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Constellation Energy
Nine Mile Point Nuclear Station

December 4, 2008

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 1; Docket No. 50-220

License Amendment Request Pursuant to 10 CFR 50.90: One-Time Extension of the Primary Containment Integrated Leakage Rate Test Interval – Response to NRC Request for Additional Information (TAC No. MD9453)

REFERENCES:

(a) Letter from K. J. Polson (NMPNS) to Document Control Desk (NRC), dated August 15, 2008, License Amendment Request Pursuant to 10 CFR 50.90: One-Time Extension of the Primary Containment Integrated Leakage Rate Test Interval - Technical Specification Section 6.5.7, 10 CFR 50 Appendix J Testing Program Plan

(b) Letter from R. V. Guzman (NRC) to K. J. Polson (NMPNS), dated November 6, 2008, Request for Additional Information Regarding Nine Mile Point Nuclear Station, Unit No. 1, One-Time Extension of the Primary Containment Integrated Leakage Rate Test Interval (TAC No. MD9453)

Nine Mile Point Nuclear Station, LLC (NMPNS) hereby transmits supplemental information requested by the NRC in support of a previously submitted request for amendment to Nine Mile Point Unit 1 (NMP1) Renewed Operating License DPR-63. The initial request, dated August 15, 2008 (Reference a) proposed to revise Technical Specification Section 6.5.7, "10 CFR 50 Appendix J Testing Program Plan," to allow a one-time extension of the Integrated Leakage Rate Test (ILRT) interval for no more than five (5) years. The supplemental information, provided in Attachment 1 to this letter and Attachments 2 and 3 referenced therein, responds to the request for additional information (RAI) documented in the NRC's letter dated November 6, 2008 (Reference b).

This supplemental information does not affect the No Significant Hazards Determination analysis provided by NMPNS in Reference (a). Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this supplemental information to the appropriate state representative. This letter contains no new regulatory commitments.

A017
NRR

Should you have any questions regarding the information in this submittal, please contact T. F. Syrell, Licensing Director, at (315) 349-5219.

Very truly yours,

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

I, Keith J. Polson, 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 document 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 4th day of December, 2008.

WITNESS my Hand and Notarial Seal:

Sandra A. Oswald
Notary Public

My Commission Expires:

10/25/09
Date

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified in Oswego County
Commission Expires 10-25-09

KJP/DEV

Attachments:
1. Nine Mile Point Unit 1 – Response to NRC Request for Additional Information Regarding the Proposed One-Time Extension of the Primary Containment Integrated Leakage Rate Test Interval
2. Review of the NMP1 PRA Model Update Peer Review Findings (RAI-5)
3. Annotated Pages from Attachment 2 to the NMPNS Submittal dated August 15, 2008

cc: S. J. Collins, NRC
R. V. Guzman, NRC
Resident Inspector, NRC
J. P. Spath, NYSERDA

ATTACHMENT 1

NINE MILE POINT UNIT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE PROPOSED ONE-TIME EXTENSION OF THE

PRIMARY CONTAINMENT INTEGRATED LEAKAGE RATE

TEST INTERVAL

NINE MILE POINT UNIT 1
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PROPOSED ONE-TIME EXTENSION OF THE PRIMARY CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL

By letter August 15, 2008, Nine Mile Point Nuclear Station, LLC (NMPNS) requested an amendment to the Nine Mile Point Unit 1 (NMP1) Renewed Facility Operating License DPR-63. The proposed change would revise Technical Specification (TS) Section 6.5.7, "10 CFR 50 Appendix J Testing Program Plan," to allow a one-time extension of the Integrated Leakage Rate Test (ILRT) interval for no more than five (5) years. This attachment provides supplemental information in response to the request for additional information documented in the NRC's letter dated November 6, 2008. Each individual NRC question is repeated (in italics), followed by the NMPNS response.

Containment Integrity

RAI-1

Please discuss and provide the following:

a. *A summary list of those containment penetrations (including their test schedule intervals) that have not demonstrated acceptable performance history in accordance with the primary containment leakage rate program.*

b. *A summary table for Type B and Type C tests, including the interval schedule dates, that are planned to be performed prior to and during the requested 5-year extension period of the ILRT interval.*

c. *Type B and Type C test results and their comparison with the allowable leakage rate specified in the plant Technical Specifications.*

d. *Testing and schedule of those penetrations with seals and gaskets, and bolted connections that are frequently disassembled or are not routinely disassembled.*

Response

a. Summary list of those containment penetrations (including their test schedule intervals) that have not demonstrated acceptable performance history in accordance with the primary containment leakage rate program.

Containment penetrations that have experienced Appendix J local leak rate test failures and their test schedule intervals are listed in Table 1, beginning with the 1999 NMP1 refueling outage (when the last ILRT was performed). A test failure represents leakage that exceeds the administrative criteria established in accordance with 10 CFR 50, Appendix J, Option B.

b. Summary table for Type B and Type C tests, including the interval schedule dates, that are planned to be performed prior to and during the requested 5-year extension period of the ILRT interval.

The planned test schedules for Type B and Type C leak rate tested components for the next three NMP1 refueling outages are provided in Table 2 (Type B tests) and Table 3 (Type C tests). With approval of the 5-year ILRT interval extension request, the next ILRT would be performed during the 2013 refueling outage (N1R22). These planned test schedules were developed assuming that there are no leak rate test failures, and do not account for any leak rate tests that may be required to support maintenance activities.

As indicated by Tables 2 and 3, the total number of planned Type B and Type C leak rate tests over the next three refueling outages is: 2009 (N1R20) - 82; 2011 (N1R21) - 92; and 2013 (N1R22) - 87.

c. Type B and Type C test results and their comparison with the allowable leakage rate specified in the plant Technical Specifications.

The NMP1 combined local leak rate test (Type B and Type C tests including airlocks) acceptance criterion (0.6 L_a) is 388.44 scfh. The maximum and minimum pathway leak rate summary totals for the last two refueling outages are shown below.

	Maximum Pathway		Minimum Pathway	
Refueling Outage	**Leakage (scfh)**	**% of 0.6 L_a**	**Leakage (scfh)**	**% of 0.6 L_a**
2007 (N1R19)	240.38	61.9%	89.734	23.1%
2005 (N1R18)	222.169	57.2%	83.492	21.5%

d. Testing and schedule of those penetrations with seals and gaskets, and bolted connections that are frequently disassembled or are not routinely disassembled.

The current test schedule interval and date last tested for Type B penetrations (i.e., those with seals and gaskets, and bolted connections) are listed in Table 4 for frequently disassembled penetrations and in Table 5 for infrequently disassembled penetrations. Note that electrical and mechanical penetrations and airlocks are not included in these two tables.

RAI-2

Regulatory Position C.3 of Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," recommends that visual examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test based on a 10-year ILRT interval. Please describe, with a schedule, how you would supplement this 10-year interval-based visual inspection requirement for the requested 15-year ILRT interval.

Response

As stated in Section 3.1.2.2 of the Enclosure to the August 15, 2008, NMPNS submittal letter, the general visual examination requirements specified in the American Society of Mechanical Engineers (ASME) Code Section XI (Subsection IWE) containment inspection program will continue to be performed during the proposed 5-year extension of the ILRT interval. In addition, visual inspections of accessible interior surfaces of the primary containment are conducted each refueling outage in accordance with approved plant procedures to provide reasonable assurance that the effects of aging will be adequately managed, as described in the NMPNS License Renewal application (Reference 1). These visual inspections include the following:

Drywell and Drywell Head Interior

- Vicinity of drywell penetrations for obvious structural discontinuities (cracks).
- Support attachments and brackets for obvious defects (missing or broken bolts/nuts, bent rods, plate buckling, etc.).
- Internal surface area for gross signs of corrosion and deterioration (depth greater than approximately 1/16"; indications of leak).
- Internal coated surface area for any visible defects including blistering, cracking, flaking, peeling and physical or mechanical damage (area larger than approximately 6 square feet).

Suppression Chamber Interior

- Vicinity of any penetrations for obvious structural discontinuities (cracks).
- Vent pipe expansion joints, support structures, brackets and bolting for obvious defects (missing or broken nuts/bolts, bent rods, plate buckling, etc.).
- Internal surface area, including water line regions, for gross signs of corrosion or buckling.

The above-described inspections are scheduled to be performed each refueling outage during the proposed 5-year extension of the ILRT interval.

RAI-3

Section 3.1.2.4 of the enclosure to your August 15, 2008, submittal, discusses IWE-1240 augmented inspection of the interior surface of the drywell shell. Please discuss whether there are other areas requiring augmented examination.

Response

Other than the six localized drywell shell interior surface areas discussed in Section 3.1.2.4 of the Enclosure to the August 15, 2008, NMPNS submittal letter, there are no other areas requiring augmented examination in accordance with IWE-1240.

During the License Renewal application process, NMPNS committed to perform an augmented VT-1 visual inspection of the containment penetration stainless steel bellows using enhanced techniques qualified for detecting stress corrosion cracking (see NMP1 Updated Final Safety Analysis Report Appendix C). These inspections are beyond the scope of examinations required by Table IWE-2500-1 of the ASME Code Section XI and thus are referred to as augmented examinations in the IWE containment inspection program plan. However, they are not considered augmented examinations as defined in IWE-1240.

RAI-4

As part of the NMP1 drywell augmented inspection/monitoring program, Section 3.1.2.4 of the enclosure to your submittal describes the volumetric and visual examinations of the drywell shell during the 2003 and 2007 refueling outages. Please provide further discussion relative to the following:

a. *General description and correlation between the 2003 and 2007 examination results.*

b. *General corrosion condition in the monitored areas.*

c. *Based on the results of 2007 examinations and anticipated corrosion rate, please discuss the schedule for the next ultrasonic testing measurements, root cause determination, and any planned or already implemented corrective actions.*

Response

a. General description and correlation between the 2003 and 2007 examination results.

