Steve Snider
Vice President

Nuclear Engineering
526 South Church Street, EC-07H
Charloote, NC 28202

980-373-6195
Steve.Snider@duke-energy.com

Serial: RA-19-0352 10 CFR 50.55a
December 1, 2020

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

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2, DOCKET NOS. 50-325, 50-324

CATAWBA NUCLEAR STATION, UNITS 1 AND 2, DOCKET NOS. 50-413, 50-414

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, DOCKET NOS. 50-369, 50-370

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, DOCKET NO. 50-400

SUBJECT: Relief Request for Alternative for Reactor Vessel Closure Stud Examinations

Pursuant to 10 CFR 50.55a(z)(1), Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (hereafter referred to collectively as Duke Energy) request NRC approval to implement a proposed alternative to ASME section XI IWB-2500(a), Table IWB-2500-1, Examination Category B-G-1, Item No. B6.20. Relief is being requested on the basis that the alternative provides an acceptable level of quality and safety.

The proposed alternative is requested for the remainder of the currently licensed operating periods for the Brunswick Steam Electric Plant Units 1 and 2; Catawba Nuclear Station Units 1 and 2; McGuire Nuclear Station Units 1 and 2; and Shearon Harris Nuclear Power Plant Unit 1.

A pre-submittal meeting was held with NRC staff on November 18, 2019 (ML19318F291).

If you have questions concerning this request, please contact Art Zaremba, Director - Fleet Licensing, at (980) 373-2062.

Sincerely,

Steve Snider
Vice President – Nuclear Engineering

Enclosure:

1. Request for Alternative for Reactor Vessel Closure Stud Examinations

Attachments:

1. EPRI Report Summary
2. Plant-Specific Information for: Brunswick Unit 1
3. Plant-Specific Information for: Brunswick Unit 2
4. Plant-Specific Information for: Catawba Unit 1
5. Plant-Specific Information for: Catawba Unit 2
6. Plant-Specific Information for: McGuire Unit 1
7. Plant-Specific Information for: McGuire Unit 1
8. Plant-Specific Information for: Shearon Harris Unit 1

cc : (all with Enclosure unless otherwise noted)

L. Dudes, Regional Administrator USNRC Region II
J. D. Austin, USNRC Senior Resident Inspector – CNS
G. A. Hutto, USNRC Senior Resident Inspector - MNS
J. Zeiler, USNRC Senior Resident Inspector – HNP
G. Smith, USNRC Senior Resident Inspector - BNP
M. Mahoney, NRR Project Manager – HNP
E. Miller, NRR Project Manager – MNS
K. Cotton, NRR Project Manager - CNS
A. L. Hon, NRR Project Manager - BNP

Enclosure 1

Duke Energy Carolinas, LLC

Duke Energy Progress, LLC

Request for Alternative RA-19-0352

Request for Alternative in Accordance with 10 CFR 50.55a(z)(1)

Request for Alternative for Reactor Vessel Closure Stud Examinations
(7 pages including cover page)

1.0 ASME CODE COMPONENT(S) AFFECTED:

Component:	Reactor Pressure Vessel Closure Studs
Code Class:	Class 1
Examination Category:	B-G-1 (Pressure Retaining Bolting, Greater than 2 in. in Diameter)
Code Item Number:	B6.20 – Closure Studs
Description:	Reactor Pressure Vessel Closure Studs volumetric and surface examinations

2.0 APPLICABLE CODE EDITION AND ADDENDA:

Table 1 summarizes the plants included in this Request for Alternative, the current Inservice Inspection (ISI) Interval, and the applicable Edition and Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

Table 1: Plants Included in This Request for Alternative and Their Current ISI Intervals and Applicable ASME Code Section XI Editions/Addenda

Plant/Unit(s)	ISI Interval	ASME Section XI Code Edition/Addenda	Interval Start Date	Interval End Date
Brunswick Steam Electric Plant Units 1 and 2	Fifth	2007 Edition, Through 2008 Addendum	05/11/2018	05/10/2028
Catawba Nuclear Station Units 1 and 2	Fourth	2007 Edition, Through 2008 Addendum	08/19/2015	12/06/2024 (Unit 1) 02/24/2026 (Unit 2)
McGuire Nuclear Station Units 1 and 2	Fourth	2007 Edition, Through 2008 Addendum	12/01/2011 (Unit 1) 07/15/2014 (Unit 2)	11/30/2021 (Unit 1) 12/14/2024 (Unit 2)
Shearon Harris Nuclear Power Plant Unit 1	Fourth	2007 Edition, Through 2008 Addendum	09/09/2017	09/08/2027

3.0 APPLICABLE CODE REQUIREMENT:

ASME Section XI IWB-2500(a), Table IWB-2500-1, Examination Category B-G-1, Item No. B6.20 requires volumetric examination of the reactor pressure vessel (RPV) closure studs once during each Section XI inspection interval, nominally every ten years. Per Note (1) of Table IWB-2500-1 (B-G-1), the "bolting may be examined: (a) in place under tension, (b) when the connection is disassembled, (c) when the bolting is removed." The examination volume is the entire engaged length of the stud to a depth of 1/4 in. from the thread root, as shown in ASME Section XI Figure IWB-2500-12. Additionally, per Note (7) of Table IWB-2500-1 (B-G-1), "when bolts or studs are removed for examination, surface examination meeting the acceptance criteria of IWB-3515 may be substituted for volumetric examination." The surface examination area is the stud external surface along the entire engaged length of the stud, as shown in ASME Section XI Figure IWB-2500-12.

4.0 REASON FOR REQUEST:

The Electric Power Research Institute (EPRI) recently developed a technical basis report [8.1] that determined the requirement to perform inservice volumetric or surface exams of RPV closure studs (Examination Category B-G-1, Item No. B6.20) could be eliminated for the current operating periods identified in Table 2 without increasing plant risk or posing any safety concerns for the RPV. The EPRI report provides the supporting technical basis to allow longer inspection intervals for the closure studs while maintaining the appropriate safety margins required by ASME Section XI, Division 1.

The analysis methodology and the results of the EPRI report are summarized in Attachment 1. Attachments 2 through 8 demonstrate that the EPRI report analysis methods are applicable to the plants listed in Table 1.

As noted in Attachment 1, the EPRI report technical basis considers the degradation mechanisms applicable to RPV closure studs, including (1) fatigue, (2) stress corrosion cracking, (3) boric acid corrosion (pressurized water reactors only), and (4) steam cutting. Based on a review of operating experience, the quantitative assessments in this technical basis report focus on the potential for RPV stud degradation caused by fatigue mechanisms and the time for the postulated flaw to propagate beyond an acceptable flaw size can be used to optimize an appropriate inspection frequency. Historically, stress corrosion cracking (SCC) is the degradation mechanism that has led to failures of Class 1 structural bolting. However, the causes of SCC degradation were identified and are now addressed procedurally through the tensioning process and lubricant chemical compatibility. Leakage from the RPV flange is detectable by the equipment that monitors the leak-off space between the two concentric vessel o-rings and plant operating procedures require shutdown in the event detected leakage exceeds technical specification leakage limits.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

In accordance with 10 CFR 50.55a(z)(1), Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (Duke Energy) are requesting a proposed alternative to the requirement to perform inservice volumetric or surface examinations of Examination Category B-G-1, Item Number B6.20, RPV Closure Studs on a 10-year ISI interval. The proposed alternative is to extend the frequency of reactor vessel closure stud volumetric or surface examination for the remainder of the currently licensed operating periods for the plants listed in Table 1. The current licensing periods for these plants are summarized in Table 2.

