Constellation Energy
Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, NY 13093

October 2, 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

American Society of Mechanical Engineers (ASME) Code, Section XI, Inservice Inspection Program for the Fourth Ten-Year Inservice Inspection Interval and Associated 10 CFR 50.55a Requests – Response to NRC Request for Additional Information (TAC No. ME0993)

REFERENCES: (a) Letter from P. A. Mazzaferro (NMPNS) to Document Control Desk (NRC), dated March 16, 2009, American Society of Mechanical Engineers (ASME) Code, Section XI, Inservice Inspection Program for the Fourth Ten-Year Inservice Inspection Interval and Associated 10 CFR 50.55a Requests

(b) Letter from R. V. Guzman (NRC) to S. L. Belcher (NMPNS), dated August 18, 2009, Request for Additional Information Regarding Nine Mile Point Nuclear Station, Unit No. 1, Relating to Relief Request, 1ISI-003, Associated with the Fourth 10-Year Inservice Inspection Interval (TAC No. ME0993)

Nine Mile Point Nuclear Station, LLC (NMPNS) hereby transmits supplemental information requested by the NRC in support of a previously submitted request for alternative (No. 1ISI-003) under the provision of 10 CFR 50.55a(a)(3). This 10 CFR 50.55a request was included within the Nine Mile Point Unit 1 Fourth Ten-Year Inservice Inspection Plan and Schedule that was submitted by letter dated March 16, 2009 (Reference a). The supplemental information, provided in the Attachment to this letter, responds to the request for additional information documented in the NRC's letter dated August 18, 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,

Joseph E. Pacher
Manager Engineering Services

JEP/DEV

Attachment: Nine Mile Point Unit 1 – Response to NRC Request for Additional Information Regarding Fourth Ten-Year Inservice Inspection Interval Request No. 1ISI-003

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 FOURTH TEN-YEAR INSERVICE INSPECTION

INTERVAL REQUEST NO. 1ISI-003

Nine Mile Point Nuclear Station, LLC
October 2, 2009

ATTACHMENT

NINE MILE POINT UNIT 1
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING FOURTH TEN-YEAR INSERVICE INSPECTION INTERVAL REQUEST NO. 1ISI-003

By letter dated March 16, 2009, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted the Nine Mile Point Unit 1 (NMP1) Fourth Ten-Year Inservice Inspection (ISI) Plan and Schedule and associated 10 CFR 50.55a request pursuant to 10 CFR 50.55a(a)(3). This attachment provides supplemental information in response to the request for additional information (RAI) documented in the NRC's letter dated August 18, 2009, concerning request no. 1ISI-003 (alternative risk-informed, safety-based ISI program). Each individual NRC request is repeated (in italics), followed by the NMPNS response.

RAI No. 1

In Section 3.0 of 1ISI-003 you state that there were no deviations to the process described in Code Case N-716. Code Case N-716 dated April 19, 2006, does not include the guidance that any segments with a large early release frequency (LERF) greater than 10^{-7} per year should be assigned as high safety significant (HSS). In Section 3.1, however, you state that a LERF greater than 10^{-7} per year was also used to determine high safety significance. Please identify all differences between the guidance in N-716 and your analysis (i.e., even if the differences are not proposed deviations, state the differences which involve the use of additional guidance from N-716).

Response

As identified above, the NMP1 application included an additional metric for identifying HSS piping (i.e., LERF). This is not considered a deviation, but rather an additional consideration based on lessons learned from the Code Case N-716 pilot plant applications. Two other similar items are noted below:

- A sensitivity study was conducted to assess the impact on delta risk when taking credit for enhanced Probability of Detection (POD). Enhanced POD was only credited for locations susceptible to thermal fatigue (Thermal Transients [TT] and/or Thermal Stratification, Cycling, and Striping [TASCS]). If TT and Intergranular Stress Corrosion Cracking (IGSCC) were both identified, no credit for enhanced POD was taken. This explains why the calculated risk impact analysis results (Core Damage Frequency (CDF) and LERF) in Table 3.4 of request no. 1ISI-003 sometimes differ in the "w/POD" and "w/o POD" columns.

