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NL-17-137

December 8, 2017

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

SUBJECT: License Amendment Request (LAR) for One-Time Extension of the Containment Type A Leak Rate Testing Frequency from 15 to 16 Years
Indian Point Unit No. 3
Docket No. 50-286
License No. DPR-64

REFERENCES:

1) Entergy Letter, "Amendment to License Renewal Application – Reflecting Shortened License Renewal Terms for Units 2 and 3," dated February 8, 2017 (NL-17-019) (ML17044A005)

2) Entergy Letter, "Proposed License Amendment Regarding Extending the Containment Type A Leak Rate Testing Frequency to 15 Years," dated February 4, 2014 (NL-14-014)

3) NRC Letter, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment re: Extension of the Type A Containment Integrated Leak Rate Test Frequency From 10 to 15 Years (TAC No. MF3426)," dated March 13, 2013 (ML15028A308)

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (Entergy) is submitting a request for an amendment to the Technical Specifications (TS) for the Indian Point Nuclear Generating Unit No. 3 (IP3). The proposed license amendment request (LAR) would allow for a one-time extension to the 15-year frequency of the IP3 containment leakage rate test (i.e., Integrated Leakage Rate Test (ILRT) or Type A test). This test is required by TS 5.5.15 "Containment Leakage Rate Testing Program." The proposed change would permit the existing ILRT frequency to be extended from 15 years to 16 years.

Because IP3 operates on a twenty-four (24) month refueling cycle, the next ILRT is required to be performed approximately one year prior to the 15th year anniversary of the completion of the last ILRT (March 2005). If granted, this revision would extend the period from 15 years to 16 years between tests. In terms of refueling outages (RFOs), this extension would allow the performance of the next ILRT to be scheduled during the Spring 2021 (3R21) RFO. As discussed in Reference 1, Entergy has entered into a settlement agreement with the Attorney General of the State of New York, and Riverkeeper, Inc. to shut down IP3 by April 30, 2021, subject to operating extension through, but not beyond 2025.

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Attachment 1 contains a description of the proposed changes, the supporting technical analyses, and the significant hazards considerations determination. Attachment 2 provides the existing TS pages marked-up to show the proposed changes. Attachment 3 provides the retyped TS pages.

A copy of this application and the associated attachments are being submitted to the designated New York State official in accordance with 10 CFR 50.91.

Entergy requests approval of the proposed amendment in one calendar year and an allowance of 30 days for implementation. There are no new commitments being made in this submittal. If you have any questions or require additional information, please contact Mr. Robert W. Walpole, Regulatory Assurance Manager at (914) 254-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 12/8, 2017.

Sincerely,

AJV/rl

Attachments: 1. Description and Assessment of Technical Specification Changes
2. Proposed Technical Specification Changes (Marked-Up)
3. Revised Technical Specification Pages (Clean)

Enclosure: 1. Risk Impact Assessment of Extending the Indian Point 3 ILRT Interval

cc: with Attachment(s)

Mr. Daniel H. Dorman, Regional Administrator, NRC Region I
Mr. Richard V. Guzman, Senior Project Manager, NRC NRR DORL
Mr. William Burton, Senior Project Manager, NRC DLR
Ms. Bridget Frymire, New York State Department of Public Service
Ms. Alicia Barton, President and CEO NYSERDA
NRC Resident Inspector's Office

ATTACHMENT 1 TO NL-17-137

Description and Assessment of Technical Specification Changes

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

EVALUATION OF THE PROPOSED CHANGE

Subject: License Amendment Request (LAR) for One-Time Extension of the Containment Type A Leak Rate Testing Frequency from 15 to 16 Years

1. INTRODUCTION

Pursuant to 10 CFR 50.90, Entergy Nuclear Operation, Inc. (Entergy) requests an amendment to Operating License (OL) No. DPR-64 for the Indian Point Nuclear Generating Unit 3 (IP3) to allow for a one-time extension to the 15-year frequency of the IP3 containment leakage rate test (i.e., Integrated Leakage Rate Test (ILRT) or Type A test). Specifically, the proposed change is a request to revise TS 5.5.15 "Containment Leakage Rate Testing Program" to allow a one-time extension to the 15-year frequency of the IP3 containment leakage rate test. The proposed change would permit the existing ILRT frequency to be extended from 15 years to 16 years.

Because IP3 operates on a twenty-four month refueling cycle, the next ILRT is required to be performed approximately one year prior to the 15th year anniversary of the completion of the last ILRT (March 2005). If granted, this revision would extend the period from 15 years to 16 years between tests. In terms of refueling outages (RFOs), this extension would allow the performance of the next ILRT to be scheduled during the Spring 2021 (3R21) RFO. As discussed in Reference 1, Entergy has entered into a settlement agreement with the Attorney General of the State of New York, and Riverkeeper, Inc. to shut down IP3 by April 30, 2021, subject to operating extension through, but not beyond 2025.