Table 2: Current ISI Intervals and License Periods

Plant/Unit	Current ISI Interval End Date	Current License Period End Date	Date of Last Category B-G-1 Examination	Length of Relief Requested (years)
Brunswick Unit 1	05/10/2028	09/08/2036	03/30/2018	18
Brunswick Unit 2	05/10/2028	12/27/2034	04/11/2017	17
Catawba Unit 1	12/06/2024	12/05/2043	12/05/2015	28
Catawba Unit 2	02/24/2026	12/05/2043	03/19/2015	28
McGuire Unit 1	11/30/2021	06/12/2041	03/31/2013	28
McGuire Unit 2	12/14/2024	03/03/2043	09/19/2015	27
Shearon Harris Unit 1	09/08/2027	10/24/2046	04/27/2009 (See Note 1)	37

Notes:

1. For the 3rd Interval RPV Stud volumetric exams at HNP, 1/3 of the RPV studs were performed in each of the following outages:
   a. RFO-15 (4/27/2009) – RPV Studs 1-19
   b. RFO-16 (10/13/2010) – RPV Studs 20-38
   c. RFO-20 (10/15/2016) – RPV Studs 39-58

   No relevant indications were identified in any of the examinations at HNP listed above. Conservatively, the date of the last B-G-1 examination and length of the request for HNP was based on exams performed in RFO-15 (RPV Studs 1-19). However, RPV studs 39-58 were most recently performed in 2016.

As indicated in Table 2, the proposed alternative results in a maximum effective operating time period of 37 years, 5 months, and 28 days from the last inspection for the Pressurized Water Reactor (PWR) plants and 18 years, 5 months, and 10 days from the last inspection for the Boiling Water Reactor (BWR) plants included in this Request for Alternative. As summarized in Attachment 1, the EPRI report [8.1] demonstrates that time intervals longer than 30 years (80 years for PWR and 37.9 years for BWR) are required for flaws to reach ASME Section XI acceptance limits.

The EPRI report [8.1] provides the basis for the frequency extension of the RPV closure studs examination requirement (ASME Section XI Examination Category B-G-1, Item Number B6.20). A review of Duke Energy's past ISI examination records for RPV closure studs indicates there has been no occurrences of service-induced degradation. Also, there are negative impacts for performing these exams related to work dose, personnel safety, radwaste, and critical path time. Also, additional time at reduced RCS water inventory for BWRs, since the volumetric examinations are typically performed with the studs installed in the RPV flange.

System leakage tests are performed each refueling outage for both PWRs and BWRs in accordance with ASME Code, Section XI, Table IWB-2500-1; Examination Category B-P, during which a VT-2 visual examination is performed to detect evidence of leakage. For the PWR plants, to protect against non-service-related degradation, detailed procedures are used during each refueling outage for the removal, care, and visual inspection of the RPV studs and the threads in flange. Prior to reinstallation, the studs and stud holes are cleaned and lubricated with a stable lubricant (avoiding use of molybdenum disulfide). Controls are in place to ensure chemical compatibility and stability of lubricants and surface treatments. The studs are then replaced into the RPV flange, the top head is replaced, and the RPV closure studs are tensioned. These activities are performed during each refueling outage and each step is documented per plant procedures. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service and, coupled with the B-P (VT-2) examinations, they provide additional assurance of pressure boundary integrity.

For BWR plants, to protect against non-service-related degradation, detailed procedures are used during each refueling outage for the removal, care, and visual inspection of the RPV studs and the threads in flange. Unlike a PWR, all the RPV studs are not removed each outage from the vessel flange. Typically, only 4 RPV studs are removed in support of refueling activities and stored on the refuel floor. The remaining studs are left in place. Care is taken to not only remove the 4 RPV studs, but once the studs are removed, the 4 RPV threads are inspected for damage. Prior to reinstallation, the 4 studs and 4 stud holes are cleaned and lubricated with a stable lubricant (avoiding use of molybdenum disulfide). Controls are in place to ensure chemical compatibility and stability of lubricants and surface treatments. The 4 studs are then replaced into the RPV flange, the top head is replaced, and the RPV closure studs are tensioned. These activities are performed during each refueling outage and each step is documented per plant procedures. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service and, coupled with the B-P (VT-2) examinations, they provide additional assurance of pressure boundary integrity.

For both the BWR and PWR units, the RPV closure head is sealed using two self-energized metallic O-rings. The volume between these O-rings are instrumented to monitor for changes in temperature or pressure and any leakage through the inner seal. In combination with low administrative limits for leakage through the O-ring joint, this monitoring ensures that any increase in main flange leakage is recognized and the potential for steam cutting (erosion corrosion) to occur can be minimized.

Ongoing review of both plant specific and industry Operating Experience (OE), including relevant research and development, ensures Aging Management Programs will be enhanced or modified to continue to remain effective in managing aging effects such that intended functions will be maintained for the current licensed operating period. The established methods for determining aging effects and degradation mechanisms for RPV closure studs are reasonable and no unpredicted aging unique to RPV closure studs have yet been identified. The review of applicable OE will ensure the aging management program is enhanced, as appropriate, when it is determined through the evaluation of OE that the effects of aging may not be adequately managed. This should provide objective evidence to support the conclusion that the effects of aging will be managed adequately. Furthermore, actions for developing concerns regarding aging phenomena which may be identified during the extended period of no examinations is assured by Duke Energy's administratively mandated corrective action program and its participation in and review of industry OE.

For these reasons, Duke Energy requests authorization to use the proposed alternative pursuant to 10 CFR 50.55a(z)(1) on the basis that the alternative provides an acceptable level of quality and safety.

6.0 DURATION OF PROPOSED ALTERNATIVE:

Typically, the duration for requests approved under the provisions of 10 CFR 50.55a(z) are limited to the active inservice inspection ten-year interval for which they are submitted. However, there are instances where the NRC approved requests under the provisions of 10 CFR 50.55a(z) for durations longer than the 10 years associated with an inservice inspection interval. For example, licensee's requests to reduce examinations of BWR reactor vessel circumferential welds based on BWRVIP-05 [8.2].

As stated herein, the EPRI report [8.1] demonstrates that time intervals longer than 30 years (80 years for PWR and 37.9 years for BWR) are required for flaws to reach ASME Section XI acceptance limits. Therefore, Duke Energy requests approval of this proposed alternative for the remainder of the currently licensed operating periods for the plants listed in Tables 1 and 2.

7.0 PRECEDENTS:

There have not been any previous submittals that request relief from the ASME Examination Category B-G-1, B6.20 examination of RPV closure studs. However, Section 2.2 of the EPRI report [8.1] summarizes reactor vessel closure stud operating experience and cites other related NRC requests for RPV closure studs.

The NRC has recently authorized the use of an alternative to examination of the RPV threads in flange, ASME Examination Category B-G-1, Item Number B6.40. The NRC approval for these alternative relief requests was provided as noted below.

7.1. **ADAMS Accession Number ML17331A086. NRC approved dated December 26, 2017.** Duke Energy Carolinas, LLC & Duke Energy Progress, LLC, "Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1) for Examination of ASME Section XI, Examination Category B-G-1, Item Number B6.40, Threads in Flange (17-GO-001)", dated March 29, 2017 (ML17088A846).

7.2. **ADAMS Accession Number ML17006A109. NRC approved dated January 26, 2017.** Vogtle Electric Generating Plant, Units 1 and 2 , and Joseph M Farley Nuclear Plant, Unit 1, "Alternative to Inservice Inspection Regarding Reactor Pressure Vessel Threads in Flange Inspection (CAC Nos. MF8061, MF8062, MF8070)", dated August 4, 2016 (ML16221A072).

