ELECTRIC POWER RESEARCH INSTITUTE

MRP Materials Reliability Program MRP 2014-019

(via e-mail)

July 8, 2014

Subject: Transmittal of Westinghouse Guidelines for Responding to U.S. NRC Questions Related to MRP-227-A Applicant/Licensee Action Item 7 for Combustion Engineering and Westinghouse Pressurized Water Reactor Lower Support Column Designs

To: MRP IC and Assessment TAC

This letter transmits a non-proprietary utility guideline prepared by Westinghouse to support responding to United States (U.S.) Nuclear Regulatory Commission's (NRC's) questions related to MRP-227-A Applicant/Licensee Action Item 7 (A/LAI 7) for Combustion Engineering and Westinghouse pressurized water reactor lower support column designs. The information provided in the Attachment was provided to the Electric Power Research Institute (EPRI) to transmit to U.S. Materials Reliability Program members. This information is intended to complement the discussions of the November-December 2013 industry meetings with the NRC on A/LAI 7.

NRC's Safety Evaluation (SE) on MRP-227-A includes eight Applicant/Licensee Action Items (A/LAIs). This guideline is specifically related to Action Item 7 and its application to cast austenitic stainless steel (CASS) lower support columns (LSCs). In some Westinghouse and Combustion Engineering (CE) designed plants the LSC length is manufactured using CASS. Components manufactured using CASS materials and those in regions of high neutron fluence (e.g., LSCs) have been identified as being potentially susceptible to thermal embrittlement (TE) and/or irradiation embrittlement (IE). A/LAI 7 requires plants to demonstrate, by plant-specific analysis or evaluation, that the MRP-227-A recommended inspections will ensure functionality of the LSCs until the next scheduled inspections. These guidelines were prepared to support licensees in responding to A/LAI 7.

Sincerely,

B. C. Rudell
Chairman, Integration Committee
EPRI-Materials Reliability Program

Anne Demma
Program Manager
EPRI- Materials Reliability Program

Attachments: LTR-RIAM-14-46
NSD-EPRI-14-6

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Westinghouse

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cc: Laura L. Genutis, Rick J. Gradomski, Jeffrey R. Stukus, Patricia C. Paesano, Jun C. Bae, Cheryl L. Boggess, David C. Radonovich, Nicholas A. Szweda

Date: June 27, 2014

From: Reactor Internals Aging Management
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Your ref: N/A
Our ref: LTR-RIAM-14-46

Subject: **Guidelines for Responding to U.S. NRC Questions Related to MRP-227-A Applicant/Licensee Action Item 7 for Combustion Engineering and Westinghouse Pressurized Water Reactor Lower Support Column Designs**

References:

1. EPRI Contract, EP-P44330/C19232, "Westinghouse Reactor Internals Management Support," Including:
   a) MA 10001231, "NRC Follow-up Items Related to Generic Applicability of MRP-227 (amendment MA10001231)," October 3, 2013.
   b) Amendment 1 to Agreement MA 10001231, "NRC Follow-up Items Related to Generic Applicability of MRP-227," January 23, 2014.
2. U. S. Nuclear Regulatory Commission Letter, "Summary of November 19, 2013, Public Meeting to Discuss on the Resolution of Plant-specific Action Items Related to Materials Research Program-227-A Reactor Internals Aging Management Programs/Inspection Plans," January 15, 2014 (ADAMS Accession No: ML13345A020).
3. U. S. Nuclear Regulatory Commission Letter, "Summary of the December 3, 2013, Closed Meeting on the Resolution of Applicant/Licensee Action Items Related to Materials Reliability Program-227-A Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," February 10, 2014 (ADAMS Accession No: ML13345A320).

Attachment: 1. Guidelines for Responding to U.S. NRC Questions Related to MRP-227-A Applicant/Licensee Action Item 7 for Combustion Engineering and Westinghouse Pressurized Water Reactor Lower Support Column Designs

The purpose of this letter is to request transmittal of Attachment 1. Attachment 1 contains the non-proprietary utility guideline to support responding to United States (U.S.) Nuclear Regulatory Commission's (NRC's) questions related to MRP-227-A Applicant/Licensee Action Item 7 (A/LAI 7) for Combustion Engineering and Westinghouse pressurized water reactor lower support column designs. The information provided in Attachment 1 is for the Electric Power Research Institute (EPRI) to transmit to U.S. Materials Reliability Program members [1]. This information is intended to complement the discussions of the November-December industry meetings [2 and 3] on A/LAI 7.

