December 17, 2015

MEMORANDUM TO: Kevin P. Hsueh, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

FROM: John J. McHale, Chief ***/RA/***
Vessels and Internals Integrity Branch
Division of Engineering
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY ASSESSMENT OF REPORT PWROG-14048-P "FUNCTIONALITY ANALYSIS: LOWER SUPPORT COLUMNS"

The Pressurized Water Reactor Owner's Group (PWROG) developed report PWROG-14048-P, "Functionality Analysis: Lower Support Columns," which contains a generic methodology for evaluating the functionality of lower support columns (LSCs), and sent it to the U. S. Nuclear Regulatory Commission (NRC) for information only. PWROG-14048-P was provided to support the NRC in activities with regard to licensee submittals for reactor internals aging management and implementation of MRP-227-A. In particular, PWROG-14048-P was provided to support resolution of Action Item 7 of the staff's safety evaluation of MRP-227-A. Action Item 7 has to do, in part, with demonstrating functionality of LSCs that are made of cast austenitic stainless steel (CASS).

LSCs are support structures within pressurized water reactor vessels. Some LSCs are made of CASS, which is known to be susceptible to loss of fracture toughness due to thermal embrittlement and irradiation embrittlement. Loss of toughness of LSCs can potentially lead to the loss of their support function. The Vessels and Internals Integrity Branch staff performed an assessment of report PWROG-14048-P, which is provided in the enclosure. The Westinghouse Program Team has reviewed this assessment and by electronic correspondence dated November 11, 2015 (ADAMS Accession No. ML15335A399), has determined that there is nothing that would be considered Westinghouse Proprietary contained in the assessment.

Enclosure:
Summary Assessment

CONTACTS: David Dijamco, NRR/DE/EVIB
(301) 415-1502

Ganesh Cheruvenki, NRR/DE/EVIB
(301) 415-2501

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OFFICE	NRR/DE/EVIB	NRR/DE/EVIB	NRR/DE/EVIB: BC
NAME	DDijamco	GCheruvenki	JMcHale
DATE	12/14/2015	12/16/2015	12/17/2015

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SUMMARY ASSESSMENT OF PWROG-14048-P, REVISION 0

"FUNCTIONALITY ANALYSIS: LOWER SUPPORT COLUMNS"

1.0 BACKGROUND

Applicant/Licensee Action Item (A/LAI) 7 from the U. S. Nuclear Regulatory Commission (NRC) staff's final safety evaluation (SE) of Materials Reliability Program Report (MRP)-227-A, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (Reference 1) addresses the aging management program (AMP) for cast austenitic stainless steel (CASS) reactor vessel internal components in PWR units. A/LAI 7 states in part:

> "…applicants/licensees shall develop a plant-specific analysis to demonstrate that these components will maintain their functions during the period of extended operation. These analyses shall consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The plant-specific analyses shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation."

Thermal embrittlement (TE) and irradiation embrittlement (IE) are aging-related degradation mechanisms that are active in CASS materials and are concerns addressed in NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report" (Reference 2), in AMP XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel," and in AMP XI.M16A, "PWR Vessel Internals."

For some Westinghouse Electric Company (Westinghouse) PWR units, the lower support column (LSC) bodies are made of CASS materials. A/LAI 7 requires that the LSCs in Westinghouse units be evaluated for their continued functional capability during the period of extended operation (PEO). Therefore, the Pressurized Water Reactor Owner's Group (PWROG) developed report PWROG-14048-P, "Functionality Analysis: Lower Support Columns" (Reference 3) that addresses the generic methodology for evaluating the functionality of CASS LSC bodies, taking into account loss of fracture toughness due to TE and IE. The PWROG submitted this report to the NRC for information only. Section 2 of this summary assessment presents the NRC staff's assessment of the technical analyses in PWROG-14048-P for addressing functionality of the LSCs with consideration of TE and IE. This assessment is not an SE.