- Consistent with previous risk-informed, safety-based (RIS_B) inspection program applications, including the Code Case N-716 pilot applications, locations that only receive a surface exam and are not subject to outside diameter attack were not included in the delta risk assessment.

RAI No. 2

Section 1.2 states that the updated probabilistic risk assessment (PRA) model meets the Capability Category II (CCII) supporting requirements (SRs) and combined CCII and CCIII SRs where both requirements are equivalent. The NRC staff has concluded that additional work may be needed beyond CCII in order for the PRA technical adequacy to be consistent with that determined to be acceptable for PRAs which supported the Electric Power Research Institute TR-112657 RI-ISI process. Please explain how the following three issues are addressed.

- *Supporting requirement IF-C3 (IFSN-A6) identifies the failure mechanisms that shall be evaluated to determine the susceptibility of each structure, system, and component (SSC) in a flood area to flood-induced failures. CCII identifies failure modes by submergence and spray as requiring identification but may not require assessment. CCIII requires identification and assessment of all failure modes including submergence, spray, jet impingement, pipe whip, and humidity, condensation, and temperature concerns. Risk Informed Inservice Inspection (RI-ISI) methods require that all SSC failures induced by a pipe break be considered. Please demonstrate that all SCCs failures that are induced by a pipe break are adequately assessed in your analysis.*

- *Supporting requirement IF-D3a (IFEV-A3) Category II permits grouping or subsuming flood initiated scenarios with existing plant initiating event groups. A Category III analysis which does not permit grouping is more consistent with previous RI-ISI analyses. If grouping of flood scenarios with other initiating events groups was done, please confirm that the subsumed flooding scenarios were identified during the flooding analysis and extracted during the RI-ISI analysis in order to insure that their contribution to the RI-ISI analysis was properly included.*

- *Supporting requirements IF-C6 (IFSN-A14) and IF-C8 (IFSN-A16) permit screening-out of flood areas and sources respectively based on, in part, the success of human actions to isolate and terminate the flood before equipment is damaged. RI-ISI methods require determination of the flood scenario with and without human intervention which corresponds to the capability Category III, i.e. scenarios are not screened out based on human actions. Therefore, a Category III analysis is more consistent with previous RI-ISI analyses. If capability Category II is used, high reliability of the human actions relied upon to screen out scenarios should be demonstrated using methods consistent with the supporting requirement IF-E5 (IFQU-A5) in the standard. Please re-evaluate the credit given to human actions to provide confidence that scenarios that might exceed the quantitative guideline are identified.*

Response

SR IF-C3
An assessment of the impact (initiating events and SSC failures) of submergence, spray, jet impingement, pipe whip and environmental impacts (e.g. humidity, temperature, condensation) were included in the internal flooding evaluation.

SR IF-D3a
There was no grouping of internal flooding initiating events with other internal initiating events. Flooding sources were either modeled as flood initiating events or screened. Those that were screened would have CDF/LERF values much less than 1E-06/1E-07 and as such would be of low safety significance for this application. Grouping was only done for certain pipe segments and sources when the breaks had similar impacts. The impacts were developed for the worst case, and timing was estimated for the largest flow when an operator action was involved.

SRs IF-C6 and IF-C8
The internal flooding PRA was done to meet Capability Category II, which ensures that only low risk segments are screened from the PRA model. Internal flooding initiating events are modeled to capture both operator success and failure impacts as applicable. These operator actions required to support

quantification of flood scenarios have a human reliability analysis performed in accordance with SR IF-E5.

For the limited cases where NMPNS did screen certain piping (e.g. smaller lines) and locations, as allowed per Capability Category II, there is: (1) significant time available before relevant equipment is impacted; (2) immediate detection with indication provided in the control room; and (3) procedural direction for operator response. For these scenarios, the combination of initiating frequency and highly reliable operator actions ensures that this piping is not high safety significant (i.e., CDF/LERF values are less than 1E-06/1E-07). This was confirmed by reviewing unscreened internal flooding events that are modeled in the PRA having the same three attributes noted above (i.e., significant time available, control room indication, and procedural direction for operator response). The review determined that the CDF/LERF results for these modeled cases are well below the 1E-06/1E-07 criterion for being high safety significant and are bounding relative to the events screened.

