November 17, 2004

LICENSEE: Constellation Energy Group, Inc.

FACILITY: Nine Mile Point Nuclear Station, Units 1 and 2

SUBJECT: SUMMARY OF A MEETING HELD ON OCTOBER 28, 2004, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND THE CONSTELLATION ENERGY GROUP INC. CONCERNING THE REVIEW FOR THE NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC NOS. MC3272 AND MC3273)

The U.S. Nuclear Regulatory Commission staff and representatives of Constellation Energy Group Inc. (CEG or the applicant) held a drop-in meeting on October 28, 2004, to discuss questions pertaining to the Nine Mile Point Nuclear Station, Units 1 and 2 (NMP) license renewal application.

The meeting was useful in further clarifying the intent of the staff's questions and the applicant's proposed responses. On the basis of the discussion, the applicant was able to better understand the staff's questions. No staff decisions were made during the meeting, and the applicant agreed to provide information for clarification in their final responses.

Enclosure 1 provides a list of the meeting participants. Enclosure 2 contains a listing of the staff's draft questions and the applicant's corresponding proposed responses which were used during the telephone conference, including a brief description on the status of the items discussed. The applicant has had an opportunity comment on this summary.

/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

Enclosures: As stated

cc w/encls.: See next page

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FACILITY: Nine Mile Point Nuclear Station, Units 1 and 2

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CONSTELLATION ENERGY GROUP, INC.

OCTOBER 28, 2004

Participants	Affiliation
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John Fair	NRC
Peter Mazzaforro	Constellation Energy Group (CNG)
Mike Fallins	CNG

Enclosure 1

REVIEW OF LICENSE RENEWAL APPLICATION (LRA) FOR NINE MILE POINT UNITS 1 AND 2 (NMP 1 AND NMP 2)

October 28, 2004

The staff has previously sent the following draft questions via D-RAI 4.3.1-1, D-RAI 4.3.1-2, D-RAI 4.3.1-3, D-RAI 4.3.1-4, D-RAI 4.6.2-1, D-RAI 4.6.2-1 to the applicant. The applicants provided the staff with the proposed response on October 25, 2004. The staff reviewed the applicant's proposed responses and held a meeting with the applicant on October 28, 2004, to discussion the applicant's proposed responses. The staff draft questions, the applicants proposed responses, and the staff follow-up questions are as followed:

Section 4.3.1-1 Reactor Vessel Fatigue Analysis

D-RAI 4.3.1-1

Section 4.3.1 of the license renewal application indicates that the fatigue usage will be monitored at critical locations for NMP 1 and NMP 2. The application further indicates that these locations would include the components identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." Tables 4.3-3 and 4.3-4 list the Reactor Pressure Vessel locations that will be monitored by the Fatigue Monitoring Program (FMP). The application does not list all of the locations identified in NUREG/CR-6260 as locations that will be monitored FMP. Please clarify that all locations identified in NUREG/CR-6260 will be monitored by the FMP. Please provide a complete list of all locations that will be monitored by the FMP for NMP 1 and NMP 2.

CEG Response

All locations identified in NUREG/CR-6260, or their equivalent, will be monitored at NMP, unless no equivalent location exists. NMP 1 does not have a residual heat removal (RHR) system, so the location described as "RHR Return Line Class 1 Piping" for an older vintage BWR in NUREG CR/6260 does not exist. Also, for the NUREG CR/6260 location described as "Feedwater Line Class 1 Piping," there are no existing ASME Section III fatigue analyses since NMP 1 feedwater piping was designed to ASA B31.1-1955. As noted in LRA Section 4.3-4, the Feedwater/High Pressure Coolant Injection System piping has been identified as requiring additional analysis, which would consist of developing an ASME Section III type fatigue analysis for portions of the piping. Bounding location(s) for monitoring will be determined based on the ASME III type analysis. Table 1 below provides the correlation between the NUREG locations and the equivalent NMP 1 locations.

For NMP 2, all equivalent locations to those identified in NUREG CR/6260 for a "newer vintage BWR" will be monitored. Table 2 below provides the correlation between the NUREG locations and the equivalent NMP 2 locations.

All locations currently identified as requiring monitoring for fatigue are listed in LRA Tables 4.3-3, 4.3-4, 4.3-5, and 4.3-7. Additional locations may be identified based on ASME Section III-type fatigue analyses to be performed for the systems listed in Section 4.3-4.

