Constellation Energy
Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, NY 13093

December 20, 2005
NMP1L 2009

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 – Annual Update Information Required by 10CFR54.21(b) (TAC Nos. MC3272 and MC3273)

Gentlemen:

By letter dated May 26, 2004, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted a License Renewal Application (LRA) for the operating licenses of Nine Mile Point Units 1 and 2. In accordance with 10CFR 54.21(b), Nine Mile Point Nuclear Station (NMPNS) is required to submit a summary of the current licensing basis (CLB) changes that have occurred during the NRC review of the Application that materially affects the contents of the Application, including the Updated Final Safety Analysis Report (UFSAR) supplement (Unit 1) and Updated Safety Analysis Report (USAR) supplement (Unit 2).

NMPNS has completed a review of pertinent documents, including the UFSAR and USAR supplements, and identified changes that materially affect the Application.
Attachment 1 provides a description of each of the changes to the Application due to modifications. Attachment 2 provides Application changes due to new or revised analyses.

If you have any questions about this submittal, please contact David Dellario, NMPNS License Renewal Project Manager, at (315) 349-7141.

Very truly yours,

Timothy J. O'Connor
Plant General Manager

JAS/MSL/sac

A107

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

I, Timothy J. O'Connor, being duly sworn, state that I am Nine Mile Point Plant General Manager, and that I am duly authorized to execute and file this 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.

Timothy J. O'Connor
Plant General Manager

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 20th day of December, 2005.

WITNESS my Hand and Notarial Seal:

TONYA L. JONES
Notary Public in the State of New York
Oswego County Reg. No. 01JO6083354
My Commission Expires 11/12/2006

Tonya L. Jones
Notary Public

My Commission Expires: 11/12/2006
Date

Attachments:
1: Application changes due to modifications.
2: Application changes due to new or revised analyses.

cc: Mr. S. J. Collins, NRC Regional Administrator, Region I
Mr. L. M. Cline, NRC Senior Resident Inspector
Mr. T. G. Colburn, Senior Project Manager, NRR
Mr. N. B. Le, License Renewal Project Manager, NRR
Mr. J. P. Spath, NYSERDA

ATTACHMENT 1 to NMP1L 2009

The following are nine (9) modifications to NMP1 and NMP2, since the submittal of the initial LRA, that materially affect the Amended License renewal Application (ALRA).The first five (5) of these modifications are for NMP1 and the last four (4) are for NMP2.

Revisions to the existing ALRA are shown with *italics* for additions and ~~strikethroughs~~ for deletions.

MODIFICATION 1 – NMP1

A Zinc Injection System was installed at NMP1. The purpose is to inject zinc ions into the reactor coolant via the Feedwater System to reduce the corrosion of stainless steel surfaces in the reactor coolant system. This lowers radiation levels due to ^{60}Co deposition. The NSR system is not within scope of license renewal since it does not meet any of the three scoping criteria which would bring it into scope as follows:

54.4(a)(1) – The system is non-safety-related (NSR); therefore, this criterion does not apply.

54.4(a)(2) – 1. The system does not contain any NSR components credited in the current licensing basis to accomplish safety functions (i.e., protection from missiles, overhead cranes, flooding, or high energy line breaks).

2. The system does not contain NSR components that are within the boundary between a connection to safety-related components and the first seismic or equivalent anchor.

3. The system does not contain any NSR components that could spatially interact (via spray, leakage, jet impingement, or provision of any other harsh environment) with safety-related components.

54.4(a)(3) – The system does not contain any NSR components that are relied upon to provide compliance with the regulations for fire protection, environmental qualification, anticipated transients without scram, or station blackout.

The change to the ALRA as a result of the installation of this system is its addition to Table 2.2-1, NMP1 Plant Level Scoping Results, on ALRA p. 2.2-6 as shown below.

Mechanical Systems		
System or Commodity	**Within Scope of License Renewal?**	**Comments**
Turbine Building HVAC System (Section 2.3.3.A.26)	Yes	
Zinc Injection System	No	

MODIFICATION 2 – NMP1

A NMP1 Containment Spray pump was replaced. The material of the original pump casing was gray cast iron that has been replaced with stainless steel. As a result of this modification, the changes to ALRA p. 3.2-38 are as shown below.

Table 3.2.2.A-1 Engineered Safety Features Systems
NMP1 Containment Spray System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Pumps (cont'd)	PB (cont'd)	Gray Cast Iron	Raw Water	Loss of Material	Open-Cycle Cooling Water System Program Selective Leaching of Materials Program	VII.C1.5-a	3.3.1.A-29	A
			Treated Water, temperature < 140°F, Low Flow	Loss of Material	One-Time Inspection Program Water Chemistry Control Program	VII.C2.3-a	3.3.1.A-15	E
					Selective Leaching of Materials Program	VII.C2.3-a	3.3.1.A-29	A
		Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	Loss of Material	One-Time Inspection Program Water Chemistry Control Program	VIII.E.5-b	3.4.1.A-02	D

Modification 3 – NMP1

The carbon steel pressure safety valve that serves as the air release valve for the NMP1 diesel-driven fire pump was replaced with a cast iron model. The resultant changes to the ALRA from this modification are on p. 3.3-144 as shown below.

