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

July 17, 2014

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: REVISED CONTAINMENT INTEGRITY ANALYSIS (TAC NO. MF0591)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 253 to Facility Operating License No. DPR-64 for Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 28, 2013, as supplemented by letters dated August 21, 2013, and April 22, 2014.

Westinghouse Nuclear Safety Advisory Letter 11-5 identified methodology errors in the long-term mass and energy release rates during a large break loss-of-coolant accident. These errors impacted the containment integrity analysis for Indian Point Unit No. 3 and necessitated revisions to limiting initial operating conditions (i.e., containment temperature, containment pressure, and refueling water storage tank temperature) resulting in changes to TS 3.5.4, "Refueling Water Storage Tank (RWST)," and 3.6.4, "Containment Pressure." In addition, the amendment revised TS 3.6.3, "Containment Isolation Valves," to delete a redundant surveillance requirement and TS 5.5.15, "Containment Leakage Rate Testing Program," to reflect a slightly higher calculated containment peak pressure.

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,

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures:
1. Amendment No. 253 to DPR-64
2. Safety Evaluation

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ENTERGY NUCLEAR INDIAN POINT 3, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

AND TECHNICAL SPECIFICATIONS

Amendment No. 253
License No. DPR-64

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (Entergy, the licensee) dated January 28, 2013, as supplemented on August 21, 2013, and April 22, 2014, 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-64 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A, B and C, as revised through Amendment No. 253, are hereby incorporated in the license. ENO 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley

Benjamin G. Beasley, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: July 17, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 253

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

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

Remove Page	Insert Page
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
3.5.4-2	3.5.4-2
3.6.3-6	3.6.3-6
3.6.4-1	3.6.4-1
5.0-31	5.0-31

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; Amdt. 203 11/27/00

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Amdt. 203 11/27/00

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 3216 megawatts thermal (100% of rated power).

(2) Technical Specifications

The Technical Specifications contained in Appendices A, B, and C, as revised through Amendment No. 253 are hereby incorporated in the License. ENO shall operate the facility in accordance with the Technical Specifications.

(3) (DELETED) Amdt. 205 2-27-01

(4) (DELETED) Amdt. 205 2-27-01

D. (DELETED) Amdt.46 2-16-83

E. (DELETED) Amdt.37 5-14-81

F. This amended license is also subject to appropriate conditions by the New York State Department of Environmental Conservation in its letter of May 2, 1975, to Consolidated Edison Company of New York, Inc., granting a Section 401 certification under the Federal Water Pollution Control Act Amendments of 1972.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	----------NOTE---------- Not required to be performed when ambient air temperature is ≥ 35°F and ≤ 95°F if heating steam supply isolation valves are locked closed. ---------- Verify RWST borated water temperature is ≥ 35°F and ≤ 105°F.	24 hours
SR 3.5.4.2	Verify RWST borated water level is ≥ 35.4 feet.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2400 ppm and ≤ 2600 ppm.	31 days
SR 3.5.4.4	Perform CHANNEL CHECK of RWST level	7 days
SR 3.5.4.5	Perform CHANNEL CALIBRATION of RWST level switch and ensure the low level alarm setpoint is ≥10.5 feet and ≤12.5 feet.	184 days
SR 3.5.4.6	Perform CHANNEL CALIBRATION of RWST level transmitter and ensure the low level alarm setpoint is ≥10.5 feet and ≤12.5 feet.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	24 months
SR 3.6.3.7	Verify each 10 inch containment pressure relief line isolation valve is blocked to restrict valve opening to ≤ 60 degrees.	24 months
SR 3.6.3.8	Perform one complete cycle of each manually operated containment isolation valve on essential lines.	24 months
SR 3.6.3.9	Verify leakage rate into containment from isolation valves sealed with service water system is within limits.	In accordance with the Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be maintained as follows:

a. If RWST temperature is > 95^0F or containment temperature is > 125^0F, Containment pressure shall be ≥ -2.0 psig and ≤ +1.5 psig.