2. PROPOSED CHANGE

Current Containment Leakage Rate Testing Program

Current TS 5.5.15 specifies, "A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008, as modified by the following exception:

> ANS 56.8-2002, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

The maximum allowable primary containment leakage rate, L_a, at a minimum test pressure equal to P_a, shall be 0.1% of primary containment air weight per day. Pa is the peak calculated containment internal pressure related to the design basis accident.

Leakage acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

    1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

2) For each door, leakage rate is ≤ 0.01 La when pressurized to ≥ P_a,

c. Isolation Valve Seal Water System leakage rate acceptance criterion is ≤ 14,700 cc/hr at ≥ 1.1 P_a.

d. Acceptance criterion for leakage into containment from isolation valves sealed with the service water system is ≤ 0.36 gpm per fan cooler unit when pressurized at ≥ 1.1 P_a. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.38 psig. The containment design pressure is 47 psig.

The maximum allowable primary containment leakage rate, L_a, at Pa, shall be 0.1% of primary containment air weight per day."

TS Change Description

The proposed changes to TS 5.5.15, Containment Leakage Rate Testing Program, will add a new exception to allow for the performance of the next Type A test no later than the Spring 2021 (3R21) RFO. The proposed change, shown in bold italicized text, will revise TS 5.5.15, as follows, to state:

"A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008, as modified by the following exception***s***:

a. ANS 56.8-2002, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

b. The next Type A test to be performed after the March 2005 Type A test shall be performed no later than the plant restart after the Spring 2021 (3R21) RFO.

The maximum allowable primary containment leakage rate, L_a, at a minimum test pressure equal to P_a, shall be 0.1% of primary containment air weight per day. P_a is the peak calculated containment internal pressure related to the design basis accident.

Leakage acceptance criteria are:

a. Containment leakage rate acceptance criterion is ≤ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a,

   2) For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ P_a,

c. Isolation Valve Seal Water System leakage rate acceptance criterion is ≤ 14,700 cc/hr at ≥ 1.1 P_a.

d. Acceptance criterion for leakage into containment from isolation valves sealed with the service water system is ≤ 0.36 gpm per fan cooler unit when pressurized at ≥ 1.1 P_a. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.38 psig. The containment design pressure is 47 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day."

3. BACKGROUND

Chronology of 10 CFR 50 Appendix J Testing Requirements

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per 10 years. The revised Type A 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 performance leakage was less than the normal containment leakage of 1.0La (allowable leakage). The basis for a 10 year test interval is provided in Section 11.0 of NEI 94-01, Revision 0 (Reference 4), 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," (Reference 5) provides the technical basis to support

rulemaking 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 rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

The NRC report on performance-based leak testing, NUREG-1493, analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Because ILRTs represent substantial resource expenditures, it is desirable to show that extending the ILRT interval one time will not lead to a substantial increase in risk from containment isolation failures to support a reduction in the test frequency for IP3.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In October 2008, EPRI 1018243 was issued to develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily NRC Regulatory Guide 1.174. This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and containment conditional failure probability (CCFP), whereas EPRI TR-104285 considered only the change in risk based on the change in population dose.

Current IP3 10 CFR 50 Appendix J Option B Requirements

Title 10 CFR Part 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."

By letter dated January 13, 1997, Indian Point Unit 3 submitted a TS change that was supplemented by letters dated March 24, 1997, May 13, 1997 and May 23, 1997 requesting the implementation of 10 CFR 50, Appendix J, Option B. The NRC approved this request as Amendment No. 174 issued in NRC letter of June 17, 1997. The NRC noted that the proposed TS changes were in compliance with the requirements of Option B, and were consistent with the guidance in RG 1.163. With the approval of the amendment, IP3 transitioned to a performance-based 10 year frequency for the Type A tests.

In a letter dated September 6, 2000 that was supplemented by letters dated January 18, 2001 and April 2, 2001, Entergy submitted a license amendment request to extend the ILRT interval one time from 10 years to 15 years. This one-time extension was approved by the NRC, as Amendment No. 206 on April 17, 2001.

By letter dated August 31, 2007, NEI submitted NEI 94-01, Revision 2, and EPRI report No. 1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," to the NRC Staff for review. NEI 94-01, Revision 2, describes an approach for implementing the optional performance-based requirements of Option B,

which includes provisions for extending Type A intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163. It delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. This method uses industry performance data, plant-specific performance data, and risk insights in determining the appropriate testing frequency. NEI 94-01, Revision 2, also discusses the performance factors that licensees must consider in determining test intervals. The NRC final Safety Evaluation (SE) issued by letter dated June 25, 2008, documents the evaluation and acceptance of NEI 94-01, Revision 2, subject to the specific limitations and conditions listed in Section 4.1 of the SE. The accepted version of NEI 94-01 was subsequently issued as Revision 2-A dated October 2008. The NEI guideline does not address how to perform the tests because these details are included in referenced industry documents (e.g., American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002).

