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NMP1L3612
RS-24-126

10 CFR 50.55a

June 16, 2025

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

Braidwood Station, Unit 1
Renewed Facility Operating License No. NPF-72
NRC Docket No. STN 50-456

Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

James A. FitzPatrick Nuclear Power Plant
Renewed Facility Operating License No. DPR-59
NRC Docket No. 50-333

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF- 18
NRC Docket Nos. 50-373 and 50-374

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Nine Mile Point Nuclear Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-63 and NPF-69
NRC Docket Nos. 50-220 and 50-410

Peach Bottom Atomic Power Station, Units 2 and 3
Subsequent Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Relief Request for Alternative to Examination Category B-G-1, Item B6.20, Reactor Vessel Closure Stud Examinations

In accordance with 10 CFR 50.55a(z)(1), Constellation Energy Generation, LLC (CEG), requests 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, Reactor Vessel Closure Studs. This proposed alternative requests to extend the inspection interval for the ASME Code, Section XI, Table IWB- 2500-1, Examination Category B-G-1, Item Number B6.20 examinations for the remainder of the currently licensed operating periods and through the Sixth Inservice Inspection (ISI) interval for Peach Bottom Atomic Power Station. 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 Braidwood Station, Unit 1, Byron Station, Units 1 and 2, Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Clinton Power Station, Unit 1, James A. FitzPatrick Nuclear Power Plant, LaSalle County Station, Units 1 and 2, Limerick Generating Station, Units 1 and 2, Nine Mile Point Nuclear Station, Units 1 and 2, and through the Sixth ISI interval for Peach Bottom Atomic Power Station, Units 2 and 3.

CEG requests your review and approval of this fleet request by June 30, 2026.

There are no regulatory commitments contained in this letter.

If you have any questions concerning this letter, please contact Jesse Brown at 267-533-6355

Respectfully,

Knowles, Justin W

Digitally signed by Knowles, Justin W
Date: 2025.06.16 10:55:08 -04'00'

Justin Knowles
Sr. Manager - Licensing
Constellation Energy Generation, LLC

Enclosure: Relief Request for Alternative to Examination Category B-G-1, Item B6.20, Reactor Vessel Closure Stud Examinations

Attachment: 1) Summary of EPRI Report
2) Plant-Specific Information for: Braidwood Station, Unit 1
3) Plant-Specific Information for: Byron Station, Units 1 and 2
4) Plant-Specific Information for: Calvert Cliffs Nuclear Power Plant, Units 1 and 2
5) Plant-Specific Information for: Clinton Power Station, Unit 1
6) Plant-Specific Information for: James A. FitzPatrick Nuclear Power Plant

7) Plant-Specific Information for: LaSalle County Station, Units 1 and 2
8) Plant-Specific Information for: Limerick Generating Station, Units 1 and 2
9) Plant-Specific Information for: Nine Mile Point Nuclear Station, Unit 1
10) Plant-Specific Information for: Nine Mile Point Nuclear Station, Unit 2
11) Plant-Specific Information for: Peach Bottom Atomic Power Station, Units 2 and 3

cc: Regional Administrator - NRC Region I
Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
NRC Senior Resident Inspector - Byron Station
NRC Senior Resident Inspector - Calvert Cliffs Nuclear Power Plant
NRC Senior Resident Inspector - Clinton Power Station
NRC Senior Resident Inspector - James A. FitzPatrick Nuclear Power Plant
NRC Senior Resident Inspector - LaSalle County Station
NRC Senior Resident Inspector - Limerick Generating Station
NRC Senior Resident Inspector - Nine Mile Point Nuclear Station
NRC Senior Resident Inspector - Peach Bottom Atomic Power Station
NRC Project Manager - Braidwood Station
NRC Project Manager - Byron Station
NRC Project Manager - Calvert Cliffs Nuclear Power Plant
NRC Project Manager - Clinton Power Station
NRC Project Manager - James A. FitzPatrick Nuclear Power Plant
NRC Project Manager - LaSalle County Station
NRC Project Manager - Limerick Generating Station
NRC Project Manager - Nine Mile Point Nuclear Station
NRC Project Manager - Peach Bottom Atomic Power Station
Illinois Emergency Management Agency - Division of Nuclear Safety
D. Baracco, Commonwealth of Pennsylvania
S. Seaman, State of Maryland
A. L. Peterson, NYSERDA
B. Frymire, NYSPSC
A. Kauk, NYSPSC

