Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. Fitzpatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315 349 6024 Fax 315 349 6480

Pete Dietrich
Site Vice President - JAF

July 2, 2008
JAFP-08-0057

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

**Subject:** **Entergy Nuclear Operations, Inc.**
**James A. FitzPatrick Nuclear Power Plant**
**Docket No. 50-333**
**License No. DPR-59**

**James A. FitzPatrick Nuclear Power Plant – Response to Request For Additional Information Regarding Relocation of Pressure and Temperature Curves to the Pressure and Temperature Limits Report (TAC No. MD8556)**

References:
1) NRC Letter to P. Dietrich, Request For Additional Information Regarding Relocation of Pressure and Temperature Curves to the Pressure and Temperature Limits Report, dated June 24, 2008
2) Entergy Letter, JAFP-08-0034, Application for Amendment to Technical Specifications Regarding Relocation of Pressure and Temperature (P-T) Curves to the Pressure and Temperature Limits Report (PTLR) Consistent with TSTF-419-A, dated April 22, 2008, (TAC No. MD8556)
3) Entergy Letter, JAFP-08-0053, James A. FitzPatrick Nuclear Power Plant – Response to Request For Additional Information Regarding Relocation of Pressure and Temperature Curves to the Pressure and Temperature Limits Report (TAC No. MD8556), dated June 27, 2008

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (ENO), as operator of the James A. FitzPatrick Nuclear Power Plant (JAF), hereby submits this response to the NRC's Request for Additional Information (RAI)(Reference 1) regarding the Relocation of Pressure and Temperature Curves to the Pressure and Temperature Limits Report (Reference 2). This response supersedes Reference 3 in its entirety.

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Attachment 1 contains the responses to questions 1, 3, 4, 5, and 6

The commitments made in this letter are summarized in Attachment 4.

This letter does not affect the "No Significant Hazards" determination made in Reference 2.

Should you have any questions concerning this letter, please contact Mr. Jim Costedio, Licensing Manager, at (315) 349-6538.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 2nd day of July, 2008.

Respectfully,

Pete Dietrich
Site Vice President

Attachments:

1. Response to RAI Questions
2. Structural Integrity Associates Calculation No. FITZ-10Q-301, Revision 0, "Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts," 2/15/2008
3. Structural Integrity Associates Calculation No. FITZ-10Q-302, Revision 0, "Revised Pressure-Temperature Curves," 2/26/2008
4. List of Commitments

cc:

Mr. Samuel J. Collins
Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Office of Resident Inspector
James A. FitzPatrick Nuclear Power Plant
U. S. Nuclear Regulatory Commission
P. O. Box 136
Lycoming, New York 13093

Mr. Paul Eddy
New York State Department of
Public Service
3 Empire State Plaza
Albany, New York 12223-1350

Mr. Bhalchandra Vaidya, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-8-G14
Washington, DC 20555-0001

Mr. Paul Tonko, President
New York State Energy Research
and Development Authority
17 Columbia Circle
Albany, New York 12203-6399

# ATTACHMENT 1 to JAFP-08-0057

## Entergy Nuclear Operations, Inc.
## James A. FitzPatrick Nuclear Power Plant

## Response to RAI Questions

## ATTACHMENT 1 to JAFP-08-0057

RAI Question

1) Provide an evaluation showing that you have performed analysis for the bottom head region and the non-beltline region in accordance with SIR-05-044-A and identify the portion of the proposed P-T Limits (Figure 3 of the proposed PTLR is sufficient) that are limited by these regions.

Response:

Refer to Section 2.0 of Attachment 3 for the analysis methodology used for both the bottom head region and the non-beltline (feedwater nozzle/upper vessel) region. Both regions were evaluated consistent with SIR-05-044-A methodology.

RAI Question

2) Provide an evaluation for the small diameter, drill hole type instrument nozzle (e.g., water level nozzle) if it exists in your reactor pressure vessel (RPV) beltline.

Response:

Response to this question will be provide by separate letter to be submitted not later than July 23, 2008

RAI Question

3) Identify among the three methodologies (Page 2-13 of SIR-05-044-A) the one that you used to calculate thermal stress intensity factors for the shell regions.

Response:

As indicated in Section 2.0 (page 5) of Attachment 3, the "Section XI Non-mandatory Appendix G Method" specified in SIR-05-044-A was used to calculate thermal stress intensity factors for the shell regions.

RAI Question

4) Provide the temperature instrument uncertainty, the pressure instrument uncertainty, and the pressure head to account for the column of water in the RPV (page 2-25 of SIR-05-044-A) so that the NRC staff can assess the difference between the staff's calculated P-T Limits and your proposed P-T limits.

Response:

Refer to Section 2.0 (page 8) and Section 3.0 (page 10) of Attachment 3 for the pressure instrument uncertainty (0 psig), temperature instrument uncertainty (0°F), and pressure head to account for the column of water in the RPV (29.8 psig).

RAI Question

5) Provide Reference 6.3, "Evaluation of Adjusted Reference Temperature and Reference Temperature Shifts"," dated February 2008; Reference 6.4, "Revised P-T curves", dated February 2008, and Reference 6.14, "BWRVIP-135, Revision 1: BWR Internals Project, Integrated Surveillance Program Data Source Book and Plant Evaluation", dated June 2007, to supplement the above requested specific information and to provide information regarding data and methodology for the adjustment of chemistry factors.

Response:

References 6.3 and 6.4 are provided as Attachments 2 and 3 of this letter. Since reference 6.14 is a proprietary EPRI Document it can not be submitted at this time. Entergy will work with EPRI to provide the relevant portions of the document to the NRC by July 23, 2008.

RAI Question

6) The guidelines in SIR-05-044-A provided for analysis of feedwater nozzles. Identify the specific analysis performed for the JAF feedwater nozzles, or explain why the analysis was not necessary.

Response:

Refer to Section 2.0 (pages 5 through 8) and Section 3.0 (page 11) of Attachment 3 for discussion of the analyses performed for the feedwater nozzles. The stress intensity factor calculations for the feedwater nozzles are based on the stress results for the limiting normal/upset transient for the feedwater nozzle using a plant-specific finite element model of the JAF feedwater nozzle described in GE Report NEDC-30799-P, "James A. FitzPatrick Nuclear Power Station Feedwater Nozzle Fracture Mechanics Analysis to show Compliance with NUREG-0619". NEDC-30799-P was previously developed to provide a fracture mechanics assessment of the feedwater nozzles, and the limiting nozzle corner hoop stresses were extracted as a part of that assessment. These nozzle corner hoop stresses are directly relevant to P-T curve development, and were therefore used to determine polynomial fits of the pressure and thermal hoop stresses for use with the SIR-05-044-A methodology, as described in Sections 2.0 and 3.0 of Attachment 3. The thermal transient evaluated in NEDC-30799-P is equivalent to the limiting normal/upset design transient for a BWR feedwater nozzle, which is a Turbine Roll event that represents initiation of feedwater flow into the RPV. This event is assumed to occur immediately after RPV heatup to rated temperature and pressure, where cold (unheated = 100°F) feedwater is injected into the hot (550°F) RPV. Because the transient is an injection event, the transient is assumed to be an instantaneous (shock) event for the nozzle. All other normal/upset events occur either at slower rates or from less bounding temperatures (because of the presence of heated feedwater or lower RPV temperatures), thereby making the Turbine Roll event the most severe event for the feedwater nozzle.

An affidavit regarding the proprietary nature of NEDC-30977-P and a non-proprietary version will be obtained from General Electric and submitted by July 23, 2008.