Entergy

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June 20, 2007
JAFP-07-0079

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, Entergy to USNRC, "License Renewal Application, Amendment 8," JAFP-07-0047, (TAC No. MD2666) dated April 6, 2007
3. Letter, Entergy to USNRC, "License Renewal Application, Amendment 10," JAFP-07-0053, (TAC No. MD2666) dated April 24, 2007
4. Letter, Entergy to USNRC, "License Renewal Application, Amendment 6," JAFP-07-0021, (TAC No. MD2666) dated February 12, 2007
5. Letter, Entergy to USNRC, "License Renewal Application, Amendment 9," JAFP-07-0048, (TAC No. MD2666) dated April 6, 2007
6. Letter, Entergy to USNRC, "License Renewal Application, Amendment 11," JAFP-07-0067, (TAC No. MD2666) dated May 17, 2007

SUBJECT: **Entergy Nuclear Operations, Inc.**
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333, License No. DPR-59
License Renewal Application, Amendment 12

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.

Attachment 1 contains a response to questions concerning LRA section 4.3.1, Class 1 Fatigue, previously committed to in Reference 2. Based on the information provided in Attachment 1 Entergy determined a need to update the 60 year cycle projections, due to the identification of additional startup/shutdown transients not previously considered which are detailed in Attachment 1. These updated projections will be included in the future fatigue management activities described in LRA Commitment #20 documented in Reference 5. Attachment 2 contains clarifications to previous RAIs provided to the NRC in References 3, 4, 5, and 6, as requested by the NRC license renewal staff.

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

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NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 26TH day of June, 2007.

Sincerely,

PETE DIETRICH
SITE VICE PRESIDENT

PD/cf

Attachments 1 and 2

cc:

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Attachment 1

James A. FitzPatrick Nuclear Power Plant

License Renewal Application – Amendment 12

RAI section 4.3.1, Class 1 Fatigue
SIR-07-084-NPS, Rev. 1

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June 8, 2007
SIR-07-084-NPS, Rev. 1

Mr. Kenneth Phy
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James A. Fitzpatrick Nuclear Power Plant
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SUBJECT: SI Response to NRC Requests for Additional Information at Fitzpatrick

References:

1. James A. FitzPatrick Nuclear Power Plant License Renewal Application, submitted July 31, 2006, SI File No. FITZ-08Q-210.
2. USNRC Letter, Requests for Additional Information Regarding the Review of the License Renewal Application for James A. FitzPatrick Nuclear Power Plant (TAC No. MD2666), February 23, 2007, SI File No. FITZ-08Q-210.
3. General Electric Report No. EAS-149-1286, DRF B13-01391, "Reactor Pressure Vessel Fatigue Evaluation for the James A. FitzPatrick Nuclear Power Plant," January 1987, SI File No. FITZ-08Q-201.
4. Structural Integrity Associates Report No. SIR-02-045, Revision 1, "Updated Fatigue Analysis for James A. Fitzpatrick Nuclear Power Plant Reactor Pressure Vessel Components," September 2002, SI File No. W-NYPA-78Q-401.

Dear Ken:

Structural Integrity Associates, Inc. (SI) is pleased to provide responses to U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information (RAIs) related to Section 4.3.1 of the License Renewal Application (LRA) for the James A. Fitzpatrick Nuclear Power Plant (JAFNPP). Our proposed responses are included as Attachment 1 to this letter.

Entergy Nuclear Northeast (ENN) is in the process of getting the LRA for JAFNPP approved by the NRC. Section 4.3.1 of the LRA addresses the effects fatigue may have on ASME Class 1 structures, systems and components. Prior to License Renewal activities, an evaluation was performed for JAFNPP by General Electric to update the original RPV analysis to determine fatigue usage factors resulting from plant operating data through 1986 [3]. In 1992, the fatigue usage factors were revised to include the effects of Power Uprate and an updated feedwater nozzle fracture mechanics evaluation.

The NRC staff conducted an on-site Time-Limited Aging Analysis (TLAA) audit on January 8th and 9th, 2007 and has identified areas where additional information is required. The questions are divided into three parts; Parts A, B & C. The attachment to this letter summarizes the NRC staff questions and SI's proposed responses.

Please note that during the course of our review, an error was identified in a supporting calculation used to develop the Reference [4] report. SI has determined that the number of Scram events after 11.4 years of plant operation in the Reference [3] GE report was not correctly translated into the Reference [4] SI report. This resulted in an inaccurate estimate of current cycle counts and projected 60-year transient cycles. In accordance with the SI Quality Assurance Program, Corrective Action Report (CAR-07-06) and Nonconformance Report (NCR 07-05) were initiated and transmitted to ENN.

Following further review of plant data, SI was able to re-classify the twelve missing transients as Startup and Shutdown events, identified an additional startup/shutdown transient not previously considered, and re-classified six Scram events as startup/shutdown events. With the incorporation of these 19 events, JAFNPP remains within the current allowable cycle limit in the UFSAR, and is expected to remain within the current design basis analyzed number of cycles for the current licensed operating period. However, this will require a revision to the JAFNPP cycle counting procedure and LRA Table 4.3-2 to incorporate the updated 60-year cycle projections for the affected events.

Since JAFNPP has already committed to the NRC to update existing fatigue evaluations to address environmentally assisted fatigue, we recommend that this information be addressed as a part of that effort.

Please feel free to contact me with any questions you have.

Prepared by:

Date: 6/8/2007

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Senior Consulting Engineer

Reviewed by:

Gary L. Stevens

Date: 6/8/2007

Gary L. Stevens, P.E
Senior Associate

Approved by:

Date: 6/8/2007

Terry J. Herrmann, P.E.
Senior Consulting Engineer

cc: FITZ-08Q-401
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Entergy License Renewal Services (ANO-1):

M. Stroud	S. Batch	A. Cox

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J

REFERENCES

1. Reactor Thermal Cycles, GE Drawing No. 729E762, Revision 0, JAFNPP Drawing 11825-5.15-1A, September 15 1967, SI File No. W-NYPA-78Q-205.

2. Structural Integrity Associates Report No. SIR-02-045, Revision 1, "Updated Fatigue Analysis for James A. Fitzpatrick Nuclear Power Plant Reactor Pressure Vessel Components," September 2002, SI File No. W-NYPA-78Q-401.

3. JAFNPP Technical Requirements Manual, Revision 18, SI File No. FITZ-08Q-207.

4. JAFNPP Updated Final Safety Analysis Report, Chapter 4, Table 4.2-3 "Reactor Pressure Vessel Thermal Cyclic/Transient Limits", Rev 5/05, SI File No. FITZ-08Q-206.

5. New York Power Authority Memorandum JTS-95-0684, "Response to DER 95-1499, Discrepancy Between GE Fatigue Report and Reactor Analyst Procedure for Tracking Thermal Cycles", March 15, 1986, SI File FITZ-08Q-212.

6. Entergy Nuclear Northeast Document No. JAF-SE-03-002, "Updated Reactor Pressure Vessel Fatigue Analysis," December 2003, SI File No. FITZ-08Q-205.

7. Combustion Engineering Report No. CENC-1159, "Analytical Report for PASNY Reactor Vessel for FitzPatrick Station," August 1971, SI File No. W-NYPA-78Q-204.

8. General Electric Report No. EAS-149-1286, DRF B13-01391, "Reactor Pressure Vessel Fatigue Evaluation for the James A. FitzPatrick Nuclear Power Plant," January 1987, SI File No. FITZ-08Q-201.