bcc:	Sr. Vice President, Northeast Operations	w/o attachments
	Sr. Vice President, Mid-Atlantic Operations	“
	Sr. Vice President, Midwest Operations	“
	Vice President, Nuclear Security, Licensing and Regulated Programs	“
	Site Vice President, Braidwood Station	“
	Site Vice President, Byron Station	“
	Site Vice President, Calvert Cliffs Nuclear Power Plant	“
	Site Vice President, Clinton Power Station	“
	Site Vice President, James A. FitzPatrick Nuclear Power Plant	“
	Site Vice President, LaSalle County Station	“
	Site Vice President, Limerick Generating Station	“
	Site Vice President, Nine Mile Point Nuclear Station	“
	Site Vice President, Peach Bottom Atomic Power Station	“
	Director Licensing – KSA/CAN	with attachments
	Sr. Manager Licensing – KSA/CAN	“
	Regulatory Assurance Manager, Braidwood Station	“
	Regulatory Assurance Manager, Byron Station	“
	Regulatory Assurance Manager, Calvert Cliffs Nuclear Power Plant	“
	Regulatory Assurance Manager, Clinton Power Station	“
	Regulatory Assurance Manager, James A. FitzPatrick Nuclear Power Plant	“
	Regulatory Assurance Manager, LaSalle County Station	“
	Regulatory Assurance Manager, Limerick Generating Station	“
	Regulatory Assurance Manager, Nine Mile Point Nuclear Station	“
	Regulatory Assurance Manager, Peach Bottom Atomic Power Station	“
	Licensing Records – KSA	“

ENCLOSURE

Relief Request for Alternative to Examination Category B-G-1, Item Number B6.20, Reactor Vessel Closure Stud Examinations

Braidwood Station, Unit 1
Byron Station, Units 1 and 2
Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Clinton Power Station, Unit 1
James A. FitzPatrick Nuclear Power Plant
LaSalle County Station, Units 1 and 2
Limerick Generating Station, Units 1 and 2
Nine Mile Point Nuclear Station, Units 1 and 2
Peach Bottom Atomic Power Station, Units 2 and 3

Request for Alternative to Reactor Vessel Closure Stud Examinations in Accordance with 10 CFR 50.55a(z)(1)

1. ASME CODE COMPONENT(S) AFFECTED:

Component:	Reactor Vessel Closure Studs
Code Class:	Class 1
Examination Category:	B-G-1 (Pressure Retaining Bolting, Greater than 2 in. (50 mm) in Diameter)
Code Item Number:	B6.20 – Closure Studs
Description:	Reactor Vessel Closure Studs volumetric and surface examinations

2. 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
Braidwood Station, Unit 1	Fourth	2013 Edition	August 29, 2018	July 28, 2028
Byron Station, Units 1 and 2	Fourth	2007 Edition with the 2008 Addenda	July 16, 2016	July 15, 2025
Byron Generating Station, Units 1 and 2	Fifth	2019 Edition	July 16, 2025	July 15, 2037
Calvert Cliffs Nuclear Power Plant, Units 1 and 2	Fifth	2013 Edition	July 1, 2019	June 30, 2029
Clinton Power Station, Unit 1	Fourth	2013 Edition	July 1, 2020	June 30, 2030
James A. FitzPatrick Nuclear Power Plant	Fifth	2007 Edition with the 2008 Addenda	August 1, 2017	June 15, 2027
LaSalle County Station, Units 1 and 2	Fourth	2007 Edition with the 2008 Addenda	October 1, 2017	September 30, 2027
Limerick Generating Station, Units 1 and 2	Fourth	2007 Edition with the 2008 Addenda	February 1, 2017	January 31, 2027

Nine Mile Point Nuclear Station, Unit 1	Fifth	2013 Edition	August 23, 2019	August 22, 2029
Nine Mile Point Nuclear Station, Unit 2	Fourth	2013 Edition	August 23, 2018	August 22, 2028
Peach Bottom Atomic Power Station, Units 2 and 3	Fifth	2013 Edition	January 1, 2019	December 31, 2028

3. <u>APPLICABLE CODE REQUIREMENT:</u>

ASME Section XI IWB-2500(a), Table IWB-2500-1, Examination Category B-G-1, Item Number B6.20 requires volumetric examination of the closure studs once during each Section XI inspection interval. 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 (2007 Edition with the 2008 Addenda and 2013 Edition) or IWB-2500-12(a) (2019 Edition). 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 or IWB-2500-12(a) as applicable.

4. <u>REASON FOR REQUEST:</u>

The Electric Power Research Institute (EPRI) recently developed a technical basis report [Reference 8.1] that determined the requirement to perform inservice volumetric or surface examinations of Reactor Pressure Vessel (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. It is noted that there are negative impacts for performing the RPV stud volumetric examinations related to worker dose, personnel safety, radwaste, and critical path time. Also, examinations result in additional time at reduced Reactor Coolant System (RCS) water inventory for Boiling Water Reactors (BWRs), since the volumetric examinations are typically performed with the studs installed in the RPV flange.