b. If RWST temperature is ≤ 95^0F and containment temperature is ≤ 125^0F, Containment pressure shall be ≥ -2.0 psig and ≤ +2.5 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

5.5 Programs and Manuals

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)

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 253 TO FACILITY OPERATING LICENSE NO. DPR-64

ENTERGY NUCLEAR INDIAN POINT 3, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

1.0 INTRODUCTION

By application dated January 28, 2013 (Reference 1, Agencywide Documents Access and Management System (ADAMS) Accession No. ML13042A224), as supplemented by letters dated August 21, 2013 (Reference 2, ADAMS Accession No. ML13239A477), and April 22, 2014 (Reference 6, ADAMS Accession No. ML14121A113), Entergy Nuclear Operations, Inc. (Entergy, the licensee) submitted a License Amendment Request (LAR) for Indian Point Nuclear Generating Unit No. 3 (IP3) in the form of proposed changes to the Technical Specifications (TSs).

The proposed changes would revise:

(a) TS 3.5.4, "Refueling Water Storage Tank (RWST)," Surveillance Requirement (SR) 3.5.4.1 by changing the maximum RWST temperature limit to ≤105 °F from the current temperature limit of ≤110 °F;

(b) TS 3.6.4, "Containment Pressure," Limiting Condition for Operation (LCO) 3.6.4 by dividing it into two LCOs, 3.6.4.a and 3.6.4.b. The current LCO 3.6.4 requires containment pressure to be maintained between ≥ -2.0 pounds per square inch (psig) and ≤ +2.5 psig with no dependency on RWST temperature or containment temperature. The new LCOs 3.6.4.a and 3.6.4.b would require containment pressure to be maintained between ≥ -2.0 psig and ≤ +1.5 psig and ≥ -2.0 psig and ≤ +2.5 psig, the applicability of which will be determined by the RWST temperature and/or containment temperature;

(c) TS 3.6.3, "Containment Isolation Valves," to delete a containment leakage surveillance requirement that is redundant to the requirements of TS 5.5.15, "Containment Leakage Rate Testing Program,"; and

(d) TS 5.5.15, "Containment Leakage Rate Testing Program," to change the calculated peak internal pressure for the design basis loss-of-coolant accident (LOCA), P_a, from 42.0 psig to 42.38 psig.

The licensee stated that the proposed changes would address the mass and energy (M&E) release errors identified in Westinghouse's Nuclear Safety Advisory Letter, "Westinghouse LOCA Mass and Energy Release Calculation Issues," NSAL-11-05, dated July 25, 2011" (Reference 3). These issues affect the LOCA M&E release analysis results that are used as input to the containment integrity analyses.

The supplements dated August 21, 2013, and April 22, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration.

2.0 REGULATORY EVALUATION

The following explains the applicability of General Design Criteria (GDC) for IP3. The construction permit for IP3 was issued by the Atomic Energy Commission (AEC) on August 13, 1969, and the operating license was issued on December 12, 1975. The plant GDC are discussed in the Updated Final Safety Analysis Report (UFSAR) Chapter 1.3, "General Design Criteria," with more details given in the applicable UFSAR sections. The AEC published the final rule that added Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (FR) (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971. In accordance with an NRC staff requirements memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the Appendix A GDC to plants with construction permits issued prior to May 21, 1971. Therefore, the GDC which constitute the licensing bases for IP3 are those in the UFSAR.

As discussed in the UFSAR, the licensee for IP3 has made some changes to the facility over the life of the unit that committed to some of the GDCs from 10 CFR Part 50, Appendix A. The extent to which the Appendix A GDC have been invoked can be found in specific sections of the UFSAR and in other IP3 licensing basis documentation, such as license amendments.

The NRC staff's acceptance criteria for the primary containment functional design are based on the following GDCs in 10 CFR 50, Appendix A:

Criterion-4, "Environmental and dynamic effects design bases," insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and that such SSCs be protected against dynamic effects.