9. GE Nuclear Energy Certified Design Specification No. 25A5024, Revision 0, "Reactor Vessel – Power Uprate," July 1991, SI File No. W-NYPA-78Q-203.

10. Structural Integrity Associates Calculation, Revision 1, "Updated Cycle Counts and 60 Year Projections," SI File No. W-NYPA-78Q-301.

11. Entergy Nuclear Northeast Engineering Report No. JAF-RPT-05-LRD04, Revision 1, "TLAA – Mechanical Fatigue," July 2006, SI File No. FITZ-08Q-204.

12. Structural Integrity Associates Corrective Action Report 07-06, "Calculation NYPA-78Q-301 did not Account for 12 Unidentified Transient Events", dated 4/4/2007, SI File No. FITZ-08Q-109.

13. J. A. FitzPatrick computer logs and strip charts from GE evaluation EAS-149-1286, SI File No. FITZ-08Q-203.

14. Shift Supervisor Operating Logs from J. A. FitzPatrick, SI File No. FITZ-08Q-208.

15. Scram Reports from J. A. FitzPatrick, SI File No. FITZ-08Q-209.

16. J. A. FitzPatrick Thermal Cycles Summary Reports (RAP-7.3.31), SI File No. NYPA-78Q-201.

17. Nozzle Thermal Cycles, GE Drawing No. 135B9990, Revision 1, Sheets 1 through 8, JAFNPP Drawings 11825-5.01-50A through 57A, May 22 1967, SI File No. W-NYPA-78Q-206.

18. General Electric Report No. NEDC-32068, "Reactor Pressure Vessel Power Uprate Stress Report Reconciliation for the Fitzpatrick Power Plant", March 1992, NYPA File No. S-90-00978-38A, SI File No. NYPA-58Q-214P.

19. Structural Integrity Associates Calculation, Revision 1, "Fatigue Analysis of Feedwater Nozzle," SI File No. W-NYPA-78Q-305.

20. Structural Integrity Associates Calculation, Revision 1, "Fatigue Analysis of Recirculation Inlet Nozzle," SI File No. W-NYPA-78Q-308.

21. Structural Integrity Associates Calculation, Revision 2, "Fatigue Analysis of Control Rod Drive Nozzle," SI File No. W-NYPA-78Q-311.

22. Structural Integrity Associates Calculation, Revision 0, "Revised Fatigue Calculation for the RPV Closure Region Bolts," SI File No. W-NYPA-78Q-302.

RAI 4.3.1-1, Part A

Licensing renewal application (LRA) Table 4.3-2 gives the current design basis allowable cycles and updated 60-year cycle projections for the James A. FitzPatrick Nuclear Power Plant (JAFNPP) design basis transients. The cycle values in the "Current Design Basis Cycles, Allowable" column of the table represent the updated current design basis allowable cycles performed by Structural Integrity Associates (SIA) and the cycle values in the "Updated 60-year Cycle Projection" column of the table represent 60-year cycle projections as of actual JAFNPP operations through Spring 2005. The staff requests the following additional information:

RAI 4.3.1-1, Part A (i)
The original current design basis allowable cycles for the original metal fatigue calculations were performed by General Electric Company (GE). Provide the current design basis allowable cycle values that were calculated by GE for the JAFNPP design basis transients.

Response:
Whereas the original design basis allowable transients were established by GE [1], the current design basis allowable transient cycle values are those established by SIA [2]. Table A-1 shows both the original and current design basis values.

Table A-1: Transient Event Cycle Limits

Event Number	Design Basis Transient	Original Design Basis Analyzed No. of Cycles (GE) [1]	Revised Design Basis Analyzed No. of Cycles (SIA) [2]
1	Bolt-up (70°F)	123	36
2	Design Hydro Test (1250 psig, 100°F)	130	36
3	Startup (100°F/hr to 546°F)	120	233
4	Turbine Roll and Increase to Rated Power	120	221
5	Daily Reduction to 75% Power	10000	7566
6	Weekly Reduction to 50% Power	2000	1685
7, 8	Rod Worth Test (Sequence Exchange)	400	357
	Loss of Feedwater Heater		
9	Turbine Trip at 25% Power	10	7
10	Feedwater Heater Bypass	70	34
	SCRAMs		
11	Loss of FW Pumps, MSIVs Close	10	12
12	Turbine Generator Trip, FW on, MSIVs stay open	40	12
13	Reactor Overpressure	1	1
14	Single Relief or Safety Valve Blowdown	2	2
15	All Other Scrams	147	64
16	Normal Operation	---	---
17	Improper Start of Cold Recirculation Loop	5	5
18	Sudden Start of Pump - Cold Recirculation	5	5
	Shutdowns (events 19-23)	118	233
19	Reduction to 0% Power		
20	Hot Standby		
21	Cooldown (to 375°F @100°F/hr)		
22	Vessel Flooding (375°F to 330°F in 10 minutes)		
23	Cooldown (330°F to 100°F @ 100°F/hr)		
24	Hydrostatic Test (1563 psig, 100°F)	3	1
25	Unbolt (100°F)	123	35
26	Refueling (70°F)	--	--

RAI 4.3.1-1, Part A (ii)
Clarify what regulatory process was used to allow SIA's updated current design basis allowable cycle values as the current design basis for the JAFNPP design basis transients.

Response:

The JAFNPP licensing basis was updated in accordance with 10CFR50.59 per JAF-SE-03-002, "Updated Reactor Pressure Vessel Fatigue Analysis" [6]. The review was performed considering the original CE analysis [7] and subsequent GE analyses [8, 9] and SIA analyses [2, 10].

The revised number of allowable cycles was included in UFSAR Section 4.2, Table 4.2-3 "Reactor Pressure Vessel Thermal Cyclic/Transient Limits" [4]. Per procedure, JAFNPP monitors transients affecting reactor vessel fatigue usage as required by Section 5.5.5 "Component Cyclic or Transient Limit" of the Technical Specifications [3].

RAI 4.3.1-1, Part A (iii)
Discuss the methods used to establish the original current design basis allowable cycles performed by GE and the updated current design basis allowable cycles by SIA. Identify the differences in the methods used by GE and SIA and justify why SIA's updated current design basis allowable cycle assessment is acceptable to use as the current design basis for JAFNPP.

Response:

During the original design of JAFNPP, GE, as supplier of the Reactor Pressure Vessel (RPV) and associated ASME Section III Class 1 components, developed thermal and pressure cycle diagrams for the RPV and nozzles [1, 17]. These thermal, pressure, and flow rate cycle diagrams established the transient events to be evaluated for cyclic operating conditions. The number of transients was originally established based on an estimate of what would be experienced by the RPV during a 40-year operating period. The original design basis allowable cycles were considered to be conservative bounding values for design.

Combustion Engineering (CE), the RPV fabricator, used the cycles defined by GE as input to calculate fatigue usage for the RPV in accordance with the ASME Boiler and Pressure Vessel Code. The CE analysis [7] was the original design basis fatigue calculation for the JAFNPP RPV.

