From: Mahoney, Michael
Sent: Wednesday, December 15, 2021 7:54 PM
To: Sigmon, Chet Austin
Cc: Grzeck, Lee
Subject: Request for Additional Information - Duke Fleet Request RA-19-0352 - Alternative for RPV Closure Stud Exams (L-2020-LLR-0156)
Attachments: RAIs - Duke Fleet - Alternative Request for Reactor Closure Studs - EPID L-2020-LLR-0156.docx

Hi Chet,

Attached are RAIs for the subject Duke Fleet relief request RA-19-0352 dated December 1, 2020, for a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the reactor vessel head closure studs of Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick), Catawba Nuclear Station, Units 1 and 2 (Catawba), McGuire Nuclear Station, Units 1 and 2 (McGuire), and Shearon Harris Nuclear Power Plant, Unit 1 (Harris).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR Part 50), Paragraph 50.55a(z)(1), the licensee is proposing to extend the frequency of the reactor vessel head closure stud volumetric or surface examination required by ASME Code, Section XI for the remainder of the currently licensed operating periods for the subject units of the Duke Energy fleet from the current frequency of every 10 years. The licensee referred to the results of the deterministic fracture mechanics analyses in the following Electric Power Research Institute (EPRI) non-proprietary report as the primary basis for proposing to extend the frequency of examinations for the requested reactor vessel head closure studs: EPRI report 3002014589, "Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs," November 2018 (i.e., EPRI report 14589).

As agreed to on December 13, 2021, during the RAI clarification call, please provide the response to the attached RAIs within 45 days of the date of this electronic correspondence.

Thanks

Mike Mahoney
Project Manager, LPL2-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
Desk: (301)-415-3867
Mobile: (301)-250-0450
Email: Michael.Mahoney@nrc.gov

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Created By: Michael.Mahoney@nrc.gov

Recipients:
"Grzeck, Lee" <Lee.Grzeck@duke-energy.com>
Tracking Status: None
"Sigmon, Chet Austin" <Chet.Sigmon@duke-energy.com>
Tracking Status: None

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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE NO. RA-19-0352

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NOS.: 50-325, 50-324, 50-413, 50-414, 50-369, 50-370, AND 50-400

EPID NO: L-2020-LLR-0156

Background

By letter dated December 1, 2020 (Agencywide Document Access and Management System Accession Number ML20336A033), Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (Duke Energy, the licensee) submitted to the United States Nuclear Regulatory Commission (NRC), a proposed alternative to the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the reactor vessel head closure studs of Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick), Catawba Nuclear Station, Units 1 and 2 (Catawba), McGuire Nuclear Station, Units 1 and 2 (McGuire), and Shearon Harris Nuclear Power Plant, Unit 1 (Harris).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR Part 50), Paragraph 50.55a(z)(1), the licensee is proposing to extend the frequency of the reactor vessel head closure stud volumetric or surface examination required by ASME Code, Section XI for the remainder of the currently licensed operating periods for the subject units of the Duke Energy fleet from the current frequency of every 10 years. The licensee referred to the results of the deterministic fracture mechanics analyses in the following Electric Power Research Institute (EPRI) non-proprietary report as the primary basis for proposing to extend the frequency of examinations for the requested reactor vessel head closure studs: EPRI report 3002014589, “Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs,” November 2018 (i.e., EPRI report 14589).

The NRC staff (the staff) needs additional information to complete its review, as identified as below.

Regulatory Basis

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in United States nuclear power plants. Among these requirements are the ISI examinations of Section XI of the ASME Code, incorporated by reference in 10 CFR Part 50.55a, to ensure that adequate structural integrity of the reactor vessel head closure studs is maintained through the service life of the reactor vessel. Therefore, the regulatory basis for the following requests for additional information (RAIs) is associated with the need to

demonstrate that the proposed alternative ISI examinations would ensure continued adequate structural integrity of the reactor vessel head closure studs of the subject units of the Duke Energy fleet for the remainder of the current licensed operating period, and thereby would provide an acceptable level of quality and safety per 10 CFR 50.55a(z)(1) for the reactor vessel head closure studs.

Request for Additional Information (RAI)-1

Issue

The licensee's proposed alternative request relies heavily on the results of the evaluation in EPRI report 14589, which has not been submitted to the NRC for review. Because of this reliance on the results of a report that has not been submitted for NRC review, the licensee's plant-specific request must include EPRI report 14589 for the NRC staff to make its regulatory findings on the request.

