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

April 2, 2015

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 RE: PROPOSED LICENSE AMENDMENT REGARDING A CHANGE TO TECHNICAL SPECIFICATION 3.1.4 "REACTIVITY CONTROL SYSTEMS" (TAC NO. MF5747)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 280 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 February 12, 2015, as supplemented by letters dated March 10 and April 1, 2015.

The amendment revises the TSs by changing the acceptance criteria for Surveillance Requirement (SR) 3.1.4.2 for Control Rod G-3. The change includes a footnote to the SR stating that Control Rod G-3 need not be moved until repaired in the next forced outage of sufficient duration prior to the refuel outage of 2016 or during the refuel outage in 2016.

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. 50-247

Enclosures:
1. Amendment No. 280 to DPR-26
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY NUCLEAR INDIAN POINT 2, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

AND TECHNICAL SPECIFICATIONS

Amendment No. 280
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 February 12, 2015, as supplemented on March 10 and April 1, 2015, 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, B and C, as revised through Amendment No. 280, 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

NJ Pannuthi for

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: April 2, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 280

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
3.1.4-3	3.1.4-3

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, B, and C, as revised through Amendment No. 280, 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.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is within the limits specified in the COLR.	1 hour
	<u>OR</u>	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	-- **– NOTE –** Not required to be met for individual control rods until 1 hour after completion of control rod movement. -- Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod* not fully inserted in the core $\geq$ 10 steps in one direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 2.4 seconds from the gripper release to dashpot entry, with: a. $T_{avg} \geq 500°F$ and b. All reactor coolant pumps operating.	Prior to criticality after each removal of the reactor head

*Control Rod G-3 need not be moved until repaired in the next forced outage of sufficient duration prior to the refuel outage of 2016 or during the refuel outage in 2016.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 280

TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR INDIAN POINT 2, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

1.0 INTRODUCTION

By letter dated February 12, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15044A471), as supplemented by letters dated March 10 (ADAMS Accession No. ML15076A009), and April 1, 2015, Entergy Nuclear Operations, Inc. (Entergy, the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs). The amendment revises the TSs by changing the acceptance criteria for Surveillance Requirement (SR) 3.1.4.2 for Control Rod G-3. The change includes a footnote to the SR stating that Control Rod G-3 need not be moved until repaired in the next forced outage of sufficient duration prior to the refuel outage of 2016 or during the refuel outage in 2016.

The supplemental letters dated March 10 and April 1, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration.

2.0 REGULATORY EVALUATION

The following explains the applicability of General Design Criteria (GDC) to 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 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* (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

Enclosure

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 licensee for IP2 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 IP2 licensing basis documentation, such as license amendments.

2.1 Applicable Regulatory Requirements

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls.

SRs in 10 CFR 50.36 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 LCO will be met.

As stated in 10 CFR 50, Appendix A, the GDC for nuclear power plants applicable to CR design requirements are as follows:

> GDC 26, "Reactivity control system redundancy and capability," states that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions;
>
> GDC 26 is comparable to the version of GDC 27 that was issued for comment by the Atomic Industrial Forum on July 11, 1967. The 1967 version of GDC 27 is part of the licensing basis for IP2 and is discussed in IP2 UFSAR Section 7.2.1.9. GDC 27 states "Two independent control systems, preferably of different principles, shall be provided."
>
> GDC 27, "Combined reactivity control systems capability," states that the reactivity systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained;

GDC 27 is comparable to the version of GDC 29 that was issued for comment by the Atomic Industrial Forum on July 11, 1967. The 1967 version of GDC 29 is part of the licensing basis for IP2 and is discussed in IP2 UFSAR Section 9.2.1.4. GDC 29 states "One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn."

