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Constellation Energy
Nine Mile Point Nuclear Station

June 16, 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 - Technical Specification Section 6.5.7, 10 CFR 50 Appendix J Testing Program Plan

Pursuant to 10 CFR 50.90, Nine Mile Point Nuclear Station, LLC, (NMPNS) hereby requests an amendment to the Nine Mile Point Unit 1 (NMP1) Renewed Facility Operating License DPR-63. The proposed change to the Technical Specifications (TS) contained herein would revise NMP1 TS Section 6.5.7, "10 CFR 50 Appendix J Testing Program Plan," to allow a one-time extension of the Integrated Leak Rate Test (ILRT) interval for no more than five (5) years. The proposed change would allow the next ILRT for NMP1 to be performed within 15 years from the last ILRT as opposed to the current 10-year interval.

The proposed amendment is considered risk-informed. An evaluation has been performed to assess the risk impact of the proposed change. The risk assessment follows the guidelines of NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J;" the methodology used in Electric Power Research Institute Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals;" and the guidance of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The Enclosure provides a description and technical bases for the proposed change, an existing TS page marked up to show the proposed change, and the NMP1 risk assessment. NMPNS has concluded that the activities associated with the proposed change represent no significant hazards consideration under the standards set forth in 10 CFR 50.92. This submittal contains no regulatory commitments.

A017
NRR

Approval of the proposed amendment is requested by February 13, 2009, with implementation within 30 days of receipt of the approved amendment. Approval by the requested date is needed to support planning activities for the next NMP1 refueling outage, which is currently scheduled to begin in March 2009.

Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this license amendment request, with Enclosure, to the appropriate state representative.

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 license amendment request 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 16th day of June, 2008.

WITNESS my Hand and Notarial Seal:

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

Enclosure: Evaluation of the Proposed Change

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

ENCLOSURE

EVALUATION OF THE PROPOSED CHANGE

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ATTACHMENTS

1. Proposed Technical Specification Change (Mark-up)
2. Risk Impact Assessment of Extending Containment Type "A" Test Interval for Nine Mile Point Nuclear Station Unit 1 (NER-1M-013)

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Operating License DPR-63 for Nine Mile Point Unit 1 (NMP1). The proposed change would revise NMP1 Technical Specification (TS) Section 6.5.7, "10 CFR 50 Appendix J Testing Program Plan," to allow a one-time extension of no more than five (5) years for the Type A, Integrated Leakage Rate Test (ILRT) interval. This change is a one-time exception to the 10-year frequency of the performance-based leakage rate testing program for Type A tests as defined in NEI 94-01, Revision 0, "Industry Guideline For Implementing Performance-Based Option of 10CFR Part 50, Appendix J" (Reference 1), pursuant to 10 CFR 50, Appendix J, Option B. The requested exception is to allow the ILRT to be performed within 15 years from the last ILRT, which was completed on June 8, 1999.

The current 10-year ILRT for NMP1 is due by June 8, 2009, which would require the test to be performed during the spring 2009 refueling outage. The proposed exception would allow the next ILRT for NMP1 to be performed within 15 years from the last ILRT as opposed to the current 10-year interval. Deferral of the ILRT for an additional five years would result in significant savings in radiation exposure to personnel, cost, and critical path time during the 2009 refueling outage.

The technical analysis for the proposed license amendment is based on risk related and non-risk related considerations. A risk analysis performed by Nine Mile Point Nuclear Station, LLC (NMPNS) has concluded that the increases in estimated person-rem and large early release frequency (LERF) are consistent with the guidance provided in Regulatory Guide (RG) 1.174 (Reference 2) and NUREG-1493 (Reference 3). The technical analysis also demonstrates that defense-in-depth would be provided by the low increase in the conditional containment failure probability, and by non-risk based considerations such as the ILRT and containment inspection history, and the ongoing primary containment local leakage rate testing and inservice inspection programs.

2.0 DETAILED DESCRIPTION

2.1 Description of the Proposed Change

In NMP1 TS Section 6.5.7.a, "10 CFR 50 Appendix J Testing Program Plan," currently states:

> *A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:*
>
> 1. *Type A tests will be conducted in accordance with ANSI/ANS 56.8-1994 and/or Bechtel Topical Report BN-TOP-1, and*
> 2. *The first Type A test following approval of this Specification will be a full pressure test conducted approximately 70, rather than 48, months since the last low pressure Type A test.*

The proposed change would add a third exception to TS 6.5.7.a to specify the date of the next required Type A test. Specifically, the added exception would state:

3. *Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Section 9.2.3: The first Type A test performed after the June 8, 1999 Type A test shall be performed no later than June 7, 2014.*

Attachment 1 to this Enclosure contains existing TS page 355 marked up to show the proposed changes to TS Section 6.5.7.a.

2.2 Background

2.2.1 Description of Primary Containment System

The primary containment is described in Updated Final Safety Analysis Report (UFSAR) Sections VI and XVI-B. The Mark I pressure suppression containment system consists of the drywell, the pressure suppression chamber (torus), the connecting vent system between the drywell and suppression chamber, penetrations, isolation valves, access openings (hatches, air locks), vacuum relief system, and containment cooling systems. The primary containment was designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition.

The drywell, which houses the reactor vessel and reactor coolant recirculation loops, is a free-standing, low-leakage steel pressure vessel designed to contain the reactor coolant that would be released during a postulated pipe rupture, and acts as an essentially leak-tight boundary against the uncontrolled release of radioactive materials to the environment. The vessel is in the shape of an inverted light bulb. The lower section of the drywell is spherical with a 70 foot diameter and the upper section is cylindrical with a 33 foot diameter. The cylindrical section has a flanged and bolted elliptical top cover. The overall height of the drywell is approximately 118 feet. The drywell is designed for an internal pressure of 62.0 psig, an external design pressure of 2.0 psig, and a maximum internal temperature of 310°F. The drywell is enclosed in a reinforced concrete structure with a 2 to 3 inch air gap between the shell plate and concrete. The concrete provides radiological shielding during normal plant operations. Shielding above the drywell is provided by removable, segmented reinforced concrete plugs.

The suppression chamber (torus) is a free-standing, toroidal shaped steel pressure vessel designed to hold a large volume of water (suppression pool) for use as a heat sink for postulated transients and accident conditions in which the normal heat sink may become unavailable. The suppression pool serves as the water source for the Core Spray System and the Containment Spray System. The suppression chamber is located below and completely encircles the drywell, with a major centerline diameter of 123 feet 4 inches and a cross-sectional inside diameter of 27 feet. It is constructed of twenty (20) mitered sections to eliminate the need for compound curves in the shell. The suppression chamber shell is stiffened by an internal ring girder located at each of the mitered joints, and is externally supported by a series of columns, saddles, and cross-bracing at each of the mitered sections.

Ten (10) vent lines from the drywell enter the suppression chamber and connect to a vent header. The vent header forms a continuous ring around the inside of the suppression chamber. Connected to the vent header are 120 downcomers that terminate below the water level maintained in the suppression chamber.

2.2.2 Testing Requirements of 10 CFR 50, Appendix J

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. The limitation of containment leakage provides assurance that the containment would perform its design function following an accident up to and including the

plant design basis accident. Appendix J identifies three types of required tests: (1) Type A tests, intended to measure the primary containment overall integrated leakage rate; (2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for primary containment penetrations; and (3) Type C tests, intended to measure containment isolation valve leakage rates. Type B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (integrated) containment leakage rate and serve to ensure continued leakage integrity of the containment structure by evaluating those structural parts of the containment not covered by Type B and C testing.

10 CFR 50, Appendix J, was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." On February 10, 1997, the NRC approved License Amendment No. 159 for NMP1 (Reference 4), authorizing the implementation of 10 CFR 50, Appendix J, Option B. The amendment added TS Section 6.16, "Primary Containment Leakage Rate Testing Program," to require Type A, B and C testing in accordance with RG 1.163, "Performance-Based Containment Leak-Test Program" (Reference 5). TS Section 6.16 was subsequently re-numbered as TS Section 6.5.7 in License Amendment No. 181, issued by NRC letter dated April 23, 2003 (Reference 6). RG 1.163 specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994 (Reference 7), subject to several regulatory positions stated in the RG. NEI 94-01 specifies an initial Type A test interval of 48 months but allows an extended interval of ten years, based upon two consecutive successful tests.

Adoption of the Option B performance-based containment leakage rate testing program altered the frequency of measuring primary containment leakage in Type A, B, and C tests but did not alter the basic method by which Appendix J leakage rate testing is performed. The test frequency is based on an evaluation of the "as found" leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded. The changes to the Type A (ILRT) test frequency allowed by Option B do not result in an increase in containment leakage. Similarly, the proposed one-time five year extension for the ILRT frequency will not result in an increase in containment leakage.

The allowed frequency for Type A testing, as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493 (Reference 3). The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident risks on containment leak-tightness for five reactor/containment types, including a General Electric-designed boiling water reactor with a Mark I containment (NMP1 has a Mark I containment). The following observations regarding extending the test frequency are made in NUREG-1493, Section 10.1.2:

- Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between ILRTs is possible with minimal impact on public risk.

- Type B and C tests can identify the vast majority (greater than 95 percent) of all potential leakage paths.

- Performance-based alternatives to current local leakage-testing requirements are feasible without significant risk impacts. Since under existing requirements, leakage contributes less than 0.1 percent of overall accident risk, the overall impact is very small.

10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," allows exceptions to the guidelines of RG 1.163. That section states: "The regulatory guide or other implementation document used by a licensee or applicant for an operating license under this part or a combined license under part 52 of this chapter to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide." Since exceptions meeting the stated requirements are permitted, this license amendment request does not require an exemption from 10 CFR 50, Appendix J, Option B.

3.0 TECHNICAL EVALUATION

The following sections describe both traditional engineering considerations and the plant-specific risk assessment that support the proposed change to TS Section 6.5.7.

3.1 Engineering Evaluations

3.1.1 10CFR50, Appendix J, Option B Plant Specific Implementation

As noted previously, License Amendment No. 159 implemented 10 CFR 50, Appendix J, Option B, for NMP1 by adding TS Section 6.16 (subsequently re-numbered as Section 6.5.7). TS Section 6.5.7 requires Type A, B, and C testing in accordance with RG 1.163, which endorses the methodology for complying with Option B identified in NEI 94-01. The surveillance frequency for Type A testing in NEI 94-01 is at least once every 10 years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart demonstrate the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

The performance leakage rates are calculated in accordance with NEI 94-01, Section 9.1.1. The performance leakage rate includes the Type A Upper Confidence Limit (UCL) at 95% plus the as-left minimum pathway leakage rate for all Type B and C pathways not in service, isolated, or not lined up in their test position.

The two most recent Type A tests at NMP1 have been satisfactory with leakage rates for the 1993 and 1999 Type A tests being 0.4634 wt% / day and 0.5045 wt% / day, respectively. These results are less than the maximum allowable containment leakage rate (L_a at P_a), of 1.5% containment air weight per day at a pressure of 35 psig. Based on these two consecutive successful tests and License Amendment No. 159, the current ILRT interval requirement for NMP1 is 10 years.

3.1.2 Plant Testing and Inspection Programs

In addition to periodic Type A testing, various inspections and tests are routinely performed to assure primary containment integrity. These include Type B and C testing performed in accordance with Appendix J, Option B; inspection activities performed as part of the American Society of Mechanical Engineers (ASME) Code Section XI (Subsection IWE) inspection program; inspection of drywell and torus surfaces and structural elements; inspections of torus exterior surfaces and supports; and inspections

of drywell interior coatings. Further discussion of these testing and inspection programs is provided below. The aggregate results of these tests and inspections provide a high degree of assurance of continued primary containment integrity.

3.1.2.1 Type B and Type C Testing Program

The NMP1 Appendix J, Type B and Type C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and valves within the scope of the program as required by 10 CFR 50, Appendix J, Option B, and RG 1.163. The Type B and C test program consists of local leak rate testing of penetrations with a resilient seal, expansion bellows, double-gasketed manways, hatches and flanges, drywell airlocks, and containment isolation valves that serve as a barrier to the release of the post-accident primary containment atmosphere. These components are tested with air or nitrogen at a pressure greater than or equal to 35 psig (P_a). The results of the test program are used to ensure that proper maintenance and repairs are made on the primary containment components over their service life. The Type B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining the leakage from these components below appropriate limits.

As previously noted, Type B and Type C testing evaluates all but a small portion of potential containment leakage pathways. This amendment request does not affect the scope, performance, or scheduling of Type B or Type C tests. Type B and Type C testing will continue to provide a high degree of assurance that primary containment integrity is maintained.

3.1.2.2 Primary Containment Inspection Requirements – ASME Code Section XI, Subsection IWE

10 CFR Part 50, Appendix J, Option B, Section III.A, states: "A general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment system." This inspection is also conducted during two other refueling outages before the next Type A test if the interval for the Type A test has been previously extended to 10 years, in order to allow for early discovery of structural deterioration.

Effective September 1996, the NRC amended 10 CFR 50.55a to endorse Subsections IWE and IWL of the ASME Code, Section XI, 1992 Edition including 1992 Addenda. These subsections contain inservice inspection (ISI) and repair/replacement rules for Class MC (metal containment) and Class CC (concrete containment) components. The NMP1 primary containment is a free-standing steel containment, to which the requirements of Subsection IWE apply.

The Subsection IWE containment inspection requirements are implemented at NMP1 through the Containment Inservice Inspection Plan and Schedule (NMP1-IWE-003, referred to herein as the IWE ISI program). The program contains detailed inservice inspection requirements for Class MC components in accordance with 10 CFR 50.55a(b)(2)(vi) and (ix), and the ASME Code, Section XI, 1998 Edition with no addenda. The use of the 1998 Edition of ASME Section XI, Subsection IWE for the NMP1 IWE ISI program was requested in a relief request that was submitted to the NRC by letter dated October 28, 1999 (Reference 14). The NRC approved this relief request by letter dated August 17, 2000 (Reference 15). There are no other relief requests in effect for this program.

The general visual examination requirements specified in the IWE ISI program satisfy the visual examination requirements specified in Option B. The First Interval Containment Inservice Inspection Program became effective on December 26, 1999 and is scheduled to end December 25, 2009. The three inspection periods during the first inspection interval are shown in the following table.

NMP1 INSERVICE INSPECTION PERIODS				
INSPECTION PERIOD	PERIOD START DATE	PERIOD END DATE	REFUEL OUTAGE	REFUEL OUTAGE YEAR
1	December 26, 1999	December 25, 2002	RFO-15 RFO-16	1999 2001
2	December 26, 2002	December 25, 2006	RFO-17 RFO-18	2003 2005
3	December 26, 2006	December 25, 2009	RFO-19 RFO-20	2007 2009

Examinations are performed in accordance with non-destructive examination procedure NDEP-VT-2.05, "ASME Section XI IWE/IWL Visual Examination." This procedure provides the overall requirements and acceptance criteria for visual examinations in accordance with ASME Section XI, Article IWE, as delineated in the NMP1 IWE ISI program, in accordance with the ASME Code, Section XI, 1998 Edition with no Addenda.

The containment shell acceptance criteria are based on the calculated corrosion rate and the projected wall thickness at the end of the period of extended operation (i.e., 2029). The corrosion rate criteria are based on maintaining a wall thickness of greater than the minimum design value. Acceptance criteria have been established and are documented in procedure NDEP-VT-2.05 and in the IWE ISI program. Flaws identified during inspections are described as nicks, gouges, arc strikes, cracking, rust, or pitting. For each flaw, varying levels of severity are described and are evaluated as acceptable, unacceptable, or requiring further evaluation. Examples of criteria are provided below:

- Nicks & Gouges: depth > 10% of metal thickness; unacceptable for continued service.
- Arc Strikes (all) and Metal Cracking (all): unacceptable for continued service.
- Rust, no base metal loss (minor): acceptable.
- Rust, < 5% base metal loss (medium): acceptable.
- Rust, > 5% base metal loss (major): unacceptable for continued service.

Examination of pressure-retaining bolted connections and evaluation of containment bolting flaws or degradation are performed in accordance with the requirements of 10 CFR 50.55a(b)(2)(ix)(G) and 10 CFR 50.55a(b)(2)(ix)(H).

Containment inspections will continue to be performed during the proposed 5-year extension of the Type A test interval (June 2009 through June 2014), in accordance with the IWE ISI program. The IWE ISI program requires a minimum of one inspection during each inspection period of the inspection interval. This extension will coincide with the first and second inspection periods of the second interval of the IWE ISI program.

3.1.2.4 Drywell IWE-1240 Augmented Inspection

During the 2003 NMP1 refueling outage, a general visual examination of 100 percent of the accessible portions of the interior surface of the drywell shell was performed. Six localized areas, coinciding with the location of the drywell area coolers, were observed to have significant corrosion. In accordance with ASME Section XI, Subsection IWE, a detailed visual examination (VT-1) was performed of the six localized areas and characterized the corrosion as "major" (i.e., greater than 5 percent of the base metal was judged to be lost). The condition was entered into the corrective action program and a rigorous engineering evaluation was performed.

To ascertain the actual thickness of the drywell shell at these locations, volumetric (UT) examinations were performed at the four most severe locations. Four individual UT measurements were taken by cleaning the corrosion from the base metal, conducting a continuous scan, and recording the lowest value. The results of the UT examinations ranged from 1.106 inches to 1.131 inches. A comparison of these results against the minimum design thickness value of 1.049 inches concluded that the drywell shell was acceptable for continued service.

Subsequent to the evaluations performed during the 2003 refueling outage, an engineering calculation was performed that projected the time necessary to reach minimum design thickness for the drywell shell. Using the volumetric examination results (minimum value of 1.106 inches) and minimum design value (1.049 inches), the available margin was determined to be 57 mils. Using the originally assumed corrosion allowance (62.5 mils over 40 years), it was calculated that it would take 36 years from 2003 (i.e., until 2039) to reach the minimum design thickness. This is 10 years beyond the end of renewed operating license, which expires in 2029.

Another method, using a newer approved corrosion rate, was also used to project the year that minimum wall thickness would be reached for the drywell shell. A corrosion rate of 1.26 mils/yr is used in the Torus Corrosion Monitoring Program (described below) to evaluate volumetric examination results. This value is documented in an NRC safety evaluation dated August 11, 1994 (Reference 8). The use of this corrosion rate is appropriate since the material of the drywell shell is essentially the same as the material for the torus shell for the purposes of corrosion resistance. The drywell shell is made of ASTM-212 Grade B Firebox steel whereas the torus shell is made of ASTM-201 Grade B Firebox steel. The environment in the torus is also the same as, or more severe than, the drywell environment. The torus is approximately half full of demineralized water and the remainder is a nitrogen inerted atmosphere. The drywell is entirely a nitrogen inerted atmosphere. Therefore, since the materials and environments are essentially the same, it is appropriate to use the approved torus corrosion rate for the drywell shell. Performing the calculation, 57 mils of margin divided by the 1.26 mils/yr corrosion rate yields 45 years until the minimum design thickness of the drywell shell is reached (19 years beyond the end of the renewed operating license). Note that the drywell shell minimum wall thickness is based on a design pressure of 62 psig, whereas the peak calculated accident pressure is 35 psig. Calculation of the minimum wall thickness based on a containment pressure of 35 psig would yield additional margin.

Based on the analyses described above, NMPNS concluded that the NMP1 drywell shell could perform its intended function following the 2003 refueling outage and throughout the remainder of the renewed operating license. Continued monitoring of the above-identified localized areas of the drywell shell has been implemented. The attributes of this additional monitoring program are described in the NMPNS letter to the NRC dated April 4, 2006 (Reference 17).

During the March 2007 NMP1 refueling outage the drywell shell was inspected per the IWE ISI examination schedule. Volumetric and visual examinations of the drywell shell were performed in specified areas in order to monitor the corrosion rate. The minimum recorded UT measurement of 1.089 inches exceeded the established minimum wall thickness criteria of 1.049 inches and was therefore acceptable.

To more accurately determine material loss over a period of time, grids overlapping the four areas measured in 2007 were marked on the shell using a permanent marking means so that these same points can be measured again in future refueling outages. Each of the grids provides for a potential of 121 data points. The surfaces where the grids were applied were cleaned to the extent practical; however, some measurements may not be obtainable due to surface irregularities. The significant number of data points where wall thickness can be measured provides a solid basis for determining corrosion rates during the remaining period of plant operation.

