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

October 3, 2008

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: RELOCATION OF PRESSURE AND TEMPERATURE CURVES TO THE PRESSURE AND TEMPERATURE LIMITS REPORT CONSISTENT WITH TSTF-419-A (TAC NO. MD8556)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 292 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 April 22, 2008, as supplemented by letters dated July 2, July 22, and September 24, 2008.

The amendment modified Technical Specification (TS) 1.0, "Definitions," Limiting Conditions for Operation and Surveillance Requirement Applicability, Section 3.4.9, "RCS [Reactor Coolant System] Pressure and Temperature (P-T) Limits," and Section 5.0, "Administrative Controls," to delete reference to the pressure and temperature curves, and include reference to the Pressure and Temperature Limits Report.

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,

B.K.Vaidya

Bhalchandra K. Vaidya, 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. 292 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

DATED: October 3, 2008

AMENDMENT NO. 292 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-59
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

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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. 292
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 April 22, 2008, as supplemented by letters date July 2, July 22, and September 24, 2008, 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 renewed facility operating 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 Appendix A, as revised through Amendment No. 292, are hereby incorporated in 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

Mark G. Kowal, 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: October 3, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 292

RENEWED FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

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

Remove Page	Insert Page
3	3

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

Remove Pages	Insert Pages
1.1-4	1.1-4
3.4.9-1	3.4.9-1
3.4.9-3	3.4.9-3
3.4.9-4	3.4.9-4
3.4.9-5	3.4.9-5
3.4.9-6	3.4.9-6
3.4.9-7	none
3.4.9-8	none
5.6-3	5.6-3

(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) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 292, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

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:

1.1 Definitions (continued)

LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power that exists in the core for each type of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------ NOTE -------- Required Action A.2 shall be completed if this Condition is entered. ---------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	------------ NOTE ------------ Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ----------------------------- Verify: a. RCS pressure and RCS temperature are within the limits specified in the curves in the PTLR as applicable; and b. RCS temperature change averaged over a one hour period is: 1. ≤ 100°F when the RCS pressure and RCS temperature are on or to the right of curve C in the PTLR as applicable, during inservice leak and hydrostatic testing; 2. ≤ 20°F when the RCS pressure and RCS temperature are to the left of curve C in the PTLR as applicable, during inservice leak and hydrostatic testing; and 3. ≤ 100°F during other heatup and cooldown operations.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	------------------------ NOTES ------------------------ 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be performed if SR 3.4.9.4 is satisfied. --- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	------------------------ NOTES ------------------------ 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be met if SR 3.4.9.3 is satisfied. --- Verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	------------ NOTES ------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. ----------------------------- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.6	------------ NOTES ------------ Only required to be performed when tensioning the reactor vessel head bolting studs. ----------------------------- Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	30 Minutes
SR 3.4.9.7	------------ NOTES ------------ Not required to be performed until 30 minutes after RCS temperature ≤ 80°F with any reactor vessel head bolting stud tensioned. ----------------------------- Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.8	------------ NOTES ------------ Not required to be performed until 12 hours after RCS temperature ≤ 100°F with any reactor vessel head bolting stud tensioned. ------------------------------ Verify, when the reactor vessel head bolting studs are under tension, reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	12 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
   i) Limiting Conditions for Operation Section 3.4.9 "RCS Pressure and Temperature (P/T) Limits"
   ii) Surveillance Requirements Section 3.4.9 "RCS Pressure and Temperature (P/T) Limits"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
   i) SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors"
   ii) SIA Calculation 0800846.301, "2" Instrument Nozzle Stress Analysis"

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 292

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-59

ENTERGY NUCLEAR OPERATIONS, INC.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated April 22, 2008, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081200881), as supplemented by letters dated July 2, July 22, and September 24, 2008 (ADAMS Accession Nos. ML081910140, and ML082100454 respectively[1]), Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Technical Specifications (TS). The supplements dated July 2, July 22, and September 24, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination. The supplement dated July 22, 2008, superseded the supplement dated July 2, 2008.

