Thomas D. Ray
Site Vice President
McGuire Nuclear Station

Duke Energy
12700 Hagers Ferry Rd
Huntersville, NC 28078

980.875.4805
Tom.Ray@duke-energy.com

Serial: RA-19-0005 10 CFR 50.90
January 30, 2019

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

MCGUIRE NUCLEAR STATION, UNIT NOS. 1 AND 2
DOCKET NOS. 50-369 AND 50-370
RENEWED LICENSE NOS. NPF-9 AND NPF-17

SUBJECT: RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING REVIEW REQUEST OF THE AGING MANAGEMENT PROGRAM AND INSPECTION PLAN FOR THE REACTOR VESSEL INTERNALS TO IMPLEMENT MRP-227-A

REFERENCES:

1. Duke Energy letter, *Review Request for the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A,* dated December 13, 2017 (ADAMS Accession No. ML17356A184).

2. Duke Energy letter, *Resolution of Commitments related to Review Request for the Aging Management Program and Inspection Plan*, dated May 9, 2018 (ADAMS Accession No. ML18135A087).

3. NRC E-Mail, *Request for Additional Information – McGuire Nuclear Station, Units 1 and 2 – MRP-227 Review (EPID L-2017-LLA-0414)*, dated December 18, 2018 (ADAMS Accession No. ML18352A805).

Ladies and Gentlemen:

In Reference 1, as supplemented by Reference 2, Duke Energy Carolinas, LLC (Duke Energy) submitted a Review Request for the Aging Management Program (AMP) and Inspection Plan for the McGuire Nuclear Station (MNS), Units 1 and 2 Reactor Vessel Internals (RVIs).

By correspondence dated December 18, 2018 (Reference 3), the Nuclear Regulatory Commission (NRC) staff requested additional information from Duke Energy that is needed to complete the review.

Attachment 1 is a letter from Westinghouse that provides the responses to Reference 3, RAI 1 and RAI 3. Attachment 3 is a letter from Framatome to Duke Energy that provides the response to Reference 3, RAI 2. Attachment 3 contains information that is proprietary to Framatome. In accordance with 10 CFR 2.390, Duke Energy requests that Attachment 3 be withheld from public disclosure. An Affidavit is included (Attachment 2) attesting to the proprietary nature of Attachment 3. A non-proprietary version of Attachment 3 is included in Attachment 4.

This submittal contains no regulatory commitments.

Should you have any questions concerning this letter, or require additional information, please contact Art Zaremba, Manager – Nuclear Fleet Licensing, at 980-373-2062.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 1/25/2019.

Sincerely,

Thomas D. Ray

Thomas D. Ray
Site Vice President, McGuire Nuclear Station

NDE

Attachments:

1. Response to RAIs 1 and 3: Westinghouse letter, "*Responses Supporting NRC Request for Additional Information on the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals*," dated January 21, 2019.
2. Framatome Affidavit
3. Response to RAI 2: Framatome letter, "*Response to Requests for Additional Information for Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A,*" dated January 2019 (Proprietary).
4. Response to RAI 2: Framatome letter, "*Response to Requests for Additional Information for Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A,*" dated January 2019 (Redacted).

cc: (all with Attachments unless otherwise noted)

C. Haney, Regional Administrator USNRC Region II
G.A. Hutto, USNRC Senior Resident Inspector
M. Mahoney, NRR Project Manager
W.L. Cox, III, Section Chief, NC DHSR

Attachment 1

Response to RAIs 1 and 3: Westinghouse letter, "*Responses Supporting NRC Request for Additional Information on the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals*," dated January 21, 2019.

Westinghouse

To: David Kovacic

cc:

Date: January 21, 2019

From: Aging Management and License Renewal
Ext: (412) 374-3957
Fax: (724) 940-8559

Our ref: LTR-AMLR-19-1, Rev. 0

Subject: **Responses Supporting NRC Request for Additional Information on the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals**

References:

1. Westinghouse Contract Impact Notification, CIN-ES-17-0108-001, "McGuire Units 1 & 2 Aging Management Plan RAI Response Support," January 4, 2019.
2. Email Correspondence from Michael Mahoney to Art Zaremba, "Request for Additional Information - McGuire Nuclear Station, Units 1 and 2 – MRP-227 Review (EPID L-2017-LLA-0414)," December 18, 2018. (U. S. NRC ADAMS Accession No. ML18352A805)
3. Westinghouse Report, WCAP-18265-NP, Rev. 0, "Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals (Application to Implement MRP-227-A)," December 2017.