The drywell augmented inspection/monitoring plan is more effective than Appendix J Type A tests (ILRT) for identifying degrading minimum wall conditions, since the Type A test will only identify an actual breach in the pressure boundary.

3.1.2.5 Torus Corrosion Monitoring

The NMP1 suppression chamber (torus) is constructed of ASTM-201 Grade B Firebox steel plates with a certified minimum thickness of 0.460 inches. This value included an original corrosion allowance of 0.0625 inches, which was added to the minimum wall thickness required by the applicable design codes. However, subsequent consideration of hydrodynamic loads resulting from a loss of coolant accident (LOCA) and safety relief valve (SRV) actuation resulted in a reduction of the corrosion allowance. To establish reasonable assurance that the revised minimum wall thickness of 0.431 inches is not reached, NMPNS monitors torus wall thickness and corrosion rate. Determination of torus corrosion rates is an ongoing activity that considers inspection results and the remaining corrosion allowance.

The NMP1 Torus Corrosion Monitoring Program has been developed to monitor the torus shell material thickness and ensure it is maintained within the bounds of the qualification bases. Assessment of observed torus shell conditions ensures that timely action can be taken to correct degradation that could lead to loss of the intended function. The program is based on a commitment to periodically monitor torus condition as described in the NRC's safety evaluation dated August 11, 1994 (Reference 8). The program was also reviewed and accepted by the NRC as part of the License Renewal application review (Reference 9). The program focuses on condition monitoring in the form of component inspections and analysis, including the following elements:

- Periodic torus wall UT thickness measurements are obtained over a pre-defined grid system.
- Torus material coupons are periodically removed, analyzed, and compared to the results of the UT measurements.
- Visual inspections of accessible external surfaces of the torus support structure are performed. Included are inspections of the existing coatings of the base plates, anchor bolts, pipe columns, stiffener plates, tie rod cross braces, etc. Additionally, the inspections include evaluations of concrete support elements for evidence of deterioration.

Both the UT thickness measurement and the corrosion coupon data sets are analyzed, and the most conservative corrosion rate for a particular torus bay is used to evaluate torus condition. To date, the

coupon corrosion rate has been consistently lower than the corrosion rate determined via UT thickness measurement. Monitoring in this manner ensures the torus shell material will not be reduced to less than the minimum required wall thickness, and that any degradation is detected before there is a loss of intended function.

Review of previously obtained corrosion coupon data determined a maximum coupon corrosion rate of 0.534 mils/yr, which is significantly less than the 1.26 mils/yr value identified as an upper bound rate in the NRC's safety evaluation dated August 11, 1994 (Reference 8). The next coupon corrosion rate determination is scheduled for the spring 2009 refueling outage.

Analysis of February 2007 UT thickness measurements at selected torus bottom mid-bay locations indicated that the smallest local/individual wall thickness obtained during this survey was 0.4452 inches with calibration adjustment applied. All wall thickness measurements exceeded the allowable average minimum wall thickness of 0.431 inches and thus were considered acceptable.

The most recent inspections of the torus external structures, performed during the 2007 refueling outage, found the condition of the structures acceptable with no signs of missing or loose hardware, spalled concrete or major degradation that would impact the structural integrity of the torus structure.

The Torus Corrosion Monitoring Program is more effective than Appendix J Type A tests (ILRT) for identifying degrading minimum wall conditions, since the Type A test will only identify an actual breach in the pressure boundary.

3.1.3 Plant Operational Performance

During power operation the NMP1 primary containment is inerted with nitrogen to maintain oxygen concentration within TS limits. As a result, the primary containment is maintained at a slightly positive pressure during power operation. Maintaining the containment pressurized at power assures that gross containment leakage that may develop during power operation will be detected. A drywell pressure alarm will alert operators that the drywell pressure is abnormal (greater than 2 psig or less than 1 psig).

3.1.4 NRC Information Notice 92-20, Inadequate Local Leak Rate Testing

NRC Information Notice 92-20 was issued to alert licensees to problems with local leak rate testing of two-ply stainless steel bellows used on piping penetrations at some plants. Specifically, local leak rate testing could not be relied upon to accurately measure the leakage rate that would occur under accident conditions since, during testing, the two plies in the bellows were in contact with each other, restricting the flow of the test medium to the crack locations. Any two-ply bellows of similar construction may be susceptible to this problem.

As described in UFSAR Section VI-B.2.2, NMP1 fluid lines with temperatures above 150°F have a guard pipe between the hot fluid line and the penetration attachment to the drywell steel, in addition to the double-seal arrangement. In this manner, the penetration is protected against over pressurization should the hot line rupture inside the penetration. The hot fluid from a rupture of this type would be vented into the drywell by the guard pipe. The guard pipes are designed to the same pressure and temperature as the fluid line. The hot fluid penetrations have two expansion bellows, one inside the drywell and one outside. The main steam line penetration bellows have been modified by adding "clam-shell" bellows joints over the existing joints, which were removed. These are designed to accommodate the thermal expansion as well as any movement due to a line rupture. As a final precaution, the inner bellows are designed for a lower pressure (62 psig) than the outside bellows (120 psig) to assure inward leakage in the event of

failure. The pipe sleeve which attaches to the drywell is designed for 62 psig but because of structural thicknesses, the sleeve can withstand substantially higher pressures.

Type B tests are applicable to the primary containment penetrations with expansion bellows. Test results have been satisfactory, and the test interval is 120 months for all penetration expansion bellows except the main steam and feedwater penetrations. Since the main steam and feedwater lines are potential post-accident secondary containment bypass pathways, these penetration expansion bellows have a fixed 30-month Type B test interval.

The NMPNS engineering study, "Risk Impact Assessment of Extending Containment Type A Test Interval for Nine Mile Point Station Unit 1," provided as Attachment 2 to this Enclosure, takes into consideration the potential failure of containment bellows assemblies.

An augmented VT-1 visual examination of the containment penetration bellows will also be performed using enhanced techniques qualified for detecting stress corrosion cracking, per NUREG-1611. This is an addition to the IWE ISI program and has been reviewed and accepted by the NRC as part of the License Renewal application review (Reference 9).

3.1.5 Through-wall Torus Shell Crack at James A. Fitzpatrick Nuclear Power Plant

A through-wall torus shell crack was discovered at the James A. Fitzpatrick Nuclear Power Plant (JAF) on June 27, 2005. NMPNS reviewed the issue for applicability to NMP1, and documented the results in the corrective action program.

The JAF high pressure coolant injection (HPCI) turbine exhaust line that discharges into the suppression pool is open ended and does not have an end cap or a sparger. NMP1 does not have a steam turbine driven HPCI system. NMP1 systems that may discharge steam to the suppression pool are:

- A 1½ inch diameter Emergency Condenser (EC) System vent line. The vent pipe terminates 2'-6" below the torus water surface and has a sparger allowing for steam condensing.

- The 14 inch diameter Electromatic Relief Valve (ERV) discharge lines. These lines discharge through "Y" Quenchers that terminate 5'-11" below the torus water surface thereby allowing for steam condensing.

The NMP1 system configurations described above would not introduce the type of event that occurred at JAF.

3.1.6 Generic Letter 87-05, Request for Additional Information - Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells

Generic Letter 87-05 described drywell shell degradation which occurred at Oyster Creek Nuclear Generating Station as a result of water intrusion into the air gap between the outer drywell surface and the surrounding concrete and subsequent wetting of the sand cushion at the bottom of the air gap. This area of the drywell is not preserved with a protective coating as the wall surface above it is. The initial response to this generic letter for NMP1 was provided in a letter to the NRC dated May 13, 1987 (Reference 10), in which it was stated that the following actions would be taken:

1. Visual examination of the drain lines in the sand cushion will be performed using remote video equipment during the March 1988 refueling outage.

2. To assure that the reactor head cavity wall bypass leakage drain is properly functioning, a visual inspection of the drain line will be made using remote video equipment during the March 1988 refueling outage. In addition, the drain line flow switch will be checked for proper operation.

3. After the initial inspections described above are performed, the need to perform additional inspections will be re-evaluated.

The above actions were completed and reported to the NRC by letter dated June 26, 1990 (Reference 11). The inspections found all drain lines in the sand cushion to be operable, and factors that could cause a change in their condition were not present. In addition, examination for evidence of moisture in the air gap between the outer surface of the drywell and surrounding concrete indicated that mechanisms which could cause aggravated corrosion attack on the drywell wall were not present.

Additional inspections performed during the spring 2007 refueling outage were also satisfactory. Five of the ten drains were inspected by inserting a fiber optical probe from the torus room open ended drain line all the way up to the sand cushion area under the drywell shell. Review of the video recordings for all five probes indicated a dusty, dry environment. An expected nominal general corrosion of the interior of the carbon steel drain line was observed and found to be acceptable. There were no observed indications of water leakage into the sand cushion area. As a result of this 2007 sand cushion inspection, no further action is warranted.

3.1.7 Monitoring of Drywell Interior Coating

In addition to the inspections performed in accordance with the IWE ISI program, periodic visual inspections of the coating on accessible interior surfaces of the drywell shell and drywell head are performed to identify evidence of deterioration. The inspections are performed every refueling outage to identify any visible defects including blistering, cracking, flaking, peeling, and physical or mechanical damage. When degraded coatings are identified, evaluations are performed to determine any necessary actions (e.g., repair, removal, or replacement).

3.1.8 Moisture Barrier Inspection

The moisture barrier between the drywell shell and the concrete floor at elevation 225-ft prevents moisture from entering the gap between the concrete and the drywell shell and protects against corrosion of the shell in the inaccessible areas below the 225-ft level. The moisture barrier was inspected during the spring 2001 refueling outage in accordance with the IWE ISI program. The inspection found some degradation of the moisture barrier sealant; however, no unacceptable degradation in the visible areas of the drywell shell adjacent to the moisture barrier was found. The moisture barrier was repaired during the spring 2001 refueling outage by applying new sealant where required.

3.2 Plant-Specific Risk Assessment

An evaluation was performed to assess the risk impact of a one-time extension of the NMP1 containment ILRT interval from 10 years to 15 years. The risk assessment follows the guidelines of NEI 94-01, the methodology used in EPRI TR-104285 (Reference 12), and the guidance of RG 1.174 (Reference 2) on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change the licensing basis of a plant. In addition, a risk assessment was performed using two more recent studies for comparison purposes. The methodologies used are those presented in the NEI Interim Guidance (Reference 16) and in EPRI TR-1009325 (Reference 13). Although these methodologies

generally produce more conservative results than do the earlier methodologies, they build upon the work of the earlier studies, and much of the analyses developed from application of the EPRI TR-104285 methodology remains applicable for use in these more recent studies. The analysis uses the current NMP1 PRA model that has been updated to meet the guidance of RG 1.200 (Reference 18). Peer review findings were considered and found to not impact the risk assessment, as further discussed in Attachment 2 to this Enclosure.

The findings of the NMP1 assessment confirm the general findings of previous studies on a plant-specific basis, including severe accident category frequencies, the containment failure modes, the TS-allowed leakage, and the local population surrounding NMP1. The following are the conclusions regarding the risk impact of extending the ILRT interval from 10 to 15 years:

1. There is no change in the at-power Core Damage Frequency (CDF) associated with the proposed change; therefore, the RG 1.174 acceptance guidelines for CDF are met.

2. The increase in Large Early Release Frequency (LERF) is determined to be between 3.43E-10/yr and 3.81E-9/yr. The range in LERF is based on the use of the three different methodologies in the risk analysis. Guidance in RG 1.174 defines very small changes in LERF as those that are less than 1E-7/yr. Therefore, the proposed change results in a very small change to the NMP1 risk profile.

3. The total integrated plant risk increases by between 0.0056% and 0.048%. The range in total integrated plant risk increase is based on the use of the three different methodologies in the risk analysis. Therefore, the change in risk impact, when compared to that of other severe accident risks, is negligible.

4. The change in Conditional Containment Failure Probability (CCFP) is less than 1%. This is insignificant and reflects sufficient defense-in-depth. The additional risk associated with potential containment shell corrosion was also evaluated, with a similar conclusion that the proposed change poses negligible risk.

5. Incorporating external event impacts into this analysis does not change the conclusion of this risk assessment (i.e., increasing the NMP1 ILRT interval from 10 years to 15 years is an acceptable plant change from a risk perspective).

Details of the risk assessment are contained in Attachment 2 to this Enclosure.

3.3 Conclusions

Based on the previous ILRT tests conducted at NMP1, which confirm that the primary containment structure exhibits extremely low leakage, NMPNS concludes that the one-time extension of the containment ILRT interval from 10 to 15 years represents minimal risk to increased leakage. The risk is minimized by continued Type B and C testing performed in accordance with Option B of 10 CFR 50 Appendix J, inspection activities performed as part of the plant IWE ISI program, the torus corrosion monitoring program, inspections of drywell interior coatings, and by operating experience with a containment that normally operates at a positive pressure (i.e., the pressure from containment inerting). In the aggregate, these provide continuing confidence in containment integrity.

This experience is supplemented by risk analysis studies, including the NMP1 risk analysis provided in Attachment 2 to this Enclosure. The findings of the NMP1 risk assessment confirm the general findings

of previous studies, on a plant-specific basis, that extending the ILRT test interval from 10 years to 15 years results in a very small change to the NMP1 risk profile.

NMPNS therefore concludes that the requested one-time 5-year extension of the ILRT interval does not endanger public health and safety.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR Part 50, Appendix J, Option B, requires that licensees' primary reactor containments meet the leakage rate requirements as delineated by Appendix J. This requirement is met by performance of Type A, B, and C leakage rate testing on the primary containment and its associated components (e.g., valves, penetrations). The leakage rate test results are compared to allowable leakage rate acceptance criteria set forth by Appendix J. NMP1 TS Section 6.5.7, "10 CFR 50 Appendix J Testing Program Plan," invokes Appendix J requirements.

TS Section 6.5.7 requires that the leakage rate testing of the containment be performed in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and in accordance with NRC RG 1.163 dated September 1995 with NRC-approved exceptions. Regulatory Position C.1 of RG 1.163 states that licensees should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01. Section 11.0 of NEI 94-01 references Section 9.0, which would require that ILRTs be performed for NMP1 within 10 years from the date of their last performance. TS Sections 3.3.3.a and 6.5.7.c specify the maximum allowable containment leakage rate, L_a, as 1.5% of primary containment air weight per day.

NMP1 will continue to comply with the requirements of 10 CFR 50, Appendix J, with the proposed ILRT extension. No other regulations or TS are affected by the proposed amendment.

4.2 Precedent

The NRC has approved similar risk-informed license amendment requests relating to a one-time extension of the ILRT interval for a number of plants. Examples include:

- Edwin I. Hatch Nuclear Plant, Unit 2 (License Amendment No. 187 issued by NRC letter dated February 1, 2005 - TAC No. MC2761).
- Vermont Yankee Nuclear Power Station (License Amendment No. 227 issued by NRC letter dated August 31, 2005 - TAC No. MC4662).
- Cooper Nuclear Station (License Amendment No. 224 issued by NRC letter dated October 3, 2006 - TAC No. MC9732).

NMPNS considers these valid precedents because NMP1 shares the following similarities with these three plants:

- All are boiling water reactors with Mark I pressure suppression systems, consisting of the drywell, the pressure suppression chamber, and the interconnecting piping.

- All perform Type A, Type B, and Type C testing in accordance with 10 CFR 50, Appendix J, Option B, and perform inspections in accordance with Subsection IWE of Section XI of the ASME Boiler and Pressure Vessel Code.

- All operate with a positive pressure in containment which is monitored and provides an indication of leakage if a pressure decrease occurs.

- The risk assessments all used the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, and the regulatory guidance from RG 1.174.

4.3 Significant Hazards Consideration

Nine Mile Point Nuclear Station, LLC (NMPNS) is requesting revisions to Nine Mile Point Unit 1 (NMP1) Technical Specification (TS) Section 6.5.7, "10 CFR 50 Appendix J Testing Program Plan." The proposed change would allow a one-time extension to the current interval for the primary containment integrated leak rate test (ILRT), which is required to be performed by 10 CFR 50, Appendix J. The current test interval of 10 years would be extended on a one-time basis to no longer than 15 years from the last ILRT.

NMPNS has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change involves a one-time extension of the primary containment ILRT interval from 10 years to 15 years. The proposed change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The primary containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed change.

   Continued containment integrity is assured by the established programs for local leak rate testing and inservice/containment inspections, which are unaffected by the proposed change. As documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," dated September 1995, industry experience has shown that local leak rate tests (Type B and C) have identified the vast majority of containment leakage paths, and that ILRTs detect only a small fraction of containment leakage pathways.

   The potential consequences of the proposed change have been quantified by analyzing the changes in risk that would result from extending the ILRT interval from 10 years to 15 years. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be of a magnitude that NUREG-1493 indicates is imperceptible. NMPNS has also analyzed the increase in risk in terms of the frequency of large early releases

from accidents. The increase in the large early release frequency resulting from the proposed change was determined to be within the guidelines published in NRC Regulatory Guide 1.174. Additionally, the proposed change maintains defense-in-depth by preserving a reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation. NMPNS has determined that the increase in conditional containment failure probability due to the proposed change would be less than one percent. Therefore, it is concluded that the proposed one-time extension of the primary containment ILRT interval from 10 years to 15 years does not significantly increase the consequences of an accident previously evaluated.

Based on the above discussion, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change involves a one-time extension of the primary containment ILRT interval. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change in the manner in which the plant is operated or controlled.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed one-time extension of the primary containment ILRT interval does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the 10 CFR 50 Appendix J Testing Program Plan, as defined in the TS, exist to ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant safety analyses is maintained. The overall containment leakage rate limit specified by the TS is maintained, and Type B and C containment leakage tests will continue to be performed at the frequency currently required by the TS.

NMP1 and industry experience strongly support the conclusion that Type B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by the ILRT is small. Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment will not degrade in a manner that is detectable only by an ILRT. Additionally, the on-line containment monitoring capability that is inherent to inerted boiling water reactor containments allows for the detection of gross containment leakage that may develop during power operation. This combination of factors ensures that the margin of safety that is inherent in plant safety analyses is maintained. Furthermore, a risk assessment using the current NMP1

Probabilistic Risk Assessment internal events model concluded that extending the ILRT test interval from 10 years to 15 years results in a very small change to the NMP1 risk profile.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NMPNS concludes that the proposed change presents no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a surveillance requirement regarding leak rate testing of the primary containment. However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995

2. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002

3. NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995

4. Letter from D. S. Hood (NRC) to B. R. Sylvia (NMPC), dated February 10, 1997, Issuance of Amendment for Nine Mile Point Nuclear Station Unit No. 1 (TAC Nos. M96081 and M89522)

5. Regulatory Guide 1.163, "Performance Based Containment Leak-Test Program," dated September 1995

6. Letter from P. S. Tam (NRC) to J. T. Conway (NMPNS), dated April 23, 2003, Nine Mile Point Nuclear Station, Unit No. 1 - Issuance of Amendment Re: Administrative Controls (TAC No. MB2441)

7. ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements," August 1994

8. Letter from D. S. Brinkman (NRC) to B. R. Sylvia (NMPC), dated August 11, 1994, Approval of Reduction Factors for Condensation Oscillation Loads in Nine Mile Point Nuclear Station Unit No. 1 (NMP1) Torus (TAC No. M85003)

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

10. Letter from C. V. Mangan (NMPC) to Document Control Desk (NRC), dated May 13, 1987 (Response to Generic Letter 87-05)

11. Letter from C. D. Terry (NMPC) to Document Control Desk (NRC), dated June 26, 1990 (Regarding Generic Letter 87-05)

12. EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," August 1994

13. EPRI TR-1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 1, December 2005

14. Letter from R. B. Abbott (NMPC) to Document Control Desk (NRC), dated October 28, 1999, Use of 1998 Edition of ASME Code Section XI for Containment Inspections

15. Letter from M. Gamberoni (NRC) to J. H. Mueller (NMPC), dated August 17, 2000, Nine Mile Point Nuclear Station, Unit Nos. 1 and 2 - Relief from the Requirements of 10 CFR 50.55a Related to Containment Inspection (TAC Nos. MA7116, MA7117, and MA7118)

16. J. Haugh, J. Gisclon, W. Parkinson, and K. Canavan, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Intervals," Rev. 3.1, EPRI, October 2001

17. Letter from T. J. O'Connor (NMPNS) to Document Control Desk (NRC), 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)

18. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Assessment Results for Risk-Informed Activities," Revision 1," January 2007

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGE (MARK-UP)

The current version of Technical Specification Page 355 has been marked-up by hand to reflect the proposed change.