Table 3.3.2.A-8 Auxiliary Systems
NMP1 Fire Detection and Protection System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves	LBS PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Air	Loss of Material	One-Time Inspection Program	VII.H2.2-a	3.3.1.A-05	A
			Dried Air or Gas	None	None			None
			Raw Water, Low Flow	Loss of Material	Fire Water System Program	VII.G.6-b	3.3.1.A-21	A
		Copper Alloys (Zinc ≤15%)	Raw Water, Low Flow	Loss of Material	Fire Water System Program	VII.G.6-b	3.3.1.A-21	A
		Copper Alloys (Zinc > 15%) and Aluminum Bronze	Raw Water, Low Flow	Loss of Material	Fire Water System Program	VII.G.6-b	3.3.1.A-21	A
					Selective Leaching of Materials Program	VII.C1.2-a	3.3.1.A-29	A
		Gray Cast Iron	Raw Water, Low Flow	Loss of Material	Fire Water System Program	VII.G.6-b	3.3.1.A-21	A
					Selective Leaching of Materials Program	VII.C1.5-a	3.3.1.A-29	C

MODIFICATION 4 – NMP1

The carbon steel piping and valves associated with one of the main NMP1 Feedwater Pump's lube oil system were replaced with stainless steel. The resultant ALRA changes from this modification apply to pp. 3.4-40 and 3.4-42 for Piping and Fittings and Valves, respectively, and are as shown below and on the following page.

Table 3.4.2.A-2 Steam and Power Conversion System
NMP1 Feedwater/High Pressure Coolant Injection System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings	LBS SIA	Carbon Steel, Low Alloy Steel (Yield Strength < 100 Ksi)	Lubricating Oil	None	None			None
			Treated Water or Steam, temperature ≥212°F, but < 482°F	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CRF 54.21(c)	VIII.D2.1-c	3.4.1.A-01	A
				Loss of Material	Flow-Accelerated Corrosion Program	VIII.D2.1-a	3.4.1.A-06	A
				Loss of Material (cont'd)	One-Time Inspection Program Water Chemistry Control Program	VIII.D2.1-b	3.4.1.A-02	B
		Wrought Austenitic Stainless Steel	Lubricating Oil	None	None			None
	PB PH	Carbon Steel, Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, temperature ≥212°F, but < 482°F	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	IV.C1.1-d	3.1.1.A-01	A, 16
				Loss of Material	Flow-Accelerated Corrosion Program	IV.C1.1-c	3.1.1.A-25	A, 16
					One-Time Inspection Program Water Chemistry Control Program			H, 16

Table 3.4.2.A-2 Steam and Power Conversion System
NMP1 Feedwater/High Pressure Coolant Injection System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	~~Aging Effect~~ Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	LBS SIA (cont'd)	Carbon Steel, Low Alloy Steel (Yield Strength < 100 Ksi) (cont'd)	Treated Water or Steam, temperature ≥212°F, but < 482°F	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	VIII.D2.1-c	3.4.1.A-01	C, 7
				Loss of Material	Flow-Accelerated Corrosion Program	VIII.D2.2-a	3.4.1.A-06	A
					One-Time Inspection Program Water Chemistry Control Program	VIII.D2.2-a	3.4.1.A-02	B
			Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	VIII.D2.1-c	3.4.1.A-01	C, 7
				Loss of Material	One-Time Inspection Program Water Chemistry Control Program	VIII.D2.2-b	3.4.1.A-02	B
		Wrought Austenitic Stainless Steel	Lubricating Oil	None	None			None
	PB	Carbon Steel, Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, temperature ≥212°F, but < 482°F	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	IV.C1.3-d	3.1.1.A-01	A, 16
				Loss of Material	Flow-Accelerated Corrosion Program	IV.C1.3-a	3.1.1.A-25	A, 16

MODIFICATION 5 – NMP1

One of the carbon steel NMP1 Control Rod Drive Pumps was replaced with one having a stainless steel casing. The resultant changes to ALRA p. 3.1-68 are as shown below.