EPRI Report No. 1009325, Revision 2, provides a risk impact assessment for optimized ILRT intervals of up to 15 years, using current industry performance data and risk-informed guidance, primarily Revision 1 of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The NRC's final SE issued by letter dated June 25, 2008, documents the evaluation and acceptance of EPRI Report No. 1009325, Revision 2, subject to the specific limitations and conditions listed in Section 4.2 of the SE. An accepted version of EPRI Report No. 1009325 has subsequently been issued as Revision 2-A (also identified as Technical Report TR-1018243) dated October 2008.

By letter dated February 4, 2014 (Reference 2), as supplemented by letter dated December 9, 2014, Entergy submitted an amendment request to allow a permanent extension of the Type A primary containment integrated leak rate test frequency from once every 10 years to once every 15 years. On March 13, 2015, the NRC approved Amendment No. 256 for IP3 authorizing the implementation of 10 CFR Part 50, Appendix J, Option B for Types A, B and C tests (Reference 3).

Current Technical Specification 5.5.15 requires that a program be established to comply with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The program is required to be in accordance with the guidelines contained in the Safety Evaluation issued by the Office of Nuclear Reactor Regulation dated March 13, 2015 (Reference 3).

As stated previously, compliance with the current TS would require the Type A test to be performed approximately one year prior to the 15th year anniversary of the completion of the last ILRT (March 2005). The proposed change would defer the Type A ILRT by approximately 12 months. If granted, this revision would extend the period until the Spring 2021 outage at which time Entergy will cease IP3 power operation in accordance with Reference 1.

4. TECHNICAL ANALYSIS

The allowable frequency for the Type A ILRT is based, in part, upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leak-Test Program"

(Reference 5). NUREG-1493 made the following observations with regard to changing the test frequency:

- Reducing the Type A ILRT frequency to once per twenty years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because the Type A ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A ILRTs have only been marginally above the existing requirements. Given the insensitivity of risk to the containment leakage rate, the small fraction of leakage detected solely by Type A ILRTs, increasing the interval between Type A ILRTs has minimal impact on public risk.
- While Type B and C tests identify the vast majority (i.e., greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

Enclosure 1 provides an assessment of the risk impact associated with implementing this one-time extension of the IP3 containment Type A integrated leak rate test (ILRT) interval from ten years to sixteen years. This assessment represents an update of the analysis previously developed to support the extension of the ILRT interval from ten years to 15 years for both IP2 and IP3 (see Reference 2). The previous calculation was updated solely to reflect the change from 15 to 16 years (unless where otherwise noted) to reflect IP3 specific assumptions.

Earlier ILRT frequency extension submittals used the methodology contained in EPRI TR-104285 to perform the risk assessment. In October 2008, EPRI 1018243 was issued to develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily NRC Regulatory Guide 1.174. This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and containment conditional failure probability (CCFP), whereas EPRI TR-104285 considers only the change in risk based on the change in population dose. This ILRT interval extension risk assessment for IP3 employs the EPRI 1018243 methodology, with the affected system, structure, or component (SSC) being the primary containment boundary.

The risk assessment follows the guidelines from NEI 94-01, Revision 3-A (Reference 6), the methodology outlined in EPRI TR-104285 (Reference 7), the EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (Reference 8), the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 (Reference 9), and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of a steel liner going undetected during the extended test interval (Reference 10). The format of Enclosure 1 is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the October 2008 EPRI final report (Reference 8).

Acceptance Criteria

The acceptance guidelines in RG 1.174 were used to assess 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 1.0E-06 per reactor year and increases in large early release frequency (LERF) less than 1.0E-07 per reactor year. A review of the IP3 ECCS NPSH calculations made in support of the GSI-191 effort confirms that containment overpressure is not required to obtain adequate NPSH. This confirms that the CDF is not impacted by the proposed change for IP3. Therefore, since the Type A test does not impact CDF for IP3, the relevant criterion is the change in LERF. RG 1.174 also defines small changes in LERF as below 1.0E-06 per reactor year, provided that the total LERF from all contributors (including external events) can be reasonably shown to be less than 1.0E-05 per reactor year. RG 1.174 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 (CCFP) was calculated to help ensure that the defense-in-depth philosophy was maintained.

