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

September 1, 2009

Mr. Mark B. Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Mail Stop A-PY-A290
P.O. Box 97, 10 Center Road
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY AND A CLARIFICATION OF A FREQUENCY EXAMPLE (TAC NO. ME0886)

Dear Mr. Bezilla:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 153 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications (TSs) in response to your application dated March 11, 2009 (Agencywide Documents Access and Management System Accession No. ML090760866).

This amendment revises the TS surveillance requirement frequency in TS 3.1.3, "Control Rod OPERABILITY," and revises Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension.

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

Sincerely,

Stephen P. Sands

Stephen Sands, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures:
1. Amendment No. 153 to NPF-58
2. Safety Evaluation

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FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.153
License No. NPF-58

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

   A. The application for license filed by FirstEnergy Nuclear Operating Company, et al., (the licensee, FENOC) dated March 11, 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. NPF-58 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 153 are hereby incorporated into this license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of its issuance and shall be implemented within 90 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen J. Campbell

Stephen J. Campbell, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Facility Operating License

Date of Issuance: September 1, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 153

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Facility Operating License and 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	Insert
License NPF-58	License NPF-58
Page 4	Page 4
TSs	TSs
1.0-27	1.0-27
1.0-28	1.0-28
3.1-8	3.1-8
3.1-10	3.1-10
3.1-11	3.1-11
3.1-14	3.1-14

renewal. Such sale and leaseback transactions are subject to the representations and conditions set forth in the above mentioned application of January 23, 1987, as supplemented on March 3, 1987, as well as the letter of the Director of the Office of Nuclear Reactor Regulation dated March 16, 1987, consenting to such transactions. Specifically, a lessor and anyone else who may acquire an interest under these transactions are prohibited from exercising directly or indirectly any control over the licenses of PNPP Unit 1. For purposes of this condition the limitations of 10 CFR 50.81, as now in effect and as may be subsequently amended, are fully applicable to the lessor and any successor in interest to that lessor as long as the license for PNPP Unit 1 remains in effect; these financial transactions shall have no effect on the license for the Perry Nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of these transactions; (ii) the PNPP Operating Agreement; (iii) the existing property insurance coverage for PNPP Unit 1; and (iv) any action by a lessor or others that may have an adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now and hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at reactor core power levels not in excess of 3758 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 153 , are hereby incorporated into the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

a. FirstEnergy Nuclear Generation Corp. and Ohio Edison Company

1.4 Frequency

EXAMPLES

<u>EXAMPLE 1.4-2</u> (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

<u>EXAMPLE 1.4-3</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be performed until 12 hours after ≥ 25% RTP. -- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required <u>performance</u> of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-------------------NOTE------------------ Only required to be met in MODE 1. --------------------------------------- Verify leakage rates are within limits.	 24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour (plus the extension allowed by SR 3.0.2) interval, but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR, except as provided by SR 3.0.3 and LCO 3.0.4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.2 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than or equal to the low power setpoint (LPSP) of the Rod Pattern Control System (RPCS).
	<u>AND</u>		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- Inoperable control rods may be bypassed in RACS in accordance with SR 3.3.2.1.9, if required, to allow insertion of inoperable control rod and continued operation. --------------------- Fully insert inoperable control rod.	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	-------------------NOTE-------------------- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RPCS. --- Insert each withdrawn control rod at least one notch.	31 days
SR 3.1.3.3	Verify each control rod scram time from fully withdrawn to notch position 13 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.4	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position AND Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

Table 3.1.4-1
Control Rod Scram Times

-----------------------------------NOTES-----------------------------------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 13. These control rods are inoperable, in accordance with SR 3.1.3.3, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds) REACTOR STEAM DOME PRESSURE(c) 950 psig	REACTOR STEAM DOME PRESSURE(c) 1050 psig
43	0.30	0.31
29	0.78	0.84
13	1.40	1.53

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero.

(b) Scram times as a function of reactor steam dome pressure when < 950 psig are within established limits.

(c) For intermediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 153 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated March 11, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090760866), FirstEnergy Nuclear Operating Company, et al. (the licensee) requested changes to the technical specifications (TSs) for the Perry Nuclear Power Plant, Unit No. 1 (PNPP). The proposed amendment would revise the TS surveillance requirement frequency in TS 3.1.3, "Control Rod OPERABILITY," and revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension.

The changes are consistent with the NRC-approved Technical Specification Task Force (TSTF) change traveler TSTF-475, Revision 1, "Control Rod Notch Testing and [Source Range Monitor] SRM Insert Control Rod Action," with some deviations as discussed below. TSTF-475, Revision 1, was approved for use by the NRC on November 5, 2007 (ADAMS Accession No. ML073050017). This operating license improvement was made available to the industry by the NRC on November 13, 2007 (72 FR 63935) through the consolidated line item improvement process (CLIIP).

The licensee is proposing a plant-specific deviation from the TS changes described in the TSTF-475, Revision 1, and the NRC staff's model safety evaluation dated November 13, 2007. The deviation is discussed in the Technical Evaluation Section of this safety evaluation. The deviation does not affect the applicability of either the safety evaluation or the no significant hazards consideration determination published in the *Federal Register* as part of the CLIIP.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix A, General Design Criterion (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 [AOOs], 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 29, "Protection against anticipated occurrence," states, "the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences."