The proposed changes would revise the JAFNPP TS as necessary to relocate the pressure and temperature limit curves and associated references to a Pressure and Temperature Limits Report (PTLR). The request is submitted consistent with the guidance contained in Technical Specifications Task Force (TSTF) Report 419-A, "Revised PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (Reference 2). The request would modify TS 1.0, "Definitions," Limiting Conditions for Operation and Surveillance Requirement Applicability Section 3.4.9, "RCS [Reactor Coolant System] Pressure and Temperature (P-T) Limits," and Section 5.0, "Administrative Controls," to delete reference to the P-T curves, and includes reference to the PTLR.

TSTF-419-A provides for the relocation of P-T limit curves from the plant TS to a PTLR, which remains under administrative control, and is incorporated by reference into the plant TS. TSTF-419-A was approved by the NRC by letter dated March 21, 2002. To implement TSTF-419-A, a licensee is required to use, and reference in the TS, NRC-approved methodologies to develop the P-T limits. In this case, the licensee is using the methodologies described in: (1)

Note 1 ADAMS Accession No.for September 24, 2008 letter was not available at the time of the issuance this amendment.

Boiling Water Reactor Vessel Internals Project Topical Report BWRVIP-114, "RAMA Fluence Methodology Theory Manual" (Reference 3); and (2) Structural Integrity Associates Topical Report SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors" (Reference 4).

The NRC staff approved the RAMA fluence calculation code for pressure vessel licensing actions (Reference 5). Reference 5 discusses the NRC staff's review of RAMA and concludes that the code is acceptable based on its adherence to the guidance contained in Regulatory Guide 1.190, "Calculational and Dosimetry Techniques for Determining Pressure Vessel Neutron Fluence" (RG 1.190, Reference 6). The fluence values used for the determination of the JAFNPP P-T limit curves used the RAMA code.

The NRC staff approved the methodology described in SIR-05-044-A in the safety evaluation report dated February 6, 2007 (ADAMS Accesion No. ML070180483).

2.0 REGULATORY EVALUATION

In 10 CFR 50.36, the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

In general, there are two classes of changes to TSs: (1) changes needed to reflect modifications to the design basis (TSs are derived from the design basis), and (2) changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TSs over time. In determining the acceptability of such changes, the NRC staff interprets the requirements of 10 CFR 50.36, using as a model the accumulation of generically approved guidance in the improved Standard Technical Specifications (STS). For this review, the NRC staff used NUREG-1433, Revision 3, "Standard Technical Specifications, General Electric Plants BWR/4."

The NRC has established requirements in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates the acceptability of a facility's proposed PTLR based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Appendix H to 10 CFR Part 50; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); Generic Letter (GL) 92-01, Rev. 1; GL 92-01, Rev. 1, Supplement 1; Standard Review Plan (SRP) Section 5.3.2; and GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Reference 7). Appendix G to 10 CFR Part 50 requires that facility P-T limits for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers *Boiler and Pressure Vessel* Code (ASME Code). Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs. RG 1.99, Rev. 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation.

GL 92-01, Rev. 1 requested that licensees submit the RPV data for their plants to the NRC staff for review, and GL 92-01, Rev. 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. SRP Section 5.3.2 provides an acceptable method for determining the P-T limits for ferritic materials in the beltline of the RPV based on the ASME Code, Section XI, Appendix G methodology.

The most recent version of Appendix G to Section XI of the ASME Code which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G, is the 2001 Edition through the 2003 Addenda of the ASME Code. The P-T limit methodology based on this edition of Appendix G to Section XI of the ASME Code (the ASME Code, Section XI, Appendix G methodology) incorporates the provisions of ASME Code Cases N-588 and N-640. Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20% of the preservice hydrostatic test pressure.

The NRC staff's review of the requested change is based on the requirements in GL 96-03. Attachment 2 to GL 96-03 provides the complete regulatory basis for the requested TS change. The guidance contained in GL 96-03 provides three separate actions to be completed by the licensee to allow relocating P-T limit curves to a licensee controlled document. As stated in GL 96-03, the licensee must satisfy the following criteria:

(1) Have NRC-approved methodologies to reference in its TS,

(2) Develop a PTLR to contain the figures, values, and any explanation necessary, and

(3) Modify the applicable sections of the TS accordingly, as described in Section 1.0, "Introduction" of this SE.

