UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 5, 2002

Michael C. Farrar, Chairman
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Charles N. Kelber
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Richard F. Cole
Administrative Judge
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of
Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations Inc.
(Indian Point Nuclear Generating Unit No. 2)
Docket No. 50-247-OLA

Dear Administrative Judges:

This letter is to inform the Board that, pursuant to 10 C.F.R. §50.91(a)(4), the Staff has made a final determination that no significant hazards consideration is involved in Entergy's amendment request for a one-time change to Technical Specification Surveillance Requirement 4.4.A.3 to revise the frequency of the containment integrated leak rate test (ILRT, Type A test) from at least once per 10 years to once per 15 years and issued the amendment today. A copy of the safety evaluation, final no significant hazards consideration determination, and amended technical specifications pages are enclosed for your information. In addition, the technical specification pages are being faxed to all parties as the pages were not immediately available in electronic form. The ADAMS accession number for the entire package is ML021860223.

Sincerely,

Sara Brock

Sara E. Brock
Counsel for NRC Staff

Enclosures: As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 5, 2002

Mr. Michael R. Kansler
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
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SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - AMENDMENT RE: ONE-TIME DEFERRAL OF CONTAINMENT INTEGRATED LEAK RATE TEST (TAC NO. MB2414)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 232 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 (IP2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated July 13, 2001, and supplemented by letters dated November 30, 2001, March 13, April 3, May 30, and June 13, 2002. The amendment revises TSs to allow a one-time deferral of the Type A containment integrated leak rate test (ILRT). This results in an extension of the surveillance interval from a once-per-10-year frequency to a once-per-15-year frequency for performance of the test. Under the allowed extension, IP2 shall perform its next ILRT within 15 years of the last successful ILRT, which was conducted on June 20, 1991.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

Daniel S. Collins for

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures: 1. Amendment No. 232 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY NUCLEAR INDIAN POINT 2, LLC

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 232
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated July 13, 2001, and supplemented November 30, 2001, March 13, April 3, May 30, and June 13, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 232 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days. Additionally, as set forth in the licensee's May 30, 2002, supplemental submittal, the containment integrated leakage test procedure shall be revised prior to the next performance of the containment integrated leakage test to ensure that the abandoned portions of the Weld Channel and Penetration Pressurization System are subjected to containment atmospheric pressure during the performance of the test.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Laufer

Richard J. Laufer, Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 5, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 232

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Pages	Insert Pages
4.4-2	4.4-2

e. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

2. Acceptance Criteria

The As Found measured leakage rate shall be less than 1.0 L_a where L_a is equal to 0.1 w/o per day of containment steam air atmosphere at 47 psig and 271°F, which are the peak accident pressure and temperature conditions. Prior to entering a mode where containment integrity is required, the As Left leakage rate shall not exceed 0.75 L_a.

3. Frequency

The integrated leakage rate test frequency shall be performed in accordance with 10 CFR 50 Appendix J, Option B as modified by approved exemption and in accordance with guidelines contained in Regulatory Guide 1.163, dated September 1995, with the following exceptions:

Exception 1: The Type A testing frequency specified in NEI 94-01 paragraph 9.2.3 as at-least-once-per-10 years based on acceptable performance history is changed to allow a Type A testing frequency of at-least-once-per-15 years based on acceptable performance history. This is a one-time-only exception that applies only for the interval following the Type A test performed in June 1991.

B. SENSITIVE LEAKAGE RATE

1. Test

A sensitive leakage rate test shall be conducted with the containment penetrations, weld channels, and certain double-gasketed seals and isolation valve interspaces at a minimum pressure of 52 psig and with the containment building at atmospheric pressure.

2. Acceptance Criteria

The test shall be considered satisfactory if the leak rate for the containment penetrations, weld channel and other pressurized zones is equal to or less than 0.2% of the containment free volume per day.

3. Frequency

A sensitive leakage rate test shall be performed at every Refueling Interval (R#).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 232 TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated July 13, 2001, Consolidated Edison Company of New York, Inc. (Con Edison) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs). On September 6, 2001, Con Edison transferred its ownership interests in IP2 to Entergy Nuclear Indian Point 2, LLC and its operating authority under the IP2 license to Entergy Nuclear Operations, Inc. (ENO). By letter dated September 20, 2001, ENO requested that the U.S. Nuclear Regulatory Commission (NRC) continue to review and act on all requests before the Commission which had been submitted before the transfer. Accordingly, the NRC staff has acted upon the request. The request for an amendment was supplemented by ENO in letters dated November 30, 2001, March 13, April 3, May 30, and June 13, 2002.

The requested change would revise the IP2 TSs to allow a one-time deferral of the Type A containment integrated leak rate test (ILRT) resulting in an extended interval of up to 15 years from the last ILRT for performance of the test. The November 30, 2001, March 13, April 3, May 30, and June 13, 2002, letters provided clarifying information that did not expand the application beyond the scope of the *Federal Register* notice or change the initial proposed no significant hazards consideration determination. As discussed in Section 2.2 of this safety evaluation (SE), ENO's amendment request is consistent with similar requests from other licensees to extend ILRT intervals, which the NRC has approved.