As discussed in Section 3.1.2.4 of the Enclosure to the August 15, 2008, NMPNS submittal letter, detailed visual examinations of six localized areas of the drywell shell, coinciding with the locations of the drywell area coolers, were performed in 2003 in accordance with the ASME Section XI, Subsection IWE inspection program. These examinations identified corrosion that was characterized as "major" (i.e., greater than 5 percent of the base metal was judged to be lost). The code-required evaluation of this condition included taking volumetric (UT) thickness measurements to confirm that the drywell shell was acceptable for continued service (i.e., minimum wall thickness had not been violated). Due to the radiological conditions existing in the drywell during the 2003 refueling outage, the investigation of the condition was limited to four areas of the drywell shell (around 3 of the area coolers) that were considered to represent the worst areas of major corrosion. A UT thickness reading was taken at each of these four identified locations. The thickness reading locations were defined by measured distances from the floor and nearby support beams, but no grids were applied to the shell to facilitate future location of the exact spots where the thickness readings were taken. The evaluation performed in 2003 evaluated the lowest readings found at each measured location against the minimum required wall thickness and concluded that the drywell shell was acceptable for continued service.

In accordance with the drywell supplemental inspection program (submitted to the NRC by NMPNS letter dated April 4, 2006 (Reference 2) and accepted by the NRC as part of the License Renewal application review (Reference 3)), UT thickness measurements were taken during the 2007 refueling outage at the reported locations where the 2003 measurements had been taken. It was anticipated that a corrosion rate could be determined from a comparison of the 2003 and 2007 readings; however, the corrosion rates derived from that limited set of data points were widely scattered, unrealistic (one location showed a gain in wall thickness) and inconsistent with the observed condition of the drywell shell (see Item b below). It was concluded that this limited data could not be used as the sole basis for determining a corrosion rate. This result was attributed to the likelihood that the exact same spots had not been measured in 2003 and in 2007. Therefore, actions were taken during the 2007 refueling outage to establish a more repeatable means of determining wall thickness measurements so that a truly representative corrosion rate can be determined. Grids were painted on the drywell shell at the areas of interest and readings taken at multiple grid points. Measurements taken during the 2009 refueling outage at the same grid points will allow actual corrosion rates to be established and addressed in accordance with the drywell supplemental inspection program acceptance criteria, which have been added to the IWE Program.

b. General corrosion condition in the monitored areas.

The areas of localized drywell shell corrosion were extensively inspected by the IWE Responsible Individual during the 2007 refueling outage. These areas were observed to have a generalized corroded surface, but no evidence of loose corrosion products was present. There were no rust flakes or blisters on the surfaces, no evidence of pitting, and no build up of rust flakes on the floor below the areas. If significant shell corrosion had taken place, corrosion products should have been observed in the areas since carbon steel corrosion products expand significantly. The absence of corrosion products was inconsistent with the corrosion rates that were indicated by comparing the first 2007 set of four UT thickness measurements with the 2003 UT thickness measurement data.

c. Based on the results of 2007 examinations and anticipated corrosion rate, please discuss the schedule for the next ultrasonic testing measurements, root cause determination, and any planned or already implemented corrective actions.

In accordance with the IWE Program Plan, UT thickness measurements will be taken during the 2009 refueling outage at the grid locations established in 2007. These 2009 measurements will be compared to the baseline data established in 2007 to determine a corrosion rate for the 2-year period. The acceptance standards are tabulated in the IWE Program Plan and are the same as those given in Reference 2. The corrosion rate determined from the UT measurement data and the remaining margin to the minimum required wall thickness will determine the subsequent frequency of performing UT thickness measurements as well as the need to implement mitigative strategies (e.g., application of protective coatings, repair, or replacement of affected sections of the shell).

The apparent cause of the localized corrosion of the drywell shell in the area of each of the drywell area coolers was determined to be the cleaning practices for the area cooler coils. The procedure for cleaning the area coolers called for the coils to be rinsed with a cleaning agent. There were no protective measures for the liner and no requirement to rinse the liner after cleaning. The procedure for cleaning the cooler coils was revised in 2003 to require the use of protection on the liner before cleaning of the coolers.

Risk Analysis

RAI-5

The core damage frequency and total population dose in the ILRT analysis (based on the 2007 PRA) are about a decade lower than in the severe accident mitigation alternative analysis (based on the 2003 PRA). Provide a description of the major changes to PRA models and assumptions that account for these changes. Provide a description of the peer review comments in areas related to these reductions, the resolution of these comments, and the impact of any unresolved comments on the risk results for the requested change.

Response

A full update of the NMP1 probabilistic risk assessment (PRA) model in accordance with Regulatory Guide 1.200 was completed in January 2008 (referred to as the 2007 PRA model). Table 6 provides a description of the major changes to PRA models and assumptions that account for the differences between the PRA model supporting the severe accident mitigation alternatives (SAMA) analysis (the 2003 model) and the updated Regulatory Guide 1.200 compliant PRA model. The impacts are ranked relative to their impact on the internal events core damage frequency (CDF). The resulting model improvements caused a decrease in overall CDF, thus decreasing LERF.

An industry peer review team reviewed the updated PRA model in February 2008 and commended NMPNS on the quality of the NMP1 Level II analysis. Attachment 2 contains a summary of all of the findings from the peer review and addresses the impact of these findings on the NMP1 ILRT interval extension risk assessment. In summary, most of the findings are related to documentation and have no material impact on the ILRT interval extension risk assessment. Assessment of required model changes resulting from resolution of the peer review findings has determined that the changes would have a negligible, if any, impact on the conclusions of the ILRT interval extension risk assessment.

RAI-6

Explain how the population dose of 1.05E6 person rem per event was derived for Electric Power Research Institute (EPRI) Class 7 releases (as reported in Tables 10-2 through 10-4 of Attachment 2 to the August 15, 2008, submittal). Table 6-7 indicates that the population dose for EPRI Class 7 would be a combination of releases from collapsed accident progression bins (APBs) 3, 4, and 5. However, the NMPNS adjusted population dose for each of these APBs (last column of Table 6-3) is well below 1E6 person rem per event. It appears that the population dose for EPRI Class 7 was calculated based on the sum of the population dose values for APBs 3, 4, and 5 rather than the frequency-weighted sum. Reconcile the population dose values and update the risk assessment as appropriate.

Response

As discussed in Section 8 of Attachment 2 to the August 15, 2008, NMPNS submittal letter (page 26), the Class 7 population dose calculation utilized a modified EPRI 1009325, Revision 1, methodology. The EPRI guidance document uses a weighted average of values for accident progression bins 3, 4, and 5 to generate the population dose for Class 7. The methodology was simplified in the NMP1 calculation as follows: each accident progression bin frequency (bins 3, 4, and 5) was multiplied by the entire Class 7 frequency and summed, yielding a dose value that is always conservative with respect to a weighted average. Utilizing this approach is conservative and is deemed acceptable from a risk perspective.

RAI-7

The large early release frequency (LERF) is reported to be 3.00E-7 per year for internal events (on page 44 of Attachment 2) and 8.67E-7 per year for external events (on page B-1 of Attachment 2). This results in a total LERF of 1.17E-6 per year. However, on page B-3, it is stated that the total LERF from all hazards is 1.7E-6 per year. Address this inconsistency, and confirm the correct value for the total LERF for NMP1, with and without the requested change.

Response

Note: All of the section and page numbers referenced in the following response are referring to Attachment 2 to the August 15, 2008, NMPNS submittal letter.

As discussed in Section 6.2 (page 8), the NMP1 ILRT extension risk assessment utilized the PRA model (Level I and Level II) developed in 2007. At the time that the ILRT extension risk assessment was performed, the 2007 PRA model addressed accidents initiated by internal events at full power, and containment response to those accidents. The 2007 PRA model did not include fire and seismic event contributions. Therefore, the most recent fire and seismic models available (those that were updated in 2003) were utilized to assess the impact of external events. This bounding external events assessment is presented in Appendix B of Attachment 2.

The large early release frequency (LERF) value of 3.00E-07/year stated in Section 9.2 (page 44) is the baseline value for a 3-year ILRT interval and is from the 2007 PRA model (i.e., considers only internal events). In Appendix B, Section B.4 (page B-1), the LERF contribution from external events (fire and seismic), obtained from the 2003 PRA model, was determined to be 8.67E-07/year. Adding these two values yields a total LERF value of 1.17E-06/year. However, as also stated in Appendix B, Section B.4 (page B-3), the baseline total LERF value for all hazards (internal and external) given in the 2003 PRA is 1.7E-06/year, which is larger than the 1.17E-06/year value obtained by adding the 2007 PRA internal events contribution and the 2003 PRA external events contribution. Therefore, the baseline total LERF value of 1.7E-06/year from the 2003 PRA was conservatively chosen to determine the LERF increase for the proposed 5-year extension of the ILRT interval.

RAI-8

Table B-2 provides the external and internal event contributions to EPRI Class 3b frequency based on the results from the 2003 PRA. Explain how the Class 3b frequency value of 2.38E-9 per year for external events (first entry in column 2) was derived.

Response

The assessment summarized in Table B-1 of Attachment 2 to the August 15, 2008, NMPNS submittal letter is sufficient to support the conclusion of Appendix B of Attachment 2; i.e., that incorporating external event hazard risk into the analysis does not change the overall conclusion that extending the ILRT interval by 5 years is acceptable from a risk perspective.