The analysis methodology and the results of the EPRI report are summarized in Attachment 1. Attachments 2 through 11 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 (SCC), (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, 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. PROPOSED ALTERNATIVE AND BASIS FOR USE:

In accordance with 10 CFR 50.55a(z)(1), Constellation Energy Generation, LLC (CEG) is 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 each ISI interval. The proposed alternative is to extend the frequency of RPV closure stud volumetric or surface examination for the remainder of the currently licensed operating periods for the plants listed in Table 1 except for Peach Bottom Atomic Power station requested through the Sixth ISI interval. 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, Item Number B6.20 Examination[1]	Length of Relief Requested (years)[2]
Braidwood Station, Unit 1	07/28/28	10/17/46	09/17/13	33.1
Byron Station, Units 1 and 2	07/15/25	10/31/44 (Unit 1) 11/06/46 (Unit 2)	S01-S18: 09/19/18 (Unit 1) S19-S36: 09/21/21 (Unit 1) S37-S54: 09/22/15 (Unit 1) S01-S18: 10/07/17 (Unit 2) S19-S36: 04/28/22 (Unit 2) S37-S54: 10/14/23 (Unit 2)	29.1 (Unit 1) 29.1 (Unit 2)
Calvert Cliffs Nuclear Power Plant, Units 1 and 2	06/30/29	07/31/34 (Unit 1) 08/13/36 (Unit 2)	03/09/10 (Unit 1) 02/26/11 (Unit 2)	24.4 (Unit 1) 25.5 (Unit 2)
Clinton Power Station, Unit 1	06/30/30	04/17/27	09/18/19	7.6
James A. FitzPatrick Nuclear Power Plant	06/15/27	10/17/34	S01-S17 & S33-S52: 01/15/17 S18-S32: 09/17/18	17.8
LaSalle County Station, Units 1 and 2	09/30/27	04/17/42 (Unit 1) 12/16/43 (Unit 2)	S01-S22: 02/25/18 (Unit 1) S23-S68: 02/20/16 (Unit 1) 02/10/17 (Unit 2)	26.2 (Unit 1) 26.9 (Unit 2)
Limerick Generating Station, Units 1 and 2	01/31/27	10/26/44 (Unit 1) 06/22/49 (Unit 2)	03/28/16 (Unit 1) 04/24/17 (Unit 2)	28.6 (Unit 1) 32.2 (Unit 2)
Nine Mile Point Nuclear Station, Unit 1	08/22/29	08/22/29	S01-S22: 03/26/11 S23-S43: 04/20/13 S44-S68: 03/27/17	18.4

Plant/Unit	Current ISI Interval End Date	Current License Period End Date	Date of Last Category B-G-1, Item Number B6.20 Examination[1]	Length of Relief Requested (years)[2]
Nine Mile Point Nuclear Station, Unit 2	08/22/28	10/31/46	S01-S25: 03/11/20 S26-S50: 03/15/22 S51-S76: 04/26/16	30.6
Peach Bottom Atomic Power Station, Units 2 and 3	12/31/28	08/08/33 (Unit 2) 07/02/34 (Unit 3)	S01-S46: 10/19/10 (Unit 2) S47-S92: 10/28/16 (Unit 2) S01-S46: 09/30/11 (Unit 3) S47-S92: 09/26/15 (Unit 3)	28.2 (Unit 2)[3] 27.3 (Unit 3)[3]

Notes:
1. Some stations split up the RPV closure head stud examinations in order to decrease the impact of examining all of the studs in any one given outage. If the examinations were split into different groups of studs, the last examination date was provided for each stud group in the “Date of Last Category B-G-1, Item Number B6.20 Examination” column.
2. Conservatively, the examination date of the group of RPV studs in column “Date of Last Category B-G-1, Item Number B6.20 Examination” that would allow the longest deferral of examinations was used to calculate the “Length of Relief Requested” column.
3. Relief is only being requested for the Fifth and Sixth ISI Intervals. The “Length of Relief Requested” numbers provided are based on the expected end date of the Sixth ISI Interval which is 12/31/38.

As indicated in Table 2, the proposed alternative results in a maximum effective operating time period of 33.1 years from the last inspection for the Pressurized Water Reactor (PWR) plants and 32.2 years 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 [Reference 8.1] demonstrates that time intervals substantially longer than 30 years (80 years for Pressurized Water Reactor (PWR) and 37.9 years for BWR) are required for flaws to reach ASME Section XI acceptance limits.

The EPRI report 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 CEG’s past ISI examination records for RPV closure studs indicates there have been no relevant indications detected in the RPV closure studs.

Plant Maintenance

The RPV closure head is sealed using two self-energized metallic O-rings. The volume between these O-rings is 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. System leakage tests are performed each refueling outage 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.

The generic aging management program for RPV studs in the Generic Aging Lessons Learned (GALL) report lists four preventative actions that can effectively reduce the potential for SCC:

- Avoiding studs that are metal plated to reduce the potential for seizing. The metal plating can lead to hydrogen embrittlement or galvanic corrosion at discontinuities when wetted.

- Applying manganese phosphate or other acceptable surface treatments.
- Avoiding the use of molybdenum disulfide as a lubricant, and instead using lubricants that remain stable at operating temperatures.
- Using material with an actual yield strength confirmed by measurement to be less than 150 ksi (newly installed studs) or an ultimate strength of less than or equal to 170 ksi (existing studs).

All four bullets above are satisfied using site-specific periodic maintenance procedures and supply chain controls on procurement of new RPV studs compliant with NRC Regulatory Guide 1.65 [Reference 8.4]. These compliance commitments are established in the UFSAR or through maintenance procedures for the plants listed in Table 1.