With regard to population dose, examinations of NUREG-1493 and Safety Evaluation Reports (SERs) for one-time interval extension (summarized in Appendix G of EPRI 1018243) indicate a range of incremental increases in population dose that have been accepted by the NRC. The range of incremental population dose increases is from ≤0.01 to 0.2 person-rem/yr and 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (Figure 7-2 of NUREG-1493) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, the NRC SER on this issue [7] defines a small increase in population dose as an increase of ≤ 1.0 person-rem per year, or ≤1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This definition was adopted for the IP3 analysis.

Summary of Plant-Specific Risk Assessment Results

The findings of the IP3 risk assessment confirm the general findings of the previous studies that the risk impact associated with extending the ILRT interval one-time to one in 16 years is small. The IP3 plant-specific results for extending the ILRT interval to 16 years, taken from Enclosure 1 Section 7.0 Conclusions, are summarized below.

- Reg. Guide 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 1.0E-06/yr and increases in LERF below 1.0E-07/yr. "Small" changes in risk are defined as increases in CDF below 1.0E-05/yr and increases in LERF below 1.0E-06/yr. Since the ILRT extension was demonstrated to have no impact on CDF for IP3, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval for the base case with corrosion included for IP3 is 1.37E-07/yr (see Enclosure 1, Table 5.6-1), which is within the small change region of the acceptance guidelines in Reg. Guide 1.174. In using the EPRI Expert Elicitation methodology, the change is

estimated as 1.45E-08/yr (see Enclosure 1, Table 6.2-2), which is within the very small change region of the acceptance guidelines in Reg. Guide 1.174.

- The change in dose risk for changing the Type A test frequency from three-per-ten years to once-per-sixteen-years, measured as an increase to the total integrated dose risk for all internal events accident, it is 0.813 person-rem/yr (1.00%) using the EPRI guidance with the base case corrosion case (Enclosure 1, Table 5.6-1). The change in dose risk drops to 1.54E-01 person-rem/yr when using the EPRI Expert Elicitation methodology (Enclosure 1, Table 6.2-2). The values calculated per the EPRI guidance are lower than the acceptance criteria of ≤1.0 person-rem/yr or <1.0% person-rem/yr defined in Enclosure 1, Section 1.3.

- The increase in the conditional containment failure frequency from the three in ten year interval to one in sixteen years including corrosion effects using the EPRI guidance (see Enclosure 1, Section 5.5) is 0.92% for IP3. This value drops to 0.10% using the EPRI Expert Elicitation methodology (see Enclosure 1, Table 6.2-2). This is below the acceptance criteria of less than 1.5% defined in Section 1.3.

- To determine the potential impact from external events, a bounding assessment from the risk associated with external events utilizing information from the IP3 IPEEE similar to the approach used in the License Renewal SAMA analysis was performed. As shown in Table 5.7-2, the total increase in LERF due to internal events and the bounding external events assessment is 6.18E-07/yr. This value is in Region II of the Reg. Guide 1.174 acceptance guidelines.

- As shown in Enclosure 1, Table 5.7-4, the same bounding analysis indicates that the total LERF from both internal and external risks is 6.39E-06/yr for IP3, which is less than the Reg. Guide 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II (small change in risk).

- Finally, since the external events assessment led to a challenge of the dose risk acceptance criteria (i.e. greater than 1.0 person-rem/yr and equal to a 1.00% change in the total dose risk), an alternative detailed bounding external events assessment was also performed to demonstrate that the acceptance criterion requiring the change in dose risk to be less than 1.00% of the total dose risk could be met. In this case, as shown in Enclosure 1, Table 5.7-7, the total change in LERF from both internal and external events was 6.46E-7/yr, the change in personrem/ yr was 3.84/yr representing 0.69% of the total, and the change in the CCFP was 0.97%. All of these calculated changes meet the acceptance criteria. As shown in Enclosure 1, Table 5.7-8, this assessment indicates that the total LERF from both internal and external risks is 2.98E-06/yr, which is less than the Reg. Guide 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II (small change in risk).

- Including age-adjusted steel liner corrosion effects in the ILRT assessment was demonstrated to be a small contributor to the impact of extending the ILRT interval for IP3.
- The risk assessment for the one-time extension to 16 years was performed measuring the delta risk compared to the original 3-in-10 year ILRT test interval. The acceptance criteria were demonstrated to be met with high confidence such that the cumulative impact of the proposed change is also acceptable. The actual delta-risk compared to the currently approved 1-in-15 year test interval is minimal and would fall in Region III of the RG 1.174 acceptance guidelines for "very small" changes in risk.

Therefore, allowing a one-time increase of the ILRT interval to a one-in-sixteen year frequency is not considered to be significant since it represents only a small change in the IP3 risk profile.

