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

February 24, 2010

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. 2 - ISSUANCE OF AMENDMENT REGARDING BATTERY CAPACITY SURVEILLANCE REQUIREMENT (TAC NO. ME0985)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 264 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 29, 2009, as supplemented by letters dated September 21 and December 22, 2009.

The amendment established a more restrictive acceptance criterion for surveillance requirement (SR) 3.8.6.6 regarding periodic verification of capacity for the affected station batteries.

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,

John P. Boska

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures:
1. Amendment No. 264 to DPR-26
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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. 264
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 March 29, 2009, as supplemented by letters dated September 21 and December 22, 2009, 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. 264, 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 its date of issuance and shall be implemented within 30 days of issuance. Implementation of the amendment shall include updating the Updated Final Safety Analysis Report and other applicable documents, such as the battery program prescribed by Technical Specification (TS) 5.5.15 and the TS Bases for the batteries. These updates shall include the reduction to 18 years of the expected service life for the batteries.

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Nancy L. Salgado, 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: February 24, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 264

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

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 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 Page	Insert Page
3.8.6-4	3.8.6-4

instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(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. 42 10-17-78

(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. 220 09-06-01

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; 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. Amdt. 241 10-27-04

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 264, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

(3) The following conditions relate to the amendment approving the conversion to Improved Standard Technical Specifications:

1. This amendment authorizes the relocation of certain Technical Specification requirements and detailed information to licensee-controlled documents as described in Table R, "Relocated Technical Specifications from the CTS," and Table LA, "Removed Details and Less Restrictive Administrative Changes to the CTS" attached to the NRC staff's Safety Evaluation enclosed with this amendment. The relocation of requirements and detailed information shall be completed on or before the implementation of this amendment.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.5	Verify each battery connected cell voltage is $\geq$ 2.07 V.	92 days
SR 3.8.6.6	- NOTE - This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. Verify battery capacity is $\geq$ 85% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.	60 months AND 12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating AND 24 months when battery has reached 85% of the expected life with capacity $\geq$ 100% of manufacturer's rating

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 264 TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR OPERAT!ONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated March 29, 2009, Agencywide Documents Access and Management System (ADAMS) Accession No. ML090980300, as supplemented by letters dated September 21 and December 22, 2009, ADAMS Accession Nos. ML093010534 and ML093631146, respectively, Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs). The supplements dated September 21 and December 22, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration. The proposed change would establish a more restrictive acceptance criterion for Surveillance Requirement (SR) 3.8.6.6 regarding periodic verification of capacity for the affected station batteries.

2.0 REGULATORY EVALUATION

The following explains the applicability of general design criteria (GDC) for IP2. The construction permit for IP2 was issued by the Atomic Energy Commission (AEC) on October 14, 1966, and the operating license was issued on September 28, 1973. The plant GDC are listed 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* (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 IP2 are those in the UFSAR.

As discussed in the UFSAR, the licensees for IP2 have made some changes to the facility over the life of the unit that have committed to some of the GDCs from 10 CFR Part 50, Appendix A. The extent to which the Appendix A GDCs have been invoked can be found in specific sections of the UFSAR and in other IP2 licensing basis documentation, such as license amendments.

Based on a review of UFSAR Section 8.1, the NRC staff identified UFSAR GDCs 24 and 39 as being applicable to the proposed amendment:

- "An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single component. (GDC 39 and GDC 24)."

The following NRC requirements and guidance document are also applicable to the staff's review of the licensee's amendment request:

Paragraph 50.36(c)(2)(ii) of 10 CFR, "Technical specifications," requires that "[a] technical specification limiting condition for operation [LCO] of a nuclear reactor must be established for each item meeting one or more of the [criteria set forth in 10 CFR 50.36(c)(2)(ii)(A)-(D)]."

Paragraph 50.36(c)(3) of 10 CFR, "Technical specifications," requires that TSs include Surveillance Requirements (SRs), which "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

While IP2 is not currently committed to Regulatory Guide (RG) 1.212, "Sizing of Large Lead-Acid Storage Batteries," the NRC staff used RG 1.212 as a technical reference during its review of the license amendment request. RG 1.212 describes a method that the NRC staff considers acceptable for use in complying with requirements and regulations on the criteria for the sizing of large lead-acid storage batteries for use in nuclear power plants.

3.0 TECHNICAL EVALUATION

3.1 Background and Evaluation

IP2 has four separate safety-related 125 volt (V) direct current (DC) systems that serve various DC loads throughout the station. Each system consists of one battery, one battery charger, one main power panel, and one or more DC distribution panels (sub panels). Each of the four batteries is composed of 58 lead-calcium storage cells connected to provide a nominal terminal voltage of 125 V DC.

