temperature, and material grade for each generic component of the Westinghouse and CE-design internals were used as input to the screening process. These values were determined based on an "expert elicitation" process. Stress values were not explicitly tabulated, but were recorded as either above the threshold for stress corrosion cracking (>30 ksi), or not, based on the expert interviews.

MRP-232, Revision 0, “Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals” (Ref. 3) (proprietary document, not available to the public), reported more specific stress, operating temperature and neutron fluence values based on detailed analyses for selected components identified in MRP-191 as having a high consequence of failure.

In this context, in requests for additional information (RAIs) to several licensees, the NRC staff initially requested that licensees with PWR units designed by Westinghouse and CE (1) Describe the process used to verify that the inspection and evaluation (I&E) guidelines of the RVI components at a given facility is bounded by the assumptions related to the variables (i.e., neutron fluence, temperature, stress values, and materials) that were used for each component in the FMECA and functionality analyses supporting the development of MRP-227-A, or (2) Provide the plant-specific values of these key variables for a sampling of RVI components. Licensee responses to these requests generally indicated that the neutron fluence, temperature, and stress values requested were not available on a plant-specific basis and would require significant time to prepare.

During a November 28, 2012, public meeting, the NRC staff discussed concerns regarding industry responses to A/LAIs 1 and 2 (Ref. 4). The concerns were addressed to owners of currently operating PWR plants designed by Westinghouse and CE. At the conclusion of this meeting, the NRC staff communicated that it would be helpful to the staff to see how changes in the input assumptions related to the essential variables (e.g., stress, neutron fluence and temperature due to gamma heating) would affect the results of the process used to develop the I&E guidelines provided in the MRP-227-A. For example, would the changes in the essential variables result in either (a) the identification of new RVI components susceptible to aging degradation (or new aging mechanisms for components), or (b) changes in the categorization of components as “Primary” or “Expansion”?

Following this meeting, a series of non-public and public meetings were conducted from January to June of 2013 (References 5, 6, 7, and 8), at which the NRC, Westinghouse, the Electric Power Research Institute (EPRI), and utility representatives discussed a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A.

As a result of the technical discussions with the NRC staff, the basis for a plant to demonstrate that MRP-227-A is applicable to the facility was determined to be satisfied with plant-specific responses to the following two questions:

Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)
Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

To address the staff's concerns, Westinghouse submitted to the NRC proprietary topical report WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs" (Ref. 9), by letter dated June 28, 2013. This report includes the slides used by Westinghouse at the January 22-23, 2013, and May 22, 2013, meetings, in Appendices A and B, respectively. Section 6 of WCAP-17780-P provides a discussion of the effects of cold work on the applicability of MRP-227-A, as a basis for guidance to respond to Question 1. The bulk of WCAP-17780-P (Sections 2-5) discusses sensitivity studies intended to provide the technical basis for the response to Question 2. This report discusses the geometric and operational characteristics that impact neutron exposure and nuclear heat generation in the RVI components. To examine the potential variability of neutron exposure and nuclear heat generation rates that could occur over the life of a reactor or across the entire fleet of Westinghouse and CE designs, a series of studies were conducted to examine the sensitivity of the RVI thermal and radiation environment to differences in RVI geometry, fuel assembly geometry, fuel management strategy, and other reactor operating characteristics. The studies considered the variation in these parameters across the Westinghouse and CE fleets, and also examined the effects of power uprates on the operating environment of the RVI.

WCAP-17780-P states that the initial evaluation process used to develop MRP-227-A considered neutron fluence, temperature, stress, and material data in screening of material susceptibility to one or more of the aging degradation mechanisms and the method of categorization RVI components under inspection categories of "Primary," "Expansion," "Existing," and "No Additional Measures." Westinghouse further stated that it considered variations in operating conditions and design for 2-loop, 3-loop, and 4-loop vessels (designed by Westinghouse), as illustrated on slide 25, page A-13 in Appendix A of the WCAP-17780-P. The staff's evaluation of WCAP-17780-P is detailed in Section 2.0.

Based on the sensitivity studies documented in WCAP-17780-P, EPRI developed guidance for licensees to respond to the two questions, documented in an attachment to MRP 2013-025, "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs, Enclosure to MRP Letter 2013-025, October 14, 2013" (Ref. 10). With respect to Question 1 on cold work, Reference 10 provides guidance for determining if a plant has RVI components with 20 percent cold work or greater. With respect to Question 2, Reference 10 provides specific numeric and geometric criteria to which plant-specific values can be compared to determine whether a the plant has atypical fuel design or fuel management. The staff's evaluation of the MRP guidelines in MRP 2013-025 is detailed in Section 3.0.

2.0 TECHNICAL EVALUATION OF WCAP-17780-P

2.1 Effects of Stress on MRP-227-A Categorization of RVI Components

2.1.1 Westinghouse Evaluation

As discussed in Appendix A of WCAP-17780-P, the stress assessments included thermal stresses, weld residual stresses (in a weld joint), and normal operating stresses, which often include upset conditions. Stresses associated with faulted conditions were not directly included because they are important in determining acceptance criteria but do not contribute to aging degradation. As examples in WCAP-17780-P, stresses in several RVI components were evaluated to demonstrate that varying the stress values of these components would not change the evaluation of these components, such as screening of the components or any change in the inspection categorization of these components. The RVI components included: (a) CE components—core shroud; (b) Westinghouse components—control rod guide cards; lower flange weld. This evaluation considered the effects of stress on the aging effect of cracking due to either irradiation-assisted stress corrosion cracking (IASCC) or stress corrosion cracking (SCC) for these components. The stress values for these components are bounded by the original assumptions that were used for stress values.

2.1.2 Staff Evaluation

The staff reviewed Westinghouse's evaluation on the effects of variations in the stress values and, based on the data presented, concluded that stress values on some of the RVI components that were shown in the presentation are bounded by the original assumptions that were used for stress values in developing the inspection guidelines in the MRP-227-A report. Where this is not true, Westinghouse demonstrated that the increased stress level would not affect the categorization of the affected component. Therefore, the staff accepts the evaluation performed by Westinghouse on the issue related to effect of stress on RVI components.