In 1987, GE performed an updated fatigue evaluation to assess the effect of actual cyclic duty from the first 11.4 years of plant operation on RPV components [8] using temperature, pressure and flow data obtained from plant records. These components included the closure studs, the recirculation inlet nozzle, the feedwater nozzle and the control rod drive nozzle. The analyzed number of cycles used in this evaluation was an estimate made by GE based on plant operating data through 1986. Components were selected for evaluation considering the severity of the stresses as well as their sensitivity to changes in reactor conditions based on past GE fatigue analysis experience. Components which were originally exempted from fatigue evaluation in the original design basis fatigue calculation were reevaluated and those fatigue exemptions were confirmed to remain valid.

A review of actual plant history in 2002 revealed that seven thermal cycle events reached as high as 92% of the original design basis cycle limits. Based on the limited remaining thermal cycle margin for these events, SIA updated both the actual and projected thermal cycle count for JAFNPP [2, 10]. SIA used the same methodology as GE had previously used to update the actual and 40-year projected thermal cycle counts, except that additional plant data that was available was factored into revised projections for 40 and 60 years.

For the 2002 update, thermal cycle transient data collected from plant instrumentation and operator logs was obtained. The data was reviewed and compared to the design basis thermal cycle diagrams. Design inputs for this evaluation primarily consisted of:

1. The thermal cycle monitoring reports for the time period from January 1st, 1989 to June 30th 2001. Data for the time period between 1986 (i.e., end date of previous fatigue update) and 1/1/89 was not available at the time of the 2002 update.

2. Additional plant process computer data and strip charts retrieved from microfiche or provided on CD-ROM by JAFNPP personnel.
3. Design basis thermal cycle diagrams used to categorize the actual thermal cycles captured in the data.
4. RPV Power Uprate Design Specification, since this specification altered many of the transient pressure and temperature profiles.

The 2002 evaluation categorized each thermal transient experienced at JAFNPP based on a comparison to the design thermal cycle diagram [1]. For any transient event where detailed data was unavailable, conservative assumptions were applied to assign each transient event to a bounding design basis event. For example, the GE evaluation [8] identified that there were 4 Loss of Feedwater Pump events (Event 11) through Year 11.4, which was incremented to 5 events in the 2002 SI evaluation [2, 10]. A subsequent JAFNPP review [5] identified that the actual count through Year 19.5 (June 1994) should be 2 events. The Loss of Feedwater Pump transient would result in significantly higher temperature and pressure changes in the RPV bottom head than other events that could reasonably be used (e.g., Shutdown and Startup) if more detailed data had been available.

By reviewing actual plant operating conditions, SIA determined the actual numbers of accrued event cycles. These numbers were used to project the numbers of cycles expected through sixty years of operation [10]. The 60-year projections were used to revise the number of cycles from those originally estimated by GE [1] and included in the CE analytical report [7] and the 1987 GE evaluation [8]. The projected number of cycles for 60 years was set equal to (but never less than) the results from the straight-line projections.

SIA's updated current design basis allowable cycle assessment is acceptable to use as the current design basis for JAFNPP because: (1) associated fatigue calculations demonstrate that CUF values will remain less than allowable, and (2) the Fatigue Monitoring Program ensures that actual numbers of cycles do not exceed the numbers used in the CUF calculations. The Fatigue Monitoring Program also requires that appropriate corrective action be taken prior to any analyzed number of cycles being exceeded.

RAI 4.3.1-1, Part A (iv)
For each transient in LRA Table 4.3-2, clarify how many operational cycles have been recorded up to the time that the 60-year transient projections were calculated, as given in the "Updated 60-year Cycle Projection" column of LRA Table 4.3-2.

Response:

The requested information is provided in Table A-2. The Year 30.5 cycle count data was the most recent data available when the 60-year transient projections were calculated [11]. The cycles as of Year 30.5 from Reference [11] represent an update to the SIA work from 2002 [10], which was done as of 26.5 years of operation.

Table A-2: Number of Cycles as of Year 30.5 Compared to Year 60 Projections

Event Number	Design Transient	Year 30.5 Cycle Count [11]	Updated Year 60 Cycle Projection [11]
1	Bolt-up (70°F)	18	35
2	Design Hydro Test (1250 psig, 100°F)	18	35
3	Startup (100°F/hr to 546°F)	113	216
4	Turbine Roll and Increase to Rated Power	107	204
5	Daily Reduction to 75% Power	3,608	6,674
6	Weekly Reduction to 50% Power	814	1,526
7, 8	Rod Worth Test (Sequence Exchange)	165	310
	Loss of Feedwater Heater		
9	Turbine Trip at 25% Power	5	7
10	Feedwater Heater Bypass	27	32
	SCRAMs		
11	Loss of FW Pumps, MSIVs Close	5	10
12	Turbine Generator Trip, FW on, MSIVs stay open	10	12
13	Reactor Overpressure	0	0[(1)]
14	Single Relief or Safety Valve Blowdown	1	1
15	All Other Scrams	57	62
17	Improper Start of Cold Recirculation Loop	0	0[(1)]
18	Sudden Start of Pump - Cold Recirculation	0	0[(1)]
	Shutdowns (events 19-23)	126	244
19	Reduction to 0% Power		
20	Hot Standby		
21	Cooldown (to 375°F @100°F/hr)		
22	Vessel Flooding (375°F to 330°F in 10 minutes)		
23	Cooldown (330°F to 100°F @ 100°F/hr)		
24	Hydrostatic Test (1563 psig, 100°F)	1	1
25	Unbolt (100°F)	17	34
26	Refueling (70°F)	--	--

Note: 1. Although zero events are anticipated, the number of design cycles remains unchanged from the reference [2] analysis.

RAI 4.3.1-1, Part A (v)
Provide a technical discussion to clarify how the 60-year projections were performed based on recorded transient data. In particular, if a particular transient category in LRA Table 4.3-2 is made up of more than one specific transient, clarify which specific transient is used to define the transient and clarify how the total number of cycles were used to derive the 60 year cycle projections.

Response:

The 60-year projections in LRA Table 4.3-2 were developed based on the same approach used in both the 1987 GE evaluation and the 2002 SIA evaluation. Straight-line projections of historical data and actual transient events recorded through June 30, 2005 (or 30.5 years of plant operation) were utilized, as documented in Reference [11].

There are two instances where more than one transient event is grouped in Table 4.3-2; the Rod Worth Test/Sequence Exchange (Events 7 & 8) and the Shutdown (Events 19-23). Since the RPV does not experience any changes in pressure or temperature for Events 7 & 8, Event 7 does not contribute any fatigue usage to the RPV. The Sequence Exchange (Event 8) does involve a change in feedwater conditions, so this event is used in the evaluation of the feedwater nozzles.

Events 19 through 23 are sequential stages of a normal plant shutdown. A shutdown and cooldown to 100°F entails one each of transient Events 19 through 23. Each Shutdown event is therefore considered to result in one cycle of Events 19 through 23. In the case of the grouped shutdown transients (Events 19-23), the maximum stress intensity for the entire pressure and temperature range is used to calculate fatigue usage. Therefore, all plant shutdown events to cold ambient conditions were assumed to cause each of Events 19 through 23.

In 2002, SIA used various JAFNPP reports [13] to determine the actual cycles recorded during the first 26.5 years of operation since this is the latest data that was available at the time. The number of cycles the plant would incur in 60 years of operation was projected based on a straight-line extrapolation of the rate of cycles observed between Year 3 and Year 26.5 (i.e., the previous 23.5 years of plant operation). The data for the first 3 years of operation included a large number of events associated with initial operation and is not considered representative of future performance. Based on established maintenance rule programs and continuing advances in technology, the future rate of transient occurrence is not expected to exceed the rate of transient occurrence observed during the past 23.5 years of operation.