Table 1 – NMP 1 Equivalent Locations to NUREG/CR-6260 Environmental Fatigue Sample Locations

NUREG CR/6260 Location for an Older Vintage BWR	NMP 1 Equivalent Location	LRA Location
Reactor vessel shell and lower head	Bottom Head – Vessel Junction	Table 4.3-3
Reactor Vessel Feedwater Nozzle	Feedwater Nozzles	Table 4.3-3
Reactor Recirculation Piping (Including Inlet and Outlet Nozzles)	Recirculation Outlet Nozzle Recirculation Inlet Nozzles	Table 4.3-3
Core Spray Line Reactor Vessel Nozzle and Associated Class 1 Piping	Core Spray Nozzle Core Spray Nozzle Safe End	Table 4.3-3
RHR Return Line Class 1 Piping	none	
FW Line Class 1 Piping	To be determined. No Class 1 fatigue analysis at this time.	Section 4.3-4

Status of the discussion:

The staff finds that the applicant's proposed response provides the information requested by the D-RAI; however, the applicant should discuss why, in Table 1-NMP 1 Equivalent Locations to NUREG/CR-6260, there is no NMP 1 Equivalent location for the RHR Return Line Class 1 piping. The applicant representatives stated that this information will be provided in the final response.

Section 4.3.1-2 Reactor Vessel Fatigue Analysis

D-RAI 4.3.1-2

Tables 4.3-3 and 4.3-4 of the license renewal application indicate that stress based fatigue monitoring will be used to track the fatigue usage for the NMP 1 and NMP 2 feedwater nozzles. Please describe the method used to estimate the fatigue usage of these nozzles prior to implementation of the stress based fatigue monitoring.

CEG Response

The feedwater nozzles that will be monitored using stress-based fatigue techniques, the initial CUF is determined based on a linear projection of the design basis CUF. For example, if the design CUF for an SBF component is 0.70 and the FMP is implemented after 20 years of plant operation, the initial CUF is estimated to be (20/40) *0.70 = 0.35. Continued CUF monitoring into the future will be used to demonstrate the conservatism of this estimate (i.e., show that the rate of actual CUF accumulation is less than the rate of design basis fatigue accumulation).

Documents Used to Validate the Response

Engineering Judgment

Answer supplied by Gary Stevens of Structural Integrity Associates (Fatigue Pro Vendor)

Status of the discussion:

The staff reviewer stated that he has reviewed the proposed response; however some of the transients listed in LRA Tables 4.3-1 and 4.3-2 appear to have accumulated a larger number of cycles than would be projected using a linear assumption based on plant operating time. The applicant representatives stated that they will take a look and provide the needed information accordingly.

Section 4.3.1-3 Reactor Vessel Fatigue Analysis

D-RAI 4.3.1-3

Table 4.3-1 of the license renewal application lists the design transients for NMP 1. Note 2 to the table indicates that a number of the transients were not counted/monitored prior to 2000. The note contains the statement: "Data listed for allowable design transients are incremental values for the balance of the original license term." The intent of this statement is not clear. Please provide additional clarification. Indicate the method used to estimate the number of cycles prior to 2000 for those design transients identified by Note 2.

CEG Response

These cycles are incremental from the year 2000 onwards because it was realized in 1999 that certain transients related to operation of the emergency cooling system affecting the reactor recirculation nozzles (N1 and N2 nozzles) had not been accounted for in the original fatigue calculations (Reference 1). The reason for fatigue not being considered for these nozzle's was that thermal transients to the feedwater nozzle was considered to bound the recirculation nozzles from a fatigue standpoint.

An ASME stress and fatigue evaluation of the N1 and N2 nozzles was performed (Reference 2). Seven different transients affecting the N1 or N2 nozzle were evaluated, but only the bounding transient for each nozzle was analyzed. These were: (1) emergency condenser (EC) initiation into an isolated recirculation loop, other loops running, and (2) EC initiation in an idle loop, other loops in natural circulation.

This analysis determined the fatigue usage resulting from 30 cycles to be 0.065 for the N1 nozzle (nozzle bounds safe end) and 0.005 for the N2 nozzles (safe end bounds nozzle). Use of 30 cycles was an arbitrary number chosen for the analysis as one that would easily bound the number of cycles in the 2000-2009 period, because historically EC initiation events have been infrequent. The actual number of allowable cycles of the bounding transients for a fatigue usage of 0.8 are 369 for the N1 nozzle and 4800 for the N2 nozzle.

The values in the "Designed Cycles Analyzed" column are the allowable numbers of transients from 2000, when counting of these transients began, to 2009, when NMP 1's original operating license expires. The value in the "Cycles to August 2003" column are from 2000 to 2003. Actual cycles prior to 2000 have not been reconstituted because as mentioned above, the initiation of the emergency condenser system has been infrequent relative to the large number of cycles allowed. Since initial plant startup the actual number of times the emergency condensers have been initiated is estimated to be less than 20 and the number of instances in which those EC initiations occurred in an isolated loop is less.