Table 3.1.2.A-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	B
					One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	E, 24
Pumps	LBS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature ≥140°F, but < 212°F	Loss of Material	One-Time Inspection Program Water Chemistry Control Program	VIII.E.3-a	3.4.1.A-02	B
		Wrought Austenitic Stainless Steel	Treated Water, temperature ≥140°F, but < 212°F	Cracking	One-Time Inspection Program Water Chemistry Control Program			H
Tank	SIA	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature < 140°F, Low Flow	Loss of Material	One-Time Inspection Program Water Chemistry Control Program	VIII.E.5-a	3.4.1.A-02	B

MODIFICATION 6 – NMP2

A filtration skid for Reactor Recirculation Pump seal water was installed at NMP2. The components in this skid are within scope of license renewal since they meet the 54.4(a)(2) criterion with the intended function of Leakage Boundary (Spatial). The ALRA changes that follow are applicable with the installation of this equipment. The piping and fittings included with the skid are encompassed by the existing Piping and Fittings entries in the AMR table.

From p. 2.3-22 of the ALRA for scoping of the filters, the resultant table changes are as shown below.

Table 2.3.1.B.4-1
NMP2 Reactor Recirculation System

Component Type	Intended Functions
Closure Bolting	Pressure Boundary
Filters	Leakage Boundary (Spatial)
Piping and Fittings	Pressure Boundary Leakage Boundary (Spatial) Structural Integrity (Attached)
Pumps	Pressure Boundary
Radiation Collars	Shielding
Restriction Orifices	Throttle, Pressure Boundary
Seal Coolers	Heat Transfer, Pressure Boundary
Valves	Pressure Boundary Leakage Boundary (Spatial) Structural Integrity (Attached)

From ALRA p. 3.1-96, for the AMR of the Filters, the table changes are as shown below.

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Closure Bolting	PB	Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	IV.C1.2-f	3.1.1.B-01	A
						IV.C1.3-g	3.1.1.B-01	A
				Loss of Material	Bolting Integrity Program	IV.C1.2-d	3.1.1.B-26	A
						IV.C1.3-e	3.1.1.B-26	A
				Loss of Preload	Bolting Integrity Program	IV.C1.2-e	3.1.1.B-26	A
						IV.C1.3-f	3.1.1.B-26	A
		Wrought Austenitic Stainless Steel	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)			F
				Loss of Preload	Bolting Integrity Program	IV.C1.2-e	3.1.1.B-26	H
External Surfaces	PB	Cast Austenitic Stainless Steel	Air	None	None			None
	PB	Nickel Based Alloys	Air	None	None			None
	LBS PB SIA	Wrought Austenitic Stainless Steel	Air	None	None			None
Filters	LBS	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F	Loss of Material	One Time Inspection Program Water Chemistry Control Program	VIII.E.5-b	3.4.1.B-02	D
Piping and Fittings	LBS PB SIA	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F	Loss of Material	One-Time Inspection Program Water Chemistry Control Program	VIII.E.5-b	3.4.1.B-02	D

From ALRA p. 3.1-102, for AMR of the associated cast austenitic stainless steel valves, the table changes are as shown below.

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Seal Coolers	HT PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	One-Time Inspection Program Water Chemistry Control Program			H
				Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	IV.C1.2-a	3.1.1.B-01	C
Valves	LBS	Cast Austenitic Stainless Steel	Treated Water, temperature < 140°F	Loss of Material	One-Time Inspection Program Water Chemistry Control Program	VIII.E.5-b	3.4.1.B-02	D
	LBS PB SIA	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	Loss of Material	One-Time Inspection Program Water Chemistry Control Program	VIII.E.2-b	3.4.1.B-02	D
			Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	E, 24

MODIFICATION 7 – NMP2

Several NMP2 Turbine Building Sump Pumps that had aluminum casings were replaced with new pumps having cast iron casings. With the environment of Raw Water, the Selective Leaching of Materials Program will now be credited. The resultant changes to p. 3.3-58 to add this program are as shown below.

Aging Management Programs

The following aging management programs manage the aging effects for the NMP2 Floor and Equipment Drains System components:

- 10 CFR 50 Appendix J Program
- Bolting Integrity Program
- One-Time Inspection Program
- Preventive Maintenance Program
- Selective Leaching of Materials Program
- Systems Walkdown Program

The resultant AMR table changes on p. 3.3-241 are as shown below.

Table 3.3.2.B-14 Auxiliary Systems
NMP2 Floor and Equipment Drains System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Pumps	LBS	Aluminum	Raw Water	Loss of Material	One-Time Inspection Program			J
		Gray Cast Iron	Raw Water	Loss of Material	Preventive Maintenance Program Selective Leaching of Materials Program			J
	PB	Wrought Austenitic Stainless Steel	Treated Water, temperature <140°F	Loss of Material	One Time Inspection			J
			Treated Water, temperature ≥140°F, but < 212°F	Cracking	One-Time Inspection Program			J
Orifices	FC PB	Wrought Austenitic Stainless Steel	Air, Moisture or Wetting, temperature ≥140°F	Loss of Material	One-Time Inspection Program			J
Spray Nozzle	PB SPR	Wrought Austenitic Stainless Steel	Air, Moisture or Wetting, temperature <140°F	Loss of Material	One-Time Inspection Program			J

MODIFICATION 8 – NMP2

Several NMP2 Feedwater System check valves that were originally fabricated of carbon steel have been replaced by stainless steel check valves. The resultant changes to ALRA pp. 3.4-15 and 3.4-65 are as shown below.