Criterion-16, "Containment design," insofar as it requires that the containment and associated systems be designed to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment, and to assure that the containment design conditions important to safety are not exceeded as long as postulated accident conditions require.

Criterion-38, "Containment heat removal," insofar as it requires that a containment heat removal system be provided and that its function shall be to reduce rapidly, consistent with the

functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Criterion-50, "Containment design basis," insofar as it requires that the containment and the heat removal systems be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated temperature and pressure conditions resulting from any LOCA.

NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," provides criteria for M&E release analyses.

SRP Section 6.2.1.1.A, "PWR [pressurized water reactor] Dry Containments, Including Subatmospheric Containments," provides criteria for containment integrity analysis for large dry containments.

The licensee cited a similar TS revision for Indian Point Nuclear Generating Unit No. 2 (IP2) that revised the RWST temperature limit in the stretch power uprate application that was approved by the NRC as a precedent for the IP3 LAR.

3.0 TECHNICAL EVALUATION

IP3 is a pressurized water reactor (PWR) having a large dry containment. The containment building is a reinforced concrete vertical right cylindrical structure with a flat base and hemispherical dome lined with a steel plate on the inside. The design objective of the containment structure is to contain all radioactive material, which might be released from the core following a LOCA. The containment serves as both a biological shield and a pressure container.

Mass and Energy Release Analysis

Nuclear Safety Advisory Letter 11-05 (Reference 3) identified Westinghouse methodology errors in the long-term M&E releases during a large break loss-of-coolant accident (LBLOCA). The licensee identified the following issues in the current long-term LOCA M&E release analysis that used the Westinghouse containment analysis methodology (Reference 4):

(a) The reactor vessel stored energy initially available in the M&E model was based on an incorrect mass of the vessel and barrel/baffle downcomer region.

(b) The long-term LOCA M&E release analysis was initialized at a low (non-conservative) Steam Generator (SG) secondary pressure condition.

(c) Due to errors identified by Westinghouse in the EPITOME computer code, the long term M&E release during the post-reflood phase of a LOCA was underestimated. In response to SCVB-RAI-7 (Reference 2), the licensee stated that at the time of peak pressure, the M&E release was underestimated by 2.88-percent, and at 3600 seconds following the accident, the M&E release was underestimated by about 1.92-percent.

The licensee confirmed, in a request for additional information (RAI) response, that the Main Steam Line Break (MSLB) accident was not impacted by the issues identified in Reference 3, and therefore there are no changes in the MSLB M&E release and containment response.

The NRC staff requested that the licensee list the computer codes used for the corrected M&E release analysis during the various phases of LOCA (blowdown, refill, reflood and post-reflood) and justify their use if they differ from those used in the current licensing basis analysis. In response to SCVB-RAI-2 (Reference 2), the licensee stated that all computer codes used in the proposed analysis are the same as those used in the current licensing basis analysis based on Reference 4 methodology except for the version number for the EPITOME code which is revised for error correction to address the issues in Reference 3.

Current Containment Integrity Analysis

The current containment integrity analysis considers the single failure of one Emergency Diesel Generator (EDG) concurrent with loss of offsite power (LOOP). It assumes the EDG is the most limiting failure which leaves available the minimum safeguards equipment consisting of one containment spray pump and four containment fan cooler units. With the initial RWST temperature of 110 °F, and containment pressure and temperature of 2.5 psig and 130 °F respectively, the current analysis gives a peak containment pressure of 40.38 psig for the Double-Ended Hot Leg (DEHL) break LOCA and 42.00 psig for the Double-Ended Pipe Suction (DEPS) break LOCA which are lower than the containment design pressure of 47 psig.

Proposed Containment Integrity Analysis

The proposed containment integrity analysis also assumes the most limiting failure of the same EDG concurrent with LOOP, and therefore credits the same equipment as in the current analysis. In accordance with the response to SCVB-RAI-3 (Reference 2), the licensee performed the analysis using the current licensing basis computer code COCO (Reference 5).

