Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
James A. Fitzpatrick NPP
P.O. Box 110
Lycoming, NY 13093
Tel 315 342 3840

April 24, 2007
JAFP-07-0053

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

REFERENCES:
1. Letter, Entergy to USNRC, "James A. FitzPatrick Nuclear Power Plant, Docket No. 50-333, License No. DPR-59, License Renewal Application," JAFP-06-0109, dated July 31, 2006
2. Letter, USNRC to JAFNPP, "Requests for Additional Information Regarding the Review of the License Renewal Application for James A. FitzPatrick Nuclear Power Plant (TAC No. MD2666)," dated April 2, 2007

SUBJECT: **Entergy Nuclear Operations, Inc.**
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333, License No. DPR-59
License Renewal Application, Amendment 10

Dear Sir or Madam:

On July 31, 2006, Entergy Nuclear Operations, Inc. submitted the License Renewal Application (LRA) for the James A. FitzPatrick Nuclear Power Plant (JAFNPP) as indicated by Reference 1.

This LRA amendment consists of three attachments. Attachment 1 provides responses to the requests for additional information as detailed by the NRC in a telephone conference call on March 5, 2007. Attachment 2 contains a response to RAIs provided in Reference 2. Attachment 3 contains the updates to LRA Table 3.3.2-3 and 3.3.2-5 identified by the NRC April 9, 2007.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 24TH day of April, 2007.

Sincerely,

PETE DIETRICH
SITE VICE PRESIDENT

PD/cf

A124

Attachment 1, 2 and 3

cc:

Mr. N.B. (Tommy) Le, Senior Project Manager
License Renewal Branch B
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-11-F1Washington, DC 20555

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

NRC Resident Inspector
U. S. Nuclear Regulatory Commission
James A. FitzPatrick Nuclear Power Plant
P.O. Box 136
Lycoming, NY 13093

Mr. John P. Boska, Project Manager
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8-C2
Washington, DC 20555

Mr. Paul Eddy
New York State Department of Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Mr. Peter R. Smith, President
NYSERDA
17 Columbia Circle
Albany, NY 12203-6399

Attachment 1

James A. FitzPatrick Nuclear Power Plant

License Renewal Application – Amendment 10

RAI Responses:

2.4.1-2

2.4.1-3

2.4.1-5

2.4.3-1

3.5.2-3

3.5.2-4

The U.S. Nuclear Regulatory Commission (NRC or the staff) and representatives of Entergy Nuclear Operations, Inc., (ENO) held a telephone conference call on March 5, 2007, to discuss and seek clarification regarding the applicant's responses to the staff RAI letter dated January 12, 2007. The following are draft requests of clarification of the applicant's previous RAI responses dated February 14, 2007 concerning the James A. FitzPatrick Nuclear Power Plant (JAFNPP), license renewal application (LRA).

RAI 2.4.1-2

Drywell head closure bolts are listed in Table 2.4-4 as support bolting. Since these bolts are very important, the bolts should be described in the table clearly.

Discussion: The NRC staff indicated that the drywell head closure bolts should be included in Table 2.4-1, Reactor Building Components Subject to Aging Management Review. The applicant indicated that the head closure bolts were included in Table 2.4-4, Bulk Commodities Components Subject to Aging Management Review. The applicant has agreed to clarify whether the drywell closure bolts should be included in Table 2.4-1 or Table 2.4-4 in a supplemental response.

Response:
Drywell head closure bolts have been included in Table 2.4-1. LRA page 2.4-19, Table 2.4-1, item "Drywell head manway cover" is revised to read "Drywell head manway cover and drywell head closure bolts". Accordingly, LRA page 3.5-56, Table 3.5.2-1, item "Drywell head manway cover" is revised to read "Drywell head manway cover and drywell head closure bolts".

RAI 2.4.1-3

Is refueling drain/seal pipe obstruction free? What are the maintenance activities or administrative procedures in place for drain and in the trough area?
Oyster Creek had a crack in the steel liner of the refueling cavity pool. Based on this what are the inspection results of the liner?

Discussion: NRC personnel were concerned with obstructions in the refueling drain and how does the applicant ensures the refueling drain was obstruction free. NRC personnel also requested additional information to show inspection and preventative maintenance activities of the steel liner of the refueling pool. The applicant indicated that JAFNPP had alarms installed to indicate leakage in the refueling trough area. The applicant also indicated that preventative maintenance was performed on alarms and refueling drains every outage. It was agreed that the applicant would provide the preventative maintenance documentation for the alarms and drains in a supplemental response. The applicant would also provide additional information to show inspection and preventative maintenance activities of the steel liner of the refueling pool.