Request

Submit EPRI report 3002014589, "Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs," on the docket.

RAI-2

Issue

The NRC staff needs additional information related to the stud preload, as noted below:

a. In Section 3.3.2 of EPRI report 14589, EPRI stated that the stud average preload stress for pressurized water reactors (PWRs) is 46.1 ksi, which is different than the preload membrane stress value of 50.2 ksi given in Table 3-4 of EPRI report 14589.

b. The NRC staff noted that a previous Duke submittal for the RPV threads-in-flange (ADAMS Accession No. ML17221A305) showed actual stud preload stress values for some of the Duke Energy units requested in the current submittal, except for Brunswick, Unit 2 and Catawba, Unit 1. The NRC staff needs to confirm the stud preload stress values for Brunswick, Unit 2 and Catawba, Unit 1.

c. The NRC staff noted that the actual stud preload stress value of 44.328 ksi for Brunswick, Unit 1 (and Brunswick, Unit 2 if information in RAI-2b is confirmed) is higher than the preload stud membrane stress value of 41.6 ksi given in Table 3-3 of EPRI report 14589 for boiling water reactors (BWRs). The lower preload stud membrane stress value of 41.6 ksi in Table 3-3 of EPRI report 14589 could result in nonconservative fatigue crack growth, which could ultimately impact the requested interval extensions for Brunswick.

d. In Section 4.2.1 of EPRI report 14589, EPRI stated that "consistent with Paragraph G-2222(b) [of Section XI of the ASME Code], stresses from bolt preloading are considered primary loads." The NRC staff would expect that the licensee considers stress due to RPV internal pressure a primary stress, but the submittal is not clear whether that is the case.

e. In Attachment 1 of the submittal, the licensee stated that a fracture toughness

(symbolized by the parameter K_{IC}) value of 190 ksi√in (discussed in Section 4 of EPRI report 14589) was used for the reactor vessel head closure studs of the subject Duke units. The NRC staff is not clear about the values of the temperatures during stud tensioning and preloading of the studs of the subject Duke units relative to the temperature at the K_{IC} value of 190 ksi√in.

Request

a. Confirm that the higher stud preload membrane stress value of 50.2 ksi given in Table 3-4 of EPRI report 14589 was the value used in the analysis.

b. Confirm that Brunswick, Unit 2 has the same actual stud preload stress value of 44.328 ksi for Brunswick, Unit 1 given in Table 2 of the previous Duke submittal and that Catawba, Unit 1 has the same actual stud preload stress value of 41.144 ksi for Catawba, Unit 2 given in the table.

c. Explain the impact of the higher actual stud preload stress value of 44.328 ksi for Brunswick, as compared to the preload stress value of 41.6 ksi in Table 3-3 of EPRI report 14589 for BWRs on the requested interval extensions for Brunswick, Units 1 and 2.

d. Confirm that in addition to stresses from bolt (i.e., reactor vessel head closure stud) preloading, the other primary stress used in the allowable flaw sizes discussed in Section 4.2 of EPRI report 14589 is RPV internal pressure stress.

e. State the temperature during stud tensioning and preloading of the reactor vessel head closure studs at each of the subject Duke Energy units and compare with the temperature at the K_{IC} value of 190 ksi√in.

RAI-3

Issue

The NRC staff needs additional information related to the applied loads, as noted below:

a. Section 3.3.3 of EPRI report 14589 stated that a hydrotest was included in the evaluation. However, in its review, the staff did not receive information with respect to a leakage test. The NRC staff needs this information since the leakage test needs to be accounted for as one of the loading conditions in the flaw tolerance evaluation.

b. In Section 3.3.3 of EPRI report 14589, EPRI stated that for PWRs, the normal operating pressure of 2,185 psi and normal operating temperature of 579°F were applied. The NRC staff noted that some of the Duke Energy PWR units included in the request may have higher normal operating pressure and normal operating temperature than the values used in EPRI report 14589. Catawba, Units 1 and 2, for example, have higher operating pressures and temperatures per the Catawba Updated Final Safety Analysis Report (UFSAR).

c. The NRC staff noted that the flaw tolerance evaluation in EPRI report 14589 did not include seismic loading and loading due to loss-of-coolant accident (LOCA). Section 3.2 of EPRI report 14589, stated that other mechanical loadings, such as those from seismic cases, do not generate significant additional loads for the RPV closure head. The NRC

staff noted that seismic and LOCA events could cause the most limiting loads. Seismic loads, for instance, could generate relative motion between the reactor closure head and the reactor vessel closure flange, and thus generate additional loads on the reactor vessel head closure studs. These additional loads combined with operating loads could result in the most limiting flaw size in the studs, when the applied stress intensity factors (SIFs) are compared to fracture toughness in determining the maximum flaw sizes in Section 4.2.3 of EPRI report 14589.