Attachment 1 to GL 96-03 provides seven criteria for an approvable methodology; the first of these criteria concerns an acceptable fluence determination methodology and is the subject of this safety evaluation. According to GL 96-03, a criterion for the NRC-approved methodology is that the methodology shall describe how the neutron fluence is calculated.

TSTF-419 provides additional guidance which provides an alternative format for documenting the implementation of a PTLR in the "Administrative Controls" section of a facility's TS. Since this license amendment application requests the initial implementation of a PTLR for the JAFNPP unit, the NRC staff's review focused on both the implementation of the JAFNPP PTLR and the appropriate application of the SIR-05-044-A methodology to generate the proposed JAFNPP P-T limits. The related neutron fluence calculation was reviewed by the NRC's Division of Safety Systems.

As indicated in Section 4.2.2.2, NUREG-1905, "Safety Evaluation Report Related to the License Renewal of the James A. FitzPatrick Nuclear Power Plant," dated April 30, 2008 (ADAMS Accession No. ML081510826), the licensee made a commitment to submit revised P-T curves for NRC review and approval for "use [of P-T curves] past 32 EFPY [effective full power years], prior to reaching 32 EFPY." The NRC approval of relocation of the P-T limits from JAFNPP's TS to its PTLR will relieve such a commitment because, as long as the P-T limit methodology remains the same, the implementation of PTLR allows the licensee to revise P-T limits under the 10 CFR 50.59 process.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

3.1.1 PTLR Implementation

The licensee stated in the April 22, 2008, submittal that, "Attachment 1 of GL 96-03 contains seven technical criteria that the contents of a proposed methodology should conform to if license amendments requesting PTLRs are to be approved by the NRC staff." The NRC staff's evaluations of the contents of the BWROG [Boiling Water Reactor Owners Group] methodology against the seven criteria in Attachment 1 of GL 96-03 are provided in Section 3.1 of the SER for the SIR-05-044-A report. The licensee further stated that, "the Pressure and Temperature Limits Report (PTLR) based on the methodology and template provided in SIR-05-044-A is being supplied for review."

3.1.2 P-T limits

The adjusted reference temperature (ART) values and P-T limits valid for 32 EFPY of facility operation using the SIR-05-044-A methodology were documented in the proposed JAFNPP PTLR. The licensee identified the lower intermediate shell weld 2-233-A as the limiting material for the JAFNPP RPV. The key parameters in determining the licensee's ART value for the limiting material at the one-quarter of the RPV wall thickness (1/4T) location are shown in Table 4 of the PTLR for 32 EFPY. Corresponding parameters at the three-quarter of the RPV wall thickness (3/4T) are not provided in the PTLR because the PTLR indicated in Section 5, "Discussion," that the P-T limit curves based on the cooldown transient (where the relevant critical location is at the 1/4T depth) are more conservative than the P-T limit curves based on the heatup transient (where the relevant critical location is at the 3/4T depth).

Additional information regarding the JAFNPP P-T limits was provided for review in the licensee's July 22, 2008, response to the NRC staff's request for additional information (RAI). This submittal documented the detailed thermal analyses and fracture mechanics evaluations for the RPV beltline, the bottom head, and the upper vessel, and nozzles supporting the proposed "composite" JAFNPP P-T limits valid for 32 EFPY. Since the fluence exposure of the instrument nozzles is greater than 1×10^{17} n/cm^2 (E > 1.0 MeV) for 32 EFPY, ART, ΔRTNDT, and P-T curve calculations in comparison to the beltline P-T curves, were provided for review. For the RPV beltline and bottom head, the ASME Code, Section XI, Appendix G methodology was used to calculate the pressure and thermal stress intensity factors (K_{Im} and K_{It}), except that a stress concentration factor of 3 was applied to the K_{Im} value in the bottom head calculation. For the upper vessel, the nozzle corner pressure and thermal hoop stresses were based on plant-specific finite element model (FEM) results for the JAFNPP feedwater nozzle under the limiting turbine roll event. The formulas in the SIR-05-044-A report were then used to calculate its K_{Im} and K_{It} values. In the final step, the calculation package utilized the applied K_{It} values and the plane-strain fracture toughness (K_{Ic}) values at the crack tip to calculate the allowable pressure stress intensity factor (K_{Ip}) at the tip of the postulated flaw at the 1/4T location. Pressure was then obtained by comparing the K_{Im} value based on FEM to the K_{Ip} value.

