Constellation Energy
Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, New York 13093

December 6, 2004
NMP1L 1891

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

SUBJECT: Nine Mile Point Units 1 and 2
Docket Nos. 50-220 and 50-410
Facility Operating License Nos. DPR-63 and NPF-69

License Renewal Application – Responses to NRC Requests for Additional Information Regarding Time-Limited Aging Analyses (TAC Nos. MC0691 and MC0692)

Gentlemen:

By letter dated May 26, 2004, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted an application to renew the operating licenses for Nine Mile Point Units 1 and 2.

In a letter dated November 10, 2004, the NRC requested additional information regarding the time-limited aging analyses (TLAAs) that are described in Sections 4.3 and 4.6 of the License Renewal Application. The NMPNS responses to these requests for additional information are provided in Attachment 1. This letter contains no new regulatory commitments.

If you have any questions about this submittal, please contact Peter Mazzaferro, NMPNS License Renewal Project Manager, at (315) 349-1019.

Very truly yours,

James A. Spina
Vice President Nine Mile Point

JAS/DEV/jm

A107

STATE OF NEW YORK :
: TO WIT:
COUNTY OF OSWEGO :

I, James A. Spina, being duly sworn, state that I am Vice President Nine Mile Point, and that I am duly authorized to execute and file this supplemental information on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this submittal are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 6th day of December, 2004.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

12/6/04
Date

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified in Oswego County
Commission Expires 10/25/05

Attachment:
1. Responses to NRC Requests for Additional Information (RAI) Regarding the Time-Limited Aging Analyses (TLAAs) Described in Sections 4.3 and 4.6 of the License Renewal Application

cc: Mr. S. J. Collins, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR
Mr. N. B. Le, License Renewal Project Manager, NRR
Mr. J. P. Spath, NYSERDA

ATTACHMENT 1

Nine Mile Point Nuclear Station

Responses to NRC Requests for Additional Information (RAI)

Regarding the Time-Limited Aging Analyses (TLAAs) Described in

Sections 4.3 and 4.6 of the License Renewal Application

This attachment provides Nine Mile Point Nuclear Station, LLC (NMPNS) responses to the requests for additional information contained in the NRC letter dated November 10, 2004. For each identified License Renewal Application (LRA) section, the NRC RAI is repeated, followed by the NMPNS response for Nine Mile Point Unit 1 (NMP1) and/or Nine Mile Point Unit 2 (NMP2), as applicable.

LRA Section 4.3.1, Reactor Vessel Fatigue Analysis

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 NMP1 and NMP2. 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 by the 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 NMP1 and NMP2.

Response

All locations identified in NUREG/CR-6260, or their equivalent, will be monitored at NMP1 and NMP2. NMP1 does not have a Residual Heat Removal (RHR) system, so the location described as "RHR Return Line Class 1 Piping" for an older vintage boiling water reactor (BWR) plant in NUREG/CR-6260 does not exist. However, the NMP1 shutdown cooling return line is analogous to the RHR return line, for the purposes of identifying locations subject to environmental fatigue analysis. Also, for the NUREG/CR-6260 location described as "Feedwater Line Class 1 Piping," and for the Reactor Recirculation System piping with the exception of the reactor vessel inlet and outlet nozzles, there are no existing American Society of Mechanical Engineers (ASME) Section III fatigue analyses since NMP1 piping was designed to American Standards Association (ASA) B31.1-1955. As noted in LRA Section 4.3.4, the

Feedwater/High Pressure Coolant Injection System piping and the Reactor Recirculation System piping (and associated Shutdown Cooling System lines) have 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 Section III-type analysis. Table 1 below provides the correlation between the NUREG/CR-6260 locations and the equivalent NMP1 locations.

For NMP2, all locations equivalent to those identified in NUREG/CR-6260 for a "newer vintage BWR" will be monitored. Table 2 below provides the correlation between the NUREG/CR-6260 locations and the equivalent NMP2 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 LRA Section 4.3.4.

Table 1 (RAI 4.3.1-1)
NMP1 Locations Equivalent to NUREG/CR-6260
Environmental Fatigue Sample Locations

NUREG/CR-6260 Location for an Older Vintage BWR	NMP1 Equivalent Location	LRA Location
Reactor Vessel Shell and Lower Head	Bottom Head – Vessel/Head 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 Nozzles; Recirculation Inlet Nozzles Shutdown cooling return line tee to recirculation piping(1)	Table 4.3-3 Section 4.3.4
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	Shutdown cooling return line to reactor recirculation line. Exact location to be determined by an ASME Section III-type fatigue analysis of this piping.	Section 4.3.4
Feedwater Line Class 1 Piping	Exact location to be determined by a ASME Section III-type fatigue analysis of this piping.	Section 4.3.4

(1) NUREG/CR-6260 considered the RHR return line tee with the reactor recirculation piping to be part of the reactor recirculation piping. For the sample older vintage BWR plant considered in the NUREG, the RHR return line tee was found to be the limiting location in the reactor recirculation piping. The limiting location will be determined once the ASME Section III-type fatigue analysis is completed for the reactor recirculation piping.