3.4.2.B.3 NMP2 FEEDWATER SYSTEM

Materials

The materials of construction for the NMP2 Feedwater System components are:

- Carbon Steel, Low Alloy Steel (Yield Strength < 100 Ksi)
- Wrought Austenitic Stainless Steel

Table 3.4.2.B-3 Steam and Power Conversion System
NMP2 Feedwater System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB PH	Carbon Steel, Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, temperature ≥212°F, but < 482°F	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	IV.C1.3-d	3.1.1.B-01	A, 16
				Loss of Material	Flow-Accelerated Corrosion Program	IV.C1.3-a	3.1.1.B-25	A, 16
			Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow	Cumulative Fatigue Damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	IV.C1.3-d	3.1.1.B-01	A, 16
				Loss of Material	Flow-Accelerated Corrosion Program	IV.C1.3-a	3.1.1.B-25	A, 16
		Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F	Loss of Material	One Time Inspection Program Water Chemistry Control Program	VIII.E.5-b	3.4.1.B-02	D

MODIFICATION 9 – NMP2

Several NMP2 Turbine Building Closed Loop Cooling Water relief valves that had been fabricated of carbon steel were replaced with new valves fabricated of cast austenitic stainless steel. The resultant changes to ALRA p. 3.3-83 are as shown below and the changes to p. 3.3-307 are as shown on the next page.

3.3.2.B.40 NMP2 TURBINE BUILDING CLOSED LOOP COOLING WATER SYSTEM

Materials

The materials of construction for the NMP2 Turbine Building Closed Loop Cooling Water System components are:

- Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)
- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)
- Cast Austenitic Stainless Steel

Table 3.3.2.B-40 Auxiliary Systems
NMP2 Turbine Building Closed Loop Cooling Water System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Bolting	LBS	Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi	Air	Loss of Material	Bolting Integrity Program	VII.I.1-b	3.3.1.B-05	A
External Surfaces	LBS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi	Air	Loss of Material	Systems Walkdown Program	VII.I.1-b	3.3.1.B-05	A
Heat Exchangers	LBS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi	Demineralized Untreated Water	Loss of Material	Closed-Cycle Cooling Water System Program	VII.C2.4-a	3.3.1.B-15	C
Piping and Fittings	LBS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi	Demineralized Untreated Water	Loss of Material	Closed-Cycle Cooling Water System Program	VII.C2.1-a	3.3.1.B-15	A
Valves	LBS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi	Demineralized Untreated Water	Loss of Material	Closed-Cycle Cooling Water System Program	VII.C2.2-a	3.3.1.B-15	A
		Cast Austenitic Stainless Steel	Demineralized Untreated Water	Loss of Material	Closed-Cycle Cooling Water System Program	VII.C2.2-a	3.3.1.B-15	A

ATTACHMENT 2 to NMP1L 2009

A review of NMPNS new and revised analyses completed since submittal of the LRA resulted in changes to ALRA Time-Limited Aging Analysis (TLAA) sections. Of the following four (4) analyses, three (3) are associated with NMP1 and the fourth is associated with NMP2.

Revisions to the existing ALRA are shown with *italics* for additions and ~~strikethroughs~~ for deletions.

ANALYSIS 1 – NMP1

A revised analysis related to ALRA Section 4.6.1, Torus Shell and Vent System Fatigue Analysis (NMP1 Only), was performed that resulted in revised information in that section of the ALRA. The revisions on p. 4.6-2 of the ALRA are as shown below.

Disposition: §54.21(c)(1)(i) – The analyses remain valid for the period of extended operation;
AND
§54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation.

The design basis accident (DBA) was identified as the major load contributing to the fatigue evaluation for all high stress locations in the vent header system. The controlling usage factor was 0.86 at the vent header/vent pipe spherical intersection. Provided that a DBA (the major contributor to fatigue) does not occur during the original 40-year license period, this usage factor will not be exceeded during the period of extended operation; therefore, the NMP1 vent header fatigue usage analyses remain valid in accordance with §54.21(c)(1)(i).

Analysis 2 – NMP1

A revised analysis related to ALRA Section 4.6.5, Fatigue of Primary Containment Penetrations, was performed that resulted in revised information in that section of the ALRA. Changes to ALRA Table 4.1-1 on p. 4.1-3 are as shown on page 6 of this attachment. The changes to ALRA Section 4.6.5 on p. 4.6-10 of the ALRA are as shown below.

