Constellation Energy®
Nine Mile Point Nuclear Station

P.O. Box 63
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May 14, 2009

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 1; Docket No. 50-220

Request to Utilize an Alternative to the Requirements of 10 CFR 50.55a(g) for the Repair and Inservice Inspection of Control Rod Drive Stub Tubes for the License Renewal Period of Extended Operation – Response to NRC Request for Additional Information (TAC No. MD9604)

REFERENCES: (a) Letter from G. J. Laughlin (NMPNS) to Document Control Desk (NRC), dated August 29, 2008, Request to Utilize an Alternative to the Requirements of 10 CFR 50.55a(g) for the Repair and Inservice Inspection of Control Rod Drive Stub Tubes for the License Renewal Period of Extended Operation

(b) Letter from R. V. Guzman (NRC) to K. J. Polson (NMPNS), dated March 17, 2009, Request for Additional Information Regarding Nine Mile Point Nuclear Station, Unit No. 1, Alternative for the Repair and Inservice Inspection of Control Rod Drive Stub Tubes for the License Renewal Period of Extended Operation (TAC No. MD9604)

Nine Mile Point Nuclear Station, LLC (NMPNS) hereby transmits supplemental information requested by the NRC in support of a previously submitted 10 CFR 50.55a Request (Number 1ISI-02) under the provision of 10 CFR 50.55a(a)(3). The initial request, dated August 29, 2008 (Reference a), would allow the use of American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Code Case N-730, "Roll-Expansion of Class 1 Control Rod Drive [CRD] Bottom Head Penetrations in BWRs," as an alternative permanent repair for CRD housings that may exhibit leakage during the license renewal period of extended operation. The supplemental information, provided in the attachment to this letter, responds to the request for additional information documented in the NRC's letter dated March 17, 2009 (Reference b). This letter contains no new regulatory commitments.

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Should you have any questions regarding the information in this submittal, please contact T. F. Syrell, Licensing Director, at (315) 349-5219.

Very truly yours,

Peter A. Mazzaferro
Acting Manager Engineering Services

PAM/GNS

Attachment: Nine Mile Point Unit 1 – Response to NRC Request for Additional Information Regarding 10 CFR 50.55a Request Number 1ISI-02

cc: S. J. Collins, NRC
R. V. Guzman, NRC
Resident Inspector, NRC

ATTACHMENT

NINE MILE POINT UNIT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING 10 CFR 50.55a REQUEST NUMBER 1ISI-02

ATTACHMENT

NINE MILE POINT UNIT 1
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING 10 CFR 50.55a REQUEST NUMBER 1ISI-02

By letter dated March 17, 2009, Nine Mile Point Nuclear Station, LLC (NMPNS) received a request for additional information (RAI) (Reference 1) from the NRC. The NRC questions are reiterated below, followed by the NMPNS response.

NRC Request:

In its proposed alternative for repair of control rod drive stub tubes, the Licensee requested in Alternative 1A to apply the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME Code), Section XI, Code Case N-730, "Roll-Expansion of Class 1 Control Rod Drive [CRD] Bottom Head Penetrations in BWRs." In Alternative 2A, the licensee requested to apply ASME Code, Section XI, Code Case N-730, but with the exception to perform the post-roll expansion VT-2 examination at approximately 900 pounds-per-square inch (psig) prior to returning NMP1 to full power operation. ASME Code Case N-730 requires a post-repair leakage test at the operating pressure. As the basis for the proposed alternatives (1A and 2A), the submittal references both Title 10 to Code of Federal Regulations (10 CFR), Section 50.55a(a)(3)(i) related to demonstrating an acceptable level of quality and safety as well as Section 50.55a(a)(3)(ii) related to demonstrating hardship. Please clarify which regulatory basis is being applied for each alternative when responding to the questions below.

RAI-1. For proposed Alternative 1A, provide additional basis for using ASME Code Case N-730 instead of the ASME Code, Section XI, requirements by addressing the 4 considerations of the evaluation stated in Section 5 of the ASME Code Case N-730. If a generic evaluation is available, demonstrate that the generic evaluation is applicable to NMP1.