Westinghouse has been contracted [1] to provide support for a U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) set on the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals [2] and [3]. Per [1], Westinghouse is required to respond to RAI 1 and RAI 3 of the RAI set. This letter contains the RAI responses to RAI 1 and RAI 3.

If you have additional questions or require further information, please contact the undersigned.

Authored by: ELECTRONICALLY APPROVED*
Taylor N. Terek
Aging Management and License Renewal

Reviewed by: ELECTRONICALLY APPROVED*
Joshua K. McKinley
Aging Management and License Renewal

Verified by: ELECTRONICALLY APPROVED*
Karli N. Szweda
Aging Management and License Renewal

Approved by: ELECTRONICALLY APPROVED*
Melanie R. Fici, Manager
Aging Management and License Renewal

Electronically approved records are authenticated in the electronic document management system.

Responses to NRC Requests for Additional Information (RAIs) on the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals

1.0 Introduction and Background

Duke Energy submitted the Reactor Vessel Internals (RVI) Aging Management Program (AMP) and Inspection Plan for McGuire Unit 1 and Unit 2 [1] to the NRC on December 13, 2017 [2], as supplemented by [3]. The NRC reviewed this submittal and requested additional information from McGuire within [4].

The responses to RAI 1 and RAI 3 are within the following sections.

2.0 NRC RAI 1

RAI-1

The licensee stated, in part, in Section 6.2.2 in Attachment 1 (ADAMS Accession No. ML17356A178) of its December 13, 2017 letter, that a detailed tabulation of the McGuire RVI components was completed and compared to typical Westinghouse PWR RVI components in Table 4-4 of MRP-191, "Screening, Categorization and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design" (ADAMS Accession No. ML091910130). This effort identified one McGuire RVI component (access plug assembly spring) that has no corresponding MRP-191 component.

Further assessment of Attachment 1 in the May 9, 2018 supplement (ADAMS Package No. ML18135A087), indicated that the McGuire access plug assembly spring is made of Inconel X-750, and no degradation mechanisms were identified through the process of MRP-191. Attachment 1 of the May 9, 2018 supplement also indicated that the McGuire anti-vibration sleeves are made of 304 stainless steel (SS), and although several degradation mechanisms have been identified, the [McGuire-specific expert] panel considered the likelihood of failure and damage to be low.

Please address the following:

- In its consideration of the acceptability of MRP-191, as a basis for MRP-227, the NRC staff performed three tasks: 1) assessed the nature and qualifications of the expert panel used in developing MRP-191; 2) assessed the process used by the expert panel in screening, categorization, and ranking the RVI components, and 3) developed an independent assessment of the MRP-191 disposition for verification. The NRC staff intends to review those items applicable to McGuire that were not included in MRP-191 in the same manner as it reviewed MRP-191. To permit the NRC staff to perform its analysis, please compare the makeup and processes used by the McGuire expert panel to those used by the MRP-191 expert panel.

- The above conclusion on McGuire anti-vibration sleeves was made in Attachment 1 of the May 9, 2018 supplement, without justification. Please provide justification regarding this conclusion (e.g., negligible stresses, absence of hostile environment, or negligible neutron fluence).

Response:

The McGuire Expert Panel applied the current Failure Modes, Effects, and Criticality Analysis (FMECA) approach for Westinghouse designed plants as described within Section 6.2 of MRP-191 Revision 0 [5]. By applying Section 6.2 of [5], the McGuire Expert Panel used the same process, definitions, and inputs as the MRP-191 Revision 0 Expert Panel. The McGuire Expert Panel was composed of experts filling the following FMECA Roles:

- Component design, testing, and repair
- Structural modeling and analysis
- Thermal-hydraulics and systems analysis
- Neutron fluence and radiation analysis
- Materials degradation and failure experience
- Component inspection experience
- Risk assessment
- Inspection requirements
- System function and operating experience
- Licensing and regulatory interaction

The McGuire Expert Panel required consistent definitions for some of the key terms and concepts applied throughout assigning likelihood and consequence levels to the RVI components. Therefore, the following definitions and categories from MRP-191 Revision 0 [5] were used throughout the FMECA:

Component Failure: Material degradation of a given component by one or more credible degradation mechanisms, as identified in the Screening Evaluation, causes the component to lose its ability to perform its intended design function either during normal operation or under accident conditions. Accident conditions include design basis earthquake or pipe break with no credit for the low likelihood of these accidents actually occurring. Cosmetic wear, craze cracking, and plastic deformation (excluding springs) were not considered failures.