Response:

Refueling seal drains

The outer refueling bellows drains prevent potential refueling cavity leakage from entering the annulus air gap above the sand cushion. Inspections of outer refueling bellows drain lines were performed at JAF prior to start-up from the 1988 refuel outage. Five of the six refueling bellows leakage drain lines were inspected through inspection ports. The drain lines were found to contain minor accumulation of debris. The debris included pieces of weld rod indicating that it had been introduced during construction. The amount of debris did not impact the ability of the drain lines to perform their function. The lines were vacuumed and reinspected and determined to be clear and functional as designed. An inspection port could not be installed in the sixth line because of the piping configuration and the limited space available for access.

The bellows assembly seals the trough area where the drain lines originate, preventing ingress of debris that could lead to obstruction of the lines. The only access to the area for the 1988 inspections was through the drain lines with a boroscope. Because of this design coupled with the redundancy in the number of drain lines, periodic inspections are not warranted to verify the refueling bellows drain piping is free of obstructions.

Leakage through the outer refueling bellows, if any, flows into a common drain line equipped with a flow indicator/switch that will alarm in the control room in the event of bellows leakage. A functional test (PM) of the flow switch 19FIS-62 is performed prior to every refueling outage (2 year frequency) to verify the indicator/switch and associated control room annunciator are functional. No failures of this test have been identified during performance.

Refueling Cavity Liner

At JAF, there has been no indication of refueling cavity steel liner leakage. Leakage through the liner, if any, would enter the trough area below the refueling bellows assembly and flow from there into the drain system with the flow alarm that is discussed above. JAF has not experience flow alarms during previous refueling outages.

If leakage exceeded the capacity of the six drain lines (four 4" drain lines and two 2" drain lines), it would enter the annulus air gap and be detected flowing from three of the four annulus air gap drains in the torus room. JAF has examined the air gaps through the drain lines using fiber optic cables in 1988 and recently in April 2007. No evidence of moisture potentially causing corrosion of the drywell shell has been identified to date. In the future, if any evidence of moisture is identified, JAF will determine additional inspection activities, as appropriate.

Although no formal inspection or maintenance procedure is required for the refueling cavity liner, routine observation during refueling operations and monitoring of the

refueling bellows drain system and associated alarm have indicated no leakage of the liner.

RAI 2.4.1-5

LRA Table 2.4-4 should list reinforced shield plugs separately from manway and hatches.

Discussion: NRC indicated that these items should not be listed on Table 2.4-4 together because they are different items. The applicant indicated that they would investigate further and provide a supplemental response.

Response:
Concrete shield plugs have been added as separate line item on Table 2.4-1. The new line item below is revised to LRA Table 2.4-1, under component "concrete".

Component	Intended Function
Concrete	
Concrete shield plugs	Shelter or protection

Accordingly, LRA Table 3.5.2-1, is revised to add the new line item below with material grouping "concrete".

Concrete shield plugs	EN, SSR	Concrete	Protected from weather	None	Structures Monitoring			I, 501

RAI 2.4.3-1

To justify the main stack is sufficiently located far from the Seismic class I structures, provide sketch to show distance from the stack to those structures seismic Class I and II and have potential tornado induced interaction.

Discussion: Applicant has agreed to provide a clarification of potential tornado induced interaction and will provide a drawing showing main stack location relative to seismic Class I structures.

Response:
The main stack is designed as a Seismic Class 1 structure but is not designed for tornado loads. The nearest Seismic Class 1 or 2 structure to the stack is the standby gas treatment (SBGT) room that is located at a distance that is slightly less than the 'topple' zone of the main stack. The site drawings FY-12B and FY-12D (attached) show the main stack and reactor track bay (which contains the SBGT room). Calculation JAF-CALC-BYM-04122 was performed to confirm the failure mode for the main stack is crushing and 'breaking' at a location above the base. The conclusion states that it is unlikely the main stack failure would interact with the SBGT room. For license renewal,

the main stack is in scope and subject to aging management review. The effects of aging on the main stack are managed by the Structures Monitoring Program.

