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

September 30, 2014

Vice-President, Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT RE: REVISED SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (TAC NO. MF4164)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 307 to Renewed Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 1, 2014, as supplemented by letter dated August 21, 2014.

The amendment revises TS 2.0, "Safety Limits (SLs)," by changing the safety limit minimum critical power ratio for both single and dual recirculation loop operation.

Enclosure 2 includes a copy of the Proprietary Safety Evaluation where the proprietary information is identified within double brackets **[[ ]]**. Enclosure 3 includes a copy of the Non-proprietary Safety Evaluation. 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. 50-333

Enclosures:
1. Amendment No. 307 to DPR-59
2. Proprietary Safety Evaluation
3. Non-proprietary Safety Evaluation

cc w/encls 1 and 3: Distribution via Listserv

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| <u>NOTICE</u>: Enclosure 2 to this letter contains Proprietary Information. Upon separation from Enclosure 2, this letter is DECONTROLLED. |

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

ENTERGY NUCLEAR FITZPATRICK, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 307
Renewed Facility Operating License No. DPR-59

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated May 1, 2014, as supplemented on August 21, 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 Renewed Facility Operating License No. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 307, are hereby incorporated into the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 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 Renewed Facility Operating License and Technical Specifications

Date of Issuance: September 30, 2014

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 307 RENEWED FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

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 |
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| 2.0-1 | 2.0-1 |

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use, at any time, any byproduct, source and special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration; or associated with radioactive apparatus, components or tools.

(5) 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.

C. This renewed operating 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) <u>Maximum Power Level</u>

ENO is authorized to operate the facility at steady state reactor core power levels not in excess of 2536 megawatts (thermal).

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 307, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Fire Protection</u>

ENO shall implement and maintain in effect all provisions of the approved fire protections program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated November 20, 1972; the SER Supplement No. 1 dated February 1, 1973; the SER Supplement No.2 dated October 4, 1974; the SER dated August 1, 1979; the SER Supplement dated October 3, 1980; the SER Supplement dated February 13, 1981; the NRC Letter dated February 24, 1981; Technical Specification Amendments 34 (dated January 31, 1978), 80 (dated May 22, 1984), 134 (dated July 19, 1989), 135 (dated September 5, 1989), 142 (dated October 23, 1989) 164 (dated August 10, 1990), 176 (dated January 16, 1992), 177 (dated February 10, 1992), 186 (dated February 19, 1993), 190 (dated June 29, 1993), 191 (dated July 7, 1993), 206 (dated February 28, 1994) and 214 (dated June 27, 1994); and NRC Exemptions and associated safety evaluations dated April 26, 1983, July 1, 1983, January 11, 1985, April 30, 1986, September 15, 1986 and September 10, 1992 subject to the following provision:

# 2.0 SAFETY LIMITS (SLs)

## 2.1 SLs

### 2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.13 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

### 2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

## 2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATED TO AMENDMENT NO. 307

## ENTERGY NUCLEAR FITZPATRICK, LLC

## AND ENTERGY NUCLEAR OPERATIONS, INC.

## DOCKET NO. 50-333

## JAMES A. FITZPATRICK NUCLEAR POWER PLANT

## TO RENEWED FACILITY OPERATING LICENSE NO. DPR-59

## 1.0 INTRODUCTION

By letter dated May 1, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14143A316) (Reference 1), as supplemented by letter dated August 21, 2014 (ADAMS Accession No. ML14233A381) (Reference 2), Entergy Nuclear Operations, Inc. (Entergy, or the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Technical Specifications (TSs). The amendment revises TS 2.0, "Safety Limits (SLs)," by changing the safety limit minimum critical power ratio (SLMCPR) for both single and dual recirculation loop operation.

The supplemental letter dated August 21, 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 Comission (NRC) staff's original proposed no significant hazards consideration (NSHC) determination as published in the *Federal Register* on August 5, 2014 (79 FR 45487).

## 2.0 REGULATORY EVALUATION

The following explains the use of general design criteria for JAFNPP. The construction permit for JAFNPP was issued by the Atomic Energy Commission (AEC) on May 20, 1970, and the operating license was issued on October 17, 1974. The plant design criteria for the construction phase are listed in the Updated Final Safety Analysis Report (UFSAR) Chapter 1.5, "Principal Design Criteria." The AEC published the final rule that added Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* on February 20, 1971 (36 FR 3255), 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 final GDC to plants with construction permits issued prior to May 21, 1971, which includes JAFNPP. However, the JAFNPP UFSAR, Chapter 16.6, "Conformance to AEC Design Criteria," evaluates JAFNPP against the 10 CFR Part 50, Appendix A, General Design Criteria (GDC). Also, the initial AEC safety evaluation of JAFNPP, dated November 20, 1972, Chapter 14.0, stated "Based on our evaluation of the design and design criteria for the James A. FitzPatrick Nuclear Power Plant, we conclude that there is reasonable assurance that the intent of the General Design Criteria for Nuclear Power Plants, published in the Federal Register on May 21, 1971 as Appendix A to 10 CFR part 50, will be met." Therefore, the NRC staff reviews amendments to the JAFNPP license using the 10 CFR 50 Appendix A GDC unless there are specific criteria identified in the UFSAR.