3.1.3 Fluence Methodology

In March 2001, the NRC staff issued RG 1.190 (Reference 6) which provides methods for determining RPV fluence. Fluence calculations are acceptable if they are performed with approved methodologies or with methods which are shown to conform with the guidance in RG 1.190. Attachment 1 to Reference 1 noted that the NRC SER related to the license renewal of JAFNPP contained an open item related to reactor vessel neutron fluence, in which the method supporting the license renewal application did not adhere to the RG 1.190 guidance. The licensee submitted new fluence calculations for 32 EFPY and 54 EFPY by letter dated November 5, 2007 (Reference 8). These calculations were performed using the RAMA methodology (Reference 3), which was approved by the NRC staff (Reference 5).

3.2 Staff Evaluation

3.2.1 PTLR Implementation

As mentioned in Section 3.1.1 of this SE, Attachment 1 of GL 96-03 requires the licensee evaluate and document seven technical criteria to demonstrate the acceptability of its PTLR methodology. The NRC staff examined the proposed PTLR and determined that it was developed from the template PTLR of the SIR-05-044-A report and meets the seven technical criteria:

(1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluences (Section 3.0, "Methodology," page 3 of the JAFNPP PTLR).

(2) The PTLR methodology describes the surveillance program (Appendix A, "JAF Reactor Vessel Material Surveillance Programs," page 22 of the JAFNPP PTLR).

(3) The PTLR methodology describes how the low temperature overpressure protection system limits are calculated applying system/thermal hydraulics and fracture mechanics (not applicable to BWRs).

(4) The PTLR methodology describes the method for calculating the ART values using RG 1.99, Rev. 2 (Section 3.0, page 3 of the JAFNPP PTLR).

(5) The PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on ASME Code, Section XI, Appendix G and the guidance in the NRC's SRP. The JAFNPP PTLR provided information regarding the finite element analyses performed to generate part of the P-T limits. The July 22, 2008 submittal stated that all of the equations and values were calculated in accordance with the SIR-05-044-A report. This description is sufficient because the SIR-05-044-A report contained detailed information regarding the application of fracture mechanics in the construction of P-T limits based on the ASME Code, Section XI, Appendix G and the

NRC SRP.

Since the instrument nozzle geometry does not correspond to the nozzle geometry contained in SIR-05-044-A, the licensee provided the instrument nozzle fracture mechanics analysis in the July 22, 2008 submittal. The licensee's methodology for the analysis of the JAFNPP RPV instrument nozzles within its overall PTLR methodology is reflected in revisions to the JAFNPP TS 5.6.7b.

(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T limits for boltup temperature and hydrotest temperature. (Page 3 of the JAFNPP PTLR stated that the P-T limits were calculated in accordance with the SIR-05-044-A report. This description is sufficient because the SIR-05-044-A report contained detailed information regarding the minimum temperature requirements for boltup temperature and hydrotest temperature.)

(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation. The NRC staff accepts the JAFNPP PTLR because this generic information is available in Appendix A of the SIR-05-044-A report.

In addition, the NRC staff reviewed corresponding changes in the "notes" associated with JAFNPP Surveillance Requirements (SR) 3.4.9.7 and 3.4.9.8 in the TS. These corresponding changes, which establish the temperature at which JAFNPP operators are expected to check reactor coolant system temperatures to ensure compliance with the JAFNPP P-T limits, are acceptable as the new values implemented in accordance with this TS amendment are consistent with the 32 EFPY P-T limit curves provided in the JAFNPP PTLR. That is, the verification step document in the "notes" associated with SR 3.4.9.7 and SR 3.4.9.8 will ensure that RPV flange temperatures remain consistent with the stated requirements.

