ELECTRIC POWER RESEARCH INSTITUTE

MRP Materials Reliability Program MRP 2013-025

(via email)

October 14, 2013

To: MRP Assessment TAC and MRP IC

Subject: MRP-227-A Applicability Template Guideline

Enclosed please find an MRP-227-A related guidance document for MRP member use in developing reactor internals related information for plant-specific inspection programs. Over the past year, several public meetings were held with the NRC staff to discuss NRC expectations and concerns regarding industry responses to MRP-227-A, Applicant/Licensee Action Items (A/LAIs) 1 and 2. The concerns were addressed to owners of currently operating pressurized water reactor plants designed by Westinghouse and Combustion Engineering (CE). At these meetings, the NRC, Westinghouse, Electric Power Research Institute (EPRI), and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A.

The information provided by the industry to the NRC staff demonstrated that the MRP-227-A I&E Guidelines are applicable for the range of conditions expected at the currently operating Westinghouse and CE-designed plants in the United States. As a result of the technical discussions with the NRC staff, the Enclosure was developed to provide utilities with the basis for a plant to respond to the NRC's Request for Additional Information (RAI) to demonstrate compliance with the basic technical applicability assumptions in MRP-227-A for originally licensed and uprated conditions.

Best Regards,

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EPRI Materials Reliability Program

Tim Wells
Chairman, MRP IC
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ENCLOSURE TO
EPRI letter
MRP 2013-025

Attachment 1:

MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs

BACKGROUND

The Safety Evaluation (SE) [1] issued on Materials Reliability Program (MRP) technical report MRP-227 by the U.S. Nuclear Regulatory Commission (NRC) contained eight Applicant/Licensee Action Items (A/LAIs). These eight action items must be completed in the implementation of the Inspection and Evaluation (I&E) Guidelines outlined in MRP-227-A [2].

On November 28, 2012, a public meeting was held [3] at the NRC office to discuss staff expectations and concerns regarding industry responses to A/LAIs 1 and 2. The concerns were addressed to owners of currently operating pressurized water reactor plants designed by Westinghouse and Combustion Engineering (CE). A series of proprietary and public meetings were conducted from January to June of 2013 [4, 5, 6, and 7]. At these meetings, the NRC, Westinghouse, The Electric Power Research Institute (EPRI), and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A.

Westinghouse summarized the proprietary meeting presentations and supporting proprietary generic design basis information in WCAP-17780-P [8], and provided it to the NRC. WCAP-17780-P provides background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by Westinghouse and CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A. 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 [5],
2. criteria defined in MRP-227-A, Section 2.4, and
3. plant-specific regulatory commitments for managing aging in reactor internals.

The NRC staff indicated in [7] that the information provided by the industry to the NRC staff demonstrated that the MRP-227-A I&E Guidelines are applicable for the range of conditions expected at the currently operating Westinghouse and CE-designed plants in the United States. As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC's Request for Additional Information (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 [5 and 7]:

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 objective of this document is to provide a simple, non-proprietary means of demonstrating plant-specific applicability of the MRP-227-A inspection sampling recommendations for managing aging in currently operating U.S. CE and Westinghouse plants for NRC Questions 1 and 2. Plants that exceed the thresholds defined in this document do not necessarily fall outside the MRP-227-A recommendations. Instead, they may require additional plant-specific evaluations to fully demonstrate plant-specific applicability. Technical background and direction to support developing the plant-specific responses to NRC Questions 1 and 2 are included in this document. Appendices A and B provide a sample template for submittal of responses for Questions 1 and 2, respectively.

GUIDANCE ON PLANT-SPECIFIC APPLICABILITY DEMONSTRATION

Guidance for demonstrating compliance with MRP-227-A conditions in responding to NRC Questions 1 and 2 and assessments for uprated conditions (identified as Question 3), if applicable, is provided in the following sections.