2.2 Effects of Neutron Fluence and Temperature on MRP-227-A Categorization of RVI Components

Sections 2 through 5 of WCAP-17780-P address the sensitivity studies performed by Westinghouse to assess the effects of variations in fuel management and fuel design on the applicability of MRP-227-A. Section 2 provides a general description of the factors influencing neutron fluence and heat generation in the RVI. Section 3 provides details on the sensitivity study results with respect to neutron fluence and heat generation variations for components moving radially outward from the active core and components located axially above and below the active core. Section 4 provides recommendations for numerical parameters that can be verified on a plant-specific basis to validate that MRP-227-A is applicable to the plant. Section 5 discusses how power uprates are accounted for in the sensitivity studies.

For neutron fluence and temperature, Westinghouse performed sensitivity studies to assess the effect of radial and axial fluence variation on aging degradation of the components The aging mechanisms evaluated are IASCC, irradiation embrittlement (IE), irradiation-induced stress relaxation (ISR), and void swelling. According to Westinghouse, the assumptions of MRP-227-A assumed high leakage core (called "out-in" core loading) operation for 30 years followed by 30 years of low leakage core operation. The power distribution associated with out-in core loading maximizes power densities in the peripheral fuel assemblies, producing high neutron doses for the RVI.

2.2.1 Effects of Axial Variations in Fluence and Heat Generation

2.2.1.1 Westinghouse Evaluation

Axial Fluence Variance Above the Active Core

As described in Section 3.2 of WCAP-17780-P, Westinghouse provided its evaluation related to the effects of axial fluence variance for RVI components located above and below the active core region. In its evaluation, Westinghouse concluded that the guidelines for various RVI components addressed in MRP-227-A are not affected. The sensitivity studies considered the effect of variations in the geometric and operational parameters across the Westinghouse and CE fleets, with the core power density (reactor power/total core volume) the essential variable.

The sensitivity studies included a range of values, spanning several inches, for the distance from the active core to the upper core plate (UCP) for Westinghouse reactors, and from the active core to the fuel alignment plate (FAP) for CE reactors. Based on this study, the screening criterion for IE will be exceeded over a certain height above the active core, at levels higher than those identified in the development of MRP-227-A. However, this area will be contained within the UCP and the FAP. With the new sensitivity studies presented in WCAP-17780-P, the UCP in Westinghouse units and CE's FAP were screened in for IE.

Axial Fluence Variance Below the Active Core

As described in Section 3.3 of WCAP-17780-P, Westinghouse performed a bounding analysis for the lower internals, taking into account significant variations of fluence, and determined that no additional components will be added to the inspection category. Westinghouse further stated that all the components that are susceptible to any aging degradation in the lower internals were already included in various inspection categories as addressed in the MRP-227-A report. Therefore, this evaluation has no impact on the MRP-227-A inspection recommendations.

Plant-Specific Applicability for Axial Components Above the Active Core

As described in Section 4.2 of WCAP-17780-P, the neutron fluence for Westinghouse units at the UCP does not exceed the neutron fluence at the lower core plate (LCP), hence the UCP is not categorized as a leading indicator for detecting IE. Westinghouse stated that IASCC in the UCP is unlikely and therefore, did not require any inspections for the UCP. Westinghouse concluded that, since the UCP is already classified under the "Expansion" inspection category for wear and fatigue, any aging degradation due to IE would be managed under the current MRP-227-A guidelines. Therefore, Westinghouse determined that no inspections are needed for the UCP in Westinghouse units.

From the sensitivity studies, Westinghouse also identified that the FAP in CE units is susceptible to IE. For CE plants with core shrouds assembled with full-height shroud plates (System 80 designs only), the FAP was originally categorized under the "Primary" inspection category for monitoring aging effects due to fatigue, which is evaluated by a time limited aging analysis (TLAA) or by an enhanced visual testing (EVT-1) inspection. This categorization applies only to the units with the CE System 80 design. Thus, the MRP-227-A proposed aging management of the FAP for System 80 plants is adequate to address the susceptibility to IE identified by the sensitivity studies.

The FAPs in the rest of the CE fleet are categorized as "No Additional Measures" components, and therefore require no augmented inspections during the period of extended operation (although some may be subject to ASME Code, Section XI inspection requirements) because no cracking mechanism was identified. Combining the lack of a credible cracking mechanism and the limited extent of potential embrittlement, Westinghouse concluded that no additional inspections are required for the FAP in CE units.

Based on this information, Westinghouse determined that no changes were necessary to the MRP-227-A categorization and inspection recommendations for the UCP and the FAP.

To demonstrate the applicability of MRP-227-A for the upper axial boundary components, WCAP-17780-P Section 4.0 recommends that each plant identify:

(1) the distance between the UCP and the active fuel for Westinghouse units (with an acceptance criterion of greater than 12.2 inches) and the distance between the active fuel and the FAP for CE units (with an acceptance criterion of greater than 12.4 inches), and,

(2) the average core power density for Westinghouse units (with an acceptance criterion of less than 124 Watts/cm^3) and CE units (with an acceptance criterion of less than 110 Watts/cm^3) .

A plant-specific evaluation to assess the need for additional aging management measures is necessary if the unit does not conform to the prescribed limits.

Axial Heat Generation Variance: Components Above and Below the Active Core

As described in Section 3.5 of WCAP-17780-P, the gamma heating rates are expected to be lower in the UCP than in the baffle-former assembly in Westinghouse units. Westinghouse assessed the axial heat generation variance as part of its evaluation on void swelling of the UCP and the LCP. Westinghouse concluded that aging degradation due to void swelling is not likely to occur in the UCP and the LCP for Westinghouse units. Therefore, Westinghouse determined that the UCP in Westinghouse units is not screened in for void swelling. For CE units, the core shroud structure is the leading location with respect to void swelling. The FAP is not expected to experience void swelling due to its exposure to lower fluence and gamma heating than the CE shroud structure.

The new sensitivity studies indicated that the assumptions that were used in developing the I&E guidelines for the RVI components that are addressed in MRP-227-A were conservative and, as such, these guidelines will remain the same for the other RVI components. This is due to the fact that the RVI components that are already categorized under “Primary” inspection category in MRP-227-A are more susceptible to IE than the newly identified, from the new sensitivity studies, UCP in Westinghouse units and FAP in CE units.