The NRC staff requested the licensee to compare the initial conditions used in the current most limiting large break LOCA analysis against the proposed analysis for the containment peak pressure and temperature response. The licensee was also requested to provide justification for any initial condition in the proposed analysis that is less conservative than the current analysis. In response to SCVB-RAI-1 (Reference 2), the licensee listed all the initial conditions and provided the following justification for those in which the conservatism is reduced in the proposed analysis: (a) the recirculation spray flow rate provided by the residual heat removal (RHR) pumps was reduced from 970 gallons per minute (gpm) to 960 gpm for consistency with the flow used in the offsite dose calculation, (b) the RHR heat exchanger overall heat transfer coefficient changed based on the emergency core cooling system (ECCS) flow during the various recirculation periods, (c) the closed cooling water (CCW) heat exchanger UA (overall heat transfer coefficient multiplied by the heat transfer area) has increased from 1.44 BTU/Hr-°F to 2.98 BTU/Hr-°F as a result of accounting for two CCW heat exchangers in operation post-LOCA and a small change related to GSI-191, "Experimental Studies of Loss-of-Coolant-Accident-Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation (NUREG/CR-6874, LA-UR-04-1227), (d) the CCW heat exchanger shell side flow decreased from 7420 gpm to 3980 gpm. The current analysis used two times 3710 gpm for two CCW heat exchangers in operation. According to the licensee, the actual CCW flow should have been 3710 gpm. The change from 3710 gpm to 3980 gpm is due to

updating of the current CCW analysis and for a planned enhancement of the CCW pump performance margin, (e) the CCW heat exchanger tube side flow has decreased from 7221 gpm to 5500 gpm accounting for conservatism in the service water pump performance, and (f) the total hot leg recirculation flow changed from 822 gpm to 1096 gpm, as a result of an error correction. The NRC staff finds the changes acceptable, as they represent operating experience, error corrections, future margin enhancements, and incorporation of results from GSI-191 calculations.

By using the corrected M&E release data and keeping initial conditions the same as in the current analysis, i.e. RWST temperature of 110 °F and containment pressure and temperature of 2.5 psig and 130 °F respectively, the licensee calculated the bounding peak containment pressure to be 44.26 psig for the DEPS break LOCA, which is lower than the containment design pressure of 47 psig.

However, the licensee desires to maintain the peak containment accident pressure similar to the value in the current analysis of record (42.0 psig) to reduce the impact on other programs. Therefore, the licensee performed peak containment pressure sensitivity studies to evaluate the reduction in peak containment accident pressure that can be achieved by decreasing: (a) initial RWST temperature, (b) initial containment/accumulator temperature, and (c) initial containment pressure. All sensitivity studies were based on the corrected M&E release data. Further evaluations by the licensee were based on the following results:

(a) Assuming an initial RWST temperature of 105 °F, initial containment pressure and temperature of 1.5 psig and 130 °F respectively, the licensee calculated the peak containment pressure to be 39.71 psig for the DEHL break LOCA and 42.38 psig for the DEPS break LOCA.

(b) Assuming an initial RWST temperature of 95 °F, initial containment/accumulator temperature of 125 °F, and initial containment pressure of 2.5 psig, will result in a peak containment pressure of 42.31 psig.

Based on the evaluations, the licensee proposed changes to the following TSs:

<u>TS 3.5.4 "Refueling Water Storage Tank (RWST)", SR 3.5.4.1</u>

Surveillance Requirement (SR) 3.5.4.1 requires verification of RWST borated water temperature to be ≥ 35 °F and ≤ 110 °F every 24 hours with a note stating that verification is not required to be performed when ambient temperature ≥ 35 °F and ≤ 110 °F if the heating steam supply isolation valves are locked closed.

