



Entergy Nuclear Northeast  
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Proprietary Information  
Withhold from Public Disclosure  
Pursuant to 10 CFR 2.390(a)(4)

Pete Dietrich  
Site Vice President - JAF

July 22, 2008  
JAFP-08-0067

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 (TAC No. MD8556), 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
  - 4) Entergy Letter, JAFP-08-0057, 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 July 2, 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

Attachments 2, 6, and 8 contain proprietary information as described in 10 CFR 2.390. When separated from these attachments this letter and its contents are non-proprietary.

ADD  
NRR

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Pressure and Temperature Limits Report (Reference 2). This response supersedes References 3 and 4 in their entirety.

Attachment 1 contains the responses to questions 1 through 6. Attachments 2 through 10 contain the reference documents requested by the NRC Staff to support review of Reference 2. Attachments 2, 6, and 8 contain proprietary information; affidavits attesting to the proprietary nature of the documents are included in the attachments. Non-proprietary versions of those documents are provided in Attachments 3, 7, and 9.

There are no new commitments made in this letter.

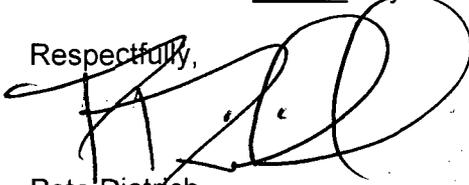
This letter does not affect the "No Significant Hazards" determination made in Reference 2.

Should you have any questions concerning this letter, please contact Mr. Jim Costedio, Licensing Manager, at (315) 349-6358.

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

Executed on the 22<sup>nd</sup> day of July, 2008.

Respectfully,



Pete Dietrich  
Site Vice President

List of Attachments (next page)

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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 (Proprietary Version with Affidavit)
3. Structural Integrity Associates Calculation No. FITZ-10Q-301, Revision 0, "Evaluation of Adjusted Reference Temperatures and Reference Temperature Shifts," 2/15/2008 (Non-proprietary Version)
4. Structural Integrity Associates Calculation No. FITZ-10Q-302, Revision 0, "Revised Pressure-Temperature Curves," 2/26/2008
5. Structural Integrity Associates Calculation No. 0800846.302, Revision 0, "Comparison of Instrument Nozzle (N16) and Bellline P-T Curves," 7/22/2008
6. BWR Vessels and Internals Project, "Integrated Surveillance Program (ISP) Data for FitzPatrick", (Excerpted from BWRVIP-135, Revision 1) July, 2008 (Proprietary Version with Affidavit)
7. BWR Vessels and Internals Project, "Integrated Surveillance Program (ISP) Data for FitzPatrick", (Excerpted from BWRVIP-135, Revision 1) July, 2008 (Non-proprietary Version)
8. NEDC-30799-P, "James A. FitzPatrick Nuclear Power Station Feedwater Nozzle Fracture Mechanics Analysis to Show Compliance with NUREG-0619", December, 1984 (Proprietary Version with Affidavit)
9. NEDO-30799, "James A. FitzPatrick Nuclear Power Station Feedwater Nozzle Fracture Mechanics Analysis to Show Compliance with NUREG-0619", December, 1984 (Non-proprietary Version)
10. Structural Integrity Associates Calculation No. 0800846.301, Revision 0, "2" Instrument Nozzle Stress Analysis," 7/16/2008

cc:

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## **ATTACHMENT 1 to JAFP-08-0067**

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

### **Response to RAI Questions**

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**ATTACHMENT 1 to JAFP-08-0067 CONTENTS**

**1) Response to RAI Questions**

**3 Pages**

Attachments 2, 6, and 8 contain proprietary information as described in 10 CFR 2.390.  
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**ATTACHMENT 1 to JAFP-08-0067**

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 4 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:

Refer to Structural Integrity Calculation 0800846.302 Revision 0, provided in Attachment 5.

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 4, 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 4 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).

**ATTACHMENT 1 to JAFP-08-0067**

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:

Reference 6.3 is provided in Attachment 2 of this letter. Because Reference 6.3 contains proprietary information, obtained from an EPRI BWRVIP document, it is a proprietary calculation. Attachment 2 includes the calculation and the appropriate affidavit attesting to its proprietary nature, while Attachment 3 provides a non-proprietary version of the calculation. Reference 6.4 is provided as Attachment 4 to this letter. Reference 6.14 contains proprietary information on multiple facilities, therefore, the pages addressing JAF have been excerpted and are provided in Attachment 6 with an affidavit attesting to their proprietary nature and Attachment 7 provides a non-proprietary version of the excerpted pages.

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 4 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 4. 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

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**ATTACHMENT 1 to JAFP-08-0067**

(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.

NEDC-30977-P is a proprietary document and is provided as Attachment 8 to this letter with an affidavit attesting to the proprietary nature of document. Attachment 9 provides the non-proprietary version of the analysis, NEDO-30799.

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**ATTACHMENT 2 to JAFP-08-0067**

**Entergy Nuclear Operations, Inc.  
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**Structural Integrity Associates  
Calculation No. FITZ-10Q-301, Revision 0,  
"Evaluation of Adjusted Reference Temperatures and  
Reference Temperature Shifts," 2/15/2008  
(Proprietary Version with Affidavit)**