Documents Used to Validate the Response

1. DER-1999-3551

2. Calculation S0VESSELM26 Revision 01 Attachment E (Calculation prepared by MPR Associates entitled "Thermal Transient Analysis of N1 and N2 Nozzles"

Status of the discussion:

The staff indicated that the response should address all of the design transients that are covered by Footnote 2. The applicant agreed to provide the needed information.

Section 4.3.1-4 Reactor Vessel Fatigue Analysis

D-RAI 4.3.1-4

Table 4.3-2 of the license renewal application lists the deign transients for NMP 2. The Table does not list the daily reduction to 75% power that is listed in USAR Table 3.9B-1. Please explain why this transient was not included in Table 4.3-2 of the application.

Response:

The transient of "Daily Reduction To 75% Power" has been combined with the transient "Weekly Reduction to 50% Power" for counting purposes. These transients have historically not been counted separately at NMP 2. The "Daily Reduction To 75% Power" transient has an allowable number of cycles of 10,000 per USAR Table 3.9B-1, while the Weekly Reduction to 50% Power" transient has an allowable number of cycles of 2000 per USAR Table 3.9B-1. Therefore, allowing a combined number of transients of 2000 for both levels of power reduction is conservative. The transient listed in LRA Table 4.3-2 as "Power Change $ 25%" combines the transients listed in USAR Table 3.9B-1 as "Daily Reduction to 75% Power" and "Weekly Reduction to 50% Power."

Documents Used to Validate the Response

1. LRA Table 4.3-4 (Note 1 explains).

2. NIP-REL-06, Revision 01, "Fatigue Monitoring Program," Attachment 2: Unit 2 Recordable Plant Events" (Note 4 explains).

3. NER-1S-035 Revision 00, "Report On System Review and Recommendations for a Transient and Fatigue Monitoring System at the Nine Mile Point Nuclear Station,"

SIR-03-140 Revision No. 01, February 2004, Structural Integrity Associates, Inc., Greenwood Village Colorado (Table 2-2).

Status of the discussion:

The staff stated that the applicant proposed response is adequate, and no followup request for information is needed at this time.

Section 4.6.2-1 Containment Liner Plate, Metal Containment & Penetrations Fatigue Analysis

D-RAI 4.6.2-1

Section 4.6.2 of the license renewal application addresses the torus attached piping for NMP 1. The application indicates that the existing fatigue usage factors are less than 0.5 and, therefore, the fatigue usage factors will remain less than 1.0 for sixty years of plant operation. Please identify the location containing the bounding fatigue usage for the torus attached piping. List the design transients, including the number used in the fatigue analysis and associated fatigue usage, for this bounding location. Provide the number of these design transients that have been experienced since initial plant operation.

CEG Response

The statement that "the existing fatigue usage factors are less than 0.5" is based on the results of a generic fatigue study of torus attached piping for all BWRs (Reference 1). This study determined fatigue usage for specific piping locations at each plant, but not all locations for all plants. Reference 1 indicates the stress results for the most limiting piping systems and locations were selected for each plant, so the remainder of piping systems for each plant should have lower fatigue usage. However, the conclusion that the fatigue usage factors are less than 0.5 was based on the conclusion of Reference 1 that 100% of SRV and torus attached piping (for all BWRs) had a 40-year CUF less than 0.5.

The generic analysis in Reference 1 assumed the following transients:

1. Periodic SRV actuations over the life of the plant with the total number of actuations determined for the specific plant. One combined thermal and anchor motion load is assumed to act during each initiation. (For NMP 1, Reference 2 indicates up to 4500 stress cycles can be expected due to SRV discharge. The equivalent of five full stress cycles per actuation is typically assumed per SRV discharge, corresponding to a design assumption of 900 SRV discharges over the 40 year original operating license).

2. Five operating basis earthquakes.

3. One accident condition consisting either of a design basis accident (DBA) or intermediate break accident/small break accident (SBA/IBA) which includes: (i) one combined thermal and anchor motion loading, (ii) operating basis earthquake (OBE) and safe shutdown earthquake (SSE) earthquake stresses, and (iii) periodic SRV actuations during IBA/SBA with the total number of actuations determined for the specific plant.