Question 1 Cold-worked Materials

> *" 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 1 Response Guidance

The cold work process (also known as "work hardening" or "strain hardening") strengthens an alloy by deforming it in a plastic manner, generating dislocations that prevent further dislocations from moving. Cold working can be performed purposefully to add strength to a material or as a byproduct of a forming process to achieve a desired shape. MRP-175 [10] provides the foundational criteria for the industry screening limits that were used in the MRP-191 [9] susceptibility ranking assessments supporting the MRP-227-A inspection sampling recommendations. MRP-175, Section 3.2.3 on cold work contains the following:

> *"Manufacturing records and practices, to the extent that these are recoverable, should be reviewed to determine locations that may have been cold-worked during fabrication. Alternatively, this information may be available through expert knowledge. Cold-work (giving rise to an elevation of material yield strength) results from a number of things, including procurement in the cold-worked condition (e.g., cold-worked bolting), intentional grinding or bending during fabrication, and shrinkage strains associated with welded component items. The screening criterion applies to anything that creates the equivalent of ≥ 20% cold-work in an austenitic stainless steel component item or weld. For austenitic PH stainless steel (Alloy A-286) material, hot-heading of bolts, which can create a HAZ between the head and shank, is another known adverse factor. Shot-peened bolts that are preloaded to high stress levels, where a stress reversal of the compressively stresses layer might occur, are also of concern. Component items that fall into these categories should be regarded as having potentially high residual tensile stresses for the screening process."*

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% cold work outside of those already identified in MRP-191 [9] and managed under the requirements of MRP-227-A. Specifically, the following actions are identified:

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 do 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.

The plant materials list shall be binned according to the following categories:

Category 1: Cast Austenitic Stainless Steel (CASS)
Category 2: Hot-formed Austenitic Stainless Steel
Category 3: Annealed Austenitic Stainless Steel
Category 4: Fasteners Austenitic Stainless Steel
Category 5: Cold-formed Austenitic Stainless Steel (without subsequent solution annealing)

Component material Categories 1, 2, and 3 do not exceed the 20% cold work criterion in the generic assessments. Components in Categories 4 and 5 are generically identified as cold worked, however, manufacturer's specifications for fastener materials may include limits on yield or tensile strength that would preclude cold work greater than 20%. Components in Categories 1, 2 and 3 are generically considered as not cold-worked and therefore not susceptible to SCC. Category 4 components may employ materials that exceed the 20% cold worked limit. Licensees are responsible for determining that plant specific implementations do not exceed the 20% cold worked limit. Plant material records and specifications may identify those Category 4 fastener components that would exhibit less than 20% cold-work. These components would be considered to be not susceptible to SCC. The remaining Category 4 components along with the Category 5 components are generically considered susceptible to SCC and should be screened in for this mechanism if the component's tensile stress exceeds 30 ksi. If a new component subject to SCC degradation is confirmed, placement in the ranking susceptibility order relative to the MRP-232 [11] generic evaluation and the basis for the decision should be documented as part of the plant-specific evaluation.

The impact of any auxiliary manufacturing or installation processes that could have introduced significant strain hardening in the component should also be considered. These processes could potentially move Category 1, 2, 3 or 4 components into Category 5. Assessments of field fit-up and auxiliary processes that could introduce cold work are very subjective but such adjustments would be expected to provide only minor increases of cold work and minor increases of potential susceptibility to SCC. Additional macroscopic strain hardening induced by fit-up would be expected to be small compared to the 20% allowable. The most significant additional cold work induced by these processes would be expected to be quite local and would be considered to primarily affect the surface of the component. The plant owner should be aware of the impact of these activities on SCC initiation. There is no requirement for the owner to search for records of field fit-up or operational activities, nor is there a requirement for the owner to change the MRP-227-A inspection strategy to incorporate these effects. However, any records of such

actions being performed during component assembly and installation that may be discovered during the execution of the component categorization process should be noted.