The 60-year number of cycles for four transients, Event 9 (Turbine Trip at 25% Power), Event 10 (Feedwater Heater Bypass), Event 12 (Turbine Generator Trip with MSIVs Open and Feedwater On) and Event 15 (All Other Scrams), were projected based on the rate of cycles occurring over 13.5 years of operation (Year 13.0 to Year 26.5). Projections for these events did not use data from the first 13 years of operation because there was a significant reduction in the accumulation of these events after Year 13. The projections for these four events based on only the last 13.5 years of operation are considered realistic and more representative for future plant operation.

In preparing the JAFNPP LRA, ENN compiled four additional years of thermal cycle data that were collected since the plant 60-year thermal cycle limits were established by SIA in 2002 [11]. The most recent semi-annual report for the component cyclic or transient limit monitoring program at the time the LRA was being prepared included a record of actual transient events through June 30, 2005, or 30.5 years of operation. Reference [11] performed revised projections to determine whether the thermal cycle limits established by SIA in 2002 remained valid for 60 years of operation, based on the updated Year 30.5 thermal cycle data. The same methods used to project 60-year counts in 2002 were applied to the Year 30.5 data. The results of this analysis are shown in the final column of Table A-2 and Table 4.3-2 of the JAFNPP LRA.

<u>RAI 4.3.1-1, Part A (vi)</u>
Explain how the cycles were recorded prior to 1988, when JAFNPP did not implement a plant computer to track transient events.

Response:

The method used to record transient events has evolved and improved since initial plant operation due to improved procedures and widespread availability of computer technology.

When JAFNPP entered commercial operation, operator log entries were the primary means used to determine when transient events occurred. Periodically, engineering personnel reviewed operator logs, strip charts and plant process computer printouts retained in plant records. A plant procedure [16] was implemented in the 1980s to provide a more structured approach to accounting for the information collected from operator logs, strip charts and computer printouts. This procedure listed each transient event, the number of transient cycles experienced during the reporting period, the total number of cycles accumulated to date and the projected number of cycles to Year 40. The information collected from this procedure is maintained in plant records.

RAI 4.3.1-1, Part A (vii)

Justify why the following values in LRA Table 4.3-2 are acceptable:

(a). A "Current Design Basis Cycles, Allowable" value of "1" and an "Updated 60 Year Cycle Projection" value of "0" for transient category 13, "Reactor Overpressure."

Response:

Refer to Tables A-1 and A-2. The revised design basis number of analyzed cycles for Event 13 is one cycle, as shown in Table A-1. To date, none of these events have occurred and none are expected to occur for 60 years, which is reflected in Table A-2. Since the projected number of cycles through 60 years of operation does not exceed the number assumed in the fatigue analyses, the value is acceptable. The JAFNPP Fatigue Monitoring Program assures that the analyzed number of cycles is not exceeded.

RAI 4.3.1-1, Part A (vii)

(b). A "Current Design Basis Cycles, Allowable" value of "2" and an "Updated 60 Year Cycle Projection" value of "1" for transient category 14, "Single Relief Valve Blowdown."

Response:

Refer to Tables A-1 and A-2. The revised design basis number of analyzed cycles for Event 14 is two cycles, as shown in Table A-1. To date, one of these events have occurred and none are expected to occur in the future, which is reflected in Table A-2. Since the projected number of cycles through 60 years of operation does not exceed the number assumed in the fatigue analyses, the value is acceptable. The JAFNPP Fatigue Monitoring Program assures that the analyzed number of cycles is not exceeded.

RAI 4.3.1-1, Part A (vii)

(c). A "Current Design Basis Cycles, Allowable" value of "5" and an "Updated 60 Year Cycle Projection" value of "0" for transient category 17, "Improper Start of Cold Recirculation Loop."

Response:

Refer to Tables A-1 and A-2. The revised design basis number of analyzed cycles for Event 17 is five cycles, as shown in Table A-1. To date, none of these events have occurred and none are expected to occur for 60 years, which is reflected in Table A-2. Since the projected number of cycles through 60 years of operation does not exceed the number assumed in the fatigue analyses, the value is acceptable. The JAFNPP Fatigue Monitoring Program assures that the analyzed number of cycles is not exceeded.

RAI 4.3.1-1, Part A (vii)

(d). A "Current Design Basis Cycles, Allowable" value of "5" and an "Updated 60 Year Cycle Projection" value of "0" for transient category 18, "Sudden Start of Pump-Cold Recirculation."

Response:

Refer to Tables A-1 and A-2. The revised design basis number of analyzed cycles for Event 18 is five cycles, as shown in Table A-1. To date, none of these events have occurred and none are expected to occur for 60 years, which is reflected in Table A-2. Since the projected number of cycles through 60 years of operation does not exceed the number assumed in the fatigue analyses, the value is acceptable. The JAFNPP Fatigue Monitoring Program assures that the analyzed number of cycles is not exceeded.

RAI 4.3.1-1, Part A (vii)

(e). A total "Current Design Basis Cycles, Allowable" value of "233" and a total "Updated 60 Year Cycle Projection" value of "244" for "Shutdowns", which comprises transient categories Nos. 19, "Reduction to 0% Power;" 20, "Hot Standby;" 21, "Cooldown (100°F/hr to 375°F);" 22, "Vessel Flooding (375°F to 330°F in 10 min.);" and 23, "Cooldown (100°F/hr to 100°F)."

Response:

Refer to Tables A-1 and A-2. The revised design basis number of analyzed cycles for Events 19-23 is 233 cycles, as shown in Table A-1. To date, 126 of these events have occurred and 244 were previously projected to occur for 60 years, which is reflected in Table A-2.

However, as discussed in the answer to Part C below, a review of plant records was performed to classify the twelve unidentified events mentioned in the 1987 GE evaluation [8]. This review was able to identify that these twelve previously unidentified events, as well as one additional event, should be classified as shutdown and startup events. This results in a revised projection of 270 cycles for Events 19-23 for 60 years, as shown in Table C-2 below. The 40-year projected number of cycles for Events 19-23 is 186 cycles. Since the projected number of cycles through 40 years of operation does not exceed the number assumed in the fatigue analyses, the value is acceptable for the remaining operating period. However, since the projected number of cycles through 60 years of operation exceeds the number assumed in the fatigue analyses, corrective action may be required if the trend continues and accumulated numbers of cycles approach the analyzed value of 233. The JAFNPP Fatigue Monitoring Program assures that corrective action will be taken prior to the accumulation of more than 233 cycles.

RAI 4.3.1-1, Part A (vii)

(f). A "Current Design Basis Cycles, Allowable" value of "1" and an "Updated 60 Year Cycle Projection" value of "1" for transient category 24, "Hydrostatic Test (1563 psig)."

Response:

Refer to Tables A-1 and A-2. The revised design basis number of analyzed cycles for Event 24 is one cycle, as shown in Table A-1. To date, one of these events have occurred and none are expected to occur in the future, since these events are no longer required by the ASME Code. This is reflected in Table A-2. Since the projected number of cycles through 60 years of operation does not exceed the number assumed in the fatigue analyses, the value is acceptable. The JAFNPP Fatigue Monitoring Program assures that the analyzed number of cycles is not exceeded.

RAI 4.3.1-1, Part A (vii)

(g). A "Current Design Basis Cycles, Allowable" value of "35" and an "Updated 60 Year Cycle Projection" value of "34" for transient category 25, "Unbolt."