Table 2 (RAI 4.3.1-1)
NMP2 Locations Equivalent to NUREG/CR-6260
Environmental Fatigue Sample Locations

NUREG/CR-6260 Location for a Newer Vintage BWR	NMP2 Equivalent Location	LRA Location
Reactor Vessel Shell and Lower Head	Control Rod Drive (CRD) Penetration, Stub Tube (Inconel portion)	Table 4.3-4
Reactor Vessel Feedwater Nozzle	Feedwater Nozzle – Low Alloy Steel Nozzle Body; Feedwater Nozzle – Stainless Steel Safe End	Table 4.3-4
Reactor Recirculation Piping (Including Inlet and Outlet Nozzles)	Recirculation Inlet Nozzle (N2) Inconel Safe End; Recirculation Outlet Nozzle (N1) Low Alloy Steel Nozzle-to-Shell Junction RHR Return Line Tee, Loops A and B	Table 4.3-4 Section 4.3.4
Core Spray Line Reactor Vessel Nozzle and Associated Class 1 Piping	Core Spray (N16) Nozzle Inconel Safe End	Table 4.3-4
Residual Heat Removal Nozzles and Associated Class 1 Piping	Limiting location of RHR Return Line Class 1 piping	Section 4.3.4
Feedwater Line Class 1 Piping	Feedwater Primary Containment North Loop Node 267; Feedwater Primary Containment South Loop Node 210; Feedwater Secondary Containment Node 590	Table 4.3-5

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 NMP1 and NMP2 feedwater nozzles. Please describe the method used to estimate the fatigue usage of these nozzles prior to implementation of the stress based fatigue monitoring.

Response

Since the actual number of cycles for certain types of events, such as startups and shutdowns, is a high percentage of design cycles (i.e., more than would be estimated by a linear projection based on years of operation), the fatigue usage to date will be determined based on the number of cycles as a percentage of design cycles for each type of event. Essentially, a cycle-based fatigue (CBF) calculation will be performed for cycles accumulated to date and the result used as the baseline value. Continued cumulative usage factor (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).

RAI 4.3.1-3

Table 4.3-1 of the license renewal application lists the design transients for NMP1. 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.

Response

The "Design Cycles Analyzed" for the noted transients in Table 4.3-1 are incremental from the year 2000 and beyond due to the discovery in 1999 that certain transients related to operation of the Shutdown Cooling and Emergency Cooling systems affecting the reactor recirculation nozzles (N1 and N2 nozzles) had not been accounted for in the original fatigue calculations. The reason for fatigue not being considered for these nozzles was that thermal transients to the feedwater nozzles were considered to bound the recirculation nozzles from a fatigue standpoint.

An ASME stress and fatigue evaluation of the N1 and N2 nozzles was performed. 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 into an idle loop, other loops in natural circulation.

These analyses determined the fatigue usage resulting from 30 cycles of the bounding transient to be 0.065 for the N1 nozzle (nozzle bounds safe end) and 0.005 for the N2 nozzle (safe end bounds nozzle). The use of 30 cycles was an arbitrary number chosen for the analyses 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 is 369 for the N1 nozzle and 4,800 for the N2 nozzle.

The numbers of cycles listed for the other five transients covered by Note 2 to Table 4.3-1 were also arbitrarily chosen to easily bound the cycles during the 2000-2009 period.

The "Emergency Condenser Initiation Into Isolated Loop" transient is the bounding transient for the N1 nozzle, and bounds the "Unisolation of an Isolated Loop" transient. However, if the fatigue usage was assumed to be the same for both transients, the fatigue usage resulting from 30 cycles of each transient, for a total of 60, would be 0.13 compared to an allowable value of 0.8. The other five transients covered by Note 2 apply to the N2 nozzle and are bounded by the transient described as "Emergency Condenser Initiation into Idle Loop." The total number of design cycles of these five transients is 1,090. If all five transients affecting the N2 nozzle had equal fatigue usage per occurrence of the bounding transient, the fatigue usage would be 0.18 for 1,090 cycles.