6.5.7 10 CFR 50 Appendix J Testing Program Plan

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance-Based Containment Leak-Test Program," dated September 1995 with the following exceptions:

1. Type A tests will be conducted in accordance with ANSI/ANS 56.8-1994 and/or Bechtel Topical Report BN-TOP-1, and

Insert A

2. The first Type A test following approval of this Specification will be a full pressure test conducted approximately 70, rather than 48, months since the last low pressure Type A test.

b. The peak calculated containment internal pressure (Pac) for the design basis loss of coolant accident is 35 psig.

c. The maximum allowable primary containment leakage rate (La) at Pac shall be 1.5% of primary containment air weight per day.

d. Leakage Rate Surveillance Test acceptance criteria are:

1. The as-found Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than 1.0 L_a.

2. The as-left Primary Containment Integrated Leak Rate Test (Type A Test) acceptance criteria is less than or equal to 0.75 L_a, prior to entering a mode of operation where containment integrity is required.

3. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.

4. The combined Local Leak Rate Test (Type B & C Tests including airlocks) acceptance criteria is less than 0.6 L_a, calculated on a minimum pathway basis, at all times when containment integrity is required.

e. The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the 10 CFR 50 Appendix J Testing Program Plan.

The provisions of Specification 4.0.3 are applicable to the 10 CFR 50 Appendix J Testing Program Plan.

INSERT A (for TS Page 355)

3. Exception to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Section 9.2.3: The first Type A test performed after the June 8, 1999 Type A test shall be performed no later than June 8, 2014.

ATTACHMENT 2

RISK IMPACT ASSESSMENT OF EXTENDING CONTAINMENT TYPE "A" TEST INTERVAL FOR NINE MILE POINT NUCLEAR STATION UNIT 1 (NER-1M-013)

Nine Mile Point Nuclear Station, LLC
June 16, 2008

Constellation Energy Group

Nine Mile Point
Nuclear Station

NUCLEAR ENGINEERING REPORT
NINE MILE POINT UNIT 1

Risk Impact Assessment of Extending Containment Type "A" Test Interval for Nine Mile Point Nuclear Station Unit 1

NER-1M-013, Rev. 0

Prepared By:	Nicholas Lovelace *PRA Engineer (see note)*	**Date:** 6/9/08
Reviewed By:	Rick Anoba *PRA Engineer (see note)*	**Date:** 6/9/08
Approved By:	Theodore Kulczycki *NMP PRA Engineering General Supervisor*	**Date:** 6/9/08

John Raby [per telecon] 6/9/08
TGK

PURPOSE: The purpose of this Engineering Report is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) frequency from ten years to fifteen years for a one time extension for the Nine Mile Point Nuclear Station (NMPS) Unit 1. The proposed change would impact testing associated with the current surveillance test for Type A leakage. The risk assessment follows the guidelines from NEI 94-01, the methodology used in EPRI TR-104285, and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change a plant's licensing basis as outlined in Regulatory Guide 1.174. In addition, for comparison purposes, the risk assessment was also performed using two more recent studies. These methodologies are presented in the NEI Interim Guidance, and in EPRI TR-1009325 Revision 1. Although these methodologies generally produce more conservative results than do the earlier methodologies, they build upon the work of the earlier studies, and much of the analyses developed from application of the EPRI TR-104285 methodology remains applicable for use in these more recent studies.

NOTE: Document preparation and review performed under PO #7703384 and internally Owner Accepted by Constellation Energy Nine Mile Point Nuclear Station PRA Engineering.

Revisions:

Number	Description / Prepared By / Date	Reviewed By	Date	Approved By	Date

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EXECUTIVE SUMMARY

An evaluation was performed to assess the risk impact of extending the currently allowed containment Type A integrated leak rate test (ILRT) frequency from ten years to fifteen years for a one time extension for the Nine Mile Point Nuclear Station Unit 1 (NMPS). For the purpose of this report, all references to NMPS are for Unit 1 only. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for NMPS. The proposed change would impact testing associated with the current surveillance test for Type A leakage. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change a plant's licensing basis as outlined in Regulatory Guide 1.174 [3]. In addition, for comparison purposes, the risk assessment was also performed using two more recent studies. These methodologies are presented in the NEI Interim Guidance [23], and in EPRI TR-1009325 [17]. Although these methodologies generally produce more conservative results than do the earlier methodologies, they build upon the work of the earlier studies, and much of the analyses developed from application of the EPRI TR-104285 methodology remains applicable for use in these more recent methodologies.

The findings for the NMPS assessment confirm the general findings of previous studies on a plant specific basis, including severe accident category frequencies, the containment failure modes, the Technical Specification allowed leakage, and the local population surrounding the NMPS station. The following conclusions are provided with regard to the risk impact of extending the Type A ILRT test from ten years to fifteen years:

- There is no change in the at-power CDF associated with the ILRT test interval extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

- Reg. Guilde 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from once-per-ten-years to once-per-fifteen years is between 3.43E-10/yr and 3.81E-9/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1E-7/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the NMPS risk profile.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the total integrated plant risk (total dose to the public) by between 0.0056% and 0.048%. Therefore, the risk impact change when compared to other severe accident risks is negligible.

- This change in Conditional Containment Failure Probability (CCFP) of less than 1% is judged to insignificant and reflects sufficient defense-in-depth.

- Incorporating external event impacts into this analysis does not change the conclusion of this risk assessment (i.e., increasing the NMPS ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

Risk Impact Assessment of Extending Containment Type A Test Interval for Nine Mile Point Nuclear Power Station

1. Introduction

The purpose of this analysis is to provide a risk assessment of extending the currently allowed containment Type A integrated leak rate test (ILRT) frequency from ten years to fifteen years for a one time extension for the Nine Mile Point Nuclear Station (NMPS). The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for NMPS. The proposed change would impact testing associated with the current surveillance test for Type A leakage. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request to change a plant's licensing basis as outlined in Regulatory Guide 1.174 [3]. In addition, for comparison purposes, the risk assessment was also performed using two more recent studies. These methodologies are presented in the NEI Interim Guidance [23], and in EPRI TR-1009325 [17]. Although these methodologies generally produce more conservative results than do the earlier methodologies, they build upon the work of the earlier studies, and much of the analyses developed from application of the EPRI TR-104285 methodology remains applicable for use in these more recent studies. Therefore, the calculations and results from these analyses are presented in Sections 9 and 10, with previous EPRI TR-104285 results referenced as necessary for efficient reporting of the study results.

2. Limitations

This PRA Engineering Study evaluates the risk significance of extending ILRT test frequency based on NMPS Unit 1 Quantification Notebook [22] and the 2007 update of the Level II Analysis [24]. External events were addressed as a sensitivity study but not included in the final results. The NUREG/CR-4551 doses were adjusted for the population surrounding the NMPS and the NMPS ILRT leakage, volume, and power level criteria.

3. Background

10CFR50, Appendix J, Option B, allows individual plants to extend the Type A Integrated Leak Rate Test (ILRT) surveillance test interval from three-in-ten years to at least once per ten years. The revised Type A test frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated leakage performance was less than 1.0La. NMPS meets these requirements.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based

Containment Leak Test Program," September 1995, provides the technical basis to support rule making to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rule making basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285 [2]. This analysis has also taken the draft form of NEI 94-01 Rev 1J [13] into consideration and complies with updates that are proposed therein. The major update to NEI 94-01 was the change in methodology to EPRI 1009325 Rev 1 which was utilized in section 10.

NUREG-1493 [4] analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing and determined that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day results in a small increase in total population exposure. In addition, increasing the leak rate to 50 percent per day increases the total population risk by less than 1 percent. Consequently, extending the ILRT interval should not result in a substantial increase in risk. The current analysis is being performed to confirm these conclusions based on NMPS specific models and available data.

EPRI TR-104285 (Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals) is a follow-on report to NUREG-1493 that provides a methodology for use in preparing PRA analysis to support a submittal to extend ILRT test intervals. This methodology is followed to determine the appropriate risk information for use in evaluating the impact of the proposed ILRT changes.

It should be noted that containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct a general visual inspection of the accessible areas of the interior of the containment in accordance with Subsection IWE once each period. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B and Type C local leak tests performed to verify the leak-tight integrity of containment penetration valves, air locks, seals, and gaskets are also not affected by the change to the Type A test frequency.

4. Criteria

The acceptance guidelines in RG 1.174 are used to access the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the

relevant criterion is the change in LERF. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability, which helps to ensure that the defense-in-depth philosophy is maintained, will also be calculated. In addition, the total risk (person rem/yr population dose) is examined to demonstrate the relative change in this parameter.

5. Methodology

A simplified bounding analysis approach consistent with the EPRI approach is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The approach is consistent with that presented in EPRI TR-104285 [2] and NUREG-1493 [4]. The analysis uses the current NMPS PRA model that includes the results from the NMPS Level II analysis of core damage scenarios and subsequent containment response resulting in various fission product release categories (including no release).

The four general steps of this risk assessment are as follows:

1) Quantify the baseline risk in terms of frequency events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report.

2) Develop plant-specific person-rem (population dose) per reactor year for each of the eight containment release scenario types from plant specific consequence analyses.

3) Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.

4) Determine the change in risk in terms of LERF in accordance with Regulatory Guide (RG) 1.174 [3] and compare with the acceptance guidelines of RG 1.174.

This approach is based on the information and methodology contained in the previously mentioned studies and is consistent with the following:

- Other industry risk assessments for ILRT test interval extensions. The NMPS assessment uses population dose as one of the risk measures. The other risk measures used in the NMPS assessment are LERF, and Conditional Containment Failure Probability (CCFP) to demonstrate that the acceptance guidelines from RG 1.174 are met.

- EPRI TR-104285 and NUREG-1493. The NMPS assessment uses information from NUREG-1273 [6] regarding the low percentage of containment leakage events that would only be detected by an ILRT as input to calculate the increase in the pre-existing containment leakage probability due to the testing interval extension.

- The approach used in the Indian Point 3 risk-informed submittal for a one-time extension of the Type A test interval. The NMPS evaluation uses similar ground rules and methods to calculate changes in risk metrics [14]. NRC granted approval to Indian Point 3 by License Amendment No. 206 dated April 17, 2001 (TAC No. MB0178) [21].

6. Inputs

This section summarizes the general resources available as input (Section 6.1) and the plant specific resources required (Section 6.2).

6.1 General Resources Available

Various industry studies on containment leakage risk assessment are briefly summarized here:

1) NUREG/CR-3539 [7]
2) NUREG/CR-4220 [8]
3) NUREG-1273 [6]
4) NUREG/CR-4330 [9]
5) EPRI TR-105189 [10]
6) NUREG-1493 [4]
7) EPRI TR-104285 [2]

1. NUREG/CR-3539 [7]

The study is applicable because it provides one basis for the threshold that could be used in the Level II PRA for the size of containment leakage that is considered significant and to be included in the model. Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on Light Water Reactor (LWR) accident risks is relatively small.

2. NUREG/CR-4220 [8]

This study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories for the NRC in 1985. The study

reviewed over two thousand License Event Reports (LERs), ILRT reports, and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages. It is the latter category that is applicable to containment isolation modeling and that is the focus of this risk assessment.

NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event. It should be noted that all of the 4 identified large leakage events were PWR events.

3. NUREG-1273 [6]

The study is applicable because it is a subsequent study to NUREG/CR-4220 which undertook a more extensive evaluation of the same database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradation" of the containment isolation system.

4. NUREG/CR4330 [9]

This study provides an assessment of the impact of different containment leakage rates on plant risk. NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "The effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

5. EPRI TR-105189 [10]

This study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) of the impact of extending ILRT and LLRT test intervals on shutdown risk for two reference plants (a BWR-4 and a PWR).

The result of the study concluded that a small but measurable safety benefit is realized from extending the test intervals. Extending the ILRT frequency from 3 per 10 years resulted in a reduction in shutdown CDF of approximately 1E-7/yr. This risk reduction is due to the following issues:

- Reduced potential of vessel drain down events
- Reduced time in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS-draindown events caused by ILRT/LLRT activities, and the other 5 were events involving loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the improvement in safety due to extending the ILRT test interval.

6. NUREG-1493 [4]

The study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2-1.0%) population risk.

NUREG-1493 used information from NUREG-1273 regarding the low percentage of containment leakage events that would only be detected by an ILRT in the calculation of the increase in the pre-existing containment leakage probability due to the testing interval extension. NUREG-1493 makes the following assumptions in this probability calculation:

- The average time that a pre-existing leakage may go undetected increases with the length of the testing interval.
- Only 3% of all pre-existing leaks can be detected only by an ILRT (i.e., and not by LLRTs).

This same approach that was used in a previously approved ILRT test interval extension submittal [14, 21] is also proposed here for the NMPS ILRT test interval extension risk assessment.

7. EPRI TR-104285 [2]

The study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending ILRT and LLRT test intervals on at-power public risk. This study combined IPE Level II models with NUREG/CR-4551 Level III population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage frequencies into eight (8) classes of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures dependent upon the core damage accident
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Containment isolation failures not identified by LLRT (e.g., isolation failures due to testing or maintenance)
7. Containment failure due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> "These study results show that the proposed CLRT [containment leak rate tests] frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year."

6.2 Plant Specific Inputs

The information used to perform the NMPS ILRT Extension Risk Assessment includes the following:

- Population Dose Calculations by release category.
- NMPS PRA Model
- ILRT results to demonstrate adequacy of the administrative and hardware issues. The two most recent Type A ILRT tests for NMPS were successful, so the current Type A test interval is 10 years.

<u>Population Dose Calculations</u>

The population dose is calculated by using Peach Bottom data provided in NUREG/CR-4551 [5] and adjusting the results for NMPS using guidelines from EPRI 1009325 [17]. Each accident sequence was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551[5].

The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 [5] and are reproduced in Table 6-1 for reference purposes. Table 6-2 summarizes the calculated population dose associated with each APB from NUREG/CR-4551[5].

Population Estimate Methodology

The person-rem results in Table 6-2 can be used as an approximation of the dose for NMPS if it is corrected for the population surrounding NMPS and the difference in Technical Specifications leak rate, power level and volume. For the 2000 population estimate, data is available for population by county from the US Census Bureau on the website (http://quickfacts.census.gov/qfd/states/36000.html). This data was used to estimate the population within a 50-mile radius of the plant and matches reference [27]. If any part of a county fell within the 50-mile radius (based on a review of a map containing a mileage scale and state/county borders), then the entire population was included in the population estimate. The results of this population estimate are presented in Table 6-3.

The total population shown in Table 6-3 is compared to the total population that is provided in NUREG/CR-4551 in order to get a "Population Dose Factor" that was applied to the APBs to get dose estimates for NMPS.

Total NMPS Population = 5.81E+5 [Table 6-3]
Peach Bottom Population from NUREG/CR-4551 = 3.2E+6 [5]
Population Dose Factor = 5.81E+5 / 3.2E+6 = 0.181

This population dose factor was applied to the APBs from NUREG/CR-4551. Additionally, correction factors for power level, volume and leakage allowable were required to be applied to the NUREG/CR-4551 calculation to account for differences in those values for Accident Progression Bins 1,2,3,4,5,6,7,9 and 10. For Accident Progression Bin 8 the Leakage allowable factor is not included per EPRI 1009325 guidelines. The Technical Specification containment available leak rate for NMPS is 01.5% (L_a^{NMPS}) versus the 0.5% (L_a^{PB}) for the NUREG/CR-4551 plant, Peach Bottom. Therefore, the leakage (L_a^{PB}) person-rem calculated for Peach Bottom that is scaled by population for the NMPS analysis must be multiplied by a factor of 3.00 (L_a^{NMPS} / L_a^{PB}) to account for the differences in Technical Specification leakage rates. Similarly power levels and containment volumes were collected and population dose factors were generated and applied. Table 6-4 shows the results of applying the population dose factor, the allowable leakage factor, the volume factor, and the power level factor to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for NMPS. Since NMPS has a lower power rating than Peach Bottom, it is reasonable to assume that the dose release for the NMPS containment would be lower.

NMPS PRA Model

The version of the PRA model used for the NMPS ILRT Extension Risk Assessment is comprised of a Level I PRA Model and Level II PRA model developed in 2007. This version of the PRA model addresses accidents initiated by internal events at full power, and containment responses to these accidents. The NMP model was updated to meet RG 1.200 and ASME Category II requirements. It has recently been peer reviewed and F&Os submitted. A review of the F&Os was performed for applicability to the ILRT extension. This review determined that

there should be no significant change in CDF, LERF, or recategorization of the radionuclide release end state frequencies that would impact the conclusions of this report.

For the NMPS ILRT Extension Risk Assessment, the controlling risk parameters are the frequency of the intact containment release category relative to the total core damage frequency, and the magnitude of the intact containment population dose relative to the total dose (i.e., the higher the intact frequency relative to the total core damage frequency and/or the intact population dose relative to the total dose, the larger the impact on ILRT risk). Any increases in initiating event frequencies, basic event probabilities, or human failure event probabilities would have a tendency to reduce the relative frequency of the intact containment release category relative to the total core damage frequency. Any reductions in initiating event frequencies, basic event probabilities, or human failure event probabilities would have a tendency to increase the relative frequency of the intact containment release category relative to the total core damage frequency.

As shown in Table 6-6, the 2007 update of the NMPS Level II PRA [24] was used to quantify the contributions for radiological release categories based on containment failure mode. The Level II PRA model was also used to quantify frequencies for the radiological release categories for ILRT Extension. The core damage frequency (CDF) is 3.3E-6 per-yr and the Large Early Release Frequency (LERF) is 3.0E-7 per-yr [22].

EPRI Release Category Definition

Table 6-7 defines the accident classes used in the ILRT extension evaluation consistent with the EPRI methodology [2].

Table 6-9 defines the mapping of NMPS Containment Failure Modes to EPRI release classes. Page 4.4-8 of the LE Notebook [24] defines the containment isolation node of the CET. Conservatively all of the containment isolation failures (IIV, IIID, IIA, IIL, and IV) were grouped into EPRI Class II. If the containment phenomena contribution was removed from EPRI Class II and binned into EPRI Class VII, sensitivity analyses show it would only decrease the risk for extending the ILRT. EPRI Class I is defined as the containment remains intact which includes accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values La, under Appendix J for that plant. EPRI Class VII involves containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents. Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in EPRI Class VIII. Changes in Appendix J testing requirements do not impact these accidents.

Table 6-1 COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTION [5]

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.

Collapsed APB Number	Description
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

Table 6-2 CALCULATION OF PEACH BOTTOM (NUREG/CR-4551) POPULATION DOSE

Collapsed Accident Progression (APB) Number	Fractional APB Contributions to Risk (MFCR) (1)	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 7.9 person-rem/yr, mean) (2)	NUREG/CR-4551 Collapsed Bin Frequencies (per year) (3)	NUREG/CR-4551 Population Dose at 50 miles (Person-rem) (4)
1	0.021	0.1659	9.55E-8	1.74E+6
2	0.0066	0.05214	4.77E-8	1.09E+6
3	0.556	4.3924	1.48E-6	2.97E+6
4	0.226	1.7854	7.94E-7	2.25E+6
5	0.0022	0.01738	1.30E-8	1.34E+6
6	0.059	0.4661	2.04E-7	2.28E+6
7	0.118	0.9322	4.77E-7	1.95E+6
8	0.0005	0.00395	7.99E-7	4.94E+3
9	0.01	0.079	3.86E-7	2.05E+5
10	0	0	4.34E-8	0
Totals	1.0	7.9	4.34E-6	

(1) Mean Fractional Contribution to Risk from Table 5.2-3 of NUREG/CR-4551

(2) The total population dose risk at 50 miles from internal events in person-rem is provided in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

(3) NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-6. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

(4) Obtained from dividing the population dose risk shown in the third column of this table by the collapsed bin frequency shown in the fourth column of this table.