4.6.5 FATIGUE OF PRIMARY CONTAINMENT PENETRATIONS

NMP1 - Summary Description

The NMP1 drywell was designed as a Class B Vessel in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, 1965 Edition (ASME Section III, 1965). The 1965 Edition of the ASME Section III B&PV Code did not require fatigue analysis of Class B vessels. The drywell penetrations were considered an extension of the drywell and thus did not require fatigue analysis. For NMP1, fatigue of torus penetrations was addressed in the same analysis as the torus attached piping, the "Plant Unique Analysis Report of the Torus Attached Piping for Nine Mile Point Unit 1 Nuclear Generating Station," which was transmitted to the NRC in a letter dated May 22, 1984 (Reference 4.8-61). Additional fatigue analyses were performed for a number of specific penetrations subsequent to the submittal of Reference 4.8-61. These analyses were performed in accordance with ASME Section III, 1977 Edition, through the Summer 1977 Addenda. Fatigue analyses were performed for the safety/relief valve (SRV) penetration (where the SRV line penetrates the vent header spherical intersection) and torus attached piping penetrations.

Changes to p. 4.6-11 are as shown below starting with the insertion of a new paragraph after the first full paragraph on the page.

In addition to the bounding penetration fatigue evaluations reported in the Plant Unique Analysis Report (Reference 4.8-61), specific stress and fatigue analyses were performed for 26 torus piping penetrations. These analyses were performed in accordance with the 1977 ASME Code Section III, Subsection NC through the 1977 Summer Addenda. The loading conditions considered in the calculations were developed from the 27 possible cases listed in Table 2-5 of the Mark I Containment Program Structural Acceptance Criteria Plant Unique Application Guide, NEDO-24583-1 (Reference 4.8-47). For the fatigue evaluation, the load case having the maximum alternating stress was assumed for each fatigue cycle. This load case included deadweight, thermal, safe shutdown earthquake, and condensation oscillation loads. The number of cycles for all load events was estimated to be 10,000, including accident, earthquake, and normal operating conditions. The fatigue usage determined by the analyses is considered to be conservative because the stress used for each cycle includes stresses from accident and earthquake conditions which do not normally occur. By contrast, the only types of cyclic loads that occur during normal operation are safety/relief valve discharges. The alternating stresses resulting from SRV discharges alone is a fraction of the stress for the load case assumed in the analyses; therefore, SRV discharge will result in negligible fatigue usage per cycle. Also, only 520 SRV discharges are projected for NMP1 which will result in 4,500 effective stress cycles compared to the 10,000 assumed in the analyses. The subject calculations were dispositioned in 2003 for a hypothetical 5 degree increase in torus water temperature, resulting in a minimal change to the calculated fatigue usages. Therefore, since the actual fatigue usage for these penetrations through the period of extended operation will be low compared to the calculated values, the analyses remain valid for the period of extended operation in accordance with §54.21(c)(1)(i).

A CUF was not reported for the torus attached piping penetrations. However, considering the major loads listed above to be significant contributors to fatigue usage, each causing a load cycle equal to the maximum load, there are 951 significant loading events during the 40-year design life. Section 3.4.7 of the PUAR indicates that other loads such as normal SRV actuation, intermediate break accident condensation oscillation (IBA.CO), and chugging can cause up to 10,450 cycles, but only at greatly reduced stress levels; therefore, these loads are assumed to produce a negligible contribution to the overall fatigue usage of the penetrations. Assuming 10,000 cycles of the maximum load equates to a CUF of 1.0, the fatigue CUF resulting from these loads is 0.0951. Projecting this to 60 years by multiplying by 1.5, the 60-year fatigue usage is 0.143, which is well below the code allowable of 1.0.

<u>Disposition</u>: §54.21(c)(1)(i) – The analyses remain valid for the period of extended operation; OR §54.21(c)(1)(ii) – The analyses have been projected to the end of the period of extended operation.

The paragraph that starts on the bottom of p. 4.6-11 and carries over to p. 4.6-12 is revised as shown below.

For the fatigue analyses reported in the Plant-Unique Analysis Report (Reference 4.8-61), the number of anticipated significant transient cycles for a 40-year life divided by the maximum number of allowable cycles for the transient producing the maximum stress was used to estimate the 40-year design CUF. Linear projection of this CUF to 60 years results in a CUF far below the allowable. Therefore, the fatigue analysis of the torus attached piping penetrations has been projected to the end of the period of extended operation in accordance with §54.21(c)(1)(ii). For the additional fatigue analyses of specific penetrations, the actual fatigue usage based on transients that occur during normal operation is predicted to be negligible. Therefore, these analyses remain valid for the period of extended operation in accordance with §54.21(c)(1)(i).

Additionally, on p. 4.8-6, Reference 4.8-47 is changed as shown below.