Response:

Refer to Tables A-1 and A-2. The revised design basis number of analyzed cycles for Event 25 is 35 cycles, as shown in Table A-1. To date, 17 of these events have occurred and 34 are projected to occur for 60 years, which is reflected in Table A-2. Since the projected number of cycles through 60 years of operation does not exceed the number assumed in the fatigue analyses, the value is acceptable. The JAFNPP Fatigue Monitoring Program assures that the analyzed number of cycles is not exceeded.

RAI 4.3.1-1, Part B

Page 19 of GE Design Calculation EAS-149-1286 / DRF B13-01391 discusses GE's evaluation of 12 transients (i.e., nine reactor SCRAMS, one turbine trip, two feedwater pump trips) that had been grouped into the "Shutdown" transient for the plant. The report stated that the change in reactor coolant temperature (ΔT) for six of these events had exceeded the ΔT value for this transient. The staff noted that the bases provided on page 19 for justifying why these events can be categorized as plant heatups or cooldowns are based on qualitative analysis without using any temperature gradient data. The staff requests the following additional information:

RAI 4.3.1-1, Part B (i)
Explain why the six transients specified in GE calculation can be grouped into "Shutdown" transient for the plant when the ΔT values for these six events were determined to be excessive and the temperature gradients for the transients are not defined.

Response:

Each of the six transients listed on page 19 of the 1987 GE report was reviewed against the record copy of the original reactor pressure strip charts and computer logs [0], as well as station Scram reports [15], to identify the time frame and rate of temperature change for each transient. Whereas the ΔT for these events exceeded the ΔT for the Rapid Heatup and Rapid Cooldown events, the rate of change of temperature for the actual events was significantly less than the Rapid Heatup and Rapid Cooldown events. In fact, per the assessment provided below for each event, the rate of temperature change remained well within the limits of the Shutdown and Startup events. Therefore, based on the review provided below, these six events have been re-classified.

The information obtained for each of the six transients is summarized as follows:

1-9-83 Scram
The Scram event occurred on 1-9-83 due to a failure of a relay that caused the 345kV breakers to open. The event continued until the plant reached atmospheric pressure in the RPV. The time it took for the change from 430°F to 100°F (-330°F) was greater than 48 hours. The rate of temperature change was at all times less than the design basis transient rate of 100°F/hr for the shutdown transient.

11-4-84 Scram
The Scram event occurred on 11-4-84 with the plant at 30% power during startup from a planned maintenance outage when the condensate bypass controller failed with the reactor at low power. The High Pressure Coolant Injection system started and injected to the reactor to maintain level. Shutdown continued until zero pressure. The shutdown from 397°F to 100°F (-297°F) took approximately 12.5 hours. The temperature change in any 1 hour period remained less than 100°F. Review of the event strip charts showed that the maximum rate of temperature change for the remainder of the shutdown event was approximately 50°F/hr. Following the shutdown, the plant entered a startup to a pressure of approximately 940 psig. The startup event from 100°F to 537°F began on 11-5-84 around 6:30 PM and took approximately 19.5 hours.

7-19-85 Scram

The Scram occurred on 7-19-85 due to a turbine trip concurrent with a loss of the 10300 and 10500 buses while transferring house loads that led to a loss of condenser vacuum when the "A" Circulating Water pump tripped and air admission valves to 3 condenser water boxes opened. The recovery event began on 7-20-85 with the reactor taking approximately 15 hours to increase from 364°F to 540°F (+176°F). The highest rate of temperature increase occurred in the first three hours with temperature increasing from 364°F to 539°F at a rate of less than 60°F/hr.

7-26-85 Scram

The Scram event occurred on 7-26-85 during turbine control valve testing when a Scram signal was received on two Reactor Protection System channels, triggering a full Scram signal. The recovery began on 7-27-85 and continued through 7-28-85. The time to increase 168°F was greater than 48 hours. For much of the time, the pressure held relatively constant. The highest rate of temperature increase occurred between 364°F to 411°F, which occurred over a time period of 1.5 hours with a rate of slightly more than 30°F/hr.

8-20-85 Scram

The Scram event occurred on 8-20-85 when the "B" inboard Main Steam Isolation Valve (MSIV) closed while testing Main Steam line radiation monitors. The event occurred due to a latent failure of one of the solenoid valve coils designed to maintain the MSIV open during the surveillance test. The recovery began on 8-23-85 and continued through 8-25-85. The times were not consistently noted on the charts and the computer logs were illegible during much of that time. The highest observed rate of temperature increase occurred between 470°F to 518°F, which occurred over a time period of 1.5 hours with a rate of slightly more than 30°F/hr.

Based on the above review, the six transient events listed in Table 5 of the GE evaluation were re-classified as Shutdown and Startup events.

RAI 4.3.1-1, Part B (ii)
For the scram event that occurred on November 4, 1984, a ΔT of -297°F and a ΔT of +437°F occurred on the same day, when did ΔT events occur and what were the actual temperature gradients associated with these events.

Response:

Based on a review of all available plant records for this event, the Scram event occurred on 11-4-84 with the plant at 30% power during startup from a planned maintenance outage. The event occurred when the condensate bypass controller failed with the reactor at low power. The High Pressure Coolant Injection system started and injected to the reactor to maintain level. Shutdown continued until zero pressure. The shutdown from 397°F to 100°F (-297°F) took approximately 12.5 hours. The temperature change in any 1 hour period remained less than 100°F. Review of the event strip charts showed that the maximum rate of temperature change for the remainder of the shutdown event was approximately 50°F/hr. Following the shutdown, the plant entered a startup to a pressure of approximately 940 psig. The startup event from 100°F to 537°F began on 11-5-84 around 6:30 PM and took approximately 19.5 hours.

RAI 4.3.1-1, Part B (iii)

Clarify how your response to this part (Part B) factors into your response to Part A, particularly with respect to the number of recorded occurrences for the transient Categories in LRA Table 4.3-2.

Response:

Based on the response to RAI 4.3.1-1, Part B (i) above, the number of recorded occurrences for some of the transient categories in LRA Table 4.3-2 will change. The changes are described in the response to RAI 4.3.1-1, Part C below.

RAI 4.3.1-1, Part C

RAI 4.3.1-1, Part C (i)
In the GE stress report, GE characterized 12 unidentified operational transients as reactor SCRAMS. GE identified that 63 occurrences of these transients had occurred prior to 1987. Verify the operational transients and occurrences identified in the GE stress report and provide your evaluations.

Response:

The operational transients and occurrences identified in the GE report were compared to the data supplied to GE by JAFNPP in 1986. This review verified that the number of operational transients, including the number of unidentified operational transients, is consistent with the input provided to GE.

In order to determine if the unidentified operational transients could be more accurately classified, a review of operating logs [14] for the time period in question was performed. From this data, all of the twelve unidentified operational transients were able to be located, identified and categorized. One additional shutdown event on 6/17/76 was identified during the review and some dates were modified slightly to coincide with the beginning of the events. The resulting classification of these events is provided in Table C-1.

Based on the information provided in Table C-1, the twelve unidentified events and the one additional event were all classified as Shutdown (Event 19-23) and Startup (Event 3) events.