On March 18, 2002, Riverkeeper, Inc., filed a Petition for Leave to Intervene and Request for Hearing, which it subsequently amended on April 30, 2002. The Atomic Safety and Licensing Board issued an Order dated July 17, 2002, setting a schedule for prehearing conference. As of the date of issuance of this amendment, the Board has not yet acted on the Petition.

2.0 REGULATORY EVALUATION

Containment structures, including access openings and penetrations, are designed to ensure that they can accommodate the calculated pressure and temperature conditions resulting from a loss-of-coolant accident (LOCA) without exceeding the design-basis leakage rate. The design-basis leakage rate is specified such that leakage at that leakage rate of radioactive materials to the environment resulting from a LOCA would not result in off-site doses greater than the limits of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 100.

In order to ensure that reactor containments are maintained properly such that they will be able to perform their design-basis functions, the NRC's regulations require that licensees conduct leakage rate testing and inspections of containments and associated pressure retaining components at periodic intervals. Title 10 of the *Code of Federal Regulations*, Section 50.54(o) specifies that primary reactor containments for water-cooled power reactors shall be subject to the leakage testing requirements contained in 10 CFR 50, Appendix J (Appendix J). That appendix defines three types of leakage rate tests that must be conducted on containment pressure boundaries, and provides two options for testing programs that licensees may implement to satisfy the requirements of the appendix. Additionally, Appendix J and 10 CFR 50.55a specify containment inspection requirements that licensees must satisfy.

2.1 Leakage Rate Tests

Type A - A Type A test (also known as an ILRT) is an overall leakage rate test of the containment structure, which measures the integrated leakage rate from all potential leakage paths including containment liner welds, valves, fittings, and components that penetrate containment. These tests typically involve pressurizing the containment atmosphere to a specified test pressure for a duration sufficient to determine what the containment leakage would be under design-basis accident conditions.

The acceptance criteria for the Type A test and the TS leakage limits are conservatively established to ensure that, in the event of a design-basis accident, the dose received by a member of the general public will not exceed the limits specified in 10 CFR Part 100.

Type B - A Type B test, (also known as a local leakage rate test (LLRT)), is intended to detect or measure leakage across pressure-retaining or leakage-limiting boundaries other than valves, such as (1) containment penetrations whose design incorporates resilient seals, gaskets, sealant compounds, expansion bellows, or flexible seal assemblies, (2) seals, including door operating mechanism penetrations, which are part of the primary containment, or (3) doors and hatches with resilient seals or gaskets except for seal-welded doors.

This type of test typically involves pressurizing the penetration/seals with air (or dry nitrogen) to a specified test pressure and determining the leakage through the penetration.

Type C - A Type C tests (also known as an LLRT) is a pneumatic test to measure containment isolation valve leakage rates.

2.1.1 Leakage Rate Test Acceptance Criteria

The acceptance criteria for containment leakage rate tests are typically expressed in terms of the maximum allowable containment leakage rate, L_a, that would occur at the calculated peak containment internal pressure related to the design-basis LOCA. Plant TSs typically specify values for L_a in terms of the allowable weight percent (w/o) of the containment atmosphere that may leak per 24 hours. The acceptance criteria for Type A tests, and the combined Type B and Type C tests are typically specified as multiples of L_a. For example, typical acceptance criteria for the ILRT are $1L_a$ for "as-found" tests and $.75\ L_a$ for "as-left" tests. Typical acceptance criteria for the combined Type B and Type C tests is $.6\ L_a$.

2.2 Test Program Options

Appendix J provides licensees two alternatives for leakage testing programs. The first, Option A, provides prescriptive requirements with specific test methods, test frequencies, and acceptance criteria for all three types of leakage rate tests. Regarding the ILRT, Option A specifies that three tests must be conducted during each 10-year interval with the third test being conducted when the plant is shut down for the 10-year plant inservice inspections (ISI).

In 1995, the NRC amended the Appendix J requirements to provide the second alternative program, Option B, which allows licensees to adjust the frequency of leakage rate testing based on the performance history of the tested components. In other words, under Option B, containment pressure boundary components that have a poor leakage rate test performance history are required to be leak rate tested more frequently than components that have a good performance history. The NRC's analysis to support that 1995 rule change is discussed in NUREG-1493, "Performance-Based Containment Leak-Test Program," dated January 1995. That analysis included evaluations of historical leak-rate test experience, which found that Type A testing detected breeches of the containment pressure boundary that were not identifiable by Type B or Type C testing in only about 3 percent of the ILRT failures that had occurred prior to April 1993. That is, LLRTs would have identified the containment pressure boundary leakages in over 97 percent of the cases. The NRC's analysis to support the 1995 rule change also included a risk impact assessment associated with a range of extended leakage rate test intervals, which is discussed further in Section 3.5 of this SE.

Title 10 of the *Code of Federal Regulations* Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. Plant TSs typically require that the integrated leakage rate test frequency shall be performed in accordance with 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 (RG 1.163). This regulatory guide endorses, with certain exceptions, NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995 (NEI 94-01). NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval for an additional 15 months in certain circumstances.