Based on this evaluation, Westinghouse concluded that variance in gamma heating will have no impact on the MRP-227-A recommendations for the RVI components located above or below the active core, except for the identification susceptibility to IE for the UCP in Westinghouse units and the FAP in CE units.

2.2.1.2 Staff Evaluation

For the lower internals below the active core, Westinghouse performed a bounding analysis taking into account significant variations of fluence and determined that no additional components will be added to the inspection category. The staff agrees with the Westinghouse evaluation because, considering the bounding evaluation, no new components would screen in for any aging mechanism; therefore, the MRP-227-A inspection recommendation would remain valid for the RVI components below the active core.

For Westinghouse units, the neutron fluence at the UCP does not exceed the neutron fluence at the LCP; hence the UCP is not categorized as a leading indicator for detecting IE. Westinghouse stated that IASCC in the UCP is unlikely and, therefore, did not require any inspections for the UCP. Westinghouse concluded that since the UCP is already classified under the "Existing Programs" inspection category for wear and fatigue, any aging degradation due to IE would be managed under the current MRP-227-A guidelines, and no additional inspections are needed for the UCP in Westinghouse units. The staff agrees with Westinghouse's conclusion based on the following assessment. Fatigue is the only applicable cracking mechanism in the UCP in MRP-227-A, and fatigue of the UCP is adequately managed as an "Expansion" component through the "Primary" inspection of the CRGT lower flange weld via EVT-1 visual examination. As long as cracking is managed, there is no concern with IE, which cannot cause failure without the existence of a crack.

From the sensitivity studies, Westinghouse identified that the FAP in CE units is susceptible to IE. The FAP of System 80-design CE RVI is categorized under the "Primary" inspection category for monitoring aging effects due to fatigue, which is evaluated by a TLAA or by inspection. The FAPs in the rest of the CE fleet are categorized as "No Additional Measures," thus augmented inspections are not required for the period of extended operation

The staff determined that compliance with the limits of the following attributes addressed in the plant-specific applicability guidance in Section 4.0 of the WCAP is essential to ensure adequate aging management for axial components located above the active core in Westinghouse and CE units. The attributes addressed in the checklist and their acceptance criteria are:

(1) the distance between the UCP and the active fuel for Westinghouse units must be greater than 12.2 inches and the distance between the active fuel and the FAP for CE units must be greater than 12.4 inches, and,

(2) the average core power density for Westinghouse units must be less than 124 Watts/cm^3 and for CE units must be less than 110 Watts/cm^3.

These criteria may be exceeded for a period of up to two years of operation.

Compliance with the aforementioned attributes by each CE and Westinghouse unit will adequately demonstrate the applicability of MRP-227-A for the UCP in Westinghouse units and the FAP in System 80 CE units. Failure to meet these limits requires a plant-specific evaluation that would justify applicability of MRP-227-A for the unit, and the staff's approval of the plant-specific evaluation is required. This plant-specific evaluation would be necessary to determine if components located above the UCP or the FAP would screen in for aging mechanisms such as IE. Based on this evaluation, the staff concludes that the proposed aging management for the newly identified IE of RVI components, i.e., the UCP in Westinghouse units and the FAP in System 80 CE units, are adequately addressed in MRP-227-A. The staff agrees with Westinghouse's conclusion that the existing guidance for the UCP is adequate because fatigue is the only applicable cracking mechanism in the UCP, and fatigue of the UCP is adequately managed in MRP-227-A through the "Primary" inspection of the CRGT lower flange weld via EVT-1 visual examination. As long as cracking is managed, there is no concern with IE, which cannot cause failure without the existence of a crack.

EPRI determined all mechanisms for cracking do not require aging management in the FAP (other than System 80).

However, in MRP-191, fatigue and SCC of welds screened in for the FAP. MRP-191 (Ref. 2) does not distinguish between the FAP in System 80 and non-System 80 designs. The basis for the final aging management recommendation for the FAP in MRP-227-A, in which the "Primary" category inspection of the FAP is only applicable to System 80 designs, is not documented in MRP reports. If cracking can be determined to not require aging management for the FAP, the possibility that some portion of the FAP thickness has IE is not significant, since IE does not cause a problem except when cracks are present.

To verify that the applicability of the MRP-227-A categorization of non-System 80 FAP as "no additional measures," the staff in an e-mail dated August 14, 2014, issued an RAI. The staff requested that Westinghouse provide an explanation how cracking due to SCC and fatigue was determined to be non-significant for CE's FAP in non-System 80 design. In response to this RAI, by a letter dated September 9, 2014, Westinghouse stated that FAPs in non-System 80 CE design, are not susceptible to fatigue and SCC because there are no structural welds. The staff reviewed this response and determined that in the absence of a structural weld in CE FAPs in non-System 80 design, aging degradation due to IE is less likely to occur. The staff believes that IE in CE units is adequately monitored by the inspections of core support plate and core shroud assembly which are binned under the "Primary" inspection category. Therefore, the staff accepts the Westinghouse response and concludes that CE FAPs in non- System 80 design can be binned under the "No Additional measures" category. The gamma heating rates in the UCP and the LCP in Westinghouse units are lower than in the baffle-former assembly. Similarly, in CE units, the gamma heating rate is lower in the FAP than the core shroud structures. Both the baffle-former assembly in Westinghouse units and the core shroud structure in CE units were determined to be the lead component with respect to void swelling, and thus were categorized under the "Primary" inspection category for detecting void swelling in MRP-227-A using the original, conservative assumptions. Based on its evaluation of the information provided, the staff concluded that variance in gamma heating will have no impact on the MRP-227-A recommendations for the RVI components located above or below the active core. Specifically, this conclusion is valid for the UCP and LCP in Westinghouse units and for the FAP in CE units.

2.2.2 Effects of Radial Variations in Fluence and Heat Generation

2.2.2.1 Westinghouse Evaluation

Radial Fluence and Heat Generation Variance in Components Surrounding the Core

As described in Section 3.1 of WCAP-17780-P, the fluence effects on the aging degradation due to IASCC, IE, and ISR on the radial RVI components (Primary/Expansion categories) were studied and the screening results did not alter the I&E guidelines of the RVI components. This is due to the fact that that the conservative assumptions with out-in core loading, though not bounding, maximizes power densities in the peripheral fuel assemblies. However, the relative fluences on the RVI along a radius from the core do not result in any changes in identification of the leading components nor identification of components that exceed the fluence threshold level, and, therefore, the identification of "Primary" and "Expansion" components is not changed with the studied variations in the neutron fluence levels.