3.0 TECHNICAL EVALUATION

3.1 System Description

The Rod Cluster Control Assemblies (RCCAs) for IP2 are used for controlling reactor temperature and power distribution within the core. The RCCAs are divided into two sub groups, Control Bank and Shutdown Bank Control Rods. The CR G-3 is in Shutdown Bank B. The primary function of Shutdown Bank RCCAs is to release the CRs, which fall by gravity to the bottom of the core, in response to manual or automatic reactor trip signals.

3.2 Proposed Changes

SR 3.1.4.1.2 states the following:

> Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in one direction.

SR 3.1.4.1.2 would be revised to state the following:

> Verify rod freedom of movement (trippability) by moving each rod* not fully inserted in the core ≥ 10 steps in one direction.

A footnote will be added to TS page 3.1.4-3 to state the following:

> *Control Rod G-3 need not be moved until repaired in the next forced outage of sufficient duration prior to the refuel outage of 2016 or during the refuel outage in 2016.

3.3 Summary of Technical Information Provided by Licensee

The licensee requested to revise the IP2 TSs. SR 3.1.4.2 requires that control rods not fully inserted into the core be moved ≥ 10 steps in one direction once every 92 days. The licensee requested to defer testing of CR G-3.

The licensee stated that during the September 18, 2014, CR surveillance testing in accordance with TS SR 3.1.4.2, CR G-3 inserted to 195 steps with Shutdown Bank B demanding 218 steps. Performing the surveillance testing on December 11, 2014, CR G-3 inserted to 208 steps with Shutdown Bank B demand at 221 steps. The licensee stated that there is industry history of intermittent rod misalignment issues that have been attributed to corrosion related unidentified deposit (CRUD). CRUD is defined as corrosion and wear products, i.e., rust particles, which become radioactive when exposed to radiation.

The licensee stated that the purpose of SR 3.1.4.2 is to demonstrate that the CRs are operable to perform their design function of tripping and dropping by gravity into the core. The licensee evaluated the degrading condition to avoid a rod drop accident or plant transient. The licensee concluded that CR G-3 remains trippable and is expected to remain so.

TS 3.1.1, "Shutdown Margin (SDM)," specifies that the SDM shall be greater than or equal to that specified in the Core Operating Limits Report (COLR) when in MODE 2 with $k_{eff} < 1.0$, and in MODES 3, 4, and 5. The COLR for IP2 states that the SDM shall be greater than or equal to 1.3% $\Delta k/k$. The licensee performed an analysis to supplement this application to verify that the required SDM is maintained if a reactor scram occurred and CR G-3 and the control rod of greatest reactivity worth did not insert in the core.

3.4 Summary of NRC Staff Review

The NRC staff reviewed the application and determined that additional information was required regarding the evaluation of the request. In a letter dated March 10, 2015, the licensee responded to the NRC staff's request for additional information (RAI) regarding Entergy's proposal to defer testing of CR G-3 in accordance with SR 3.1.4.2.

The NRC staff requested that the license confirm that the SDM analysis was performed using NRC-approved methodologies for IP2 by assuming that both CR G-3 and the next highest reactive rod fail to insert into the reactor core following a reactor scram. The licensee confirmed that the SDM analysis was determined using previously approved NRC methodologies and analytic codes. The minimum calculated SDM was found to exceed the minimum required TS SDM of 1.3% $\Delta k/k$ for MODE 2 with $k_{eff} < 1.0$, and in MODES 3, 4, and 5. Based on the considerations discussed above, the staff determined that the licensee adequately determined that SDM is maintained with CR G-3 and the next highest reactive rod not inserting into the reactor core using NRC-approved methodology. Therefore, the staff concludes that the licensee's SDM analysis adequately addressed the intent of GDC 27, "Combined reactivity control systems capability."