The licensee is proposing to revise the acceptable range of the RWST borated water temperature to be ≥ 35 °F and ≤ 105 °F. The licensee is also proposing a change to the note to state that verification is not required to be performed when the ambient air temperature is ≥ 35 °F and ≤ 95 °F if the heating steam supply isolation valves are locked closed. The frequency of verification will remain at every 24 hours.

TS 3.6.3, "Containment Isolation Valves," SR 3.6.3.9

SR 3.6.3.9 requires verification that the combined leakage rate for all containment bypass leakage paths is ≤ 0.6 L_a when pressurized ≥ 42.42 psig and that this be performed in accordance with the Containment Leakage Rate Testing Program. The licensee proposes to delete SR 3.6.3.9 because it is redundant to the containment leakage rate testing specified in TS 5.5.15, "Containment Leakage Rate Testing Program."

TS 3.6.4 "Containment Pressure", LCO 3.6.4

LCO 3.6.4 currently states that the containment pressure shall be maintained ≥ -2.0 psig and ≤ +2.5 psig.

The licensee proposed change to LCO 3.6.4 reads as follows:

"Containment pressure shall be maintained as follows:

a. If RWST temperature is > 95 °F or containment temperature is > 125 °F, containment pressure shall be ≥ -2.0 psig and ≤ +1.5 psig.

b. If RWST temperature is ≤ 95 °F and containment temperature is ≤ 125 °F, containment pressure shall be ≥ -2.0 psig and ≤ +2.5 psig."

TS 5.5.15 "Containment Leakage Rate Testing Program"

The licensee is proposing a change to the calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, from 42.0 psig to 42.38 psig.

Evaluation of the Proposed TS Changes

The proposed change to SR 3.5.4.1 to modify the upper limit of RWST borated water temperature verification from ≤ 110 °F to ≤ 105 °F is acceptable to the NRC staff because 105 °F represents the maximum limit of the RWST water temperature assumed in the containment integrity analysis. The decrease in the temperature from ≤ 110 °F to ≤ 95 °F in the note associated SR 3.5.4.1 is also acceptable to the staff because the proposed revision to LCO 3.6.4 requires initial containment pressure to be reduced from ≤ 2.5 psig to ≤ 1.5 psig when RWST temperature is >95 °F.

The proposed LCOs 3.6.4.a and 3.6.4.b are acceptable because sensitivity studies demonstrated that the peak containment accident pressure, P_a, of 42.38 psig, that is proposed to be included in TS 5.5.15, is shown to bound all combinations of initial RWST temperature, initial containment temperature, and initial containment pressure that are within the limitation of the new LCO.

The proposed revision to change the calculated peak containment internal pressure (P_a) in TS Section 5.5.15, "Containment Leakage Rate Testing Program", from 42.0 psig to 42.38 psig is acceptable because it represents the maximum pressure calculated for the design basis LOCA using a range of initial containment conditions that are controlled and monitored by TSs. The licensee stated that the increase in the peak containment pressure does not affect the accident

dose because the pressures used for the 10 CFR 50 Appendix J Type A, B and C tests are currently performed at pressure above 42.38 psig. The NRC staff finds this acceptable.

The licensee also proposed to delete SR 3.6.3.9 which requires verification that the combined leakage rate for all containment bypass leakage is ≤ $0.6L_a$. The licensee noted that SR 3.6.3.9 is redundant because containment bypass leakage limits are part of the overall containment leakage rate testing program that is described in TS 5.5.15. The NRC staff agrees with the licensee's assessment that SR 3.6.3.9 is redundant to TS Section 5.5.15 and, therefore, finds its removal acceptable.

TS 3.6.5, "Containment Air Temperature," requires that the containment air temperature shall remain between 50 °F and 130 °F. The licensee stated that the containment air temperature limit in TS LCO 3.6.5 does not need to be lowered to 125 °F to match the limit of TS 3.6.4 because the calculated containment peak pressure of 42.38 psig is met assuming an initial containment temperature of 130 °F, a RWST temperature of 105 °F and an initial containment pressure of 1.5 psig. The NRC staff finds this acceptable.