Westinghouse further reiterated that the current I&E guidelines in MRP-227-A are valid only if lowleakage core loading is maintained beginning no later than 30 years after the plant begins operations. A plant-specific aging management program would be required for a plant returning to out-in core loading in the future.

As described in Section 3.4 of WCAP-17780-P with respect to radial heat generation rate in components surrounding the core, Westinghouse stated that, when a plant switched to low-leakage operation before the completion of 30 years, the plant-specific nuclear heat generation rate at a hot spot (baffle plate interior corner) seems to be higher than the heat generation values used in the MRP-227-A evaluation. As explained in its plot of heat generation rate versus operating time in terms of effective full power years (EFPY), Westinghouse used heat generation rate with the first 30 year of out-in operation followed by low-leakage operation for 30 years. The average heat generation value used by MRP (as shown in the plot) is higher than the plant-specific average heat generation value. This is because, although the heat generation rate for the plant-specific example was slightly higher during low-leakage operation than assumed by Westinghouse, the plant switched to low-leakage much earlier in life than 30 years. Westinghouse claims that similar plots, between displacements per atom (dpa) rate value and operating time in EFPY, showed similar results. Therefore, a higher average value of the heat generation rate, which is conservative, was used for developing the I&E guidelines for MRP-227-A. Westinghouse stated that the original assumptions (used in MRP-227-A) with out-in core loading for the initial 30 year period of operation result in projected fluence values which are very conservative, even for plants with power uprates which began to operate with a low leakage core loading earlier in plant life. Hence, Westinghouse's report states that no changes in the I&E guidelines are required.

Plant-Specific Applicability for Radial Components

As described in Section 4.1 of WCAP-17780-P, the out-in core loading for 30 years of initial operation produces increased rates of heat generation in the radial components, resulting in a higher likelihood of aging degradation in the components. To reduce the heat generation in the radial components, low-leakage core loading is required during the period of extended operation (PEO). Westinghouse stated that it performed a functionality analysis to evaluate the aging degradation in the RVI components and that the heat generation rate at any re-entrant corner location can be determined by plant-specific core power density (reactor power/total core volume), generic corner weighting factors and relative fuel assembly power. These three variables are used in calculating a "figure of merit" or "F" value, which will be used by each plant to assess the applicability of MRP-227-A. The F value is higher for an out-in core configuration than for a low-leakage core configuration. Westinghouse further stated that in its original assumptions, the estimates of end-of-life fluence in RVI components, such as the core barrel and thermal shield, are conservative based on comparison with surveillance capsule dosimetry. The MRP-227-A I&E guidelines were developed using conservative estimates of heat generation rate and fluence values.

Westinghouse developed a maximum limit for the F value of 68 Watts/cm^3, and a value higher than this limit was used to evaluate the aging degradation of the RVI components for the extended period of operation. Any plant that maintains core loading patterns that are below this maximum limit will satisfy the MRP-227-A fluence screening criteria. Plants that exceed this limit for more than two years would be required to demonstrate their compliance with applicability of MRP-227-A.

2.2.2.2 Staff Evaluation

Radial Fluence and Heat Generation Variance Surrounding the Core

With respect to the evaluation of radial fluence and temperature variance surrounding the core, the staff determined that Westinghouse adequately demonstrated that the radial variations would not change the relative susceptibility of the components from the original evaluation. The staff's conclusions are based on the fact that an increase in the fluence would proportionately increase the fluence to all radial components, but would not change the relative susceptibility of the components. The staff also agrees that the original assumption of 30 years of operation with an out-in core design is generally conservative, and should have resulted in conservative end-of-license fluence and temperature values that were used for the original screening and functionality analyses. Further, the staff finds the proposed heat generation figure of merit "F" provides a quantitative measure that will ensure operation going forward with a low-leakage core, thus ensuring the heating rates at the re-entrant corner locations during the second 30 years of operation levels do not exceed the assumptions of the original MRP-227-A functionality analyses. Otherwise, a plant-specific evaluation of the applicability of MRP-227-A would be necessary. This was confirmed by the new sensitivity studies for some RVI components categorized under "Primary" inspection category. The sensitivity studies did not result in categorization of any new RVI components in "Primary" or "Expansion" inspection categories and, therefore, the assumptions used in developing the I&E guidelines in MRP-227-A remain valid.

For plant-specific applicability of the heat generation rate for radial components, the aging degradation (i.e., void swelling) is a function of average core power density. Westinghouse's functionality analysis that was used for the development of the I&E guidelines was based on the conservative assumption of implementing the out-in core loading for the first 30 years of plant operation and then switching to a low leakage core loading. This assumption resulted in establishing a limiting value of an average core power density at which highest degradation occurs. Since most of the plants switched to low leakage loading at early stages of plant operation (addressed in Section 3.1, page 3-8, of the WCAP-17780-P), the original assumptions are conservative and, as such, they remain valid unless a specific plant operates at a higher average core power density than the limiting value recommended in Section 4 of WCAP-17780-P. The original assumptions used in screening RVI components for radiation damage due to gamma heating do not need to be altered. The sensitivity studies did not identify the need to re-categorize any RVI components (such as elevation from "Expansion" to "Primary" inspection category or "No Additional Measures" to "Expansion") and, as such, the I&E guidelines in MRP-227-A are acceptable unchanged.

Plant-Specific Applicability for Radial Components

Regarding the plant-specific applicability for radial RVI components, out-in core loading for the first 30 years of operation produces increased rates of heat generation in the radial components, resulting in a higher degree of aging degradation in the components. The RVI components that were selected for inspections in MRP-227-A already take into account the conservative values of neutron fluence and heat generation rates. The new sensitivity studies validated that the original assumptions used in developing the I&E guidelines in MRP-227-A are conservative, since low-leakage core loading, was adopted by many in the fleet after 12 years of operation.