Question 2 Fuel Design or Fuel Management

> *"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?"*

Question 2 Response Guidance

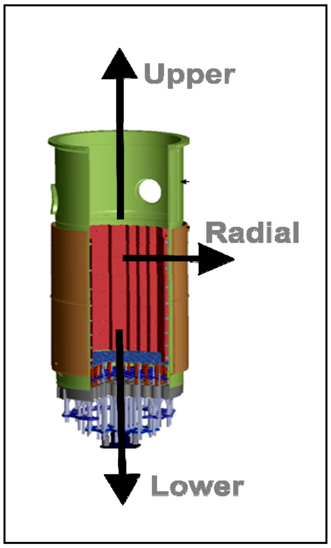
The MRP-227-A inspection recommendations require extensive inspections of the components surrounding an active core. Additional inspection requirements are not anticipated. The aging degradation analysis completed in the development of MRP-227-A indicated decreased degradation rates under the low leakage core loading assumptions assumed during the second 30 years of operation. To provide assurance that there would not be higher than anticipated rates of degradation in the later years of operation, MRP-227-A guidance on applicability of the recommendations precludes return to out-in core loading patterns. MRP-227-A does not provide a quantitative definition of out-in core loading that could be used to evaluate the potential impact on internals degradation. The primary impact of out-in core loading is increased rates of nuclear heat generation in the reactor internals.

Plant-specific fuel design or fuel management application were identified as having potential to invalidate the fluence, temperature, and stress assumptions used to develop MRP-227-A. The goal in the development of this guideline was to define a simple parameter to demonstrate that the assumptions of MRP-227-A are representative of the plant. Subsection 4.3.2 of MRP-191 states that:

> *"The core power density that was assumed for projecting the fluence values for the Westinghouse plant was 104.5 W/cm^3. The core power density assumed in the analysis of the CE-designed plant was 83.0 W/cm^3. Higher core power densities result in higher neutron fluence values."*

The largest impact of changes in the fluence distribution on recommendations occurs in components near the lower bound fluence for susceptibility. Any limit on fluence or on the heat generation rate is effectively a limit on core power spatial distribution.

Three different boundaries were explored in developing this guideline:



1. radial boundary (components laterally surrounding core)
2. upper axial boundary (components above core)
3. lower axial boundary (components below core)

Radial Boundary Limitations

The primary driver for the radial core power distribution is the MRP-227-A basis of 30 years of "out-in" management, where fresh fuel is placed in peripheral core locations, followed by 30 years of low leakage fuel management. Any change in this scenario has the potential to impact re-inspection, but not the initial inspection timing or affected components. Due to design similarities across the currently operating Westinghouse and CE U.S. fleet, in most, but not all, cases, internals component geometry is a secondary effect. Neutron flux and heating rate could be expected to vary by as much as a factor of 5, depending on radial core power distribution and absolute rated power. Local effects at key locations are dominated by a few (typically 3-50) fuel assemblies located on the core periphery. There is no impact on the initial inspection, but a change from the low leakage operating characteristics during the second 30 years of operation could impact the MRP-227-A re-inspection recommendations. To provide assurance that there would not be higher than anticipated rates of degradation in the later years of operation, MRP-227-A guidance on applicability of the recommendations precludes return to out-in core loading patterns. The limitations on power for demonstrating applicability in the peripheral assemblies provided in this guideline preclude return to the more damaging out-in core loading pattern.

Plant-specific applicability of MRP-227-A in the radial direction with no further evaluation required is demonstrated by meeting the following limits:

CE: heat generation figure of merit, $F \leq 68$ Watts/cm^3
average core power density < 110 Watts/cm^3

Westinghouse: heat generation figure of merit, $F \leq 68$ Watts/cm^3
average core power density < 124 Watts/cm^3

The figure of merit (Figure 1), F, is given by:

$$F = P_{avg}(W_1 * R_1 + W_2 * R_2 + W_3 * R_3 + W_4 * R_4)$$

In the previous equation:

P_{avg} = average core power density
W_i = generic inside corner weighting factor (Figures 1 and 2)
R_i = relative fuel assembly power

The limiting core power values determined for the plant-specific assessment apply to operation going forward. Plants that do not meet these guidelines may require additional evaluation to demonstrate the applicability of the MRP-227-A recommendations. Plants that exceed the radial boundary limitations should evaluate the impact on the re-inspection schedules for components in the Westinghouse baffle/former/barrel/shield or the CE core shroud/barrel/shield structure.