NINE MILE POINT UNIT 1
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PROPOSED ONE-TIME EXTENSION OF THE PRIMARY CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL

NMPNS has determined that the information in Table B-2 and the associated text discussing Table B-2 is extraneous and should therefore be disregarded. Annotated pages of Attachment 2 to the August 15, 2008, NMPNS submittal letter, showing the deletion of Table B-2 and associated text, are provided in Attachment 2 to this letter.

A correction for Section 11 of Attachment 2 to the August 15, 2008, NMPNS submittal letter, is also provided in Attachment 2 to this letter. The value for the estimated change in LERF associated with the increase in ILRT interval from 10 years to 15 years was originally stated in Section 11 (page 58, second paragraph), as 1.91E-8/year. This value was improperly transposed from Table B-2 rather than from Table B-1. The correct value, from Table B-1, is 8.79E-08/year. The derivation of this value is as explained in Section B.4 of Attachment 2 to the August 15, 2008, NMPNS submittal letter.

References

1. Letter from J. A. Spina (NMPNS) to Document Control Desk (NRC), dated July 14, 2005, Recovery of Nine Mile Point License Renewal Application Quality (TAC Nos. MC3272 and MC3273)

2. Letter from T. J. O'Connor (NMPNS) to Document Control Desk, dated April 4, 2006, Safety Evaluation Report (SER), With Open Items Related to the License Renewal of Nine Mile Point Nuclear Station, dated March 2006 - SER Open Item 3.0.3.2.17-1 (TAC Nos. MC3272 and MC3273)

3. NUREG-1900, "Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2," September 2006

Table 1 (RAI-1a)
10 CFR 50 Appendix J Local Leak Rate Test Failures

N1R15, Spring 1999 Refueling Outage (When Last ILRT Performed)

Penetration	Comp ID	System	Test Type	Test Result (scfh) (1)	Test Schedule Interval	Comments (2)
X-241	RPV Stabilizer B Access Cover	Containment	B	17.7 (AF) 0.042 (AL)	60 to 30 months	Exceeded Admin Limit, returned to 30 month interval
X-2B	Test Station 6M (3)	Containment	B	39.6 (AF) 0.061 (AL)	30 months fixed	Exceeded Admin Limit, MSIV bellows leak identified, bellows replaced
X-2A	IV-01-03	Main Steam	C	138.48 (AF) 1.07 (AL)	30 months fixed	Exceeded Admin Limit, due to packing leak
X-2B	IV-01-04	Main Steam	C	Gross (AF) 0.045 (AL)	30 months fixed	Unquantified gross leakage, MSIV internal modification implemented
X-4A	CKV-31-01R	Feedwater	C	53.3 (AF) 0.99 (AL)	30 months fixed	Exceeded Admin Limit, valve repaired
X-5A	CKV-39-04	Emerg. Cooling	C	Gross (AF) 14.7 (AL)	30 months	Unquantified gross leakage, valve repaired and kept at 30 month interval
X-19	IV-201.1-09	Containment	C	117.6 (AF) 0.046 (AL)	60 to 30 months	Exceeded Admin Limit, valve repaired and returned to 30 month interval
X-19	IV-201.1-11	Containment	C	94.2 (AF) 9.34 (AL)	60 to 30 months	Exceeded Admin Limit, valve repaired and returned to 30 month interval
X-40	CKV-201.2-68	Containment	C	55.4 (AF) 0.042 (AL)	30 months	Exceeded Admin Limit, remained on 30 month interval
XS-321	CKV-201.2-71	Containment	C	Gross (AF) 0.042 (AL)	30 months	Unquantified gross leakage, valve repaired and kept at 30 month interval

N1R16, Spring 2001 Refueling Outage

Penetration	Comp ID	System	Test Type	Test Result (scfh) [1]	Test Schedule Interval	Comments [2]
X-4B	CKV-31-02	Feedwater	C	Gross (AF) 0.042 (AL)	30 month fixed	Unquantified gross leakage, valve repaired
X-9	IV-33-02	Cleanup	C	59.96 (AF) 3.466 (AL)	30 months	Exceeded Admin Limit, valve replaced and kept at 30 month interval
X-5A	CKV-39-04	Emerg. Cooling	C	Gross (AF) 2.77 (AL)	30 months	Unquantified gross leakage, valve repaired and kept at 30 month interval

N1R17, Spring 2003 Refueling Outage

Penetration	Comp ID	System	Test Type	Test Result (scfh) [1]	Test Schedule Interval	Comments [2]
X-4A	CKV-31-01	Feedwater	C	Gross (AF) 3.811 (AL)	30 month fixed	Unquantified gross leakage, valve repaired
X-4B	CKV-31-02	Feedwater	C	Gross (AF) 8.5 (AL)	30 month fixed	Unquantified gross leakage, valve repaired
X-154	CKV-33-03	Cleanup	C	Gross (AF) 0.174 (AL)	60 to 30 months	Unquantified gross leakage, valve replaced and returned to 30 month interval
X-5A	CKV-39-04	Emerg. Cooling	C	Gross (AF) 6.5 (AL)	30 months	Unquantified gross leakage, valve internals replaced and kept at 30 month interval
X-174	CKV-44.3-13	Control Rod Drive	C	Gross (AF) 0.85 (AL)	24 month fixed	Unquantified gross leakage, valve repaired. On IST 24 month test interval

N1R18, Spring 2005 Refueling Outage

Penetration	Comp ID	System	Test Type	Test Result (scfh) [1]	Test Schedule Interval	Comments [2]
X-247	RPV Stabilizer H Access Cover	Containment	B	Gross (AF) 0.046 (AL)	120 to 30 months	Unquantified gross leakage, flange O-ring replaced and returned to 30 month interval

N1R19, Spring 2007 Refueling Outage

Penetration	Comp ID	System	Test Type	Test Result (scfh) [1]	Test Schedule Interval	Comments [2]
X-243	RPV Stabilizer D Access Cover	Containment	B	22.424 (AF) 0.046 (AL)	120 to 30 months	Exceeded Admin limit, O-rings replaced and returned to 30 month interval
X-4A	IV-31-07	Feedwater	C	17.124 (AF) 19.424 (AL)	30 month fixed	Exceeded Admin limit, valve seats flushed with no improvement, accepted AL leakage value
X-174	CKV-44.3-12	Control Rod Drive	C	41.924 (AF) 27.024 (AL)	30 month fixed	Exceeded Admin limit, performed minor seat maintenance, leakage improved to less than admin limit but still elevated

NOTES:
(1) AF = As-found leak test; AL = As-left leak test
(2) Admin limits are: Type B Tests - 16.1 scfh; Type C Tests - 32.3 scfh
(3) Containment penetration test stations provide a single location from which multiple penetrations can be tested at once. If elevated leakage is detected, penetrations can be individually isolated to locate the leaking penetration.

Table 2 (RAI-1b)
NMP1 Appendix J Type B Test Schedule

Comp ID	Penetration #	Option "B" Interval (1)	Last Test Date	N1R20 03/2009 (2)	N1R21 03/2011 (2)	N1R22 (ILRT) 03/2013 (2)
Escape Airlock	X-1B	30/f	5/22/07	1	1	1
Personnel Airlock	X-1A	30/f	5/22/07	1	1	1
Equipment Hatch	X-1	30/f	4/4/07	1	1	1
Drywell	Flange	30/f	4/12/07	1	1	1
Tophead	Manway	30/f	3/28/07	1	1	1
RPV Stabilizer A	X-240	120	3/28/01	1	X	X
RPV Stabilizer B	X-241	120	3/24/07	X	X	1
RPV Stabilizer C	X-242	120	3/24/07	X	X	1
RPV Stabilizer D	X-243	30/p	3/30/07	1	1	1
RPV Stabilizer E	X-244	120	3/28/01	1	X	X
RPV Stabilizer F	X-245	120	3/26/01	X	1	X
RPV Stabilizer G	X-246	120	3/22/03	X	1	X
RPV Stabilizer H	X-247	30/p	3/24/07	1	X	1
201-08	I Flange	60	2/16/05	1	X	X
201-10	I Flange	120	3/14/05	X	X	1
201-16	I Flange	120	3/14/05	X	X	1
201-32	I Flange	120	3/14/05	X	X	1
68-01	Cover	30/f	3/24/07	1	1	1
68-01	N Shaft	120	3/24/07	X	X	1
68-01	S Shaft	120	3/23/07	X	X	1
68-02	Cover	30/f	3/26/07	1	1	1
68-02	W Shaft	120	4/9/05	X	1	X
68-02	E Shaft	120	4/9/05	X	1	X
68-08	I Flange	120	3/12/01	1	X	X
68-03	Cover	30/f	3/24/07	1	1	1
68-03	N Shaft	120	4/30/04	1	X	X
68-03	S Shaft	120	4/30/04	1	X	X
68-09	I Flange	120	2/14/03	X	1	X
68-04	Cover	30/f	5/19/07	1	1	1