RAI 3.5.2-3

1. Under paragraph 'Drywell Shell Exterior', the sentence reads" JAF determined that only one out of four air gap drain lines is required to performed the function that they were designed for, the draining of condensates which may form in the air gap". The staff requests JAF to explain how these bases were established?

   Response:

   The architect engineer, Stone & Webster (SWEC) stated that one of the four upper sand cushion drain lines would be sufficient to perform the function of draining any condensates which may form in the two inch air gap. Per Attachment 1 to NYPA Memorandum #JTS-88-0848, dated November 8, 1988, SWEC stated that their search of the project files, including specifications, calculations and job books did not locate any design basis for these drains. SWEC further stated that based on a preliminary evaluation, one 2" drain line would have enough capacity to drain any moisture resulting from condensation on the drywell shell. Also, any condensation caused by cooling of the drywell would return to the vapor state when the drywell heats up. Therefore, condensation should not be considered a major concern.

2. In the same paragraph, the staff requests to address conditions of the stainless steel plates and adhesive that are used to cover the sand cushion.

   Discussion: The applicant will provide NRC a copy of the evaluation performed by Stone & Webster for drain lines. Question 2 was closed when applicant provided proof that this was addressed in document JAFP-07-0021, page 16.

   Response:

   As stated in LRA Amendment #6, JAFP-07-0021, page 16, JAF stated that additional examination of the drywell air gap will be performed in 2007. Preliminary details of the examination are discussed in question 2.4.1-3 above.

RAI 3.5.2-4

How is JAFNPP monitoring the vent pipe bellows? Has JAFNPP considered a Type B test?

Discussion: The applicant indicated that a Type B test is performed once every 10 years. The applicant will provide a supplemental response.

Response:
JAF performs the Type "B" Leak Rate Test once every ten years in accordance with ST-39B and ST-39B-X201. The testing interval is in accordance with the requirements of Appendix J.

Attachment 2

James A. FitzPatrick Nuclear Power Plant

License Renewal Application – Amendment 9

Updated RAI Responses:

RAI 3.1.2-1

RAI

Background:

In Table 3.1.2-1, the applicant applied TLAA-metal fatigue as an aging management program to manage fatigue cracking for many Class 1 components such as reactor vessel internal attachments and welds, the incore monitor housing bolting, CRD housings, CRD stub tubes, the CRD return line, the reactor vessel (the shell, upper and bottom head and the closure flange), and the reactor vessel nozzles (including safe ends and thermal sleeves) for various piping and instrumentation connections.

Chapter 4.3 of NUREG-1800 provides guidance on the metal fatigue analysis which is focused on the fatigue analysis based on the ASME Code, Section III, of using cumulative usage factor concept. The ASME Code, Section III analysis assumes that no crack exists in the components. Chapter 4.3 of the FitzPatrick license renewal application has addressed the requirements of the ASME Section III fatigue calculations.

However, a metal fatigue analysis based on a known initial flaw should also be performed for those components that contain flaw(s). The calculation is performed to assess the stability of the final crack size of the affected component at the end of the license. This analysis will demonstrate that the component has sufficient fracture toughness to resist rapid crack propagation and thus arrest the crack. The method for this calculation would follow the ASME Code, Section XI. Chapter 4.3 of the FitzPatrick license renewal application discussed this analysis without providing much details.

RAI 3.1.2-1

(a) Please identify all the Class 1 components in Table 3.1.2-1 that contain indications or flaws remained in service at the FitzPatrick nuclear plant based on the acceptance criteria of the ASME Code, Section XI,

(b) Discuss the flaw evaluations (such as procedures and assumptions) performed for the affected components in accordance with the ASME Code, Section XI,

(c) Discuss the number of years assumed in the associated fatigue crack growth analysis, and

(d) Discuss whether the affected components are demonstrated to be acceptable for the extended period of operation.

Response to RAI 3.1.2-1

During the integrated plant assessment for license renewal, JAFNPP reviewed the analyses of flaws discovered during in-service inspections (ISI). The only TLAA identified during this review was the fatigue analysis of the shroud repair hardware installed as a repair for shroud cracking. This TLAA was discussed in the application with other fatigue TLAA in Section 4.3.1.2.

Eight component indications were identified. Six of these indications (items 1 through 6 below) were justified for further operation using methods other than ASME Section XI; consequently, they do not meet the criterion of RAI-3.1.2-1 a) and are not discussed in detail. Items 7 and 8 have subsections that respond to the four subparts of the RAI.