Table 6-3 POPULATION WITHIN 50 MILES OF NMPS

State	County	Population (2000 US Census Bureau) [27]
New York		
	Oswego County	122,377
	Onondaga County	458,336
	Total	**580,713**

Table 6-4 CALCULATION OF NMPS POPULATION DOSE AT 50 MILES

Peach Bottom Accident Progression Bin #	NUREG/CR-4551 Population Dose at 50 miles (Person-rem)	Bin Multiplier used to obtain NMPS Population Dose	NMPS Adjusted Population Dose at 50 miles (Person-rem)
1	1.74E+6	.102	1.77E+05
2	1.09E+6	.102	1.11E+05
3	2.97E+6	.102	3.03E+05
4	2.25E+6	.102	2.29E+05
5	1.34E+6	.102	1.36E+05
6	2.28E+6	.102	2.33E+05
7	1.95E+6	.102	1.99E+05
8	4.94E+3	.315	1.56E+03
9	2.05E+5	.102	2.09E+04
10	0	0	0.00E+00

Table 6-5 CORE DAMAGE FREQUENCIES

SUMMARY OF THE CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE SUBCLASS FOR NMP1

Accident Class Designator	Subclass	Definition	CAFTA Model (per Rx Yr)
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	1.44E-6
	B	Accident sequences involving a station blackout and loss of coolant inventory makeup.	1.03E-6 (IBE: 5.50E-7) (IBL: 4.83E-7)
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	8.59E-8
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.	1.34E-8
Class II	A/T	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage; core damage induced post containment failure.	5.22E-7
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage induced post containment failure.	2.25E-10
	V	Class IIA and III except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	5.42E-9
Class III (LOCA)	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	3.54E-8
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	3.13E-8
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	3.32E-10
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	2.09E-8
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g. LOCA or SORV); core damage induced post containment failure.	3.01E-8
Class V	---	Unisolated LOCA outside containment.	8.88E-8
		Total CDF	3.30E-6

Table 6-6 SUMMARY OF NMPS RELEASE FREQUENCY BY CONTAINMENT FAILURE MODE

Summary of Radionuclide Release End State Frequencies

Class	Base CDF	All Releases	Non H/E	Intact	H/E	H/I	L/E	L/I	LL/E	LL/I	LL/L	M/E	M/I	M/L
IA	1.44E-06	7.17E-07	6.77E-07	7.19E-07	3.96E-08	4.29E-07	9.26E-09	1.93E-09	1.45E-09	6.17E-08	0.00E+00	6.18E-10	1.74E-07	0.00E+00
IBE	5.50E-07	5.50E-07	4.43E-07	0.00E+00	1.07E-07	4.08E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.19E-09	0.00E+00	2.95E-08	0.00E+00
IBL	4.83E-07	4.83E-07	4.61E-07	0.00E+00	2.22E-08	3.61E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.39E-08	0.00E+00	7.53E-08	0.00E+00
IC	3.59E-08	3.92E-09	3.53E-09	8.19E-08	3.94E-10	1.10E-12	4.74E-10	0.00E+00	0.00E+00	1.79E-09	0.00E+00	0.00E+00	1.28E-09	0.00E+00
ID	1.34E-08	1.43E-09	1.33E-09	1.20E-08	1.02E-10	3.14E-10	3.58E-11	0.00E+00	1.27E-10	4.01E-10	0.00E+00	0.00E+00	4.47E-10	0.00E+00
IIA/IIT	5.22E-07	5.22E-07	5.22E-07	0.00E+00	0.00E+00	2.88E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.33E-07	0.00E+00
IIL	2.25E-10	2.25E-10	0.00E+00	0.00E+00	2.25E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IIV	5.42E-09	4.81E-09	0.00E+00	6.15E-10	4.81E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IIIB	3.54E-08	2.17E-08	2.16E-08	1.36E-08	1.20E-10	2.19E-09	0.00E+00	0.00E+00	2.05E-10	1.92E-08	0.00E+00	0.00E+00	6.05E-12	0.00E+00
IIIC	3.13E-08	2.82E-08	2.78E-08	3.07E-09	3.61E-10	2.61E-09	0.00E+00	3.93E-10	0.00E+00	2.05E-08	0.00E+00	0.00E+00	2.61E-10	0.00E+00
IIID	3.32E-10	3.32E-10	0.00E+00	0.00E+00	3.32E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IVA/L	5.11E-08	5.11E-08	2.41E-08	0.00E+00	2.69E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.41E-08	0.00E+00	0.00E+00
V	8.88E-08	8.88E-08	0.00E+00	0.00E+00	8.88E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Totals	3.30E-06	2.47E-06	2.18E-06	8.30E-07	2.91E-07[5]	1.49E-06	9.77E-09	2.32E-09	1.78E-09	1.04E-07	3.11E-08	2.48E-08	5.14E-07	0.00E+00
%CDF	100.00%	74.85%	66.03%	25.15%	8.82%	45.23%	0.30%	0.07%	0.05%	3.14%	0.94%	0.75%	15.55%	0.00%

(1) Due to truncation, the Class IBE end state frequency total was initially calculated slightly below the accident total of 5.50E-7/yr at 5.45E-7/yr. The individual Class IBE end state totals were increased proportionally by a factor of 1.01 (i.e. 5.45E-7/5.50E-7) to equal the total class IBE Level 1 CDF of 5.50E-7/yr.

(2) Due to truncation, the Class IBL end state frequency total was initially calculated slightly below the accident total of 4.83E-7/yr at 4.69E-7/yr. The individual Class IBL end state totals were increased proportionally by a factor of 1.03 (i.e. 4.69E-7/4.83E-7) to equal the total class IBL Level 1 CDF of 4.83E-7/yr.

(3) Due to minimization problems, the Class IIA/IIT end state frequency total was initially calculated slightly above the accident total of 5.22E-7/yr at 5.50E-7/yr. The individual Class IIA/IIT end state totals were decreased proportionally by a factor of 0.95 (i.e. 5.22E-7/5.50E-7) to equal the total class IIA/IIT Level 1 CDF of 5.22E-7/yr.

(4) Due to truncation, the Class IVA/IVL end state frequency total was initially calculated slightly below the accident total of 5.11E-8/yr at 4.99E-8/yr. The individual Class IVA/IVL end state totals were increased proportionally by a factor of 1.02 (i.e. 4.99E-8/5.11E-8) to equal the total class IVA/IVL Level 1 CDF of 4.83E-7/yr.

(5) The accident class LERF total of 2.91E-7/yr is slightly higher than the base level 1 LERF total of 2.83E-7/yr from the single top model. Quantifying the LERF on an accident class level may result in the generation of non-minimal cutsets between accident classes. Therefore, the reported LERF may appear to be slightly higher.

Table 6-7 EPRI CONTAINMENT FAILURE CLASSIFICATIONS

Class	Description
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values La, under Appendix J for that plant.
2	Containment isolation failures include accidents for which there is a failure to Isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-Existing isolation failure to seal is not dependent on the sequence in progress. This Class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and Maintenance requirements or verified per in service inspection and testing (ISI/IST).
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

Table 6-8 MAPPING OF PEACH BOTTOM ACCIDENT PROGRESSION BINS TO EPRI ACCIDENT CLASSES

EPRI Accident Classes	Description	Peach Bottom Collapsed Accident Progression Bin (APB)
1	No Containment Failure	8
2	Large Isolation Failures (Fail to Close)	3
3A	Small Isolation Failures (Liner Breach)	
3B	Large Isolation Failures (Liner Breach)	
4	Small Isolation Failures (Fail to Seal - Type B)	
5	Small Isolation Failures (Fail to Seal - Type C)	
6	Other Isolation Failures (e.g., dependent failures)	
7	Failures induced by Phenomena (early and late)	3,4,5
8	Bypass (Interfacing Systems LOCA)	3

Table 6-9 MAPPING OF NMPS CONTAINMENT FAILURE MODES TO EPRI RELEASE CLASSES

L1 Acc. Class/ L2 Release Category	Description	EPRI Release Category
Intact	Containment remains intact	1
Class IIV	Class IIA and III except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	2
Class IIID	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	2
Class IIA	Accident sequences involving a loss of containment heat removal with RPV initially intact; core damage;core damage induced post containment failure	2
Class IIL	Accident sequences involving a loss of containment heat removal with RPV breached but no initial core damage; core damage induced post containment failure.	2
Class IV	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	2
(other L2)	Other Containment Failure Release Categories	7
Class V (ISLOCA)	LERF bypass scenarios that result from early SGTR scenarios	8

Note: for EPRI class 2 bins only high releases were inputted see table 6-6 above [24].

6.3 Conditional Probability of ILRT Failure (Small and Large)

The ILRT can detect a number of failures such as liner breach, failure of certain bellows arrangements, and failure of some sealing surfaces. The proposed ILRT test interval extension may influence the conditional probability associated with the ILRT failure. To ensure that this effect is properly accounted for, the Class 3 Accident Class is divided into two sub-classes, Class 3a and Class 3b, representing small and large leakage failures, respectively.

Data presented in NUREG-1493 [4] was used to calculate the probability that a liner leak will be large (Event Class 3b). The data found in NUREG-1493 states that 144 ILRTs were conducted. The largest reported leak rate from those 144 tests was 21 times the allowable leakage rate (L_a). Because $21L_a$ does not constitute a large release, no releases have occurred based on the 144 ILRTs reported in NUREG-1493 [4].

To estimate the failure probability given that no failures have occurred, a conservative estimate is obtained from the 95th percentile of the χ^2 distribution. In statistical theory, the χ^2 distribution can be used for statistical testing, goodness-of-fit tests, and evaluating s-confidence [25]. The χ^2 distribution is really a family of distributions, which range in shape from exponential to normal. Each distribution is identified by the degrees of freedom, v. For time-truncated tests (versus

failure-truncated tests), an estimate of the probability of a large leak using the χ^2 distribution can be calculated as χ^2_{95} (v = 2n + 2) / 2N, where n represents the number of large leaks and N represents the number of ILRTs performed to date. With no large leaks (n = 0) in 144 events (N = 144) and χ^2_{95} (2) = 5.99, the 95th percentile estimate of the probability of a large leak is calculated as 5.99 / (2 * 144) = 0.021.

To calculate the probability that a liner leak will be small (event Class 3a), use was made of the data presented in NUREG-1493 [4]. The data found in NUREG-1493 states that 144 ILRTs were conducted. The data reported that 23 of 144 tests had allowable leak rates in excess of $1.0L_a$. However, of these "failures" only 4 were found by an ILRT; the others were found by Type B and C testing on errors in test alignments. Therefore, out of the 144 ILRTs, 4 failures were categorized as "small releases". Similar to the event Class 3b probability, the estimated failure probability for small release is found by using the χ^2 distribution. The χ^2 distribution is calculated by n = 4 (number of small leaks) and N = 144 (number of events) which yields a. χ^2_{95} (10) = 18.3070. Therefore, the 95th percentile estimate of the probability of a small leak is calculated as 18.3070 / (2 * 144) = 0.064.

Using the methodology discussed above is conservative compared to the typical mean estimates used for PRA analysis. For example, the mean probability of a Class 3a failure would be the (number of failures) / (number of tests) or 4/144 = 0.03 compared with 0.064 used here.

6.4 Impact of Extension on Leak Detection Probability

The NRC in NUREG-1493 [4] has determined from a review of operating experience data that only 3% of the ILRT failures were found which local leakage-rate testing could not and did not detect. In NUREG-1493 [4], it is noted that based on a review of leak rate testing experience, a small percentage (3%) of leakages that exceed current requirements are detectable only by Type A testing (ILRT). Further, in NUREG-1493 it is noted that the leakage rates observed in these few Type A test failures were only marginally above currently prescribed limits and could be characterized by a leakage rate of about two times the allowable.

Also in NUREG-1493 [4], it was assumed that the characteristic magnitude of leakages detectable only by ILRTs would not change, but the probability of leakage would change due to the longer intervals between tests. The change in probability was estimated by comparing the average time that a leak could exist without detection. For example, the average time that a leak could go undetected with a three-year test interval is 1.5 years (3yrs/2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10yrs/2). This change would lead to a non-detection probability that is a factor of 3.33 (5.0/1.5) higher for the probability of a leak that is detectable only by ILRT testing. However, since ILRTs have been demonstrated to improve the residual leak detection by only 3%, the interval change noted above would only lead to about a 10% (3.33 x 3%) non-detection leak probability. It is assumed that LLRT will continue to provide leak detection for 97% of leakages. Correspondingly, an extension of the ILRT interval to fifteen years is estimated to result in approximately a 15% (7.5/1.5x3%)

non-detection probability of a leak. These are approximations assumed by the NRC and EPRI because the current 3 ILRTs in 10 years would have a T/2 = 1.67 years instead of 1.5 years.

Therefore, the failure rate of ILRTs for which the LLRTs do not provide adequate backup is 0.03/1.5 year average detection time. Applying a constant failure rate model, the failure probability of ILRTs, P_f, can be estimated as follows:

For 3 Year Interval

$$P_f = \frac{1}{2}\lambda T = \left(\frac{.03}{1.5yr}\right)\left(\frac{3yr}{2}\right) = 0.03$$

For 10 Year Interval

$$P_f = \frac{1}{2}\lambda T = \left(\frac{.03}{1.5yr}\right)\left(\frac{10yr}{2}\right) = 0.10$$

For 15 Year Interval

$$P_f = \frac{1}{2}\lambda T = \left(\frac{.03}{1.5yr}\right)\left(\frac{15yr}{2}\right) = 0.15$$

EPRI has previously interpreted this to mean that the failure to detect probability values is as follows:

ILRT FAILURE TO DETECT PROBABILITY

ILRT Interval	EPRI Assessment [2]	IP3 [14]	Constant Failure Rate Model
3 yr	0.03	0.03	0.03
10 yr	0.13	0.13	0.10
15 yr	NA	0.18	0.15

In addition, IP3 [14] has used this same estimate of changes in detection probability in a submittal to extend the ILRT interval on a one-time basis. The IP3 request for a one-time ILRT extension was approved by the NRC on April 17, 2000 (TAC No. MB0178) [21].

The analysis included in this report follows the precedence set by the EPRI report and the IP3 analysis. The use of the constant failure rate model is conservatively represented by the assumed "failure to detect" probabilities used by EPRI and in the IP3 submittal.

7. Assumptions

The following assumptions and ground rules were used in the engineering evaluation:

- The NMPS Level I and Level II internal events PRA model provides representative results for the analysis. The NMPS Level I does not include fire, floods, and shutdown events. However, a bounding sensitivity analysis is performed to assess the impact of external events.
- It is appropriate to use the NMPS internal events PRA model as a gauge to effectively describe the risk changes attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if fire, floods and shutdown events were to be included in the calculations.
- An evaluation of the risk trade-off impact of performing the ILRT during shutdown is addressed using the generic results from EPRI TR 105189 [10].
- Dose results for the containment failures modeled in the PRA can be characterized by the NUREG/CR-4551 population dose results [5] with corrections for NMPS-specific population density and ILRT leakage, volume, and power level criteria.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [2] and are summarized on Table 6-7.
- The maximum containment leakage for Class 1 sequences is 1.0 L_a. Class 3 accounts for increased leakage due to Type A inspection failures.
- The maximum containment leakage for Class 3a sequences is 10 L_a, based on the previously approved methodology [14, 21, 23].
- The maximum containment leakage for Class 3b sequences is 35 L_a, based on the previously approved methodology [14, 21, 23].

8. Application of EPRI TR-104285 Methodology

The application of the EPRI TR-104285 approach is based on EPRI-TR-105189 [10] and previous risk assessment submittals on this subject [14]. The approach has established a clear process for the calculation and presentation of results.

The method chosen to display the results is according to the eight (8) accident classes consistent with these two reports. Table 8-1 lists these accident classes.

The analysis performed examined NMPS specific accident sequences in which the containment either remains intact or is impaired. Specifically, the break down of the severe accident contribution to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI TR-104285 Class 1 sequences).

- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage. (EPRI TR-104285 Class 3 sequences).

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test - EPRI TR-104285 Class 6 sequences).

- Accident sequences involving containment bypass (EPRI TR-104285 Class 8 sequences), large containment isolation failures (EPRI TR-104285 Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI TR-104285 Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the ILRT frequency change.

- Class 4 and 5 sequences are impacted by changes in Type B and C test intervals; therefore, changes in the Type A test interval do not impact these sequences.

The steps taken to perform this risk assessment evaluation are as follows:

Step 1 - Quantify the baseline risk in terms of frequency per reactor year for each of the applicable eight accident classes presented in Table 8-1.

Step 2 - Develop plant specific person-rem dose (population dose) per reactor year for each of the eight accident classes evaluated in EPRI TR-104285.

Step 3 - Evaluate the risk impact of extending Type A test interval from 10 to 15 years.

Step 4 - Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174.

Step 5 - Determine Impact on the Conditional Containment Failure Probability (CCFP).

Step 1 - Quantify the Base-line Risk in Terms of Frequency per Reactor Year

As discussed in Section 6.2 and as shown on Table 6-4, the population doses for the Peach Bottom Plant in NUREG/CR-4551 [5] were used to develop the population doses for NMPS. The population doses for the ten Peach Bottom Accident Progression Bins were adjusted for population differences and ILRT leakage, power level and volume differences to obtain the equivalent NMPS doses. As shown on Table 6-8, the ten Peach Bottom Accident Sequence Bins were mapped to the EPRI accident classes.

As discussed in Section 6.2, the 2007 update of the NMPS Level II analysis was used to quantify frequencies for the radiological release categories based on NMPS containment failure modes. The frequencies were obtained from the Quantification Notebook [22]. As shown in Table 6-9, the NMPS containment failure modes were mapped to the EPRI accident classes.

The extension of the Type A test interval does not influence those accident progressions that involve large containment isolation failures, Type B or Type C testing, or containment failure induced by severe accident phenomena.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks are included in the model. Specifically, a simplified model based on NUREG-1493 results is used to predict the likelihood of having a small/large breach in the containment liner that is undetected by the Type A ILRT test. These events are represented by the "Class 3" sequence depicted in EPRI TR-104285 [2]. The Class 3 leakage includes the probability of a liner breach or bellows failure (due to excessive leakage) at the time of core damage. Two failure modes, event Class-3a (small breach) and event Class-3b (large breach) were considered to ensure proper representation of available data.

After including the respective "large" and "small" liner breach leak rate probabilities (Classes 3a and 3b), the eight severe accidents class frequencies were developed consistent with the definitions in Table 8-1 and described below.

Class 1 Sequences

This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is 8.30E-07/year based on the value in Table 6-6 and the mapping relationship in Table

6-9. After all accident class frequencies (Classes 2 through 8) were developed, frequencies for Classes 3A and 3B were summed. This was then subtracted from the total 8.30E-7 to obtain the Class 1 frequency of "No Containment Failure" of 7.59E-7/yr. For this analysis, the associated maximum containment leakage for this group is $1.0L_a$, consistent with an intact containment evaluation.

Class 2 Sequences.

This group consists of all core damage accident progression bins for which a failure to isolate the containment occurs. These sequences are dominated by failure-to-close of large containment isolation valves. The frequency per year for these sequences is 3.20E-7/year based on the values on Table 6-6 and the mapping relationship on Table 6-9. In order to calculate this number, high release values were taken from classes IIV, IID, IIA, IIL, and IV as designated in the 2007 PRA model [24].

Class 3 Sequences.

This group consists of all core damage accident progression bins for which a pre-existing leakage in the containment structure (e.g., containment liner) exists. The containment leakage for these sequences can be either small ($2.0L_a$ to $35L_a$) or large ($>35L_a$).

The respective frequencies per year are determined as follows:

$PROB_{Class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.064 [see Section 6.3]

$PROB_{Class\ 3b}$ = probability of large pre-existing containment liner leakage
= 0.021 [see Section 6.3]

CLASS_3a_FREQUENCY = 0.064 * 8.30E-7/year = 5.31E-8/year

CLASS_3b_FREQUENCY = 0.021 * 8.30E-7/year = 1.74E-8/year

For this analysis, the associated containment leakage for Class 3a is $10L_a$ and for Class 3b is $35L_a$. These assignments are consistent with the Indian Point 3 ILRT submittal [14] which was approved by the NRC [21]. For 10-yr and 15-yr test intervals, there is a likelihood that corrosion related containment leakage may not be detected. Therefore, the baseline frequency for Class 3b sequences is increased by a factor of 1.003007 to account for undetected corrosion related containment leakage. (Appendix A presents the basis and supporting calculations). Note that this factor is conservatively based on a test interval increase from 3 years to 15 years and is used for the 10-year and 15-year cases.