7.3. **ADAMS Accession Number ML17170A013. NRC approved dated June 26, 2017.** Exelon Generation Co. LLC, "Proposed Alternative for Examination of ASME Section XI, Examination Category B-G-1, Item Number B6.40, Threads in Flange", dated October 31, 2016 (ML16306A270).

7.4. **ADAMS Accession Number ML17332A663. NRC approved dated December 6, 2017.** Virginia Electric and Power Company (Dominion), "North Anna Power Station, Units 1 and 2, ASME Section XI Inservice Inspection Program Request for Proposed Alternative N1-14-NDE-009 and N2-14-NDE-004", dated November 30, 2016 (ML16340B092).

7.5. **ADAMS Accession Number ML17132A187. NRC approved dated May 25, 2017.** Dominion Nuclear Connecticut, Inc. Millstone Power Station Units 2 and 3, "Proposed Alternative Requests RR-04-24 an 1R-3-30 for Elimination of the Reactor Pressure Vessel Threads in Flange Examination", dated October 6, 2016 (ML16287A724).

8.0 REFERENCES:

8.1. *Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs.* EPRI, Palo Alto, CA: 2018. 3002014589.

8.2. **ADAMS Accession Number ML032200246.** BWRVIP-05; *"BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations"*, EPRI TR-105697, September 1995.

8.3. Duke Energy Presentation for the Pre-submittal Meeting on November 18, 2019, Regarding the RPV Stud Examination Relief Request, November 18, 2019 (ADAMS Accession Number ML19318F291).

8.4. U.S. NRC, *Materials and Inspections for Reactor Vessel Closure Studs*, Regulatory Guide 1.65, Rev. 1, April 2010.

A report from the Electric Power Research Institute (EPRI) [1] provides a technical basis evaluation of the inspection period for RPV closure studs. This attachment provides a summary of the EPRI report. The applicability for each plant included in this Request for Alternative is demonstrated in Attachments 2 through 8.

Potential Degradation Mechanisms

Section 2 of the EPRI report provides an evaluation of potential degradation mechanisms that could impact flange/threads reliability. The evaluation considers the aging effects for reactor vessel closure studs identified in the Generic Aging Lessons Learned (GALL) report [2] and the GALL Report for Subsequent License Renewal (GALL-SLR) [3]. Those effects are: (1) cumulative fatigue damage or fatigue cracking, (2) stress corrosion cracking (SCC), and (3) loss of material due to wear, general corrosion, pitting or crevice corrosion.

The EPRI report notes that the primary initiating causes for SCC in Class 1 structural bolting have been aggressive environment primarily caused by certain lubricants and hydrogen embrittlement of steels with yield strengths above 150 ksi. In particular, the GALL-SLR report Section XI.M3 specifies as a preventative action for reactor head closure stud bolting the use of bolting material with an actual measured ultimate tensile strength less than 170 ksi for existing studs and or 150 ksi yield strength for newly installed studs. These initiating causes have been removed from operating plants. The plant-specific historical usage of such lubricants is described in Attachments 2 through 8. The EPRI report also notes that loss of material due to wear, general corrosion, pitting or crevice corrosion are generally predicated on the presence of active leakage. Because the reactor vessel closure is closely monitored for leakage, and because active leakage would be apparent during vessel disassembly, degradation due to loss of material is a condition that is already monitored. The EPRI report concludes that the most plausible degradation mechanisms for the reactor vessel closure studs is mechanical fatigue and fatigue cracking. Both of these mechanisms were addressed in the stress and flaw tolerance assessments performed for the closure studs in the EPRI report.

Stress and Flaw Tolerance Assessments

Sections 3 and 4 of the EPRI report document generic stress and flaw tolerance assessments performed for the reactor vessel closure studs. The method of evaluation for the closure studs was comprised of the following elements: (1) a series of static and transient stress analyses to define the operating stresses in the reactor vessel closure studs, (2) a fatigue crack growth calculation using the operating stresses, and (3) a limiting flaw size calculation.

Section 3 of the EPRI report describes the two stress analysis models that were used in the evaluation: one model representing a bounding pressurized water reactor (PWR) closure stud geometry and one model representing a bounding boiling water reactor (BWR) closure stud geometry. The limiting nature of the bounding geometries was confirmed using multiple representative reactor vessel closure flange dimensions; the limiting plant configurations were determined to be those with the largest ratios for: (1) the reactor vessel head inside radius to reactor vessel head thickness, and (2) the reactor vessel head inside radius to closure stud diameter. Each model considered stud preload by itself and in combination with normal operating temperature and pressure along with a full range of representative plant operating transients. The reactor vessel closure preload results in a net tensile stress in the stud that is combined with a stud bending stress caused by deflection of the reactor vessel head and flange. The stud tension and bending stresses change during operating transients due primarily to differential thermal expansion between the closure flange and the closure studs.

Section 4 of the EPRI report describes flaw tolerance assessment of the closure studs. For this assessment, the closure stud tension and bending stresses determined in Section 3 of the EPRI report are used to calculate values for the crack tip stress intensity factor (SIF) of a postulated surface flaw originating from the edge of stud at the point of maximum tension plus bending stress. The postulated surface flaw propagates horizontally across the stud cross section; a fully-circumferential (360°) flaw was not considered. This approach is supported by Welding Research Council Bulletin (WRCB) 175 Paragraph 7 [4], which considered, but did not use, the more conservative 360° flaw case when developing minimum toughness requirements for bolting. The crack tip SIF solution for a surface flaw from a threaded bolt is calculated using published influence coefficients for tension and bending stress. The variation in crack tip SIF is used to calculate the growth of a postulated initial flaw with a depth equal to 0.30 inches, or 5% of the nominal 6.0-inch bolt diameter. Using typical values for numbers of transient cycles per year of operation, crack growth as a function of time is also calculated.

The limiting flaw size was determined in the flaw tolerance assessment, and was calculated using the methodology specified in ASME Section XI Appendix G. The structural factors provided in Article G-2000 for vessels are applied to the crack tip SIF values calculated as described above. While Appendix G also includes Article G-4000 "Bolting," this Article makes reference to WRCB 175 Paragraph 7, which does not evaluate flaws using defined structural factors. Therefore, consistent with Paragraphs G-2215 and G-2222, the limiting flaw size was determined using structural factors of 2.0 on the applied K_I due to primary loads and 1.0 on the applied K_I due to secondary loads, and comparing the sum of these two values to the allowed toughness, K_{IC}. Consistent with Paragraph G-2222(b), stresses from bolt preloading are considered primary loads. A significant amount of data on the fracture toughness of the SA-540 steels used for RPV studs formed the basis for establishing an appropriate K_{IC} for bolting material [5] of 190 ksi√in (209 MPa√m). In this paper, a correlation between the SA-540 material yield strength and the fracture toughness is not directly considered. Therefore, the limits on yield and ultimate strength previously noted are for preventative measures against SCC and not limitations on fracture toughness.

The results of the calculation demonstrate the following:

- The fatigue crack growth for PWR RPV studs for a postulated flaw after 80 years of assumed loading cycles remains less than the allowable flaw size using methods consistent with ASME Code, Section XI, Nonmandatory Appendix G
- The fatigue crack growth for BWR RPV studs for a postulated flaw after 37.9 years of assumed loading cycles remains less than the allowable flaw size using methods consistent with ASME Code, Section XI, Nonmandatory Appendix G

Application of the EPRI Report

Section 5.2 of the EPRI report provides criteria to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the closure studs. These criteria, and the applicability evaluation for each plant, are provided in Attachments 2 through 8. The GALL-SLR values for measured ultimate tensile strength and yield strength, as applicable, are also evaluated in these Attachments.