Request

a. Explain how the leakage test performed for each of the Duke Energy units included in the request is bounded, in terms of stress and cycles, by the transients selected in the flaw tolerance evaluation in EPRI report 14589.

b. For each PWR unit included in the request, either confirm that the normal operating pressure of 2,185 psi and temperature of 579°F used in EPRI report 14589 bound the corresponding values for the unit or explain how the 2,185 psi and 579°F are adequate for each PWR unit.

c. Explain how seismic and LOCA events do not generate significant additional loads for the reactor vessel head closure studs of the Duke Energy units in the request.

RAI-4

Issue

Section 4.2.1 of EPRI report 14589 discusses the methodology for determining the limiting flaw size in the reactor vessel head closure studs. EPRI stated that a safety factor of 2.0 was applied on the primary loads based on the methods in nonmandatory Appendix G, Paragraphs G-2215 and G-2222 of the ASME Code, Section XI. EPRI cited a 2017 NRC safety evaluation (ADAMS Accession No. ML17006A109) that authorized a plant-specific alternative examination request for the reactor vessel threads-in-flange. The reactor vessel threads-in-flange are the components into which the reactor vessel head closure studs are threaded. EPRI stated that the use of the methods of Appendix G of ASME Code, Section XI, is consistent with the NRC position in the 2017 NRC safety evaluation regarding the plant-specific reactor vessel threads-in-flange.

The EPRI report noted that the methods in Paragraphs G-2215 and G-2222 of Appendix G of ASME Code, Section XI, are for vessel components. The report also noted that for bolting materials (i.e., the reactor vessel head closure studs), the recommended methods for evaluating fracture prevention are in Article G-4000 of ASME Code, Section XI, which refers to Welding Research Council Bulletin (WRCB) 175, “PVRC Recommendations on Toughness Requirements for Ferritic Materials.” EPRI stated that the evaluation methods in WRCB 175 are used primarily to define toughness criteria for bolts with a reference flaw size, and not to evaluate flaws with defined structural (i.e., safety) factors. EPRI stated that WRCB 175 is considerably older than other references cited for fracture mechanics evaluations in bolted joints and none of the solutions discussed in it regarding bolting are specific for bolted joints. Thus, EPRI used the safety factors in Paragraphs G-2215 and G-2222 of Appendix G of ASME Code, Section XI, for vessels, with the SIFs described in Section 4.1 of EPRI report 14589, to define the limiting flaw size for the postulated flaws in the reactor vessel head closure studs.

The NRC staff confirmed the information in Appendix G of ASME Code, Section XI and WRCB

175 that EPRI cited, as discussed above. Even though EPRI acknowledged that the safety factor of 2.0 in Appendix G of ASME Code, Section XI is for vessels, the NRC staff assessed the basis for the safety factor. Chapter 30 of the Companion Guide to the ASME Code (Volume 2) explains that a safety of factor of 2.0 applied on the SIF due to the pressure loading (i.e., primary load) combined with a conservative postulated flaw size (i.e., a depth of one-quarter of the vessel thickness) ensures a safety factor of 3 on primary load that is consistent with the ASME Code, Section III design stress limits on vessels. Given the basis of the safety factor of 2.0 and that the postulated flaw sizes in the reactor vessel head closure studs assumed in EPRI report 14589 are relatively small, the staff is not clear whether applying a safety factor of 2.0 for the reactor vessel head closure studs achieves the same level of margin as the corresponding concept in vessels with regard to the ASME Code, Section III, design stress limits on the reactor vessel head closure studs. The staff accepted the use of a safety factor of 2.0 for the plant-specific reactor vessel threads-in-flange evaluated in the 2017 NRC safety evaluation because the postulated flaw sizes in the reactor vessel threads-in-flange were large (full 360-degree flaw and relatively deep), and therefore provided adequate margin for the reactor vessel threads-in-flange analyses.