Class 4 Sequences.

This group consists of all core damage accident progression bins for which a containment isolation failure-to-seal of Type B test components occurs. Because

these failures are detected by Type B tests which are unaffected by the Type A ILRT, this group is not evaluated any further in the analysis.

Class 5 Sequences.

This group consists of all core damage accident progression bins for containment isolation failure-to-seal of Type C test components. Because the failures are detected by Type C tests which are unaffected by the Type A ILRT, this group is not evaluated any further in this analysis.

Class 6 Sequences.

This group is similar to Class 2. These are sequences that involve core damage accident progression bins for which a failure-to-seal containment leakage due to failure to isolate the containment occurs. However, as these failures are unaffected by changes in the Type A ILRT frequency, this group is not evaluated any further in this analysis.

Class 7 Sequences.

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., direct containment heating, melt-through, over-pressure). The baseline frequency per year for these sequences is 2.06E-6/year and is based on the values on Table 6-6 and the mapping relationship on Table 6-9. The mapped NMPS bin frequencies are summed to obtain the EPRI Class 7 frequency.

Class 8 Sequences.

This group consists of all core damage accident progression bins in which containment bypass occurs. The frequency per year for these sequences is 8.88E-8/year and is based on the values on Table 6-6 and the mapping relationship on Table 6-9.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to radionuclide release to the public have been derived consistent with the definition of Accident Classes defined in EPRI-TR-104285. Table 8-2 summarizes these accident frequencies by Accident Class.

Step 2 – Develop Plant-Specific Person-Rem Dose (Population Dose) Per Reactor Year

In Section 6.2, a release analysis was performed to estimate the person-rem doses to the population, within a 50-mile radius from NMPS. The releases are based on NUREG/CR-4551 [5] with adjustments for site-specific population and ILRT test leakage, power level and volume differences. The population doses for the EPRI classes were calculated using the population doses in Table 6-4 and the mapping relationships in Table 6-8.

Class 1 = 1.56E+3 person-rem (at $1.0L_a$)

Class 2 = 3.03E+5 person-rem

Class 3a = 1.56E+3 person-rem x $10L_a$ = 1.56E+4 person-rem

Class 3b = 1.56E+3 person-rem x $35L_a$ = 5.45E+4 person-rem

Class 4 = Not analyzed (Assigned a zero value)

Class 5 = Not analyzed (Assigned a zero value)

Class 6 = Not analyzed (Assigned a zero value)

Class 7 = 6.68E+5 person-rem

Class 8 = 3.03E+5 person-rem

The population dose estimates derived for use in the risk evaluation are summarized in Table 8-3.

The above results, when combined with the results presented in Table 8-2, yield the baseline mean consequence measures for each accident class. These results are presented in Table 8-4.

The total dose per year is compared with the other sites as shown below:

Plant	Annual Dose (Person-Rem/yr)	Reference
Indian Point 3	14.515	14
Peach Bottom	6.2	15
Crystal River	1.4	16
NMPS	1.5017	Table 8-4

Based on the risk values from Table 8-4, the percent risk contribution ($\%Risk_{BASE}$) for Class 3 (i.e., the Class affected by the ILRT interval change) is as follows:

$\%Risk_{BASE} = [(CLASS3a_{BASE} + CLASS\ 3b_{BASE})/\ Total_{BASE}] \times 100$

Where:
$CLASS3a_{BASE}$= Class 3a person-rem/year = 8.275E-4 person-rem/year [Table 8-4]
$CLASS3b_{BASE}$= Class 3b person-rem/year = 9.504E-4 person-rem/year [Table 8-4]
$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 1.5017 person-rem/yr [Table 8-4]

$\%Risk_{BASE} = [(8.275E\text{-}4 + 9.504E\text{-}4)/1.5017] \times 100$
$\%Risk_{BASE} = 0.118\%$

Step 3 – Evaluate Risk Impact of Extending Type A Test Interval from 10-to-15 Year

According to NUREG-1493 [4], relaxing the Type A ILRT interval from 3-in-10 years to 1-in-10-years will increase the average time that a leak detectable only by an ILRT goes undetected from 1.5 years to 5 years. The average time for failure to detect is calculated using the approximation 1/2 λT where T is the Test interval and λ, the leakage failure rate, is (3%)/1.5 year. If the test interval is extended to 1 in 15 years, the average time that a leak detectable only by an ILRT test goes undetected increases to 7.5 years (1/2 * 15 years.). Because ILRTs only detect about 3% of leaks (the rest are identified during LLRTs), the result for a 10-yr ILRT interval is a 10% undetectable rate in the overall probability of leakage 1/2 * (3% /1.5 years) * 10 years.

This value is determined by multiplying 3% and the ratio of the average time for non-detection for the increased ILRT test interval to the baseline average time for non-detection. For a 15-yr-test interval, the result is a 15% overall probability of leakage (i.e., 1/2 * (3% /1.5 yrs) * 15 years). Thus, increasing the ILRT test interval from 10 years to 15 years translates into a 5% increase in the overall leakage probability.

Risk Impact due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval, (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3 sequences is impacted. Therefore, for Class 3 sequences, the risk contribution is determined by multiplying the Class 3 accident frequency by the increase in probability of leakage of 1.1 (7% increase [10%-3%] which is approximated here as a factor of 1.1 consistent with the approach used by Indian Point 3 [14]). Specifically, there is a factor of 1.1 increase in the Class 3a and 3b frequencies relative to the baseline associated with increasing the ILRT test interval from 3 yrs to 10 yrs. (See Section 6.4)

Risk Impact of Corrosion Related Leakage due to Increase to 15-year Test Interval

Increasing the test interval from 3 to 15 years may reduce the chance of detecting corrosion related leakage. The likelihood of not detecting corrosion related leakage due to increased test interval from 3 to 15 years is calculated to be 0.3007%. Details of this calculation are provided in Appendix A. The calculation conservatively assumes that the total containment surface area can be exposed to corrosion. The increased likelihood of corrosion-related leakage is assumed to

increase LERF contributions from EPRI Class 3B by a factor of 1.003007. This factor is applied to both 10-year and 15-year test interval calculations.

The results of this calculation are presented in Table 8-5. Based on the Table 8-5 values, the Type A 10-year test frequency percent risk contribution ($\%Risk_{10}$) for Class 3 is computed as follows:

$$(\%Risk_{10}) = [(CLASS3a_{10} + CLASS3b_{10}) / Total_{10}] \times 100$$

Where:
$CLASS3a_{10}$ = Class 3a person-rem/year = 9.10E-4 person-rem/yr [Table 8-5]
$CLASS\ 3b_{10}$ = Class 3b person-rem/year = 1.05E-3 person-rem/yr [Table 8-5]
$TOTAL_{10}$ = Total person-rem/yr for 10-year interval = 1.5019 person-rem/yr [Table 8-5]

$\%Risk_{10} = [(9.10E\text{-}4 + 1.05E\text{-}3) / 1.5019] \times 100$
$\%\ Risk_{10} = 0.13\%$

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 0.13%.

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-year ILRT over the baseline case is computed as follows:

$$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0$$

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 1.5017 person-rem/yr [Table 8-4]
$TOTAL_{10}$ = Total person-rem/yr for 10 yr ILRT interval = 1.5019 person-rem/yr[Table 8-5]

$\Delta\%Risk_{10} = [(1.5019 - 1.5017) / 1.5017] \times 100.0$
$\Delta\%Risk_{10} = 0.011\%$

Therefore, the increase in risk due to the change in ILRT test frequency from three-in-ten years to 1-in-ten-years is 0.011%.

Risk Impact Due to 15-Year Test Interval

The risk contribution for a 15-year interval is calculated in a manner similar to the 10-year interval. The difference is in the increase in probability of leakage in Classes 3a and 3b. For this case, the value used in the analysis is 15 percent or 1.15 consistent with previously approved method [14, 21]. Specifically, there is a factor of 1.15 increase in Class 3a and 3b frequencies relative to the baseline associated with increasing the ILRT test interval from 3 yrs to 15 yrs. (See Section 4.4) The results for this calculation are presented in Table 8-6.

Based on the values from Table 8-6, the Type A 15-year test frequency percent risk contribution ($\%Risk_{15}$) for Class 3 is as follows:

$$\%Risk_{15} = [(CLASS3a_{15} + CLASS3b_{15}) / TOTAL_{15}] \times 100$$

Where:
$CLASS3a_{15}$ = Class 3a person-rem/year = 9.517E-4 person-rem/year [Table 8-6]
$CLASS3b_{15}$ = Class 3b person-rem/year = 1.096E-3 person-rem/year [Table 8-6]

$TOTAL_{15}$ = Total person-rem/yr for 15-year interval
= 1.5020 person-rem/yr [Table 8-6]

$\%Risk_{15} = [(9.517E\text{-}4 + 1.096E\text{-}3)/1.5020] \times 100$
$\%Risk_{15} = 0.136\%$

Therefore, the Total 15-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 0.136%.

The percent increase on the <u>total integrated plant risk</u> when the ILRT is extended from 10 years to 15 years is computed as follows:

$$\%TOTAL_{10\text{-}15} = [(TOTAL_{15} \text{ B } TOTAL_{10}) / TOTAL_{10}] \times 100$$

Where:
$TOTAL_{10}$ = Total person-rem/year for 10-year interval
= 1.5019 person-rem/year [Table 8-5]

$TOTAL_{15}$ = Total person-rem/year for 15-year interval
= 1.5020 person-rem/year [Table 8-6]

$\%TOTAL_{10\text{-}15} = [(1.5020 - 1.5019) / 1.5019] \times 100$
$\%TOTAL_{10\text{-}15} = 0.0056\%$

Therefore, the percent increase in total plant risk for these accident sequences, based on going from a 10-year ILRT interval to a 15-year interval, as influenced by Type A testing, is only 0..0056%.

The percent risk increase ($\Delta Risk_{15}$) due to a fifteen-year ILRT interval over the baseline is computed as follows:

$$\Delta Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100$$

Where:
$TOTAL_{BASE}$ = Total person-rem/year for baseline interval

= 1.5017 person-rem/year [Table 8-4]
$TOTAL_{15}$ = Total person-rem/year for 15-year interval
= 1.5020 person-rem/year [Table 8-6]

$\%\Delta Risk_{15}$ = [(1.5020 - 1.5017) /1.5017] x 100
$\%\Delta Risk_{15}$ = 0.0169%

Therefore, the total increase in risk contribution associated with relaxing the ILRT test frequency from three-in-ten-years to one-per-fifteen years is 0.0169%.

Step 4 – Determine the Change in Risk in Terms of Large Early Release Frequency (LERF)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. Class 3b is treated in this analysis as a potential LERF contributor. Class 3a is not treated as a "large" release. Therefore, for this evaluation, only Class 3b sequences have the potential to result in large releases if a pre-existing leak were present. Class 1 sequences are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Other accident classes such as 2, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval.

RG 1.174[3] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Because the ILRT does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the ILRT interval on the leakage probability.

Baseline (3 Yr Test Interval) LERF

From Section 6.2, the baseline LERF frequency is:

$LERF_{BASE}$ = 3.00E-7/year

LERF for 10-Yr Test Interval

The LERF increase ($\Delta LERF_{BASE-10}$) due to a 10-year ILRT over the baseline is computed as follows:

$\Delta LERF_{BASE-10} = CLASS3b_{10} - CLASS3b_{BASE}$

Where:
$CLASS3b_{BASE}$ = 1.74E-8/yr [Table 8-4]
$CLASS3b_{10}$ = 1.92E-8/yr [Table 8-5]

$\Delta LERF_{BASE-10}$ = 1.92E-8/yr - 1.74E-8/yr = 1.8E-9/yr

$LERF_{10} = LERF_{BASE} + \Delta LERF_{BASE\text{-}10}$

$LERF_{10}$ = 3.00E-7/year + 1.8E-9/yr = 3.02E-7/yr

LERF for 15-Yr Test Interval

The LERF increase ($\Delta LERF_{BASE\text{-}15}$) due to a 15-year ILRT over the baseline is computed as follows:

$\Delta LERF_{BASE\text{-}15} = CLASS3b_{15} - CLASS3b_{BASE}$

Where:
$CLASS3b_{BASE}$ = 1.74E-8/yr [Table 8-4]
$CLASS3b_{15}$ = 2.01E-8/yr [Table 8-6]

$\Delta LERF_{BASE\text{-}15}$ = 2.01E-8/yr - 1.74E-8/yr = 2.67E-9/yr

$LERF_{15} = LERF_{BASE} + \Delta LERF_{BASE\text{-}15}$

$LERF_{15}$ = 3.00E-7/year + 2.67E-9/yr = 3.03E-7/year

The LERF increase ($\Delta LERF_{10\text{-}15}$) due to a 15-year ILRT over the 10-yr ILRT is as follows:

$\Delta LERF_{10\text{-}15} = CLASS3b_{15}$ B $CLASS3b_{10}$

Where:
$CLASS3b_{10}$ = 1.92E-8/yr [Table 8-5]
$CLASS3b_{15}$ = 2.01E-8/yr [Table 8-6]

$\Delta LERF_{10\text{-}15}$ = 2.01E-8/yr -1.92E-8/yr = 8.74E-10/yr

It should be noted that the calculated changes in LERF for all cases are less than the 1.0E-7/yr screening criterion in Reg. Guide 1.174 and represent a very small change in risk.

Step 5 – Determine Impact on the Conditional Containment Failure Probability

Another parameter that the NRC Guidance Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP is calculated from the risk calculations performed in this analysis. The CCFP is "conditional" in that it identifies the probability of containment failure given that a severe accident (i.e., core damage) has occurred. Containment

failure in this context includes all radionuclide release end states other than the intact state that do not involve containment bypass. Generally, this means non-bypass, non-Class 1 sequences.

Since the only classes that are increasing are Classes 3a and 3b, the change in CCFP can be calculated by the difference in these classes. The CCFP calculation for the base case ($CCFP_{BASE}$) is shown below:

The percent increase in CCFP ($\Delta\%CCFP_{BASE\text{-}10}$) due to a 10-year ILRT over the baseline is computed as follows:

$$\Delta\%CCFP_{BASE\text{-}10} = [((F_{CLASS\ 3a\_10} + F_{CLASS\ 3b\_10}) - (F_{CLASS\ 3a\_BASE} + F_{CLASS3b\_BASE}))/CDF] \times 100$$
$$= [((5.84E\text{-}8 + 1.92E\text{-}8) - (5.31E\text{-}8 + 1.74E\text{-}8)) / 3.30E\text{-}6] \times 100$$
$$= 0.055\%$$

The percent increase in CCFP increase ($\Delta\%CCFP_{BASE\text{-}15}$) due to a 15-year ILRT over the baseline is computed as follows:

$$\Delta\%CCFP_{BASE\text{-}15} = [((F_{CLASS\ 3a\_15} + F_{CLASS\ 3b\_15}) - (F_{CLASS\ 3a\_BASE} + F_{CLASS3b\_BASE}))/CDF] \times 100$$
$$= [((6.11E\text{-}8 + 2.01E\text{-}8) - (5.31E\text{-}8 + 1.74E\text{-}8)) / 3.30E\text{-}6] \times 100$$
$$= 0.081\%$$

The percent increase in CCFP increase ($\Delta\%CCFP_{10\text{-}15}$) due to a 15-year ILRT over the 10-year ILRT is as follows:

$$\Delta\%CCFP_{10\text{-}15} = [((F_{CLASS\ 3a\_15} + F_{CLASS\ 3b\_15}) - (F_{CLASS\ 3a\_10} + F_{CLASS\ 3b\_10}))/CDF] \times 100$$
$$= [((6.11E\text{-}8 + 2.01E\text{-}8) - (5.84E\text{-}8 + 1.92E\text{-}8)) / 3.30E\text{-}6] \times 100$$
$$= 0.0265\%$$

This change in CCFP of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

Results Summary

The following are the key results of the ILRT test interval extension risk analysis (using EPRI TR-104285 methodology):

1. The baseline risk contribution (person-rem/yr) associated with containment leakage affected by the ILRT and represented by Class 3 accident scenarios is 0.118% of the total risk.

2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.130% of the total risk.

3. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-10 years is 0.0113%.

4. When the ILRT interval is 15 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.136% of the total risk.

5. The total integrated increase in risk contribution from reducing the ILRT test frequency from the once-per-10-year frequency to once-per-15 years is 0.0056%.

6. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-15 years is 0.0169%.

7. There is no change in the at-power CDF associated with the ILRT extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

8. The increase in LERF from the original 3-in-10 years test frequency to once-per-10 years is 1.8E-9/yr. This is considered to be "very small" using the acceptance guidelines in Reg. Guide 1.174.

9. The increase in LERF from the original 3-in-10 years test frequency to once-per-15 years is 2.67E-9/yr. This is also considered to be "very small" using the acceptance guidelines in Reg. Guide 1.174.

10. The increase in LERF from reducing the ILRT test frequency from once-per-10 years to one-per-15 years is 8.74E-10/yr. This is determined to be a very small using the acceptance guidelines of Reg. Guide 1.174.

11. The change in CCFP of less than 1% for both cases (when reducing test frequency to either once-per-10 or to once-per-15 years), is judged to be insignificant and reflects sufficient defense-in-depth.

Other significant results are summarized in Table 8-7.

Table 8-10 EPRI ACCIDENT CLASS DEFINITIONS

Accident Classes (Containment Release Type)	Description
1	No Containment Failure
2	Large Isolation Failures (Failure to Close)
3a	Small Isolation Failures (liner breach)
3b	Large Isolation Failures (liner breach)
4	Small Isolation Failures (Failure to seal-Type B)
5	Small Isolation Failures (Failure to seal-Type C)
6	Other Isolation Failures (e.g., dependent failures)
7	Failures Induced by Phenomena (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End states (including very low and no release)

Table 8-11 EPRI ACCIDENT CLASS FREQUENCIES BASED ON NMPS PRA - EPRI TR-104285

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Contribution to CDF (%)
1	No Containment Failure	7.59E-07	23.0341%
2	Large Isolation Failures (Fail to Close)	3.20E-07	9.7137%
3A	Small Isolation Failures (Liner Breach)	5.31E-08	1.6111%
3B	Large Isolation Failures (Liner Breach)	1.74E-08	0.5287%
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.0000%
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.0000%
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.0000%
7	Failures induced by Phenomena (early and late)	2.06E-06	62.4191%
8	Bypass (Interfacing Systems LOCA)	8.88E-08	2.6933%
Total		**3.30E-06**	

Table 8-12 POPULATION DOSE ESTIMATES FOR NMPS AT 50 MILES

Accident Classes (Containment Release Type)	Description	Person-Rem (50 miles)
1	No Containment Failure	1.56E+3
2	Large Isolation Failures (Failure to Close)	3.03E+5
3a	Small Isolation Failures (liner breach)	1.56E+4
3b	Large Isolation Failures (liner breach)	5.45E+4
4	Small Isolation Failures (Failure to seal-Type B)	0.00E+0
5	Small Isolation Failures (Failure to seal-Type C)	0.00E+0
6	Other Isolation Failures (e.g., dependent failures)	0.00E+0
7	Failures Induced by Phenomena	6.68E+5
8	Bypass (Interfacing System LOCA)	3.03E+5

Table 8-13 POPULATION DOSE RATE: 3/10-YR ILRT TEST INTERVAL - EPRI TR-104285

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	7.59E-07	1.56E+03	1.183E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	5.31E-08	1.56E+04	8.275E-04
3B	Large Isolation Failures (Liner Breach)	1.74E-08	5.45E+04	9.504E-04
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5017**

Table 8-14 POPULATION DOSE RATE: 10-YR ILRT TEST INTERVAL - EPRI TR-104285

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	7.52E-07	1.56E+03	1.172E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	5.84E-08	1.56E+04	9.103E-04
3B	Large Isolation Failures (Liner Breach)	1.92E-08	5.45E+04	1.049E-03
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5019**

Table 8-15 POPULATION DOSE RATE: 15-YR ILRT TEST INTERVAL - EPRI TR-104285

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	7.49E-07	1.56E+03	1.167E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	6.11E-08	1.56E+04	9.517E-04
3B	Large Isolation Failures (Liner Breach)	2.01E-08	5.45E+04	1.096E-03
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5020**

Table 8-16 SUMMARY OF RISK IMPACT OF TYPE A ILRT TEST FREQUENCIES - EPRI TR-104285

Risk Metric	Risk Impact (Baseline)	Risk Impact (10-years)	Risk Impact (15-years)
Class 3a and 3b Risk Contribution	0.12% of total integrated value 1.78E-3 person-rem/yr	0.13% of total integrated value 1.96E-3 person-rem/yr	0.14% of total integrated value 2.05E-3 person-rem/yr
Total Integrated Risk	1.5017 person-rem/year	1.5019 person-rem/year	1.5020 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	0.011%	0.017%
Increase in LERF over Baseline	N/A	1.8E-9/yr	2.7E-9/yr
Percent Increase in CCFP over Baseline	N/A	0.055%	0.081%
Percent Increase in Integrated Risk over 10-yr ILRT	N/A	N/A	0.0056%
Increase in LERF over 10-yr ILRT	N/A	N/A	8.7E-10/yr
Percent Increase in CCFP over 10-yr ILRT	N/A	N/A	0.0265%

9. Application of NEI Interim Guidance Methodology

9.1 Summary of Methodology

The results of the risk assessment performed using the methodology of EPRI TR-104285 [2], were provided in Section 8 of this document. In 2001 NEI recognized a need to update this methodology to support future risk-informed ILRT interval extension submittals. The methodology update focused on the following three particular areas.