The design relies on the control rod drive (CRD) system to function in conjunction with the protection systems under AOOs, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD system provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during AOOs. A compliance with GDCs 26 and 29 for the CRD system, prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during AOOs. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

Section 50.36(c)(3) of 10 CFR states that Technical Specifications (TSs) shall contain Surveillance Requirements (SRs) "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." As discussed in Section 3.0 of this attachment, revising the SR frequency for notch testing of each fully withdrawn control rod from weekly to monthly and clarifying in a TS example that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to time periods discussed in SR Notes, still assures 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.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 <u>Revise the SR Frequency for notch testing of each fully withdrawn control rod from weekly to monthly (TS 3.1.3, "Control Rod OPERABILITY")</u>

The CRD system consists of CRDs, which are hydraulically-operated stepping mechanisms mounted in CRD housings, which extend below the reactor vessel bottom head.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD, which houses the collet mechanism. The collet mechanism performs the locking and unlocking functions that allow the insertion and withdrawal of the control rod. The latch, or locking collet, is a ratchet device that allows the control rod to be freely inserted but requires a specific unlock signal for rod withdrawal.

Control rod insertion capability is demonstrated by notch testing (i.e., inserting a control rod by at least one notch). Notch testing is currently performed weekly for fully withdrawn control rods and monthly for partially withdrawn control rods. During power operation, most control rods in the core are fully withdrawn, and subjected to notch testing at weekly intervals. Notch testing can also detect a CRT that is totally severed, e.g., from a 360-degree from Intergranular Stress Corrosion Cracking (IGSCC)-initiated crack, and can identify most postulated causes of mechanical binding.

Notch testing is designed to verify the ability to move rods. The ability to scram may be inferred from notch test results; but this is confirmed through scram time testing. Scram time testing can also detect problems in CRD performance resulting from IGSCC-initiated cracks and mechanical binding. Unlike the notch tests, these single rod scram tests cover additional mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod. The hydraulic control units, CRDs, and control rods are also tested during refueling outages, approximately every 18 to 24 months. Based on the data collected during the preceding cycle of operation, selected CRDs are inspected and their internal components are replaced, as required.

In 1975, cracking was observed in some CRTs (Reference 1). Circumferential cracking could lead to failure of a CRT that would prevent movement of its CRD. Notch testing, which requires movement of CRDs, is used to demonstrate CRT integrity. Since there have been no CRT failures since cracking was first observed in 1975 (Ref. 1), and since the CRT crack growth rate is slow (Ref. 1), the applicant maintains that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly.

IGSCC growth rates were evaluated using General Electric's PLEDGE model (Ref. 1), based on fundamental principles of stress corrosion cracking that can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. This report states that adding 24 days to the surveillance interval could result in an additional 10 mils of growth in total crack length. The small addition in crack length would not amount to a significant difference in the results of two notch tests, performed 31 days apart.

The NRC staff concludes that it would be acceptable to decrease the notch testing surveillance frequency of fully withdrawn control rods from weekly to monthly, based on the following reasons:

(1) The accumulation of operating experience, as reviewed by the NRC staff, indicates there have been no immovable control rods identified via performance of rod notch surveillance for either partially or fully withdrawn control rods.

(2) The predicted crack growth rate is slow. The proposed surveillance interval (31 days) remains short enough to be effective in detecting failed CRTs.

The NRC staff finds that increasing the surveillance interval from 7 days to 31 days would not compromise the CRD system's capability to reliably control reactivity changes under normal operations, including anticipated operational occurrences, such that specified acceptable fuel design limits are not exceeded.

The NRC staff notes that General Electric recommends a limited sampling of several CRDs removed for maintenance, for evidence of discernable corrosion that is different from corrosion that was observed in the past when weekly notching was performed, and an evaluation of CRT maintenance data to assess the actual extent of CRT cracking (Ref.1). The staff's conclusions are not based upon implementation of either of these recommendations. Operational experience, slow crack growth, and potential safety benefits of reduced control rod movements were sufficient. Therefore, implementation of either or both of these recommendations remains at the discretion of the user.

3.2 Clarify in TS Example 1.4-3 that the 1.25 surveillance test interval extension in SR 3.0.2 is also applicable to Time Periods discussed in SR Notes

The NRC staff has reviewed the proposal to amend Example 1.4-3 in Section 1.4 "Frequency," to clarify that the 1.25 provision in SR 3.0.2 is equally applicable to time periods specified in the Note in the "Surveillance" column. The NRC staff finds this change acceptable because the revision clarifies the example to make it consistent with the definition of "specified frequency" provided in the second paragraph of Section 1.4, which states that "the 'specified frequency' is referred to throughout this section and each of the specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The 'specified frequency' consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements."

3.3 Clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, Required Action E.2, "Source Range Monitoring Instrumentation"

The licensee did not apply for this change because the word "fully" is already contained in PNPP TS 3.3.1.2, Required Action E.2, "Source Range Monitoring Instrumentation." The NRC staff agrees that this change is not needed for PNPP.

3.4 Summary

The NRC staff has reviewed the licensee's proposal to amend the PNPP TSs, and has concluded that the TS revisions will have a minimal effect on the reliability of the CRD system while reducing the opportunity for potential reactivity events, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Therefore, the NRC staff concludes that the amendment request is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This 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 or 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 effluent 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (74 FR 20748; May 5, 2009). Accordingly, this 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 this amendment.

6.0 CONCLUSION

The NRC staff 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. CRD Notching Surveillance Testing for Limerick Generating Station (Report originally prepared in 2004), GE-NE-0000-0024-9858 R3, November 2006 (ADAMS Accession No. ML063250258).

2. Letter TSTF-07-19, Response from the Technical Specifications Task Force to the NRC, "Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," dated February 28, 2007, (TSTF-475, Revision 1 is an enclosure), ADAMS accession number ML071420428.

Principal Contributor:

Date: September 1, 2009

September 1, 2009

Mr. Mark B. Bezilla
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SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY AND A CLARIFICATION OF A FREQUENCY EXAMPLE (TAC NO. ME0886)

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Sincerely,

/RA/
Stephen Sands, Project Manager
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Docket No. 50-440

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Amendment Accession No. ML092220694

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