The design of the RPV closure studs at Braidwood, Byron, and Calvert Cliffs allow them to be completely removed during each refueling outage permitting visual inspection of the RPV stud and the threads in flange to assess protection against degradation. Refueling procedures require that each stud be removed, visually inspected, and placed in a stud rack. After the studs are removed, the stud holes in the RPV flange are sealed with a special plug. The studs are lifted and moved to a storage area prior to the water level being raised in the refueling cavity. Thus, the bolting materials and stud holes are not exposed to the borated refueling cavity water. These procedural steps mitigate exposing the studs to chlorides and potential degradation mechanisms during refueling activities. Additional protection against the possibility of incurring corrosion effects is assured by using a manganese base phosphate surfacing treatment applied to each RPV closure stud. 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.

For 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 closure studs and the stud holes are inspected for presence of water and foreign material. 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 top head is replaced, the studs are replaced into the RPV flange, 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 Examination Category 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 closure studs and the stud holes are inspected for presence of water and foreign material. Typically, only 4 to 6 RPV studs are removed, depending on the site, 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 necessary RPV studs, but once the studs are removed, the related RPV threads are inspected for damage. 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 top head is replaced, the studs are replaced into the RPV flange, 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 Examination Category B-P (VT-2) examinations, they provide additional assurance of pressure boundary integrity.

Condition Monitoring

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 CEG administratively mandated corrective action program and its participation in and review of industry OE.

Site-specific maintenance procedures and practices related to RPV closure studs are compliant with NRC Regulatory Guide 1.65 [Reference 8.4] and reflect lessons learned from industry OE. These maintenance procedures have been revised to ensure potential degradation mechanisms associated with RPV closure studs are mitigated. As noted previously, the upper and lower stud threads on all removed studs are cleaned every outage and evaluated for mechanical damage; therefore, the critical volume around threads in the closure studs will be observed for new degradation or a change in degradation.

RPV closure stud examinations in accordance with ASME Section XI requirements continue to be performed at nuclear plants within the CEG Fleet (Braidwood Station, Unit 2, Dresden Nuclear Power Station, Units 2 and 3, R. E. Ginna Nuclear Power Plant, and Quad Cities Nuclear Power Station, Units 1 and 2) and within the Industry. This would result in approximately 28.6% of the CEG fleet (6 out of the 21 operating units) continuing to undergo performance monitoring under the requirements of ASME Section XI. Considering the individual number of reactor vessel closure studs, this would result in approximately 31.6% (469 out of 1483 studs) of the CEG Fleet studs examined under the performance monitoring plan. Any relevant indications or new degradation mechanisms identified during those volumetric examinations of the RPV closure studs would be entered into the CEG Corrective Action Program as required by the applicable administrative procedures. This OE would be evaluated and extent of condition examinations (if required per the evaluation) would be performed at the plants listed in Table 1. If the extent of condition examinations for any of the plants listed in Table 1 identify relevant indication or new degradation mechanisms, then the plant where the indication(s) was identified will be removed from consideration of this relief request and will resume the ASME Section XI examination frequency.

Therefore, CEG 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. DURATION OF PROPOSED ALTERNATIVE:

The proposed alternative is requested for the remainder of the currently licensed operating periods for the plants listed in Tables 1 and 2 except for Peach Bottom Atomic Power Station requested through the Sixth ISI Interval. A similar duration has been approved by the NRC Safety Evaluation (ADAMS Accession No. ML22096A003) listed in the Precedent section, and the NRC position to allow approval of relief requests longer than one ISI Interval is addressed in SECY-23-0061 [Reference 8.5].

7. PRECEDENT:

The following previous submittal has been made by Duke Energy to provide relief from the ASME Section XI Examination Category B-G-1 Item Number B6.20 volumetric examinations based on the Reference 8.1 technical basis report:

- Letter from S. Snider (Duke Energy) to the U.S. NRC, “Relief Request for Alternative for Reactor Vessel Closure Stud Examinations,” dated December 1, 2020, ADAMS Accession No. ML20336A033 [Reference 8.2].

The U.S. NRC issued a safety evaluation of the Duke Energy request for alternative on November 18, 2022.

- Letter from D. Wrona (U.S. NRC) to S. Gibby (Duke Energy), “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 – Authorization of RA-19-0352 Regarding Use of Alternative for Reactor Pressure Vessel Head Closure Stud Examinations (EPID L-2020-LLR-0156),” dated November 18, 2022, ADAMS Accession No. ML22096A003 [Reference 8.3].