Please transmit Attachment 1 to Kyle Amberge, Robin Dyle, and Anne Demma of EPRI, and include Glenn Gardner (Millstone) and Bernard Rudell (CENG) on copy. Please include all Westinghouse addressees on copy in the project letter distribution.

If there are any questions or the need for additional information, please contact Jun Bae at (860) 731-1778.

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Attachment 1:

Guidelines for Responding to U.S. NRC Questions Related to MRP-227-A Applicant/Licensee Action Item 7 for Combustion Engineering and Westinghouse Pressurized Water Reactor Lower Support Column Designs

BACKGROUND

The United States (U.S.) nuclear industry [through the Materials Reliability Program (MRP)] has developed inspection and evaluation guidelines (MRP-227-A [1]) for the management of reactor internals age-related degradation issues in the U.S. pressurized water reactor (PWR) fleet. MRP-227-A focuses on those internals components identified as susceptible to aging effects. MRP-227-A provides information to support effective aging management, while simultaneously maintaining safety and reliability. The inspection and evaluation recommendations in MRP-227-A were issued under the Nuclear Energy Institute NEI-03-08 protocol for industry-wide implementation. MRP-227-A provides a common basis for reactor internals inspection plans for all U.S. PWR plant-specific Aging Management Program Plans to support the concept of an industry-wide living program.

The NRC Safety Evaluation (SE) on MRP-227-A includes eight Applicant/Licensee Action Items (A/LAIs). This guideline is specifically related to Action Item 7 and its application to cast austenitic stainless steel (CASS) lower support columns (LSCs). In some Westinghouse and Combustion Engineering (CE) designed plants, at least a portion of the LSC length is manufactured using CASS. Components manufactured using CASS materials and those in regions of high neutron fluence (e.g., LSCs) have been identified as being potentially susceptible to thermal embrittlement (TE) and/or irradiation embrittlement (IE). Because of this potential for embrittlement and a concern for the combined effects of TE and IE, there is a concern that small flaws, which are inherent in the casting process, could grow and cause failures without detection. As a result of this concern, the NRC has included A/LAI 7 as part of the SE to MRP-227-A. A/LAI 7 requires plants to demonstrate, by plant-specific analysis or evaluation, that the MRP-227-A recommended inspections will ensure functionality of the LSCs until the next scheduled inspections.

A/LAI 7 indicates that a plant-specific evaluation to demonstrate functionality of the LSCs would need to consider the following items:

1) The possible loss of fracture toughness in these components due to TE and IE.
2) Limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques.
3) All licensing basis conditions of operation.

On November 19, 2013, a public meeting was held [2] at the NRC office to discuss staff expectations and questions regarding industry responses to A/LAI 7. The questions were addressed to owners of currently operating PWR plants designed by Westinghouse and CE. A follow-on proprietary meeting was conducted on December 3, 2013 [3]. At these meetings, the NRC, Westinghouse, the Electric Power Research Institute (EPRI), and utility representatives discussed regulatory questions. These groups determined a path for a comprehensive and consistent utility response to demonstrate that the MRP-227-A recommended inspections will ensure functionality of the LSCs until the next scheduled inspections. Westinghouse submitted the proprietary meeting presentations containing supporting proprietary generic design basis information in LTR-NRC-14-7 [4]. Non-proprietary presentations from the public meeting with the NRC on November 19, 2013[2] and the closed meeting on December 3, 2013 [4] were also provided.

The NRC indicated in [3] that, based on the information provided by the industry, it may be possible to resolve A/LAI 7 for some plants using a screening approach if plant-specific material

information is available (in particular, the ferrite content of the CASS LSC bodies). To use such a screening approach, the NRC indicated that the acceptable maximum ferrite content would need to be revised from existing NRC guidance for aging management of CASS to account for the higher neutron fluence received by CASS reactor vessel internals (RVI) components.

The objective of this document is to provide a means of demonstrating LSC functionality in U.S. CE and Westinghouse plants by employing the ferrite content screening option to resolve A/LAI 7. Westinghouse supports use of the industry recommendations for ferrite screening as provided in BWRVIP-234; however, the approach outlined in this document is independent of the screening criteria. Plants that exceed the screening limits referenced in this document may require additional plant-specific evaluations to fully demonstrate functionality of potentially embrittled LSCs. In that case, alternative responses referring to ongoing work through the Pressurized Water Reactor Owners Group (PWROG) are provided. Technical background and direction to support developing the plant-specific responses are included in this document.