Failure Likelihood: The likelihood that component failure(s) will occur during 60 years of operation. The four categories of failure likelihood are defined in Table 6-2 of [5].

Core Damage: Physical damage to one or more fuel assemblies or other internals components - either through direct impact with the fuel, flow-jetting, loss of core support / fuel spring hold-down force, loose parts, blockage / diversion of coolant flow, or loss of insertion ability for more than one control rod that would impair the ability to safely shut down the reactor.

Damage Likelihood: The conditional likelihood that component failure(s) results in core damage given that the failure occurs irrespective of the actual failure likelihood. The four categories of conditional damage likelihood are defined in Table 6-3 of [5].

Based on failure likelihood and damage likelihood, the access plug assembly spring and flux thimble anti-vibration sleeves were assigned a FMECA Group, as seen in Table 1, which is Table 6-4 in [5].

Table 1: Reactor Internals FMECA (Significance) Groups Used by the McGuire Expert Panel

Failure Likelihood	Consequence (Damage Likelihood)		
	Low	**Medium**	**High**
High	2	3	3
Medium	1	2	3
Low	1	1	2
None	0	0	0

The McGuire Expert Panel discussed the component design, key parameters, and function of the access plug assembly spring and flux thimble anti-vibration sleeves, in addition to material information of each component. The McGuire Expert Panel then reviewed the geometry, location, and function of each component, identified screened-in degradation mechanisms for each component, and discussed relevant information from previous Westinghouse Pressurized Water Reactor Owners Group (PWROG) sponsored FMECAs and the MRP Issue Management Table [8]. A likelihood of failure was determined for each component based on the component review in addition to a review of degradation and failure experience. A likelihood of damage was determined for each component based on a review of the effects and consequences of degradation and failure.

The access plug assembly spring was determined to have no screened-in degradation mechanisms since the component experiences negligible stresses. Therefore, the McGuire Expert Panel did not evaluate the access plug assembly spring for likelihood of failure or likelihood of damage and the access plug assembly spring was assigned a FMECA ranking of 0 and a Category of A.

The flux thimble anti-vibration sleeves were determined to screen in for irradiation assisted stress corrosion cracking (IASCC), wear, fatigue, irradiation embrittlement (IE), void swelling (VS), and irradiation-induced stress relaxation/irradiation creep (ISR/IC). The McGuire Expert Panel first considered failure of the anti-vibration sleeves and then consequences of failure. If the anti-vibration sleeves were to fail, wear would increase on the flux thimble tubes. This, however, does not compromise safety as McGuire periodically inspects the flux thimble tubes by eddy current examination and monitors wall thinning of the flux thimble tubes in order to predict when tubes should be repaired or replaced [1]. Leakage due to failure would be detectable at the flux thimble tube seal table. The McGuire Expert Panel considered loose parts as a concern; however the anti-vibration sleeves are not expected to become a loose part based on design.

The McGuire Expert Panel assigned a likelihood of failure ranking of “Low” for the flux thimble anti-vibration sleeves based on a consideration of the screened in degradation mechanisms. ISR/IC would

reduce the anti-vibration sleeve's preload, which may allow for more vibration. However, the anti-vibration sleeve would still be locked in place during operation and would still contribute to vibration reduction. SCC would also reduce preload, but the anti-vibration sleeve is not expected to become a loose part based on design, and would still be present during operation to reduce vibration. The effects of IE and VS are considered to be low due to the anti-vibration sleeve's proximity to the core. IE and VS are not expected to prevent the anti-vibration sleeve's function. The McGuire Expert Panel assigned a likelihood of damage ranking of "Low" to the flux thimble anti-vibration sleeves, since loose parts are not a concern and loss of vibration reduction would not cause core damage. Therefore, the flux thimble anti-vibration sleeves were assigned to FMECA Group 1 and Category A based on the likelihood of failure ranking of "Low" and the likelihood of damage ranking of "Low".