The NRC staff requested that the licensee provide possible causes, other than CRUD, for the difficulties encountered when performing SR 3.1.4.2. The licensee responded that the following possible causes, other than CRUD, for CR G-3 misalignment during surveillance testing are: 1) improper engagement of the Control Rod Drive Mechanism (CRDM) moveable gripper assembly during rod motion, 2) insufficient current to the lift coil to allow proper engagement of the moveable gripper, 3) intermittent pin connection on the connector located on the reactor head, 4) intermittent electrical connection of the cable/connectors at the buttsplice connections on the bedspring, or the vapor containment penetrations, and 5) physical degradation of the grooves in the CRDM shaft. Based upon operating experience and the staff's knowledge of control rod drive mechanisms, the staff concludes that none of the above identified misalignment mechanisms will prevent the rod from dropping into the core when the reactor trip breakers are open. Therefore, the staff concludes that the licensee adequately addressed the intent of GDC 26, "Reactivity control system redundancy and capability."

The NRC staff also requested that the licensee describe if CRUD, or any of the identified misalignment causes, would prevent the rod from dropping into the core when the reactor trip breakers are open. The licensee stated that none of the possible causes, including CRUD, would prevent the CR from fully inserting into the core. Rod drop testing demonstrates that there is no binding or deformation of the fuel assembly that would prevent the CR from fully inserting into the core. The staff concludes that these possible causes do not affect the ability of the stationary coil to de-energize and the stationary grippers to release when the reactor trip breakers are open, causing the rod to insert. Therefore, the staff concludes that the licensee adequately addressed the intent of GDC 26, "Reactivity control system redundancy and capability."

Based on the considerations discussed above, the NRC staff determined that the licensee demonstrated that the identified misalignment causes, including CRUD, would not prevent the CR from dropping into the core when the reactor trip breakers are open.

In summary, the NRC staff reviewed the license amendment request to defer testing of CR G-3 from the requirements of SR 3.1.4.2 and concludes that the licensee provided adequate justification to defer CR G-3 from the requirements of SR 3.1.4.2. Specifically, the staff concludes that:

- The license confirmed that the SDM analysis was performed using NRC-approved methodologies for IP2;
- The licensee demonstrated that CR G-3 will trip with reasonable certainty and that there is adequate SDM assuming that both CR G-3 and the control rod of greatest reactivity worth fail to insert into the reactor core following a reactor scram.
- The licensee identified and described possible causes for the difficulties encountered while performing SR 3.1.4.2;
- The licensee identified possible misalignment mechanisms that could prevent the CR from dropping into the core when the reactor trip breakers are open;

Additionally, the NRC staff concludes that deferring testing of CR G-3 during the current operating cycle will reduce the possibility of an inadvertent rod drop and resulting plant transient. Therefore, the staff concludes that the licensee's proposal to defer testing of CR G-3 from the requirements of SR 3.1.4.2 until repaired in the next forced outage of sufficient duration prior to the refuel outage of 2016 or during the refuel outage in 2016, to be acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (NSHCD)

The Commission may issue the license amendment before the expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. This amendment is being issued prior to the expiration of the 60-day period. Therefore, a final finding of no significant hazards consideration follows.

The Commission has made a final determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment does 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 has provided its analysis of the issue of no significant hazards consideration which is presented below.

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

   Response: No.

   The proposed change revises the requirement to perform SR 3.1.4.2 testing on Control Rod G-3 until the next refuel outage or forced outage of sufficient duration. Performing a technical specification surveillance test is not an accident initiator and does not increase the probability of an accident occurring. Since the control rod remains operable, the proposed change does not affect or create any accident initiators or precursors. The proposed revision to the test frequency is based on the ability of the control rod to continue to be able to perform its design function. The safety analyses assume control rod full insertion be [by] de-energizing the CRDM coils and not the ability to move a full length control rod by its drive mechanism. The last rod drop test verified this ability so there is no increase in the consequences of an accident.

   Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change revises the requirement to perform SR 3.1.4.2 testing on Control Rod G-3 by changing the frequency of the test. The proposed change does not involve installation of new equipment or modification of existing equipment, so that no new equipment failure modes are introduced. Also, the proposed change in test frequency does not result in a change to the way that the equipment or facility is operated so that no new accident initiators are created.