GUIDANCE ON PLANT-SPECIFIC DEMONSTRATION OF LSC FUNCTIONALITY

The presentations given during the November and December public and closed meetings provide non-proprietary and proprietary information on Westinghouse and CE plants to support NRC reviews of utility submittals to demonstrate functionality of the LSCs. During these meetings, the following topics were covered:

- material
- fabrication
- failure modes, effects, and criticality analysis (FMECA)
- operating experience
- inspection accessibility and feasibility
- design and safety considerations

The basic conclusions resulting from the interactions with the NRC staff were:

- The plant-specific acceptance criteria for dispositioning MRP-227-A LSC inspection results used commercially bounding acceptance criteria that are more limiting than what would be required to support safety considerations. Therefore, the results of these analyses should not be considered an accurate reflection of the level of redundancy available for the lower support structure to maintain its intended safety function.
- LSCs that screen out for TE will potentially remain susceptible to IE. However, in this case, the embrittlement of the CASS material would be no different than that of wrought material.
- Fabrication of LSC included manufacturing controls and non-destructive examination (NDE) requirements that are in line with current industry standards for core support structure castings to limit the potential flaw sizes that could have potentially been placed into service.
- The standard FMECA process [5] used in the development of MRP-191 [9] component categorization identified TE and IE as potential degradation mechanisms for the LSCs. They

identified these as FMECA Group 1, with a low likelihood of damage as a result of LSC failure.

- Predominate functional loadings on the LSC are compressive in nature. They are not conducive to propagating small manufacturing flaws or causing catastrophic failures of LSC.
- Potential failure modes as a result of cracking are not likely to result in the inability of the lower support structure to perform its function.
- There is no industry evidence of material degradation of the LSC bodies.
- The columns are in an unfavorable location for performing detailed inspections using currently available inspection techniques.

For responses to the NRC related to functionality of the CASS LSCs, it is recommended that the utilities reference the documentation from the November and December meetings and cite these general conclusions to provide a generic qualitative demonstration of functionality. However, as stated in [3], the NRC would still expect to have, paired with these arguments, a demonstration of low ferrite content and a summary of pre-service fabrication inspection requirements. Therefore, confirmation of plant-specific fabrication records should also be conducted to complete this demonstration. To satisfy this need, plants should undergo a search and review of LSC fabrication records, drawings, and pertinent reports to obtain the following plant-specific information:

1) Chemistry of the material used in the fabrication of each of the installed CASS LSCs, generally available in certified material test reports (CMTRs) or applicable ASME/ASTM specifications.
2) Casting method used (static or centrifugal).
3) NDE techniques and acceptance criteria specified for the installed CASS LSCs.

The elemental percentages from the chemical data retrieved from CMTRs for the LSCs are input into Hull's formula (per guidance of [6]) to calculate the delta ferrite content of the CASS material. Based on input from the NRC, the preference is to have a complete set of delta ferrite data available for all installed LSCs.

It is preferred that the following data be provided for each support column or set of support columns grouped by heat number:

1) supplier heat code
2) casting method
3) calculated ferrite content

In the event that not all data is available, it may be possible to use statistics to estimate the ferrite content of LSCs where data is missing. Furthermore, in the event that the required data is sparse or unavailable for a given plant, and the column vendor is known, it may be possible to perform a statistical evaluation using fabrication records from columns manufactured from that same vendor for a different plant or set of plants. It should be noted that there is currently no formally established precedence for NRC acceptance of delta ferrite data based on statistical evaluations. However, this is likely to be a reasonable alternative approach provided that a statistically

significant amount of data is used to perform the evaluations and appropriate justification is provided for the applicability of the data that is used.

Currently, there are two different guidance documents related to the screening limits for TE: the industry position provided in BWRVIP-234 [7] and the draft NRC position provided in ADAMS Accession No ML14072A012 [8]. Presently, Westinghouse supports the use of the industry guidance in [7] and recommends the use of this information for responding to requests from the NRC related to LSC screening for TE. It is recommended, however, that the plants keep current with industry developments on this topic and adjust their responses appropriately.