3.0 NRC RAI 3

RAI-3

Regarding Applicant/Licensee Action Item 7 in the NRC safety evaluation of MRP-227-A, dated June 22, 2011 (ADAMS Accession No. ML111600498), there is new NRC staff guidance on the threshold limits for thermal embrittlement (TE) and IE of Cast Austenitic Stainless Steel (CASS). The bases for the NRC staff's new consensus on the threshold limits are described in "NRC Position on Aging Management of CASS Reactor Vessel Internal Components" (ADAMS Accession No. ML14163A112), as amended in the NRC staff's SE of the Boiling Water Reactor Vessel Internals Project (BWRVIP)-234, "Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals," June 22, 2016 (ADAMS Accession No. ML16096A002).

a. Please address any difference between the new guidance and the evaluation performed for McGuire, Units 1 and 2. In particular, please address, the new screening guidelines of CASS materials for loss of fracture toughness of highly irradiated components (i.e., components susceptible to IE), in addition to TE. If any changes to the evaluation are necessary, please submit the re-evaluation, if not, please explain why not. This evaluation could affect some of the CASS components that are listed as susceptible to TE in Tables 6-2 and 6-3 of Attachment 1 of the December 13, 2017, letter, for McGuire, Units 1 and 2, respectively.

b. The NRC staff notes that MRP-191, Rev. 1 has not been reviewed by the NRC. Therefore, placing the following CF8 components in Category A in accordance with MRP-191, Rev. 1, is not considered by the NRC staff as having sufficient support:

    - CF8 upper guide tube enclosures
    - CF8 intermediate flanges
    - CF8 Brackets, clamps, terminal blocks, and conduit straps

   Please demonstrate that the screening and the failure modes, effects and criticality analyses and ranking considerations in MRP-191, Rev. 1 are equivalent or better than MRP-191, Rev. 0, to support the Category A determination for these three RVI components.

c. Section 6.2.7 states, "The lower support column bodies are not CASS material for either McGuire Unit 1 or Unit 2." WCAP-17397-NP, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Salem Nuclear Generating Station, Salem 1," states that, "The lower internals assembly column cap is a CASS piece welded onto the top of the core support column shaft. These two pieces together constitute the lower internals assembly - column body." For Salem 1, only the lower support column caps were identified as CASS. To ensure that McGuire units do not have the same lower support column bodies as Salem 1, please confirm that the lower support column caps are not CASS material. If this cannot be confirmed, please provide an explanation of how aging degradation due to TE and IE of the lower support column caps is being managed and will be managed during the period of extended operation because the NRC staffs initial review indicated that, in addition to TE, the lower support column caps are also susceptible to IE.

Response:

Part a:

The McGuire Unit 1 and Unit 2 CASS evaluation within the McGuire Units 1 and 2 Aging Management Program Plan (AMP) documented in WCAP-18265 [1] relies on the scoping and screening evaluations conducted for MRP-191 Revision 0 [5] and Revision 1 [9]. MRP-191 Revision 0 and Revision 1 reference the screening criteria of MRP-175 Revision 0 [6] for irradiation embrittlement (IE) and thermal embrittlement (TE) of CASS material components. MRP-175 Revision 0 includes the degradation mechanism screening for IE and TE. The new guidance on screening criteria for IE and TE provided within the safety evaluation (SE) of BWRVIP-234 [7] is almost the same as those provided in MRP-175 Revision 0. The only relevant difference is the increase of the TE threshold for centrifugally cast materials with low molybdenum (Mo) content (<0.5 wt%) from 20% ferrite content (MRP-175, Revision 0 [6]) to 25% ferrite content (SE on BWRVIP-234 [7]). Since the new guidance is equivalent to or slightly relaxed from the guidance used in the McGuire Units 1 and 2 CASS evaluations, a re-evaluation of the McGuire CASS components is not necessary.