Comp ID	Penetration #	Option "B" Interval (1)	Last Test Date	N1R20 03/2009 (2)	N1R21 03/2011 (2)	N1R22 (ILRT) 03/2013 (2)
68-04	N Shaft	120	3/26/07	X	X	X
68-04	S Shaft	120	3/26/07	X	X	X
68-10	I Flange	120	2/14/03	X	1	X
Test Station	1E	120	3/5/07	X	X	X
Test Station	2E	120	4/25/01	1	X	X
Test Station	3E	120	3/15/05	1	X	X
Test Station	4E	120	3/14/05	X	1	X
Test Station	5E	120	3/15/05	X	1	X
Test Station	6E	120	3/18/05	X	1	X
Test Station	7E	120	3/15/05	X	X	1
Test Station	8E	120	2/23/05	X	X	1
Test Station	9E	120	3/5/07	X	X	X
Test Station	1M	120	3/11/07	X	X	X
Test Station	2M	120	3/26/05	X	X	1
Test Station	3M	120	1/27/03	1	X	X
Test Station	4M	120	1/30/03	X	1	X
Test Station	5M	120	3/28/03	X	1	X
Test Station	6M	30/f	3/21/07	1	1	1
Test Station	7M	120	3/9/07	X	X	X
Test Station	11M	120	2/13/03	1	X	X
Test Station	12M	120	3/17/05	X	X	1
TIP #3	X-23B	120	3/27/01	1	X	X
TIP #4	X-23C	120	3/27/01	1	X	X
TIP #1	X-23D	120	3/28/03	X	1	X
TIP #2	X-23E	120	3/28/03	X	1	X
Hatch #1	XS-310	30/f	4/6/07	1	1	1
Hatch #2	XS-311	30/f	4/6/07	1	1	1
Hatch #3	XS-312	30/f	4/4/07	1	1	1
58.1-07	Flange	120	3/8/07	X	X	X
68-01	N Flange	120	3/24/07	X	X	1
68-01	S Flange	120	3/24/07	X	X	1
68-02	W Flange	120	4/9/05	X	1	X

Comp ID	Penetration #	Option "B" Interval (1)	Last Test Date	N1R20 03/2009 (2)	N1R21 03/2011 (2)	N1R22 (ILRT) 03/2013 (2)
68-02	E Flange	120	4/9/05	X	1	X
68-03	N Flange	120	4/30/04	1	X	X
68-03	S Flange	120	4/30/04	1	X	X
68-04	N Flange	120	3/26/07	X	X	X
68-04	S Flange	120	3/26/07	X	X	X
81-241	I Flange	30/p	4/1/07	1	1	X
81-242	I Flange	30/p	4/1/07	1	1	X
81-243	I Flange	30/p	3/19/07	1	1	X
81-244	I Flange	30/p	3/20/07	1	1	X
			Type B Test Totals	33	33	28

NOTES: (1) Option "B" Interval key: 30/f = 30 month fixed interval
30/p = 30 month performance-based interval
60 = 60 month performance-based interval
120 = 120 month performance-based interval

(2) Test Schedule Interval key: 1 = Scheduled test; X = Test not scheduled

Table 3 (RAI-1b)
NMP1 Appendix J Type C Test Schedule

Comp ID	Penetration #	Option "B" Interval (1)	Last Test Date	N1R20 03/2009 (2)	N1R21 03/2011 (2)	N1R22 (ILRT) 03/2013 (2)	Type C Test ILRT Comments (3)
81-241	XS-335	30/p	4/1/07	1	X	1	Normal lineup
81-242	XS-335	30/p	4/1/07	1	X	1	Normal lineup
81-243	XS-334	30/p	3/19/07	1	1	X	Normal lineup
81-244	XS-334	30/p	3/19/07	1	X	1	Normal lineup
01-01	X-2A	30/f	3/17/07	1	1	1	ILRT penalty
01-02	X-2B	30/f	3/17/07	1	1	1	ILRT penalty
01-03	X-2A	30/f	3/28/07	1	1	1	ILRT penalty
01-04	X-2B	30/f	3/28/07	1	1	1	ILRT penalty
31-01R	X-4A	24	3/21/07	1	1	1	ILRT penalty
31-02R	X-4B	24	3/21/07	1	1	1	ILRT penalty
31-07	X-4A	30/f	3/22/07	1	1	1	ILRT penalty
31-08	X-4B	30/f	3/21/07	1	1	1	ILRT penalty
33-01R	X-154	60	3/23/07	X	1	1	ILRT penalty
33-02R	X-9	60	4/7/05	1	X	1	ILRT penalty
33-03	X-154	60	3/23/07	X	1	1	ILRT penalty
33-04	X-9	60	4/17/05	1	X	1	ILRT penalty
36-147	X-23B	24	3/23/07	1	1	1	Vented
36-148	X-23C	24	3/23/07	1	1	1	Vented
36-149	X-23D	24	3/23/07	1	1	1	Vented
36-150	X-23E	24	3/23/07	1	1	1	Vented
39-03	X-5B	30/p	3/22/07	1	1	1	ILRT penalty
39-04	X-5A	30/p	3/23/07	1	X	1	ILRT penalty
39-05	X-5B	60	3/24/07	X	1	1	ILRT penalty
39-06	X-5A	60	3/30/07	X	1	1	ILRT penalty
39-07R	X-3A	60	3/26/07	X	1	1	ILRT penalty
39-09R	X-3A	60	3/26/07	X	1	1	ILRT penalty
39-08R	X-3B	60	3/26/07	X	1	1	ILRT penalty
39-10R	X-3B	60	3/26/07	X	1	1	ILRT penalty
42.1-02	X-131	24	3/22/07	1	1	1	ILRT penalty

Comp ID	Penetration #	Option "B" Interval (1)	Last Test Date	N1R20 03/2009 (2)	N1R21 03/2011 (2)	N1R22 (ILRT) 03/2013 (2)	Type C Test ILRT Comments (3)
42.1-03	X-131	24	3/22/07	1	1	1	ILRT penalty
44.2-15	SDV Vent	60	3/30/07	X	1	X	Normal lineup
44.2-16	SDV Vent	60	3/30/07	X	1	X	Normal lineup
44.2-17	SDV Drain	60	3/30/07	X	1	X	Normal lineup
44.2-18	SDV Drain	60	3/30/07	X	1	X	Normal lineup
68-05/ 68-08	XS-313 & XS-317	24	3/27/07	1	1	1	Normal lineup
68-06/ 68-09	XS-314 & XS-318	24	3/24/07	1	1	1	Normal lineup
68-07/ 68-10	XS-316 & XS-320	24	3/27/07	1	1	1	Normal lineup
72-479	X-122	60	3/20/07	X	1	1	ILRT penalty
72-480	X-122	60	3/20/07	X	1	1	ILRT penalty
83.1-09	X-26	60	4/15/05	1	X	1	ILRT penalty
83.1-10	X-26	60	4/6/05	1	X	1	ILRT penalty
83.1-11	X-25	60	4/16/05	1	X	1	ILRT penalty
83.1-12/ 83.1-35	X-25	60	4/16/05	1	X	1	ILRT penalty
110-127	X-139	60	4/4/05	1	X	1	ILRT penalty
110-128/ 110-640	X-139	30f	4/12/07	1	1	1	ILRT penalty
114-114	X-121	30p	3/20/07	1	X	1	Vented
114-116	X-121	60	3/19/07	1	X	1	Vented
122-03	X-82	60	4/3/07	X	1	1	ILRT penalty
201-07/ 201-08	XS-340	30/f	3/27/07	1	1	1	Vented
201-09/ 201-10	X-18	30/f	3/27/07	1	1	1	Vented
201-16/ 201-17	XS-327	30/f	3/27/07	1	1	1	Vented
201-31/ 201-32	X-19	30/f	4/3/07	1	1	1	Vented
201.1-09	X-19	60	3/23/07	X	1	X	Vented
201.1-11	X-19	60	3/23/07	X	1	X	Vented
201.1-14	X-59	60	3/16/05	1	X	1	Vented
201.1-16	X-59	60	3/16/05	1	X	1	Vented
201.2-03	X-19	60	2/23/07	X	1	X	Vented
201.2-06	XS-327	60	2/22/07	X	1	X	Vented
201.2-23	XS-321	60	3/16/05	1	X	1	Vented
201.2-24	XS-321	60	3/16/05	1	X	1	Vented
201.2-29	X-49	60	3/22/07	X	1	X	Vented

Comp ID	Penetration #	Option "B" Interval (1)	Last Test Date	N1R20 03/2009 (2)	N1R21 03/2011 (2)	N1R22 (ILRT) 03/2013 (2)	Type C Test ILRT Comments (3)
201.2-30	X-49	60	3/20/07	X	1	X	Vented
201.2-32	X-19	60	2/23/07	X	1	X	Vented
201.2-33	XS-327	60	2/22/07	X	1	X	Vented
201.2-39	X-23D	60	3/17/05	1	X	1	Vented
201.2-40	X-23D	60	3/17/05	1	X	1	Vented
201.2-67	X-40	60	2/22/07	X	1	X	Vented
201.2-68	X-40	60	2/22/07	X	1	X	Vented
201.2-70	XS-321	60	3/16/05	1	X	1	Vented
201.2-71	XS-321	60	3/16/05	1	X	1	Vented
201.2-109	XS-328	60	8/15/08	X	1	X	Vented
201.2-110	XS-328	60	8/15/08	X	1	X	Vented
201.2-111	XS-328	60	1/23/07	X	1	X	Vented
201.2-112	XS-328	60	1/23/07	X	1	X	Vented
201.7-01	X-64	60	1/23/07	X	1	X	Vented
201.7-02	X-64	60	1/23/07	X	1	X	Vented
201.7-08	X-134	60	3/19/07	X	1	X	Vented
201.7-09	X-134	60	3/19/07	X	1	X	Vented
201.7-10	X-20	60	8/20/04	1	X	1	Vented
201.7-11	X-20	60	8/20/04	1	X	1	Vented
44.3-12	X-174	24	4/2/07	1	1	1	ILRT penalty
44.3-13	X-174	24	3/27/07	1	1	1	ILRT penalty
			Type C Test Totals	49	59	59	

NOTES: (1) Option "B" Interval Key: 24 = IST required test, performed every 24 months
30/f = 30 month fixed interval
30/p = 30 month performance-based interval
60 = 60 month performance-based interval

(2) Test Schedule Interval Key: 1 = Scheduled test; X = Test not scheduled

(3) Type C Test ILRT Comments: Normal Lineup - Component inherently exposed to Type A test pressure
Vented - Penetration exposed to Type A test pressure and is vented
ILRT Penalty - Penetration not exposed to Type A test pressure, penalty taken. Type C results added to ILRT.

