Attachment 2: Supplemental Clarification Material Requested by the NRC Staff in Support of the Safety Evaluation of MRP-227, Revision 1

The NRC staff requested further supplemental clarification material based on the industry presentation at the February 15, 2018 [1]. This request was transmitted via email and recorded in the NRC's public document repository [2]. Portions of the supplemental information requested have already been included in the responses provided in Attachment 1 of this letter. The remaining information requested falls into two categories:

- Expected industry response to the observation of degradation in the core barrel welds
- Additional detail on the comparison between boiling water reactor (BWR) and pressurized water reactor (PWR) experience and environmental conditions

The February 15, 2018 meeting included a discussion of the limitations on core barrel weld inspection coverage due to accessibility issues and the serious risks associated with reactor vessel internals disassembly. This discussion has been included here as relevant supplemental information.

Industry Response to Observed Degradation

Observation of relevant indications in the core barrel welds or in any MRP-227-A [3] or Revision 1 [4] inspection component would trigger a multi-part response from the affected licensee. Notification of the industry and the NRC would be included in this response along with evaluation of the extent of condition and evaluation and disposition of the relevant finding.

Section 7 of MRP-227-A [3] and Revision 1 [4] governs the response to relevant findings. Quotes from this section provided here are from MRP-227-A. Section 7.5 "Examination Results Requirement" dictates that examination results which do not meet the acceptance criteria of MRP-227, Section 5 "shall be recorded and entered in the plant corrective action program and dispositioned." Section 7.6 "Aging Management Program Results Requirement" requires that a summary report of inspection and evaluation experience be provided to the MRP Program Manager within 120 days of the outage completion. These inspection and evaluation reports are compiled biennially into a summary report [5]. Finally, Section 7.7 "Evaluation Requirement" requires that any engineering evaluation used to disposition relevant indications must be conducted in accordance with an NRC-approved methodology. WCAP-17096-NP-A is an example of such a methodology [6].

These requirements from MRP-227, Sections 7.5, 7.6, and 7.7 are all "Needed" elements under the NEI 03-08 protocol [7]. The protocol provides further governance of licensee response to emergent issues. In Appendix B of NEI 03-08, the following requirement is provided for emergent issues:

> *4. EMERGENT ISSUES*
>
> *Utilities shall inform the applicable IP* [issue program] *of significant emergent materials-related issues occurring at their plants when they have potential generic implications. In order to support this communication, each IP shall be prepared to perform a timely evaluation of the significance of emergent materials issues that fall within the scope of its program. The IP evaluation should be performed within a timeframe that supports the utility's needs where possible. Items that should be considered in the IP's evaluation include:*
>
> - *Safety significance*
> - *Demonstration of a new degradation type*
> - *Effect on the basis of industry guidance*
> - *Effect on the existing knowledge base*
> - *Expected regulatory significance*

Thus, observation of cracking degradation in a PWR core barrel weld would lead to notification of the appropriate industry issue program and transmittal of the inspection results to the EPRI program manager who would then transmit them to the NRC through the periodically updated summary report. The operating experience would also be entered into the licensee's corrective action program and require evaluation of the extent of condition and disposition through engineering evaluation.

Comparison of BWR and PWR Operating Experience and Environmental Conditions

The core shroud in the BWR design and the core barrel in the PWR design are very similar, in that they are large-diameter, thick-walled cylinders which are largely used to guide the coolant flow through the internals. They are even constructed of essentially the same materials: austenitic stainless steel plate rolled and welded into a cylinder. Although the function, fabrication, and materials of construction are similar for the two configurations, service experience at PWRs has been very different from that at BWRs.

A high-level discussion of the differences between the occurrence of SCC and IASCC in the two reactor designs was presented during the February 15, 2018 meeting between the NRC staff and industry representatives [1]. More detail supporting these discussions is provided in letter LTR-AMLR-18-18 [8], titled, "Comparison of Boiling Water Reactor and Pressurized Water Reactor Chemistry and Operating Experience with Austenitic Stainless Steel." The goal of this white paper is to clearly identify the differences between the two designs, and to justify differences in the treatment of PWR core barrels as compared to BWR core shrouds.

The white paper provides technical arguments to identify the theoretical basis for the significant differences in cracking susceptibility, and it provides a summary review of the service experience with each reactor type to demonstrate the actual differences observed. These two approaches provide a complementary argument to support different treatments of these BWR and PWR components. The mechanical design criteria and temperatures of operation for the BWR core shroud and the PWR core barrel are similar, but there are major differences in the reactor coolant water chemistry to which they are exposed.