<u>Conclusion</u>

This attachment summarizes the EPRI report for the closure studs. The applicability of the EPRI report to each of the plants included in this Request for Alternative is provided in Attachments 2 through 8.

<u>REFERENCES:</u>

1. *Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs*. EPRI, Palo Alto, CA: 2018. 3002014589.
2. U.S. NRC, *Generic Aging Lessons Learned (GALL) Report,* NUREG-1801, Rev. 2, December 2010.
3. U.S. NRC, *Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report*, NUREG-2191, July 2017.
4. Welding Research Council Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," August 1972.
5. Seeley, R.R. et al., "Fracture Toughness Properties of SA-540 Steels for Nuclear Bolting Applications," Journal of Pressure Vessel Technology, August 1977.

10 CFR 50.55a REQUEST FOR ALTERNATIVE
Attachment 2
Plant-Specific Information for: Brunswick Unit 1

This attachment provides a plant-specific assessment of the criteria in Section 5.2 of the EPRI report and the plant-specific historical use of thread lubricants to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the closure studs at Brunswick Unit 1. This assessment is as follows:

Historical Lubricant Use: The current Reactor Vessel Closure Stud program includes preventive measures as recommended in Regulatory Guide 1.65, Rev. 1, *Materials and Inspections for Reactor Vessel Closure Studs*, to use stable lubricants (i.e., prohibit the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide).

- Yes; Acceptable.

Assessment Criteria:

1. The reactor vessel head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (9’-1 15/16" / 3 5/8”) = 30.33 < 34.8 (BWR); Acceptable.

2. The reactor vessel head inner radius to stud diameter ratio is smaller than 22.4 for BWRs and 14.0 for PWRs:

    - Value = (9’-1 15/16" / 5 1/2”) = 19.99 < 22.4 (BWR); Acceptable.

3. The applicable transients for the reactor vessel closure are bounded by the transients shown in Appendix A of the EPRI report, and the number of transients projected through the end of the applicable operating period is less than the number of transients identified in Section 4.3.1 of the EPRI report. These criteria are satisfied based on the following:

    a. The maximum annual number of RPV head de-tensioning events is not greater than 1 per year:

    Acceptable. Brunswick Unit 1 operates on 24-month fuel cycles, and the head is de-tensioned once during each refueling outage. Therefore, the number of cycles per year for RPV head de-tensioning events is 1 cycle/2 years = 0.50 cycle/year.

    b. The maximum annual number of passes for any single stud during an RPV head tensioning event is not greater than 4 per year.

    Acceptable.

    - The RPV Stud tensioning procedure at Brunswick Unit 1 is a two-pass process in which the studs are normally tensioned a maximum of two times. The procedure also allows for trim passes in which an individual stud is tensioned to allow adjustment of the nut to bring the stud elongation within the desired range. Additionally, the procedure allows for studs to be re-tensioned in the case that an adjacent nut or tensioning tool is stuck.

- Although the procedures do not limit the maximum number of times hydraulic pressure is applied to an individual stud, it is unlikely that an individual stud would be tensioned more than 8 times during a refueling outage. Brunswick Unit 1 operates on 24-month fuel cycles; therefore, the studs are tensioned and de-tensioned 8 cycles/2 years = 4 cycles/year.

c. All plant heatup events maintain an average linear RPV fluid temperature increase that does not exceed 100°F/hr.

   Acceptable. Brunswick Unit 1 Technical Specification 3.4.9 requires all heatup events to maintain within the Limits of the Pressure and Temperature Limits Report (PTLR). From the PTLR, for Normal Operational conditions, the RCS heat-up and cool-down rates are limited to ≤100ºF/hour in any 1-hour period.

d. All planned plant cooldown events maintain an average linear RPV fluid temperature decrease that does not exceed 100°F/hr.

   Acceptable. Brunswick Unit 1 Technical Specification 3.4.9 requires all cooldown events to maintain within the Limits of the PTLR. From the PTLR, for Normal Operational conditions, the RCS heat-up and cool-down rates are limited to ≤100ºF/hour in any 1-hour period.

e. The RPV fluid temperature for any unplanned rapid cooldown events that exceed 100°F/hr do not exceed a temperature drop of 100°F in less than 30 seconds.

   Acceptable. From the PTLR, the Safety Valve (SRV) Blowdown thermal transient event from the RPV thermal cycle diagram (TCD), has the maximum cooldown rate of 954°F/hr (15ºF/min) and is the limiting Service Level A/B event. This is lower than 100°F in 30 seconds (12,000°F/hr).

f. The sum of the counts of all thermal transients tracked by the plant Fatigue Management Program does not exceed 1,000 events per year.

   Acceptable. The Update of RPV Component Fatigue Cumulative Usage Factor (CUF) calculation for Brunswick Unit 1 includes a listing of past and postulated future transients that impact the closure studs on the RPV. For Brunswick Unit 1 the calculation postulates a total of 633 transients for a 60-year service life, which is less than 1,000.

4. All reactor vessel closure studs remain in service and are successfully tensioned:

   - Yes, all studs are currently in service and successfully tensioned; Acceptable.

5. All reactor vessel closure studs are fabricated from material with an ultimate strength of less than or equal to 170 ksi (existing studs) or a yield strength of less than or equal to 150 ksi (newly installed studs):

   - The original 132 studs provided for Brunswick Units 1 and 2 (64 for Unit 1, 64 for Unit 2 and 4 spares) were all fabricated from the same heat of material. Tensile testing documented on the Certified Material Test Report (CMTR) consisted of 18 specimens. The 0.2% offset Yield Strengths for the 18 specimens ranged from 142,250 psi to 152,750 psi. The average of the 18 tests was 145,875 psi. Only two of the specimens exhibited yield strengths in excess of the 150 ksi (Specimen 44 = 150,250 psi and Bar 44-1 = 152,750 psi), which is not significantly in excess of 150 ksi.

- GALL-SLR Section XI.M3, "Reactor Head Closure Stud Bolting" allows bolting material for closure studs that has an actual measured yield strength of less than 150 ksi for newly installed studs, or 170 ksi ultimate tensile strength for existing studs.
- All existing Brunswick Unit 1 studs have an ultimate tensile strength of less than 170 ksi; Acceptable
- The CMTRs associated with Purchase Order 8Q6836AF for replacement RPV studs note that actual measured yield strengths are less than 145 ksi; Acceptable.

6. Reactor vessel closure studs are specified as SA-540 Grades B23 or B24 material, or closure stud material specification is consistent with all SA-540 Grade B23/B24 requirements:

    - Per the Chicago Bridge and Iron fabrication drawings for Brunswick Unit 1, the Closure Flange Bushing, Stud, Nut and Washer are all constructed from SA-540 Grade B23 or B24 material; Acceptable.

7. No leakage from the reactor vessel closure flange has been observed since the most recent volumetric/surface examination:

    - No unacceptable leakage has been observed from the reactor vessel closure flange during the RPV ASME Section XI Pressure Test, since the most recent volumetric examination of the RPV closure studs; Acceptable.

This attachment provides a plant-specific assessment of the criteria in Section 5.2 of the EPRI report and the plant-specific historical use of thread lubricants to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the closure studs at Brunswick Unit 2. This assessment is as follows:

Historical Lubricant Use: The current Reactor Vessel Closure Stud program includes preventive measures as recommended in Regulatory Guide 1.65, Rev. 1, *Materials and Inspections for Reactor Vessel Closure Studs*, to use stable lubricants (i.e., prohibit the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide).

- Yes; Acceptable.

Assessment Criteria:

1. The reactor vessel head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (9’-1 15/16” / 3 5/8”) = 30.33 < 34.8 (BWR); Acceptable.