In 1998, the NRC staff issued Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (RG 1.174). Since September 2000, many licensees have used the guidance in RG 1.174 to support amendment requests for one-time deferrals of containment ILRTs from 10 to 15-year intervals. To date, the NRC has approved license amendments to extend the ILRT intervals for 18 nuclear generating units. The ENO amendment request for IP2 is consistent with the requests that the NRC has approved.

In an effort to reduce the need for individual plant specific applications to extend ILRT intervals, the Nuclear Energy Institute (NEI) and the Electric Power Research Institute (EPRI) are developing a proposal for a generic change to NEI 94-01. Based on presentations that NEI and EPRI have made to the NRC staff, the staff expects the proposed change will use insights of RG 1.174 to establish the maximum ILRT interval at 15 years, or perhaps as much as 20 years. While that effort is still ongoing, some licensees, such as ENO, have decided to request plant specific amendments to extend their ILRT intervals.

2.3 <u>Inspections</u>

In addition to the leakage rate tests discussed above, Appendix J specifies that visual examinations of the accessible interior and exterior surfaces of containment structures and components shall be performed prior to any Type A test, and at periodic intervals between Type A tests (Option B) to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak tightness. Furthermore, 10 CFR 50.55a(b)(2)(viii), (b)(2)(ix), and (g)(4)(v) specify ISI, repair, and replacement requirements that licensees must meet with regard to reactor containment structures and associated pressure retaining components. Specifically, 10 CFR 50.55a incorporates by reference the requirements of the *American Society of Mechanical Engineers Boiler and Pressure Vessel Code* (ASME Code), Section XI, Subsections IWE and IWL, and specifies additional requirements.

2.4 <u>IP2 Containment and Associated License Requirements</u>

The IP2 containment structure is a steel-lined reinforced concrete vertical cylinder with a flat base mat and hemispherical dome that completely encloses the entire reactor and reactor coolant system (RCS). Its purpose is to ensure that any leakage of radioactive materials to the environment, even if gross failure of the RCS were to occur, does not result in off-site doses greater than the limits of 10 CFR Part 100. The IP2 containment pressure boundary consists of the steel-lined containment structure, containment access penetrations, and other process piping and electrical penetrations. IP2's maximum allowable containment leakage rate and leakage test acceptance criteria, as specified in the TSs, are as follows:

L_a = 0.1 w/o per day of the containment steam air atmosphere at 47 psig and 271 °F.

ILRT acceptance criteria:	As-found:	1.0 L_a	
	As-left:	.75 L_a	
LLRT (combined Type B & C):		.6 L_a	for containment isolation valves subject to gas pressurization, airlocks, penetrations and certain double-gasketed seals.

2.4.1 <u>IP2 Leakage Testing Program</u>

IP2 adopted Option B of Appendix J, as approved by the NRC in Amendment No. 190 to the IP2 license on April 10, 1997. IP2 TS 4.4.A.3 requires that the ILRT frequency shall be performed in accordance with Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in RG 1.163. The two most recent Type A tests at IP2 were successful. Thus, as discussed in Section 2.2 above, the minimum required ILRT frequency for IP2 is once per 10 years.

The licensee is requesting additions to item 4.4.A.3 of TS 4.4, "Containment Tests," which would indicate that IP2 is allowed to take an exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS says that the Type A testing frequency specified in NEI 94-01, paragraph 9.2.3, as at least once per 10 years based on acceptable performance history is changed to allow a Type A testing frequency of at least once per 15 years based on acceptable performance history. It further specifies that this is a one-time only exception that applies only for the interval following the Type A test performed on June 20, 1991.

2.4.2 IP2 Containment Inspection Requirements

With regard to the containment inspection requirements contained in 10 CFR 50.55a, the licensee is using the 1992 Edition and the 1992 Addenda of Subsections IWE and IWL of Section XI of the ASME Code for conducting its ISI of the IP-2 containment with approved relief from certain Code requirements. The ISI interval began in March 2000 and will end in March 2010. The licensee's request for a one-time only deferral of the Type A test does not affect these inspection requirements or the Appendix J inspection requirement to perform visual inspections at periodic intervals between Type A tests.

2.4.3 Related IP2 Engineered Safety Features

IP2 is one of a very few U.S. plants to have a system that pressurizes the containment weld channels and certain containment penetrations during normal plant operation. The containment weld channels are long, narrow enclosures which cover the welds between the steel plates that make up the containment liner. The containment liner, which covers the inner surface of the concrete containment structure, provides an essentially leak-tight barrier against leakage out of the containment during an accident. Nuclear power plants are typically built with containment weld channels. During the initial construction of the containment, the containment weld channels are pressurized to test the containment liner welds to assure proper fabrication of the welds. However, in the case of IP2 there is a Containment Weld Channel and Penetration Pressurization System (WC&PPS) that keeps the channels pressurized during normal plant operation and, by design, during accidents. The pressure is maintained at or above calculated peak containment accident pressure and will prevent containment atmosphere from leaking out of the containment through the liner welds during an accident in the unlikely event that an undetected significant flaw exists in a liner weld. Although the system is not credited in the calculation of the radiological consequences of an accident, it is an engineered safety feature and would act to prevent containment leakage through any flaws that might be present in the liner welds.