NMPNS Response:

For proposed Alternative 1A, NMPNS has determined that implementation of ASME Code Case N-730 provides an acceptable level of quality and safety and satisfies the requirements of 10 CFR 50.55a(a)(3)(i).

As requested, the four (4) ASME Code Case N-730, Section 5, evaluation considerations are addressed below for the Nine Mile Point, Unit 1 (NMP1) roll expansion repairs:

Code Case Evaluation Requirement 5.1: Analysis shall be performed to show that the thickness of the CRD housing after rolling is sufficient to meet the primary stress limits of the Construction Code.

Compliance with Requirement 5.1:

NMPNS performed a plant-specific analysis to demonstrate compliance with the ASME Section III Construction Code basic sizing requirements and primary stress requirements. The analysis conservatively assumed the thickness of the CRD housing was reduced by the maximum 6.5% wall thinning allowed by Code Case N-730 Section 1.3(c). The following is a summary of the detailed plant-specific code reconciliation analysis:

- The CRD housing wall thickness, after 6.5% wall thinning, is 0.520 inches, which exceeds the code required minimum wall thickness of 0.216 inches. Therefore, the basic sizing requirements of the code are satisfied.
- The primary loadings which the thinned CRD housing is subjected to in the rolled area are tensile or compressive axial loads. The axial mechanical loads are transmitted through the area of the thinned CRD housing wall to the reactor vessel via the CRD stub-tube-to-housing J-groove weld.

This CRD penetration load path was evaluated before and after roll expansion of the CRD housing. The calculated load capacity of the CRD housing before roll expansion, i.e., before CRD housing thinning, is greater than the load capacity of the stub-tube-to-housing J-groove weld. After 6.5 % wall thinning the load capacity of the CRD housing remains greater than the load capacity of the stub-tube-to-housing J-groove weld. From this, the code analysis concluded that the primary load carrying capacity of the CRD penetration had not been compromised by roll expansion and thinning of the housing.

- Secondary stresses and fatigue usage factors were also reviewed in the analysis. The secondary stresses and fatigue usage factors were found to remain essentially the same after implementation of the roll expansion. The analysis concluded that the CRD housing and reactor vessel continue to meet the ASME Section III fatigue requirements after implementation of rolling.

The above described plant-specific construction code analysis is the analysis described in the Code Case N-730 Technical Basis Report, XGEN-2005-10, Revision 3, Section 8.1.6, (Reference 2).

Code Case Evaluation Requirement 5.2: The expanded penetration shall satisfy all plant-specific design criteria related to structural integrity. All specified load combinations and design-basis events shall be addressed.

Compliance with Requirement 5.2:

A NMP1 plant-specific evaluation for roll expanded CRD penetrations which considered specified load combinations and design-basis events was performed. The evaluation considered normal, seismic, main steam line break, recirculation line break and scram reaction loads. For most loads, the pressure loading on the CRD housing is downward and the cracked stub tube will take compression and transfer the load to the vessel, thus preventing CRD housing failure or ejection. The upward scram loads are the limiting loads that tend to lift the CRD housing relative to the vessel head. The roll expansion repair is designed to prevent lifting the CRD housing during the maximum scram loads. This evaluation was previously submitted to the NRC and the NRC's review is summarized in the Reference 3.

Additionally, Combustion Engineering, NMP1's original roll expansion repair vendor and reactor manufacturer, evaluated the effects of thermal transients on the roll expansion contact pressures and effectiveness of the rolled joint. The thermal cycling evaluations confirmed the long-term adequacy of the roll expanded joints to prevent leakage, which is consistent with the NMP1 operating experience.

The ASME code case requires that the roll expanded joint be capable of resisting the scram reaction loads to prevent the CRD housing from lifting. Section 2(b) of the code case requires that the roll-band length not be less than the roll-band length required to resist end-of-scram loads. The code case provides the following equation for determining the roll band length:

$L = (SF)\ F / [0.4\pi\ (1\text{-}p) \times T \times Sy]$, where:

F = Maximum upward end-of-scram force, Scram End of Stroke, No Buffer, use 17.1 kips
(Ref: BWRVIP-55-A, Table 7-1. The value used is more conservative than the CRD scram loads shown on the original NMP1 vessel loading drawing to ensure the "no buffer" condition is addressed)
SF = Structural Factor = 2
p = Nominal wall-thinning fraction (use .035 for minimum 3.5% wall thinning)
T = Thickness of housing, 0.556 in.