Based on this review, the staff concludes that the limit for the F value of 68 Watts/cm^3 is adequate for a plant to demonstrate applicability of the MRP-227-A I&E guidelines, and any plant that maintains core loading patterns that are below this maximum limit will satisfy the MRP-227-A fluence screening criteria. Plants that exceed this limit for more than two years would be required to demonstrate their compliance with applicability of MRP-227-A. Therefore, the staff believes that Westinghouse adequately addressed the radial effect of temperature and neutron fluence on the aging degradation of the RVI components and as such, the I&E guidelines in MRP-227-A are acceptable.

2.3 Extended Power Uprate

2.3.1 Westinghouse Evaluation

Section 5 of WCAP-17780-P provides a brief discussion of the effects of power uprates on fluence and heat generation rates. The increase in fluence is approximately proportional to the percentage increase in power prorated over the remaining life of the reactor, and the increased heat generation rate is approximately proportional to the percentage increase in power. Since both fluence and heat generation rates are a function of the core power density, the MRP-227-A inspection recommendations remain applicable as long as the plant continues to meet the guidelines for average core power density and power distribution (Section 4 of WCAP-17780-P), regardless of whether it has undergone a power uprate.

2.3.2 Staff Evaluation

The staff reviewed the information on the effects of power uprates provided in Section 5 of WCAP-17780-P, and finds that Westinghouse has adequately addressed the effects of power uprate conditions, since the quantitative applicability criteria for core power density in Section 4 of WCAP-17780-P are applicable to all reactors, regardless of whether the reactor has undergone a power uprate. Plants that exceed the guidelines for average core power density and power distribution would need to provide additional evaluations to demonstrate the applicability of the MRP-227-A I&E guidelines to the plant.

2.4 Cold Work

2.4.1 Westinghouse Evaluation

Section 6 of WCAP-17780-P provided a summary of the screening criteria process used to assess stress corrosion cracking (SCC) susceptibility documented in MRP-191. Any stainless steel component with a stress level greater than or equal to 30 ksi and cold work greater than or equal to 20 percent, or a stainless steel component with a weld, was considered susceptible to SCC. Additionally, any components thought to be heavily deformed during manufacturing were screened in regardless of stress level. Westinghouse believes that the data gathered by the expert panel to support the screening would have identified any components with 20 percent or greater cold work during the manufacturing process. Therefore, the only components of concern for individual plants, when preparing responses to Question 1 on cold work, are components that may have received 20 percent cold work as a result of plant-specific modifications. Also, Westinghouse stated that the licensees should have knowledge of the components with 20 percent or greater cold work by a combination of the aging management reviews conducted as a part of license renewal and the work necessary to respond to A/LAIs 1 and 2 of MRP-227-A.

2.4.2 Staff Evaluation

The staff reviewed the information provided in Section 6 of WCAP-17780-P. The staff finds the conclusions of Section 6 reasonable because the process used for the screening and FMECA was sufficiently robust that it should have identified RVI components that would have been generically subject to high levels of cold work during manufacture. Further, development of the aging management review for license renewal and a response to A/LAI 2 of the staff's SE of MRP-227-A requires identification of any plant-specific components that are not captured in MRP-191, including those made from different materials than the material assumed in the screening of MRP-191; this would include materials that have been cold worked. Therefore, an applicant or licensee must review its plant-specific design data (including material) for each RVI component in order to prepare its response to A/LAI 2, which should include a review of documents such as drawings and manufacturing specifications.

2.5 Staff Overall Conclusions for WCAP-17780-P Evaluation

The staff reviewed the subject report and concluded that, based on the sensitivity studies documented in the report, Westinghouse adequately demonstrated that the original assumptions used in the development of the I&E guidelines are conservative for all the RVI components, with the exception of the UCP in Westinghouse units and the FAP in CE units. The sensitivity studies, and the discussion on power uprates in Section 5, provide an adequate technical basis for the recommended plant-specific applicability criteria in Section 4 of WCAP-17780-P, which provide a basis for developing guidance to licensees for responding to generic RAI Question 2 related to fuel management and fuel design. The recommended plant-specific applicability criteria should provide reasonable assurance that the MRP-227-A I&E guidelines are conservative for the specific plants. The basis for the staff's conclusion is that the recommended criteria provide a quantitative measure of whether a plant is operating with a low-leakage core design as assumed by MRP-227-A. In addition, the discussion on cold work in Section 6 of the report provides an adequate technical basis for the industry guidance on assessing whether specific plants have components with 20 percent or greater cold work that were not identified by the generic screening and FMECA process used to develop MRP-227-A.

With respect to the UCP and System 80 and non-System 80 FAP, although the sensitivity studies showed that a new aging mechanism, IE, can be applicable, the staff determined the existing aging management requirements in MRP-227-A for these components do not need to change because IE is not significant in the absence of a relevant cracking mechanism and the MRP-227-A inspection recommendations adequately manage cracking for these components. For the FAP in non-System 80 CE-design RVI, fatigue and SCC were screened out due to absence of any structural weld. Therefore, the staff concludes that the applicability of the MRP-227-A categorization of CE's non-System 80 FAP as "no additional measures," is acceptable.

3.0 TECHNICAL EVALUATION OF MRP-227-A APPLICABILITY GUIDELINES FOR COMBUSTION ENGINEERING AND WESTINGHOUSE PRESSURIZED WATER REACTOR DESIGNS

3.1 Background

The guidance document-- MRP 2013-025, "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," Attachment 1 to MRP 2013-025, October 14, 2013 (ADAMS Accession No. ML13322A454). (Ref. 10), states that, as a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC's RAI to demonstrate compliance with MRP-227-A for originally licensed and uprated conditions was determined to be satisfied with plant-specific responses to the following two questions (References 6 and 8):

Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)

Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

The guidance document states that plant-specific evaluation to demonstrate the applicability of MRP-227-A for managing aging would need to consider the following items:

1. Designated design specific criteria in responding to specific NRC requests for additional information,
2. Criteria defined in MRP-227-A, Section 2.4, and
3. Plant-specific regulatory commitments for managing aging in reactor internals.

3.2 Question 1 - Cold Work

3.2.1 MRP Evaluation

The guidance document stated that it is expected that the aging management reviews (AMRs) conducted as part of license renewal and/or the component reviews conducted in responding to MRP-227-A SE A/LAIs 1 and 2 would be sufficient to determine if there were components with ≥ 20 percent cold work outside of those already identified in MRP-191 and managed under the requirements of MRP-227-A. The guidance document identified the following specific actions to address the possibility of components with cold work exceeding 20 percent:

1. Confirm that plant-specific components identified for aging management were included in the MRP-191 component reviews.
2. Confirm that the design and operating history of those components are consistent with MRP-191.
3. Confirm that modifications or plant-specific activities performed on the component did not introduce cold-worked conditions. Actions to consider for focused assessment of

potential impact may include, but is not limited to: annealing, cold bending, or surface grinding.