Title 10 of the *Code of Federal Regulations,* Part 50 (10 CFR 50), Appendix A, GDC 10 states, in part, that the reactor core and associated coolant, control, and protection systems shall be designed to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Fuel design limits can be exceeded if the core exceeds critical power. Critical power is a term used for the power at which the fuel departs from nucleate boiling and enters a transition to film boiling. For boiling water reactors (BWRs), the critical power is predicted using a correlation known as the GE (General Electric) critical quality boiling length correlation, or better known as the GEXL correlation. Due to core wide and operational variations, the margin to boiling transition is most easily described in terms of a critical power ratio (CPR), which is defined as the rod critical power as calculated by GEXL divided by the actual rod power. The more a CPR value exceeds 1.0, the greater the margin to boiling transition is. The SLMCPR is calculated using a statistical process that takes into account operating parameters and uncertainties. The operating limit MCPR (OLMCPR) is equal to the SLMCPR plus a CPR margin for transients. At the OLMCPR, at least 99.9 percent of the rods avoid boiling transition during steady state operation and transients caused by a single operator error or equipment malfunction.

Safety Limits are required to be included in the TS by 10 CFR 50.36. The SLMCPR is calculated on a cycle-specific basis because it is necessary to account for the core configuration-specific neutronic and thermal-hydraulic response.

## 3.0 TECHNICAL EVALUATION

### 3.1 JAFNPP Cycle 22 Core

JAFNPP is a BWR/4 which has two recirculation loops. The licensee proposed to change the SLMCPR value in TS 2.1.1.2 from 1.08 to 1.10 for two-recirculation-loop operation, and from 1.11 to 1.13 for single-recirculation-loop operation with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10 percent of rated core flow.

JAFNPP Cycle 22 core loading consists of 560 GNF2 (Global Nuclear Fuels – Americas, LLC, GNF) fuel bundles in the core. There will be 184 fresh fuel bundles, 196 once burned fuel bundles, and 180 twice burned fuel bundles.

### 3.2 Methodology

GNF developed the JAFNPP Cycle 22 SLMCPR values using the following NRC approved methodologies and uncertainties:

- NEDC-32601P "Methodology and Uncertainties for Safety Limit MCPR Evaluations"; Non-Power Distribution Uncertainty (Reference 4)
- NEDC-32694P "Power distribution Uncertainties for Safety Limit MCPR Evaluations"; Power Distribution Methodology and Uncertainty (Reference 6)
- NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Reference 3)
- NEDC-32505P-A "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel"; R-Factor Calculation Methodology (Reference 5)

Plant specific use of these methodologies must adhere to certain restrictions.

#### 3.2.1 Methodology Restrictions

Based on the review (Reference 6) of Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24011-P-A (GESTAR II), the NRC staff identified the following restrictions for the use of these Topical Reports:

1. The TGBLA (lattice physics code) fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table 3.1 of NEDC-32601P, since changes in fuel design can have a significant effect on calculation accuracy.

2. The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of R-Factor uncertainty when the methodology is applied to a new fuel lattice.

3. In view of the importance of MIP (MCPR Importance Parameter) criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601-P is applicable to future designs and operating strategies.

##### 3.2.1.1 Restrictions (1) and (2)

In addressing restrictions (1) and (2) in a letter dated September 24, 2001, from GNF to the NRC "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies" (Reference 9), GNF states that the rod power calculational uncertainties are dominated by geometrical considerations in which GE14 is identical to GE12.

The NRC staff determined that GNF2 (Reference 8) is designed for mechanical, nuclear, and thermal-hydraulic compatibility with the other 10X10 GNF fuel designs. The design has features of the currently operating GE 10, GE 11/13 and GE 12/14 fuel including pellet-cladding interaction resistant barrier cladding, high performance spacers, part length rods, interactive thick corner/thin wall channel, and axial enrichment loading. The GNF2 design is a l0xl0 array with 92 fuel rods and two large central water rods, fourteen part length fuel rods. The part length rod configuration improves efficiency and reactivity margins. Table 3.1 of NEDC-32601P provides GE12 10x10 lattices. The values given in Table 3.1 for GE12 are representative of the values being calculated for GE14 and GNF2.