Each battery charger is supplied from a different 480 V alternating current (AC) switchgear bus. Under normal conditions, the battery charger supplies the DC loads and float charges the battery. The IP2 UFSAR states that the battery provides power to the DC loads under the following conditions:

(a) When the load exceeds the capacity of the battery charger, such as during DC motor starting or simultaneous breaker operation.

(b) When the battery charger is not available, such as a battery charger failure or loss of input voltage.

According to the IP2 UFSAR, each battery has been sized to carry its expected shutdown loads for a period of at least 2 hours following a plant trip and a loss of all AC power. All equipment supplied by the batteries is maintained operable with minimum expected voltages at the battery terminals during the 2 hours. Each of the four battery chargers has been sized to recharge its own discharged battery within 15 hours while simultaneously carrying its normal load.

Each battery is maintained under continuous charge by its associated self-regulating battery charger so that the batteries will always be at full charge in anticipation of a loss of AC power incident. This ensures that adequate DC power will be available for starting and loading the emergency diesel generators and for other emergency uses.

TS SR 3.8.6.6 currently requires the licensee to verify the battery capacity is greater than or equal to 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.

During an engineering review, the licensee determined that the 80% value was non-conservative with respect to the existing design basis calculation for battery capacity under minimum design temperature conditions. Since battery performance is affected by, among other factors, the environmental temperature in the vicinity of the batteries, the licensee's engineering review concluded that the battery capacity acceptance criterion for TS SR 3.8.6.6 should be changed from greater than or equal to 80% to greater than or equal to 85%. The licensee performed a review of the current surveillance test results to verify that the more restrictive acceptance criterion was met and controls have been established in accordance with Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety," ADAMS Accession No. ML031110108, pending approval and implementation of this license amendment request.

Based on this information, the licensee proposed revising TS SR 3.8.6.6 to require verification that the battery capacity is greater than or equal to 85% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.

By letter dated July 23, 2009, the staff requested that the licensee provide the full calculation for the proposed capacity rating, including any assumptions and supporting documentation, to show that the 125 V DC batteries will perform their intended design functions.

In its response to the staff request for additional information, the licensee stated that the change to 85% battery capacity was originally determined by calculation FEX-00062-01, "Minimum Operating Electrolyte Temperature for 125 V DC Batteries 21, 22, 23, and 24," and provided margin beyond the test procedure acceptance criteria of 90%. This calculation also established a minimum electrolyte temperature of 59 degrees Fahrenheit. The licensee included calculation FEX-00062-01 in its response. The licensee also stated that the current DC system calculations use the same 85% capacity for battery sizing. The licensee provided system calculation FEX-00204, "Station Battery 22 System Calculation," as a sample. According to the licensee, the purpose of this calculation is to ensure the intended design functions of the station Battery 22 system can be met.

The NRC staff's review of these calculations was limited to the general battery loading assumptions and the battery cell sizing worksheets for Battery 22, as all four battery calculations were done using the same methodology. If the methodology is found to be acceptable for

Battery 22, it will also be acceptable for the other batteries. The staff reviewed specific areas to verify that the assumptions were consistent with the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 485-1997, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," which is endorsed by RG 1.212.

The recommended guidance in IEEE Std. 485-1997 is to use a 25% aging factor, a 5% design margin, and a 5% temperature correction factor when sizing a vented lead-acid battery. The licensee for IP2 applied a 17.6% aging factor versus 25%. The 17.6% aging factor corresponds to an end-of-life criterion of 85% battery capacity instead of the 80% recommended by IEEE Std. 485-1997. While the aging factor is less than that recommended by IEEE Std. 485-1997, the methodology was followed since the aging factor is appropriate for the assumed end-of-life criterion (i.e., 85% battery capacity).

The licensee's analysis also noted that the licensee would replace a battery the next refueling outage (two-year frequency) after the battery reaches 90% capacity. In delaying the battery replacement until the next refueling outage, the licensee is assuming that the battery capacity will not reach 85% capacity by the time they replace the battery. In response to a staff request for additional information, the licensee stated that this replacement criterion is based on the manufacturer's battery life curves and operational experience. The staff notes that according to IEEE Std. 485-1997, a vented lead-acid battery remains relatively stable throughout most of its life and typically reaches the 'knee' in the battery life versus performance curve when the capacity is 80%. Based on this information, the staff finds that there is reasonable assurance that a battery with 90% capacity can perform its design function for an additional 2 years provided the battery is not showing signs of degradation or has reached 85% of the expected service life. According to IEEE Std. 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," degradation is indicated when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is below 90% of the manufacturer's rating. If either of these conditions were present, the licensee would be required to increase surveillance testing to annually in accordance with the criteria specified TS SR 3.8.6.6.