Part b:

The MRP-191 Revision 1 Expert Panel applied the current FMECA approach for Westinghouse designed plants as described within Section 6.2 of MRP-191 Revision 0 [5]. By applying Section 6.2 of [5], the MRP-191 Revision 1 Expert Panel used the same process, definitions, and inputs as the MRP-191 Revision 0 Expert Panel. The MRP-191 Revision 1 Expert Panel was composed of experts filling the following FMECA Roles:

- Component design, testing, and repair
- Structural modeling and analysis
- Thermal-hydraulics and systems analysis
- Neutron fluence and radiation analysis
- Materials degradation and failure experience
- Component inspection experience
- Risk assessment
- Inspection requirements

- System function and operating experience
- Licensing and regulatory interaction

The MRP-191 Revision 1 Expert Panel required consistent definitions for some of the key terms and concepts applied throughout assigning likelihood and consequence levels to the RVI components. Therefore, the following definitions and categories from MRP-191 Revision 0 [5] were used throughout the FMECA:

Component Failure: Material degradation of a given component by one or more credible degradation mechanisms, as identified in the Screening Evaluation, causes the component to lose its ability to perform its intended design function either during normal operation or under accident conditions. Accident conditions include design basis earthquake or pipe break with no credit for the low likelihood of these accidents actually occurring. Cosmetic wear, craze cracking, and plastic deformation (excluding springs) were not considered failures.

Failure Likelihood: The likelihood that component failure(s) will occur during 60 years of operation. The four categories of failure likelihood are defined in Table 6-2 of [5].

Core Damage: Physical damage to one or more fuel assemblies or other internals components - either through direct impact with the fuel, flow-jetting, loss of core support / fuel spring hold-down force, loose parts, blockage / diversion of coolant flow, or loss of insertion ability for more than one control rod that would impair the ability to safely shut down the reactor.

Damage Likelihood: The conditional likelihood that component failure(s) results in core damage given that the failure occurs irrespective of the actual failure likelihood. The four categories of conditional damage likelihood are defined in Table 6-3 of [5].

Based on failure likelihood and damage likelihood, the CF8 upper guide tube enclosures and CF8 brackets, clamps, terminal blocks, and conduit straps were assigned a FMECA Group, as seen in Table 2, which is Table 6-4 in [5]. The CF8 intermediate flanges were evaluated within MRP-191 Revision 0 [5] and therefore, the likelihoods and FMECA Group were applied as identified in [5].

Table 2: Reactor Internals FMECA (Significance) Groups Used by the MRP-191, Revision 0 and MRP-191, Revision 1 Expert Panels

Failure Likelihood	Consequence (Damage Likelihood)		
	Low	**Medium**	**High**
High	2	3	3
Medium	1	2	3
Low	1	1	2
None	0	0	0

The MRP-191 Revision 1 Expert Panel discussed the component design, key parameters, and function of the upper guide tube enclosures and the brackets, clamps, terminal blocks, and conduit straps fabricated from CF8 material. The MRP-191 Revision 1 Expert Panel then reviewed the geometry, location, and function of each component, identified screened-in degradation mechanisms for each component, and discussed relevant information from previous Westinghouse PWROG sponsored FMECAs and the MRP Issue Management Table [8]. A likelihood of failure was determined for each component based on the component review in addition to a review of degradation and failure experience. A likelihood of damage was determined for each component based on a review of the effects and consequences of degradation and failure.

The CF8 upper guide tube enclosures were determined to screen in for stress corrosion cracking (SCC) of the weld, like their 304 stainless steel counterpart in MRP-191 Revision 0 [5], and additionally screen in for TE and IE due to the CF8 material[1]. The MRP-191 Revision 1 Expert Panel determined that the addition of potential embrittlement for the enclosures fabricated from CF8 will have no effect on the failure and damage likelihoods given to the 304 stainless steel upper guide tube enclosures within MRP-191 Revision 0 [5]. Therefore, the MRP-191 Revision 1 Expert Panel assigned a likelihood of failure ranking of “Low” and a likelihood of damage ranking of “Medium” to the CF8 upper guide tube enclosures. Based on these FMECA rankings, the CF8 upper guide tube enclosures were assigned to FMECA Group 1.

The CF8 intermediate flanges were evaluated within MRP-191 Revision 0 [5]. This component screened in for SCC of the weld, fatigue, and TE. The CF8 intermediate flanges were assigned a likelihood of

[1] The upper guide tube enclosures were conservatively screened in for IE within MRP-191 Revision 1 [5]. However, this has separately been determined to be overly conservative due to the component’s distance from the core [10]. Despite this, the more conservative assumption was applied for the CASS evaluation supporting the McGuire AMP.

failure ranking of “Low”, a likelihood of damage ranking of “Medium”, and were assigned to FMECA Group 1 [5].