Hence, the implementation of the JAFNPP PTLR is acceptable.

3.2.2 P-T limits

To evaluate the proposed P-T limits for the JAFNPP RPV, the NRC staff confirmed the licensee's selection of the lower intermediate shell weld 2-233-A as the limiting beltline material and performed an independent calculation of the ART values for this material using the RG 1.99, Rev. 2 methodology. The NRC staff's ART values for the limiting beltline material at the 1/4T and 3/4T locations are 112.2 °F and 81.7 °F for 32 EFPY, which were calculated using materials information for JAFNPP in the NRC Reactor Vessel Integrity Database (RVID) and the RPV inner diameter (ID) fluence in the JAFNPP PTLR. The licensee's ART value of 109.6 °F at the 1/4T location for 32 EFPY for the limiting beltline material is close to the NRC staff's value based on identical initial reference temperature (RT_{NDT}) and copper (Cu) and nickel (Ni) values for the limiting material. The licensee did not calculate the ART value at the 3/4T location, which is relevant to the heatup P-T limit calculation, because the SIR-05-044-A report concluded that P-T limits for the cooldown transient are bounding. The NRC staff will discuss this SIR-05-044-A report conclusion later.

As explained in Section 3.1.2 of this SE, the proposed JAFNPP P-T limits are composite curves, representing the most limiting P-T limits for the RPV beltline, the bottom head, and the upper vessel. The NRC staff has verified that, for a cooldown of 100 °F per hour, the bottom head P-T limits are the least limiting. Consequently, only the RPV beltline P-T limits (upper segment) and the upper vessel P-T limits (lower segment) are represented in the proposed normal operation core not critical heatup/cooldown curve (Curve B). This is also true for the proposed JAFNPP P-T limits for pressure test (Curve A), which are obtained by setting the thermal contribution to zero. The NRC staff performed independent calculations for both segments of the proposed P-T limits valid for 32 EFPY.

For the RPV beltline P-T limit segment, the NRC staff utilized the ASME Code, Section XI, Appendix G methodology in its evaluation, using the K_{IC} curve as resistance and the pressure-dependent K_{Im} formula and the cooldown rate dependent K_{It} formula as driving forces. The NRC staff used plant-specific information submitted by the licensee, which included: the temperature measurement instrument uncertainty; the pressure measurement instrument uncertainty, and the pressure adjustment for the hydrostatic pressure head. The NRC staff produced almost identical beltline P-T limits for the two ends of the upper segment. The NRC staff's calculation indicated, however, that the licensee's "Minimum Reactor Vessel Metal Temperature (degrees F)" in the proposed P-T limits is actually the metal temperature at 1/4T.

This represents a deviation from the SIR-05-044-A approach which recommended use of the coolant temperature, instead of the metal temperature at 1/4T, to evaluate K_{IC}. The SIR-05-044-A report further stated that "[t]he use of the coolant temperature is considered to be a necessary conservatism in P-T curve development to ensure that all design margins and safety factors are maintained." However, since this "necessary conservatism" is not required by the ASME Code, Section XI, Appendix G or by 10 CFR Part 50, Appendix G, the NRC staff considers the licensee's deviation acceptable.

The NRC staff's experience with prior P-T limit reviews suggested that heatup P-T limits could be limiting at certain range of the pressure and temperature. As a result, the NRC staff also performed calculations using the ART value of 81.7 °F at the 3/4T location for the limiting beltline material. This exercise confirmed that for the JAFNPP RPV, the cooldown P-T limits are more limiting than the heatup P-T limits, giving additional credibility to the proposed JAFNPP P-T limits.