In addition to the ferrite content information, the response should also include a summary of the NDE techniques that were specified for the installed columns [i.e., radiography (RT) and liquid penetrant (PT)]. NDE acceptance criteria, such as allowable flaw sizes and distribution (typical for PT) or reference standard (typical for RT), should be included in this summary to provide the staff with a relative understanding of the flaw sizes that could have potentially placed into service. This also provides plant-specific confirmation of the general arguments provided during the November and December meetings.

The following is an example of wording that could be used in the response:

> *In a closed meeting with the Nuclear Regulatory Commission (NRC) staff on December 3, 2013 [ADAMS Accession No: ML13345A320] proprietary information related to the following topics were presented to support NRC reviews of utility submittals related to the functionality of the lower support columns (LSCs):*
>
> - *material*
> - *fabrication*
> - *failure modes, effects, and criticality analysis (FMECA)*
> - *operating experience*
> - *inspection accessibility and feasibility*
> - *design and safety considerations*
>
> *The basic conclusions resulting from the interactions with the NRC staff were:*
>
> - *The plant-specific acceptance criteria for dispositioning MRP-227-A LSC inspection results used commercially bounding acceptance criteria that are more limiting than what would be required to support safety considerations. Therefore, the results of these analyses should not be considered an accurate reflection of the level of redundancy available for the lower support structure to maintain its intended safety function.*
> - *LSCs that screen out for thermal embrittlement (TE) will potentially remain susceptible to irradiation embrittlement (IE). However, in this case, the embrittlement of the cast austenitic stainless steel (CASS) material would be no different than that of wrought material.*
> - *Fabrication of LSC included manufacturing controls and non-destructive examination (NDE) requirements that are in line with current industry*

standards for core support structure castings to limit the potential flaw sizes that could have potentially been placed into service.

- *The standard FMECA process [ANSI/IEEE Standard 352-1987] used in the development of MRP-191 component categorization identified TE and IE as potential degradation mechanisms for the LSCs. They identified these as FMECA Group 1, with a low likelihood of damage as a result of LSC failure.*
- *Predominate functional loadings on the LSC are compressive in nature. They are not conducive to propagating small manufacturing flaws or causing catastrophic failures of LSC.*
- *Potential failure modes as a result of cracking are not likely to result in the inability of the lower support structure to perform its function.*
- *There is no industry evidence of material degradation of the LSC bodies.*
- *The columns are in an unfavorable location for performing detailed inspections using currently available inspection techniques.*

*In addition to the proprietary information on lower support column design, fabrication, and functionality provided in the December meeting [ADAMS Accession No: ML14063A070], a plant-specific screening of the CASS LSC was conducted. The results of this screening evaluation, summarized in Tables 1 and 2, show that delta ferrite percentages are less than the criteria (**reference the applicable screening criteria**). Therefore, it is concluded that the **plant XXX** CASS LSCs are not subject to TE. NDE and acceptance criteria used during the fabrication of the **plant XXX** LSCs are summarized in Table 3. This data provides plant-specific confirmation that the quality level of the LSCs is consistent with the general discussions provided on this topic during the November and December meetings with the NRC staff. Based on this information, it is concluded that the MRP-227-A recommended inspections are appropriate to ensure functionality of the LSCs until the next scheduled inspections.*

Table 1 Lower Internals Assembly – Lower Support Column Heat and Ferrite Content

Supplier Heat Number	***Quantity of CASS LSC***	***Calculated Ferrite Content (Percent)***

Table 2 Lower Internals Assembly – Lower Support Column TE Screening

Molybdenum Content	***Casting Method***	***Delta Ferrite Level***	***Susceptibility to TE***
			Not Susceptible to TE

Table 3 Lower Internals Assembly – Lower Support Column Fabrication NDE Summary (Example)

NDE Method	*Acceptance Criteria*
Radiography (RT)	*Industry Standard XXX*
Liquid Penetrant (PT) Examination	*Industry Standard XXX or provide acceptance limits if not an industry standard*

If, during use of this screening assessment, a plant finds that the LSCs installed are in excess of the applicable screening limits, the NRC requires, through A/LAI 7, that a functionality analysis be performed to justify the use of MRP-227-A inspection sampling plan. In that case, an alternative response would be required that defines how a plant will comply. An example would be to refer to the ongoing work through the PWROG. The following is an example of wording that could be used in the response:

> *The results of the screening evaluation show that delta ferrite percentages are in excess of the screening criteria (**reference the applicable screening criteria**). As such, the Safety Evaluation to MRP-227-A alternatively recommends the use of a functionality analysis to demonstrate that the lower support columns will maintain their safety function through the period of extended operation. The analysis to provide this demonstration would need to show that there is sufficient redundancy within the lower support structure such that the failure of a number of columns would not impact the safe operation or safe shutdown of the plant. The original design specifications were established to ensure safe operation, long-term reliability, and to allow for the completion of maintenance activities. Currently there is no established methodology for assessing the potential safety impact of operating outside of the original specification. The exclusive use of the original design criteria is expected to result in a conservative conclusion with respect to determining the point at which loss of lower support columns results in the loss of safety function that misrepresents the structural redundancy of the assembly. As a result, work is ongoing through the Pressurized Water Reactor Owners Group to establish a common methodology and appropriate acceptance criteria to use when performing the functionality analysis. **Plant XXX** is an active participant in this industry program.*
>
> *Through this industry program a generic methodology and acceptance criteria for performing a functionality analysis for the lower support columns will be developed. Once established, the methodology and acceptance criteria will be applied to pilot plants to demonstrate generically, to the extent possible, that the lower support structure will maintain its intended safety function with a potentially reduced set of functional lower support columns.*
>
> *As a participant in the industry program, **plant XXX** will follow this methodology, once established, when performing any plant specific evaluations required to support the functionality analysis.*

Due to the time required to complete the activities associated with PWROG efforts, as well as any follow-on plant-specific evaluations that may be needed, it is recommended that an extension will need to be requested should the plant determine the need to complete this analysis.

CONCLUSION

The information discussed during the November/December meetings with the NRC and associated conclusions related to the information presented provide utilities with some key points which can be leveraged in responses to the NRC related to demonstrating functionality of the LSC. In addition to this information, the guidance provided in this document related to the pertinent plant specific material and fabrication data will assist utilities in developing an appropriate demonstration to the NRC that the CASS lower support columns at their plants are not susceptible to TE; thereby eliminating the need to perform a functionality analysis.

Plants that exceed the screening criteria for TE, discussed in this document, may require additional evaluations to fully demonstrate functionality of potentially embrittled LSCs. The work being conducted through the PWROG will provide the methodology and criteria used to perform functionality analysis for the LSCs.

REFERENCES

1. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863.
2. U. S. Nuclear Regulatory Commission Letter, "Summary of November 19, 2013, Public Meeting to Discuss on the Resolution of Plant-specific Action Items Related to Materials Research Program-227-A Reactor Internals Aging Management Programs/Inspection Plans," January 15, 2014 (ADAMS Accession No: ML13345A020).
3. U. S. Nuclear Regulatory Commission Letter, "Summary of the December 3, 2013, Closed Meeting on the Resolution of Applicant/Licensee Action Items Related to Materials Reliability Program-227-A Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," February 10, 2014 (ADAMS Accession No: ML13345A320).
4. Westinghouse Letter, LTR-NRC-14-7, Rev. 0, "Submittal of "Westinghouse Lower Support Columns Design and Safety Considerations" (Proprietary/Non-Proprietary)," January 30, 2014 (ADAMS Accession No: ML14063A070).
5. ANSI/IEEE Standard 352-1987, "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Safety Systems," 1987.
6. U.S. Nuclear Regulatory Commission, NUREG/CR-4513, Rev. 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," May 1994.
7. *BWRVIP-234: BWR Vessel and Internals Project, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steels for BWR Internals.* EPRI, Palo Alto, CA: 2009. 1019060.
8. U. S. Nuclear Regulatory Commission Letter, "NRC position on Aging Management of CASS Reactor Vessel Internal Components," June 11, 2014 (ADAMS Accession No: ML14163A112).
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.

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Our ref: NSD-EPRI-14-6

June 30, 2014

Transmittal of Guidelines for Responding to U.S. NRC Questions Related to MRP-227-A Applicant/Licensee Action Item 7 for Combustion Engineering and Westinghouse Pressurized Water Reactor Lower Support Column Designs

Dear Mr. Amberge:

This letter officially documents transmittal of NRC Follow-up Items Related to Generic Applicability of MRP-227 project. This document is being provided as a deliverable of the EPRI Project Agreement MA 10001231, corresponding to Westinghouse Sales Order 101303.

If you have any questions, please contact the Westinghouse Project Manager, Jun C. Bae, at (860) 731- 1778.

Regards,
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W. Anthony Nowinowski

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Attachment: LTR-RIAM-14-46

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