1. The methodology for determining the overall probability of leakage resulting from extending surveillance intervals was revised. For an ILRT interval extension of 3 in 10 years to 1 in 10 years, the overall 10-year dose should have been calculated using an increased probability of an undetected leak (a leak detectable only by an ILRT that goes undetected due to the increased test interval) of 333.3% (increased by a factor of 3.33), as opposed to the 10% value used in the EPRI TR-104285 methodology. However, NEI also showed this methodology change to have a very small incremental risk contribution, since ILRTs only address a very small portion of the severe accident risk.

2. The methodology used to determine the frequencies of leakages detectable only by ILRTs (EPRI Classes 3a and 3b) was revised. Updated ILRT failure data was incorporated into the calculation of these containment failure classes. The guidance recommended use of a mean frequency calculation for the Class 3a distribution, and recommended the use of a Jeffery's non-informative prior distribution for the Class 3b distribution. The impact of this methodology change was to increase the probability of Class 3b releases. However, it was noted that no observed failure to date was even close in size to that necessary to cause a large release.

3. The updated guidance included provisions for utilizing NUREG/CR-4551 dose calculations, a necessary improvement to make the methodology usable for plants that do not have a Level-3 PRA.

Other improvements in the methodology include use of a simplified risk model (as opposed to the Containment Event Tree model used in EPRI TR-104285) to distinguish between those accident sequences that are affected by the status of the containment isolation system versus those that are a direct function of severe accident phenomena, and evaluation of the change in LERF by manipulating the probability of a pre-existing leak (for either Class 3a and 3b end states) of sufficient leak size to produce a large, early release.

9.2 Analysis Approach

This section presents the steps involved in performing the ILRT extension risk assessment based on the methodology of the 2001 NEI Interim Guidance.

The nine analysis steps identified in the NEI Interim Guidance are:

1. Quantify the base line (nominal three year ILRT interval) risk in terms of frequency per reactor year for the EPRI accident classes of interest. Note that Classes 4, 5, and 6 are not affected by changes in ILRT test frequency. Therefore, these classes are not considered in this assessment methodology.
2. Determine the containment leakage rates for applicable cases, 3a and 3b.
3. Develop the baseline population dose (person-rem, based on leakage, volume and power level) for the applicable accident classes.
4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in step (3) by the associated frequency calculated in step (1).
   (Note: The method provides for use of the NUREG/CR-4551 population dose methods. If plant-specific values are available, they may be used. The net result is expressed as a percentile change.)
5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.
6. Determine the population dose rate for the new surveillance intervals of interest.
7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
8. Evaluate the risk impact in terms of LERF.
9. Evaluate the change in conditional containment failure probability.

Each of these steps is described in detail below. Note that this methodology builds upon the methodology of EPRI TR-104285. Therefore, most of the plant specific information necessary to perform the assessment using this methodology was presented in Sections 6 and 8 above (reference is made as necessary to the appropriate section in Section 6 or 8 for the development of the common information).

Step 1: Quantify the base line (nominal three year ILRT interval) risk in terms of frequency per reactor year for the EPRI accident classes of interest.

The baseline EPRI accident class frequencies used in the NEI methodology case are unchanged from those calculated in Sections 6 and 8 above, with the exceptions of the frequencies for EPRI categories 1 (No Containment Failure) and 3a (Small Containment Isolation Failures due to Liner Breach) and 3b (Large Containment Isolation Failures due to Liner Breach). As described above, the frequencies of leakages detectable only by ILRTs (EPRI Classes 3a and 3b) was revised. The NEI Interim Guidance included the results of additional, updated ILRT failure data (38 more

industry tests conducted since 1/1/1995). Adding these to the NUREG-1493 data (144 ILRTs) resulted in a total population of 182 tests. One more failure was added (due to construction debris from a penetration modification), resulting in a total of 5 failures over these 182 tests. The guidance recommended use of a mean frequency (5/182 = 0.027) for the Class 3a distribution, and recommended the use of a Jeffery's non-informative prior distribution for the Class 3b distribution:

$$\begin{aligned}\text{Failure Probability}_{3b} &= (\text{Number of Failures} + ½) / (\text{Number of Tests} + 1)\\ &= (0 + ½) / (182 + 1)\\ &= 0.0027\end{aligned}$$

Using these values, the calculation of the baseline Class 3a and 3b frequencies was performed by multiplying them to IPE Class 1 frequency as follows:

CLASS_3a_FREQUENCY = 0.027 * 8.30E-7/year = 2.24E-8/year

CLASS_3b_FREQUENCY = 0.0027 * 8.30E-7/year = 2.28E-9/year

In order to maintain the sum of the frequencies of the accident classes equal to the CDF, the NEI Interim Guidance specifies that the Class 1 frequency be adjusted for the Class 3 sequences. The baseline Class 1 frequency was determined as follows:

CLASS_1_FREQUENCY
= (IPE Class 1) - (Class 3a + Class 3b)
= 8.30E-7/year - (2.24E-8/year + 2.28E-9/year)/yr
= 8.05E-7/year

Table 9-1 below provides the NMPS accident class frequencies that were used in the application of the NEI Interim Guidance methodology.

Step 2: Determine the containment leakage rates for applicable cases, 3a and 3b.

Step 3: Develop the baseline population dose (person-rem calculated based on leakage, volume and power level) for the applicable accident classes.

Step 4: Determine the population dose rate (person-rem/year) by multiplying the dose calculated in step (3) by the associated frequency calculated in step (1).

Each of the calculations necessary for these steps were performed exactly as presented in Section 8. The resulting population dose rates for all accident classes are identical to that presented in Section 8, with the exception of Classes 1, 3a and 3b (the accident sequence frequencies of which were modified per the NEI guidance as described in Step 1 above). Table 9-2 provides the baseline results for the population dose rates by accident class.

The calculation of the baseline risk contribution from Class 3 (i.e., the class affected by the ILRT interval change) was also done consistent with the method presented in Section 8.2. Based on the risk values from Table 9-2, the percent risk contribution ($\%Risk_{BASE}$) for Class 3 is as follows:

$\%Risk_{BASE} = [(CLASS3a_{BASE} + CLASS\ 3b_{BASE})/\ Total_{BASE}] \times 100$

Where:
$CLASS3a_{BASE}$ = Class 3a person-rem/year = 3.49E-4 person-rem/year [Table 9-2]
$CLASS3b_{BASE}$ = Class 3b person-rem/year = 1.24E-4 person-rem/year [Table 9-2]

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval
= 1.5005 person-rem/yr [Table 9-2]

$\%Risk_{BASE} = [(3.49E\text{-}4 + 1.24E\text{-}5)/\ 1.5005] \times 100$
$\%Risk_{BASE} = 0.0316\%$

Steps 5, 6, and 7 are performed below the description for Step 7.

Step 5: Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

Step 6: Determine the population dose rate for the new surveillance intervals of interest.

Step 7: Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

The increase in the Class 3 leakage frequencies for the surveillance intervals of interest (10 years and 15 years) were computed using the same methodology used in Section 8 above, except that the overall 10-year dose was calculated using an increased probability of an undetected leak of 333.3% (increased by a factor of 3.33), as opposed to the 10% value (factor of 1.1) used in the EPRI TR-104285 methodology. Likewise, the overall 15-year dose was calculated using an increased probability of an undetected leak of 500% (increased by a factor of 5.0). As described in the NEI Interim Guidance, increasing the test interval from 3 in 10 years to 1 in 10 years increases the average time that a leak (detectable only by an ILRT) goes undetected from 18 (3yrs/2) to 60 (10 yrs/2) months. This is a factor of 60/18=3.333. By the same logic, increasing the test interval from 3 in 10 years to 1 in 15 years increases the average time that a leak goes undetected from 18 (3yrs/2) to 90 (15 yrs/2) months, a factor of 90/18 = 5.0.

The increase in Class 3B frequency due to undetected corrosion-related leakage, calculated in Appendix A, was included in the calculation as described in Section 8 above.

Tables 9-3 and 9-4 provide the results of the population dose rate calculations for the cases where the ILRT interval is extended to 10 years and 15 years, respectively.

Based on the risk values from Tables 9-3 and 9-4, the percent risk contribution for Class 3 over the two proposed ILRT extension intervals ($\%Risk_{10}$ and $\%Risk_{15}$) was calculated as follows:

$CLASS3a_{10}$ = Class 3a person-rem/year = 1.16E-3 person-rem/year [Table 9-3]
$CLASS\ 3b_{10}$ = Class 3b person-rem/year = 4.16E-4 person-rem/year [Table 9-3]

$CLASS3a_{15}$ = Class 3a person-rem/year = 1.75E-3 person-rem/year [Table 9-4]
$CLASS\ 3b_{15}$ = Class 3b person-rem/year = 6.24E-4 person-rem/year [Table 9-4]

$TOTAL_{10}$ = Total person-rem/yr for 10-year interval = 1.5015 person-rem/yr [Table 9-3]
$TOTAL_{15}$ = Total person-rem/yr for 15-year interval = 1.5022 person-rem/yr [Table 9-4]

$\%Risk_{10}$ = [(1.16E-3 + 4.16E-4) / 1.5015] x 100
$\%Risk_{10}$ = 0.105%

$\%Risk_{15}$ = [(1.75E-3 + 6.24E-4) / 1.5022] x 100
$\%Risk_{15}$ = 0.158%

Therefore, the Total Type A 10-year ILRT interval risk contribution of leakage, represented by Class 3 accident scenarios is 0.105% for the ILRT interval extension to 1 in 10 years, and 0.158% for the ILRT interval extension to 1 in 15 years.

The percent risk increase ($\Delta\%Risk$) for each ILRT extension case over the baseline case is computed as follows:

$\Delta\%Risk_{10} = [(Total_{10} - Total_{BASE}) / Total_{BASE}] \times 100.0$
$\Delta\%Risk_{15} = [(Total_{15} - Total_{BASE}) / Total_{BASE}] \times 100.0$

$TOTAL_{BASE}$ = Total person-rem/yr for baseline interval = 1.5005 person-rem/yr [Table 9-2]
$TOTAL_{10}$ = Total person-rem/yr for 10 yr ILRT interval = 1.5015 person-rem/yr [Table 9-3]
$TOTAL_{15}$ = Total person-rem/yr for 15 yr ILRT interval = 1.5022 person-rem/yr [Table 9-4]

$\Delta\%Risk_{10}$ = [(1.5015 – 1.5005) / 1.5005] x 100.0
$\Delta\%Risk_{10}$ = 0.068%

$\Delta\%Risk_{15}$ = [(1.5022 - 1.5005) / 1.5005] x 100.0
$\Delta\%Risk_{15}$ = 0.116%

The percent risk increase ($\Delta\%Risk$) for ILRT extension from 1 in 10 years to 1 in 15 years is computed as follows:

$\Delta\%Risk_{15\text{-}10} = [(Total_{15} - Total_{10}) / Total_{10}] \times 100.0$

$\Delta\%Risk_{15\text{-}10} = [(1.5022 - 1.5015) / 1.5015] \times 100.0$
$\Delta\%Risk_{15\text{-}10} = 0.048\%$

Therefore, the increase in risk due to changing the ILRT test interval of three-in-ten years to 1-in-ten-years is 0.068%, while the increase due to changing the ILRT test interval of three-in-ten years to a 1-in-15 year test interval is 0.116%. The increase due to changing the ILRT test interval of 1-in-10 years to a 1-in-15 year test interval is 0.048%

Step 8: Evaluate the risk impact in terms of LERF.

Baseline (3 Yr Test Interval) LERF

From Section 6.2, the baseline LERF frequency is:

$LERF_{BASE} = 3.0E\text{-}7/year$

LERF for 10-Yr Test Interval

The LERF increase ($\Delta LERF_{BASE\text{-}10}$) due to a 10-year ILRT over the baseline is as follows:

$\Delta LERF_{BASE\text{-}10} = CLASS3b_{10} - CLASS3b_{BASE}$

Where:
$CLASS3b_{BASE} = 2.28E\text{-}9/yr$ [Table 9-2]
$CLASS3b_{10} = 7.62E\text{-}9/yr$ [Table 9-3]

$\Delta LERF_{BASE\text{-}10} = 7.62E\text{-}9/yr - 2.28E\text{-}9/yr = 5.34E\text{-}9/yr$

$LERF_{10} = LERF_{BASE} + \Delta LERF_{BASE\text{-}10}$

$LERF_{10} = 3.00E\text{-}7/year + 5.34E\text{-}9/yr = 3.05E\text{-}7/yr$

LERF for 15-Yr Test Interval

The LERF increase ($\Delta LERF_{BASE\text{-}10}$) due to a 15-year ILRT over the baseline is as follows:

$\Delta LERF_{BASE\text{-}15} = CLASS3b_{15} - CLASS3b_{BASE}$

Where:
$CLASS3b_{BASE} = 2.28E\text{-}9/yr$ [Table 9-2]
$CLASS3b_{15} = 1.14E\text{-}8/yr$ [Table 9-4]

$\Delta LERF_{BASE\text{-}15} = 1.14E\text{-}8/yr - 2.28E\text{-}9/yr = 9.15E\text{-}9/yr$

$LERF_{15} = LERF_{BASE} + \Delta LERF_{BASE-15}$

$LERF_{15}$ = 3.00E-7/year + 9.15E-9/yr = 3.09E-7/year

The LERF increase ($\Delta LERF_{10-15}$) due to a 15-year ILRT over the 10-yr ILRT is as follows:

$\Delta LERF_{10-15} = CLASS3b_{15} - CLASS3b_{10}$

Where:
$CLASS3b_{10}$ = 7.62E-9/yr [Table 9-3]
$CLASS3b_{15}$ = 1.14E-8/yr [Table 9-4]

$\Delta LERF_{10-15}$ = 1.14E-8/yr – 7.62E-9/yr = 3.8E-9/yr

It should be noted that the calculated changes in LERF for all cases are less than the 1.0E-7/yr screening criterion in RG 1.174 and represent a very small change in risk.

Step 9: Evaluate the change in conditional containment failure probability.

The assessment of conditional containment failure probability (CCFP) for each of the cases (base, 10-year ILRT interval extension, 15-year ILRT interval extension) is performed in accordance with the NEI Interim Guidance methodology [23].

The CCFP calculation for the base case ($CCFP_{BASE}$) is shown below [23]:

$CCFP_{BASE}$ = 1 – (Intact Containment $Frequency_{BASE}$/Total CDF)
= {1 – (Class 1_{BASE} + Class $3a_{BASE}$)/CDF}*100
= {1 – (8.05E-7 + 2.24E-8)/3.30E-6}*100
= 74.90%

The CCFP calculation for the ILRT extension cases ($CCFP_{10}$ and $CCFP_{15}$) is performed in a similar manner:

$CCFP_{10}$ = 1 – (Intact Containment $Frequency_{10}$/Total CDF)
= {1 – (Class 1_{10} + Class $3a_{10}$)/CDF}*100
= {1 – (7.48E-7 + 7.47E-8)/3.30E-6}*100
= 75.06%

$CCFP_{15}$ = 1 – (Intact Containment $Frequency_{15}$/Total CDF)
= {1 – (Class 1_{15} + Class $3a_{15}$)/CDF}*100
= {1 – (7.07E-7 + 1.12E-7)/3.30E-6}*100
= 75.17%

The percent increase in CCFP ($\Delta\%CCFP_{BASE-10}$) from a 10-year to a 15-year ILRT is computed as follows:

$\Delta\%CCFP_{BASE\text{-}10} = CCFP_{10} - CCFP_{BASE}$
$= 75.06\% - 74.90\%$
$= 0.16\%$

The percent increase in CCFP increase ($\Delta\%CCFP_{BASE\text{-}15}$) due to a 15-year ILRT over the baseline is as follows:

$\Delta\%CCFP_{BASE\text{-}15} = CCFP_{15} - CCFP_{BASE}$
$= 75.17\% - 74.90\%$
$= 0.27\%$

The percent increase in CCFP increase ($\Delta\%CCFP_{10\text{-}15}$) due to a 15-year ILRT over the 10-year ILRT is as follows:

$\Delta\%CCFP_{10\text{-}15} = CCFP_{15} - CCFP_{10}$
$= 75.17\% - 75.06\%$
$= 0.11\%$

This change in CCFP is judged to be insignificant and reflects sufficient defense-in-depth.

9.3 Results Summary

The following are the key results of the ILRT test interval extension risk analysis based on the NEI Interim Guidance Methodology:

1. The baseline risk contribution (person-rem/yr) associated with containment leakage affected by the ILRT and represented by Class 3 accident scenarios is 0.032% of the total risk.

2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.105% of the total risk.

3. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-10 years is 0.068.

4. When the ILRT interval is 15 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.158% of the total risk.

5. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-15 years is 0.116%.

6. There is no change in the at-power CDF associated with the ILRT extension. Therefore, this is within the RG 1.174 acceptance guidelines.

7. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-10 years is 5.34E-9/yr. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

8. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-15 years is 9.15E-9/yr. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

9. The risk increase in LERF from reducing the ILRT test frequency from once-per-10 years to one-per-15 years is 3.81E-9/yr. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

10. The change in CCFP of less than 1% for both cases, reducing test frequency to either once-per-10 or once-per-15 years, is judged to be insignificant and reflects sufficient defense-in-depth.

Other results are summarized in Table 9-5.