8. REFERENCES:

1. *Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs*. EPRI, Palo Alto, CA: 2018. 3002014589.
2. Letter from S. Snider (Duke Energy) to the U.S. NRC, “Relief Request for Alternative for Reactor Vessel Closure Stud Examinations,” dated December 1, 2020, ADAMS Accession No. ML20336A033.
3. Letter from D. Wrona (U.S. NRC) to S. Gibby (Duke Energy), “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 – Authorization of RA-19-0352 Regarding Use of Alternative for Reactor Pressure Vessel Head Closure Stud Examinations (EPID L-2020-LLR-0156),” dated November 18, 2022, ADAMS Accession No. ML22096A003.
4. U.S. NRC, Materials and Inspections for Reactor Vessel Closure Studs, Regulatory Guide 1.65, Rev. 1, April 2010.
5. U.S. NRC, Clarification of the Staff’s Position on Certain American Society of Mechanical Engineers Code Alternatives for More Than One 10-Year Inservice Inspection Interval Under Title 10 of the Code of Federal Regulations 50.55a, SECY-23-0061, dated July 21, 2023, ADAMS Accession No. ML23094A178.

6. Letter from S. Gibby (Duke Energy) to the U.S. NRC, “Response to Requests for Additional Information for Reactor Vessel Closure Stud Exam Extension Alternative,” dated January 31, 2022, ADAMS Accession No. ML22032A142.

ATTACHMENT 1

Summary of EPRI Report

A report from the Electric Power Research Institute (EPRI) [Reference 1] provides a technical basis evaluation of the inspection period for reactor vessel 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 11.

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 pressure vessel (RPV) closure studs identified in the Generic Aging Lessons Learned (GALL) report [Reference 2] and the GALL Report for Subsequent License Renewal (GALL-SLR) [Reference 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 high-strength bolting steels. High-strength has been designated in the past as materials with an ultimate strength greater than 170 ksi, as well as materials with a yield strength greater than 150 ksi. These initiating causes have been removed from operating plants, including via Regulatory Guide 1.65, Revision 1 [Reference 6]. Additionally, GALL-SLR XI.M3, "Reactor Head Closure Stud Bolting" restricts bolting material for closure studs to those having an actual measured yield strength of less than 150 ksi for newly installed studs, or 170 ksi ultimate tensile strength for existing studs. The plant-specific historical usage of RPV closure stud lubricants is described in Attachments 2 through 11. 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 RPV 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 by existing refueling activities and procedures. The EPRI report concludes that the most plausible degradation mechanisms for the RPV closure studs are 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 RPV closure studs. The method of evaluation for the RPV 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 RPV 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. It should be noted that the stud tension and bending stresses considered in the EPRI report are intended to be bounding for flaw growth considerations; therefore, these loads are not directly applicable to the maximum stud tension loads considered for RPV threads-in-flange evaluations.

It is a general characteristic of RPV closures that the stud preload required to resist internal pressure are substantially greater than seismic or accident loadings. As an example, the total preload force applied by the studs for a typical RPV closure is on the order of 50,000,000 pounds; the design preload is generally set to a few percent greater than the blowoff force caused by design pressure acting to the o-ring sealing radius. In contrast, the weight of the reactor closure head is on the order of 500,000 pounds, a factor of 100 less than the preload force. Therefore, a bounding vertical seismic acceleration of 5g would result in a negligible uplift force relative to the stud force holding the head down. Likewise, a bounding horizontal acceleration of 5g would result in a lateral force of 2,500,000 lbs, which is substantially lower than the force required to overcome static friction; assuming a lower bound coefficient of static friction equal to 0.2, a force of 10,000,000 lbs (i.e., 0.2 times the preload force) would be required to overcome it. Other accident loadings, such as LOCA events, tend to reduce the internal pressure and would therefore not change stud loads beyond the transients already considered in the analysis.

Section 4 of the EPRI report describes flaw tolerance assessment of the closure studs. For this assessment, the RPV 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 [Reference 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 Nonmandatory 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 Nonmandatory 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 closure studs formed the basis for establishing an appropriate K_{IC} for bolting material [Reference 5] of 190 ksi√in (209 MPa√m).

The results of the calculation demonstrate the following:

- The fatigue crack growth for PWR RPV closure 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 closure 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 eight criteria to assess the plant-specific applicability of the technical basis and the supporting stress and flaw tolerance assessments for the RPV closure studs. These criteria, and the applicability evaluation for each plant, are provided in Attachments 2 through 11.

Conclusion

This attachment summarizes the EPRI report for the RPV 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 11.

REFERENCES:

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

ATTACHMENT 2

Plant-Specific Information for: Braidwood Station, Unit 1

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at Braidwood Station (Braidwood), 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (88.34" / 6.75") = 13.09 < 14.5 (PWR); Acceptable.

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

    - Value = (88.34" / 6.87") = 12.86 < 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. Braidwood 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 detensioning procedures at Braidwood, Unit 1, both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. Braidwood operates on 18-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/1.5 years = 2.7 cycles/year.