The NRC staff asked the licensee (RAI-1) to explain the differences in design and geometrical considerations between GNF2 and GE 14 fuel. In response to the staff's question (Reference 2), the licensee stated that GNF2 is an evolutionary fuel product based on GE14 and that it is not considered a new fuel design as it maintains the previously established 10X10 array.

Based on the above discussion, the NRC staff concludes that the rod power calculational uncertainties used by GNF to develop the JAFNPP Cycle 22 SLMCPR values are valid for GNF2 fuel and that the response to RAI-1 (Reference 2) addresses the staff's concern and is acceptable.

3.2.1.2 Restriction (3)

For JAFNPP Cycle 22, the minimum core flow SLMCPR calculation performed at 79.8 percent core flow and rated core power condition was limiting as compared to the rated core flow and rated core power condition. The analysis indicates that, at low core flows, the limiting rod pattern and the nominal rod pattern are essentially the same. In its application, GNF determined that the rod pattern used to calculate the SLMCPR at 100 percent rated power and 79.8 percent rated flow reasonably assures that at least 99.9 percent of fuel rods in the core would not be expected to experience boiling transition during normal operation or anticipated operational occurrences during the operation of JAFNPP Cycle 22.

The NRC staff determined that the rod patterns used to calculate the SLMCPR at 79.8 percent of rated core flow and 100 percent of rated core power produce a limiting MCPR distribution that reasonably bounds the MCPR distributions that would be expected during the operation of the JAFNPP core throughout Cycle 22. Therefore, the staff concludes that the licensee's submittal demonstrates the validity of the criterion in restriction (3) (MIP criterion) for GNF2 fuel and the minimum core flow condition.

In summary, the NRC staff concludes that the licensee has adequately addressed the restrictions of Topical Reports NEDC-32601P-A, NEDC-32694P-A, Amendment 25 to NEDE-24011-P-A (GESTAR II), and NEDC-32505P-A and that the use of these reports to evaluate the JAFNPP Cycle 22 SLMCPR is acceptable.

3.3 Departures from NRC-Approved Methodology

No departures from NRC-approved methodologies were identified in the JAFNPP Cycle 22 SLMCPR calculations.

## 3.4 Deviations from the NRC-Approved Calculational Uncertainties

### 3.4.1 R-Factor

The R-factor is an input into the GEXL correlation used to describe the local pin-by-pin power distribution and the fuel assembly and channel geometry on the fuel assembly critical power. The R-factor uncertainty analysis includes an allowance for power peaking modeling uncertainty, manufacturing uncertainty and channel bow uncertainty. GNF has increased this uncertainty for all SLMCPR calculations to account for the potential impact of control blade shadow corrosion induced bow. GNF has generically increased the GEXL R-Factor uncertainty from **[[ ]]** to account for an increase in channel bow due to the emerging unforeseen phenomenon called control blade shadow corrosion-induced channel bow, which is not accounted for in the channel bow uncertainty component of the approved R-Factor uncertainty. The JAFNPP Cycle 22 analysis shows an expected channel bow uncertainty of **[[ ]]**, which is bounded by a GEXL R-Factor uncertainty of **[[ ]]** that accounts for a channel bow uncertainty of up to **[[ ]]** (Reference 5). Thus, the NRC staff finds that the use of a GEXL R-Factor uncertainty of **[[ ]]** adequately accounts for the expected control blade shadow corrosion-induced channel bow for JAFNPP Cycle 22.

### 3.4.2 Core Flow Rate and Random Effective TIP Reading

GNF has committed (Reference 7) to the expansion of the state points used in the determination of the SLMCPR. Consistent with the Reference 7 commitments, GNF performs analyses at the rated core power and minimum licensed core flow point in addition to analyses at the rated core power and rated core flow point. The NRC-approved SLMCPR methodology is applied at each statepoint that is analyzed.

The core flow and random traverse in-core probe (TIP) reading uncertainties used in single loop operation (SLO) minimum core flow SLMCPR analysis remain the same as in the rated core flow SLO SLMCPR analysis because these uncertainties (which are substantially larger than used in two loop operation (TLO) analysis) already account for the effects of operating at reduced core flow.

For TLO calculations performed at 79.8 percent core flow, the NRC-approved uncertainty values for the core flow rate (2.5 percent) and the random effective TIP reading (1.2 percent) are adjusted by dividing them by 79.8/100. The treatment of the core flow and random effective TIP reading uncertainties is based on the assumption that the signal to noise ratio deteriorates as core flow is reduced. GNF states that this increase is conservative based on the expectation that the variability in the absolute flow will decrease as flow decreases. The NRC staff finds that this increase in the uncertainty should bound the original non-flow dependent uncertainties and, therefore, the staff finds it acceptable for JAFNPP Cycle 22.