Table C-1: Classification of Unidentified Transients for JAFNPP

Event Date	Description	Classification
3/18/75	Shutdown due to "D" Emergency Diesel Generator failure and inoperable "A" EDG. Plant shut down to cold condition.	Shutdown & Startup
5/22/75	Shutdown following Turbine Trip/Reactor Scram as part of startup test program. Plant shut down to cold condition.	Shutdown & Startup
5/28/75	Shutdown to repair #1 main steam stop valve. Plant shut down to cold condition.	Shutdown & Startup
6/2/75	Reactor Scram and MSIVs closed during test. Continued in Hot Standby until startup commenced later that same day.	Shutdown (1) & Startup
6/11/75	Reactor Scram on MSIV isolation due to loose PCIS connection. Continued in Hot Standby until startup commenced later that same day.	Shutdown (1) & Startup
6/17/75	Reactor Scram on Low RPV water level. Plant shut down to cold condition.	Shutdown & Startup
12-25-75	Shutdown to hot standby to repair a main turbine hydraulic control system (EHC) oil leak from 40% power. Continued in Hot Standby until startup commenced 12-26-75.	Shutdown (1) & Startup
6-4-76	Turbine Trip/Reactor Scram due to Moisture Separator Reheater drain tank high level. Continued in Hot Standby until startup commenced later that same day.	Shutdown (1) & Startup
6-17-76	Reactor Scram during surveillance test. Continued in Hot Standby until startup commenced later that same day.	Shutdown (1) & Startup
6-18-76	Reactor Scram during startup due to exceeding 1st stage pressure setpoint with Turbine Stop Valves shut. Continued in Hot Standby until startup commenced 2 hours later.	Shutdown (1) & Startup
6-30-76	Turbine trip/Rx Scram due to loss of 125VDC input to EHC/load unbalance circuit. Continued in Hot Standby until startup commenced later that same day.	Shutdown (1) & Startup
8-30-76	Reactor Scram while replacing Reactor Water Recirculation flow transmitter. Continued in Hot Standby until startup commenced 9 hours later.	Shutdown (1) & Startup
1-22-77	Reactor Scram on high neutron flux at 44% power. Continued in Hot Standby until startup commenced 11 hours later.	Shutdown (1) & Startup

Note: 1. Classifying the event as a Shutdown (including each of Events 19 through 23) and Startup (Event 3) is conservative because the temperature and pressure changes are significantly greater than for the Scram event (Event #15).

RAI 4.3.1-1, Part C (ii)
In LRA Table 4.3-2, Entergy projects that the number of SCRAM events occurring through 60 years of operation for the "All Other SCRAM" events will be 62. Explain how the number of cycles projected through 60 years of operation can be 62 when 63 occurrences had been recorded through 1987.

Response:

ENN has determined that the number of cycles used in the 2002 SIA evaluation [2, 10] to project through 60 years of operation was incorrectly determined [12] because the evaluation did not properly account for the twelve unidentified transients assigned to the Scram transient category from the 1987 GE analysis [8]. This resulted in an inaccurate estimate of current transient cycles and an inaccurate 60 year transient cycle projection for the Scram event (Event 15). The corrected number of events is provided in Table C-2, considering the updated classifications shown in Table C-1, and the re-classification of the six Scram events discussed in the response to RAI 4.3.1-1, Part B (i).

RAI 4.3.1-1, Part C (iii)
In the GE stress report, GE also mentioned that the change in ΔT associated with these 12 unidentified transients was approximately 330°F. The staff requests the following additional information:

(a). Please define these unidentified transients and list the pressure-temperature data for these transients.

Response:

The change in ΔT of 330°F was not for an unidentified transient, but rather for the Scram which occurred on 1-9-1983. The information related to this transient event was previously discussed in the response to RAI 4.3.1-1, Part B (i). Available data was reviewed and used to further investigate the twelve unidentified transients. As indicated in the response to the previous question, these events have been re-classified as shutdown and startup events. Thus, the maximum possible temperature differential has been assumed to occur for all of the previously unidentified events, thereby making these classifications conservative.

RAI 4.3.1-1, Part C (iii)

(b). Please define the pressure-temperature (P-T) data that were used for the limiting SCRAM event used in SIA's updated 60 year cumulative usage factor calculations.

Response:

The pressure-temperature data is obtained from the original thermal cycle diagrams [1, 17] for JAFNPP and was modified for power uprate, according to References [9, 18]. The evaluation used the governing stress analyses for the closure bolts as a basis to estimate fatigue usage. The remaining three components (recirculation inlet nozzle, feedwater nozzle and CRD penetrations) were reanalyzed using finite element methods. These transients are defined in the SIA evaluation [19, 20, 21]. Figures C-1 through C-3 identify the Scram transients used in the SIA calculations for the feedwater nozzle. Figures C-4 through C-8 identify the Scram transients used in the SIA calculations for the recirculation inlet nozzle. Figures C-9 through C-13 identify the Scram transients used in the SIA calculations for the control rod drive nozzles. These transient definitions reflect the unique pressure and temperature severity of the design basis Scram event, modified for power uprate conditions, for each evaluated component. The severity is different for each component due to the different thermal regions of the vessel, e.g., the feedwater nozzle is affected by incoming feedwater flow. Consistent with the design basis, the Scram event definitions also include the subsequent return to full power conditions after the initiating Scram.