Table 4 (RAI-1d)
NMP1 Frequently Disassembled Gasketed / Bolted Penetrations
Type B Testing Schedule

Penetration Description	Test Schedule Interval	Date Last Tested
X-1, Equipment Hatch	30 month fixed	4/04/07
X-127, Drywell Head Manway	30 month fixed	3/28/07
Drywell Flange	30 month fixed	4/12/07
XS-310, Torus Access Manway	30 month fixed	4/06/07
XS-311, Torus Access Manway	30 month fixed	4/06/07
XS-312, Torus Access Manway	30 month fixed	4/07/07
CKV-68-01, Containment Vacuum Breaker Cover	30 month fixed	3/24/07
CKV-68-02, Containment Vacuum Breaker Cover	30 month fixed	3/26/07
CKV-68-03, Containment Vacuum Breaker Cover	30 month fixed	3/24/07
CKV-68-04, Containment Vacuum Breaker Cover	30 month fixed	5/19/07
PSV-81-241, Safety Valve Flange	30 month performance-based	4/01/07
PSV-81-242, Safety Valve Flange	30 month performance-based	4/01/07
PSV-81-243, Safety Valve Flange	30 month performance-based	3/19/07
PSV-81-244, Safety Valve Flange	30 month performance-based	3/20/07

Table 5 (RAI-1d)
NMP1 Infrequently Disassembled Gasketed / Bolted Penetrations
Type B Testing Schedule

Penetration Description	Test Schedule Interval (All Performance-Based)	Date Last Tested
X-240, RPV Stabilizer A Access Cover	120 months	3/28/01
X-241, RPV Stabilizer B Access Cover	120 months	3/24/07
X-242, RPV Stabilizer C Access Cover	120 months	3/24/07
X-243, RPV Stabilizer D Access Cover	30 months	3/30/07
X-244, RPV Stabilizer E Access Cover	120 months	3/28/01
X-245, RPV Stabilizer F Access Cover	120 months	3/26/01
X-246, RPV Stabilizer G Access Cover	120 months	3/22/03
X-247, RPV Stabilizer H Access Cover	30 months	3/24/07
IV-58.1-07, Inboard Flange	120 months	3/08/07
IV-201-08, Inboard Flange	60 months	2/16/05
IV-201-10, Inboard Flange	120 months	3/14/05
IV-201-16, Inboard Flange	120 months	3/14/05
IV-201-32, Inboard Flange	120 months	3/14/05
IV-68-08, Inboard Flange	120 months	3/12/01
IV-68-09, Inboard Flange	120 months	2/14/03
IV-68-10, Inboard Flange	120 months	2/14/03
CKV-68-01, Valve Shaft Seals	120 months	3/24/07

Penetration Description	Test Schedule Interval (All Performance-Based)	Date Last Tested
CKV-68-02, Valve Shaft Seals	120 months	4/09/05
CKV-68-03, Valve Shaft Seals	120 months	4/30/04
CKV-68-04, Valve Shaft Seals	120 months	3/26/07
TIP #1, Flange	120 months	3/28/03
TIP #2, Flange	120 months	3/28/03
TIP #3, Flange	120 months	3/27/01
TIP #4, Flange	120 months	3/27/01

Table 6 (RAI-5)
Summary of Major Changes in the 2007 NMP1 PRA Model and
a Qualitative Assessment of Their Importance to Internal Events CDF

PRA Model	Change Summary	CDF Increase	CDF Decrease
Original Initiating Events	More realistic frequency plus added availability	---	High
Manual Shutdown Initiator	Improved modeling (was conservative)	---	High
High Energy Line Breaks/Floods	Added these initiators to model	Medium	---
Equipment Reliability Data	Data is much improved since the IPE	---	Medium
Consequential Loss of Offsite Power (LOSP)	Increased the probability that trip causes LOSP	Medium	---
Human Reliability Analysis	Improved modeling of dependencies plus Emergency Condenser (EC) control	Medium	---
Load Management	Improved modeling (was conservative)	---	Medium
Loss of RPS Buses and Instruments	Improved modeling (was conservative)	---	Low
Unit 2 Firewater Crosstie	Added to the model	---	Low
Station Blackout Model	Improved Accident Sequence Model	---	Low
Success Criteria	2 HPCI versus 1 for MLOCAW and ATWS 3 Electromatic Relief Valves (ERVs) versus 2 with firewater Long term makeup to EC for depressurization	Low	---

ATTACHMENT 2

REVIEW OF THE NMP1 PRA MODEL UPDATE

PEER REVIEW FINDINGS (RAI-5)

SR	F&O ID	F&O Level	Category II	F&O Description	F&O Basis	F&O Pos	UNIT 1 ILRT IMPACT
DA-E1	DA-E1-01	Finding	DOCUMENT the data analysis in a manner that facilitates PRA applications, upgrades, and peer review.	The documentation of the special basic events (Section 5) should be enhanced to provide a better basis for the values used. Sensitivity studies should also be considered to assess the impact of these events on the overall risk results.	Changes in the values used for the special events could impact the PRA results	The documentation of the special basic events should be enhanced to better justify the basis for these items, particularly those that involve a combination of equipment issues and human actions. Consider noting some of these events as sources of uncertainty and consider the use of sensitivity studies to assess the significance of these events on the overall PRA results.	**DOCUMENTATION ONLY** This section and table of special variables were made explicit to ensure their visibility, documentation to facilitate future peer, applications, etc. Also, it is not very specific with regard to which special variables are of concern if any. Also, Section 6 of the DA (Data Analysis) notebook explicitly refers to these as containing potentially important assumptions that can be assessed from the QU (Quantification) notebook. This is an opportunity for potential improvement in the future as are numerous assumptions throughout the PRA (this is not a finding). Some of these variables are set to 1.0 and are place holders for future updates (e.g., Bennett's Bridge, Portable Charger)
SY-A11	SY-A11-01	Finding	INCORPORATE the effect of variable success criteria (i.e., success criteria that change as a function of plant status) into the system modeling. Example causes of variable system success criteria are: (a) different accident scenarios. Different success criteria are required for some systems to mitigate different accident scenarios (e.g., the number of pumps required to operate in some systems is dependent upon the modeled initiating event); (b) dependence on other components. Success criteria for some systems are also dependent on the success of another component in the system (e.g., operation of additional pumps in some cooling water systems is required if noncritical loads are not isolated); (c) time dependence. Success criteria for some systems are time-dependent (e.g., two pumps are required to provide the needed flow early following an accident initiator, but only one is required for mitigation later following the accident); (d) sharing of a system between units. Success criteria may be affected when both units are challenged by the same initiating event (e.g., LOOP).	The system notebooks in section 2.6 provide the success criteria but do not evaluate DG mission time for different LOSP type initiators.	Improved modeling could impact PRA results.	Evaluate DG mission time for different LOSP type initiators.	**NO IMPACT.** Constellation disagrees this is a finding. In future updates, will add explanation that adding convolution for each LOSP cause and recovery would result in different EDG recovery times, but once weighted correctly it would give the same result as already done with average weighted LOSP and recovery convolution case. Breaking out LOSP causes and modeling this level of detail is only required if due to a future application.
SY-B10	SY-B10-01	Finding	When modeling a system, INCLUDE appropriate interfaces with the support systems required for successful operation of the system for a required mission time. (See also AS-A6.) Examples include: (a) actuation logic (b) support systems required for control of components (c) component motive power (d) cooling of components (e) any other identified support function (e.g., heat tracing) necessary to meet the success criteria and associated systems	Diesel generator modeling needs to include actuation logic and air compressors.	Support systems, particularly for major components such as EDGs, need to be included in the PRA model.	The Diesel generator does not include actuation logic, (Item a - See F&O SY-B11-01). The diesel model omits the diesel air compressors from the system boundary. This requires that the air receivers for auto start to stay at proper pressure for 24 hours with no air compressors. There does not seem to be documentation to support this (Item b).	**DOCUMENTATION ONLY.** EDG failure history shows that the fast start system (actuation logic) is very reliable and has not resulted in any failures to start the EDG. As such the reliability of the fast start sequence can be assumed to be fully considered in the failure data used to determine the EDG failure rates. Modeling the sequencer along with detailed modeling of its components will not provide additional risk insights and will add unnecessary complexity to the model and as such precludes the need to explicitly model the fast start sequencer as a subsystem outside the EDG boundary. All the components of the air start system are tested during the monthly EDG operability test; procedure N1-ST-M4 A(B). As such, air start system failures are captured in the data used to evaluate EDG failures. This precludes the need to explicitly model the air start system as a subsystem outside the EDG boundary. Regarding the air receivers stay at proper pressure for 24 hours, this is dependent on the availability of AC power, the reliability of the two air compressors, and the condition of the piping. Condition reports were reviewed and no evidence was found that leakage and air compressor reliability were problematic to the extent that additional modeling detail was required. Also, AC power recovery is very likely well within the 24 hour time frame. Regardless, Diesel notebook documentation will be enhanced to provide a basis for this conclusion.
IE-D3	IE-D3-01	Finding	DOCUMENT the key assumptions and key sources uncertainty with the initiating event analysis.	The IE notebook (as well as the other PRA notebooks) contains a discussion of the assumptions used. The notebooks also provide a discussion of plant-specific sources of uncertainty. However, this SR requires a systematic evaluation of all sources of uncertainty, including industry-wide issues with data and modeling approaches. Recent EPRI reports are available that document generic industry uncertainty sources. These items should be reviewed for applicability to NMP1 and added to the PRA documentation.	While primarily a documentation issue, the sources of uncertainty are needed to support sensitivity studies both for the base PRA and for risk-informed applications.	EPRI TR-1009652 provides a more extensive listing of generic industry sources of uncertainty that should also be considered for applicability to NMP1.	**DOCUMENTATION ONLY.** From initial reviews we agree that this will mostly be more documentation and sensitivity studies.
SY-B13	SY-B13-01	Finding	DO NOT USE proceduralized recovery actions as the sole basis for eliminating a support system from the model; however, INCLUDE these recovery actions in the model quantification. For example, it is not acceptable to not model a system as HVAC or CCW on the basis that there are procedures for dealing with losses of these systems.	Section 3.1.1 assumption in system notebook SY-01c (page SY-01-21) violates the Capability Category 2 requirements.	The current assumption only meets Category I. Modeling of this support system failure (and operator actions to recover it may have some impact on the PRA results.	Resolve the assumption stated in section 3.1.1 in the Diesel system notebook (SY-01c, page SY-01-21) that violates cat 2.	**DOCUMENTATION ONLY** This refers to assumption 11 where it was noted that failure of the lockout relays 86-16 and 86-17 to reset was modeled within the basic event for operator action. Based on the evaluation provided in appendix H of *SY-01b-4160/600V/480V System Notebook*, overloading is unlikely since the EDG start would have to be concurrent with a large LOCA to achieve the maximum pump loads. In the event an overload would occur, the EDG can be restarted and loads managed to preclude overload. Therefore this failure was not considered in the model.