For the upper vessel P-T limit segment, the NRC staff utilized the K_{It} and K_{Im} formulas in the SIR-05-044-A report (also available in the licensee's calculation package) to calculate driving forces and the ASME Code, Section XI, Appendix G K_{IC} curve to calculate resistance. The input nozzle corner pressure and thermal hoop stresses were based on plant-specific FEM results for the JAFNPP feedwater nozzle under the limiting turbine roll event. The licensee explained in its July 22, 2008, response that this transient results in an injection of cold feedwater (100 °F) into the hot RPV (550 °F) and represents the most severe event for the feedwater nozzle. The NRC staff, therefore, agrees with the licensee that this event is equivalent to the limiting normal/upset design transient for a BWR feedwater nozzle. The NRC staff's calculation produced almost identical P-T values for a randomly selected points along the lower segment of the proposed P-T limits.

Additional requirements are contained in 10 CFR Part 50, Appendix G for the minimum metal temperature of the closure head flange and vessel flange regions. These considerations were reflected in the "notch" of the upper vessel P-T limits. The NRC staff confirmed that when the pressure is greater than 20% of the hydro test pressure (0.20 x 1563 = 312 psig) the temperature for the pressure test P-T limits is greater than the RT_{NDT} of the limiting flange material plus 90 °F and the temperature for the core not critical P-T limits is greater than the RT_{NDT} of the limiting flange material plus 120 °F. The JAFNPP pressure test P-T limits also show a 60 °F straight line on the low pressure end. This was made to meet the 10 CFR Part 50, Appendix G minimum temperature requirement for pressure test which limits the operating temperature to the highest RT_{NDT} of the closure flange that is highly stressed by the bolt preload. Since this value for the JAFNPP RPV is 30 °F, the licensee's approach is conservative.

Recently, concern has arisen over whether the P-T limits for BWRs appropriately considered small bore instrument nozzles located within the beltline region. The licensee's July 22, 2008, response to the NRC staff's RAI included calculations to assess the 2" instrument nozzles (N16A and N16B) located in the JAFNPP reactor vessel, from a brittle fracture perspective. The nozzles are located in lower-intermediate shell plates (G-3414-2 and G-3413-7) with fluence exposure greater than 1 x 10^{17} n/cm^2 (E > 1.0 MeV) at 32 EFPY. The nozzles consist of a stainless steel pipe welded into the RPV wall with austenitic material. The NRC staff verified the ART and reference temperature shift (ΔRT_{NDT}) calculations using the methodology of RG 1.99, Rev. 2. P-T curves were developed consistent with the ASME Code, Section XI, Appendix G (2001 Edition including the 2003 Addenda), and 10 CFR Part 50, Appendix G methodology. Although a nozzle solution was provided in the SIR-05-044-A report, it applied to a different nozzle geometry than that of the instrument nozzles. The July 22, 2008, submittal detailed calculations for the N16 instrument nozzle geometry. The NRC staff verified that P-T curves for the N16 instrument nozzles are bounded by the beltline P-T curves. Hence, the NRC staff agrees with the licensee that the P-T curves remain valid and bound the N16 nozzles through 32 EFPY of operation.

Based on the above evaluation, the NRC staff determined that the licensee's proposed P-T limits are in accordance with the SIR-05-044-A report and satisfy the requirements of Appendix G to Section XI of the ASME Code and Appendix G to 10 CFR Part 50. Hence, the licensee's proposed P-T limit curves are acceptable for operation of the JAFNPP RPV valid for 32 EFPY. As a separate issue, the NRC staff determined that the licensee is relieved from its commitment to submit revised P-T limits for NRC review and approval for a 60-year operating license prior to reaching 32EFPY, as part of the NRC approval of the JAFNPP license renewal application as long as the P-T limit methodology remains the same. Other changes to the PTLR are subject to the regulations in 10 CFR 50.59.

3.2.3 Fluence Methodology

The NRC staff determined in the SER for license renewal (Reference 9) that the new fluence calculations submitted by the licensee in Reference 8 are acceptable and confirmed that the use of this fluence methodology is an acceptable basis for support of the PTLR development.