4.8-47 Mark 1 Containment Program Structural Acceptance Criteria Plant Unique Analysis Application Guide, Task 3.1.3, NEDO 24583-1 79 NED 125, Class 1, October 1979

ANALYSIS 3 – NMP1

A new fatigue flaw growth analysis was performed for an NMP1 Reactor Water Cleanup System weld overlay repair that generated a new TLAA. The new TLAA comprises new ALRA Section 4.7.5. The resultant changes to ALRA p. 4.1-3 are as shown on page 6 of this attachment. The new Section 4.7.5 that would start on new ALRA p. 4.7-8 is as shown below.

4.7.5 REACTOR WATER CLEANUP SYSTEM WELD OVERLAY FATIGUE FLAW GROWTH EVALUATIONS (NMP1 ONLY)

Fatigue crack growth analyses have been performed for two weld overlays in the reactor water cleanup system. The weld overlay design is in accordance with Code Case N-504 (Weld 33-FW-22) and N-504-2 (Weld 33-FW-23A). The repaired welds are located at the inlet nozzle of the regenerative heat exchanger and the transition pipe between the upper and lower shells of the regenerative heat exchanger, respectively. The weld overlays consist of IGSCC-resistant austenitic stainless steel material and, thus, are not susceptible to continued IGSCC crack propagation. However, the first 1/16" thick layer of weld metal deposited is not assumed to be IGSCC-resistant due to weld dilution; thus, it is assumed to be cracked. A fatigue crack growth analysis was performed in accordance with ASME Section XI, Appendix C, with the crack propagating into the overlay from the hypothetical 1/16" deep crack. The acceptance criteria for fatigue crack growth analyses are based on the depth to thickness ratios from Tables IWB-3641-5 and IWB-3641-6.

Disposition: §54.21(c)(1)(i) - The analyses remain valid for the period of extended operation.

The analysis for weld 33-FW-22 assumed 154 startup/shutdown cycles. The flaw depth to weld overlay thickness ratio remained acceptable at 0.22 versus an allowable of 0.29. The number of cycles assumed was based on an estimate of 7 cycles/year for 22 years (12 remaining years of the original license period plus 10 additional years). There have been 34 startup/shutdown cycles since the beginning of 1997; this overlay was installed in May, 1997. Therefore, the weld overlay could experience 120 more cycles before exceeding the assumptions of the analysis. During the last ten years of operation, NMP1 has been experiencing startups and shutdown cycles at a rate of approximately 4/year. Therefore, it is expected that the actual number of startups/shutdowns will remain below 120 cycles in the next 24 years (4 years original license plus 20 year period of extended operation). Therefore, the fatigue crack growth analysis for the weld 33-FW-22 overlay is expected to remain valid for the period of extended operation.

The analysis for weld 33-FW-23A was performed in 2005 and assumed 168 startup/shutdown cycles based on 7 cycles/year for 24 years (the 4 remaining years of original license plus the 20 year period of extended operation). The final flaw depth to thickness ratio was 0.17 compared to an allowable of 0.42; thus, the weld was acceptable for the cycles analyzed. Therefore, since this analysis was performed considering the period of extended operation, the analysis remains valid for the period of extended operation.

This TLAA results in the addition of new ALRA Section A1.2.5.2 starting on p. A1-34 as shown below.

A1.2.5.2 REACTOR WATER CLEANUP SYSTEM WELD OVERLAY FATIGUE FLAW GROWTH EVALUATIONS

Fatigue crack growth analyses have been performed for two weld overlays in the reactor water cleanup system. The repaired welds are located at the inlet nozzle of the regenerative heat exchanger and the transition pipe between the upper and lower shells of the regenerative heat exchanger, respectively. The weld overlays consist of IGSCC-resistant austenitic stainless steel material and, thus, are not susceptible to continued IGSCC crack propagation. However, the first 1/16" thick layer of weld metal deposited is not assumed to be IGSCC-resistant due to weld dilution; thus, it is assumed to be cracked. A fatigue crack growth analysis was performed in accordance with ASME Section XI, Appendix C, with the crack propagating into the overlay from the hypothetical 1/16" deep crack. The fatigue crack growth analyses for the weld overlays are expected to remain valid for the period of extended operation.

ANALYSIS 4 – NMP2

There was a new fatigue analysis performed for the NMP2 Downcomer and Safety/Relief Discharge Line that generated a new TLAA. The new TLAA comprises new ALRA Section 4.6.6. The resultant changes to ALRA Table 4.1-1 on p. 4.1-3 are shown below.