Another function of the WC&PPS is to provide a continuous measure of leakage through the portions of the containment boundary which it pressurizes. Technical Specification 3.3.D requires the system to be operable in modes above cold shutdown and places a limit on the total rate of leakage out of the WC&PPS during normal operation. If the leakage rate exceeds the limit, the licensee must reduce the leakage rate below the limit or shut down the plant.

A portion of the containment liner is attached to the containment basemat. It is covered over by several additional feet of concrete, which forms the bottom floor of the containment. Thus, the containment weld channels on the basemat portion of the containment liner, and on the lowest several feet of the containment wall, are under the floor and are not accessible for inspection.

3.0 EVALUATION

The NRC staff reviewed the following in its evaluation of the licensee's amendment request:

- ILRT performance history
- recent LLRT performance history
- recent containment inspection results
- WC&PPS performance
- risk impact assessment associated with extending the ILRT interval to 15 years

3.1 ILRT Performance History

ENO's July 13, 2001, application included information regarding the plant's ILRT performance history. Since 1979, four ILRTs have been conducted at IP2 and all were completed with satisfactory results. The application states that the most recent ILRT was completed on June 20, 1991, with an "as-left" test result of .047791 w/o of containment air per day leakage (.47791L_a), which is well within the .075 w/o per day (.75L_a) acceptance limit. The previous ILRT, conducted on December 19, 1987, showed similar results with a measured leakage of .047726 w/o per day (.47726L_a). The docketed test reports submitted to the NRC after these tests, dated December 12, 1987, and September 20, 1991, respectively indicate that the only containment pressure boundary repairs conducted during outages prior to the 1987 and 1991 ILRTs were performed on valves and penetrations that are subject to Type B or Type C testing. Hence, the containment structure and liner had performed acceptably during the operating cycles prior to these Type A tests.

3.2 Recent LLRT performance history

As discussed in Section 2.2 above, industry experience has shown that the vast majority of containment pressure boundary breeches have been through components that are subject to Type B and Type C LLRTs. Therefore, in evaluating the current condition of the IP2 containment, the NRC staff reviewed recent LLRT performance information provided by the licensee. In its June 13, 2002, supplemental submittal, ENO stated that the current running total of the combined Type B and Type C LLRTs is 1.8275 standard cubic feet per minute (scfm), which converts to approximately .24L_a. This value is well within the .6L_a acceptance limit for combined Type B and Type C LLRTs, and indicates that the licensee is effectively maintaining the leak tightness of the containment pressure boundary components that are subject to LLRTs.

3.3 Recent containment inspection results

The July 13, 2001, application referenced an ISI program summary report, "2000 Refueling Outage Inservice Inspection (ISI) Program Summary Report - Second Outage, Second Period, Third Interval," dated April 2, 2001, that the licensee had previously submitted. That report included a discussion of results from containment liner and concrete surface examinations that the licensee conducted during the 2000 refueling outage as required by 10 CFR 50.55a.

As discussed in the ISI program summary report, the licensee identified some areas of degradation in the containment concrete, reinforcing bars (rebar), cadweld splices[1], liner plate and penetrations. With the exception of the corrosion identified on the liner, the summary report characterized the identified degradation as minor.

Because the ISI of the containment, in combination with the leakage rate tests discussed above, ensures the structural integrity and leak-tightness of the containment, the NRC staff sought additional assurance that the conclusions drawn by the licensee regarding the identified degradation were appropriate. By letter dated October 4, 2001, the NRC issued a request for additional information (RAI) regarding the ISI of containment and potential areas of weaknesses in the containment that may not be apparent in the risk assessment.

ENO responded to the NRC staff's RAI in a letter dated November 30, 2001. In its reply, the licensee indicated that:

- the corrosion identified on the containment penetrations during the IWE inspections was characterized as non-aggressive surface corrosion which had not resulted in significant loss of material.
- corrosion identified on the containment liner was limited to areas slightly above and slightly below the concrete containment floor. Volumetric examination of the containment liner in these areas had determined that the minimum remaining liner thickness is sufficient to meet design requirements.
- Although the visual inspection of the concrete containment structure had identified some degradation, evaluation by the Responsible Professional Engineer for the IP2 IWE program, with support from Raytheon Engineering and Sargent and Lundy, had determined that the identified conditions do not adversely affect the ability of the containment to meet its design-basis requirements. Corrosion identified on rebar and cadweld splices did not exhibit signs of flaking or aggressive corrosion processes, and the reduction on cross-section of rebar or cadwelds was less than 10 percent. Additionally, the evaluation considered the location of the degraded rebar and cadweld splices within the containment structure in order to account for variations in the actual stresses and resulting margins within this reinforcing steel.
- The accessible areas of the containment pressure boundary will be periodically monitored for signs of degradation.