Upper Axial Boundary Limitations

The limitations for the upper axial components or components above the reactor core, typically the Westinghouse upper core plate or the CE fuel alignment plant, are based on the MRP-175 and MRP-191 fluence threshold for irradiation embrittlement. The primary driver is the fuel assembly geometry, noting the position of the active fuel stack and the use of axial blankets. Similar to the radial components due to design similarities across the fleet, in most cases, upper axial component geometry is a secondary effect. Contrary to the radial boundary components, out-in versus low leakage operation has only a small effect on the maximum exposure. Neutron flux and heating rate above 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. Variations could impact screening results for components above the core; some plants may exceed the screening criterion, others may not.

Plant-specific applicability of MRP-227-A in the upper axial direction with no further evaluation required is demonstrated by meeting the following limits:

CE: active fuel to fuel alignment plate distance > 12.4 inches
average core power density < 110 Watts/cm^3

Westinghouse: active fuel – upper core plate distance > 12.2 inches
average core power density < 124 Watts/cm^3

Evaluations shall consider the entire plant operational period, original and extended life. Plants that exceed the limits for more than two years of operation would need to provide further evaluations to demonstrate compliance with this applicability requirement. A plant-specific analysis may be required to demonstrate that the fluence above the upper core plate or fuel alignment plate does not exceed the irradiation embrittlement screening threshold. In the event that this fluence limit is not met, an evaluation of potential irradiation embrittlement in components immediately above the plate may be required.

Lower Axial Boundary Limitations

The limits for the lower axial components or components below the reactor core were evaluated based on the MRP-175 [10] and MRP-191 [9] fluence threshold for irradiation embrittlement. The parameters affecting the lower axial components are identical to those for the upper axial components. The primary driver is the fuel assembly geometry. Due to design similarities across the fleet, in most cases, lower axial component geometry is a secondary effect. Out-in versus low leakage operation has only a small effect on the maximum exposure. 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. However, 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.

Plant-specific applicability of MRP-227-A in the lower axial direction with no further evaluation required is demonstrated by meeting the MRP-227-A, Section 2.4 criteria.