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

   Acceptable. Braidwood Technical Specification 3.4.3, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.3.1, requires all heatup events to maintain within the limits of the Pressure and Temperature Limits Report (PTLR). From the PTLR, Section 2.1.1.a requires all heatup events to maintain 100°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. Braidwood Station Technical Specification 3.4.3, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.3.1, requires all cooldown events to maintain within the limits of the PTLR. From the PTLR, Section 2.1.1.b requires all cooldown events to maintain 100°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 Inadvertent Startup of an Inactive Loop transient is bounding compared to other upset transients. The Inadvertent Startup of an Inactive Loop transient results in the active loop temperature variation of approximately 40°F over 8 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. The number of actual occurrences of reactor coolant system transients as monitored in the Braidwood thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All Braidwood Unit 1 RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

- Acceptable. All existing Braidwood, Unit 1, RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
- Acceptable. All newly installed Braidwood, Unit 1, RPV closure studs have a yield strength of less 150 ksi.

6. RPV 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:

    - Acceptable. All Braidwood RPV closure studs are specified as SA-540 Grade B23.

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

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

ATTACHMENT 3

Plant-Specific Information for: Byron Station, Units 1 and 2

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at Byron Station (Byron), Units 1 and 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (88.34" / 6.75") = 13.09 < 14.5 (PWR); Acceptable.

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

    - Value = (88.34" / 6.87") = 12.86 < 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. Byron 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 detensioning procedures at Byron both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. Byron operates on 18-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/1.5 years = 2.7 cycles/year.

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

   Acceptable. Byron Technical Specification 3.4.3, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.3.1, requires all heatup events to maintain within the limits of the Pressure and Temperature Limits Report (PTLR). From the PTLR, Section 2.1.1.a requires all heatup events to maintain 100°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. Byron Technical Specification 3.4.3, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.3.1, requires all cooldown events to maintain within the limits of the PTLR. From the PTLR, Section 2.1.1.b requires all cooldown events to maintain 100°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 Inadvertent Startup of an Inactive Loop transient is bounding compared to other upset transients. The Inadvertent Startup of an Inactive Loop transient results in the active loop temperature variation of approximately 40°F over 8 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. The number of actual occurrences of reactor coolant system transients as monitored in the Byron thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All Byron RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

- Acceptable. All existing Byron RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
- Acceptable. All newly installed Byron RPV closure studs have a yield strength of less than 150 ksi.

6. RPV 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:

   - Acceptable. All Byron RPV closure studs are specified as SA-540 Grade B23.

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

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

ATTACHMENT 4

Plant-Specific Information for: Calvert Cliffs Nuclear Power Plant, Units 1 and 2

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (83.25" / 7.50") = 11.1 < 14.5 (PWR); Acceptable.

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

    - Value = (85.69" / 6.82") = 12.56 < 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. Calvert Cliffs 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 tensioning and detensioning procedures at Calvert Cliffs both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. Calvert Cliffs operates on 24-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/2 years = 2.0 cycles/year.

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

Acceptable. Calvert Cliffs Technical Specification Section 3.4.3, Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.3.1, requires all heatup events to maintain 100°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. Calvert Cliffs Technical Specification Section 3.4.3, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.3.1, requires all cooldown events to maintain 100°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 Abnormal Loss of Load transient is bounding compared to other upset transients. The Abnormal Loss of Load transient results in an RCS temperature variation of approximately 28°F within 11 seconds (2.54°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 RCS transients as monitored in the Calvert Cliffs thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All Calvert Cliffs RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

   - Acceptable. All existing Calvert Cliffs RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
   - Acceptable. All Calvert Cliffs RPV closure studs have a yield strength of less than or equal to 150 ksi.

6. RPV 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:

    - Acceptable. All Calvert Cliffs RPV closure studs are specified as ASTM A-540 Grade B24, which is consistent with all SA-540 Grade B24 requirements and permitted for use under ASME Section II.

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

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

ATTACHMENT 5

Plant-Specific Information for: Clinton Power Station, Unit 1

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at Clinton Power Station (Clinton), 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (125.00" / 4.94") = 25.30 < 34.8 (BWR); Acceptable.

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

    - Value = (125.00" / 6.00") = 20.83 < 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. Clinton 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 tensioning and detensioning procedures at Clinton both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. Clinton operates on 24-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/2 years = 2.0 cycles/year.

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

Acceptable. Clinton Technical Specification Section 3.4.11, Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.11.1, requires all heatup events to maintain 100°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. Clinton Technical Specification Section 3.4.11, Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.11.1, requires all cooldown events to maintain 100°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 Automatic Blowdown (ADS) transient is bounding compared to other upset transients. The ADS transient results in an RCS temperature variation of approximately 177°F within 3.3 minutes (0.89°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 RCS transients as monitored in the Clinton thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All Clinton RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

   - Acceptable. All existing Clinton RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND

6. RPV 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:

    - Acceptable. All Clinton RPV closure studs are specified as SA-540 Grade B23 or B24.

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

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

ATTACHMENT 6

Plant-Specific Information for: James A. FitzPatrick Nuclear Power Plant

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at James A. FitzPatrick Nuclear Power Plant (FitzPatrick). 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (109.50" / 3.1875") = 34.35 < 34.8 (BWR); Acceptable.

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

    - Value = (109.50" / 6.00") = 18.25 < 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. FitzPatrick 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 tensioning and detensioning procedures at FitzPatrick both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. FitzPatrick operates on 24-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/2 years = 2.0 cycles/year.