<u>Request</u>

Explain how a safety factor of 2.0 applied on the SIF due to primary loads on a postulated semi-circular flaw in the reactor vessel head closure stud ensures an adequate safety factor on primary load that is consistent with the ASME Code, Section III design stress limits on the reactor vessel head closure stud.

RAI-5

<u>Issue</u>

Section 4 of EPRI report 14589 states a K_{IC} value of 190 ksi√in based on Charpy impact testing and fracture toughness data of SA-540 steels used for reactor vessel closure studs reported in a 1977 paper in the Journal of Pressure Vessel Technology (JPVT). Figures 6 and 7 of the 1977 JPVT paper show the Charpy impact (and lateral expansion) testing and fracture toughness data, respectively. The staff needs clarification on the information in Figure 7 of the 1977 JPVT paper, more information on whether a K_{IC} value of 190 ksi√in is appropriate for the flaw tolerance evaluation, and confirmation of Charpy impact values of the reactor vessel head closure studs of the subject Duke Energy units.

a. The staff noted that Figure 7 of the 1977 JPVT paper has two y-axes: fracture toughness and impact energy. The staff understands that the Charpy impact curves in Figure 7 of the 1977 JPVT paper represent the range of impact energies from Figure 6 of the 1977 JPVT paper. However, because the y-axes of fracture toughness and impact energy are presented side-by-side, it appears that the figure shows the correspondence of fracture toughness and impact energy. For example, the upper-shelf impact energy value at room temperature (70°F) from the lower Charpy impact curve in Figure 7 of the 1977 JPVT paper is about 40 ft-lb, which appears to correspond to a fracture toughness value of 100 ksi√in.

b. The staff verified from Figure 7 of the 1977 JPVT the K_{IC} value of 190 ksi√in selected for analysis. It is the minimum of the range of upper-shelf K_{IC} values at room temperature, as stated in the 1977 JPVT and shown in Figure 7 of the paper. However, since pressure loading occurs at temperatures higher than room temperature and the upper-shelf K_{IC} data in Figure 7 of the 1977 JPVT show that upper-shelf K_{IC} values decrease with

increasing temperature, the selection of an upper-shelf K_{IC} value at room temperature may be nonconservative.

c. Figure 7 of the 1977 JPVT paper shows a lower Charpy impact property curve for the SA-540 steel heats used to generate the curve. NB-2333 of Section III of the ASME Code specifies requirements for Charpy impact property values for bolting materials. Because of this requirement, Charpy impact values of the reactor vessel head closure studs of the subject Duke Energy units should be available, and thus may be compared to the lower Charpy impact property curve in Figure 7 of the 1977 JPVT paper.

Request

a. Clarify whether there is correspondence between fracture toughness and impact energy in Figure 7 of the 1977 JPVT paper and if there is correspondence, justify the selection of the high fracture toughness value of 190 ksi√in for the flaw tolerance evaluation.

b. Re-perform the flaw tolerance evaluation using a minimum upper-shelf K_{IC} value at temperatures corresponding to full pressure of PWRs and BWRs or justify the selection of an upper-shelf K_{IC} value of 190 ksi√in at room temperature for the flaw tolerance evaluation, given that pressure loading occurs at temperatures higher than room temperature and upper-shelf K_{IC} values decrease with increasing temperature.

c. Confirm that the available Charpy impact values of the reactor vessel head closure studs of each of the subject Duke Energy units are above the lower Charpy impact property curve shown in Figure 7 of the 1977 JPVT paper.

RAI-6

Issue

In Section 4.3.2 of EPRI report 14589, EPRI performed generic fatigue crack growth (FCG) calculations for the reactor vessel head closure studs and stated that the FCG rate was from Nonmandatory Appendix A, Subarticle A-4300 of ASME Code, Section XI. The reactor vessel head closure studs of the subject Duke Energy units are made of high strength bolting materials specified as SA-540 Grade B23 or B24 (yield strength up to 150 ksi or ultimate strength up to 170 ksi), as the licensee stated in Attachments 2 through 8 to the submittal. The NRC noted that, as stated in Subarticle A-1100 of ASME Code, Section XI, the scope of Appendix A applies to ferritic materials 4 inches and greater in thickness with specific minimum yield strengths of 50.0 ksi or less, which implies that the A-4300 FCG rate applies only to vessel materials with specific minimum yield strengths of 50.0 ksi or less.