2. The reactor vessel head inner radius to stud diameter ratio is smaller than 22.4 for BWRs and 14.0 for PWRs:

    - Value = (9’-1 15/16” / 5 ½”) = 19.99 < 22.4 (BWR); Acceptable.

3. The applicable transients for the reactor vessel closure are bounded by the transients shown in Appendix A of the EPRI report, and the number of transients projected through the end of the applicable operating period is less than the number of transients identified in Section 4.3.1 of the EPRI report. These criteria are satisfied based on the following:

    a. The maximum annual number of RPV head de-tensioning events is not greater than 1 per year:

       Acceptable. Brunswick Unit 2 operates on 24-month fuel cycles, and the head is de-tensioned once during each refueling outage. Therefore, the number of cycles per year for RPV head de-tensioning events is 1 cycle/2 years = 0.50 cycle/year.

    b. The maximum annual number of passes for any single stud during an RPV head tensioning event is not greater than 4 per year.

       Acceptable.

       - The RPV Stud tensioning procedure at Brunswick Unit 2 is a two-pass process in which the studs are normally tensioned a maximum of two times. The procedure also allows for trim passes in which an individual stud is tensioned to allow adjustment of the nut to bring the stud elongation within the desired range. Additionally, the procedure allows for studs to be re-tensioned in the case that an adjacent nut or tensioning tool is stuck.

- Although the procedures do not limit the maximum number of times hydraulic pressure is applied to an individual stud, it is unlikely that an individual stud would be tensioned more than 8 times during a Refueling outage. Brunswick Unit 2 operates on 24-month fuel cycles; therefore, the studs are tensioned and de-tensioned 8 cycles/2 years = 4 cycles/year.

c. All plant heatup events maintain an average linear RPV fluid temperature increase that does not exceed 100°F/hr.

Acceptable. Brunswick Unit 2 Technical Specification 3.4.9 requires all heatup events to maintain within the Limits of the Pressure and Temperature Limits Report (PTLR). From the PTLR, for Normal Operational conditions, the RCS heat-up and cool-down rates are limited to ≤100ºF/hour in any 1-hour period.

d. All planned plant cooldown events maintain an average linear RPV fluid temperature decrease that does not exceed 100°F/hr.

Acceptable. Brunswick Unit 2 Technical Specification 3.4.9 requires all cooldown events to maintain within the Limits of the PTLR. From the PTLR, for Normal Operational conditions, the RCS heat-up and cool-down rates are limited to ≤100ºF/hour in any 1-hour period.

e. The RPV fluid temperature for any unplanned rapid cooldown events that exceed 100°F/hr do not exceed a temperature drop of 100°F in less than 30 seconds.

Acceptable. From the PTLR, the Safety Valve (SRV) Blowdown thermal transient event from the RPV thermal cycle diagram (TCD), has the maximum cooldown rate of 954°F/hr (15ºF/min) and is the limiting Service Level A/B event. This is lower than 100°F in 30 seconds (12,000°F/hr).

f. The sum of the counts of all thermal transients tracked by the plant Fatigue Management Program does not exceed 1,000 events per year.

Acceptable. The Update of RPV Component Fatigue Cumulative Usage Factor (CUF) calculation for Brunswick Unit 2 includes a listing of past and postulated future transients that impact the closure studs on the RPV. For Brunswick Unit 2 the calculation postulates a total of 742 transients for a 60-year service life, which is less than 1,000.

4. All reactor vessel closure studs remain in service and are successfully tensioned:

   - Yes, all studs are currently in service and successfully tensioned; Acceptable.

5. All reactor vessel closure studs are fabricated from material with an ultimate strength of less than or equal to 170 ksi (existing studs) or a yield strength of less than or equal to 150 ksi (newly installed studs):

   - The original 132 studs provided for Brunswick Units 1 and 2 (64 for Unit 1, 64 for Unit 2 and 4 spares) were all fabricated from the same heat of material. Tensile testing documented on the CMTR consisted of 18 specimens. The 0.2% offset Yield Strengths for the 18 specimens ranged from 142,250 psi to 152,750 psi. The average of the 18 tests was 145,875 psi. Only 2 of the specimens exhibited yield strengths in excess of the 150 ksi (Specimen 44 = 150,250 psi and Bar 44-1 = 152,750 psi), which is not significantly in excess of 150 ksi.

- GALL-SLR XI.M3, "Reactor Head Closure Stud Bolting" allows bolting material for closure studs that has an actual measured yield strength of less than 150 ksi for newly installed studs, or 170 ksi ultimate tensile strength for existing studs.
- All existing Brunswick Unit 2 studs have an ultimate tensile strength of less than 170 ksi; Acceptable.
- The CMTRs associated with Purchase Order 8Q6836AF for replacement RPV studs note that actual measured yield strengths are less than 145 ksi; Acceptable.

6. Reactor vessel closure studs are specified as SA-540 Grades B23 or B24 material, or closure stud material specification is consistent with all SA-540 Grade B23/B24 requirements:

    - Per the Chicago Bridge and Iron fabrication drawings for Brunswick Unit 2, the Closure Flange Bushing, Stud, Nut and Washer are all constructed from SA-540 Grade B23 or B24 material; Acceptable.

7. No leakage from the reactor vessel closure flange has been observed since the most recent volumetric/surface examination:

    - No unacceptable leakage has been observed from the reactor vessel closure flange during the RPV ASME Section XI Pressure Tests, since the most recent volumetric examination of the RPV closure studs; Acceptable.

This attachment provides a plant-specific assessment of the criteria in Section 5.2 of the EPRI report and the plant-specific historical use of thread lubricants to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the closure studs at Catawba Unit 1. This assessment is as follows:

Historical Lubricant Use: The current Reactor Vessel Closure Stud program includes preventive measures as recommended in Regulatory Guide 1.65, Rev. 1, *Materials and Inspections for Reactor Vessel Closure Studs*, to use stable lubricants (i.e., prohibit the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide).

- Yes; Acceptable.

Assessment Criteria:

1. The reactor vessel head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

   - Value = (88.346" / 6.496") = 13.60 < 14.5 (PWR); Acceptable.

2. The reactor vessel head inner radius to stud diameter ratio is smaller than 22.4 for BWRs and 14.0 for PWRs:

   - Value = (88.346" / 6.772") = 13.05 < 14.0 (PWR); Acceptable.

3. The applicable transients for the reactor vessel closure are bounded by the transients shown in Appendix A of the EPRI report, and the number of transients projected through the end of the applicable operating period is less than the number of transients identified in Section 4.3.1 of the EPRI report. These criteria are satisfied based on the following:

   a. The maximum annual number of RPV head de-tensioning events is not greater than 1 per year:

      Acceptable. Catawba Unit 1 operates on 18-month fuel cycles, and the head is de-tensioned once during each refueling outage. Therefore, the number of cycles per year for RPV head de-tensioning events is 1 cycle/1.5 years = 0.67 cycle/year.

   b. The maximum annual number of passes for any single stud during an RPV head tensioning event is not greater than 4 per year.

      Acceptable. The tensioning and de-tensioning procedures at Catawba Unit 1 both specify less than two full tensioning passes. Therefore, each stud is tensioned/de-tensioned no more than 4 times every head removal cycle. Catawba Unit 1 operates on 18-month fuel cycles; therefore, the studs are tensioned and de-tensioned 4 cycles/1.5 years = 2.67 cycles/year.

c. All plant heatup events maintain an average linear RPV fluid temperature increase that does not exceed 100°F/hr.