2.0 TECHNICAL ASSESSMENT

The technical assessment focuses on the three sections of PWROG-14048-P that are related to the functionality analysis of the LSCs: Section 4, "Lower Core Support Structure," Section 5, "Assessment of Failure Likelihood," and Section 6, "LSC Failure Tolerance Analysis."

The NRC staff notes that Section 3, "Limits of Applicability," of the PWROG-14048-P report states that further evaluations will likely be needed to reconcile plant-specific design details to ensure that conclusions of the report are applicable to the plant. Each Westinghouse PWR unit with CASS LSCs should perform plant-specific evaluations to assess continued operation through the PEO.

2.1 Section 4, "Lower Core Support Structure"

2.1.1 Summary

Section 4, "Lower Core Support Structure," of PWROG-14048-P provides an overall description of the lower support structure (LSS) assembly, whose three major components are the lower core plate (LCP), the LSCs, and the lower support plate. Specifically, the section discusses the function of the LSCs, the loading conditions applicable to the LSCs, and information about the fabrication of the LSCs that includes inspection acceptance criteria (i.e., size of allowable indications). The subsection on loading conditions applicable to the LSCs discusses all the possible primary and secondary loads during normal, upset, and accident conditions.

2.1.2 NRC Staff Assessment

Section 4 provides adequate description of the structures and components needed for evaluating the functionality of the LSCs through the PEO.

In Section 4.3, "Lower Support Column Fabrication," the PWROG stated that, during fabrication, all CASS LSCs in Westinghouse and Combustion Engineering designed units were inspected using radiography testing and dye penetrant testing, with the acceptance criteria for these inspections from American Society for Testing & Materials (ASTM) International Standards and American Society of Mechanical Engineers (ASME) code provisions. Because all of the CASS LSCs were initially inspected consistent with ASTM International and ASME code provisions, the NRC staff believes that casting defects would have been repaired before the LSCs were put in service. In the absence of any existing fabrication casting defect that could serve as an initiation site for cracking, it is unlikely that cracking would be initiated during service in these LSCs. Therefore, it can be expected that there would be minimum impact of TE and IE on the LSCs during operation due to lack of fabrication defects.

The other subsections in Section 4.3 provide adequate description of the LSS and LSCs that serve as input to the evaluations performed in Sections 5 and 6 of PWROG-14048-P.

2.2 Section 5 "Assessment of Failure Likelihood"

2.2.1 Summary

Section 5, "Assessment of Failure Likelihood," of PWROG-14048-P addresses the likelihood of full-section failure of the LSCs (i.e., complete loss of structural support function of the LSCs) that are embrittled due to thermal aging and irradiation. A failure modes and effects analysis (FMEA) was first conducted and a ranking established for each potential failure mode. Then evaluations were performed to support that the failure prevention measures that are in place make it unlikely that the failure mode will result in actual failures. These evaluations include neutron fluence estimates in the LSCs for a 60-year life, stress and subsequent flaw tolerance analyses, and discussions of potential mechanisms that can initiate (and extend) a crack.

In addition, operating experience associated with LSC failures, accessibility of LSCs, and potential detection methods if LSC failures occur are discussed.

2.2.2 NRC Staff Assessment

The FMEA focused on full-section failure of the LSCs as the main potential failure mode and considered the different effects that this failure mode could have on maintaining the safety function of the LSC, which is primarily as structural support for the fuel assemblies. The purpose of the evaluations in Section 5 was to assess the likelihood that the potential failure mode would lead to the different failure effects (i.e., actual failures).

One of the evaluations was determining the stress levels in the LSCs using finite element analysis (FEA). Standard techniques of FEA were used. Sufficient detail of the LSS geometry was included to reasonably simulate the stiffness of the LSS. Normal, upset, and accident loading conditions were considered. Accident conditions include, among other loads, safe shutdown earthquake concurrent with loss of coolant accident. The resulting highest stresses in the LSCs were local bending stresses that were below yield stress. Stress evaluations were performed for both 3-loop and 4-loop plant designs.