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

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

   Response: No.

   The conduct of performance tests on safety-related plant equipment is a means of assuring that the equipment is capable of performing its intended safety function and therefore maintaining the margin of safety established in the safety analysis for the facility. The proposed change

> revises the requirement to perform SR 3.1.4.2 testing on Control Rod G-3 by changing the frequency of the test. The proposed change is based [on] the fact that there have been no problems with past tests of the Control Rod G-3 indicating the [that] there are no problems with binding that could prevent the rod from inserting and a 12 hour surveillance on rod position that would indicate any changes in position. There are no indications that the trip function would not work assuring the reduction in margin of safety is not significant.
>
> Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and based on this review, determined that the three standards of 10 CFR 50.92 are satisfied. Therefore, the NRC staff has determined that the amendment involves no 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 provided comments by letter dated March 20, 2015 (ADAMS Accession No. ML15085A135).

The State of New York recommended that the NRC not approve Entergy's application and requested that Entergy schedule a maintenance outage within the next 60 days for the purpose of repairing the G-3 control rod. The State characterized Entergy's application as incomplete, questioned whether the G-3 control rod would either inadvertently drop into the core during full power operations or fail to insert upon demand, questioned the extent of condition regarding other control rods, and asserted that Entergy's response to the no significant hazards consideration determination of 10 CFR 50.92(c) is incomplete.

As stated in 10 CFR 50.91(a)(2), the Commission provides a 30 day comment period on its proposed determination under the standard of 10 CFR 50.92(c) for determining that a proposed license amendment to an operating plant involves no significant hazards consideration provided that operation of the facility 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.

The State of New York asserts that Entergy failed to adequately conclude that the proposed amendment would not "involve a significant increase in the probability or consequences of an accident previously evaluated" because "Entergy only answers the question of whether the delay may further increase the likelihood that the Control Rod could not be tripped to fall into the reactor; Entergy fails to address the question of whether the postponement and continued build-up of CRUD or exacerbation of another cause could lead to the inadvertent drop of the Control Rod into the reactor."

The NRC staff's safety evaluation focused on the ability of the G-3 control rod to perform its safety function, i.e., its ability to fully insert itself into the core upon demand. The staff examined all possible causes of misalignment mechanisms, including CRUD, and concluded that none of the potential mechanisms would prevent the rod from dropping into the core when the reactor trip breakers open. Furthermore, the licensee demonstrated that sufficient SDM would be maintained following a reactor trip signal if both the G-3 control rod and the rod of greatest reactivity worth failed to insert into the core.

The control rods are electronically held in place by stationary gripper coils and the NRC staff concluded that the gripper coils remain functional because CRUD or any other misalignment causes would not prevent the rod from dropping into the core when the reactor trip breakers open. An inadvertent drop of a control rod into the reactor core during power operation is an analyzed plant transient as described in the IP2 Updated Final Safety Analysis Report (UFSAR) Section 14.1.4, "Rod Cluster Control Assembly Drop." The UFSAR concludes that following an inadvertent drop of a rod cluster control assembly (i.e., a control rod), the departure from nucleate boiling ratio criterion continues to be met, conditions would not lead to core damage, and all applicable safety criteria remain satisfied.

The NRC staff review of the proposed amendment determined that deferring testing of the G-3 control rod would reduce the probability of a dropped rod and ensuing plant transient. Therefore, the staff continues to conclude that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The State of New York asserts that Entergy did not adequately conclude that the proposed amendment would not "create the possibility of a new or different kind of accident from any accident previously evaluated" because Entergy did not address the question of (1) "whether a year-long postponement of addressing a 'misalignment' of one of its control rods creates a possibility of the rod inadvertently dropping into the reactor or of not dropping into the reactor," and (2) "whether such a potential accident caused by a misaligned control rod has been 'previously evaluated.'"