The values in the "Designed Cycles Analyzed" column of LRA Table 4.3-1 are the allowable numbers of transients from 2000, when counting of these transients began, to 2009, when the NMP1 current operating license expires. The values in the "Cycles to August 2003" column of LRA Table 4.3-1 are from 2000 to 2003. Actual cycles prior to 2000 have not been reconstituted because, as mentioned above, the initiation of the EC system has been infrequent relative to the large number of cycles allowed. Since initial plant startup, the actual number of times the ECs have been initiated is estimated to be less than 20, with an even fewer number of instances in which the EC initiations occurred into an isolated or idle loop.

RAI 4.3.1-4

Table 4.3-2 of the license renewal application lists the design transients for NMP2. 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 50% Power" for counting purposes. These transients have historically not been counted separately at NMP2. The "Daily Reduction to 75% Power" transient has an allowable number of cycles of 10,000 per NMP2 Updated Safety Analysis Report (USAR) Table 3.9B-1, while the Weekly Reduction 50% Power" transient has an allowable number of cycles of 2,000 per USAR Table 3.9B-1. Therefore, allowing a combined number of transients of 2,000 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 50% Power."

LRA Section 4.6.2, Torus Attached Piping Analysis (NMP1 Only)

RAI 4.6.2-1

Section 4.6.2 of the license renewal application addresses the torus attached piping for NMP1. 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.

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. The Reference 1 study indicates that 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 the Reference 1 study that 100% of safety relief valve (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:

- 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 NMP1, the Mark I containment plant unique analysis (LRA Reference 4.8-61) indicates that up to 4,500 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.)

- Five operating basis earthquakes.

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

Two NMP1 specific locations were analyzed in the Reference 1 study, consisting of one small-bore location and one large-bore location. The highest usage factor for the two NMP1 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 CUF of 0.036 for 40 years based on the case of normal operating conditions (NOC) plus a SBA/IBA condition (NOC+SBA/IBA). For the

case of NOC+DBA, this location had a CUF 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 a calculated 40-year CUF of 0.012 for NOC+DBA and 0.000 for NOC+SBA/IBA.

Since NMP1 has not experienced a DBA, a SBA/IBA, an OBE, or an SSE, the primary contributor to actual fatigue usage is SRV discharge during normal operation. NMP1 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.

References (RAI 4.6.2-1)

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 (General Electric) to D.B. Vassalo, NRC, re: Fatigue Evaluation Method and Results for Torus and SRV Piping for Mark I Plants, dated November 30, 1982 (referenced in LRA Section 4.6.2)

LRA Section 4.6.4, Containment Liner Analysis (NMP2 Only)

RAI 4.6.4-1

Section 4.6.4 of the license renewal application addresses the NMP2 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 CUF 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 OBE and SSE, and 0.0 inches above the basemat for SBA plus IBA pressure loading and 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.

Table 1 below shows the load events considered in the fatigue analysis, the number of events and cycles per event assumed in 40 years, and the fatigue usage corresponding to each event. SRV

actuation is the primary contributor to fatigue usage. The number of SRV actuations assumed for the original 40-year life is 4,943. Actual SRV actuations are occurring at a far lower rate, with 189 SRV actuations recorded by December, 2002. Since NMP2 commenced operation in 1986, the plant has been in operation for about one quarter of its anticipated 60-year life. Multiplying 189 SRV actuations by four yields a lifetime number of actuations of 756, which is far less than the 4,943 actuations assumed in the original design. Other events that contribute to the CUF, such as a DBA, SBA, IBA, OBE, and SSE, have not occurred at all to date at NMP2. Therefore, since the trend for the number of occurrences of the events which contribute to the NMP2 containment liner CUF indicates that the numbers of design cycles assumed for 40 years is unlikely to be exceeded during a 60-year life, a simple projection to 60 years performed by multiplying the 40-year CUF by 1.5 is conservative. The projection yields a 60-year CUF of 0.081 (0.054 x 1.5 = 0.081). The NMP2 containment liner fatigue analysis has, therefore, been projected in accordance with 10CFR54.21(c)(1)(ii).

Table 1 (RAI 4.6.4-1)
NMP2 Containment Liner Fatigue Analysis
Loading Events and Fatigue Usage

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
Safe Shutdown Earthquake (SSE)	1	20	~0
Safety Relief Valve Actuation (SRV)	4,943	10	0.05
SRV+Seismic Event	15	10	Included with other load cases
Small or Intermediate Break Accident (SBA/IBA)	20	1	0.002 pressure load; 0.001 temperature load
SRV+SBA/IBA	10	10	Included with other load cases
Operating Temperature	400	1	Not calculated since the criteria of ASME Section III, Div I, NB 3222.4(d) are met
Operating Pressure	100	1	Not calculated since the criteria of ASME Section III, Div I, NB 3222.4(d) are met