In summary, the licensee provided its basis for accepting various degradations of the containment concrete, rebar, cadweld splices, liner plate and penetrations relying on the analyses performed separately by its consultants, and accepted by the licensee.

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[1]A cadweld is a heavily walled metal cylinder that is used to create splices between two pieces of rebar. The ends of the rebar are placed into the cylinder and molten metal is then injected into the cylinder to fuse the rebar together. A cadweld splice typically has a diameter twice that of the rebar being joined.

Based on its review of the November 30, 2001, RAI response, the NRC staff issued a second RAI, dated February 5, 2002, which requested additional information regarding (1) why the licensee should not perform an ILRT during the next outage to verify that the "as is" containment is able to withstand the design pressure without exceeding the allowable leakage rate criteria; and (2) in conjunction with a potential for degradation in uninspectable areas of the steel liner, how potentially degraded conditions were factored into the risk assessment for the proposed ILRT interval extension.

By letter dated March 13, 2002, the licensee provided responses to the February 5, 2002, RAI. The responses are summarized in the following paragraphs:

The liner plate at IP2 is fabricated from ASTM A442, Grade 60 steel. Its nominal thickness is ½ in. from the top of the basemat, at approximately the 43 ft. elevation to the 72 ft. - 9 in. elevation. Above 72 ft. - 9 in. elevation, the thickness in the cylindrical portion of the containment liner is ⅜ in. During the refueling outage in 2000, the liner plate was examined in accordance with requirements of the Code. The examination indicated (1) coating deterioration and minor corrosion on 40 out of 116 electrical and mechanical penetrations; (2) coating degradation and minor corrosion at elevation 134 ft.; (3) portions of the moisture barrier at the liner basemat interface were deteriorated; (4) liner corrosion between 2 in. above and 3 in. below the concrete containment floor, which had resulted from an event in 1980 in which the containment had been flooded by a leak in the service water piping within the containment.

As required by the ASME Code, the licensee performed ultrasonic testing (UT) of the liner plates in the corroded areas near the 46 ft. elevation (containment concrete floor) to determine the extent of corrosion. The licensee examined 10 areas of the liner at the 46 ft. elevation. Included in the areas examined were all locations where there was either a degradation of the moisture barrier or the concrete floor. In one of the 10 sample areas, the minimum general area liner thickness was determined to be 0.355 in. and in two of the other 10 sample areas the minimum general area liner thickness was determined to be 0.360 in. In each of these cases, the moisture barrier performance had been degraded either by damage to the adjacent concrete or damage to the moisture barrier itself. In each of the remaining sample areas, the minimum general area liner thickness was greater than this 0.355 - 0.360 in. range. Therefore, the NRC staff has reasonable assurance that the results of the examinations have identified the minimum liner thickness in the areas where the containment liner is most likely to have been affected by the 1980 flooding event.

The licensee's consultant, Raytheon Engineers, had established a minimum required liner thickness of 0.34 in. This is based on a conservatively established critical buckling stress value associated with thermally induced compressive stresses that would result in bulging or buckling of the liner, assuming that the containment liner is un-insulated. ENO noted in its March 13, 2002, RAI response that the portions of the containment liner that have identified corrosion degradation are insulated, and not subject to significant thermal stresses. Thus, the licensee states, the minimum liner thickness required to maintain containment pressure boundary integrity is less than the Raytheon specified 0.34 in. and there is sufficient margin in the available liner thickness for the liner to continue to perform its design basis function. Based on the NRC staff's review of the results of the licensee's analysis and examinations, the NRC staff concludes that the structural integrity of the containment is acceptable because the remaining liner thickness is sufficient to withstand the loading associated with design-basis accident conditions.

In order to prevent any further corrosion of the liner below the cylinder/containment floor interface, the licensee repaired the moisture barrier seal, between the concrete containment floor and the insulation mounted on the containment wall, in areas where it was deteriorated. In its November 30, 2001, RAI response, ENO stated that portions of the containment liner in the general area around the 46 ft. elevation are planned for reinspection during the upcoming 2002 refueling outage (RFO). Additionally, in its March 13, 2002, RAI response, ENO stated that the areas of observed corrosion will be reinspected during the next inspection period (2004 RFO) as required by IWE-2420 of the ASME Code.

Therefore, based on the facts that the licensee has repaired the moisture barrier in order to eliminate the past corrosion mechanism (moisture from a service water leak), and will be inspecting the areas of the containment liner where corrosion was identified during the next two upcoming outages, the NRC staff finds the licensee's approach for ensuring that the integrity of the containment liner will be maintained to be acceptable.

With regard to the areas of the liner plate degraded by corrosion, it is the NRC staff's conclusion that the most appropriate method of assessing and monitoring the condition of the containment liner is through the ASME Code-required re-inspections that the licensee is still required to perform.

In the March 13, 2002, RAI response, the licensee also provided information regarding the manner in which a hypothetical flaw in uninspectable areas of the liner was incorporated into the licensee's risk assessment. That portion of the licensee's response is discussed further in Section 3.5 of this SE.