Integrated Leakage Rate Testing (ILRT) History

Past IP3 ILRT results have confirmed that the containment is acceptable with respect to the design criterion of 0.1% leakage of containment air weight at the design basis loss of coolant accident pressure (La). Since the last two Type A "as found" tests for IP3 had "as found" test results of less than 1.0La, a test frequency of 15 years in accordance with NEI 94-01 Revision 2A would be acceptable. The last two tests were:

Last ILRT in March 2005

The measured containment leak rate (Ltm) at the test pressure of 60.61 psia was 0.0565 % containment air weight / day with a 95% confidence level.

Prior ILRT in December 1990

The measured containment leak rate (Ltm) at the test pressure of 59.49 psia was 0.032 % containment air weight / day with a 95% confidence level.

For background, the prior three Type A tests had the following results:

Date	As found Leakage (% Containment weight per day)	Test Pressure (psia)
July 27, 1987	0.34(1)	59.89
August 4, 1982	0.034	60.00
August 2, 1978	0.14(2)	60.00

Notes: (1) There was a leak through the reactor coolant pump seal water return valve MOV- 222 on penetration R, Line 17. Valve 221A was closed to isolate MOV-222 and the leakage returned to normal.

(2) There was a leak in the #33 and #34 containment fan cooler service water supply and return lines inside containment.

Type B and C Testing

The IP3 Appendix J, Type B and Type C testing program requires testing of the components required by 10 CFR 50, Appendix J, Option B. Technical Specification Amendment 174, dated June 17, 1997, approved the adoption of 10 CFR 50, Appendix J, Option B performance based testing requirements for containment leakage testing. The minimum pathway combined Type B and Type C leakage from the March 2005 outage, when the last Type A test was performed, is provided below. The subsequent combined as found Type B and Type C test values during each successive outage since the last Type A test are also provided below. The data is provided in percentage of leakage allowed (0.6La).

Table 4-1

Date	**As-Found Leakage (cc/min)**	**0.6La (cc/min)**	**Percent ((As-Found/La) x 100)**	**Percent ((As-Found/0.6La)) x 100)**
April 2005	41585	119689	0.208	0.347
April 2007	30352	119689	0.152	0.254
April 2009	44621	119689	0.224	0.373
April 2011	51878	119689	0.260	0.433
March 2013	36669	119689	0.184	0.306
April 2015	56557	119689	0.283	0.473
June 2017	59327	119689	0.297	0.496

Based upon the information shown in Tables 4-1 and 4-2, the largest as-found and as-left conditions are within the acceptance criterion associated with the 15 year ILRT.

Table 4-2 provides a listing of the containment penetrations subject to Type B and C testing, the test frequency, the last test date and the next test date, and the as-left leakage. Notes are provided for test failures.

Table 4-2

Penetration	Description	Type	Test Frequency (Months)	Last Test Date	Next Test Date	"as-left" Leakage (cc/min)
	Fuel Transfer Tube	B	30	5/13/17	5/2019	0
	Equipment Hatch Seal	B	30	6/20/17	5/2019	9312.5
	Personnel Airlock – 80 foot	B	30	1/27/16	1/2018	4053
	Personnel Airlock – 95 foot	B	30	1/27/16	1/2018	2797
	WCCPP Zone 2 – Racks 10, 11	B(1)	36	2/12/16	2/2018	34513
	WCCPP Zone 2 – Racks 12, 13	B(1)	36	2/12/16	2/2018	200
Y	Pressurizer relief tank N2 supply tank RCS – Valve RC-518	C	30	3/17/17	3/2019	888
Y	Pressurizer relief tank N2 supply tank RCS – Valve RC-550	C	60	3/17/17	3/2021	0
GG	Containment spray headers – Valves SI-867A, SI-878A	C	60	3/17/17	3/2021	111.75
P	Containment spray headers – Valves SI-867B, SI-878B	C	60	3/17/17	3/2021	154
RR	Accumulator N2 supply – Valve NNE-1610	C	30	3/23/17	3/2019	41.25
RR	Accumulator N2 supply – Valve NNE-863	C	60	3/23/17	3/2021	13.5
V	Primary system vent and N2 supply – Valve WD-1610	C	60	3/31/17	3/2021	30
V	Primary system vent and N2 supply – Valve WD-1616	C	30	3/31/17	3/2019	1200
RR	Containment Air Sample In (Rad) – Valves PCV-1234, PCV-1235	C	60	3/28/17	3/2021	33
RR	Containment Air Sample In (Rad) – Valves PCV-1236, PCV-1237	C	60	3/28/17	3/2021	0.5

