December 3, 2004

Mr. James A. Spina
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC NOS. MC3272 AND MC3273)

Dear Mr. Spina:

By letter dated May 26, 2004, Constellation Energy Group Inc., submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54), to renew the operating licenses for the Nine Mile Point Nuclear Station (NMP), Units 1 and 2, for review by the U.S. Nuclear Regulatory Commission (NRC). The NRC staff is reviewing the information contained in the license renewal application (LRA) and has identified, in the enclosure, areas where additional information is needed to complete the review.

Based on discussions with Mr. Peter Mazzaferro of your staff, a mutually agreeable date for your response is within 30 days from the date of this letter. If you have any questions regarding this letter or if circumstances result in your need to revise the response date, please contact me at 301-415-1458 or by e-mail at nbl@nrc.gov.

Sincerely,
/RA/

N. B. (Tommy) Le, Senior Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-220 and 50-410

Enclosure: As stated

cc w/encl: See next page

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Dated: December 3, 2004, Accession No.: **ML043420049**

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NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2 (NMP1 and NMP2)
LICENSE RENEWAL APPLICATION (LRA)
REQUEST FOR ADDITIONAL INFORMATION (RAI)
RELATED TO:
3.1.2.4.3- Reactor Coolant System – Recirculation System

(1) The following RAIs are related to NMP1:

RAI 3.1.2.A.4-1
In section 3.1.2.A.4 of the LRA, the applicant lists the following environments which the NMP 1 Reactor Recirculation System components are exposed:
- o Closure bolting for non-borated water systems with operating temperatures equal to or greater than 212 degree F (degF)
- o Treated water, temperature < 140 degF, low flow
- o Treated water or steam, temperature > 482 degF
- o Treated water or steam, temperature > 482 degF, low flow

However, some of the components such as valves and piping are also exposed to air and hydraulic fluid. Please provide information as to why air and hydraulic fluid are not listed in the environment.

RAI 3.1.2.A.4-2

The applicant identifies cracking as an applicable aging effect for the recirculation system closure bolting, piping and fittings, recirculation pumps, and valves; however, the applicant has not considered cumulative fatigue damage for NMP1 piping as an aging effect. Please provide information relative to exclusion of cumulative fatigue damage for NMP 1 reactor recirculation piping.

RAI 3.1.2. A.4-3

In LRA Table 3.1.2.A-4, the applicant identifies cracking as an applicable aging effect for the recirculation system austenitic stainless steel components (piping and fittings, tubing, valve bodies, flow elements, thermowells, restricting orifices) and for the high-strength low-alloy steel primary pressure closure bolting exposed to reactor coolant water. The applicant also identifies this aging effect for cast stainless steel components exposed to reactor coolant water. The applicant identifies crack initiation and growth due to thermal and mechanical loading as an applicable aging effect for small-bore stainless steel piping and fittings and low-alloy steel pressure boundary closure bolting in the reactor recirculation system. Please provide information to indicate that, for NMP1, the applicant has no flaws evaluated in accordance with IWB-3600 “Analytical Evaluation of Flaws” under the inservice inspection program of ASME Code, Section XI since this would require a time-limited aging analysis (TLAA) under 10 CFR 54.21©).

(2) The following RAIs are related to both NMP1 and NMP2:

RAI 3.1.2.C.4-1

In LRA Tables 3.1.2.A-4 and 3.1.2.B-4, there are non-safety related (NSR) piping, fittings, and equipment of any material whose failure could affect safety-related equipment within the scope of license renewal. The applicant identified cracking and loss of material as aging effects requiring management. The aging management program to manage the aging effects is the Water Chemistry Control Program. Please provide additional information to indicate if these components in both NMP1 and NMP2 are to be covered in the One-Time Inspection Program for condition evaluation prior to the extended period of operation and to perform periodic inspection of the components under the preventative maintenance (PM) Program.

RAI 3.1.2.C.4-2

In LRA Table 3.1.2.A-1 for NMP 1 applicable to vessel drain line, the applicant identifies ASME Section XI Inservice Inspection as an aging management program to detect loss of material. Because of the size of the drain line, volumetric examination is not required by the Code. Please provide additional information on the aging management program on how it is applicable to reactor vessel drain line [due to its size and being mostly inaccessible] in the detection of loss of material under the inservice inspection program for NMP1 as well as NMP2.