Deferring testing of the G-3 control rod for the remainder of the current operating cycle would not introduce a new or different accident from any accident previously evaluated. As previously discussed, the control rod is electronically held in place by the stationary gripper coils and the NRC staff concluded that the gripper coils remain functional. An inadvertent dropped control rod into the reactor core during power operation is an analyzed event described in IP2 UFSAR Section 14.1.4. Furthermore, the licensee demonstrated that sufficient shutdown margin will exist following a reactor scram concurrent with the failure of the G-3 control rod along with the rod of greatest reactivity worth to fully insert into the core.

The only failure modes of the G-3 control rod would be either the rod inadvertently drops into the core during full power operation or fails to drop into the core upon demand. As previously discussed in this safety evaluation, an inadvertent CR drop is analyzed in IP UFSAR Section 14.1.4 and sufficient SDM exists if CR G-3 and the CR of greatest reactivity worth fails to insert into the reactor core upon demand. Therefore, the NRC staff continues to conclude that deferring testing of the G-3 control rod would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The State of New York asserts that Entergy did not adequately conclude that the proposed amendment would not "involve a significant reduction in a margin of safety" because Entergy did not address "whether a current potential impact of the CRUD build-up could be an inadvertent drop of the Rod, nor does Entergy address the potential problems that may be caused to the Control Rod at issue or any other control rod or cluster of rods if the build-up is allowed to go unaddressed for another year."

As discussed in Section 3.4 of this safety evaluation, the licensee determined that the build-up of CRUD is responsible for the misalignment during the last two quarterly tests of the G-3 control rod. Deferring testing of the G-3 control rod reduces the possibility of an inadvertent rod drop during power operations. Any additional build-up of CRUD on the G-3 control rod would not have any impact because CRUD or any other misalignment causes would not prevent the rod from dropping into the core when the reactor trip breakers open. Testing of the remaining control rods as required by the IP2 TSs will continue to demonstrate their operability. Misalignment has not been observed in any of the other control rods during previous testing. Therefore, the NRC staff has no reason to believe that the build-up of CRUD is having detrimental effects on the operation of the remaining control rods. Thus, the staff continues to conclude that the proposed amendment does not involve a significant reduction in a margin of safety.

In summary, after careful consideration of the comments provided by the State of New York, the NRC staff did not identify any deficiency in Entergy's application based on the information provided by the State. Furthermore, the staff concludes that operation of the facility as proposed by the licensee would not involve a significant hazards consideration.

6.0 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 a surveillance requirement. 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 (80 FR 11236 and 80 FR 13915). 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) 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: Matthew Hardgrove

Date: April 2, 2015

April 2, 2015

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 RE: PROPOSED LICENSE AMENDMENT REGARDING A CHANGE TO TECHNICAL SPECIFICATION 3.1.4 "REACTIVITY CONTROL SYSTEMS" (TAC NO. MF5747)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 280 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 February 12, 2015, as supplemented by letters dated March 10 and April 1, 2015.

The amendment revises the TSs by changing the acceptance criteria for Surveillance Requirement (SR) 3.1.4.2 for Control Rod G-3. The change includes a footnote to the SR stating that Control Rod G-3 need not be moved until repaired in the next forced outage of sufficient duration prior to the refuel outage of 2016 or during the refuel outage in 2016.

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. 50-247

Enclosures:
1. Amendment No. 280 to DPR-26
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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ADAMS ACCESSION NO.: ML15083A490 ***by memo**

OFFICE	LPL1-1/PM	LPL1-1/PM	LPL1-1/LA	DSS/SRXB	OGC	LPL1-1/BC
NAME	DRender	DPickett	KGoldstein	CJackson*	MYoung	BBeasley (RPascarelli for)
DATE	03/25/2015	04/01/2015	03/25/2015	03/25/2015	04/01/2015	04/02/2015

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