   Acceptable. Catawba Unit 1 Technical Specification 3.4.3 requires all heatup events to maintain 100 degrees F/hour or less.

d. All planned plant cooldown events maintain an average linear RPV fluid temperature decrease that does not exceed 100°F/hr.

   Acceptable. Catawba Unit 1 Technical Specification 3.4.3 requires all cooldown events to maintain 100 degrees F/hour or less.

e. The RPV fluid temperature for any unplanned rapid cooldown events that exceed 100°F/hr do not exceed a temperature drop of 100°F in less than 30 seconds.

   Acceptable. For the non-faulted condition transients, the temperature versus time variation for the Excessive Feedwater Flow transient is bounding compared to other upset transients. The Excessive Feedwater Flow transient results in a cold leg temperature variation of 125°F within 50 seconds (2.50°F/sec) and satisfies the above temperature rate limit of 100°F within 30 seconds (3.33°F/sec).

f. The sum of the counts of all thermal transients tracked by the plant Fatigue Management Program does not exceed 1,000 events per year.

   Acceptable. The number of actual occurrences of reactor coolant system transients as monitored in the Catawba thermal fatigue management program is significantly under 1000 events/year. Plant heatup, cooldown transients and stud tensioning/de-tensioning activities are critical with respect to fatigue effects on the studs. The occurrences of these events as well as other reactor coolant system transients over plant life are significantly lower than the 1000 events/year limit.

4. All reactor vessel closure studs remain in service and are successfully tensioned:

   - Yes, all studs are currently in service and successfully tensioned; Acceptable.

5. All reactor vessel closure studs are fabricated from material with an ultimate strength of less than or equal to 170 ksi (existing studs) or a yield strength of less than or equal to 150 ksi (newly installed studs):

   - Yes, all existing studs have a yield strength of less than 150 ksi; Acceptable.

6. Reactor vessel closure studs are specified as SA-540 Grades B23 or B24 material, or closure stud material specification is consistent with all SA-540 Grade B23/B24 requirements:

   - Studs are specified as SA-540 Grade B24; Acceptable.

7. No leakage from the reactor vessel closure flange has been observed since the most recent volumetric/surface examination:

   - No unacceptable leakage has been observed from the reactor vessel closure flange during the RPV ASME Section XI Pressure Tests, since the most recent volumetric examination of the RPV closure studs; Acceptable.

This attachment provides a plant-specific assessment of the criteria in Section 5.2 of the EPRI report and the plant-specific historical use of thread lubricants to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the closure studs at Catawba Unit 2. This assessment is as follows:

Historical Lubricant Use: The current Reactor Vessel Closure Stud program includes preventive measures as recommended in Regulatory Guide 1.65, Rev. 1, *Materials and Inspections for Reactor Vessel Closure Studs*, to use stable lubricants (i.e., prohibit the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide).

- Yes; Acceptable.

Assessment Criteria:

1. The reactor vessel head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (86.000" / 7.000") = 12.29 < 14.5 (PWR); Acceptable.

2. The reactor vessel head inner radius to stud diameter ratio is smaller than 22.4 for BWRs and 14.0 for PWRs:

    - Value = (86.000" / 6.822") = 12.61 < 14.0 (PWR); Acceptable.

3. The applicable transients for the reactor vessel closure are bounded by the transients shown in Appendix A of the EPRI report, and the number of transients projected through the end of the applicable operating period is less than the number of transients identified in Section 4.3.1 of the EPRI report. These criteria are satisfied based on the following:

    a. The maximum annual number of RPV head de-tensioning events is not greater than 1 per year:

       Acceptable. Catawba Unit 2 operates on 18-month fuel cycles, and the head is de-tensioned once during each refueling outage. Therefore, the number of cycles per year for RPV head de-tensioning events is 1 cycle/1.5 years = 0.67 cycle/year.

    b. The maximum annual number of passes for any single stud during an RPV head tensioning event is not greater than 4 per year.

       Acceptable. The tensioning and de-tensioning procedures at Catawba Unit 2 both specify less than two full tensioning passes. Therefore, each stud is tensioned/de-tensioned no more than 4 times every head removal cycle. Catawba Unit 2 operates on 18-month fuel cycles; therefore, the studs are tensioned and de-tensioned 4 cycles/1.5 years = 2.67 cycles/year.

c. All plant heatup events maintain an average linear RPV fluid temperature increase that does not exceed 100°F/hr.

   Acceptable. Catawba Unit 2 Technical Specification 3.4.3 requires all heatup events to maintain 100 degrees F/hour or less.

d. All planned plant cooldown events maintain an average linear RPV fluid temperature decrease that does not exceed 100°F/hr.

   Acceptable. Catawba Unit 2 Technical Specification 3.4.3 requires all cooldown events to maintain 100 degrees F/hour or less.

e. The RPV fluid temperature for any unplanned rapid cooldown events that exceed 100°F/hr do not exceed a temperature drop of 100°F in less than 30 seconds.

   Acceptable. For the non-faulted condition transients, the temperature versus time variation for the Excessive Feedwater Flow transient is bounding compared to other upset transients. The Excessive Feedwater Flow transient results in a cold leg temperature variation of 125°F within 50 seconds (2.50°F/sec) and satisfies the above temperature rate limit of 100°F within 30 seconds (3.33°F/sec).

f. The sum of the counts of all thermal transients tracked by the plant Fatigue Management Program does not exceed 1,000 events per year.

   Acceptable. The number of actual occurrences of reactor coolant system transients as monitored in the Catawba thermal fatigue management program is significantly under 1000 events/year. Plant heatup, cooldown transients and stud tensioning/de-tensioning activities are critical with respect to fatigue effects on the studs. The occurrences of these events as well as other reactor coolant system transients over plant life are significantly lower than the 1000 events/year limit.

4. All reactor vessel closure studs remain in service and are successfully tensioned:

   - Yes, all studs are currently in service and successfully tensioned; Acceptable.

5. All reactor vessel closure studs are fabricated from material with an ultimate strength of less than or equal to 170 ksi (existing studs) or a yield strength of less than or equal to 150 ksi (newly installed studs):

   - Tensile testing documented on the Catawba Unit 2 Closure Head Stud Material CMTR consisted of 20 specimens. The 0.2% offset Yield Strengths for the 20 specimens ranged from 138 ksi to 153.5 ksi. The average of the 20 tests was 145.14 ksi. Only 2 of the specimens exhibited yield strengths in excess of the 150 ksi (Bar No. 146 = 153.5 ksi and Bar No. 153 = 152 ksi), which is not significantly in excess of 150 ksi.
   - GALL-SLR Section XI.M3, "Reactor Head Closure Stud Bolting" allows bolting material for closure studs that has an actual measured yield strength of less than 150 ksi for newly installed studs, or 170 ksi ultimate tensile strength for existing studs.

- All existing Catawba Unit 2 studs have an ultimate tensile strength of less than 170 ksi; Acceptable.

6. Reactor vessel closure studs are specified as SA-540 Grades B23 or B24 material, or closure stud material specification is consistent with all SA-540 Grade B23/B24 requirements:

    - Studs are specified as SA-540 Grade B24; Acceptable.

7. No leakage from the reactor vessel closure flange has been observed since the most recent volumetric/surface examination:

    - No unacceptable leakage has been observed from the reactor vessel closure flange during the RPV ASME Section XI Pressure Tests, since the most recent volumetric examination of the RPV closure studs; Acceptable.

This attachment provides a plant-specific assessment of the criteria in Section 5.2 of the EPRI report and the plant-specific historical use of thread lubricants to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the closure studs at McGuire Unit 1. This assessment is as follows:

Historical Lubricant Use: The current Reactor Vessel Closure Stud program includes preventive measures as recommended in Regulatory Guide 1.65, Rev. 1, *Materials and Inspections for Reactor Vessel Closure Studs*, to use stable lubricants (i.e., prohibit the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide).