In addition, the guidance document categorized various RVI components into the following five categories with respect to the potential for cold work greater than 20 percent: (1) cast austenitic stainless steel (CASS); (2) hot formed austenitic stainless steel; (3) annealed stainless steels; (4) fasteners [fabricated from] austenitic stainless steels; (5) cold-formed austenitic stainless steel without solution annealing. Only RVI components in Categories 4 and 5 may exceed 20 percent cold work and therefore should be screened in as susceptible to stress corrosion cracking (SCC) and irradiation-assisted stress corrosion cracking (IASCC) if the component stress levels are 30 ksi or greater. The guidance document also recommends that licensees assess the impact of any auxiliary manufacturing or installation processes that have the potential to cause high levels of cold work, such as grinding or cold bending. The guidance document further states that there is no requirement for the licensee to search for the records of field fit-up or operational effects and that the licensee need not change the MRP-227-A inspection strategy to account for these effects. However, components known to have higher levels of cold work must be addressed per the guidance.

3.2.2 Staff Evaluation

Based on the information provided, the staff concluded that the guidance document would enable licensees to determine if cold-worked components exist at their plant for which modifications to the MRP-227-A aging management recommendations are necessary to address SCC and IASCC. The staff's basis for this conclusion is as follows: (1) Plant-specific components not identified as having >20 percent cold work in the generic assessment should be identified as having > 20 percent cold work via the licensee's process of performing AMRs for license renewal and reviews to respond to MRP-227-A, A/LAIs 1 and 2, (2) Material types with the potential for cold work > 20 percent are identified, which will assist in screening plant-specific components for high levels of cold work and thus susceptibility to SCC and IASCC; and (3) The guidance requires licensees to evaluate the potential for modifications and auxiliary manufacturing processes to induce higher levels of cold work thus increasing the susceptibility of components to SCC or IASCC.

3.3 Question 2 - Fuel Management

3.3.1 MRP Evaluation

The effect of variations in fuel management on the original assumptions used for stress, fluence and temperature in developing MRP-227-A, was evaluated for three different groups of components: (1) radial boundary components laterally surrounding the core; (2) upper axial boundary components (i.e., above the core); and, (3) lower axial boundary components (i.e., below the core).

As noted in Section 2.4 of MRP-227-A, one of the original assumptions used in the functionality analyses that support the MRP-227-A I&E guidelines was operation with a high leakage (out-in) core for 30 years of operation followed by operation with a low leakage core loading pattern for 30 years. The power distribution associated with out-in core loading maximizes power densities in the peripheral fuel assemblies and results in high neutron fluence; a low-leakage core loading places lower power assemblies in the peripheral locations, which reduces irradiation to the reactor pressure vessel and improves fuel utilization.

Radial boundary components are components located adjacent to the active core, such as the baffle-former assembly in Westinghouse-design RVI, the core shroud assembly in CE-design RVI, and the core barrel and the thermal shield for both designs. The guidance document indicated that the neutron flux and the gamma heating rate could vary by as much as a factor of five due to variations in radial core power distribution and absolute rated power. Therefore, to verify plant-specific applicability of MRP-227-A for the radial boundary components, the guidance document recommends that licensees verify the following variables are within a specified limit:

1. The heat generation figure of merit “F” must be ≤ 68 Watts/cm^3 for both CE and Westinghouse plants, and
2. The maximum average core power density must be less than 110 Watts/cm^3 for CE-design reactors, and less than 124 Watts/cm^3 for Westinghouse-design reactors.

The heat generation figure of merit “F” is calculated from the average core power density, relative fuel assembly power, and generic fuel assembly weighting factors provided in the guidelines. The I&E guidelines addressed in MRP-227-A are valid provided the CE or Westinghouse unit meets these two criteria. The guidance document also indicates that if a plant switched from a low leakage to out-in core loading pattern after 30 years, there would be no impact on the initial inspection, but the re-inspection intervals could be affected.

Upper axial boundary components typically include the Westinghouse UCP and the CE FAP. In its new sensitivity studies, Westinghouse discovered that the UCP and the FAP may exceed the screening limits for IE, and that the elevation of this boundary varies. Therefore, the criteria in the guidance document for the upper axial boundary components were chosen to prevent exceeding the IE threshold above the UCP or the FAP. The criteria include these two limits:

1. The active fuel to FAP distance must be greater than 12.4 inches for CE-design reactors and the active fuel to UCP distance must be greater than 12.2 inches for Westinghouse-design reactors
2. The maximum average core power density must be less than 110 Watts/cm^3 for CE-design reactors and 124 Watts/cm^3 for Westinghouse-design reactors.

The guidance document states that if these criteria are exceeded for more than two years of the plant’s operating life, a plant-specific fluence evaluation should be implemented to determine if the IE fluence threshold is exceeded for components above the UCP or the FAP, and a plant-specific evaluation of the aging management requirements for the affected components would be necessary if the fluence above the UCP and FAP exceeds the IE thresholds.

The guidance document states that the limits for the lower axial components, or components below the reactor core, were evaluated based on the MRP-175 and MRP-191 fluence threshold for IE, and the parameters affecting the lower axial components are identical to those for the upper axial components. The guidance document further states that the primary driver is the fuel assembly geometry, and that, due to design similarities across the fleet, in most cases, the lower axial component geometry is a secondary effect. The guidance document indicates that out-in versus low-leakage operation has only a small effect on the maximum exposure. The guidance document states that, although the neutron flux and the heating rate below the reactor core could be expected to vary in the fleet by as much as a factor of 4 depending on fuel assembly design, core power distribution, and absolute rated power, the variations do not impact the MRP-227-A recommendations for managing aging for the lower axial components in the currently operating CE and Westinghouse fleet.

The guidance document also states that plant-specific applicability of MRP-227-A in the lower axial direction with no further evaluation is demonstrated by meeting the criteria in Section 2.4 of MRP-227-A.