The staff consulted the Companion Guide to the ASME Code (Volume 2) for additional guidance on the materials under the scope of Appendix A of ASME Code, Section XI. Section 32.1.5 of the guide states that the majority of the A-4300 reference FCG rate were from SA-508 and SA-533 materials, which are common steels used for vessels and have minimum specified yield strengths of 50 ksi or less. The guide also cites Barsom[1] that explained that ferritic steels having a range of yield strengths from 45 ksi to 300 ksi showed similar crack growth behavior. The staff noted that this large range of yield strength includes the yield strength of SA-540 Grade B23 or

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[1] Barsom, J. M., "Fatigue Crack Growth Propagation in Steels of Various Yield Strengths," Trans. of ASME, Journal of Engineering for Industry, Series B, Vol. 93, No. 4, pp. 1190-1196, Nov. 1971.

B24 used for reactor vessel head closure studs. However, the staff compared the FCG rate cited by Barsom with the A-4300 FCG rates in Figure A-4300-1 of the ASME Code, Section XI and noted that the Barsom FCG rate could be higher than the A-4300 FCG rates.

Given the discussion above, the staff noted that the A-4300 FCG rate may not be appropriate for the high strength bolting materials of the reactor vessel head closure studs, specified as SA-540 Grade B23 or B24, of the subject Duke Energy units. The staff is also not clear whether the FCG analysis described in Section 4.3.2 of EPRI report 14589 included safety factors on the applied SIFs due to membrane and bending stresses as the guidance in C-7000 of the ASME Code, Section XI specifies for evaluations using linear elastic fracture mechanics. The staff notes that the evaluation procedures for FCG in Appendix A of the ASME Code, Section XI, cited in Section 4.3.2 of EPRI report 14589 are typically for the reactor vessel and that the evaluation procedures in C-7000 of the ASME Code, Section XI, that specify safety factors for the applied SIFs due to membrane and bending stresses would be more applicable for the reactor vessel head closure studs.

The staff further noted that the licensee is using only deterministic analysis (versus probabilistic analysis) in EPRI report 14589 as basis for eliminating the required ASME Code, Section XI volumetric or surface examination for the reactor vessel head closure studs of the subject Duke units. The A-4300 FCG rates in Figure A-4300-1 of the ASME Code, Section XI used in the deterministic analysis in EPRI report 14589 are based on the median of the data (specifically, 95 percent confidence that the A-4300 FCG rate bounds the median of the data) used to establish those rates. Thus, the A-4300 FCG rates used in EPRI report 14589 are not upper bound rates, and therefore may not be conservative.

Request

a. Provide alternate FCG calculations using an FCG rate applicable to high strength bolting materials appropriate for the reactor vessel head closure studs (address using an upper bound curve for the alternate FCG rate, similar to RAI-6c), specified as SA-540 Grade B23 or B24, of the subject Duke Energy units, or justify that the FCG rate in A-4300 of the ASME Code, Section XI, is adequate for the reactor vessel head closure studs specified as SA-540 Grade B23 or B24.

b. Clarify whether appropriate safety factors were applied on the membrane and bending stresses used for calculating the applied SIFs for the reactor vessel head closure studs. If safety factors were not applied, either recalculate the FCG calculations with the appropriate safety factors on the membrane and bending stresses used for the applied SIFs or justify not using safety factors on membrane and bending stresses used for the applied SIFs.

c. Either recalculate the FCG calculations based on upper bound FCG rates or justify how the use of median-based FCG rates (i.e., the FCG rate in A-4300 of the ASME Code, Section XI) provides reasonable assurance of structural integrity of the reactor vessel head closure studs of the subject Duke Energy units without periodic performance monitoring of the studs.

RAI-7

Issue

In Section 5.0 of Enclosure 1 of the submittal, the licensee stated that a review of Duke Energy's past ISI examination records for reactor vessel head closure studs indicates there have been no occurrences of service-induced degradation. The staff is not clear whether "no occurrences of service-induced degradation" means no relevant indications were detected in the reactor vessel head closure studs during the ASME Code, Section XI ISI examinations.