Table 9-17 EPRI ACCIDENT CLASS FREQUENCIES FOR NMPS - NEI Interim Guidance

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Contribution to CDF (%)
1	No Containment Failure	8.05E-07	24.4250%
2	Large Isolation Failures (Fail to Close)	3.20E-07	9.7137%
3A	Small Isolation Failures (Liner Breach)	2.24E-08	0.6797%
3B	Large Isolation Failures (Liner Breach)	2.28E-09	0.0692%
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.0000%
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.0000%
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.0000%
7	Failures induced by Phenomena (early and late)	2.06E-06	62.4191%
8	Bypass (Interfacing Systems LOCA)	8.88E-08	2.6933%
Total		**3.30E-06**	

Table 9-18 POPULATION DOSE RATE: 3/10-YR ILRT TEST INTERVAL - NEI Interim Guidance

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	8.05E-07	1.56E+03	1.255E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	2.24E-08	1.56E+04	3.491E-04
3B	Large Isolation Failures (Liner Breach)	2.28E-09	5.45E+04	1.243E-04
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5005**

Table 9-19 POPULATION DOSE RATE: 10-YR ILRT TEST INTERVAL - NEI Interim Guidance

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	7.48E-07	1.56E+03	1.165E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	7.47E-08	1.56E+04	1.164E-03
3B	Large Isolation Failures (Liner Breach)	7.62E-09	5.45E+04	4.156E-04
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5015**

Table 9-20 POPULATION DOSE RATE: 15-YR ILRT TEST INTERVAL - NEI Interim Guidance

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	7.07E-07	1.56E+03	1.101E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	1.12E-07	1.56E+04	1.746E-03
3B	Large Isolation Failures (Liner Breach)	1.14E-08	5.45E+04	6.235E-04
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5022**

Table 9-21 SUMMARY OF RISK IMPACT ON TYPE A ILRT FREQUENCY - NEI Interim Guidance

Risk Metric	(Baseline) Risk Impact	Risk Impact (10-years)	Risk Impact (15-years)
Class 3a and 3b Risk Contribution	0.031% of total integrated value 4.73E-4 person-rem/yr	0.105% of total integrated value 1.58E-3 person-rem/yr	0.158% of total integrated value 2.37E-3 person-rem/yr
Total Integrated Risk	1.5005 person-rem/year	1.5015 person-rem/year	1.5022 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	0.0677%	0.1161%
Increase in LERF over Baseline	N/A	5.34E-9/yr	9.15E-9/yr
Percent Increase in CCFP over Baseline	N/A	0.16%	0.27%
Percent Increase in Integrated Risk over 10-yr ILRT	N/A	N/A	0.048%
Increase in LERF over 10-yr ILRT	N/A	N/A	3.81E-9/yr
Percent Increase in CCFP over 10-yr ILRT	N/A	N/A	0.12%

10. Application of EPRI TR-1009325 Methodology

10.1 Summary of Methodology

EPRI TR-1009325 [17] is an update to EPRI TR-104285 [2] (which, in turn, was built upon the guidance of NUREG-1493 [4]) that includes the changes to the methodology included in the NEI Interim Guidance [23], plus additional enhancements that were obtained through an expert elicitation process. In addition, the methodology incorporates the results of NRC comments on various industry ILRT interval extension submittals. The expert elicitation was aimed at reducing the conservatisms associated with the various containment leakage methodologies available that were found to provide widely differing risk results when applied to the same problem. The methodology enhancements support relaxation of ILRT intervals up to 20 years (ILRT extension had only requested interval extensions up to 15 years).

The enhancements in TR-1009325 are generally in the following three areas:

1. Definition (in terms of the required resulting L_a leakage term) of the assumed containment leakage size that could lead to a large, early release (LERF), i.e., EPRI accident Class 3b. Whereas previous submittals assumed a very conservative leakage term (35 L_a) would have the potential to result in a LERF event, the methodology provides a basis for using a (still conservative) value of 100 L_a instead. For the smaller pre-existing leak (accident Class 3a) size, the previously used conservative value of 10 L_a was retained by the methodology.
2. Development of specific probabilities for pre-existing containment leakage sizes. This was done through the expert elicitation process. EPRI TR-1009325 states that this method provides a considerable improvement over the use of non-informative priors (as has been done in previous licensee submittals based on application of the previous EPRI TR-104285 methodology).
3. Consideration of the potential risk benefits associated with other containment inspections (non-ILRT) and potential indirect containment monitoring techniques that would provide indications of a containment leak (determination of the probability of leakage detection over an increased ILRT interval, again through use of the expert elicitation process).

Application of the EPRI TR-1009325 methodology generally produces results that indicate lower population dose risk than previous methodologies due to the reduction in the conservatisms noted above.

10.2 Analysis Approach

Implementation of the methodology of EPRI TR-1009325 is very similar to the implementation of the NEI Interim Guidance discussed in Section 9.2 (the steps required for the analysis

identified in TR-1009325 are identical with those presented in the NEI Interim Guidance). The practical differences between the two analyses are in the inputs used for determining the leak size requirements for LERF categorization (EPRI Class 3b), and in the probability values applied to the assumed undetected leakage categories. Therefore, in this section, the calculation discussion focuses on the changes in these inputs only. The calculation details followed are identical to those shown for the NEI Interim Guidance (Section 9.2). The presentation of results in Section 10.3 mirrors that provided for the other two methodologies.

Step 1: Quantify the base line (nominal three year ILRT interval) risk in terms of frequency per reactor year for the EPRI accident classes of interest.

Step 1 was quantified as described in Section 9.2, except in the leakage size and probabilities determined for Class 3a and Class 3b accident sequences.

Licensee ILRT extension submittals based on previous methodologies (EPRI TR-104285, 2001 NEI Interim Guidance) relied upon statistical failure data updates using non-informative priors in order to determine the probability values for containment leakage identifiable only through ILRTs (particularly Class 3b). As the risk results are sensitive to the 3b values, the choice of statistical methodology applied was seen to produce a somewhat wide range of risk results. EPRI TR-1009325 used expert elicitation to develop a relationship between the size of potential containment leakage pathways, expressed as L_a, and the probability of occurrence. This methodology was seen as a considerable improvement over the use of non-informative priors.

A summary of the final results of the statistical analysis of the expert elicitation (leak size vs. probability) are given in Table D-1 of EPRI TR-1009325. As stated in Section 10.1, for Class 3 leakage scenarios, the EPRI TR-1009325 methodology specifies the use of 10 L_a as a conservative upper bound leakage size for Class 3a sequences, and 100 L_a as a conservative upper bound leakage size for Class 3b sequences. From Table D-1 of EPRI TR-1009325, the mean probability of occurrence for a 10 L_a (Class 3a) leak is 3.88E-3, and the mean probability of occurrence for a 100 L_a (Class 3b) leak is 2.47E-4. Using these values, the calculation of the baseline Class 3a and 3b distributions was performed as follows:

CLASS_3a_FREQUENCY = 3.88E-3 * 8.30E-7/year = 3.22E-9/year

CLASS_3b_FREQUENCY = 2.47E-4 * 8.30E-7/year = 2.05E-10/year

These values are about an order of magnitude lower than the values calculated in Sections 8.1 (TR-104285 methodology) and 9.2 (NEI Interim Guidance methodology).

The remainder of the Step 1 calculation follows the same process as that presented in Section 9.2 above.

Steps 2 – 9:

The process followed to complete Steps 2 – 9 for the EPRI TR-1009325 methodology was the same as that presented in Section 9.2. Tables 10-1 through 10-4 provide the interim results of the EPRI TR-1009325 methodology.

10.3 Results Summary

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis (as calculated in Section 10 – EPRI TR-1009325 Methodology):

1. The baseline risk contribution (person-rem/yr) associated with containment leakage affected by the ILRT and represented by Class 3 accident scenarios is 0.00547% of the total risk.

2. When the ILRT interval is 10 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.0183% of the total risk.

3. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-10 years is 0.0120%.

4. When the ILRT interval is 15 years, the risk contribution of leakage (person-rem/yr) represented by Class 3 accident scenarios is increased to 0.0274% of the total risk.

5. The total integrated increase in risk contribution from reducing the ILRT test frequency from 3-per-10-year (baseline) frequency to once-per-15 years is 0.0205%.

6. There is no change in the at-power CDF associated with the ILRT extension. Therefore, this is within the Reg. Guide 1.174 acceptance guidelines.

7. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-10 years is 4.80E-10/yr. This is within the acceptance guidelines in Reg. Guide 1.174.

8. The risk increase in LERF from the original 3-in-10 years test frequency to once-per-15 years is 8.23E-10/yr. This is within the acceptance guidelines in Reg. Guide 1.174.

9. The risk increase in LERF from reducing the ILRT test frequency from once-per-10 years to one-per-15 years is 3.43E-10/yr. This is within the acceptance guidelines in Reg. Guide 1.174.

10. The change in CCFP of less than 1% for both cases, reducing test frequency to either once-per-10 or once-per-15 years, is judged to be insignificant and reflects sufficient defense-in-depth.

Other significant results are summarized in Table 10-5.

Table 10-22 EPRI ACCIDENT CLASS FREQUENCIES FOR NMPS - EPRI TR-1009325

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Contribution to CDF (%)
1	No Containment Failure	8.27E-07	25.0700%
2	Large Isolation Failures (Fail to Close)	3.20E-07	9.7137%
3A	Small Isolation Failures (Liner Breach)	3.22E-09	0.0977%
3B	Large Isolation Failures (Liner Breach)	2.05E-10	0.0062%
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.0000%
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.0000%
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.0000%
7	Failures induced by Phenomena (early and late)	2.06E-06	62.4191%
8	Bypass (Interfacing Systems LOCA)	8.88E-08	2.6933%
Total		**3.30E-06**	

Table 10-23 POPULATION DOSE RATE: 3/10-YR ILRT TEST INTERVAL - EPRI TR-1009325

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	8.27E-07	1.56E+03	1.288E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	3.22E-09	1.56E+04	5.017E-05
3B	Large Isolation Failures (Liner Breach)	2.05E-10	1.56E+05	3.194E-05
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5001**

Table 10-24 POPULATION DOSE RATE: 10-YR ILRT TEST INTERVAL - EPRI TR-1009325

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	8.19E-07	1.56E+03	1.275E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	1.07E-08	1.56E+04	1.672E-04
3B	Large Isolation Failures (Liner Breach)	6.85E-10	1.56E+05	1.068E-04
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5003**

Table 10-25 POPULATION DOSE RATE: 15-YR ILRT TEST INTERVAL - EPRI TR-1009325

EPRI Accident Class	Description	EPRI Accident Class Frequency (per year)	Population Dose (Entire Region, person-rem)	Population Dose Rate (Entire Region, person-rem/yr)
1	No Containment Failure	8.13E-07	1.56E+03	1.266E-03
2	Large Isolation Failures (Fail to Close)	3.20E-07	3.03E+05	9.690E-02
3A	Small Isolation Failures (Liner Breach)	1.61E-08	1.56E+04	2.508E-04
3B	Large Isolation Failures (Liner Breach)	1.03E-09	1.56E+05	1.602E-04
4	Small Isolation Failures (Fail to Seal - Type B)	0.00E+00	0.00E+00	0.000E+00
5	Small Isolation Failures (Fail to Seal - Type C)	0.00E+00	0.00E+00	0.000E+00
6	Other Isolation Failures (e.g., dependent failures)	0.00E+00	0.00E+00	0.000E+00
7	Failures induced by Phenomena (early and late)	2.06E-06	6.68E+05	1.375E+00
8	Bypass (Interfacing Systems LOCA)	8.88E-08	3.03E+05	2.687E-02
Total		**3.30E-06**		**1.5004**

Table 10-26 SUMMARY OF RISK IMPACT OF TYPE A ILRT TEST FREQUENCIES - EPRI TR-1009325

Risk Metric	Risk Impact (Baseline)	Risk Impact (10-years)	Risk Impact (15-years)
Class 3a and 3b Risk Contribution	0.0055% of total integrated value 8.21E-5 person-rem/yr	0.0183% of total integrated value 2.74E-4 person-rem/yr	0.0274% of total integrated value 4.11E-4 person-rem/yr
Total Integrated Risk	1.5001 person-rem/year	1.5003 person-rem/year	1.5004 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	0.0120%	0.0205%
Increase in LERF over Baseline	N/A	4.80E-10/yr	8.23E-10/yr
Percent Increase in CCFP over Baseline	N/A	0.0146%	0.0250%
Percent Increase in Integrated Risk over 10-yr ILRT	N/A	N/A	0.0085%
Increase in LERF over 10-yr ILRT	N/A	N/A	3.43E-10/yr
Percent Increase in CCFP over 10-yr ILRT	N/A	N/A	0.0104%

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 3.81E-9/yr. 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. It should be noted that Reference [B-6] did not identify any unique containment vulnerabilities for the seismic and fire evaluations.

The acceptance guidelines for Region III of the RG 1.174 LERF acceptability curve are that it can be reasonably shown that the total LERF is less than 1E-05/yr and that the cumulative changes be tracked. The baseline LERF is 3.00E-07. Based the LERF increase calculated using the NEI Interim Guidance (i.e., 3.81E-09), the total LERF for the requested change is 3.09E-07/yr which meets the total LERF criterion.

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 very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.
- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment liner.

NMPS Specific Risk Results

The findings for NMPS confirm the general findings of previous studies on a plant specific basis, including severe accident category frequencies, the containment failure modes, the Technical Specification allowed leakage, and the local population surrounding the NMPS station. Based on the results from Sections 8 through 11, the following conclusions regarding the assessment of the plant risk are associated with extending the Type A ILRT test from ten years to fifteen years:

- There is no change in the at-power CDF associated with the ILRT test interval extension. Therefore, this is within the RG 1.174 acceptance guidelines.

- RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases in CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test frequency from once-per-ten-years to once-per-fifteen years is between 3.43E-10/yr and 3.81E-9/yr. Guidance in RG 1.174 defines very small changes in LERF as below 1E-7/yr. Therefore, increasing the ILRT interval from 10 to 15 years is considered to result in a very small change to the NMPS risk profile.

- The change in Type A test frequency from once-per-ten-years to once-per-fifteen-years increases the total integrated plant risk by between 0.0056% and 0.0483%. Therefore, the risk impact change when compared to other severe accident risks is negligible.

- The change in CCFP of less than 1% is judged to be insignificant and reflects sufficient defense-in-depth.

- Incorporating external event results into this analysis does not change the conclusion of

this risk assessment (i.e., increasing the NMPS ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

Table 12-1 below summarizes the NMPS-specific results of this risk evaluation.

Table 12-27 OVERALL SUMMARY OF RISK IMPACT OF VARIOUS TYPE A ILRT TEST FREQUENCIES

Risk Metric	Risk Impact (Baseline)			Risk Impact (10-years)			Risk Impact (15-years)		
Calculation Methodology	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325
Class 3a and 3b Risk Contribution	0.12% of total integrated value 1.78E-3 person-rem/yr	0.032% of total integrated value 4.73E-4 person-rem/yr	0.0055% of total integrated value 8.21E-5 person-rem/yr	0.130% of total integrated value 1.96E-3 person-rem/yr	0.105% of total integrated value 1.58E-3 person-rem/yr	0.0183% of total integrated value 2.74E-4 person-rem/yr	0.136% of total integrated value 2.048E-3 person-rem/yr	0.158% of total integrated value 2.37E-3 person-rem/yr	0.0274% of total integrated value 4.11E-4 person-rem/yr
Total Integrated Risk	1.5017 person-rem/year	1.5005 person-rem/year	1.5001 person-rem/year	1.5019 person-rem/year	1.5015 person-rem/year	1.5003 person-rem/year	1.5020 person-rem/year	1.5022 person-rem/year	1.5004 person-rem/year
Percent Increase in Integrated Risk over Baseline	N/A	N/A	N/A	0.0113%	0.068%	0.012%	0.017%	0.116%	0.0205%
Increase in LERF over Baseline	N/A	N/A	N/A	1.80E-9/yr	5.34E-9/yr	4.80E-10/yr	2.67E-9/yr	9.15E-9/yr	8.23E-10/yr
Percent Increase in CCFP over Baseline	N/A	N/A	N/A	0.055%	0.16%	0.015%	0.081%	0..278%	0.0250%
Percent Increase in Integrated Risk over 10-yr ILRT	N/A	N/A	N/A	N/A	N/A	N/A	0.0056%	0.048%	0.0085%

Risk Metric	Risk Impact (Baseline)			Risk Impact (10-years)			Risk Impact (15-years)		
Calculation Methodology	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325	EPRI TR-104285	NEI Interim Guidance	EPRI TR-1009325
Increase in LERF over 10-yr ILRT	N/A	N/A	N/A	N/A	N/A	N/A	8.74E-10/yr	3.81E-9/yr	3.43E-10/yr
Percent Increase in CCFP over 10-yr ILRT	N/A	N/A	N/A	N/A	N/A	N/A	0.0265%	0.1156%	0.0104%

12.3 Risk Trade-off

The performance of an ILRT occurs during plant shutdown and introduces some small residual risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real shutdown risk impacts associated with the setup and performance of the ILRT during shutdown operation [10]. While these risks have not been quantified for NMPS, it is judged that there is a positive (yet un-quantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT test interval extension, there are, in fact, positive safety benefits that reduce the already small risk associated with the extension of the ILRT test interval.

13. References

1) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 1995.
2) EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI, Palo Alto, CA," August 1994.
3) Regulatory Guide 1.174 Rev 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.
4) NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995.
5) NUREG/CR-4551, SAND86-1309, Volume 4, Revision 1, Part 1, "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2, Main Report," December 1990.
6) NUREG-1273, "Technical Findings and Regulatory Analysis for Genetic Study Issue II.e.43 Containment Integrity Check," April 1988.
7) NUREG/CR-3539, ORNL/TM-8964, "Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory," April 1984.
8) NUREG/CR-4220, PNL-5432, "Reliability Analysis of Containment Isolation Systems, Pacific Northwest Laboratory," June 1985.
9) NUREG/CR-4330, PNL-5809, Vol. 2, "Review of Light Water Reactor Regulatory Requirements, Pacific Northwest Laboratory," June 1986.
10) TR-105189, Final Report, "Shutdown Risk Impact Assessment for Extended Containment Leakage Testing Intervals Utilizing ORAM™, EPRI, Palo Alto, CA," May 1995.
11) Individual Plant Examination Peach Bottom Atomic Power Station Units 2 and 3, Volumes 1 and 2 Philadelphia Electric Company, 1992.
12) DE-ACOG-87RL11313, "ALWR Severe Accident Dose Analysis," March 1989
13) NEI 94-01, Revison 1J (Draft), "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," December 2005

14) Letter from R. J. Barrett (Entergy) to U.S. Nuclear Regulatory Commission, IPN-01-007, dated January 16, 2001.
15) Letter from J. A. Hutton (Exelon, Peach Bottom) to U.S. Nuclear Regulatory Commission, Docket No. 50-278, License No. DDPR-56, LAR 01-00430, dated May 30, 2001.
16) Letter from D. E. Young (Florida Power) to U.S. Nuclear Regulatory Commission, 3F0401-11, dated April 25, 2001.
17) EPRI TR-1009325, Revision 1, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," December, 2005.
18) NUREG/CR-3539, "Burns, T. J., Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory," April 1984.
19) WASH-1400, "United States Nuclear Regulatory Commission, Reactor Safety Study," October 1975.
20) Letter from SNC (H. L. Summer, Jr.) to USNRC dated July 26, 2000.
21) United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment (Amendment No. 206) Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.
22) J.H. Moody, "Nine Mile Point Unit 1 RPA Quantification Notebook QU,"17 Revison 0 (2007) Internal Events PRA
23) J. Haugh, John Gisclon, W. Parkinson, Ken Canavan, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Intervals", Rev. 3.1, EPRI, October 2001.
24) E.T. Burns "Nine Mile Point Unit One Level II PRA", 2007 (LE Notebook)
25) Patrick D.T. O'Connor, "Practical Reliability Engineering", John Wiley & Sons, 2nd Edition, 1985.
26) NMP1 Individual Plant Examination of External Events (IPEEE), August 1996
27) NUREG-1437, Supplement 24, "SAMA Report for NMPS 1," May 2006

APPENDIX A – CONTAINMENT CORROSION ANALYSIS

EFFECT OF AGE-RELATED DEGRADATION ON RISK INFORMED/RISK IMPACT ASSESSMENT FOR EXTENDING CONTAINMENT TYPE A TEST INTERVAL

A.1. Purpose

The purpose of this calculation is to assess the effect of age-related degradation of the containment on the risk impact for extending the NMPS Integrated Leak Rate Test (ILRT) or Containment Type A test interval from ten to fifteen years for NMPS Unit 1.

A.2. Intended Use of Analysis Results

The results of this calculation will be used to indicate the sensitivity of the risk associated with the extension in the ILRT interval to potential age-related degradation of the containment shell to support obtaining NRC approval to extend the Integrated Leak Rate Test (ILRT) interval at NMPS from 10 years to 15 years. This calculation actually evaluates the impact of extending the interval from 3 years to 15 years.

A.3. Technical Approach

The present analysis shows the sensitivity of the results of the assessment of the risk impact of extending the Type A test interval for the NMPS to age-related liner corrosion.

The prior assessment included the increase in containment leakage for EPRI Containment Failure Class 3 leakage pathways that are not included in the Type B or Type C tests. These Classes (3a and 3b) include the potential for leakage due to flaws in the containment shell. The impact of increasing the ILRT Interval for these classes included the probability that a flaw would occur and be detected by the Type A test that was based on historical data. Since the historical data includes all known failure events, the resulting risk impact inherently includes that due to age-related degradation.