3.3.2 Staff Evaluation

Based on the information provided in the guidance document (Ref. 10), the staff finds the guidance provided for verifying plant-specific applicability of MRP-227-A with respect to radial boundary components is acceptable because it provides a quantitative measure of whether a plant's core loading patterns meet the definition of low-leakage used in the development of the I&E guidelines in MRP-227-A, thus providing assurance that the results of the functionality analyses will remain applicable for the plant. Applicants should provide the heat generation figure of merit "F." With respect to upper axial boundary components, the staff reviewed the criteria proposed in the guidance document (the distance from the active core to the UCP or FAP, and the maximum average core power density), and the proprietary sensitivity studies in WCAP-17780-P (Ref. 9) of the UCP and FAP fluence, and finds that compliance with these criteria provides assurance that components above the UCP and the FAP will not exceed the IE threshold, and thus applicability of the I&E guidelines of MRP-227-A for axial RVI components will be demonstrated. Although the IE threshold may be exceeded in the lower part of the UCP or the FAP for some plants meeting the criteria, the staff agrees that no changes to the MRP-227-A aging management recommendations for the UCP are necessary since IE is not significant without the existence of a crack, and the current MRP-227-A recommendations adequately manage cracking of the UCP through a linked "Primary" component. For the FAP in System 80 CE designs, cracking is managed either through a "Primary" inspection or a TLAA; thus it is adequately managed. For the FAP in non-System 80 CE-design RVI, fatigue and SCC were screened out due to absence of any structural weld. Therefore, the staff concludes that the applicability of the MRP-227-A categorization of CE's non-System 80 FAP as "no additional measures," is acceptable.

As detailed in Section 2.2.6, the staff reviewed the results of the sensitivity studies for the lower axial boundary components in Reference 9, and finds that these studies support the position that there is no effect of these variations on the aging management recommendations of MRP-227-A. Therefore, the staff finds that no specific actions are necessary for lower axial boundary components.

3.4 Question 3 – Extended Power Uprate (EPU)

3.4.1 MRP Evaluation

The MRP evaluation states that, if a plant has implemented an EPU, then all changes, both operational and any physical modifications, should be assessed against the plant-specific AMRs, aging management plan, and any other documents that support the aging management of the RVI. Further, the evaluation states that the effects of the EPU should be considered in the responses to Questions 1 and 2. As long as the average core power and peripheral heat generation limits identified in Question 2 are met, the MRP-227-A I&E guidelines should remain applicable; otherwise, plant-specific evaluations may be necessary.

3.4.2 Staff Evaluation

Based on the information provided in the guidance document (Ref. 10), the staff finds the guidance provided for verifying plant-specific applicability of MRP-227-A with respect to EPU is acceptable because the effects of the EPU are directly identifiable in the plant-specific values of average core power density and peripheral heat generation figure of merit. Therefore, the staff finds that the effects of EPU need not be evaluated separately from plant-specific responses on average core power density and peripheral heat generation figure of merit.
If a plant identifies physical modifications that have been made for the EPU, then a plant-specific evaluation should identify and justify the need for changes to the I&E guidelines for the plant.

3.5 Staff Conclusions – MRP Guidance Document

The staff reviewed the guidance in MRP Letter 2013-025 and concludes that the guidance provides a straight-forward means for applicants and licensees to assess the fabrication and operational characteristics of their plant and to demonstrate if they are consistent with the limits specified in the guidance document, thereby providing assurance that the I&E guidelines of MRP-227-A are applicable to the plant. Therefore, for plants meeting the limits of MRP Letter 2013-025, no modifications to the recommendations of the I&E guidelines in MRP-227-A are necessary, provided that the plant does not implement an out-in core loading pattern, which could result in the invalidation of the I&E guidelines of the MRP-227-A report for the plant. In addition, the staff concludes the guidance on assessing cold work components in MRP Letter 2013-025 provides adequate guidance for responding to generic RAIs on cold worked component. Attachment A to this evaluation provides a template for the staff's evaluation of the licensee's responses to (generic RAI) Question 1 and 2 in SEs of plant-specific RVI Inspection Plans or AMPs.

4.0 CONCLUSIONS

The staff reviewed MRP Letter 2013-025, and the technical basis for the guidance in the letter contained in WCAP-17780-P, and concludes that if an applicant or licensee demonstrates that its plant(s) comply with the guidance in MRP Letter 2013-025, there is reasonable assurance that the I&E guidance of MRP-227-A will be applicable to the specific plant(s). The guidance in MRP Letter 2013-025 provides an acceptable basis for licensees to prepare responses to the generic RAI questions detailed in Section 1.0 of this evaluation:

> Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)
>
> Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

The staff concludes that the information provided on evaluation of cold work in WCAP-17780-P provides an adequate technical basis for the guidance in MRP Letter 2013-025 for responding to Question 1. The staff concludes that the sensitivity studies of variations in neutron fluence, RVI geometry and temperature documented, and the information on power uprate effects on fluence and temperature, documented in WCAP-17780-P, provide an acceptable technical basis for the guidance in MRP Letter 2013-025 for responding to Question 2, For responses to Question 2, an applicant/licensee should provide the plant-specific range or value of the numerical parameters (e.g., the average core power density, the heat generation figure of merit “F”, and the active fuel to FAP distance for CE-design reactors or the active fuel to UCP distance for Westinghouse-design reactors) rather than just stating that the plant complies with the parameter.