Containment Pressure Monitoring

The licensee stated that containment pressure monitoring during normal operation is currently performed by control room instruments which has an uncertainty of ± 1.5 psi. Based on this indication, containment purging at ≤ 1 psi satisfies the current accident analysis initial containment pressure of 2.5 psig. However, the proposed revision to LCO 3.6.4 requires initial containment pressure to be ≤ +1.5 psig when the RWST temperature is >95 °F or containment/accumulator temperature is >125 °F. Therefore, containment pressure indication with a higher accuracy instrument is necessary for maintaining the accident initial condition. Under these conditions, the licensee intends to monitor and control by a locally mounted higher accuracy containment pressure indicator which will have an uncertainty of no more than ± 0.5 psi. With this improved uncertainty, the accident analysis initial containment pressure of 1.5 psig will be satisfied with containment purging at ≤ 1.0 psi. The licensee stated that the proposed TS change allows continuous monitoring of the containment pressure from the control room with the exception of a few hot summer days when the RWST temperature is >95 °F or containment/accumulator temperature is >125 °F which would require operators to monitor the containment pressure from the locally mounted instrument.

In response to staff RAIs, the licensee provided additional information concerning the safety classification and the impact on operator actions imposed by the revised LCOs as follows (Reference 6):

The local pressure monitoring gauges are safety-related and installed at elevation 41 feet of the fan house in the Primary Auxiliary Building (PAB). The pressure gauges will be provided with root isolation valves that can be closed to isolate the gauges from their respective loops in the event of a failure or if maintenance is required. The pressure gauges will provide operations with a more accurate means of monitoring containment pressure, when needed, to maintain containment pressure at ≤ +1.5 psig (≤ 1.0 psig indicated) by performing containment pressure reliefs. The system operating procedures for Containment Pressure Relief and Purge Systems operation and Containment Recirculation Fan Cooler Unit Operators will be revised to implement the proposed changes.

The proposed TSs requires operator actions to monitor containment temperature and RWST temperature. The containment temperature is monitored and logged every 12 hours by the Reactor Operator from an indicator in the control room. The RWST temperature is monitored and logged every 12 hours by the Nuclear Plant Operator (NPO) during the course of Field TS Rounds. If the containment temperature or RWST temperature exceed pre-designated values, the Control Room Supervisor will be notified to initiate a special log to monitor containment pressure using the new local gauges installed in the PAB and to initiate additional actions to maintain containment pressure in accordance with the proposed LCO 3.6.4. There is no additional impact on time available for operators to complete the actions, since the control room rounds and NPO rounds are already routine actions taken every 12 hours. The local pressure indicators are on the route of the NPO rounds, and would not be a burden on the operator time to complete the action.

The training related actions taken as a result of the proposed changes include revisions to the Vapor Containment System Description Training Module and Operator Rounds. In addition, the Licensed Operator Requalification Training Lesson Plan includes information on the pending TS change.

Based on the information provided by the licensee (Reference 6), the NRC staff concludes that the impact on operator actions imposed by the proposed changes is minimal, and therefore, acceptable.

Additional Considerations of the Proposed Containment Integrity Analysis

The NRC staff requested the licensee to describe the impact of changes in the M&E input to the containment gas temperature response for Equipment Environment Qualification (EEQ). In response to SCVB-RAI-4c (Reference 2), the licensee stated that in the proposed analysis, the LOCA peak containment gas temperature for EEQ is determined to be 262.95 °F, compared to the current containment peak gas temperature of 260.4 °F. The licensee stated that the temperature determined in the proposed analysis is increased due to the M&E release error correction in EPITOME computer code. The licensee further evaluated the slight increase in temperature for LOCA equipment qualification and concluded that all equipment in the containment remains environmentally qualified.