Table 4.1-1
Time-Limited Aging Analyses Applicable to NMPNS

TLAA Category	Description	Disposition Category	Section
1.	**Reactor Vessel Neutron Embrittlement Analysis**		4.2
	Upper-shelf Energy	§54.21(c)(1)(ii)	4.2.1
	Pressure-Temperature (P-T) Limits	§54.21(c)(1)(ii)	4.2.2
	Elimination of Circumferential Weld Inspection (NMP1 only)	§54.21(c)(1)(ii)	4.2.3
	Axial Weld Failure Probability	§54.21(c)(1)(ii)	4.2.4
2.	**Metal Fatigue Analysis**		4.3
	Reactor Vessel Fatigue Analysis	§54.21(c)(1)(iii)	4.3.1
	ASME Section III Class 1 Piping and Components Fatigue Analysis (NMP2 only)	§54.21(c)(1)(iii)	4.3.2
	Feedwater (FWS) Nozzle and Control Rod Drive Return Line (CRDRL) Nozzle Fatigue and Cracking Analyses	§54.21(c)(1)(iii)	4.3.3
	Non-ASME Section III Class 1 Piping and Components Fatigue Analysis	§54.21(c)(1)(iii)	4.3.4
	Reactor Vessel Internals Fatigue Analysis	§54.21(c)(1)(iii)	4.3.5
	Environmentally Assisted Fatigue	§54.21(c)(1)(iii)	4.3.6
	Fatigue of the Emergency Condenser (NMP1 only)	§54.21(c)(1)(iii)	4.3.7
3.	**Environmental Qualification (EQ)**		4.4
	Electrical Equipment EQ	§54.21(c)(1)(iii)	4.4.1
	Mechanical Equipment EQ (NMP2 only)	§54.21(c)(1)(iii)	4.4.2
4.	**Concrete Containment Tendon Prestress Analysis**	Not Applicable	4.5
5.	**Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analysis**		4.6
	Torus Shell and Vent System Fatigue Analysis (NMP1 only)	§54.21(c)(1)(i) and §54.21(c)(1)(ii)	4.6.1
	Torus Attached Piping Analysis (NMP1 only)	§54.21(c)(1)(iii)	4.6.2
	Torus Wall Thickness (NMP1 only)	§54.21(c)(1)(iii)	4.6.3
	Containment Liner Analysis (NMP2 only)	§54.21(c)(1)(ii)	4.6.4
	Fatigue of Primary Containment Penetrations	§54.21(c)(1)(i), §54.21(c)(1)(ii) and §54.21(c)(1)(iii)	4.6.5
	Downcomer and Safety/Relief Valve Discharge Line Fatigue Evaluation (NMP2 Only)	§54.21(c)(1)(ii) and §54.21(c)(1)(iii)	4.6.6
6.	**Other Plant-specific TLAAs**		4.7
	RPV Biological Shield (NMP2 only)	§54.21(c)(1)(ii)	4.7.1
	Main Steam Isolation Valve Corrosion Allowance (NMP2 only)	§54.21(c)(1)(iii)	4.7.2
	Stress Relaxation of Core Plate Hold-Down Bolts (NMP2 only)	§54.21(c)(1)(iii)	4.7.3
	Reactor Vessel and Reactor Vessel Closure Head Weld Flaw Evaluations (NMP1 only)	§54.21(c)(1)(i) and §54.21(c)(1)(iii)	4.7.4
	Reactor Water Cleanup System Weld Overlay Fatigue Flaw Growth Evaluation (NMP1 Only)	§54.21(c)(1)(i)	4.7.5

New ALRA Section 4.6.6 that would start on p. 4.6-14 is as shown below.

4.6.6 DOWNCOMER AND SAFETY/RELIEF VALVE DISCHARGE LINE FATIGUE EVALUATION (NMP2 ONLY)

Summary Description - Downcomer Fatigue Evaluation

The NMP2 downcomers consist of 121 pipes open to the drywell and submerged 9.5 ft below the low water level (operating minimum) of the suppression pool, providing a flow path for uncondensed steam into the pool.

Although the downcomers are a structural component of the primary containment structure, a fatigue analysis using ASME Section III Class 1 rules was performed for the downcomers. Since the analysis includes an assumption of the numbers of cycles that will occur over the 40-year life of the plant, the analysis is a time-limited aging analysis.

Disposition: §54.21(c)(1)(ii) -The analysis has been projected to the end of the period of extended operation.