1 **Torus shell**

The flaw in the torus shell was removed. As such this flaw no longer exists and has no associated flaw growth analysis.

2 **Residual heat removal shutdown cooling line**

The flaw in the residual heat removal shutdown cooling line was removed. As such this flaw no longer exists and has no associated flaw growth analysis.

3 **Steam dryer**

The steam dryer is a non-code, non-pressure boundary part. The flaws were found during inspections recommended by the BWRVIP, as implemented through the ISI program. The indications were evaluated using BWRVIP-139 guidance, not using ASME Section XI. One flaw was repaired in 2006 (RO17) and the other flaw was within acceptance criteria for continued service. Subsequent inspection revealed that the remaining flaw remains within acceptance criteria for continued service. The flaw will continue to be monitored. The BWR Vessel Internals Program in accordance with BWRVIP guidelines will continue to manage the effects of aging on the steam dryer through the period of extended operation.

4 **Core spray line (inside the vessel)**

The "B" loop core spray line within the reactor vessel has two indications that are monitored under the BWR Vessel Internals Program. The flaws were found during inspections recommended by the BWRVIP.

The first crack was discovered in 1988 between the core spray nozzle and shroud and was repaired with a clam shell sleeve. The clam shell sleeve is not an ASME code repair and does not involve a flaw growth analysis. The clam shell sleeve is the "new" pressure boundary for the weld that was cracked essentially removing the existing flaw from service. The structural integrity of the flawed weld is not credited for demonstrating acceptability for continued service. The clam shell sleeve repair is inspected as part of the BWRVIP inspection program for structural integrity and no cracking has been

detected. The core spray piping inside the vessel is not ASME Code piping but was installed as part of the RPV internals.

A second indication (at weld CSB-12 (P3)), was discovered in 2000 (RO14). There was no change in the length of this indication between 2000 (RO14) and 2002 (RO15). The 2002 inspections and more detailed inspections in 2006 indicate that the indication is a scratch, rather than a crack. Since this indication is not a flaw, no repair was required.

Cracking of the core spray lines, including these indications, will continue to be managed under the BWR Vessel Internals Program per BWRVIP-18A guidelines through the period of extended operation.

5 Shroud Cracking

Shroud inspections performed per BWRVIP-76 guidelines identified crack-like indications for vertical welds SV5A, SV5B, SV6A, SV6B and horizontal weld H4 during the 1996 (RO12) and 1998 (RO13) refueling outages. Re-inspection intervals for the vertical welds are determined based on flaw growth analyses in accordance with BWRVIP guidelines. Because these analyses justify a re-inspection interval shorter than the life of the plant, they are not TLAA. These analyses are not performed in accordance with ASME code requirements.

A shroud tie rod repair was installed to maintain shroud integrity should the horizontal welds fail. Consequently, structural integrity of the shroud does not rely on the horizontal welds. The JAFNPP shroud repair (tie rods) is a different design than that installed at the Hatch nuclear power plant. Consequently, the bracket of the JAFNPP shroud tie rods does not require repair to preclude the cracking experienced at the Hatch plant. JAFNPP inspects tie rods routinely as specified in BWRVIP program requirements. The ten tie rods were inspected in 2006 (RO17) with no degradation noted.

6 Main Steam Nozzle

UT inspection performed in 1988 as part of the JAFNPP ISI program revealed a subsurface indication on main steam nozzle N3C. Re-examination of the N3C nozzle in 1989 and 1990 revealed no discernable change in the size of the indication. This indication is believed to be a minor weld defect that was acceptable by radiography during original construction. This indication is acceptable per Section IWB-3610 (b) of ASME Section XI and no flaw growth analysis is required. This indication will continue to be monitored through the Inservice Inspection Program.

7 CRD Return Line Nozzle to End Cap Weld

a) In 2000, JAFNPP discovered a crack on the inside diameter of the weld between the CRD return line nozzle and the end cap. The CRD nozzle cap was repaired using a weld overlay.

b) JAFNPP Relief Request #26 requested relief from the repair criteria of 10CFR50.55a(c)3 as it pertains to the control rod drive return nozzle cap. This relief request cited code case N-504-1. Section (g)(2) of the code case states that an evaluation of the repair "shall demonstrate that the requirements of IWB-3640 from the 1983 Edition and Addenda, are satisfied for the design life of the repair, considering potential flaw growth due to fatigue and the mechanism believed to have caused the flaw. The flaw growth evaluation shall be performed in accordance with Appendix C."