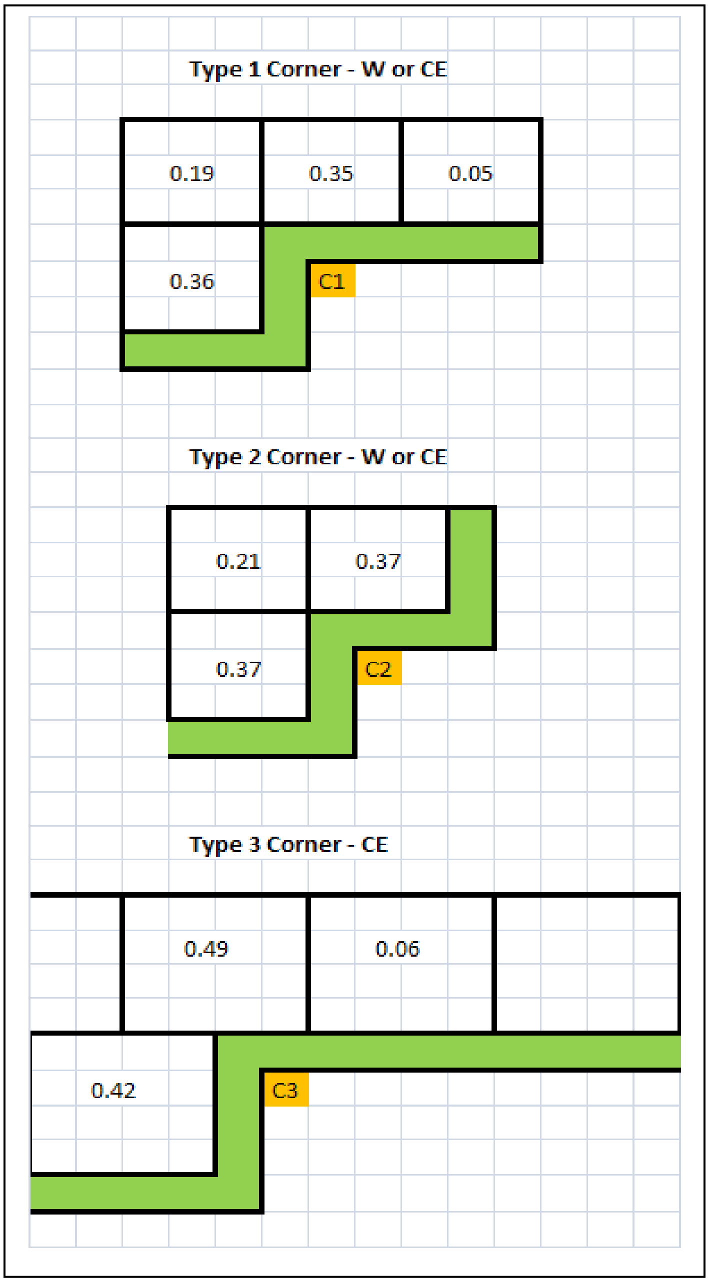


Figure 1: Weighting Factors for Typical Re-entrant Corner Configurations

Notes:

1. Mirror image configurations should also be considered.
2. In the calculation of the core power density, the core volume is taken to be equal to (number of fuel assemblies) x (fuel assembly cross-sectional area) x (active fuel height).
3. The relative fuel assembly power is taken to be the average of the fuel cycle design being evaluated.
4. The weighting factors for use in the figure of merit equation are indicated. The three corner configurations shown can be used to define key locations in all CE and Westinghouse cores.

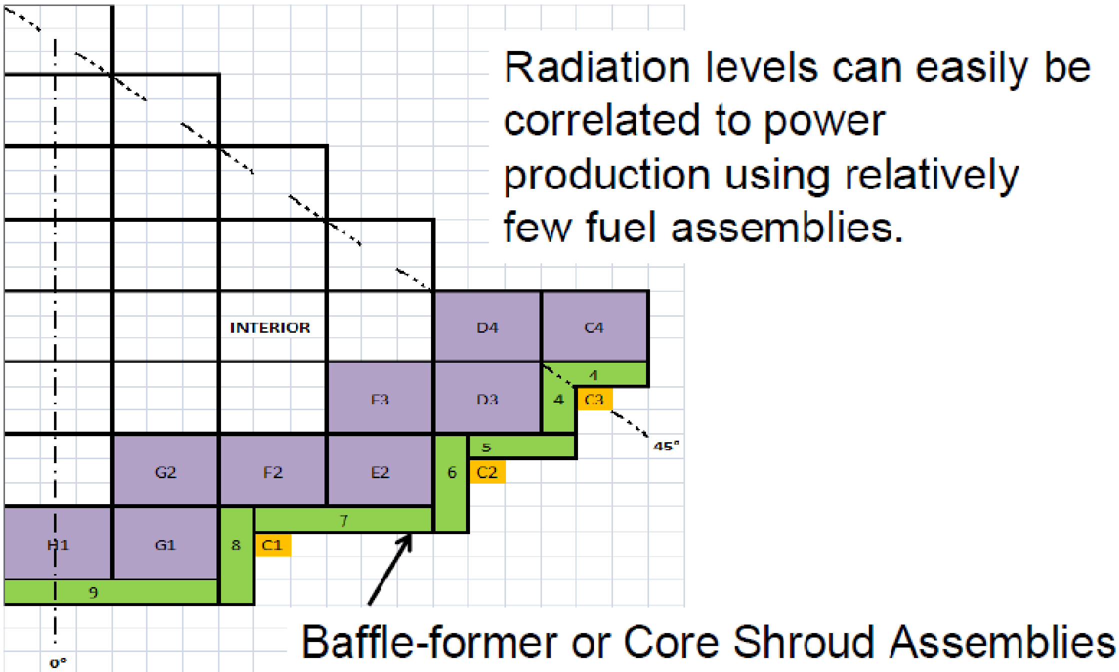


Figure 2: Typical Plant Geometry

Question 3 Extended Power Uprate (EPU)

"If the plant implemented an Extended Power Uprate (EPU), are the peak internal metal temperatures within the assumptions made in developing MRP-227-A?"