Crack initiation mechanisms, irradiation-assisted stress corrosion cracking (IASCC) and cyclic fatigue, were discussed in this section. The maximum neutron fluence estimate for a 60-year life of the LSCs and the stress levels determined in the stress evaluation were well below the conservative IASCC initiation criterion. In addition, it was stated that flow-induced, pump-induced vibration, and low-cycle fatigue usage for the LSCs were very low, but no further details are given. However, cyclic fatigue is not one of the degradation mechanisms that the LSCs screened-in for based on the MRP-191 criteria. Because neutron fluence and the stress levels for LSCs are well below the conservative IASCC criterion and cyclic fatigue usage is very low for LSCs, IASCC and cyclic fatigue may be ruled out as crack initiation mechanisms for LSCs. Therefore, it may be reasonably inferred that they are not crack extension mechanisms for LSCs. Hence, the largest potential indications in the LSCs are fabrication indications that are detectable by the pre-service inspections that were discussed in Section 4.3.

Critical flaw sizes are determined in Section 5 using standard fracture mechanics techniques. Once determined, these critical flaw sizes are compared to the largest fabrication indications that could have been missed by the pre-service inspections. The fracture mechanics evaluation used material property values of the LSCs in the 60-year embrittled condition (the actual fluence value used is slightly higher than the fluence value after 60 years). Several values of applied stresses were considered, but the highest value is much higher than the values determined in the stress analysis section. Furthermore, applied stresses are conservatively assumed as a membrane stress. The assumption of membrane stress, as compared to membrane-plus-bending stress, is conservative because membrane stress acts across the entire cross section of an LSC while membrane-plus-bending stress acts in a small, localized portion of the cross section. One of the fracture mechanics approaches used for determining critical flaw sizes is based on the saturated values of the J-resistance curve. In this approach, the J-resistance curve is the absolute lowest curve that is expected for the LSCs after embrittlement has achieved its maximum extent. The saturated J-resistance curve is then used to calculate the absolute largest flaw that could be tolerated by the LSCs. Even with the conservative assumptions described above, the resulting critical flaw sizes (projected as a length that can be seen on the surface of the LSC) are at least five times the largest fabrication indication length that can be seen on the surface of the LSC. In other words, the length of fabrication indications is much smaller than the critical flaw length.

In summary, Section 5 utilized conservative inputs, such as high membrane stresses and saturated values of material fracture toughness, to demonstrate that the likelihood of full-section failure of LSCs is low.

2.3 Section 6, "LSC Failure Tolerance Analysis"

2.3.1 Summary

Section 6, "LSC Failure Tolerance Analysis," of PWROG-14048-P discusses the evaluations performed to show redundancy in the LSS design, a redundancy that allows for retention of the LSC safety function (primarily as structural support for the fuel assemblies) if full-section failures of some LSCs do occur. The evaluations are for a representative PWR design and primarily serve as proof of concept and, consistent with Section 3, "Limits of Applicability" of the report, is not intended to generically bound all participating plants. An expert panel was formed to establish the acceptance criteria, which consist of structural limits and criteria that ensure that functionality of the LSS (and therefore the LSCs) is maintained to the point where safe shutdown of the reactor can be achieved. Changes to the modal characteristics of the LSS due to the postulated failed LSCs are addressed.

2.3.2 NRC Staff Assessment

Section 6 of PWROG-14048-P clearly states that the failure tolerance analysis is to serve primarily as proof of concept. Therefore, it should be regarded as such. Only a 3-loop plant design is considered in the failure tolerance analysis because this represents the median design among the Westinghouse fleet.