Analysis

The load combinations considered in the fatigue analysis of the downcomers include normal operating load conditions and accident conditions, including a small break accident (SBA), intermediate break accident (IBA), or design basis accident (DBA). An SBA, IBA, or DBA was assumed to occur one time during the life of the plant, combined with the fatigue usage resulting from upset conditions plus one Residual Heat Removal (RHR) System blowdown. Three different usage factors were determined: (1) upset + RHR + SBA; (2) upset + IBA + DBA; and (3) upset + RHR + DBA. The highest usage factor was for the upset + RHR +DBA, with a 40-year usage factor of 0.182. Upset loads include safety/relief valve discharge (SRV), operating basis earthquake, and thermal loads. A total of 5,154 SRV discharges were included in the analysis (including first and subsequent actuations). The largest contributor to the usage factor was from the upset loads, amounting to 0.0966. Since the upset loads were assumed to occur more than once over the 40-year life of the plant, it is appropriate to project the usage resulting form these loads for an additional 20 years of operation by multiplying the usage by 1.5. Therefore, the projected upset load usage for 60 years is 0.1449. Adding this to the contribution from DBA and RHR which is 0.0854, the total 60-year projected fatigue usage is 0.230 which remains below the ASME Section III allowable of 1.0. Therefore, the downcomer fatigue analysis has been projected in accordance with §54.21(c)(1)(ii).

Summary Description - SRV Penetration Fatigue Analysis

There are eighteen (18) twelve-inch diameter SRV lines that penetrate the drywell floor via flued head type penetrations at NMP2. For these penetrations, a fatigue analysis using ASME Section III Class 1 rules was performed for the SRV piping penetrations through the drywell floor. Although the SRV piping system is Safety Class 3, these additional requirements are considered for additional assurance that steam bypass across the drywell floor will not occur. Additionally, since the highest loads occur at the penetration, this serves to increase the confidence in the overall system.

Disposition: §54.21(c)(1)(iii) - The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Analysis

For the SRV penetration fatigue analysis, the analysis considered operating basis earthquake, SRV actuations, condensation oscillation, chugging, and thermal loads occurring during SRV discharge. A total of 5,154 SRV discharges (including first and subsequent actuations) were included in the analysis. The SRV penetration fatigue analysis was reevaluated as a result of power uprate. The revised 40-year CUF was 0.667. The contribution of the non-emergency load cases to the overall usage is 0.615. If the non-emergency usage is projected by multiplying by a 1.5 factor, the resulting non-emergency usage is 0.923. Adding the emergency usage of 0.053 to the non-emergency contribution yields a 60-year CUF of 0.976. Because this CUF approaches 1.0, NMPNS will revise the SRV penetration fatigue analysis to remove excessive conservatism, or will monitor fatigue usage of the SRV penetrations via the Fatigue Monitoring Program, with the aid of the FatiguePro software. The revision to the analysis or implementation of monitoring will be completed in conjunction with the analyses described in ALRA Section 4.3 prior to the start of the period of extended operation.

Since the SRV penetration analysis will be revised, or monitoring of the SRV penetrations will be initiated, prior to entry into the period of extended operation, the effects of aging will be adequately managed in accordance with §54.21(c)(1)(iii).

This TLAA results in the addition of new ALRA Section A2.2.4.3 starting on p. A2-29 as shown below.

A2.2.4.3 DOWNCOMER AND SAFETY/RELIEF VALVE DISCHARGE LINE FATIGUE EVALUATION

The downcomers consist of 121 pipes open to the drywell and submerged 9.5 ft below the low water level (operating minimum) of the suppression pool, providing a flow path for uncondensed steam into the pool. The load combinations considered in the fatigue analysis of the downcomers include normal operating load conditions and accident conditions, including a small break accident (SBA), intermediate break accident (IBA), or design basis accident (DBA). An SBA, IBA, or DBA was assumed to occur one time during the life of the plant, combined with the fatigue usage resulting from upset conditions plus one Residual Heat Removal (RHR) System blowdown. The largest contributor to the usage factor was from the upset loads. Since the upset loads were assumed to occur more than once over the 40-year life of the plant, it is appropriate to project the usage resulting form these loads for an additional 20 years of operation by multiplying the usage by 1.5. Adding this to the contribution from DBA and RHR the total projected fatigue usage is below the ASME Section III allowable of 1.0. Therefore, the downcomer fatigue CUF has been projected for the period of extended operation.

There are eighteen (18) twelve-inch diameter SRV lines that penetrate the drywell floor via flued head type penetrations. For these penetrations, a fatigue analysis using ASME

Section III Class 1 rules was performed for the SRV piping penetrations through the drywell floor. Although the SRV piping system is Safety Class 3, these additional requirements are considered for additional assurance that steam bypass across the drywell floor will not occur. Additionally, since the highest loads occur at the penetration, this serves to increase the confidence in the overall system. For the SRV penetration fatigue analysis, the analysis considered operating basis earthquake, SRV actuations, condensation oscillation, chugging, and thermal loads occurring during SRV discharge. The SRV penetration fatigue analysis was reevaluated as a result of power uprate utilizing significant conservatism. The projected CUF for the balance of plant life including the period of extended operation was close to the ASME Section III allowable of 1.0; therefore, prior to entry into the period of extended operation, the SRV Penetration fatigue analysis will be revised to remove the excessive conservatism or the fatigue usage of the SRV penetrations will be monitored by the Fatigue Monitoring Program.