Question 3 Response Guidance

If the plant implemented an EPU, all changes should be assessed against the plant-specific AMR, aging management plan, and any other pertinent documents that support the aging management of reactor internals (including physical modifications to reactor internals supporting the EPU). Question 1 should be evaluated for the EPU, and responses should be provided as applicable. If there were no physical modifications to the reactor internals as a result of the EPU, then the applicability requirements outlined in the response to Question 2 should be re-examined for EPU conditions. An EPU results in increases in average core power. The MRP-227-A I&E Guidelines should remain applicable as long as the average core power and peripheral heat generation limits imposed remain bounding. If these criteria are not met, plant-specific evaluations may be required to demonstrate the applicability of the guideline.

The utility response with regard to the applicability of MRP-227-A for the uprated condition should consist of appropriate text to identify the uprated plant condition and should contain responses to Questions 1 and 2.

CONCLUSION

To demonstrate plant-specific applicability of the MRP-227-A sampling inspection strategy for managing aging in reactor internals, licensees must demonstrate that the criteria of MRP-227-A, Section 2.4 are met, and that the neutron fluence and heat generation rates are within the range of the following variables summarized. The limiting threshold values are:

- active fuel – upper core plate distance > 12.2 inches for Westinghouse plants
- active fuel to fuel alignment plate distance > 12.4 inches for CE plants
- average core power density < 124 Watts/cm^3 for Westinghouse plants
- average core power density < 110 Watts/cm^3 for CE plants
- heat generation figure of merit, F $\leq$ 68 Watts/cm^3 for Westinghouse and CE plants

Plants that exceed the thresholds defined in this document do not necessarily fall outside the MRP-227-A recommendations; instead, they require additional evaluations to fully demonstrate plant-specific applicability. The limiting core power values apply to operation going forward, and should be based on the full as-licensed operational (original and extended life) life of the plant. A plant that maintains core loading patterns that meet the limits would satisfy the MRP-227-A requirement to avoid operation above this limit. Short periods of operation (fewer than two years) above this limit would not invalidate the requirement to not return to "out-in" fuel management. Plants that exceed the limits for more than two years of operation would need to provide further evaluations to demonstrate compliance with this applicability requirement.

This report provides simple guidelines to respond to NRC RAIs to demonstrate plant-specific applicability of MRP-227-A and to evaluate the impact of an EPU on the applicability of the MRP-227-A recommendations. A plant-specific evaluation demonstrating compliance with the criteria defined in this report, with the criteria defined in MRP-227-A, Section 2.4, and with any additional plant-specific regulatory commitments for managing aging in the reactor internals will be sufficient to demonstrate the applicability of MRP-227-A.

REFERENCES

1. U. S. Nuclear Regulatory Commission Letter, "Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680)," December 16, 2011. (ADAMS: ML11308A770)
2. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.
3. 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: ML13009A066)
4. 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: ML13042A048/ml13043A062)
5. 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: ML13067A262)
6. 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: ML13164A126)
7. U. S. Nuclear Regulatory Commission Presentation: "Status of MRP-227-A Action Items 1 and 7," June 5, 2013 (ADAMS: ML13154A152).
8. 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.[3]
9. *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)*. EPRI, Palo Alto, CA: 2006. 1013234.
10. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)*. EPRI, Palo Alto, CA: 2005. 1012081.
11. *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internal Components (MRP-232, Revision 1)*. EPRI, Palo Alto, CA: 2012. 1021029.

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[3] Though this reference is proprietary, it is a specific document released to the NRC on June 28, 2013 to fulfill the NRC's request to demonstrate plant-specific applicability of MRP-227-A for plants designed by Westinghouse and Combustion Engineering.