The NRC staff requested the licensee to describe the impact of changes in the M&E release on the containment wall temperature response under a LOCA. In response to SCVB-RAI-4d (Reference 2), the licensee stated that as in the current analysis documented in the FSAR, the LOCA containment uninsulated steel wall temperature response was analyzed with different initial conditions than those in design basis analysis. The analysis determined that the peak uninsulated wall temperature increased by 3.3 °F. The current peak containment uninsulated liner temperature is 234.8 °F which is 12.2 °F below its maximum limit of 247 °F. Due to sufficient margin of 12.2 °F compared to 3.3 °F increase in the uninsulated wall temperature, the licensee did not perform the specific containment liner response analysis and concluded that the liner temperature limit would be met for the corrected M&E release. The staff concludes that any impact on the LOCA containment wall liner temperature response would not result in temperature of the liner exceeding the design limit.

The NRC staff requested the licensee to describe the impact of the changes in M&E input to the following containment analyses: (a) Sump water temperature response following a LOCA, and

(b) Net Positive Suction Head (NPSH) analysis for containment spray and safety injection pumps during the recirculation phase. In response to SCVB-RAI-5 (Reference 2), the licensee provided a comparison of sump water temperature response of the current and the proposed analysis for a large break LOCA with minimum ECCS flows. The comparison showed that following the peak, the sump water temperature is slightly lower in the proposed analysis. The licensee stated that the proposed analysis does not impact the Generic Safety Issue (GSI)-191 analyses.

During the recirculation phase of a LOCA, the recirculation and RHR pumps provide containment spray and cold leg recirculation flow along with the high head safety injection pumps. In the proposed LOCA containment analysis the maximum sump temperature in the recirculation mode is determined to be 241.5 °F which is less than the sump temperature of 242.8 °F assumed for the recirculation and RHR pumps available NPSH analysis. The NPSH analysis did not credit Containment Accident Pressure (CAP) above the saturation pressure at the pump water temperature. The licensee determined that the proposed analysis does not impact the GSI-191 NPSH analyses because the change in sump water level in the proposed analysis was small and bounded by the water level assumed in the GSI-191 NPSH analysis.

The NRC staff requested the licensee to describe the impact of the changes in the M&E release on the minimum containment pressure analyses for ECCS performance using the guidance in NUREG-0800, SRP 6.2.1.5, and Regulatory Guide (RG) 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," Section 3.12.1. In response to SCVB-RAI-6 (Reference 2), the licensee stated that the Reference 4 methodology codes, including EPITOME, are not used in performing either Appendix K large break analyses or the best estimate large break analyses and, therefore, the issues identified in Reference 3 have no effect on methods used to calculate the minimum containment backpressure for the ECCS calculation.

TS Limiting Condition for Operation (LCO) 3.6.5 for IP2 and IP3 specifies 50 °F and 130 °F as the range of containment average air temperature during operating Modes 1 through 4. For both IP2 and IP3, the NRC staff requested the licensee to provide results of large break LOCA peak containment pressures, including pressure profiles, for initial containment air temperatures of 50 °F and 130 °F while assuming the remaining input parameters and assumptions remain the same in both cases. The licensee performed the sensitivity study for IP2 (Reference 6) and determined that assuming an initial containment air and heat sinks temperature of 130 °F resulted in a higher peak pressure compared to the peak pressure obtained assuming 50 °F as the initial containment air and heat sink temperature. The licensee stated that while the colder initial air temperature does contribute to a higher air partial pressure, it is more than offset by the colder heat sinks, which are able to absorb a greater amount of heat over the long-term transient. In the analysis, the licensee assumed RWST and service water temperatures at their TS maximum allowable values and stated that it is conservative to use maximum values because the lower temperatures would enhance the performance of the containment spray system and the fan cooler units and would result in a lower peak containment pressure. The licensee stated that a similar result would be expected for IP3. The NRC staff finds this trend acceptable because IP2 and IP3 have essentially the same design and have similar containments.