R	Air Ejector Discharge to Containment – Valves CA-1229, CA-1230	C	30	3/7/17	3/2019	5.75
EE	Vent Purge Supply Duct - VS-1170, VS-1171	C(1)	30	6/19/17	3/2019	5683.75
FF	Vent Purge Exhaust Duct – Valves VS-1172, VS-1173	C(1)	30	4/25/17	3/2019	4722.5
PP	Cont Pressure Relief Vent – Valves VS-1190, VS-1191	C(1)	30	3/15/17	3/2019	540
PP	Cont Pressure Relief Vent – Valve VS-1192	C(1)	30	3/15/17	3/2019	2955
R, TT, LL	Post Accident Sample system supply lines Valves SP-506, SP-507, SP-509	C	30	3/14/17	3/2019	500
O	Post Accident Sample system supply lines Valves SP-510, SP-511	C	30	3/14/17	3/2019	878.8
Z, O ,Z, R	Post Accident Sample system supply lines Valves SP-512, SP-513, SP-514, SP-515, SP-516	C	30	3/17/17	3/2019	471
Y	Instrument air (post accident vent supply) Valve IA-39	C	30	3/16/17	3/2019	0
Y	Instrument air (post accident vent supply) Valve IA-1228	C	60	3/16/17	3/2021	23
	Equipment access – Valves CB-5, CB-6 Equipment access – Valve CB-7	C	30	3/7/17	3/2019	165
	Personnel airlock – Valves CB-1, CB-2	C	30	3/7/17	3/2019	208
	Personnel airlock – Valve CB-3	C	60	3/7/17	3/2021	166

Note 1 – The Weld Channel and Penetration Pressurization System is controlled by Technical Specification 3.6.10 which requires a surveillance every 36 months to demonstrate the leakage rate for the WCPPS is ≤ 0.2% of the containment free volume per day when pressurized ≥ 43 psi above containment pressure. The leakage rate from this test is included in the containment leakage summary for the electrical penetrations and the piping penetrations. Zone 1 (Racks 12 & 13) and Zone 2 (Racks 10 & 11) test the electrical and mechanical penetrations as well as other components already Type C tested (e.g., Containment Purge supply and exhaust, equipment hatch, Containment Pressure Relief, personnel hatch) whose leakage was removed from the total since these components are already accounted for in the Rack testing.

Containment Inspections

The following discussion provides a summary of the IP3 IWE examination results of the containment metal liner completed during refueling outages 3R17 (Spring 2013) and 3R18 (Spring 2015) and the IWL examination results for the containment concrete visual inspections completed in 2015.

As discussed in the Reference 2 submittal, IP3 is committed to the 2001 Edition of the ASME Boiler and Pressure Vessel Code, Section XI with the 2002 and 2003 Addenda.

In accordance with the 10 CFR 50.55a(b)(2)(ix)(E) a general visual examination of the IP3 containment surfaces must be performed during each Section XI ISI period of the ten-year interval. IP3 completed this examination for the first period during refueling outage 3R17 (Spring 2013). In Reference 2, a summary of reported visual examinations was provided including references to the NDE examination reports. All reported visual observations were considered minor with no areas of suspect damage or deterioration which would impact the structural integrity or leak tightness of the containment liner. For the second inspection period, these general visual examinations were performed during refueling outage 3R18 (Spring 2015). 100% of the accessible portions of the containment building liner, containment surfaces pressure retaining bolting, and moisture barrier interface were examined in accordance with Section XI, Examination Category E-A, E1.11, E1.11b, and E1.30, respectively. The inspection findings for the containment liner and containment surfaces pressure retaining bolting were documented in reports VT-15-001, VT-15-002, VT-15-003, VT-15-004, VT-15-005, and VT-15-085, and the results for the containment liner were similar to the previous inspection results performed in 2013. During the inspection of the equipment hatch bolts, two bolts were found to have minimum thread engagement issues; however, they were evaluated by Engineering to be acceptable. Portions of the moisture barrier are covered by stainless steel insulation and are inaccessible at Ring Plates R-1 through R-36. In addition, the stainless steel insulation is sealed to the containment floor making the moisture barrier inaccessible for inspection. Visual inspections of the accessible portions of the moisture barrier did not reveal any signs of leakage and the findings were similar to the previous inspection results performed in 2013.

As part of the license renewal application, a one-time inspection of the IP3 containment liner was performed to determine the condition of the liner at the junction of the concrete slab at elevation 46 foot. The inspection consisted of spot observations of the liner through four approximately 6" X 6" windows cut through the insulation. Where the insulation was removed corrosion was identified at R1-Plate 7, between columns 3 and 4; R1-Plate 14, between columns 17 and 18; R1-Plate 21, between columns 14 and 15; and R1-Plate 31, between columns 9 and 10. The observations revealed discoloration and exfoliation of the liner from corrosion although no moisture was noted. Subsequent UT inspections were performed at location R1-Plate 31 between columns 9 and 10, and all points were greater than the minimum required thickness of 0.65625 inches. The observed level of corrosion was as a result of Service Water leakage from the containment fan cooling unit piping and aggravated by the process of Corrosion Under Insulation (CUI). This prevented early detection of the corrosion of the containment building liner. The level of corrosion was minimal and did not compromise the integrity of the liner. The absence of any corrosion near the upper portions of the exposed