3.4 <u>WC&PPS performance</u>

During its review, the NRC staff raised issues related to the containment WC&PPS. In 1980, the IP2 containment lower elevations were flooded with brackish river water as a result of a service water leak. During the current review, the NRC staff raised a concern that the flood might have caused water intrusion into the weld channels in the basemat and lower wall region resulting in corrosion of the liner welds. The licensee responded, in their May 30, 2002, letter, that the WC&PPS was pressurized to 52 pounds per square inch, gauge pressure (psig) during the flood, a pressure higher than that of the flood water, so that the water could not have intruded into the system.

However, the flood did apparently cause exterior corrosion on some of the piping that supplies WC&PPS air to the weld channels in the basemat and lower wall region. These small-diameter pipes pass down through the concrete containment floor to the basemat and lower wall weld channels. Starting in 1993, the licensee discovered that some of these supply pipes have corroded enough to develop leaks that exceeded the TS limit for WC&PPS leakage. Several zones in the basemat and lower wall region were affected.

The problems with the first zones affected, designated W10 and B6, generated an emergency license amendment request dated April 8, 1993. These zones exhibited increased leakage and the licensee was able to find leaks in the supply piping, by visual boroscopic examinations down the insides of the ½-inch pipes. Repairs would have required the licensee to remove several feet of the concrete floor and move major equipment, such as cable trays, to access the leaking pipes. The NRC staff issued License Amendment No. 162 on April 14, 1993, which added the following provision to the TSs:

With a portion of the weld channel pressurization system inoperable, and it is determined that it is not repairable by any practicable means, then that portion may be disconnected from the system.

In the NRC staff's SE, the staff agreed with the licensee's conclusion that the increased leakage was due to leaks in the air supply lines and not in the containment liner welds. The NRC staff's acceptance of the TS change in Amendment No. 162 was based on the fact that no credit is taken for the WC&PPS in the accident analyses, and on the continued leakage rate testing of the liner welds during Type A tests. The weld channels, including those disconnected from the WC&PPS, are vented to the containment atmosphere during Type A tests so that the liner welds are tested as part of the Type A tests. The NRC staff noted that zones W10 and B6 constituted approximately 4 percent of the containment liner welds, and that approximately 20 percent of the liner welds were beneath the concrete floor. Potential future use of the new TS provision to remove additional zones would, therefore, be relatively limited.

Since this action in 1993, the licensee has disconnected several additional zones from the WC&PPS. In each case but one, the licensee determined that the leaks were in the WC&PPS supply piping before disconnecting the zone. In its May 30, 2002, supplemental submittal, ENO committed to revise the IP2 containment integrated leakage rate test procedure prior to the next ILRT performance to ensure that the abandoned portions of the WC&PPS are subjected to containment atmospheric pressure during the performance of the test. Accordingly, this commitment is being incorporated into the amendment as a license condition.

In the case of zone W11, the licensee could not determine the location of the leak or leaks, but assumed that the leaks were from the supply piping and not through the containment welds, based on their experience with the other disconnected zones. Since there were no leaks above the concrete floor, they concluded the leaks were beneath the concrete, making them impracticable to repair, and disconnected zone W11 in March 2000.

In telephone conferences in early June 2002, the NRC staff questioned the status of zone W11 and requested clarification. The licensee documented their response in its June 13, 2002, letter.

On June 7, 2002, the licensee tested zone W11 with 52 psig service air and measured a flow rate of 2000 standard cc/min. This equals a leakage rate of approximately 0.01 L_a. Even if the measured leakage rate is conservatively assumed to all be out of the containment, it does not significantly increase the total containment leakage or cause it to exceed L_a. Furthermore, the licensee has temporarily reconnected zone W11 to the WC&PPS, and the WC&PPS is now operating within its TS limits, with zone W11 included.

Therefore, the NRC staff finds that the containment liner and liner welds currently covered by the WC&PPS are not leaking beyond regulatory limits, and that the containment liner and liner welds in disconnected zones have not indicated leakage rates beyond regulatory limits. Further, the NRC staff finds that these areas, taken together, are less likely to have developed undetected leaks than the liners and liner welds of other nuclear power plants, due to the operation of the WC&PPS. The NRC staff concludes, therefore, that the probabilistic risk assessment techniques used in Section 3.5, below, are valid for the IP2 containment.

3.5 Risk Impact Assessment

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The assessment was provided in the July 13, 2001, application. Additional analysis was provided in letters from the licensee dated November 30, 2001, and March 13, April 3, and May 30, 2002. In performing the risk assessment, ENO considered the guidelines of NEI 94-01, the methodology used in EPRI TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and Regulatory Guide 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

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, provided 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 EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 in 10 years to 1 in 10 years, will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage, in percent of person-rem/year, for the PWR representative plant was estimated to increase from .032 percent to .035 percent. This confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3-per-10-years to 1-per-10-years leads to an "imperceptible" increase in risk.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. 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. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3-in-10-year interval. 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. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided an analysis which estimated all of these risk metrics and whose methodology was consistent with previously approved submittals. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk is estimated to be 0.03 percent. The increase in the total integrated plant risk, given the change from a 3-in-10-year test interval to a 15-year test interval, was 0.08 percent. This is reasonable when compared to the range of risk increase, 0.02 to 0.14 percent, estimated in NUREG-1493 when going from a 3-in-10-year test interval to a 10-year interval. NUREG-1493 concluded that a reduction in the frequency of tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. 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 the risk-acceptance guidelines as increases in CDF less than 10^{-6} per reactor year and increases in LERF less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from 1 in 10 years to 1 in 15 years is estimated to be 3.3×10^{-8}/year. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 9.8×10^{-8}/year. Increasing the Type A interval to 15 years is considered to be a very small change in LERF when using the guidelines of RG 1.174.