- Yes; Acceptable.

Assessment Criteria:

1. The reactor vessel head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (86.000" / 7.000") = 12.29 < 14.5 (PWR); Acceptable.

2. The reactor vessel head inner radius to stud diameter ratio is smaller than 22.4 for BWRs and 14.0 for PWRs:

    - Value = (86.000" / 6.797") = 12.65 < 14.0 (PWR); Acceptable.

3. The applicable transients for the reactor vessel closure are bounded by the transients shown in Appendix A of the EPRI report, and the number of transients projected through the end of the applicable operating period is less than the number of transients identified in Section 4.3.1 of the EPRI report. These criteria are satisfied based on the following:

    a. The maximum annual number of RPV head de-tensioning events is not greater than 1 per year:

    Acceptable. McGuire Unit 1 operates on 18-month fuel cycles, and the head is de-tensioned once during each refueling outage. Therefore, the number of cycles per year for RPV head de-tensioning events is 1 cycle/1.5 years = 0.67 cycle/year.

    b. The maximum annual number of passes for any single stud during an RPV head tensioning event is not greater than 4 per year.

    Acceptable. The tensioning and de-tensioning procedure at McGuire Unit 1 specifies less than two full tensioning passes. Therefore, each stud is tensioned/de-tensioned no more than 4 times every head removal cycle. McGuire Unit 1 operates on 18-month fuel cycles; therefore, the studs are tensioned and de-tensioned 4 cycles/1.5 years = 2.67 cycles/year.

c. All plant heatup events maintain an average linear RPV fluid temperature increase that does not exceed 100°F/hr.

   Acceptable. McGuire Unit 1 Technical Specification 3.4.3 requires all heatup events to maintain 100 degrees F/hour or less.

d. All planned plant cooldown events maintain an average linear RPV fluid temperature decrease that does not exceed 100°F/hr.

   Acceptable. McGuire Unit 1 Technical Specification 3.4.3 requires all cooldown events to maintain 100 degrees F/hour or less.

e. The RPV fluid temperature for any unplanned rapid cooldown events that exceed 100°F/hr do not exceed a temperature drop of 100°F in less than 30 seconds.

   Acceptable. For the non-faulted condition transients, the temperature versus time variation for the Loss of Flow in One Loop transient is bounding compared to other upset transients. The Loss of Flow in One Loop transient results in a cold leg temperature variation of 45°F over 15 seconds and satisfies the above temperature rate limit.

f. The sum of the counts of all thermal transients tracked by the plant Fatigue Management Program does not exceed 1,000 events per year.

   Acceptable. The number of actual occurrences of reactor coolant system transients as monitored in the McGuire thermal fatigue management program is significantly under 1000 events/year. Plant heatup, cooldown transients and stud tensioning/de-tensioning activities are critical with respect to fatigue effects on the studs. The occurrences of these events as well as other reactor coolant system transients over plant life are significantly lower than the 1000 events/year limit.

4. All reactor vessel closure studs remain in service and are successfully tensioned:

   - Yes, all studs are currently in service and successfully tensioned; Acceptable.

5. All reactor vessel closure studs are fabricated from material with an ultimate strength of less than or equal to 170 ksi (existing studs) or a yield strength of less than or equal to 150 ksi (newly installed studs):

   - Tensile testing documented on the McGuire Unit 1 Closure Head Stud Material CMTR consisted of 16 specimens. The 0.2% offset Yield Strengths for the 16 specimens ranged from 140 ksi to 159 ksi. The average of the 16 tests was 146.66 ksi. Only 4 of the specimens exhibited yield strengths in excess of the 150 ksi (Heat No. 66593: Bar No. 282 = 155 ksi, Bar No. 283 = 159 ksi, Bar No. 287 = 157.5 ksi, and Bar No. 288 = 152 ksi), which is not significantly in excess of 150 ksi.
   - GALL-SLR XI.M3, "Reactor Head Closure Stud Bolting" allows bolting material for closure studs that has an actual measured yield strength of less than 150 ksi for newly installed studs, or 170 ksi ultimate tensile strength for existing studs.

- All existing McGuire Unit 1 studs have an ultimate tensile strength of less than or equal to 170 ksi; Acceptable.

6. Reactor vessel closure studs are specified as SA-540 Grades B23 or B24 material, or closure stud material specification is consistent with all SA-540 Grade B23/B24 requirements:

   - Studs are specified as SA-540 Grade B24; Acceptable.

7. No leakage from the reactor vessel closure flange has been observed since the most recent volumetric/surface examination:

   - No unacceptable leakage has been observed from the reactor vessel closure flange during the RPV ASME Section XI Pressure Tests, since the most recent volumetric examination of the RPV closure studs; Acceptable.

This attachment provides a plant-specific assessment of the criteria in Section 5.2 of the EPRI report and the plant-specific historical use of thread lubricants to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the closure studs at McGuire Unit 2. This assessment is as follows:

Historical Lubricant Use: The current Reactor Vessel Closure Stud program includes preventive measures as recommended in Regulatory Guide 1.65, Rev. 1, *Materials and Inspections for Reactor Vessel Closure Studs*, to use stable lubricants (i.e., prohibit the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide).

- Yes; Acceptable.

Assessment Criteria:

1. The reactor vessel head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (88.346" / 6.496") = 13.60 < 14.5 (PWR); Acceptable.

2. The reactor vessel head inner radius to stud diameter ratio is smaller than 22.4 for BWRs and 14.0 for PWRs:

    - Value = (88.346" / 6.772") = 13.05 < 14.0 (PWR); Acceptable.

3. The applicable transients for the reactor vessel closure are bounded by the transients shown in Appendix A of the EPRI report, and the number of transients projected through the end of the applicable operating period is less than the number of transients identified in Section 4.3.1 of the EPRI report. These criteria are satisfied based on the following:

    a. The maximum annual number of RPV head de-tensioning events is not greater than 1 per year:

    Acceptable. McGuire Unit 2 operates on 18-month fuel cycles, and the head is de-tensioned once during each refueling outage. Therefore, the number of cycles per year for RPV head de-tensioning events is 1 cycle/1.5 years = 0.67 cycle/year.

    b. The maximum annual number of passes for any single stud during an RPV head tensioning event is not greater than 4 per year.

    Acceptable. The tensioning and de-tensioning procedures at McGuire Unit 2 both specify less than two full tensioning passes. Therefore, each stud is tensioned/de-tensioned no more than 4 times every head removal cycle. McGuire Unit 2 operates on 18-month fuel cycles; therefore, the studs are tensioned and de-tensioned 4 cycles/1.5 years = 2.67 cycles/year.

c. All plant heatup events maintain an average linear RPV fluid temperature increase that does not exceed 100°F/hr.

   Acceptable. McGuire Unit 2 Technical Specification 3.4.3 requires all heatup events to maintain 100 degrees F/hour or less.

d. All planned plant cooldown events maintain an average linear RPV fluid temperature decrease that does not exceed 100°F/hr.

   Acceptable. McGuire Unit 2 Technical Specification 3.4.3 requires all cooldown events to maintain 100 degrees F/hour or less.

e. The RPV fluid temperature for any unplanned rapid cooldown events that exceed 100°F/hr do not exceed a temperature drop of 100°F in less than 30 seconds.

   Acceptable. For the non-faulted condition transients, the temperature versus time variation for the Loss of Flow in One Loop transient is bounding compared to other upset transients. The Loss of Flow in One Loop transient results in a cold leg temperature variation of 45°F over 15 seconds and satisfies the above temperature rate limit.

f. The sum of the counts of all thermal transients tracked by the plant Fatigue Management Program does not exceed 1,000 events per year.