The acceptance criteria consist of structural limits and criteria that ensure that functionality of the LSS (and therefore the LSCs) is maintained to the point where safe shutdown of the reactor can be achieved. The structural criteria are based on ASME Boiler and Pressure Vessel Code allowable stresses, using the yield stress (S_y) as the basis for allowable stress (rather than design stress intensity, S_m) for normal and upset conditions. The use of S_y instead of S_m, although less restrictive, is reasonable in the context of postulated failed LSC scenarios since the objective is to realistically assess the LSC failures and not to qualify the LSCs. The other criteria are angular and vertical deflection limits that if met would ensure functionality of the LSCs is maintained to the point where safe shutdown of the reactor can be achieved. The values and methodology for these deflection limits are reasonable.

Once the acceptance criteria were established, several scenarios of full-section failures of LSCs were evaluated. The first set of evaluations presented simplified calculations (without the use of FEA) to get a feel for the amount of redundancy in the LSS. The approach used for these simplified analyses is reasonable. The more detailed evaluations used the FEA model described in Section 5. Three cases that represent three different numbers of failed LSC columns were analyzed. Similar to the stress analysis in Section 5, normal, upset, and accident conditions were considered. Then the maximum stresses and deflections were compared to the acceptance criteria described in the previous paragraph. The results of these comparisons are reasonable. The NRC staff notes that for the cases where full-section failures are assumed, the potential for buckling of the intact LSCs that remain (which are now taking up more compressive load) should be addressed to enable generic acceptance of the analysis presented in Section 6 of this report, or for a plant-specific evaluation.

To address the changes in the modal characteristics of the LSS due to failed LSCs, a modal sensitivity study was performed. The model is simplified using a lumped mass representation of the fuel assemblies. However, there is no mention that the mass of the water was included in the lumped mass. The results of the modal analysis show that failed LSCs would result in a change in the LCP dishing frequency and introduce new LCP modal frequencies. However, because the FEA model may be oversimplified, nothing more than a general assessment of the change in modal characteristics can be made. Therefore, a modal analysis using a more detailed FEA model is necessary if plant-specific results are desired, especially regarding the effects of new LCP modal frequencies as a result of failed LSCs.

In summary, Section 6 presented a reasonable approach for addressing structural redundancy in the LSS by assuming full-section failures of some LSCs. However, due to plant-specific differences, each plant must consider its specific design parameters when addressing structural redundancy of its LSS.

3.0 <u>CONCLUSIONS</u>

The NRC staff has reviewed PWROG-14048-P. The following summarizes the NRC staff's conclusions.

- The analyses for assessing the failure likelihood of the LSCs in Section 5 of the report utilized bounding inputs, such as high membrane stresses and saturated values of material fracture toughness, to demonstrate that the likelihood of full-section failure of LSCs is low.

- The failure tolerance evaluation of the LSCs to demonstrate structural redundancy in the LSS as discussed in Section 6 of the report presents a reasonable approach for addressing structural redundancy in the LSS. However, due to plant-specific differences, each plant must consider its specific design parameters when establishing the tilt and deflection criteria and the assumed spread or cluster of failed LSCs.

- Consideration of buckling needs to be included for generic acceptance of the redundancy analysis presented in Section 6. In addition, when evaluating scenarios where an assumed spread or cluster of LSCs has lost its support function, plant-specific evaluations that consider the potential for buckling and for changes in the modal characteristics of the LSS need to be included.

- To evaluate the functionality of LSCs for a specific plant, PWROG-14048-P may be used for guidance in the evaluation methodology. However, plant-specific parameters and conditions need to be used as input into the analytical evaluations.

4.0 REFERENCES

1. Electric Power Research Institute Report, “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. U. S. Nuclear Regulatory Document, NUREG-1801, Revision 2, “Generic Aging Lessons Learned (GALL) Report,” December 2010 (ADAMS Accession No. ML103490041).

3. Pressurized Water Reactor Owners Group Report No. PWROG-14048-P, “Functionality Analysis: Lower Support Columns,” Revision 0 (ADAMS Accession No. ML15077A114).