Sy = Yield strength of the housing material at room temperature = 30 ksi
$L = 2x17.1//[0.4\pi (1-.035) \times .556 \times 30] = 1.7$ in.

The code case defines the roll-band length as the flat portion of the roll, excluding the rounded transition region at each end. The transition region is approximately 0.25 inches at each end of the roll band. Given the minimum nominal roll band length specified in the design specification for NMP1 is 4.5 inches, the flat portion of the roll is 4.5" – 2(.25") = 4.0 inch. Based on this calculation, the flat portion of the roll band length of 4.0 inches exceeds the length of 1.7 inch required to resist the maximum upward design load.

Therefore, it is concluded that the previously reviewed evaluation addressed all load combinations and design basis events and the roll expanded penetration satisfies the limiting scram design load related to structural integrity by greater than a factor of two.

Code Case Evaluation Requirement 5.3: Crack growth shall be predicted, considering stress corrosion cracking and fatigue. The analytical evaluation of the predicted crack shall satisfy the requirements of IWB-3600.

Compliance with Requirement 5.3:

The NMP1 observed cracking to date has only been seen in the 304 stainless steel furnace/weld sensitized stub tube base metal. The crack growth evaluation requirement does not apply to cracking in the stub tube base metal because it is conservatively assumed in the design of the ASME Code Case roll repair that 360 degree through-wall stub tube cracks exist (Reference 2: XGEN-2005-10, Section 5.1.1). Therefore, requirement 5.3 is not applicable to NMP1. If cracking is detected in other locations defined in code case sections 1.1(b) and 1.1(c), then NMPNS will perform the applicable evaluations required by section 5.0 of the code case.

Code Case Evaluation Requirement 5.4: If the source of the leakage is a crack in the vessel attachment weld, a postulated axial crack in the vessel attachment weld shall be evaluated. The analytical evaluation shall include an assumption that the entire weld is cracked radially and shall satisfy the requirements of IWB-3600.

Compliance with Requirement 5.4:

NMPNS has not observed cracking in the stub-tube-to-vessel attachment weld. As described above, all cracking has been confined to the stub tube base metal. However, on one stub tube the typical cracking was not seen in the stub tube base metal and while no cracking was seen in the vessel attachment weld, cracking in this weld could not be ruled out. Thus, requirement 5.4 is not directly applicable to NMP1. However, NMPNS conservatively assumed cracking can exist in the attachment weld and performed a plant-specific evaluation of a postulated flaw in the weld. The evaluation concluded that cracking in the attachment weld does not pose any structural integrity concerns for the NMP1 vessel. It was determined that the postulated flaw, based on conservative assumptions, was stable and that there was significant flaw tolerance at this location.

Additionally, the generic XGEN-2005-10, Appendix A, fracture mechanics evaluation for a postulated crack in the vessel attachment weld could be applied to NMP1. The generic analysis conservatively assumes the entire weld is cracked and it evaluates crack growth into the vessel. The analysis considered both fatigue and stress corrosion cracking for determining crack growth into the vessel. The analysis concluded that sufficient fracture margins exist for the 60 year plant life.

A comparative evaluation is included herein to demonstrate that the referenced generic fracture mechanics evaluation bounds NMP1. The table below compares the different fracture parameters between the generic assessment and NMP1. Based on the below comparison table, the pressure stress is approximately 28% lower at NMP1 than the generic analysis. Furthermore, the available facture toughness is approximately 19% higher. Therefore, the fracture margin at NMP1 is higher than that in the generic analysis.