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

Acceptable. FitzPatrick Technical Specification Section 3.4.9, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.9.1, requires all heatup events to maintain 100°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. FitzPatrick Technical Specification Section 3.4.9, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.9.1, requires all cooldown events to maintain 100°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 Reactor Overpressure with Delayed Scram transient is bounding compared to other upset transients. The Reactor Overpressure with Delayed Scram transient results in a reactor coolant system (RCS) temperature variation of approximately 67°F within 30 seconds (2.23°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 FitzPatrick thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All FitzPatrick RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

   - Acceptable. All existing FitzPatrick RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
   - Acceptable. All FitzPatrick RPV closure studs have a yield strength of less than 150 ksi.

6. RPV 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:

    - Acceptable. All FitzPatrick RPV closure studs are specified as SA-540 Grade B24.

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

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

ATTACHMENT 7

Plant-Specific Information for: LaSalle County Station, Units 1 and 2

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at LaSalle County Station (LaSalle), Units 1 and 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (125.5" / 4.93") = 25.46 < 34.8 (BWR); Acceptable.

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

    - Value = (125.5" / 5.75") = 21.83 < 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. LaSalle 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 tensioning and detensioning procedures at LaSalle both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. LaSalle operates on 24-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/2 years = 2.0 cycles/year.

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

Acceptable. LaSalle Technical Specification Section 3.4.11, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.11.1, requires all heatup events to maintain within the limits of the Pressure and Temperature Limits Report (PTLR). From the PTLR, Section 5.0 requires all cooldown events to maintain 100°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. LaSalle Technical Specification Section 3.4.11, RCS Pressure and Temperature (P/T) Limits, Surveillance Requirement 3.4.11.1, requires all cooldown events to maintain within the limits of the PTLR. From the PTLR, Section 5.0 requires all cooldown events to maintain 100°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 Reactor Overpressure with Delayed Scram transient is bounding compared to other upset transients. The Reactor Overpressure with Delayed Scram transient results in a reactor coolant system (RCS) temperature variation of 45°F within 30 seconds (1.5°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 LaSalle thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All LaSalle RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

- Acceptable. All existing LaSalle Unit 2 RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
- Section A.2.1.3 of Appendix P of the Lasalle UFSAR states that some of the closure stud bolting material in use has a measured yield stress greater than 150 ksi. The staff previously reviewed this exception to NRC Regulatory Guide 1.65 in the Safety Evaluation Report with Open Items Related to the License Renewal of Lasalle County Stations, Unit 1 (ML16271A039).

6. RPV 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:

   - Acceptable. All LaSalle RPV closure studs are specified as SA-540 Grade B24.

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

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

ATTACHMENT 8

Plant-Specific Information for: Limerick Generating Station, Units 1 and 2

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at Limerick Generating Station (Limerick), Units 1 and 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (125.5" / 3.625") = 34.62 < 34.8 (BWR); Acceptable.

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

    - Value = (125.5" / 5.625") = 22.31 < 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. Limerick 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 tensioning and detensioning procedures at Limerick both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. Limerick operates on 24-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/2 years = 2.0 cycles/year.

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

Acceptable. Limerick Technical Specification Section 3.4.6.1.a., requires all heatup events to maintain within the limits of the Pressure and Temperature Limits Report (PTLR). From the PTLR, Section 4.0 requires all heatup events to maintain 100°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. Limerick Technical Specification Section 3.4.6.1.b., requires all cooldown events to maintain within the limits of the PTLR. From the PTLR, Section 4.0 requires all cooldown events to maintain 100°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 Reactor Overpressure with Delayed Scram transient is bounding compared to other upset transients. The Reactor Overpressure with Delayed Scram transient results in a reactor coolant system (RCS) temperature variation of 45°F within 30 seconds (1.5°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 RCS transients as monitored in the Limerick thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All Limerick RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

   - Acceptable. All existing Limerick RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
   - Section A.2.1.3 of Appendix A of the Limerick UFSAR states that some of the closure stud bolting material in use has a measured yield stress greater than 150

ksi. The staff previously reviewed this exception to NRC Regulatory Guide 1.65 in the Safety Evaluation Report with Open Items Related to the License Renewal of Limerick Generating Stations, Units 1 and 2 (ML12173A470).

6. RPV 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:

    - Acceptable. All Limerick RPV closure studs are specified as SA-540 Grade B23.

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

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

ATTACHEMNT 9

Plant-Specific Information for: Nine Mile Point Nuclear Station, Unit 1

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at Nine Mile Point Nuclear Station (Nine Mile Point), 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (106.72" / 4.31") = 24.76 < 34.8 (BWR); Acceptable.

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

    - Value = (106.72" / 6.00") = 17.79 < 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. Nine Mile Point, 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 tensioning and detensioning procedures at Nine Mile Point, Unit 1, both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. Nine Mile Point, Unit 1, operates on 24-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/2 years = 2.0 cycles/year.