## 3.5 Core Monitoring System

For JAFNPP Cycle 22, the GNF 3D MONICORE System (Reference 8) will be used as the core monitoring system. The 3D MONICORE system is in widespread use throughout the GNF fueled fleet of BWRs, including BWR/4 plants like JAFNPP, and BWR/6 plants. Use of a current version of 3D MONICORE provides the plant capability to perform the reactivity anomaly surveillance. Use of 3D MONICORE has been previously evaluated and accepted by the NRC

(Reference 8). Therefore, the NRC staff finds the use of the GNF 3D MONICORE system for JAFNPP Cycle 22 to be acceptable.

3.6 Technical Evaluation Conclusion

The NRC staff finds the licensee's proposed Cycle 22 SLMCPR values of 1.10 for two-recirculation-loop operation and 1.13 for single-recirculation-loop operation acceptable for JAFNPP Cycle 22 because they were developed through the appropriate use of NRC-approved methodologies in accordance with NRC staff guidelines. The staff further finds that the licensee used methods consistent with the regulatory requirements and guidance identified in Section 2.0 above.

## 4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

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

The Commission has made a final determination that the amendment request involves NSHC. 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. The operation of JAF [James A. FitzPatrick Nuclear Power Plant] in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The basis of the Safety Limit Minimum Critical Power Ratio (SLMCPR) is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR values preserve the existing margin to transition boiling and probability of fuel damage is not increased. The derivation of the revised SLMCPR for JAF, for incorporation into the Technical Specifications and its use to determine plant and cycle-specific thermal limits, has been performed using NRC approved methods. These plant-specific calculations are performed each operating cycle and if necessary, will require future changes to these values based upon revised core designs. The revised SLMCPR values do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

   Based on the above, JAF has concluded that the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of JAF in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes result only from a specific analysis for the JAF core reload design. These changes do not involve any new or different methods for operating the facility. No new initiating events or transients result from these changes.

Based on the above, JAF has concluded that the proposed change will not create the possibility of a new or different kind of accident from those previously evaluated.

3. The operation of JAF in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

   The new SLMCPR is calculated using NRC approved methods with plant and cycle specific parameters for the current core design. The SLMCPR value remains conservative enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity. The operating MCPR limit is set appropriately above the safety limit value to ensure adequate margin when the cycle specific transients are evaluated. Accordingly, the margin of safety is maintained with the revised values.

   As a result, JAF has determined that the proposed change will not result in 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(c) are satisfied. Therefore, the 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 had no comments.

## 6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 (79 FR 45487, dated August 5, 2014). 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.

## 8.0 REFERENCES


1. Letter from Lawrence M. Coyle (Entergy Nuclear Northeast) to U.S. Nuclear Regulatory Commission, "Proposed Change to the James A. FitzPatrick Nuclear Power Plant's Technical Specification Concerning the Safety Limit Minimum Critical Power Ratio," May 1, 2014.

2. Letter from Lawrence M. Coyle (Entergy Nuclear Northeast)) to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding Proposed Safety Limit Minimum Critical Power Ratio License Amendment (SLMCPR)," August 21, 2014.

3. Global Nuclear Fuels Licensing Topical Report NEDE-24011-P-A-20, "General Electric Standard Application for Reactor Fuel," December 2013.

4. General Electric Nuclear Energy Licensing Topical Report NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," August 1999.

5. General Electric Nuclear Energy Licensing Topical Report NEDC-32505P-A, Revision 1, "R-Factor Calculation Method for GE11, GE12, and GE13 Fuel," July 1999.

6. MFN-003-99, Letter, Frank Akstulewicz (NRC) to Glen A. Watford (GE), "Acceptance for Referencing of Licensing Topical Reports NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations, NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluation; and Amendment 25 to NEDE-24011-P-A on Cycle Specific Safety Limit MCPR," March 11, 1999.

7. MFN 04-081, Letter from J.S. Post (GE) to NRC, "Part 21 Reportable Condition and 60-Day Interim Report; Notification: Non-Conservative SLMCPR," August 24, 2004.

8. NEDO-31152 Revision 9, "Global Nuclear Fuels Fuel Bundle Designs", dated May 2007.

9. Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies, dated September 24, 2001.


Principal Contributor: Fred Forsaty, NRR

Date: September 30, 2014

September 30, 2014

Vice-President, Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT RE: REVISED SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (TAC NO. MF4164)

Dear Sir or Madam:

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Enclosure 2 includes a copy of the Proprietary Safety Evaluation where the proprietary information is identified within double brackets **[[ ]]**. Enclosure 3 includes a copy of the Non-proprietary Safety Evaluation. 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. 50-333
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