Further, the NRC staff noted the following:

(1) The licensee has identified the fluence methodology in both the technical analysis supporting its license amendment request and in the PTLR itself,

(2) The fluence calculations have been previously reviewed by the NRC staff and confirmed to adhere to the guidance of RG 1.190, and

(3) The fluence calculations extend to 54 EFPY (equivalent to 60 years of operation), and contain acceptable fluence projections for 32 EFPY to support the PTLR.

Therefore, the NRC staff concludes that the licensee has satisfied Criterion 1 of GL 96-03, as discussed in Section 2.0 above, and on this basis, implementation of TSTF-419-A is acceptable with respect to the use of an NRC approved methodology for fluence calculations.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (July 1, 2008, (73 FR 37503)). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Dietrich, P., Entergy Nuclear Northeast, letter to US Nuclear Regulatory Commission, "Application for Amendment to Technical Specifications Regarding Relocation of Pressure and Temperature Curves to the Pressure and Temperature Limits Report Consistent with TSTF-419-A," Docket 50-333, Accession ML081200881, April 22, 2008.

2. Pietrangelo, A. R., Nuclear Energy Institute, letter to W. D. Beckner, US Nuclear Regulatory Commission, "Forwarding of TSTFs," Enclosure 2, "Revised PTLR Definition and References in the ISTS 5.6.6., RCS PTLR," Accession ML012690199, September 19, 2001.

3. Carter, R., Electric Power Research Institute, "BWRVIP-114: BWR Vessel and Internals Project RAMA Fluence Methodology Theory Manual," Accession ML031640195, Palo Alto, California, May 2003.

4. Stevens, G. L., Structural Integrity Associates, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," Report Number SIR-05-044-A, Accession ML072340283, Centennial, Colorado, April 30, 2007.

5. Hsii, G., US Nuclear Regulatory Commission, letter to M. Mitchell, US Nuclear Regulatory Commission, "RAMA: A Radiation Transport Code to Calculate BWR Vessel and Reactor Internals Neutron Fluence," Accession ML050960576, April 7, 2005.

6. U.S. Nuclear Regulatory Commission, "Calculational and Dosimetry Techniques for Determining Pressure Vessel Neutron Fluence," Regulatory Guide 1.190, Accession ML010890301, March 31, 2001.

7. U.S. Nuclear Regulatory Commission, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," Generic Letter 96-03, Accession ML031110004, January 31, 1996.

8. Dietrich, P., Entergy Nuclear Northeast, letter to US Nuclear Regulatory Commission, "License Renewal Application Amendment 14," Docket 50-333, Accession ML073180494, November 5, 2007.

9. Le, N. B., US Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant," ML080250372, Washington, D.C., January 24, 2008.

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Date: October 3, 2008

October 3, 2008

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: RELOCATION OF PRESSURE AND TEMPERATURE CURVES TO THE PRESSURE AND TEMPERATURE LIMITS REPORT CONSISTENT WITH TSTF-419-A (TAC NO. MD8556)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 292 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 April 22, 2008, as supplemented by letters dated July 2, July 22, and September 24, 2008.

The amendment modified Technical Specification (TS) 1.0, "Definitions," Limiting Conditions for Operation and Surveillance Requirement Applicability, Section 3.4.9, "RCS [Reactor Coolant System] Pressure and Temperature (P-T) Limits," and Section 5.0, "Administrative Controls," to delete reference to the pressure and temperature curves, and include reference to the Pressure and Temperature Limits Report.

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/

Bhalchandra K. Vaidya, 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. 292 to DPR-59
2. Safety Evaluation

cc w/encls: See next page
Distribution: See next page

ADAMS Package No.: ML082630385 Amendment No.: ML082630365
Tech Spec No.: ML082630390 *See memo dated 7/16/08 and 9/26/08

OFFICE	LPL1-1/PM	LPL1-1/LA	ITSB/BC	SRXB/BC*	CVIB/BC*	OGC	LPL1-1/BC
NAME	BVaidya	SLittle	RElliot	GCranston	MMitchell	BMizuno	MKowal
DATE	9/24/08	9/29/08	10/02/08	7/16/08	9/26/08	10/03/08	10/02/08

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