Request

For each of the Duke Energy units included in the request:

a. Clarify clear whether "no occurrences of service-induced degradation" means no relevant indications were detected in the reactor vessel closure studs during the ASME Code, Section XI ISI examinations

b. Depending on the response in RAI-7a, if relevant indications were detected, explain how the indication was dispositioned and how the size (depth and length) of the indication impacts the postulated semi-circular flaw with an initial flaw depth of 0.3 inch analyzed in the FCG calculation in Section 4.3.2 of EPRI report 14589.

RAI-8

Issue

In Attachments 2 to 8 of the submittal, the licensee stated that the Reactor Vessel Closure Stud program of each subject Duke Energy unit includes preventive measures to use stable lubricants, and specifically, to prohibit the use of molybdenum disulfide. The staff noted that this preventive measure is to mitigate the effects of stress corrosion cracking in the reactor vessel closure studs, which is discussed in Section 2.1.1 of EPRI report 14589. For BSEP Units 1 and 2, and SHNPP Unit 1, the staff verified that such a program is included in the updated final safety report (UFSAR) for the units. However, for Catawba, Units 1 and 2, and McGuire, Units 1 and 2, the staff was not able to verify which program listed in Chapter 18, "Aging Management Programs and Activities" of the corresponding UFSAR for the units includes the preventive measure that prohibits the use of molybdenum disulfide for the reactor vessel closure studs of the units.

Request

For Catawba, and McGuire, state which program listed in Chapter 18, "Aging Management Programs and Activities" of the corresponding UFSAR for the units includes the preventive measure that prohibits the use of molybdenum disulfide for the reactor vessel closure studs of the units.

RAI-9

Issue

The staff noted that in Section 5.0 of the submittal that for Catawba, Units 1 and 2, McGuire, Units 1 and 2, and Harris, the lengths of the proposed extension of the ISI intervals for the reactor vessel closure studs of these units are more than 20 years (i.e., more than two

consecutive 10-year ISI intervals). The staff notes that two consecutive 10-year ISI intervals were determined acceptable for the threads-in-flange in previous requests. Eliminating the volumetric examinations for the reactor vessel closure studs of these units during the proposed extensions eliminates the most effective method for detecting new degradation or changes in degradation *within* the reactor vessel closure studs of the units, and thereby significantly reducing condition monitoring of the reactor vessel closure studs. In Section 5.0 of the submittal, the licensee stated that the detailed procedures used during each refueling outage for the removal, care, and visual inspection of the reactor vessel closure studs and threads-in-flange provide further assurance that degradation is detected.

The staff noted that with the proposed elimination of the volumetric examination of the reactor vessel closure studs, these periodic maintenance procedures would be the only component-specific condition monitoring for the reactor vessel closure studs. It is not clear to the staff how these periodic maintenance procedures would be effective in detecting new degradation or a change in degradation within the reactor vessel closure studs because it does not examine the critical volume around the threads of the reactor vessel closure studs. Additionally, with the volumetric examination eliminated, if there is new degradation or a change in degradation, this would constitute and unanalyzed degradation and thus would not be included in the flaw tolerance analyses in EPRI report 14589. The licensee also stated that the periodic maintenance procedures, coupled with the ASME Code Section XI leak test (Examination Category B-P), provide assurance of pressure boundary integrity. However, the staff noted that the ASME Code Section XI, Examination Category B-P leak test is not component-specific and does not examine the critical volume around the threads of the reactor vessel closure studs.

<u>Request</u>

Given (1) the insufficiency of using only deterministic FCG analyses to justify elimination of volumetric examination longer than 20 years; and (2) the periodic maintenance procedures performed each refueling outage would be the only condition monitoring for the reactor vessel closure studs of Catawba, McGuire, and Harris:

a. Justify how the periodic maintenance procedures (which do not examine the critical volume around the threads of the reactor vessel closure studs) would detect new degradation or changes in degradation within the reactor vessel closure studs of these units (Catawba, Units 1 and 2, McGuire, Units 1 and 2, and Harris) for periods of longer than 20 years.

b. Explain whether these periodic maintenance procedures would be supplemented with other component-specific performance monitoring measures, such as volumetric examination of a sample of the reactor vessel closure studs of the subject Duke units, such that the critical volume around the threads of the reactor vessel closure studs is examined to ensure that new degradation or a change in degradation is detected. If not, justify how not supplementing the periodic procedures would ensure that the critical volume around the threads of the reactor vessel closure studs is examined to ensure that new degradation or a change in degradation is detected.