The CF8 brackets, clamps, terminal blocks, and conduit straps – conduit positioners were determined to screen in for TE due to the CF8 material. The conduit positioners did not screen in for IE due to proximity from the core. The MRP-191 Revision 1 Expert Panel assigned a likelihood of failure ranking of “Low” to the CF8 conduit positioners, because TE alone would not cause a loss of thermocouple function, since a cracking mechanism would be required. The MRP-191 Revision 1 Expert Panel assigned a likelihood of damage ranking of “Low” to the CF8 conduit positioners since a thermocouple failure would be detected as part of normal plant operation. Based on these FMECA rankings, the CF8 brackets, clamps, terminal blocks, and conduit straps were assigned to FMECA Group 1.

Part c:

McGuire Unit 1 and Unit 2 do not have lower support column caps. The lower support columns at McGuire Unit 1 and Unit 2 are fabricated from one piece of Type 304 stainless steel. The lower support column design at Salem is different from the design at McGuire Unit 1 and Unit 2; therefore, the component, lower support column cap, does not apply to either McGuire unit.

References

1. Westinghouse Report, WCAP-18265-NP, Rev. 0, "Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals (Application to Implement MRP-227-A)," December 2017.
2. Duke Energy Letter, MNS-17-050, "Duke Energy Carolinas, LLC (Duke Energy) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 Review Request for the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A," December 13, 2017. (U. S. NRC ADAMS Accession No. ML17356A177)
3. Duke Energy Letter, MNS-18-029, "Duke Energy Carolinas, LLC (Duke Energy) McGuire Nuclear Station, Units 1 and 2 Docket Nos. 50-369 and 50-370 Resolution of Commitments related to Review Request for the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A," May 9, 2018. (U. S. NRC ADAMS Accession No. ML18135A071)
4. Email Correspondence from Michael Mahoney to Art Zaremba, "Request for Additional Information - McGuire Nuclear Station, Units 1 and 2 – MRP-227 Review (EPID L-2017-LLA-0414)," December 18, 2018. (U. S. NRC ADAMS Accession No. ML18352A805)
5. *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.
6. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)*. EPRI, Palo Alto, CA: 2005. 1012081.
7. U.S. Nuclear Regulatory Commission Document, "Final Safety Evaluation of the BWRVIP-234: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals (TAC No. ME5060)," June 22, 2016. (U. S. NRC ADAMS Accession No. ML16096A002)
8. *Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156)*. EPRI, Palo Alto, CA: 2005. 1012110.
9. *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191, Revision 1)*. EPRI, Palo Alto, CA: 2016. 3002007960.
10. Westinghouse Letter, LTR-RIAM-16-17, Rev. 1, "Summary Letter for MRP-191, Revision 0 Update to MRP-191, Revision 1," August 30, 2016. (Westinghouse Proprietary)
11. Duke Energy License Renewal Application, "Application to Renew the Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2," June 2001.

This page was added to the quality record by the PRIME system upon its validation and shall not be considered in the page numbering of this document.

Approval Information
Author Approval Terek Taylor Jan-21-2019 10:23:40
Verifier Approval Szweda Karli N Jan-21-2019 10:27:40
Reviewer Approval Mckinley Joshua K Jan-21-2019 11:13:32
Manager Approval Fici Melanie R Jan-21-2019 14:04:55
Hold to Release Approval Terek Taylor Jan-22-2019 08:25:30

Files approved on Jan-22-2019

*** This record was final approved on 1/22/2019 8:25:30 AM. (This statement was added by the PRIME system upon its validation)

Attachment 2

Framatome Affidavit

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Philip A. Opsal. I am Manager, Product Licensing, for Framatome Inc., (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome Inc., to determine whether certain Framatome Inc. information is proprietary. I am familiar with the policies established by Framatome Inc. to ensure the proper application of these criteria.

3. I am familiar with the Framatome Inc. information contained in the following document: Framatome (formally AREVA Inc.) Document 103-3750P-000, "Response to Requests for Additional Information for Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A Technical Report; ANP-3750P Revision 0", referred to herein as "Document." Information contained in this Document has been classified by Framatome Inc. as proprietary in accordance with the policies established by Framatome Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome Inc. and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome Inc. to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome Inc.'s research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome Inc.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome Inc. in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome Inc., would be helpful to competitors to Framatome Inc., and would likely cause substantial harm to the competitive position of Framatome Inc.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c), 6(d) and 6(e) above.