   In response to the NRC staff's RAI question related to the effects of hypothetical degradation in uninspectable areas of the liner of the IP2 containment (i.e., those areas where visual inspection cannot be performed), the licensee considered the consequences of such an occurrence in the risk assessment. The methodology used by the licensee in performing its risk analysis, and the associated assumptions, are summarized below:

   a. The containment fragility curves were adjusted by reducing the median (best estimate) failure pressure by 12 percent to account for hypothetical liner degradation assumed to occur in uninspectable areas (i.e. the backside of the liner).
   b. The early containment failure probabilities in the individual plant examination (IPE) were revised based on the revised fragility curves, and the source term category frequency for early and late containment failures (i.e., Class 7) were determined.
   c. An effective Class 7 frequency and associated risk, which accounts for the potential for corrosion on the backside of the liner, was determined. This conservatively assumed that the ILRT pressure is sufficient to result in liner failure.
   d. The change in risks in absolute and percentage terms due to an increased ILRT interval was determined.
   e. The change in large early release frequency (LERF) as a function of ILRT interval was determined as the sum of the increase in LERF due to Class 3b and that portion of the Class 7 that contributes to LERF.

   The NRC staff has reviewed the methodology and associated assumptions used by ENO in this sensitivity case, and finds them to be conservative and appropriate. With regard to the assumed 12 percent reduction in median failure pressure, for reinforced concrete containments, such as the IP2 containment, NUREG/CR-6706, "Capacity of Steel and Concrete Containment Vessels with Corrosion Damage," dated February 2001, indicates that about 50 percent corrosion of the liner thickness will result in 12 percent reduction in strength. Thus, for incorporating the potential degradation in the uninspectable side of the liner into the risk assessment, the 12 percent reduction in the containment strength is acceptable.

The increase in LERF resulting from a change in the Type A test interval from 1 in 10 years to 1 in 15 years is estimated to be 3.4 x 10^{-8}/year. The increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 1.0 x 10^{-7}/year. The sensitivity assessment provides added assurance that increasing the Type A interval to 15 years is a very small change in LERF.

3. Regulatory Guide 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in the conditional containment failure probability was estimated to increase by 0.0010 for the proposed change and 0.0032 for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the very small change in the conditional containment failure probability for the proposed change.

   The staff recognizes the limitations of a conditional containment failure probability approach. For plants, such as IP2, with core damage frequency estimates well below 10^{-4}, the ability of the containment to withstand events of even lower probability becomes less clear. Therefore, it is important to consider other risk metrics in conjunction with the conditional containment failure probability, such as total LERF. The licensee has sufficiently demonstrated that the total LERF is less than 10^{-5} for the purpose of this evaluation.

Based on these conclusions, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance criteria while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

3.6 Summary

The NRC staff finds the overall procedure used by the licensee in its analysis to be reasonable.

Based on the considerations previously discussed, including the licensee's actions to preclude additional degradation of the primary containment components (e.g. repair of the moisture barrier and planned future inspections) as well as incorporation of certain degradation in the risk analysis, the NRC staff finds that granting the requested ILRT extension will not adversely affect the leak-tight integrity of the primary containment. To summarize, the key points supporting the NRC staff's finding are as follows:

- Historical performance of ILRTs and recent LLRTs at IP2 have been acceptable.

- Volumetric examination of the corroded liner areas around the perimeter of the containment floor showed that the remaining liner is adequate to perform its design function.
- These areas of identified corrosion are subject to reinspection per the ASME Code and, thus, their condition will be monitored.
- The leakage test and return to service of WC&PPS zone W11 provides reasonable assurance that liner welds in the W11 zone are sound.
- Other portions of the WC&PPS that have been retired in place do not present significant concern regarding the liner condition because: (a) all zones were in service and pressurized during the 1980 flooding event, and (b) the licensee verified the leaks that prompted the abandonment were in the air supply piping before the zones were retired.
- Historically, most leakage has occurred through containment isolation valves and penetrations that are subject to LLRTs. The LLRT program requirements are unchanged by this amendment and, thus, the LLRT program will continue to ensure the leak-tight integrity of the containment isolation valves and penetrations.
- Requirements for periodic visual inspection of the containment are unchanged by this amendment.
- A hypothetical flaw in an inaccessible area of the liner is included in the risk assessment of the effects of extending the ILRT interval. This assessment demonstrated that the increase in predicted risk due to the proposed change is within the acceptance criteria while maintaining the defense-in-depth philosophy of RG 1.174.