5.0 REFERENCES

1. Electric Power Research Institute, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A),” EPRI 1022863, Final Report, December 2011 (ADAMS Accession No. ML120170453).
2. Electric Power Research Institute, “Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191),” EPRI 1013234, November 30, 2006 (ADAMS Accession No. ML091910130).
3. Electric Power Research Institute, “Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232),” EPRI 1016593, 2008 (ADAMS Accession No. ML091671780) – Non-Publically Available.
4. U. S. Nuclear Regulatory Commission Letter, “Summary of November 28, 2012, Category II Public Meeting with the Electric Power Research Institute and Industry Representatives,” January 29, 2013 (ADAMS Accession No. ML13009A066).
5. U. S. Nuclear Regulatory Commission Letter, “Summary of January 22-23, 2013, Closed Meeting with the Electric Power Research Institute and Westinghouse,” February 21, 2013 (ADAMS Accession No. ML13042A048/ML13043A062).
6. U. S. Nuclear Regulatory Commission Letter, “Summary of February 25, 2013, Telecom with the Electric Power Research Institute and Westinghouse Electric Company,” March 15, 2013 (ADAMS Accession No. ML13067A262).
7. U. S. Nuclear Regulatory Commission Letter, “Summary of May 21, 2013, Public Meeting Regarding Pressurized Water Reactor (PWR) Vessel Internals Inspections,” June 24, 2013 (ADAMS Accession No. ML13164A126).
8. U. S. Nuclear Regulatory Commission Presentation: “Status of MRP-227-A A/LAIs 1 and 7,” June 5, 2013 (ADAMS Accession No. ML13154A152).
9. Westinghouse Electric Corporation, “Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs,” WCAP-17780-P, Rev. 0, June 2013 (ADAMS Accession No. ML13183A373) – Non-Publically Available (see ADAMS Accession No. ML13183A372).
10. Electric Power Research Institute, “MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs,” Attachment 1 to MRP 2013-025, October 14, 2013 (ADAMS Accession No. ML13322A454).

Appendix A – Template for Evaluation of MRP-227-A Plant-Specific Applicability for Safety Evaluation of Plant-Specific Reactor Vessel Internals Inspection Plans

Plant-Specific Applicability of MRP-227-A

By letter dated October 14, 2013 (Ref. 1), the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) provided to licensees a non-proprietary guidance document (MRP 2013-025) to address the U.S. Nuclear Regulatory Commission (NRC) staff requests for additional information (RAI) questions related to plant-specific applicability of the generic MRP-227-A inspection and evaluation (I&E) guidelines. MRP 2013-025 relies on a proprietary evaluation from Westinghouse Electric Company (Westinghouse) provided in WCAP-17780-P (Ref. 2), which was discussed with, evaluated by and ultimately found acceptable by the NRC, as described in Reference 3. MRP 2013-025 states that applicant/licensee responses to two questions are sufficient to demonstrate plant-specific applicability of the MRP-227-A I&E:

> Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)
>
> Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

For Question 1, the NRC staff concluded, based on the information provided, that the guidance document would enable licensees/applicants to determine if cold-worked components exist at their plant for which modifications to the MRP-227-A aging management recommendations are necessary to address SCC and IASCC. The staff's basis for this conclusion was as follows:

1. Plant-specific components not identified as having >20 percent cold work in the generic assessment should be identified as having > 20 percent cold work via the licensee's process of performing AMRs for license renewal and performing reviews to respond to MRP-227-A, A/LAIs 1 and 2,
2. Material types with the potential for cold work > 20 percent are identified, which will assist in screening plant-specific components for high levels of cold work and thus susceptibility to SCC and IASCC; and
3. The guidance requires licensees to evaluate the potential for modifications and auxiliary manufacturing processes to induce higher levels of cold work thus increasing the susceptibility of components to SCC or IASCC.

For Question 2, MRP 2013-025 indicated that applicants/licensees should provide information in three areas:

- Evaluation of a heat generation figure of merit, "F," which must be ≤ 68 Watts/cm^3 for both CE and Westinghouse plants,

- The maximum average core power density, which must be less than 110 Watts/cm^3 for CE-design reactors, and less than 124 Watts/cm^3 for Westinghouse-design reactors, and
- The active fuel to fuel alignment plate (FAP) distance for CE-design reactors or the active fuel to upper core plate (UCP) distance Westinghouse-design reactors; this distance must be greater than 12.4 inches for CE-design reactors and greater than 12.2 inches for Westinghouse-design reactors.

As described in Reference 3, the NRC staff reviewed the guidance document and WCAP-17780-P and concluded that compliance by CE and Westinghouse units with the limits specified by the guidance document provides reasonable assurance that the conservative assumptions that were used with respect to the fuel management in the development of I&E guidelines addressed in MRP-227-A are applicable to the individual plants. Therefore, for plants meeting the limits of the guidance document, no modifications to the recommendations of the I&E guidelines in MRP-227-A are necessary, provided that the plant does not implement an out-in core loading pattern, which would result in the invalidation of I&E guidelines of the MRP-227-A report.

In [RAI X], the staff requested that [licensee-plant name(s)] respond to the following questions, which are essentially identical to the questions as defined in the guidance document:

1. Do/Does the [licensee-plant name(s)] reactor vessel internals (RVI) have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so do the affected components have operating stresses greater than 30 ksi? If so, perform a plant-specific evaluation to determine the aging management requirements for the affected components.

2. Have/Has [licensee-plant name(s)] ever utilized atypical fuel design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that plant, including power changes/uprates? If so, describe how the differences were reconciled with the assumptions of MRP-227-A or provide a plant-specific aging management program for affected components as appropriate.

[Brief summary of licensee response to RAI X Question 1 and 2, including how it complies with the guidance document criteria from Reference 2]

The following template can be used as a closure statement for the A/LAI 1.

[licensee--unit name(s)] adequately addressed the two factors (e.g., cold work induced stress and atypical fuel management) that affect the active aging degradation mechanisms in the RVI components, because it confirmed that [unit names(s)] has no materials with either cold work greater than 20 percent, or with cold work greater than 20 percent and normal operating stress greater than 30 ksi, and complies with the criteria defined in the guidance document (ADAMS Accession No. ML13322A454). Furthermore, [licensee--unit name(s)] confirmed that it will continue to comply with these limits during the PEO. Therefore the staff accepts the licensee’s response because of the reasons stated above. A/LAI 1 is therefore resolved for [licensee-unit name(s)].

References [add to reference list for SE]

1. Westinghouse Report, WCAP-17780-P, Rev. 0, “Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs,” June 2013 (ADAMS Accession No. ML13183A373) – Non-Publically Available (ADAMS Accession No. ML13183A372).

2. MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs, Enclosure to MRP Letter 2013-025, October 14, 2013, transmitted via email from K. Amberge to J. Holonich, November 15, 2013. (ADAMS Accession. No. ML13322A454).

3. Office of Nuclear Reactor Regulation Evaluation of WCAP-17780-P, “Reactor Internals Aging Management MRP-227-A Applicability For Combustion Engineering and Westinghouse Pressurized Water Reactor Designs,” and MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs ADAMS Accession. No. ML14309A484).