Fracture Parameter/Attribute	**Generic Assessment**	**NMP1**
Stub tube to vessel attachment weld material	Alloy 182	Alloy 82/182
Radius of bottom head	100 in.	106.9 in.
Thickness of bottom head	6 in.	8 ¾ in.
Operating pressure	1050 psi	1030 psi
Limiting Reference Nil Ductility Temperature (RTndt) of bottom head	40 °F	40 °F
Pressure test temperature	180 °F	195 °F
App. A, Page 52 = Assumes stress corrosion cracking. "Most BWRs operate with hydrogen water chemistry….."	Normal water chemistry	Noble metals chemical addition and hydrogen water chemistry
Appendix A, Page 53: Number of cycles	Total Cycles = 810 cycles/40 years	Total projected cycles = 778 Cycles/60 years
Bottom head nominal stress (pressure stress)	PR/2t = 1.05x100/2x6 = 8.75 ksi	PR/2t = 1.03x106.9/2x8.75 = 6.29 ksi
Available fracture toughness (KIa) during pressure test	121.6 ksi√inch	= 26.8 + 12.445 exp(0.0145(T-RTndt) = 26.8 + 12.445 exp(0.0145(195-40) KIa =144.6 ksi√inch

NRC Request:

RAI-2. For proposed Alternative 2A, provide the basis for using ASME Code Case N-730 with the exception of performing a leakage test at 900 psig, instead of the ASME Code, Section XI, requirements. If 10 CFR 50.55a(a)(3)(ii) is the applicable regulatory basis, as discussed during the teleconference between NRC and licensee staff on February 19, 2009, provide a more detailed justification to support your use of 10 CFR 50.55a(a)(3)(ii) and, in particular, the following referenced items in your original submittal:

- *Basis for Relief (ii) states, "[i]nspections of CRD stub tubes for leakage during a system leakage test at the nominal operating pressure (~1030 psig) associated with 100% rated* *reactor power cannot be performed during a normal plant startup, due to the excessive temperature and radiological exposure conditions to which the NDE [nondestructive examination] examiners performing the visual VT-2 examinations would be exposed in the primary containment." Provide justification to support this statement as a hardship.*

- *Basis for Relief (iii) states, "[t]his special test usually takes one full day of plant outage time, and the additional valve lineups and system reconfigurations necessary to support this special test impose an additional challenge to the affected systems." Provide justification to support this statement as a hardship.*

NMPNS Response:

For proposed Alternative 2A, NMPNS has determined that compliance with ASME Code Case N-730 Section 6.6 post-roll expansion leakage test requirement would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety per the requirements of 10 CFR 50.55a(a)(3)(ii).

During a refueling outage, NMPNS will perform the normal ASME XI IWB-5000 required leakage test at rated test pressure of 1030 psig. If CRD penetration leakage is detected and an ASME roll expansion repair is implemented, NMPNS proposed to perform the post roll repair leakage inspection at 900 psig during the final drywell entry during plant startup. NMPNS provided the technical basis in paragraphs 2B(i) and 2B(iv through vii) in the Reference 4 submittal to justify performing the post roll expansion repair VT-2 leakage exam at approximately 900 psig in lieu of the 1030 psig required by ASME XI IWB-5000. Based on review of startup data from several past refueling outages, the 900 psig final drywell entry inspection is typically performed between 925 to 940 psig. Therefore, NMPNS proposes to increase the pressure at which the VT-2 leakage exam is performed from 900 psig to a range of 925 to 940 psig.

Below is more detailed information to justify the hardship or unusual difficulties that result from performing the post roll repair leakage exam at 1030 psig. This information is intended to supplement items 2B(ii) and 2B(iii) in our Reference 4 submittal, as requested by the staff, to further explain that performing a Code required system leakage test following the roll repair of a CRD penetration at 1030 psig to test the integrity of the mechanical joint imposes a Code test requirement that is a hardship.

There are two methods that can be utilized to perform a system leakage test in accordance with Code requirements.

1. A normal Code non-nuclear heat-up system leakage test can be performed. Isolation of only the CRD penetration that was roll repaired is not possible during this test. Therefore, the entire reactor pressure boundary and a portion of the Class 1 piping boundary would need to be isolated. This would require repeating the normal Code non-nuclear heat-up test that had just been performed. Repeating the normal Code required system leakage test requires an infrequently performed special evolution with the reactor vessel taken to a water solid condition. During this special evolution numerous safety systems are defeated during the test, such as; reactor SCRAM on high reactor pressure, emergency condenser initiation on high reactor pressure, Main Steam Isolation Valve (MSIV) closure SCRAM with reactor pressure above 600 psig, Main Condenser Lo-Lo-Lo vacuum vessel isolation, and Anticipated Transient Without Scram (ATWS) recirculation pump trip on high reactor pressure. Additionally, 4 of 6 electromatic relief valves (ERVs) are prevented from opening by pulling their control power fuses. The reactor cleanup system is shutdown and isolated which requires Chemistry technicians to enter a high radiation area to align the continuous conductivity monitor to reactor recirculation pump suction piping.