7. In accordance with Framatome Inc.'s policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome Inc. only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome Inc.'s policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Philip A. Opsal

SUBSCRIBED before me this 22nd

day of January, 201~~8~~9. HHE

Heidi H Elder

Heidi Hamilton Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/2022
Reg. # 7777873

HEIDI HAMILTON ELDER
COMMONWEALTH
REGISTRATION NO.
7777873
MY COMM. EXPIRES
12/31/2022
OF VIRGINIA
NOTARY PUBLIC

Attachment 3

Response to RAI 2: Framatome letter, "*Response to Requests for Additional Information for Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A*," dated January 2019 (Proprietary).

Note: Text that is within bolded brackets is proprietary to Framatome.

Attachment 4

Response to RAI 2: Framatome letter, "*Response to Requests for Additional Information for Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A*," dated January 2019 (Redacted).

Note: Text that is within bolded brackets is proprietary to Framatome and has been removed.

framatome

Response to Requests for Additional Information for Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A

ANP-3750NP
Revision 0

Technical Report

January 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Figures

Nomenclature

Acronym	Definition
AMP	Aging Management Program
BOL	Beginning of Life
CW	Cold-worked
dpa	Displacements Per Atom
EOL	End of Life
IC	Irradiation-Enhanced Creep
IE	Irradiation Embrittlement
ISR	Irradiation-Enhanced Stress Relaxation
MRP	Materials Reliability Program
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
SS	Stainless Steel
US	United States

ABSTRACT

By submittals dated December 13, 2017 and May 9, 2018, Duke Energy submitted for U. S. Nuclear Regulatory Commission (US NRC) staff review the Aging Management Program (AMP) and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals (Application to Implement MRP-227-A). The US NRC has issued Requests for Additional Information (RAIs) on this submittal. This report provides the Framatome Inc. response to RAI 2.

1.0 INTRODUCTION AND SUMMARY

By submittals dated December 13, 2017 (References 1, 2) and May 9, 2018 (References 3, 4), Duke Energy submitted for US NRC staff review the Aging Management Program (AMP) and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals (Application to Implement MRP-227-A). The US NRC has issued RAIs on this submittal (Reference 5) and this report provides the Framatome Inc. response to RAI 2.

Information considered proprietary to Framatome Inc. is marked in square brackets.

2.0 FRAMATOME INC. RESPONSE TO NRC RAI 2

2.1 *Statement of RAI 2*

Regarding the core barrel and lower former plate plugs, Attachment 2 of the May 9, 2018 supplement states that the McGuire core barrel and lower former plate plugs is made of Type 316L SS [stainless steel]; and irradiation embrittlement (IE), irradiation-enhanced stress relaxation and creep (ISR/IC), fatigue, and wear were identified as degradation mechanisms for them through the process of MRP-191. The NRC staff notes that the interface (contact) pressure between the plugs and the hosting components could be lost due to stress relaxation. To address this, the attachment states that the minimum required stress ratios were calculated for the core barrel plugs and the lower former plate plugs based on their respective loads and required interface pressure for 60 years. The licensee states in section 3.3.2 of Attachment 2 of the May 9, 2018, letter, these stresses ratios were less than the estimated stress ratios based on laboratory studies for these two types of plugs.

Please provide detailed information and data regarding these laboratory studies and explain why the data can be used in this application for the McGuire core barrel and lower former plate plugs.

2.2 *Response to RAI 2*

For evaluation of the stress ratio of end of life (EOL) interface pressure to beginning of life (BOL) interface pressure for the lower former plate and core barrel plugs at 60 years, recent data were reviewed. Appendix H of MRP-175, Revision 1 (Reference 6) was consulted for recent sources of irradiation-enhanced stress relaxation data. These sources were identified as:

1. Obata et al. (Reference 7)

   This paper evaluates ISR of residual stress, which is formed in solution annealed Type 304 stainless steel by welding. Data were provided up to about 2E21 n/cm^2 and reported to be about 3.6 dpa.

2. Takakura et al. (Reference 8)

   This paper evaluates the irradiation-enhanced creep rate of cold-worked (CW) Type 316 material. Data were provided up to about 1 dpa (about[1] 6.67E20 n/cm^2).

3. Garnier et al. (Reference 9)

   This paper evaluates the deformation under combined irradiation and constant applied stress of solution annealed Type 304L and CW Type 316 materials. Data were provided to 120 dpa (about[1] 8.0E22 n/cm^2).