It should also be noted that Subarticle IWE-5000 of Section XI of the ASME Code, requires leak rate testing following major repair, modification, or replacement of containment components. Thus, in the event that the licensee performs major repair, modification, or replacement of containment pressure boundary components, an ILRT might be required to confirm that the repair/replacement activities are adequate and that additional degradation does not exist in other areas of the containment. Additionally, the licensee will still be required to report serious degradation of the containment pressure boundary pursuant to 10 CFR 50.72 or 10 CFR 50.73.

On the basis of the above findings, the NRC staff finds that a one-time only extension for performing the ILRT from a 1-in-10-year to a 1-in-15-year interval, and the proposed change to TS 4.4.A.3 are acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

4.1 Introduction

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration in its July 13, 2001, amendment request. The NRC staff reviewed the licensee's analysis and, based on its review, it appeared that the three standards of 10 CFR 50.92(c) were satisfied. Therefore, the NRC staff proposed to determine that the amendment request involves no significant hazards consideration, and published its proposed determination in the *Federal Register* for public comment on August 22, 2001 (66 FR 44165).

The NRC staff has completed its evaluation of the licensee's proposed amendment as discussed above. Based on its evaluation, the NRC staff has determined that the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a significant reduction in a margin of safety. The following NRC staff evaluation in relation to the three standards of 10 CFR 50.92 supports the NRC staff's final no significant hazards consideration determination.

4.2 First Standard

"Involve a significant increase in the probability or consequences of an accident previously evaluated."

No. The containment is not an accident initiating system or structure, and its performance does not act as a precursor to any accident previously evaluated. Thus, the probability of an accident previously evaluated is unaffected by a change in the ILRT frequency.

Although the containment leak tightness can affect the consequences of a design-basis accident, the IP2 containment's ILRT history has shown performance well within its TS leakage rate acceptance limits, which are unchanged by this amendment. Additionally, as discussed in NUREG-1493, "Performance-Based Containment Leak-Test Program," dated January 1995, analyses of industry historical leak-rate test experience found that Type A testing detected breeches of the containment pressure boundary that were not identifiable by Type B or Type C testing in only about 3 percent of the ILRT failures that had occurred prior to April 1993. That is, LLRTs would have identified the containment pressure boundary leakages in over 97 percent of the cases. Since the licensee's LLRT program and the containment inspections required by other TSs and the ASME Code are not altered by this amendment, the primary mechanisms for identifying indications of containment degradation that could affect leak tightness are unchanged. Furthermore, the current TS requirements for operability of the WC&PPS during plant operation are unchanged.

Therefore, in light of the above considerations, the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

4.3 Second Standard

"Create the possibility of a new or different kind of accident from any accident previously evaluated."

No. The containment does not act as an accident initiator or precursor and the amendment does not involve any physical plant modifications or changes to the plant operation. Therefore, the proposed change in the frequency for conducting a containment ILRT does not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.4 Third Standard

"Involve a significant reduction in a margin of safety."

No. A deferral of the ILRT does not change the containment's design, the plant's accident sequences, or involve any plant modifications that would result in higher containment pressures or temperatures in the event of a design-basis accident. The leakage limits specified within the TSs are also unchanged by the amendment. The margins of safety associated with the containment are unchanged. Therefore, the amendment does not involve a significant reduction in a margin of safety.

On the basis of the above evaluation, the NRC staff concludes that the proposed amendment meets the three criteria of 10 CFR 50.92. Therefore, the NRC staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a surveillance requirement. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (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.

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Date: August 5, 2002

U. S. NUCLEAR REGULATORY COMMISSION

ENTERGY NUCLEAR OPERATIONS, INC

DOCKET NO. 50-247

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 232 to Facility Operating License No. DPR-26 issued to Entergy Nuclear Operations, Inc. (the licensee), which revised the Technical Specifications for operation of the Indian Point Nuclear Generating Unit No.2 (IP2; the facility) located in Westchester County, New York . The amendment was effective as of the date of its issuance.

The amendment made a one-time only change to Technical Specification Surveillance Requirement 4.4.A.3 to revise the frequency for the containment integrated leak rate test (ILRT, Type A test) from at least once per 10 years to at least once per 15 years. This change applies only to the interval following the last Type A test that was performed satisfactorily in June 1991 at IP2.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on August 22, 2001 (66 FR 44165). A request for a hearing was filed on March 18, 2002, by Riverkeeper, Inc.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the Safety Evaluation related to this action. Accordingly, as described above, the amendment has been issued and made immediately effective and any hearing will be held after issuance.

The Commission has determined that this amendment satisfies the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 13, 2001, as supplemented November 30, 2001, March 13, April 3, May 30, and June 13, 2002, (2) Amendment No. 232 to License No. DPR-26, and (3) the Commission's related Safety Evaluation, which are available for public inspection at the Commission's PDR, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System's (ADAMS) Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov.

Dated at Rockville, Maryland, this 5th day of August 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

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