Thermal overload heaters are installed in the motor breaker for the reactor bottom head drain valve to equipment drain tank. Vessel level is controlled by throttling CRD flow and the reactor bottom head drain valve. This puts the drywell equipment drain tank pumps under a continuous use condition. This requires dedicated operators assigned to control CRD flow control valve and the bottom head drain valve to maintain reactor pressure below the two remaining operable ERV's lift setpoints. Finally, the time required to set up to perform the special evolution including raising vessel level to fill the vessel solid, perform the pressure test, depressurize and cool down the reactor to permit reactor drain down to normal level, return the above safety systems to operable condition, and then drain the main steam lines is on the order of 24 -36 hours. During this time period, the collective radiation dose to plant personnel responsible for performing these tasks is approximately 150 mRem. This dose would be in addition to the approximately 330 mRem received during the first leakage test. As such, re-isolation of the reactor vessel to repeat the normal non-nuclear heat-up Code required system leakage test is considered a hardship when considering the increased personnel dose and required defeat of safety systems without a commensurate increase in the level of quality and safety.

2. The second method to comply with the Code required system leakage test is to perform the inspection of the roll repaired CRD housing at rated pressure during power ascension. Establishing test conditions consistent with IWB-5221(a), when not performing a normal Code non-nuclear heat-up system leakage test, would require a drywell entry to perform the VT-2 examination at reactor power greater than 80 percent (i.e., the approximate power level when 1030 psig is achieved following normal reactor startup procedures). Access to the drywell at 80 percent power would be prohibited by industrial safety and radiological safety concerns.

   Following normal plant startup procedures, the mode switch is placed in RUN soon after the final 900 psig drywell inspection. Station Technical Specifications require the drywell oxygen concentration to be less than 4 percent within 24 hours of placing the mode switch in RUN. Drywell inerting with nitrogen to purge oxygen commences at approximately 20% power and therefore, drywell entry at 80 percent power would be prohibited or would require the use of breathing apparatus. Use of breathing apparatus such as an oxygen tank and mask would make access below the bottom head very difficult. Access below the bottom head requires the examiner to first drop to his/her knees and crawl along the platform below the CRD flanges and incore instrumentation. Once the examiner reaches the specific CRD location for inspection he/she must standup between the CRD flanges and incore instrumentation without bumping into trip sensitive equipment. Wearing breathing apparatus would significantly complicate this process. In addition to these physical constraints, the examiner is limited to the approximate 30 minute capacity of the breathing apparatus for containment entry, the VT-2 examination and containment exit. Furthermore, at approximately 80% power level drywell entry for inspection would be dose prohibitive. This test also represents a hardship and presents a risk to inspection personnel without a commensurate increase in the level of quality and safety.

References:

1. Letter from R. V. Guzman (NRC) to K. J. Polson (NMPNS), dated March 17, 2009, Request for Additional Information Regarding Nine Mile Point Nuclear Station, Unit No. 1, Alternative for the Repair and Inservice Inspection of Control Rod Drive Stub Tubes for the License Renewal Period of Extended Operation (TAC No. MD9604)

2. Report XGEN-2005-10, Revision 3, dated August 2006, Technical Basis for ASME Code Case N-730 Roll-Expansion of Class 1 Control Rod Drive (CRD) Bottom Head Penetrations in BWRs

3. Letter from D. B. Vassallo (NRC) to B. G. Hooton (NMPC), dated June 29, 1984, Control Rod Drive Penetration Leakage from Stub Tube Cracking

4. Letter from G. J. Laughlin (NMPNS) to Document Control Desk (NRC), dated August 29, 2008, Request to Utilize an Alternative to the Requirements of 10 CFR 50.55a(g) for the Repair and Inservice Inspection of Control Rod Drive Stub Tubes for the License Renewal Period of Extended Operation