4. Foster et al. (Reference 10)

   This paper evaluates the relationship between stress relaxation and irradiation creep and develops a method that may be used to calculate stress relaxation using the CW Type 316 irradiation creep data. Data were provided up to 1.9 dpa (about[1] 1.27E21 n/cm^2).

5. Ishiyama et al. (Reference 11)

   This paper obtains fundamental data of ISR of stainless steels and evaluates relaxation behavior by using bent beam and C-ring specimens. The specimens were fabricated from solution annealed Type 304, 316L, and XM-19 stainless steels and irradiated up to 4.43 dpa (about 2.5E21 n/cm^2).

---

[1] Using a conversion factor of 15 dpa = 1E22 n/cm^2.

Data from Causey et al. (Reference 12) were also considered as this reference [ ]

Three of these references (Foster et al., Ishiyama et al., and Causey et al.) provide values of stress ratio based on dose. From these three references, [ ] Therefore, based on the dose and material type, the data [ ] were selected to calculate the stress ratio of EOL interface pressure to BOL interface pressure for the lower former plate and core barrel plugs at 60 years.

[ ]

Figure 2-1
Stress Ratio versus Dose

[ ]

[ ] the stress ratio of EOL interface pressure to BOL interface pressure for the lower former plate and core barrel plugs was calculated for each McGuire unit.

3.0 REFERENCES

1. Letter MNS-17-050, "Duke Energy Carolinas, LLC (Duke Energy), McGuire Nuclear Station, Units 1 and 2, Docket Nos. 50-369 and 50-370, Review Request for the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A," December 13, 2017, NRC Accession Number ML17356A177.
2. Attachment 1 (to Letter MNS-17-050), "Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals (Application to Implement MRP-227-A), WCAP-18265-NP, Rev. 0 (Non-Proprietary)," NRC Accession Number ML17356A178.
3. Letter MNS-18-029, "Duke Energy Carolinas, LLC (Duke Energy), McGuire Nuclear Station, Units 1 and 2, Docket Nos. 50-369 and 50-370, Resolution of Commitments related to Review Request for the Aging Management Program and Inspection Plan for the McGuire Nuclear Station Units 1 and 2 Reactor Vessel Internals to Implement MRP-227-A, May 9, 2018, NRC Accession Number ML18135A071.
4. Attachment 2 (to Letter MNS-18-029), "Framatome ANP-3658 Revision 0, Evaluation of the McGuire Units 1 and 2 Upflow Modification for 60 Years, Technical Report," NRC Accession Number ML18135A072.
5. Email from Michael Mahoney (US NRC) to Art Zaremba (Duke Energy), "Request for Additional Information - McGuire Nuclear Station, Units 1 and 2 - MRP-227 Review (EPID L-2017-LLA-0414)," December 18, 2018, NRC Accession Number ML18352A805.
6. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175, Revision 1). EPRI, Palo Alto, CA: 2017. 3002010268.

7. Obata, M., et al., “Radiation-Induced Stress Relaxation of Welded Type 304 Stainless Steel Evaluated by Neutron Diffraction,” Effects of Radiation on Materials: 22nd Symposium, American Society for Testing and Materials, Vol. 1475 STP, 2006, p. 15.

8. Takakura, K., et al., “In-Pile Creep Behavior of Type 316 Stainless Steel at Halden Reactor,” International Symposium: Contributions of Materials Investigations to Improve the Safety and Performance of LWRs, Fontevraud 7, SFEN, 2010.

9. Garnier, J., et al., “Irradiation Creep of SA 304L and CW316 Stainless Steels: Mechanical Behavior and Microstructural Aspects. Part I: Experimental Results,” Journal of Nuclear Materials, Vol. 413 (2011), pp. 63-69.

10. Foster, J.P., Gilbert, E.R., Bunde, K., and Porter, D.L., “Relationship Between In-Reactor Stress Relaxation and Irradiation Creep,” Journal of Nuclear Materials, 252 (1998), pp. 89-97.

11. Ishiyama, Y., et al., “Stress Relaxation Caused by Neutron-Irradiation at 561 K in Austenitic Stainless Steels,” 11th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, ANS, 2003.

12. Causey, A. R., et al., “In-Reactor Stress Relaxation of Selected Metals and Alloys at Low Temperatures,” Journal of Nuclear Materials 90 (1980) pp. 216-223.