RAI 3.1.2. C.4-3

In LRA Table 3.1.2.A-4, the applicant identifies loss of material as an aging effect for the reactor recirculation system high-strength low-alloy steel bolting exposed to air with operating temperatures in excess of 212 degF. However, under discussion for item numbers 3.1.1.A-26 and 3.1.1.B-26, the applicant states that the aging mechanism of loss of preload due to stress relaxation is not an aging effect/mechanism at NMP1 (and NMP2) for this environment. Please provide additional information on the effects of oxygenated water on the bolting material at temperatures >482 degF and a justification for excluding periodic inspection of the closure bolting for indication of loss of preload.

RAI 3.1.2.C.4-4 and RAI 3.1.2.C-4-5

In LRA Tables 3.1.1.A and 3.1.1.B, under item numbers 3.1.1.A-23 and 3.1.1.B-23 for cast authentic stainless steel (CASS) pump casing, the applicant identifies loss of fracture toughness due to thermal aging embrittlement as an applicable aging effect and inservice inspection as the aging management program. In order for the staff to properly evaluate these items, please (1) provide information regarding NMP1's and NMP2's plant-specific experience in ultrasonic examination of CASS components since

ultrasonic examination of cast austenitic stainless steel material with coarse grain structure is impractical, and
(2) provide justification for not considering cumulative fatigue damage as an aging effect for the reactor recirculation pump casing welds.

RAI 3.1.2.C.4-6

In LRA Table 3.1.1.A, item number 3.1.1.A-09, the applicant identifies stress corrosion cracking and cyclic loading as the aging effects for the isolation condenser and credits the Preventive Maintenance (PM) Program as the aging management program to manage the aging effects. The isolation condensers are part of the reactor coolant pressure boundary and, therefore, should be inspected in accordance with the ASME Code, Section XI. The PM program does not require volumetric examination to assure structural integrity of pressure boundary material or welds. Please provide additional information on how the aging effects of cracking in stainless steel tubes and in shell welds are managed through PM program that relies on visual inspection to prevent the loss of its intended function.

RAI 3.1.2 C.4-7

The applicant credits LRA Appendix B2.1.6, BWR Stress Corrosion Cracking Program, for mitigating intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel reactor coolant pressure boundary components, including piping four inches and greater nominal pipe size. The applicant also states that the BWR Stress Corrosion Cracking Program is based on industry guidelines approved by the NRC. Please provide information about its plant-specific experience related to IGSCC of the reactor coolant pressure boundary piping, mitigative actions taken, and the revised inspection schedules following the BWRVIP-75 guidelines. Please also provide information on implementation of hydrogen water chemistry and noble metal chemical application (NMCA) at both NMP1 and NMP2, and how this implementation has affected monitoring of water chemistry parameters.

RAI 3.1.2 C.4-8

In Table 3.1.1-A, under the discussion of item number 3.1.1.A-01 for feedwater nozzles, the applicant credits the BWR Feedwater Nozzle Program (Appendix B2.1.5) in addition to the Fatigue Monitoring Program because an enhanced inservice inspection program of NUREG-0619 was implemented at NMP1. The BWR Feedwater Nozzle Program is based upon an enhanced inservice inspection in accordance with the ASME Code, Section XI, 1989 Edition. However, in reviewing the program in Appendix B 2.1.5, the staff did not find any enhancement to the inservice inspection. Please provide additional information that would resolve this apparent inconsistency.

RAI 3.1.2.C.4-9

In LRA Tables 3.1.2.A-2 and 3.1.2.B-2 for orificed fuel support, the applicant states that there is no aging effect for the cast stainless steel fuel support in the treated water or steam environment at temperature exceeding 482 degree F and, therefore, has no aging management. This component is identified in the GALL under IV.B1.5.1 to be potentially susceptible to loss of fracture toughness due to thermal aging and neutron irradiation embrittlement. Furthermore, cumulative fatigue damage is also likely, thus requiring a TLAA for the period of extended operation. Please provide additional information that justify rationale for no aging management of this component.

RAI 3.1.2.C.4-10

In LRA Tables 3.1.2.A-1 and 3.1.2.B-1, the applicant states that thermal sleeves serve a pressure boundary function. However, the staff is not clear regarding the design of the thermal sleeve as a pressure boundary. Please provide additional information regarding the design of thermal sleeves and its pressure boundary function at both NMP1 and NMP2.

cc:

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