SR	F&O ID	F&O Level	Category II	F&O Description	F&O Basis	F&O Pos	UNIT 1 ILRT IMPACT
IF-D5a	IF-D5a-01	Finding	GATHER plant-specific information on plant design, operating practices, and conditions that may impact flood likelihood (i.e., material condition of fluid systems, experience with water hammer, and maintenance-induced floods). In determining the flood-initiating event frequencies for flood scenario groups, USE a combination of (a) generic and plant-specific operating experience; (b) pipe, component, and tank rupture failure rates from generic data sources and plant-specific experience; and (c) engineering judgment for consideration of the plant-specific information collected.	Provide an in depth discussion about the applicability of generic flood data to NMP1 in the notebook or update the flood frequencies over the years with NMP1 data which would be typically zero flood events.	This is considered a finding because the requirements of Cat II or Cat III can not be met. All data seems to be generic for flooding information.	It is believed that an in depth discussion to explain the NMP1 specific data interface with the generic would suffice to bring this to Cat II requirements.	**DOCUMENTATION ONLY.** Revised Section 5.2 to indicate that plant specific CR (Condition Report) search did not turn up any significant events that required Bayesian update to generic data. Note that a review of potential maintenance induced events is addressed in the notebook.
DA-C8	DA-C8-01	Finding	When required, USE plant-specific operational records to determine the time that components were configured in their standby status.	Alignment fractions are estimated based only on the number of trains available (e.g., 1 of 2 = 50% alignment fraction). This is adequate for Cat 1. For Cat II, need to base fractions on actual operating experience.	Actual data will be needed in order to meet Category II requirements.	Review the operating history to validate the alignment fractions used in the PRA. Develop documentation of this review for inclusion in the DA notebook.	**NEGLIGIBLE IMPACT** The current basis for alignment fractions is not exhaustive and additional data analysis could be performed. However, this effort will result in a very minimal impact on the model. The alignment basic events were reviewed for importance in the model. None had a significant RAW or Fussel-Vessely value. Highest RAW was 1.03 for CRD pump 11 in standby. Fussel-Vessely values were also small. The only value greater than 5E-3 was 2.6E-2 for CRD pump 11 in maintenance. From experience, one CRD pump is always in standby and equipment rotation is important for these pumps. Additionally, rotation of this equipment has been discussed in system engineer interviews. The 50% alignment fraction is deemed most appropriate and more detailed data analysis will not significantly alter the values.
MU-C1	MU-C1-01	Finding		CNG-CM-1.01-3003, PRA Configuration Control Section 5.13, PRA Applications documents a current living applications list exists but was not located.	As the applications list is a key part of the PRA update process, absence of this list indicates that the program is not fully implemented.	Document and implement "current living applications list".	**PRA CONFIGURATION CONTROL PROCEDURE – NO MODEL IMPACT.**
DA-D4	DA-D4-01	Finding	When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: (a) confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram (b) examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes (c) examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate (d) confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered (e) confirmation of the reasonableness of the posterior distribution mean value	Section 2.7 (3rd paragraph) makes a statement that Bayesian results are reasonable. However, there is no discussion of the criteria that is used. In several data variables, the plant evidence seems to be quite different from the prior. These should be investigated with regard to the Bayesian update process to assure the plant point estimate is not in the extremes of the prior distribution. Example: failure rate GAZR1 has a prior of 2.90e-3 and posterior of 4.69e-3, while the plant information is 2 failures in 240 hrs (0.008 per hr). Similarly, failure rate VMZD1 has a prior of 1.07e-3 and posterior of 3.03e-3, while the plant information is 4 failures in 550 hrs (0.007 per demand).	As some of the data values appear suspicious, the PRA results could be affected by changes to the Bayesian-updated data values.	Evaluate & document the Bayesian update process for data parameters where the plant information may be inconsistent with the prior distribution.	**DOCUMENTATION ONLY.** DA-D4 (e) was done, but not documented well. An additional explanation is required at end of first paragraph in Section 2.7. The cited examples were looked at and determined to be reasonable.
SY-B11	SY-B11-01	Finding	MODEL those systems that are required for initiation and actuation of a system. In the model quantification, INCLUDE the presence of the conditions needed for automatic actuation (e.g., low vessel water level). INCLUDE permissive and lockout signals that are required to complete actuation logic.	The diesel generator initiation system is not completely modeled.	Support systems, particularly for major components such as EDGs, need to be included in the PRA model.	The diesel generator initiation system is not completely modeled. The lockout relay is modeled with no other details besides fail to start.	**DOCUMENTATION ONLY.** EDG failure history shows that the fast start system is very reliable and has not resulted in any failures to start the EDG. As such the reliability of the fast start sequence can be assumed to be fully considered in the failure data used to determine the EDG failure rates. Modeling the sequencer along with detailed modeling of its components will not provide additional risk insights and will add unnecessary complexity to the model and as such precludes the need to explicitly model the fast start sequencer as a subsystem outside the EDG boundary. All the components of the air start system are tested during the monthly EDG operability test; procedure N1-ST-M4 A(B). As such, air start system failures are captured in the data used to evaluate EDG failures. This precludes the need to explicitly model the air start system as a subsystem outside the EDG boundary Additional discussion will be provided in the applicable system notebooks to better support these conclusions.
IF-C8	IF-C8-01	Finding	USE potential human mitigating actions as additional criteria for screening out flood sources if all the following can be shown: (a) flood indication is available in the control room; (b) the flood source can be isolated; and (c) the mitigating action can be performed with high reliability for the worst flood from that source. High reliability is established by demonstrating, for example, that the actions are procedurally directed, that adequate time is available for response, that the area is accessible, and that there is sufficient manpower available to perform the actions.	This finding has a relationship to the suggestion in F&O IF-C6-01. Table 4-5 uses the term YES with very little descriptive matter other than the criteria prior to the table in the IF notebook. In order to fully review this as per the standard more detail about alarms or operator intervention needs to be provided.	The current documentation does not meet Category II requirements	Add some specific detail in the IF notebook preferably near Table 4-5 in the cases of YES to provide the exact resolution of the requirements listed in the notebook previous to the Table.	**DOCUMENTATION ONLY** Improved Section 4.6.2 by adding reference to applicable screening criteria. Added note to Table 4-5 to reference Section 4.6.2. Also, note that ASME Quality Table references Section 4.6, which explains the screening (appears that reviewer did not see this, otherwise probably not finding)