   The purpose of the flaw growth analysis called out in the code case is to justify a short, thin overlay that uses the strength of the remaining weld as part of the structural support of the joint (as described in Section 4.4.2 of NUREG-0313). Analysis is necessary to determine how much of the original weld will remain to provide structural support at the end of service life. However, the JAFNPP overlay carries the structural load previously borne by the flawed weld. The overlay analysis conservatively assumes the underlying flaw is through wall 360° around the pipe, i.e., complete separation of the underlying pipe, as described in Section 4.4.1 of NUREG-0313. The flaw was assumed to go 100% through wall and thus no credit is taken for structural integrity. Fatigue crack growth analysis is not needed because the design assumption for the weld overlay of the affected CRD return line cap weld is that the flaw will grow 100% through wall.

c) There is no fatigue crack growth analysis associated with the repair.

d) The NRC staff accepted the FitzPatrick relief request and issued an SER on October 26, 2000. Based on its review as documented in the SER, the NRC staff concludes that the proposed alternative provides reasonable assurance of structural and pressure boundary integrity of the RPV capped N9 nozzle and, thus, provides an acceptable level of quality and safety. Potential cracking of the weld overlay is managed during the period of extended operation through inspections per the guidelines of BWRVIP-75-A.

   References:

   JAFNPP letter JAFP-00-0239, M. J. Colomb to USNRC Document Control Desk, *Proposed Alternative for the Contingency Repair of the Control Rod Drive Cap to Reactor Pressure Vessel Nozzle per Generic Letter 88-01 – Relief Request (RR-26)*, 15 October 2000

   Cases of ASME Boiler and Pressure Vessel Code, Case N-504-1, Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping, Section XI, Division 1

8 Weld Overlays to Address IGSCC Indications

a) In addition to the CRD return line cap weld overlay, JAFNPP has applied 21 weld overlays to recirculation system piping and 2 overlays to jet pump instrumentation piping to address flaw indications found during inspections performed for the IGSCC program.

b) The overlays were designed and installed in accordance with Generic Letter 88-01, NUREG-0313, Rev. 2, and ASME Code requirements. In particular, the requirements of Section 4.4.1 of NUREG-0313 suggest that the overlay be designed assuming the original crack was through wall 360 degrees. The overlay is then designed large enough to assume all the load previously borne by the flawed weld. Therefore there is no flaw growth analysis associated with the underlying flaw. (Note that Section 4.4.2 of NUREG-0313 provides alternate guidance for installing smaller overlays that rely on the strength of the underlying weld. These overlays require a flaw growth analysis to determine how much of the original (cracked) weld will be intact at the end of service life. The JAFNPP overlays are in accordance with 4.4.1 and not 4.4.2 of NUREG-0313.)

c) There is no fatigue crack growth analysis associated with the repair.

d) Because the overlays are sized based on complete weld failure despite compressing the weld and preventing future crack growth (compressive residual stress retards future crack growth), the design is conservative and the affected components are acceptable for the period of extended operation. Confirmatory inspections of the weld overlays will continue in the period of extended operation as specified in the BWR Vessel Internals Program using the guidelines of Section 3.5.1.1 of BWRVIP-75-A.

Attachment 3

James A. FitzPatrick Nuclear Power Plant

License Renewal Application – Amendment 10

Table Updates:

Table 3.3.2-3

Table 3.3.2-5

Amendment to JAFNPP LRA

During the review of the LRA, two errors were found in LRA Tables 3.3.2-3 and 3.3.2-5. In Tables 3.3.2-3 and 3.3.2-5 the following line items are incorrect:

Table 3.3.2-3

Strainer	Filtration	Copper Alloy > 15% Zn	Air-untreated (ext)	None	None			G

Table 3.3.2-5

Bolting	Pressure boundary	Stainless steel	Air – outdoor (ext)	None	None			G

These line items are replaced with the following:

Table 3.3.2-3

Strainer	Filtration	Copper Alloy > 15% Zn	Air-untreated (ext)	Loss of material	Periodic Surveillance and Preventive Maintenance	VII.G-9 (AP-78)	3.3.1-28	E, 310

Table 3.3.2-5

Bolting	Pressure boundary	Stainless steel	Air – outdoor (ext)	Loss of Material	Bolting Integrity			G