Appendix A:
MRP-227-A Applicability Guideline for CE and Westinghouse Pressurized Water Reactor Designs, Question 1 Response Template

"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.)"

RESPONSE

Plant X Unit Y has evaluated reactor internals components according to the MRP-191 [reference] industry generic component listings and screening criteria (including consideration of cold work as defined in MRP-175 [reference], noting the requirements of Section 3.2.3). In addition to consideration of the material fabrication, forming, and finishing process, a general screening definition of a resulting reduction in wall thickness of 20% was applied as an evaluation limit. It was confirmed that ***Option 1 or Option 2***...

Option 1

*"...all of **Plant X Unit Y** components, as applicable for the design, are included in the MRP-191 component lists with no exceptions. The evaluation included a review of all plant modifications affecting reactor internals and operating history, which were determined to be consistent with MRP-191 considerations. The original vendor-supplied components or any subsequently modified components were procured according to ASTM International or ASME material specifications through applicable quality controlled protocols."*

Option 2

*"...all of **Plant X Unit Y** components, as applicable for the design, are included directly in the MRP-191 component lists, except for the components identified in **Table X**.*

Table X

MRP-227 Component	***Material (Form/Fabrication)***	***Category(1)***	***Cold Worked (CW) 20% Assessment(2)***	***Comments***
Guide Card	*ASTM A351, Type 304, Gr. CF8 (Plate/CASS)*	*1*	*N*	
Baffle-former bolt	*Type 316 SS*	*4*	*Y*	*Replacement Bolt*

Notes:

(1) Categories include the following:
1. *cast austenitic stainless steel (CASS)*
2. *hot formed austenitic stainless steel*
3. *annealed austenitic stainless steel*
4. *fasteners austenitic stainless steel*
5. *cold formed austenitic stainless steel (without subsequent solution annealing)*

(2) CW potential based on MRP-227-A generic criteria:
- *N applies to categories 1, 2, and 3.*
- *Y applies to categories 4 and 5.*

*The evaluation included a review of all plant modifications affecting reactor internals and operating history. The original vendor-supplied components or any subsequent components were procured according to ASTM International or ASME material specifications through applicable quality controlled protocols. The evaluation concluded that **Option A or Option B**...*

Option A
"...there was no impact to the MRP-227-A sampling inspection aging management requirements based on... (provide basis for conclusion)."

Option B
"...there was an impact to the MRP-227-A sampling inspection aging management requirements based on... (provide basis for conclusion, detailed description of deviations from generic requirements, and actions taken to address)."

Appendix B:
MRP-227-A Applicability Guideline for CE and Westinghouse Pressurized Water Reactor Designs, Question 2 Response Template

"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?"

RESPONSE

OPTION 1

Plant X Unit Y complies with the MRP-227-A assumptions regarding core loading/core design. Neutron fluence and heat generation rates are concluded to be ***Option A or Option B***...

Option A
"...acceptable based on the following assessment to the limiting MRP guidance threshold values [MRP letter reference]:

Combustion Engineering:
- ***Plant X Unit Y*** *active fuel to fuel alignment plate distance > 12.4 inches*
- ***Plant X Unit Y*** *average core power density < 110 Watts/cm^3*
- ***Plant X Unit Y*** *heat generation figure of merit, F ≤ 68 Watts/cm^3*

Westinghouse:
- ***Plant X Unit Y*** *active fuel to upper core plate distance > 12.2 inches*
- ***Plant X Unit Y*** *average core power density < 124 Watts/cm^3*
- ***Plant X Unit Y*** *heat generation figure of merit, F ≤ 68 Watts/cm^3*

Option B
"*...unacceptable based on an assessment to the limiting MRP guidance threshold values [MRP letter reference,] but are determined to be acceptable based on…(Utility to provide basis for conclusion)."*

OPTION 2

Plant X Unit Y does not comply with the MRP-227-A assumptions regarding core loading/core design.

"(Utility to provide justification for application of MRP-227-A for aging management of reactor internals)."