SR	F&O ID	F&O Level	Category II	F&O Description	F&O Basis	F&O Pos	UNIT 1 ILRT IMPACT
IF-C3	IF-C3-01	Finding	For the SSCs identified in IF-C2c, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. EITHER: a) ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions; OR b) NOTE that these mechanisms are not included in the scope of the evaluation.	Submergence, pipe whip, and environmental effects are not addressed as per the ASME standard category II. Reg. Guide 1.200 has more information on this and there are NRC overrides identified in the standard that need to be addressed.	Requirements for Category II	Add some discussions to the IF notebook regarding these issues or specifically identify that these issues have already been addressed..	DOCUMENTATION ONLY Submergence is addressed, spray/impingement are considered and documented. Sections 4.5, 2.3, 3.4 and 4.7 revised to make it clearer that these types of impacts were considered. Note also that Table 2-1 identifies areas where HELB analysis is considered in Appendix B and IE Initiating Event Notebook.
IF-E5a	IF-E5a-01	Finding	For all human failure events in the internal flood scenarios, INCLUDE the following scenario-specific impacts on PSFs for control room and ex-control room actions as appropriate to the HRA methodology being used: (a) additional workload and stress (above that for similar sequences not caused by internal floods) (b) cue availability (c) effect of flood on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) (d) flooding-specific job aids and training (e.g., procedures, training exercises)	The requirements for this category are not met. Flooding effects on other HEPs in the Transient event accident sequence logic does not appear to be accounted for. The Internal Flooding HRA did not include an evaluation of how HFEs in the base (i.e., non-flooding) model that are credited in the flooding scenarios would be impacted by the occurrence of the flood event. The HR analysis should be updated to reflect either walkthroughs or talk-throughs of the flooding event sequences with plant operations.	The current treatment does not meet Category II requirements	The HR analysis should be updated to reflect either walkthroughs or talk-throughs of the flooding event sequences with plant operations. Qualitative analysis may be sufficient in some cases.	DOCUMENTATION ONLY This has been addressed with HR-E3-01 and is documented in Appendix C of HR Notebook. This included talk-through with Operations and consideration of impacts with and without successful isolation as well as other operator actions in the model.
HR-E3	HR-E3-01	Finding	TALK THROUGH (i.e., review in detail) with plant operations and training personnel the procedures and sequence of events to confirm that interpretation of the procedures is consistent with plant observations and training procedures.	During development of the flooding HEPs, neither plant operations nor training personnel were contacted for the review of procedures and anticipated sequence of events. Also, the flooding HEP response models were not confirmed using simulator observations/talk-throughs.	ASME Standard supporting requirement HR-E3 not met.	Update the NMP1 HRA to include interviews with plant operations and training personnel relative to development of the flooding HEPs. Conduct talk-throughs with operators to confirm flooding HEP response timing and resource availability	DOCUMENTATION ONLY This has been addressed with HR-E3-01 and is documented in Appendix C of HR (Human Reliability) Notebook. This included talk-through with Operations and consideration of impacts with and without successful isolation as well as other operator actions in the model.
LE-F1a	LE-F1a-01	Finding	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 4.5.9-3.	The LERF accident sequences are quantified as a function of the Level 1 accident sequence classes. This was determined to not satisfy the ASME requirement that the relative contribution to LERF from plant damage states and significant LERF contributors are quantified in terms of the LERF contributors identified in Table 4.5.9-3 (e.g., pressure suppression bypass, isolation condenser tube rupture, etc.).	NMP1 Level 2 analysis does not satisfy Capability Category II requirement F1a.	Create new plant damage states (PDS) that correlate with the LERF contributors identified in ASME Table 4.5.9-3, and re-quantify the Level 2 analysis based upon these new PDS.	DOCUMENTATION ONLY Include this as a useful comparison - no new PDS are required nor should they be developed to provide this breakdown
QU-C1	QU-C1-01	Finding	IDENTIFY cutsets with multiple HFEs that potentially impact significant accident sequences/cutsets by re-quantifying the PRA model with HEP values set to values that are sufficiently high that the cutsets are not truncated. The final quantification of these post-initiator HFEs may be done at the cutset level or saved sequence level.	In doing sensitivity studies to meet this SR, use HEP values higher than nominal, e.g.0.1 to ensure that all dependencies are captured.	This is needed to meet the SR.	Carry out sensitivity studies using higher than nominal values for HEPs and review the results to see all dependences are captured.	DOCUMENTATION ONLY The "0.1 HEP" Sensitivity calculation has been completed and is documented in Section 3.6 of the HRA notebook.
QU-D1a	QU-D1a-01	Finding	REVIEW a sample of the significant accident sequences/cutsets sufficient to determine that the logic of the cutset or sequence is correct.	Appendices to the QU notebook present the top 200 CDF and LERF cutsets. The top 20 CDF cutsets are specifically discussed in section 4.2.3 in the context of plant response, significant assumptions made, etc. While the top 200 cutsets are included, the analysis of these cutsets should be expanded to include a greater number of cutsets, as the top 20 only constitutes about 60% of the CDF and is dominated by cutsets with only an initiator and one failure (i.e., does not demonstrate a comprehensive review)	It is not possible to demonstrate the model is correct with such a limited review. Verification of house event and flag settings cannot be performed with such a limited review (see SR QU-D1c).	Analyze more cutsets to cover about 95 to 99% of the CDF.	DOCUMENTATION ONLY SR says to review a sample not 99%; additional sampling will be documented.
DA-C14	DA-C14-01	Finding	For each SSC for which repair is to be modeled (see SY-A22), IDENTIFY instances of plant-specific or applicable industry experience and for each repair, COLLECT the associated repair time with the repair time being the period from identification of the component failure until the component is returned to service.	The Recovery/Repair of equipment is generally neglected in the model except for offsite power recovery, diesel recovery, instrument air initiating event recovery and screenhouse recovery (these are addressed in Section 5). Industry data is used for DG recovery, and no discussion is provided concerning plant-specific repair. Also, the industry data used was not reviewed for applicability. Instrument air and screenhouse recoveries use a screening value based on long time to recover.	As repair/recovery can have a significant impact on the PRA results (particularly for major components such as EDGs), resolution of these issues could impact the PRA.	As the NRC RG 1.200 clarifications put increased emphasis on the use of plant-specific information in crediting repair, plant data should be reviewed and incorporated in the analysis and the industry data must be reviewed for applicability to NMP1. The basis for instrument air and screenhouse recovery should also be expanded to meet the intent of this SR.	NEGLIGIBLE IMPACT No screening of LOSP or EDG recovery is used - it is used as generic data from NUREG and is acceptable. While expanded data analysis could be performed, its impact will be negligible because, based on experience, NMP1 has not had a large number of initiating events and has been in operation for almost 30 years. This level of data will not appreciably impact the recovery factors used.
AS-A8	AS-A8-02	Finding	DEFINE the end state of the accident progression as occurring when either a core damage state or a steady-state condition has been reached.	In SBO trees, late recovery of power is taken to OK state without checking for system availability. This treatment can be improved by checking for availability of the mitigating systems.	The SR is not currently met. Need to address the F&O for meeting the SR.	Instead of ending the event tree with an OK end state, develop it further by modeling the availability of the mitigating systems.	NEGLIGIBLE IMPACT Resolved by model update - Injection (top event INJ) and heat removal (top event CHR) were added to SBO model and are required for success when AC is recovered. This had a negligible quantitative impact on results, but adds completeness to model.

SR	F&O ID	F&O Level	Category II	F&O Description	F&O Basis	F&O Pos	UNIT 1 ILRT IMPACT
MU-B4	MU-B4-01	Finding		CNG-CM-1.01-3003, PRA Configuration Control Section 5.12, PRA Revisions needs to implement requirement that "External peer review is required for PRA upgrades".	PRA Upgrades shall receive a peer review (in accordance with the requirements specified in Section 6 of the ASME PRA Standard) for those aspects of the PRA that have been upgraded. Refer to Section 2 of the ASME PRA Standard for the distinction of a PRA Upgrade versus PRA maintenance and update.	Revise Section 5.12.A.1.e to state "External peer review is required for PRA upgrades", currently states "External peer review is an expectation for PRA upgrades"	**DOCUMENTATION ONLY – PRA PROCEDURE.**
QU-D5b	QU-D5b-01	Finding	REVIEW the importance of components and basic events to determine that they make logical sense.	Some insights from the importance listings (for equipment and operator actions) are discussed in section 4.2.4 of QU NB However, further discussion should be provided to specifically address the requirements of this SR. Provide a more detailed discussion how this SR is met. QU NB does not have adequate discussion.	The SR is not met at present currently. The F&O needs to be resolved for SR to be met.	Provide a more detailed discussion how this SR is met. QU NB does not have adequate discussion.	**DOCUMENTATION ONLY** Add more discussion of symmetry review, plant understanding, etc. probably to Section 4.3.2, which was not referenced in this finding (Section 4.3.2 is referenced in the ASME Quality table for this SR). Make sure Section 4.2.4 reasonableness check and comparison is referenced in Section 4.3.2 and vice versa. Reasonableness check for HRA has been enhanced via comparison with Oyster Creek (Section 3.5 of the HRA Notebook).
HR-G6	HR-G6-01	Finding	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	An overall "global operator dependency failure" event is applied to account for a complete breakdown in crew functionality.	The "global operator dependency failure" event has a significant impact on the results (i.e., dominates top cutsets) without a detailed analysis supporting the estimated value assigned to the HEP.	Global event ZQQQQ_DEPOPERATO should be removed from the base model and considered as part of a sensitivity study.	**NO IMPACT.** Long Term Loss of Heat Removal Dependency group (ZQDHR) added to show that the dominant dependent groups were associated with this action. The global action (ZQQQQ) is now a small contributor and not masking other contributors.
HR-G7	HR-G7-01	Finding	For multiple human actions in the same accident sequence or cut set, identified in accordance with supporting requirement QU-C1, ASSESS the degree of dependence, and calculate a joint human error probability that reflects the dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including (a) the time required to complete all actions in relation to the time available to perform the actions (b) factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.) (c) availability of resources (e.g., personnel)[Note (1)]	The ZQQQQ_DEPOPERATO event is included in the model to consider the potential for a cross-cutting operator failure during an accident. The basis for the numerical value assigned to this event, while not unreasonable, is not well-established. The event participates in the dominant cutsets, and may be masking the risk contribution from other failures.	The ZQQQQ_DEPOPERATO event is currently a significant contributor to the overall risk. Removing this event from the base case model will impact the PRA results.	It is recommended that global event ZQQQQ_DEPOPERATO be removed from the base model and considered as part of a sensitivity study, due to its significance (i.e., dominates the top cutsets) and uncertainty (i.e., reasonableness of assigned value questionable).	**NO IMPACT.** Long Term Loss of Heat Removal Dependency group (ZQDHR) added to show that the dominant dependent groups were associated with this action. The global action (ZQQQQ) is now a small contributor and not masking other contributors.
QU-D4	QU-D4-01	Finding	REVIEW a sampling of nonsignificant accident cutsets or sequences to determine they are reasonable and have physical meaning.	Section 4.3.5 of the QU notebook briefly notes that a review was performed. However, there is no evidence presented in the notebook. The QU notebook should include a sampling of several non-significant cutsets and demonstrate that these cutsets correctly represent plant features, operator actions, and expected plant behavior.	This is needed to ensure the PRA model is correct.	Examine several, e.g., 50 or more, non-significant cutsets and demonstrate that these cutsets correctly represent plant features, operator actions, and expected plant behavior.	**DOCUMENTATION ONLY.** SR does not require explicitly that this be documented, thus the interpretation is that some evidence with a sampling be documented along with a better explanation of the review process
QU-E4	QU-E4-01	Finding	EVALUATE the sensitivity of the results to key model uncertainties and key assumptions using sensitivity analyses [Note (1)].	Section 5.2 in QU NB addresses this issue qualitatively. More sensitivity runs need to be done to evaluate model uncertainties, e.g., Set all HEPs, CCFs etc at 5th and 95th percentile during quantification.	SR is not met currently. F&O must be implemented for the SR to be met.	More sensitivity runs need to be done to evaluate model uncertainties, e.g., Set all HEPs, CCFs etc at 5th and 95th percentile during quantification.	**DOCUMENTATION ONLY.** From initial reviews we agree that this will mostly be more documentation and sensitivity studies.
QU-F6	QU-F6-01	Finding	DOCUMENT the quantitative definition used for significant basic event, significant cutset, significant accident sequence. If other than the definition used in Section 2, JUSTIFY the alternative.	This is not documented in the QU NB, This SR is therefore not met. Needed to provide a discussion in the QU NB as to how this topic is met.	SR is not met unless this F&O is addressed.	Add a discussion to the QU notebook as to how the SR is met.	**DOCUMENTATION ONLY.** Adopt ASME Section 2 and document in QU notebook