sections of the liner as well as the history of excessive leakage from the fan cooler piping located on the 68 foot level onto the 46 foot elevation indicates that this was the most likely cause of corrosion. Based on the observed condition of the liner at the window openings, and the removal of the likely cause of the corrosion by past repair/replacement of the fan cooler piping, the condition of the liner is acceptable. This condition was documented in the Corrective Action Program as Condition Report CR-IP3-2015-01888. The results can be found in Entergy Engineering Report IP-RPT-15-00010, "One Time Inspection of IP3 Containment Liner to Satisfy License Renewal Commitment".

In accordance with the ASME Boiler and Pressure Code, Section IWL, 2001 Edition, 2003 Addenda and as modified by 10 CFR 50.55a, exterior inspection of the IP3 containment building was performed in 2015. Because the previous inspection that was performed in 2009 had identified several anomalies, those areas were re-examined during the 2015 inspection to determine if further degradation has occurred. For the most part, the previous findings had remained unchanged. Inspection findings that had further degradation, such as spall areas increasing, were evaluated and deemed acceptable. Several general typical concrete conditions were identified throughout the structure such as minor cracks and pattern cracking, numerous bugholes, leaching, scaling, and spalling. These findings did not impact the structural integrity of the containment. A sample comparison of areas with multiple cracks was made using photos from past inspections and no changes were identified. Therefore, the cracking on containment was concluded to be inactive. The results of the 2015 inspection are documented in Entergy Engineering Report IP-RPT-15-00063, "IP3 ASME Section XI, IWL Concrete Containment Inspection for 2015," which was provided to the NRC by Reference 11.

In summary, the containment liner areas which had experienced some degradation were identified, analyzed and repaired as necessary to ensure an acceptable containment barrier exists.

5. REGULATORY ANALYSIS

Applicable Regulatory Requirements / Criteria

The NRC Order of February 11, 1980 required an evaluation of the degree of compliance with the GDC at the time. This section discusses continued compliance with certain of those criteria.

The plant will continue to meet Criterion 1 of 10 CFR 50.36 which says "Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems and components

important to safety shall be maintained by or under the control of the nuclear power plant licensee throughout the life of the unit" and Criterion 3 which says "Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed."

The one-time extension of the frequency of the ILRT for the containment will not affect the design, fabrication, or construction of the containment structure and the design will continue to account for the effects of natural phenomena. The ILRT of the containment will continue to be done in accordance with 10 CFR 50 Appendix J using 10 CFR 50 Appendix B quality standards. The frequency of the ILRT is being changed in accordance with standards reviewed and approved as compliant with Appendix J. Therefore, there will be no instances where the applicable regulatory criteria are not met.

Significant Hazards Considerations

Entergy has evaluated the safety significance of the proposed change to the IP3 TS which revise IP3 TS 5.5.15, "Containment Leakage Rate Testing Program," to allow a one-time only extension to the frequency of Type A testing based upon performance criteria. The proposed changes have been evaluated according to the criteria of 10 CFR 50.92, "Issuance of Amendment". Entergy has determined that the subject changes do not involve a Significant Hazards Consideration, 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 amendment involves changes to the IP3 containment leakage rate testing program. The proposed amendment 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 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 amendment.

   The proposed amendment adopts the NRC accepted guidelines of NEI 94-01, Revision 3-A, for development of the IP3 performance-based testing program for the Type A testing. Implementation of these guidelines continues to provide adequate assurance that during design basis accidents, the primary containment and its

components would limit leakage rates to less than the values assumed in the plant safety analyses. The potential consequences of extending the ILRT interval one-time to 16 years have been evaluated by analyzing the resulting changes in risk. The increase in risk in terms of person-rem per year within 50 miles resulting from design basis accidents was estimated to be acceptably small and determined to be within the guidelines published in the NRC Final Safety Evaluation for NEI Topical Report (TR) 94-01, Revision 3-A. 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. Entergy has determined that the increase in conditional containment failure probability due to the proposed change would be very small. Therefore, it is concluded that the proposed amendment does not significantly increase the consequences of an accident previously evaluated. Therefore, 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 amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for the establishment of a one-time only 16-year interval for the performance of the containment ILRT. 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 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 to 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 previously evaluated.

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

   Response: No.