The NRC staff requested that the licensee describe the impact of the proposed change on the expected life of the IP2 batteries. In its response, the licensee stated that IP2 safety-related Batteries 21 & 22 are Enersys (Exide) models GN-17 and GN-23 and that Batteries 23 & 24 are C&D Technologies model KCR-13. The licensee further stated that these batteries are qualified for 20 years of life when aged to 80% of rated capacity. Based on its review of standard battery life versus capacity curves, which indicate average cell performance, the licensee noted that aging these batteries to 85% of rated capacity would indicate an expected battery life of approximately 18 years. The licensee's discussions with both battery manufacturers revealed that a number of factors affect battery life (e.g., maintenance and discharge cycling), but in general they agree with the standard industry accepted battery life curve for flooded type cells. In reviewing this information, the NRC staff concludes that the expected service life of the IP2 safety-related batteries shall be considered 18 years as a result of the proposed change. The licensee must recognize that the change in expected service life has a direct impact on the testing requirements prescribed by TS SR 3.8.6.6, which requires specific test frequency criteria based on the expected service life of the safety-related batteries. Specifically, TS SR 3.8.6.6 currently requires verification that battery capacity is ≥ 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test. The test frequency is as follows:

60 months

AND

12 months when battery shows degradation, or has reached 85% of the expected life with capacity < 100% of manufacturer's rating

AND

24 months when battery has reached 85% of the expected life with capacity ≥ 100% of manufacturer's rating

During its review of calculation FEX-00204-01, the NRC staff noticed that the available (i.e., excess) capacity of Battery 22 decreased from 57.1% to 12.9% (page 33 of 34). The staff's initial understanding was that the licensee primarily revised this calculation to address the change in minimum design temperature (i.e., 60 degrees Fahrenheit to 59 degrees Fahrenheit). The staff requested that the licensee provide a detailed discussion on why the available capacity significantly decreased. In its letter dated December 22, 2009, the licensee stated that the available excess capacity decreased due to changes in minimum design temperature, end-of-life battery capacity (90% to 85%), and additional loading to account for automatic transfer switches. The biggest change was due to additional loading, followed by an aging factor increase from 11% to 17.6%. Based on this information, the NRC staff finds that it is reasonable to expect a drop in available battery capacity of this magnitude as a result of the aforementioned changes.

Based on the review of the calculations, the NRC staff finds that the licensee's battery sizing technique is consistent with the guidance provided in RG 1.212 and that the more restrictive acceptance criterion of greater than or equal to 85% of manufacturer's rating for TS SR 3.8.6.6 will ensure that sufficient battery capacity exists at limiting conditions.

Based on the above, the NRC staff finds that there is reasonable assurance that safe plant conditions will continue to be maintained; therefore, the proposed change is acceptable.

3.2 Implementation

The expected service life of the IP2 safety-related batteries shall be considered 18 years as a result of this proposed change. Since the expected service life affects when TS SR 3.8.6.6 is performed, this information needs to be included in the applicable documents when the amendment is implemented, rather than the normal document update period. The applicable documents include the IP2 UFSAR, the licensee's battery monitoring and maintenance program prescribed by TS 5.5.15, and the TS Bases for the batteries. Furthermore, all associated documentation for the safety-related batteries shall be revised to show the reduction in expected service life based on the proposed change.

3.3 Conclusion – Technical Evaluation

The NRC staff evaluated the licensee's request to modify TS SR 3.8.6.6 by establishing a more restrictive acceptance criterion regarding periodic verification of battery capacity.

Based on the above evaluation, the NRC staff concludes the proposed revision to the IP2 TSs provides reasonable assurance of the continued availability of the required power to shut down the reactor and to maintain the reactor in a safe condition after an anticipated operational occurrence or a postulated design-basis accident. The staff also concludes that the proposed TS change is in accordance with 10 CFR 50.36 and the requirements of UFSAR GDCs 24 and 39. Therefore, the NRC staff finds the proposed change acceptable.

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 (74 FR 23444). 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) 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: M. McConnell

Date: February 24, 2010

February 24, 2010

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

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT REGARDING BATTERY CAPACITY SURVEILLANCE REQUIREMENT (TAC NO. ME0985)

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/RA/

John P. Boska, Senior Project Manager
Plant Licensing Branch I-1
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Docket No. 50-247

Enclosures:
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ADAMS ACCESSION NO.: ML100270051 *See memo dated 1/21/10

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