ATTACHMENT 3

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11. External Event Impacts

External hazards were evaluated in the NMPS Individual Plant Examination of External Events (IPEEE) Submittal [26] in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Although the external event hazards in the NMPS IPEEE were evaluated to varying levels of conservatism, the results of the NMPS IPEEE are nonetheless used in this risk assessment to provide a conservative comparison of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment. The proposed ILRT interval extension impacts plant risk in a limited way. Specifically, the probability of a pre-existing containment leak being the initial containment failure mode given a core damage accident is potentially higher when the ILRT interval is extended. This impact is manifested in the plant risk profile in a similar manner for both internal events and external events. The spectrum of external hazards has been evaluated in the NMPS IPEEE by screening methods with varying levels of conservatism. Therefore, it is not possible at this time to incorporate a realistic quantitative risk assessment of all external event hazards into the ILRT extension assessment. As a result, external events have been evaluated as a sensitivity case to show that the conclusions of this analysis would not be altered if external events were explicitly considered.

The quantitative consideration of external hazards is discussed in more detail in Appendix B of this calculation. As can be seen from Appendix B, if the external hazard risk results of the NMPS IPEEE are included in this assessment (i.e., in addition to internal events), the change in LERF associated with the increase in ILRT interval from 10 years to 15 years is estimated at ~~1.91~~ 8.79 E-8/yr based on the most conservative methodology (NEI Interim Guidance). This increase is less than the range of 1E-07/yr to 1E-06/yr, putting it in Region III of the RG 1.174 LERF acceptability curve.

12. Conclusions

This section provides the principal conclusions of the ILRT test interval extension risk assessments as reported for the following:

- Previous generic risk assessment by the NRC
- NMPS-specific risk assessment for the at-power case, performed using three available methodologies (EPRI TR-104285, NEI Interim Guidance, and EPRI TR-1009325)
- General conclusions regarding the beneficial effects on shutdown risk

12.1 Previous Assessments

The NRC in NUREG-1493 has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years results in an imperceptible increase in risk. The estimated increase in risk is

As a sensitivity run, the estimated values for seismic and fire-induced CDF from Sections B.2 and B.3 above were used to calculate the Class 3b frequency. These values were not adjusted for sequences that will independently cause LERF, or will not cause LERF (factors used in other submittals to more accurately characterize the expected LERF from external events associated with the requested ILRT extension).

In order to determine the impact of external events on the proposed ILRT extension request, the impact on LERF was assessed in accordance with the NEI Interim Guidance. The NEI Interim Guidance was used because it yields the most conservative results relative to the other two approaches used in the Probabilistic Safety Assessment calculation.

The impact on the Class 3b frequency due to increases in the ILRT surveillance interval was calculated for external events using the relationships described in Section 6.0. The EPRI Category 3b frequencies for the 3 per 10-year, 10-year and 15-year ILRT intervals were quantified using the total external events CDF. The change in the LERF risk measure due to extending the ILRT interval from 3 in 10 years to 1 in 10 years, or to 1 in 15 years, including both internal and external hazard risk, is provided in Table B-1.

Table B-1
CALCULATION OF LERF IMPACT INCLUDING EXTERNAL EVENTS USING NEI INTERIM GUIDANCE

Baseline Case: External Events Class 3b Contribution Assumed to Equal Seismic and Internal Fires CDF						
	3b Frequency (3-per-10 year ILRT)	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15year ILRT)	LERF Increase (3-per-10 to 1-per-10)	LERF Increase (3-per-10 to 1-per-15)	LERF Increase (1-per-10 to 1-per-15)
External Event Contribution	4.35E-08	1.45E-07	2.18E-07	1.02E-07	1.75E-07	7.27E-08
Internal Event Contribution	9.06E-09	3.03E-08	4.54E-08	2.12E-08	3.64E-08	1.51E-08
Combined (Internal+External)	5.26E-08	1.76E-07	2.64E-07	1.23E-07	**2.11E-07**	8.79E-08

Table B-1 shows the sensitivity, under the bounding assumption that the entire external events CDF is applied to the Class 3b frequency, the total estimated increase in LERF is 2.11E-07/yr which is within the range of 1E-07/yr to 1E-06/yr (Region II of the RG 1.174 LERF acceptability curve). This study counted the full estimated seismic CDF and full estimated fire CDF against the 3b frequency. Note that the Class 3b frequency calculated for the internal events case (using the NEI Interim Guidance) represents only 1.38% (4.54E-8/yr / 3.30E-6/yr) of the total Internal Events CDF for the 15-year ILRT test interval.

As discussed above, significant conservatisms exist in the risk values used in the external events calculations. ~~This assessment is made more robust by including the sensitivity shown in Table B-~~

~~1 even though a calculated LERF value is available and specific calculations with these values are shown in Table B-2.~~ Per Reference B-4, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year (Region II, "Small Change" in risk), the risk assessment must also reasonably show that the total LERF from all hazards is less than 1E-5/yr. As shown in Reference B-6 the baseline total LERF from all hazards is 1.7E-06. Based on the LERF increase calculated using the NEI Interim Guidance (i.e., 2.11E-07), the total LERF for the requested change is 1.91E-06/yr. Thus these results meet the LERF criterion of RG 1.174.

~~The 2003 PRA model [B-6] seismic and fire contributions were used to create Table B-2 below. This table shows the external events LERF based on the seismic and fire percentage contributions from the 2003 PRA model [B-6]. For the most limiting case (in which the ILRT interval is extended from 3 in 10 years to 1 in 15 years), the combined delta-LERF result for the ILRT extension (from internal and external events) is calculated to be 4.59E-08/yr. These results meet the total LERF criterion of RG 1.174 (Region III of the RG 1.174 LERF acceptability curve).~~

~~Table B-2 EXTERNAL EVENTS CLASS 3b CONTRIBUTION GIVEN LERF VALUES USING 2003 PRA MODEL~~

~~Baseline Case: External Events Class 3b Contribution Given LERF Values Using Old Model~~						
	~~3b Frequency (3-per-10-year ILRT)~~	~~3b Frequency (1-per-10-year ILRT)~~	3b ~~Frequency (1-per-15year ILRT)~~	LERF ~~Increase (3-per-10 to 1-per-10)~~	~~LERF Increase (3-per-10 to 1-per-15)~~	~~LERF Increase (1-per-10 to 1-per-15)~~
~~External Event Contribution~~	~~2.38E-09~~	~~7.96E-09~~	~~1.19E-08~~	~~5.58E-09~~	~~9.56E-09~~	~~3.98E-09~~
~~Internal Event Contribution~~	~~9.06E-09~~	~~3.03E-08~~	~~4.54E-08~~	~~2.12E-08~~	~~3.64E-08~~	~~1.51E-08~~
~~Combined (Internal+External)~~	~~1.14E-08~~	~~3.82E-08~~	~~5.74E-08~~	~~2.68E-08~~	~~4.59E-08~~	~~1.91E-08~~

Therefore, incorporating external event hazard risk results into this analysis does not change the conclusion of the ILRT Extension risk assessment (i.e., increasing the Nine Mile Point ILRT interval from 3 in 10 years to either 1 in 10 years or 1 in 15 years is an acceptable plant change from a risk perspective).

B.5. References

B-1. Reference: R. P. Kennedy, "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations", Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August, 1999.

B-2. Nine Mile Point Nuclear Generating Plant Unit 1 Quantification Notebook QU Rev 0, December 2007.