The primary difference is reflected in the electrochemical potential (ECP), which measures whether or not the environment is more oxidizing or more reducing. The PWR primary water environment is the most reducing case at approximately -770 mV_{SHE}. This is at least several hundred mV lower than BWR hydrogen water chemistry (HWC) environments and nearly 1000 mV lower than the ECP in a BWR normal water chemistry (NWC) regime. Operation below approximately -230 mV_{SHE} results in mitigation of SCC and IASCC. The beneficial effects of a reducing environment have been explained in Section 2 of [8]. The service experience supports these conclusions as well and has been reviewed in Sections 3 and 4 of [8].

BWR core shrouds, as well as other internals components fabricated from austenitic stainless steels, have experienced significant levels of SCC, while little SCC of austenitic stainless steels has been observed to date in PWR environments. Multiple detailed inspections under the requirements of MRP-227 have been completed on PWR core barrels, and only one recent inspection has discovered relevant indications. Since PWRs will operate for their entire licensed lifetimes with low ECP in the primary coolant system due to hydrogen additions, it is considered extremely unlikely that SCC or IASCC will occur in a PWR core barrel.

Limitations on Core Barrel Weld Accessibility and Reactor Vessel Internals Disassembly Risks

The accessibility of core barrel welds depends on the weld location and the design of the plant. Welds in the upper core barrel in both Combustion Engineering and Westinghouse-designed plants should have close to 100% accessibility; though some obstruction could occur due to gussets or other attachments or due to the proximity of the containment cavity to the core barrel while it is in the stand. Welds that are ground flush may also present inspectors with difficulties in reliably finding the weld location.

Welds located in the lower core barrel have significantly less accessibility. Inside of the core barrel, either the baffle-former assembly (Westinghouse) or core shroud assembly (Combustion Engineering) cover 100% of the inner diameter of the barrel in the beltline region. Below the beltline region, 100% of the inner diameter of the barrel is covered by the other lower internals.

For the outer diameter of the core barrel, the welds in the beltline region have several different levels of accessibility. Combustion Engineering-designed plants with no thermal shield approach 100% accessibility. Westinghouse-designed plants with thermal shields must access the welds through the gap between the core barrel and the thermal shield and will have less than 100% accessibility (the exact coverage will depend on the plant and the specific inspection tooling). Finally, Westinghouse-designed plants with neutron panels will have access to approximately 50-60% of the circumference. Axial welds that happen to be behind neutron panels could have significantly lower accessibility. Thermal shields and neutron panels do not cover the outer diameter of the core barrel below the beltline region, but the presence of attachments like radial keys and core snubber lugs could reduce the achievable coverage slightly below 100%.

One of the basic assumptions in MRP-227 development was that component disassembly should be avoided unless absolutely warranted. This was addressed under the response to request for additional information 4-8 for MRP-227-A [3]. Disassembly carries serious risks, including:

- Personnel safety and radiation exposure during disassembly, inspection, and re-assembly
- Operations to disassemble and reassemble would have to be performed remotely, which increases difficulty significantly and may lead to irreparable damage at any point in the operations
- Cutting, shaping, or removal operations can lead to loose parts and debris, which can impact fuel integrity
- Components with elevated irradiation dose cannot be welded, so extensive modification to the base internals component design may be required

The accessibility limitations and significant risks associated with component disassembly were considered by the industry when developing the core barrel weld inspection coverage requirements.

References:

1. Materials Reliability Program Presentation, "MRP-227, Revision 1 Requests for Additional Information," NRC Public Meeting, Rockville, MD, February 15, 2018. (ML18043B155)
2. Email from Joseph Holonich (NRC) to Kyle Amberge (EPRI), Subject: "Information We Want on the Docket from MRP-227, Rev. 1 RAI Discussion," February 22, 2018. (ML18053A058)
3. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.
4. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1)*. EPRI, Palo Alta, CA: 2015. 3002005349.
5. *Materials Reliability Program: Inspection Data Survey Report* (MRP-219, Revision 11). EPRI, Palo Alto, CA: 2015. 3002005509.
6. Westinghouse Report, WCAP-17096-NP-A, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 2016.
7. Guidelines for the Management of Materials Issues, NEI 03-08, Nuclear Energy Institute, Washington, DC, Latest Edition.
8. Westinghouse Letter, LTR-AMLR-18-18, Revision 2, "Comparison of Boiling Water Reactor and Pressurized Water Reactor Experience with Cracking of Austenitic Stainless Steel," April 24, 2018.