Based on the above evaluations, the NRC staff has determined that the revised analyses, after correcting the M&E release errors, and changes based on operating experience, future margin enhancements, and incorporation of results from GSI-191 calculations are acceptable because

the licensee used NRC approved methodology and conservative inputs and assumptions and provided adequate justification for changes in the inputs and assumptions from the current licensing basis analysis. Accordingly, the staff finds the proposed TS changes acceptable.

Conclusion

The NRC staff concludes that the proposed changes, after correcting the M&E release data, meet the requirements of 10 CFR Part 50 Appendix A, (1) Criterion 4, because the licensee showed that SSCs important to safety are designed to accommodate the effects of, and are compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and that such SSCs are protected against dynamic effects, (2) Criterion 16, because the licensee showed that the reactor containment and associated systems are designed to establish an essentially leak-tight barrier and the containment design conditions important to safety are not exceeded during a postulated design-basis accident, (3) Criterion 38, because the licensee showed that the containment heat removal system would reduce the containment pressure and temperature rapidly, following a design-basis accident and would maintain them at acceptable levels and (4) Criterion 50, because the licensee showed that the containment heat removal system is designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a design basis LOCA.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (78 FR 19750). 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.

6.0 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) there is reasonable assurance that 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.

Principal Contributor: Ahsan Sallman, NRR
Ervin Geiger, NRR

Date: July 17, 2014

7.0 REFERENCES

1. Letter from Entergy to NRC dated January 28, 2013, "Proposed TS Changes Regarding RWST Temperature and Containment Pressure in Containment Integrity Analysis Indian Point Unit Number 3 Docket No. 50-286 License No. DPR-64", (ADAMS Accession Number ML13042A224).

2. Letter from Entergy to NRC dated August 21, 2013, "Response to Request for Additional Information Regarding Containment Integrity Analysis (TAC NOS. MF0590 and MF0591) Indian Point Unit Numbers 2 & 3 Docket Nos. 50-247 and 50-286 License Nos. DPR-26 and DPR-64", (ADAMS Accession Number ML13239A477).

3. NSAL-11-05, "Westinghouse LOCA Mass and Energy Release Calculation Issues," July 26, 2011 (ADAMS Accession Number ML13239A479).

4. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design", May 31, 1983, (ADAMS Accession Number ML080640615).

5. WCAP-8327, "Containment Pressure Analysis Code (COCO)", July 31, 1974, (ADAMS Accession Number ML092460709).

6. Letter from Entergy to NRC dated April 22, 2014, "Response to Request for Additional Information Regarding Containment Integrity Analysis (TAC NOS. MF0590 and MF0591) Indian Point Unit Numbers 2 and 3 Docket Nos. 50-247 and 50-286 License Nos. DPR-26 and DPR-64". (ADAMS Accession Number ML14121A113).

Vice President, Operations
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Indian Point Energy Center
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Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 - ISSUANCE OF AMENDMENT RE: REVISED CONTAINMENT INTEGRITY ANALYSIS (TAC NO. MF0591)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 253 to Facility Operating License No. DPR-64 for Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated January 28, 2013, as supplemented by letters dated August 21, 2013, and April 22, 2014.

Westinghouse Nuclear Safety Advisory Letter 11-5 identified methodology errors in the long-term mass and energy release rates during a large break loss-of-coolant accident. These errors impacted the containment integrity analysis for Indian Point Unit No. 3 and necessitated revisions to limiting initial operating conditions (i.e., containment temperature, containment pressure, and refueling water storage tank temperature) resulting in changes to TS 3.5.4, "Refueling Water Storage Tank (RWST)," and 3.6.4, "Containment Pressure." In addition, the amendment revised TS 3.6.3, "Containment Isolation Valves," to delete a redundant surveillance requirement and TS 5.5.15, "Containment Leakage Rate Testing Program," to reflect a slightly higher calculated containment peak pressure.

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,
/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-286
Enclosures:
1. Amendment No. 253 to DPR-64
2. Safety Evaluation
cc w/encls: Distribution via Listserv

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ADAMS ACCESSION NO.: ML14169A583

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