The present analysis is intended to provide additional assurance that age-related liner corrosion will not change the conclusions of the prior assessment. The methodology used for this analysis is similar to the assessments performed for Calvert Cliffs Nuclear Power Plant (CCNPP - Reference A1), Comanche Peak Steam Electric Station (CPSES - Reference A2), D. C. Cook Nuclear Plant (CNP - Reference A3) and St. Lucie (SL - Reference A4) in responses to requests for additional information from the NRC staff. The CCNPP, CPSES and CNP extension request submittals have been approved by the NRC.

The significantly lower potential for corrosion of freestanding steel shell containments, such as that at NMPS, is considered. This is due to the significantly smaller surface area susceptible to corrosion resulting from foreign material imbedded in concrete contacting the steel containment. Because of this, the analysis is carried out separately for those portions of the containment not in potential contact with foreign material and those portions in potential contact with the foreign material.

As in Reference A1, this calculation uses the following steps with NMPS values utilized where appropriate:

Step1 - Determine corrosion-related flaw likelihood.

Historical data will be used to determine the annual rate of corrosion flaws for the containment. The significantly lower potential for corrosion in the NMPS containment will be included.

Step 2 - Determine age-adjusted flaw likelihood.

The historical flaw likelihood will be assumed to double every 5 years. The cumulative likelihood of a flaw is then determined as a function of ILRT interval.

Step 3 - Determine the change in flaw likelihood for an increase in inspection interval.

The increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests is then determined from the results of Step 2.

Step 4 - Determine the likelihood of a breach in containment given a flaw.

For there to be a significant leak from the containment, the flaw must result in a gross breach of the containment. The likelihood of this occurring is a function of pressure and is evaluated at the NMPS ILRT pressure.

Step 5 - Determine the likelihood of failure to detect a flaw by visual inspection.

The likelihood that the visual inspection will fail to detect a flaw will be determined considering the portion of the containment that is uninspectable at NMPS as well as an inspection failure probability.

Step 6 - Determine the likelihood of non-detected containment leakage due to the increase in test interval.

The likelihood that the increase in test interval will result in a containment leak not detected by visual examination is determined as the product of the increase in flaw

likelihood due to the increased test interval (Step 3), the likelihood of a breach in containment (Step 4) and the visual inspection non-detection likelihood (Step 5). The results of the above for the two regions of the containment are then added to get the total increased likelihood of non-detected containment leakage due to age-related corrosion resulting from the increase in ILRT interval.

The result of Step 6 is then used, along with the results of the prior risk analysis in the body of this analysis to determine the increase in LERF as well as the increase in person-rem/year and conditional containment failure probability due to age-related liner corrosion.

A.4. Input Information

1. General methodology and generic results from the Calvert Cliffs assessment of age-related liner degradation (Reference A1).

2. The NMPS ILRT test pressure of 35.0 psig (Reference A5).

3. NMPS containment failure pressure of 100 psia based on Drywell rupture at temperatures less than 600 degrees F. (Section 3.4.1 Reference A6).

4. The surface area of the containment potentially in contact with foreign material either imbedded in the adjacent concrete or trapped in the areas of limited access is 121,667 ft^2. As depicted in Figure A-1, this is based on surface area of the Drywell and the Torus and calculated from drawings in reference [A5, A6]. Note that this is conservative since much of the containment surface area is not adjacent to concrete.

5. The number of containments, either free-standing steel shell or concrete with steel liners is 104, and the average area of steel potentially in contact with foreign material either imbedded in the adjacent concrete or trapped in the areas of limited access, is 61,900 ft^2 (Reference A10).

A.5. References

A1. "Calvert Cliffs Nuclear Power Plant Unit No. 1; Docket No. 50-317, Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," Constellation Nuclear letter to USNRC, March 27, 2002.

A2. "Comanche Peak Steam Electric Station (CPSES), Docket Nos. 50-445 and

50-446, Respond to Request for Additional Information Regarding License Amendment Request (LAR) 01-14 Revision to Technical Specification (TS) 5.5.16 Containment Leakage Rate Testing Program," TXU Energy letter to USNRC, June 12, 2002.

A3. "Donald C. Cook Nuclear Plants Units 1 and 2, Response to Nuclear Regulatory Commission Request for Additional Information Regarding the License Amendment Request for a One-time Extension of Integrated Leakage Rate Test Interval," Indiana Michigan Power Company, November 11. 2002.

A4. "St. Lucie Units 1 and 2, Docket Nos. 50-335 and 50-389, Proposed License Amendments, Request for Additional Information Response on Risk-Informed One Time Increases in Integrated Leak Rate Test Surveillance Interval," Florida Power & Light Company letter to USNRC, December 13, 2003.

A5. Nine Mile Point Nuclear Station Unit 1 Technical Specifications, October 2007

A6. E.T. Burns, ERIN Engineering and Research Inc. "Nine Mile Point Unit 1 Level 2 Probabilistic Risk Assessment (PRA)", December 2007

A7. "Containment Liner Through Wall Defect due to Corrosion," Licensing Event Report, Ler-NA2-99-02, North Anna Nuclear Power Plant Station Unit 2.

A8. "Brunswick Steam Electric Plant, Units 1 and 2, Dockets 50-325 and 50-324/License Nos. DPR=71 and DPR-62, Response tp Request for Additional Information Regarding Request for License Amendments B Frequency of Performance Based Leakage Rate Testing," CP&L letter to USNRC, February 5, 2002.

A9. "IE Information Notice No. 86-99; Degradation of Steel Containments." USNRC, December 8. 1986.

A10. E. R. Schmidt, "Calculation of Industry Average Containment Surface Area Subject to Age-Related Corrosion Due to Foreign Material," Analysis File 17547-0001-A4, Rev. 0, November 14, 2003.

A.6. Major Assumptions

1. There have been four instances of age-related corrosion that resulted in holes in steel containment liners or shells. Three were in concrete containments with steel liners and due to foreign material imbedded in the concrete in contact with the steel liner (Cook - Reference A3, North Anna - Reference A7 and Brunswick - Reference A8). The fourth was in a freestanding steel containment and occurred in an area where sand fills the gap between the steel shell and the surrounding concrete and was attributed to water accumulating in this sand (Oyster Creek - Reference A9).Based on this data, corrosion induced failures are only postulated to occur in the areas of the NMPS containment steel shell in contact with concrete

or other areas where foreign material may be trapped. For the other areas where the containment steel shell is not likely to be in contact with foreign material, the corrosion induced failure rate should be substantially lower and taken to be that based on no observations of corrosion induced failure of the containment steel shell in these regions.

2. The historical data of age-related corrosion leading to holes in the steel-containment has occurred primarily (3 out of 4 instances) for steel lined concrete containments. For these containments the surface area in contact with the concrete comprises essentially the entire surface area of the containment. As depicted in Figure A-1, it was conservatively assumed that the surface area of Drywell and the Torus [A7] is in contact with the concrete. This surface area is 121,667 square feet. Since the greater the surface area in contact with the concrete the greater the chance of foreign material being in contact with steel containment, and therefore the greater the chance of corrosion-induced flaws, the containment failure rate due to corrosion is assumed proportional to the surface area in contact with the concrete. The NMPS containment failure rate due to corrosion will be that for the industry times the ratio of the surface area at risk for NMPS to the average area at risk for the industry.

3. The visual inspection data is conservatively limited to 5.5 years reflecting the time from September 1996, when 10 CFR 50.55a started requiring visual inspection, through March 2002, the cutoff date for this analysis. Additional success data were not used to limit the aging impact of this corrosion issue, even though inspections were being performed prior to September 1996 (and after March 2002) and there is no evidence that liner corrosion issues were identified. (Step 1)

4. As in Reference A1, the containment flaw likelihood is assumed to double every 5 years. This is included to address the increased likelihood of corrosion due to aging. (Step 2)

5. The likelihood of a significant breach in the containment due to a corrosion induced localized flaw is a function of containment pressure. At low pressures, a breach is very unlikely. Near the nominal failure point, a breach is expected. As in Reference A1, anchor points of 0.1% chance of cracking near the flaw at 20 psia and 100% chance at the failure pressure (Drywell failure pressure of 100 psia for NMPS from Reference A6) are assumed with logarithmic interpolation between these two points. (Step 4)

6. In general, the likelihood of a breach in the lower head region of the containment, and this breach leading to a large release to the atmosphere, is less than that for the cylindrical portion of the containment. The assumption discussed in item 5 above is, however, conservatively applied to the lower head region of the containment, as well as to the cylindrical portions.

7. All non-detected containment overpressure leakage events are assumed to be large early releases.

8. The interval between ILRTs at the original frequency of 3 tests in 10 years is taken to be 3 years.

A.7. Identification of Computer Codes

None Used

A.8. Detailed Analysis

A.8.1 Step 1 - Determine the corrosion-related flaw likelihood.
As discussed in Assumptions 1, 2 and 3, the likelihood of through-wall defects due to corrosion for the areas of the containment potentially contacted by foreign materials is based on 4 data points in 5.5 years.

[4 failures*(121667ft^2 / 61,900ft^2 / (104 plants*5.5 years/plant) =1.37E-2 per year]

For the areas of the containment where foreign material is not likely to contact the containment the defect likelihood is taken to be that for no observed failures using a non-informative prior distribution.

Failure Frequency = [# of failures (0) + 1/2 / (Number of unit years (104*5.5)]
= 8.74E-4 per year.

A similar area-at-risk correction as above for the area in contact with concrete is not appropriate for the area where foreign material is not likely to contact the containment since the majority of the steel liner or shell for all plants has at least one side of the surface subject to this reduced corrosion (and none has been observed).

A.8.2 Step 2 - Determine age-adjusted liner flaw likelihood.

Reference A1 provides the impact of the assumption that the historical flaw likelihood will double every 5 years on the yearly, cumulative and average likelihood that an age-related flaw will occur. For a flaw likelihood of 5.2E-3 per year, the 15 year average flaw likelihood is 6.27E-3 per year for the cylinder/dome region. This result of Reference A1 is generic in nature, as it does not depend on any plant-specific inputs (i.e., except the assumed historical flaw likelihood).

For the present assumption of 4 historical failures in 104 plants, the 15 year average flaw likelihood is 263% (1.37E-2/5.2E-3 = 2.63 or 263%) of the above value (6.27E-3) or 1.65E-2 per year, and in accordance with Assumption 1, is applicable to the region of the containment potentially in contact with foreign material.

Similarly, for the region of the containment not potentially in contact with foreign material, the 15-year average flaw likelihood is 16.8%(8.74E-4/5.2E-3 = 0.168) of the above value (6.27E-3) or 1.05E-3 per year.

A.8.3 Step 3 - Determine the change in flaw likelihood for an increase in inspection interval. The increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is determined from the result of Step 2 in Reference A1 to be 8.7% for the cylinder/dome region based on assumed historical flaw likelihood and the resulting 6.27E-3 per year 15 year average flaw likelihood. This result of Reference A1 is generic in nature, as it does not depend on any plant specific, inputs except the assumed historical flaw likelihood.

For the present assumption of 4 historical failures in 104 plants, the increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is 263% (as in Step 2) of that given in Reference A1 (2.63*8.7%) or 22.88% and in accordance with Assumption 1 is applicable to only the region of the containment potentially in contact with foreign material.

Similarly, for the region of the containment not potentially in contact with foreign material, the increase in the likelihood of a flaw due to age-related corrosion over the increase in time interval between tests from 3 to 15 years is 16.8% (as in Step 2) of that given in Reference A1 or 1.46%.

A.8.4 Step 4 - Determine the likelihood of a breach in containment given a liner flaw. The likelihood of a breach in containment occurring is determined as a function of pressure as follows:

For a logarithmic interpolation on likelihood of breach

LOG (likelihood of breach) = m (pressure) + a

Where m = slope
a = intercept

The values of m and a are determined from solution of the two equations for the values of 0.1% at 20 psia and 100% of containment failure pressure at 100 psia (Reference A6).

Log 0.1 = m*20 + a

Log 100 = m*100 +a

Solving for "m",

m = (Log 100 - Log 0.1)/(100-20) = 0.0375

Solving for "a",

a = Log 0.1 - 0.0375*20 = -1.7500

The upper end of the range of NMPS ILRT pressure of 35.0 psig (Reference A5) gives the highest likelihood of breach.

At 49.7 psia (35+14.7) the above equation gives

Log (Likelihood of breach) = 0.0375*49.7-1.7500 = .11375

Likelihood of breach $=10^{0.11375}$ = 1.30%

In accordance with Reference A1, the above value is for the cylinder/dome portions of the containment. For this analysis, this value is also assumed to be applicable to the region of the containment potentially in contact with foreign material.

A.8.5 Step 5 - Determine the likelihood of failure to detect a flaw by visual inspection

A review of the geometry of the containment shell and the relative areas that are not inspectable and those in potential contact with foreign material, indicates that these two areas are essentially the same. Consequently, the portion of the containment not likely to be in contact with potential foreign material is 100% visually inspectable, while the portion that may be in contact with potential foreign material is not visually inspectable. A 10% failure rate for that portion of the containment that is visually inspectable is assumed.

A.8.6 Step 6 - Determine the likelihood of non-detected containment leakage due to the increase in test interval.

The likelihood of non-detected containment leakage in each region due to age-related corrosion of the liner considering the increase in ILRT interval is then given by:

The increased likelihood of an undetected flaw because of the increased ILRT interval (Step3)	*	The likelihood of a containment breach given a liner flaw (Step 4)	*	The likelihood that visual inspection will not detect the flaw (Step 5)

1.46% * 0.013*0.10 = 0.00190% for the regions not potentially contacted by foreign material

23.0% * 0.013*1.0 = 0.299% for the regions potentially contacted by foreign material

The total is then the sum of the values for the two regions or

Total likelihood of Non-Detected Containment Leakage = 0.0019% + 0.299% = 0.3007% for ILRT interval increase from 3 years to 15 years.

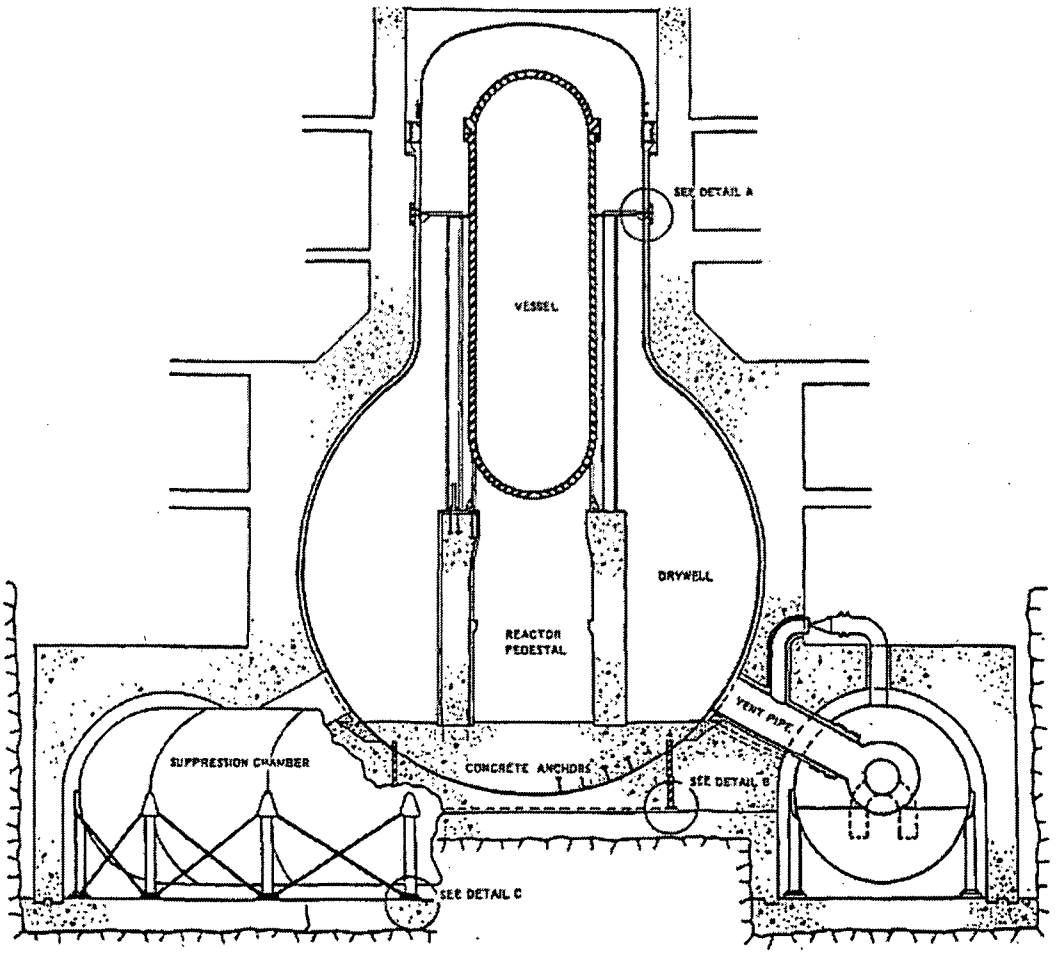

Figure A-28: NMPS CONTAINMENT AND TORUS AREA

APPENDIX B – EXTERNAL EVENT IMPACT

EFFECT OF EXTERNAL EVENTS ON RISK INFORMED/RISK IMPACT ASSESSMENT FOR EXTENDING CONTAINMENT TYPE A TEST INTERVAL

This appendix discusses the external events assessment in support of the NMPS ILRT interval extension risk assessment. In this assessment the percentage contribution of fire and seismic external events is taken from Reference B-6. High winds, flooding, transportation and nearby industrial facility accidents were addressed in the IPEEE assessment [B-5].

B.1. High Winds, Transportation, and Nearby Industrial Facility Accidents

There are no initiating events such as high winds, transportation accidents, or nearby industrial facility accidents that are judged to require inclusion in the PRA model. The evaluation that resulted in this conclusion is described in the NMP1 IPEEE [B-5]. Therefore, these external events are not evaluated further in this calculation and are expected to have an insignificant impact on the results of this calculation.

B.2. Fire

Reference B-6 utilized the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology. The NMPS 2003 PRA model quantified the percentage contribution of internal fires to be 61% to CDF and 9% to LERF. The Nine Mile Nuclear Station 2003 Probabilistic Risk Assessment [B-6] gives 2.4E-05 as the total CDF value and 1.7E-06 as the total LERF value. Total contributions to CDF and LERF from fire external events are calculated to be 1.46E-05 and 1.53E-07 respectively.

B.3. Seismic

The Nine Mile Nuclear Station 2003 Probabilistic Risk Assessment [B-6] gives seismic percent contributions to both LERF (42%) and CDF (5%) values. These percentages were used in the current assessment to calculate the total contribution to LERF and CDF. The Nine Mile Nuclear Station 2003 Probabilistic Risk Assessment [B-6] gives 2.4E-05 as the total CDF value and 1.7E-06 as the total LERF value. Total contributions to CDF and LERF from seismic external events are calculated to be 1.20E-06 and 7.14E-07 respectively.

B.4. Impact of External Events on LERF and Comparison to RG 1.174

Based on the previous discussion in Sections B.1 through B.3, the total NMPS Unit 1 external event initiated CDF is approximately 1.46E-05/yr (internal fires) + 1.20E-06/yr (seismic) = 1.58E-05/yr. The total LERF is approximately 1.53E-07/yr (fire) + 7.14E-07/yr (seismic) = 8.67E-07/yr.

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 Probabilistic Safety Assessment, 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-29
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	2.28E-09	7.62E-09	1.14E-08	5.34E-09	9.15E-09	3.81E-09
Combined (Internal+External)	4.58E-08	1.53E-07	2.30E-07	1.07E-07	**1.84E-07**	7.66E-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 1.84E-07 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 0.35% (1.14E-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., 1.84E-07), the total LERF for the requested change is 1.88E-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-4 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 1.87E-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-30 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	2.28E-09	7.62E-09	1.14E-08	5.34E-09	9.15E-09	3.81E-09
Combined (Internal+External)	4.66E-09	1.56E-08	2.34E-08	1.09E-08	**1.87E-08**	7.79E-09

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.

B-3. Level II PRA Rev 0, December, 2007, "LE Notebook".

B-4. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis".

B-5. NMP1 Individual Plant Examination of External Events (IPEEE), August 1996.

B-6. NMPS Unit 1 Probabilistic Risk Assessment (PRA) Rev. 2, September 2003.