   The proposed amendment adopts the NRC-accepted guidelines of NEI 94-01, Revision 3-A, for the establishment of a one-time only 16-year interval for the performance of the containment ILRT. This amendment 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 containment leakage rate testing program, as defined in the TS, ensure that the degree of primary containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained. The overall containment leakage rate limit specified by the TS is maintained, and the Type A, Type B, and Type C containment leakage tests would be performed at the frequencies established in accordance with the NRC-accepted guidelines of NEI 94-01, Revision 3-A. Containment inspections performed in accordance with other plant programs serve to provide a high degree of assurance that the containment would not degrade in a manner that is not detectable by an ILRT. A risk assessment using the current IP3 PSA model concluded that extending

the ILRT test interval one-time from 15 years to 16 years results in a very small change to the risk profile. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment to the Indian Point 3 Technical Specifications presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of 'no significant hazards consideration' is justified.

6. ENVIRONMENTAL CONSIDERATION

The proposed changes to the IP3 TS do 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.

7. PRECEDENT

The proposed amendment incorporates into the IP3 TS a change that is similar (i.e., greater than 15 years) to the following license amendments previously approved by the NRC to extend the Type A test frequency:

- June 29, 2007 (ML071800319), for Three Mile Island, Unit 1
- March 24, 2006 (ML060520032) for Seabrook Station
- February 9, 2006 (ML060410310), for River Bend Station
- December 23, 2005 (ML053190343), for St. Lucie Plant Unit 2

8. REFERENCES

1. Entergy Letter, "Amendment to License Renewal Application – Reflecting Shortened License Renewal Terms for Units 2 and 3," dated February 8, 2017 (NL-17-019) (ML17044A005)

2. Entergy Letter, "Proposed License Amendment Regarding Extending the Containment Type A Leak Rate Testing Frequency to 15 Years," dated February 4, 2014 (NL-14-014)

3. NRC Letter, "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment re: Extension of the Type A Containment Integrated Leak Rate Test Frequency From 10 to 15 Years (TAC No. MF3426)," dated March 13, 2013 (ML15028A308)

4. Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 26, 1995

5. NUREG-1493, "Performance-Based Containment Leak-Test Program," dated July 1995

6. Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," dated July 2012

7. Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," dated August 1994

8. Electric Power Research Institute (EPRI) TR-1018243, "Risk Impact Assessment of Extending Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, " dated October 2008

9. Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011

10. Constellation Nuclear Letter, "Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," dated March 27, 2002 (ML020920100)

11. Entergy Letter, "Entergy Transmittal of Indian Point 3 ASME Section XI, IWL Concrete Containment Inspection In Accordance With the Parties' Approved Settlement of License Renewal Contention NYS-24," dated December 2, 2016 (NL-16-134)

ATTACHMENT 2 TO NL-17-137

Technical Specification Pages (Mark-up)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

5.5.15 Containment Leakage Rate Testing Program

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, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008, as modified by the following exceptions:

a. ANS 56.8-2002, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

b. ***The next Type A test to be performed after the March 2005 Type A test shall be performed no later than the plant restart after the Spring 2021 (3R21) RFO.***

The maximum allowable primary containment leakage rate, L_a, at a minimum test pressure equal to P_a, shall be 0.1% of primary containment air weight per day. P_a is the peak calculated containment internal pressure related to the design basis accident.

Leakage acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$,

c. Isolation Valve Seal Water System leakage rate acceptance criterion is $\leq$ 14,700 cc/hr at $\geq 1.1\ P_a$.

d. Acceptance criterion for leakage into containment from isolation valves sealed with the service water system is $\leq$ 0.36 gpm per fan

(continued)

ATTACHMENT 3 TO NL-17-137

Revised Technical Specification Pages (Clean)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

5.5.15 Containment Leakage Rate Testing Program

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, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008, as modified by the following exceptions:

a. ANS 56.8-2002, Section 3.3.1: WCCPPS isolation valves are not Type C tested.

b. The next Type A test to be performed after the March 2005 Type A test shall be performed no later than the plant restart after the Spring 2021 (3R21) RFO.

The maximum allowable primary containment leakage rate, L_a, at a minimum test pressure equal to P_a, shall be 0.1% of primary containment air weight per day. P_a is the peak calculated containment internal pressure related to the design basis accident.

Leakage acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$,

c. Isolation Valve Seal Water System leakage rate acceptance criterion is $\leq$ 14,700 cc/hr at $\geq 1.1\ P_a$.

d. Acceptance criterion for leakage into containment from isolation valves sealed with the service water system is $\leq$ 0.36 gpm per fan

(continued)

5.5.15 Containment Leakage Rate Testing Program (continued)

cooler unit when pressurized at $\geq$ 1.1 P_a. This limit protects the internal recirculation pumps from flooding during the 12-month period of post accident recirculation.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10CFR50, Appendix J.

The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 42.38 psig.
The containment design pressure is 47 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

5.5.16 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Ventilation System (CRVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release,
or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

(continued)