Two NMP 1 specific locations were analyzed in Reference 1, consisting of one small-bore location and one large-bore location. The highest usage factor for the two NMP 1 locations analyzed was for the large bore location, the 12-inch core spray suction line for Pump #111 that enters the torus at penetration XS-337, which has a cumulative usage factor (CUF) of 0.036 for 40-years based on the case of normal operating conditions plus a small break accident/intermediate break accident condition (NOC+SBA/IBA). For the case of NOC+DBA, this location had a cumulative usage factor of 0.001 for 40-years. The small bore location is the 3-inch containment spray line that enters the torus at penetration XS-326; with calculated 40-year CUFs of 0.012 for NOC+DBA and 0.000 for NOC+IBA/SBA.

Since NMP 1 has not experienced a DBA, an SBA/IBA or an earthquake, the primary contributor to actual fatigue usage is SRV discharge during normal operation. NMP 1 has historically not counted SRV actuations, so the number of such transients experienced is not available. However, conservatively multiplying the 40-year maximum CUF of 0.036 by 1.5 yields a CUF of 0.054 for 60-years. This is conservative because the calculated 40-year CUF includes the effects of accident and earthquake loadings that have not been experienced during the original operating license period to date.

1. MPR-751, "Augmented Class 2/3 Fatigue Evaluation Method and Results for Typical Torus Attached and SRV Piping Systems," November 1982, MPR Associates, transmitted via letter from H. C. Pfefferlen (GE) to D. B. Vassalo, NRC, Re: "Fatigue Evaluation Method and Results for Torus and SRV Piping for Mark I Plants," dated November 30, 1982.

2. Teledyne Engineering Services, TR-5320-2, "Mark I Containment Program, Plant Unique Analysis Report of the Torus Attached Piping for Nine Mile Point Unit 1 Nuclear Generating Station," April 1984.

Status of the discussion:

The staff stated that the applicant proposed response is adequate, and no follow-up request for information is needed at this time.

Section 4.6.4 Containment Liner Analysis for NMP 2

RAI 4.6.4-1

Section 4.6.4 of the license renewal application addresses the NMP 2 containment liner analysis. The application indicates that a revised analysis will be performed prior to the period of extended operation that will demonstrate that the 60-year CUF values for all controlling locations will remain less than 1.0. Please provide the current design CUF values for the controlling containment liner locations. Explain the basis for the statement that the revised analysis will demonstrate that the 60-year CUF values for all controlling locations will remain less than 1.0, given that the revised analysis has not been completed.

Response

The design cumulative usage factor for the liner for the original 40-year operating life of the containment is 0.054. The fatigue analysis covered the liner in the suppression pool area. For different loading conditions, the peak stresses occurred at different elevations, but for the purposes of determining fatigue usage, they were assumed to occur at the same elevation. The elevations of the peak stresses were 300 inches above the basemat for the operating-basis earthquake and safe-shutdown earthquake, 0.0 inches above the basemat for small break accident (SBA) plus intermediate break accident (IBA) pressure loading and design basis accident (DBA) pressure loading. For DBA and SBA/IBA temperature loads, the peak stress occurred at 44 inches above the basemat. Stress due to SRV loading was applied uniformly to the liner.

The table below shows the load events considered in the fatigue analysis, the number of events and cycles assumed in 40-years, and the fatigue usage corresponding to each event. SRV actuation is the primary contributor to fatigue usage. It can be seen that the fatigue usage corresponding to a 60-year life could easily be projected to remain far less than 1.0.

<u>Status of the discussion</u>:

The staff stated that the applicant proposed response is adequate, however, the applicant should complete the TLAA, since it has already performed the evaluation, rather than committing to complete the TLAA at a later date. There was no follow-up request for information is needed at this time.

Load Event	Events/40-years	Stress Cycles/Event	Fatigue Usage
Design Basis Loss of Coolant Accident (DBA)	1	1	~0 pressure load 0.0016 temperature load
Operating Basis Earthquake (OBE)	5	20	~0
Design Basis Earthquake (DBE)	1	20	~0
Safety Relief Valve Actuation (SRV)	4943	10	0.05
SRV+Seismic Event	15	10	Included with other load cases
SRV+SBA/IBA	10	10	Included with other load cases
Small or Intermediate Break Accident (SBA/IBA)	20	1	0.002 pressure load 0.001 temperature load
Operating Temperature	400	1	Not calculated since the criteria of ASME III, Div I, NB 3222.4(d) are met.
Operating Pressure	100	1	Not calculated since the criteria of ASME III, Div I, NB 3222.4(d) are met.

<u>Documents Used to Validate the Response</u>

1. Calculation EM3.23 Revision 2, "Primary Containment Liner Analysis," Stone and Webster Engineering Corporation," January 3, 1985.

Nine Mile Point Nuclear Station, Unit Nos. 1 and 2

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