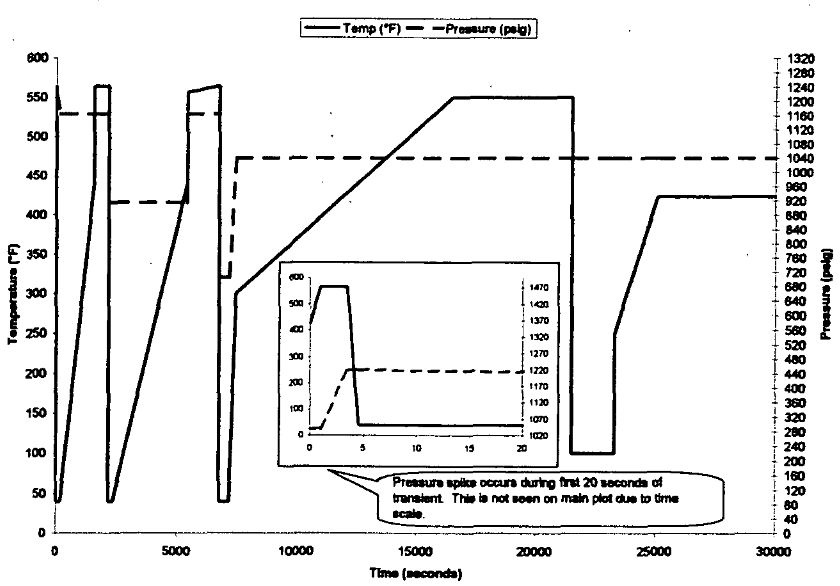

Figure C-1: Transient 11 for Feedwater Nozzle

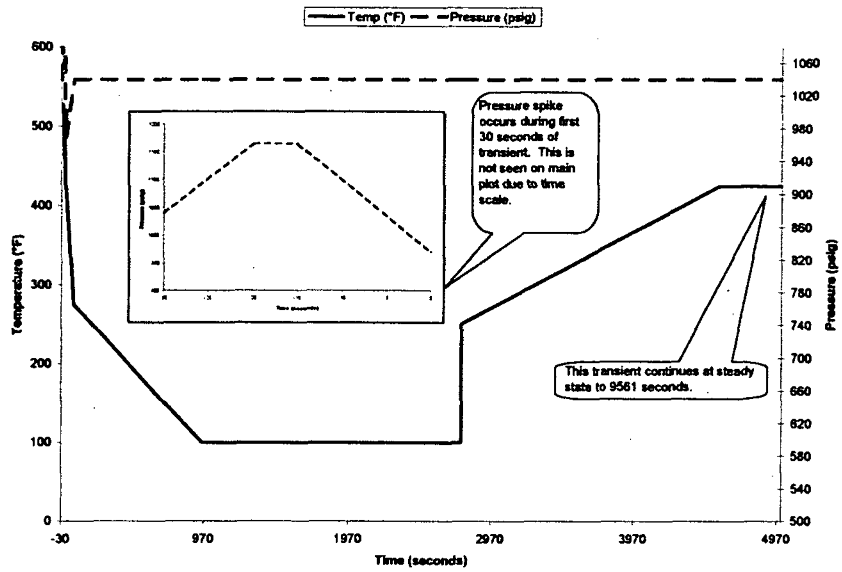

Figure C-2: Transient 12 for Feedwater Nozzle

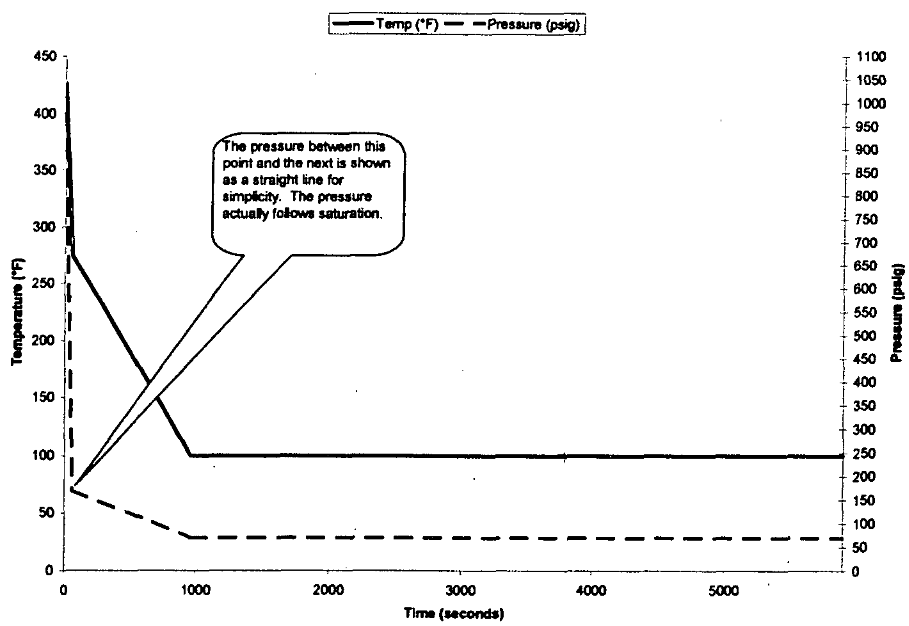

Figure C-3: Transient 14 for Feedwater Nozzle

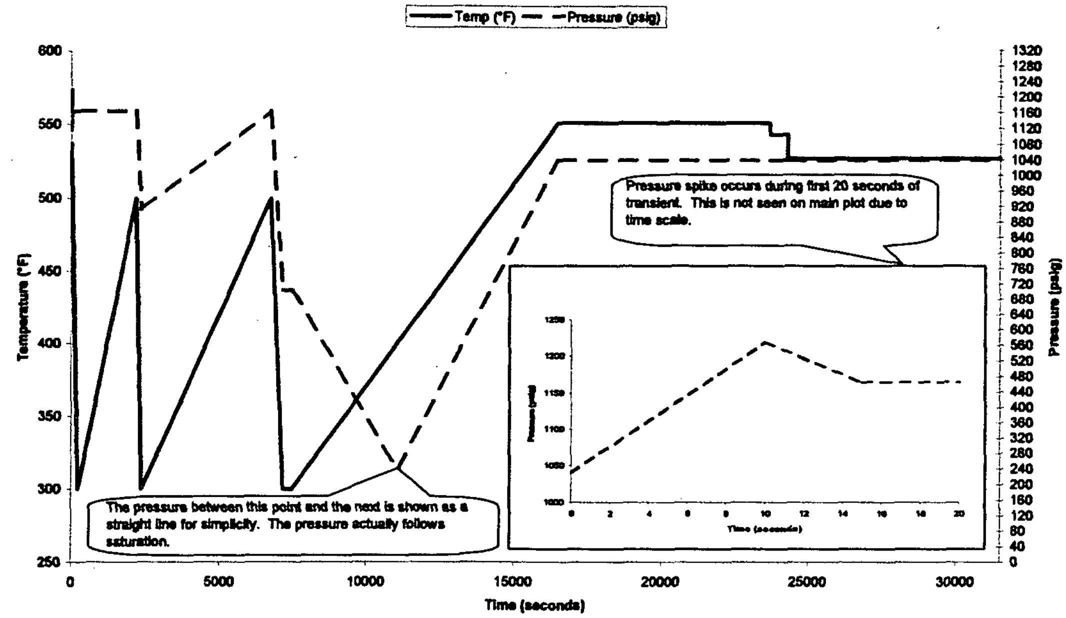

Figure C-4: Transient 11 for Recirculation Inlet Nozzle