   Acceptable. The number of actual occurrences of reactor coolant system transients as monitored in the McGuire thermal fatigue management program is significantly under 1000 events/year. Plant heatup, cooldown transients and stud tensioning/de-tensioning activities are critical with respect to fatigue effects on the studs. The occurrences of these events as well as other reactor coolant system transients over plant life are significantly lower than the 1000 events/year limit.

4. All reactor vessel closure studs remain in service and are successfully tensioned:

   - Yes, all studs are currently in service and successfully tensioned; Acceptable.

5. All reactor vessel closure studs are fabricated from material with an ultimate strength of less than or equal to 170 ksi (existing studs) or a yield strength of less than or equal to 150 ksi (newly installed studs):

   - Yes, all existing studs have a yield strength of less than 150 ksi; Acceptable.

6. Reactor vessel closure studs are specified as SA-540 Grades B23 or B24 material, or closure stud material specification is consistent with all SA-540 Grade B23/B24 requirements:

   - Studs are specified as SA-540 Grade B24; Acceptable.

7. No leakage from the reactor vessel closure flange has been observed since the most recent volumetric/surface examination:

   - No unacceptable leakage has been observed from the reactor vessel closure flange during the RPV ASME Section XI Pressure Tests, since the most recent volumetric examination of the RPV closure studs; Acceptable.

This attachment provides a plant-specific assessment of the criteria in Section 5.2 of the EPRI report and the plant-specific historical use of thread lubricants to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the closure studs at Shearon Harris Unit 1. This assessment is as follows:

Historical Lubricant Use: The current Reactor Vessel Closure Stud program includes preventive measures as recommended in Regulatory Guide 1.65, Rev. 1, *Materials and Inspections for Reactor Vessel Closure Studs*, to use stable lubricants (i.e., prohibit the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide).

- Yes; Acceptable.

Assessment Criteria:

1. The reactor vessel head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (74.71" / 5.75") = 12.99 < 14.5 (PWR); Acceptable.

2. The reactor vessel head inner radius to stud diameter ratio is smaller than 22.4 for BWRs and 14.0 for PWRs:

    - Value = (74.71" / 5.8162") = 12.84 < 14.0 (PWR); Acceptable.

3. The applicable transients for the reactor vessel closure are bounded by the transients shown in Appendix A of the EPRI report, and the number of transients projected through the end of the applicable operating period is less than the number of transients identified in Section 4.3.1 of the EPRI report. These criteria are satisfied based on the following:

    a. The maximum annual number of RPV head de-tensioning events is not greater than 1 per year:

    Acceptable. Shearon Harris operates on 18-month fuel cycles, and the head is de-tensioned once during each refueling outage. Therefore, the number of cycles per year for RPV head de-tensioning events is 1 cycle/1.5 years = 0.67 cycle/year.

    b. The maximum annual number of passes for any single stud during an RPV head tensioning event is not greater than 4 per year.

    Acceptable. The tensioning and de-tensioning procedures at Shearon Harris both specify less than two full tensioning passes. Therefore, each stud is tensioned/de-tensioned no more than 4 times every head removal cycle. Shearon Harris operates on 18-month fuel cycles; therefore, the studs are tensioned and de-tensioned 4 cycles/1.5 years = 2.67 cycles/year.

c. All plant heatup events maintain an average linear RPV fluid temperature increase that does not exceed 100°F/hr.

   Acceptable. Shearon Harris Technical Specification 3.4.9.1 requires all heatup events to maintain 100 degrees F/hour or less.

d. All planned plant cooldown events maintain an average linear RPV fluid temperature decrease that does not exceed 100°F/hr.

   Acceptable. Shearon Harris Technical Specification 3.4.9.1 requires all cooldown events to maintain 100 degrees F/hour or less.

e. The RPV fluid temperature for any unplanned rapid cooldown events that exceed 100°F/hr do not exceed a temperature drop of 100°F in less than 30 seconds.

   Acceptable. For the non-faulted condition transients, the temperature versus time variation for the Excessive Feedwater Flow transient is bounding compared to the other transients. The Excessive Feedwater Flow transient results in a cold leg temperature variation of 99°F over 27 seconds. This transient is bounded by PWR transients provided in the EPRI report.

f. The sum of the counts of all thermal transients tracked by the plant Fatigue Management Program does not exceed 1,000 events per year.

   Acceptable. A conservative estimate for the number of transients for the Shearon Harris RPV studs may be obtained using the total number of the "Cyclic or Transient Limits" for all transients counted in the Fatigue Management Program (OMM-0013), regardless of whether those transients impact the RPV head region. That total is 7,058 for the 60-year licensed life, which represents 7,058/60 years = 118 cycles/year, which is less than 1,000. (Note: This total conservatively includes 400 cycles for Tube Leakage Test).

4. All reactor vessel closure studs remain in service and are successfully tensioned:

   - Yes, all studs are currently in service and successfully tensioned; Acceptable.

5. All reactor vessel closure studs are fabricated from material with an ultimate strength of less than or equal to 170 ksi (existing studs) or a yield strength of less than or equal to 150 ksi (newly installed studs):

   - Tensile testing documented on the originally supplied Shearon Harris Closure Head Stud Material CMTR consisted of 16 specimens. The 0.2% offset Yield Strengths for the 16 specimens ranged from 137.5 ksi to 151.2 ksi. The average of the 16 tests was 145.96 ksi. Only 3 of the specimens exhibited yield strengths in excess of the 150 ksi (Heat No. 81401: Bar No. 86 = 151.2 ksi, Bar No. 91-1 = 151 ksi, and Heat No. 80751: Bar No. 75-1 = 150.2 ksi), which is not significantly in excess of 150 ksi.

- Three original studs were replaced in 2000 via WR/JO 97-AJQK1. The ASME Code Manufacturers' Data Report Form for Nuclear Vessels N-1A provided in the WR/JO, identifies the studs manufactured by CBI Nuclear Company contract 71-2633 and Westinghouse Electric Corporation (WAPD No. 150118). From Westinghouse P.O. No. W150118, the replacement studs are from the Unit 2 reactor vessel. Tensile testing for all of the Unit 2 Shearon Harris Closure Stud Material CMTR consist of 18 specimens. The 0.2% offset Yield Strengths for the 18 specimens ranged from 135.0 ksi to 151.5 ksi. The average of the 18 tested was 142.26 ksi. Only 2 of the specimens exhibited yield strengths in excess of 150 ksi (Heat No. 80751: Bar No. 75-1 = 150.25 ksi and Heat No. 80726: Bar No. 158 = 151.5 ksi) which is not significantly in excess of 150 ksi.
- GALL-SLR XI.M3, "Reactor Head Closure Stud Bolting" allows bolting material for closure studs that has an actual measured yield strength of less than 150 ksi for newly installed studs, or 170 ksi ultimate tensile strength for existing studs.
- All existing Shearon Harris studs have an ultimate tensile strength of less than 170 ksi; Acceptable.

6. Reactor vessel closure studs are specified as SA-540 Grades B23 or B24 material, or closure stud material specification is consistent with all SA-540 Grade B23/B24 requirements:

   - Studs are specified as SA-540 Grade B24; Acceptable.

7. No leakage from the reactor vessel closure flange has been observed since the most recent volumetric/surface examination:

   - No unacceptable leakage has been observed from the reactor vessel closure flange during the RPV ASME Section XI Pressure Test, since the most recent volumetric examination of the RPV closure studs; Acceptable.