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

   Acceptable. Nine Mile Point, Unit 1, Technical Specification 3.2.1 requires all heatup events to maintain within the limits of the Pressure and Temperature Limits Report (PTLR). From the PTLR, Section 4.0, Operating Limits, requires all heatup events to maintain 100°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. Nine Mile Point, Unit 1, Technical Specification 3.2.1 requires all cooldown events to maintain within the limits of the PTLR. From the PTLR, Section 4.0, Operating Limits, requires all cooldown events to maintain 100°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. The Nine Mile Point, Unit 1, UFSAR states "Stresses in various core components are at maximum during the blowdown resulting from the main steam line (MSL) break", therefore the temperature versus time variation for the Blowdown transient is bounding compared to the other upset transients. The Blowdown transient results in a reactor coolant system (RCS) temperature variation of approximately 17.5°F per minute (0.29°F/sec) over a 10 minute period 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 Nine Mile Point, Unit 1, thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All Nine Mile Point, Unit 1, RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

- Acceptable. All existing Nine Mile Point, Unit 1, RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
- Acceptable. All Nine Mile Point, Unit 1, RPV closure studs have a yield strength of less than 150 ksi.

6. RPV 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:

    - Acceptable. RPV closure studs are specified as SA-540 Grade B23 or B24.

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

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

ATTACHMENT 10

Plant-Specific Information for: Nine Mile Point Nuclear Station, Unit 2

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at Nine Mile Point Nuclear Station (Nine Mile Point), 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (124.75" / 4.94") = 25.25 < 34.8 (BWR); Acceptable.

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

    - Value = (124.75" / 6.00") = 20.79 < 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. Nine Mile Point, 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 tensioning and detensioning procedures at Nine Mile Point, Unit 2, both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. Nine Mile Point, Unit 2, operates on 24-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/2 years = 2.0 cycles/year.

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

   Acceptable. Nine Mile Point, Unit 2, Technical Specification 3.4.11 requires all heatup events to maintain within the limits of the Pressure and Temperature Limits Report (PTLR). From the PTLR, Section 4.0, Operating Limits, requires all heatup events to maintain 100°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. Nine Mile Point, Unit 2, Technical Specification 3.4.11 requires all cooldown events to maintain within the limits of the PTLR. From the PTLR, Section 4.0, Operating Limits, requires all cooldown events to maintain 100°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 Automatic Blowdown transient is bounding compared to other upset transients. The Automatic Blowdown transient results in a reactor coolant system (RCS) temperature variation of 117°F within 3.3 minutes (0.59°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 RCS transients as monitored in the Nine Mile Point, Unit 2, thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All Nine Mile Point, Unit 2, RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

- Acceptable. All existing Nine Mile Point, Unit 2, RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
- Acceptable. All Nine Mile Point, Unit 2, RPV closure studs have a yield strength of less than 150 ksi.

6. RPV 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:

   - Acceptable. RPV closure studs are specified as SA-540 Grade B23 or B24.

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

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

ATTACHMENT 11

Plant-Specific Information for: Peach Bottom Atomic Power Station, Units 2 and 3

This attachment provides a plant-specific assessment of the eight 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 reactor pressure vessel (RPV) closure studs at Peach Bottom Atomic Power Station (Peach Bottom), Units 2 and 3. 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, *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 RPV head inner radius to head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs:

    - Value = (125.68" / 4.00") = 31.42 < 34.8 (BWR); Acceptable.

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

    - Value = (125.68" / 5.83") = 21.56 < 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. Peach Bottom 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 tensioning and detensioning procedures at Peach Bottom both specify less than two full tensioning passes. Therefore, each stud is tensioned/detensioned no more than 4 times every head removal cycle. Peach Bottom operates on 24-month fuel cycles; therefore, the studs are tensioned and detensioned 4 cycles/2 years = 2.0 cycles/year.

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

Acceptable. Peach Bottom, Technical Specification Section 3.4.9. Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), requires all heatup events to maintain 100°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. Peach Bottom Technical Specification Section 3.4.9. Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR), requires all cooldown events to maintain 100°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 Reactor Overpressure with Delayed Scram transient is bounding compared to other upset transients. The Reactor Overpressure with Delayed Scram transient results in a reactor coolant system (RCS) temperature variation of 70°F within 30 seconds (2.3°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 Peach Bottom thermal fatigue management program is significantly under 1,000 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 RPV closure studs remain in service and are successfully tensioned:

   - Acceptable. All Peach Bottom RPV closure studs are currently in service and successfully tensioned.

5. All RPV 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). 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.

   - Acceptable. All existing Peach Bottom RPV closure studs have an ultimate tensile strength of less than 170 ksi; AND
   - Section R.2.1.3 of Appendix R of the Peach Bottom UFSAR states that some of the closure stud bolting material in use has a measured yield stress greater than

150 ksi. The staff previously reviewed this exception to NRC Regulatory Guide 1.65 in the Safety Evaluation Report to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3 (ML20044D902).

6. RPV 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:

   - Acceptable. All Peach Bottom RPV closure studs are specified as SA-540 Grade B23 or B24.

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

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