RAI No. 3

Please explain the difference between the "estimated" and "upper" bound values in Table 3.5 and how each is used. The explanation should clarify the differences between the 3E-2 "upper" value for "FWLOCA-OC" and the other "upper" bound values; and the difference between the "estimated" 3.4E-4 and the "upper" bound 2E-3 values for "Class 2 EC."

Response

In a traditional risk-informed ISI application (i.e., EPRI TR-112657), the risk assessment is conducted based on the results of the consequence assessment and failure potential assessment. Consequences are ranked as high (conditional core damage probability [CCDP] > 1E-04), medium (1E-06 < CCDP < 1E-04) and low (1E-06 < CCDP). Typically, piping that is ranked as high consequence uses the highest CCDP value from the plant-specific consequence assessment (e.g., Large Loss of Coolant Accident (LOCA)); piping that is ranked as medium consequence uses the upper bound for the medium consequence rank (i.e., 1E-04); and piping that is ranked as low consequence uses the upper bound for the low consequence rank (i.e., 1E-06). This process streamlines the delta risk assessment and provides stability in future updates. This philosophy has been carried forward to the Code Case N-716 application.

For the NMP1 RIS_B application, an upper bound value of 1E-04 (similar the medium consequence rank of EPRI TR-112657) is used in the delta risk assessment unless plant-specific estimates are determined to exceed this threshold value. As shown in Table 3.5 of request no. 1ISI-003, examples where this threshold value is exceeded are piping failures that result in a LOCA (2E-03), FWLOCA-OC (3E-02), ILOCA-OC (3E-3) and Class 2 EC (3.4E-04). As such, these plant-specific values are used in the delta risk assessment rather than 1E-04. Examples where the threshold value of 1E-04 is not exceeded are ILOCA (6E-06), PLOCA (2E-06), PLOCA-OC (1E-05) and Class 2 LSS (1E-04). For this piping, the delta risk assessment uses the threshold value of 1E-04 for the upper bound calculation and uses the estimated values to evaluate the more realistic case.

RAI No. 4

Applying an upper value of conditional core damage probability/conditional large early release probability (CCDP/CLERP) when adding an inspection to a location previously uninspected may result in an overestimate of the risk decrease associated with the new inspection. Please demonstrate that this nonconservative approach, if corrected in the evaluation of your proposed program, would not cause the delta risk guidelines to be exceeded.

Response

New examination locations were identified and were included in the delta risk estimate. The inspection selections for the original American Society of Mechanical Engineers (ASME) Section XI program and for the proposed Code Case N-716 program, and the difference between those selections, are provided in Table 3.4 of request no. 1ISI-003 under the columns entitled "Sec. XI," "RIS_B," and "Delta," respectively. The risk impact analysis included changes made to the original ASME Section XI inspections as a result of implementing Code Case N-716. The analysis results are displayed in the Delta column of Table 3.4 as either no change (represented by 0), an increase (represented by a positive number), or a decrease (represented by a negative number). A risk impact calculation performed using estimated CCDP and CLERP values yielded delta risk results that were similar to the results obtained using upper bound CCDP and CLERP values and met the EPRI acceptance criteria. In addition, Table 3.4 was reviewed for cases where the "RIS_B" selection value exceeded the "Sec. XI" selection value (represented by a positive number in the Delta column). This review determined that even if this delta was reduced to zero, Code Case N-716 acceptance criteria would still be met.

RAI No. 5

Please discuss your internal events PRA Peer Review Findings and Observations that are not resolved to date and may affect the planned RI-ISI Program.

Response

There are no unresolved PRA Peer Review findings that significantly impact the planned RIS_B program. As previously noted in Section 1.2 of request no. 1ISI-003, a summary of all the PRA Peer Review findings and an assessment of the impact of those findings on the PRA model were previously submitted to the NRC by NMPNS letter dated December 4, 2008. The findings were related primarily to documentation, and findings that affected the PRA model had negligible impact on the PRA results. This determination also applies to the RIS_B application of the PRA model.