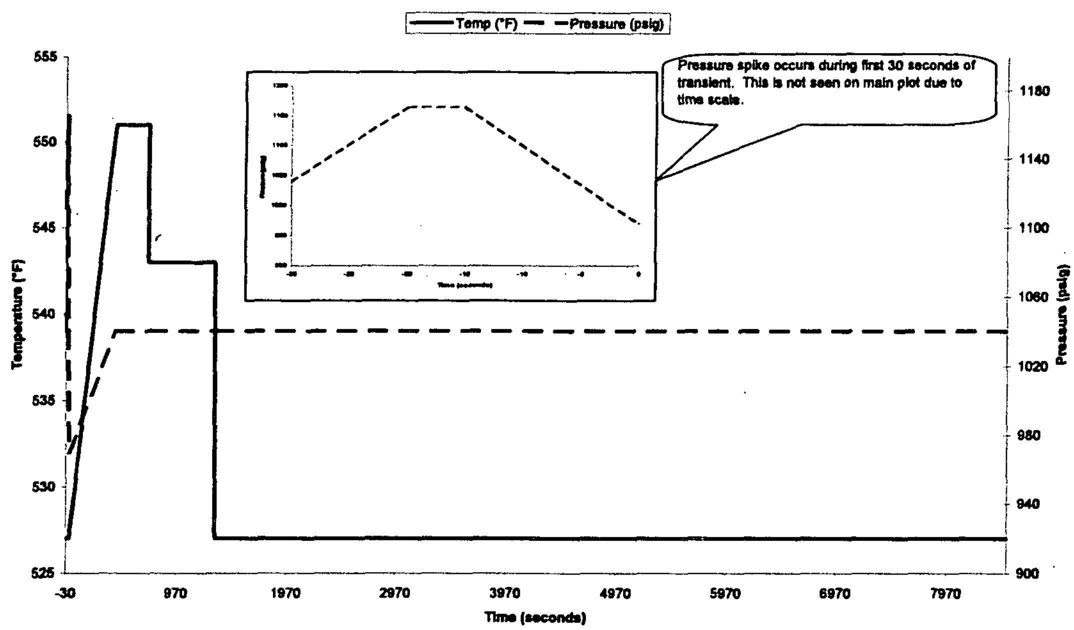

Figure C-5: Transient 12 for Recirculation Inlet Nozzle

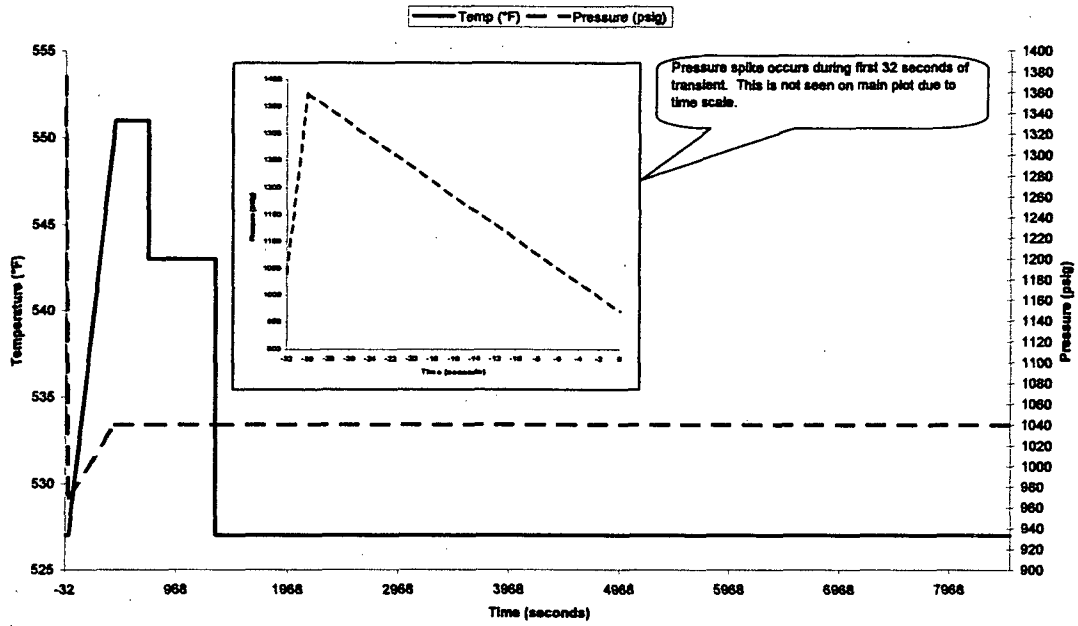

Figure C-6: Transient 13 for Recirculation Inlet Nozzle

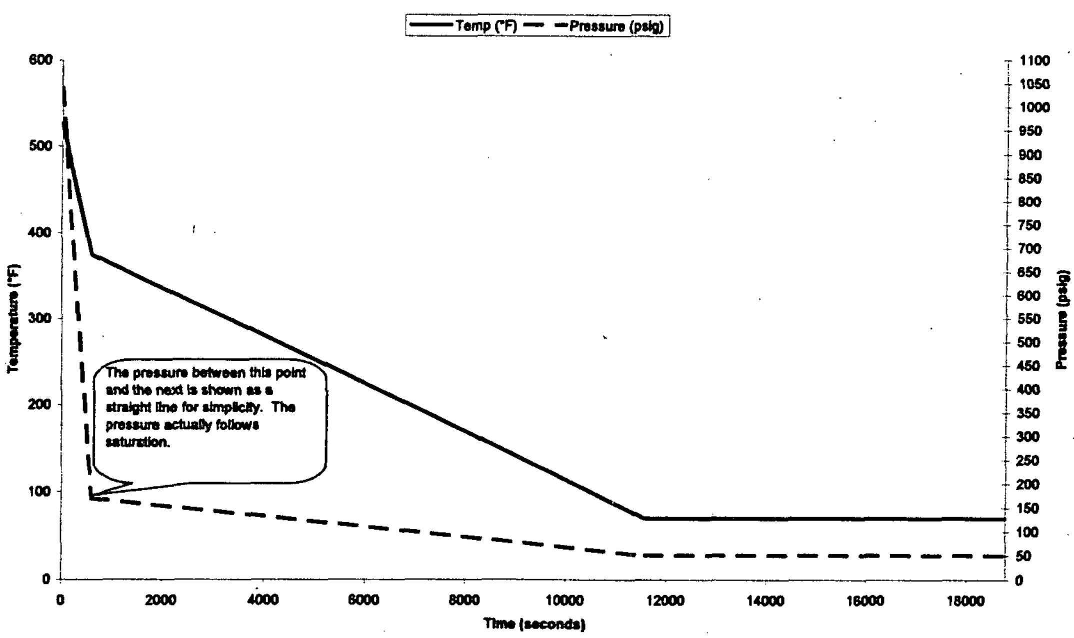

Figure C-7: Transient 14 for Recirculation Inlet Nozzle

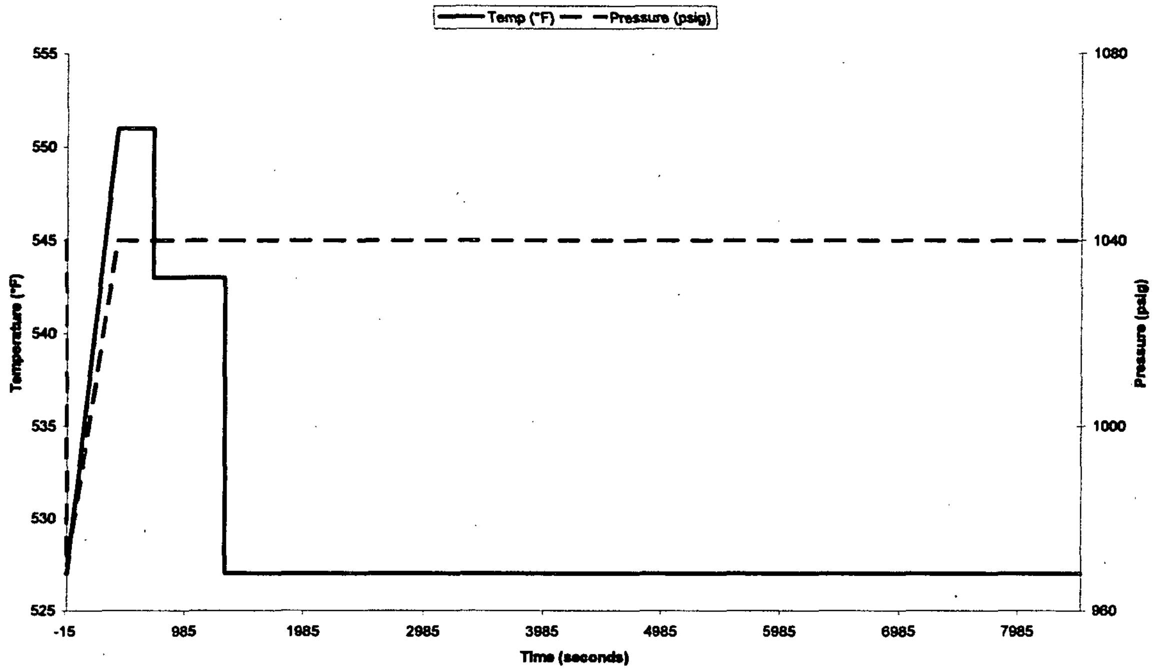

Figure C-8: Transient 15 for Recirculation Inlet Nozzle

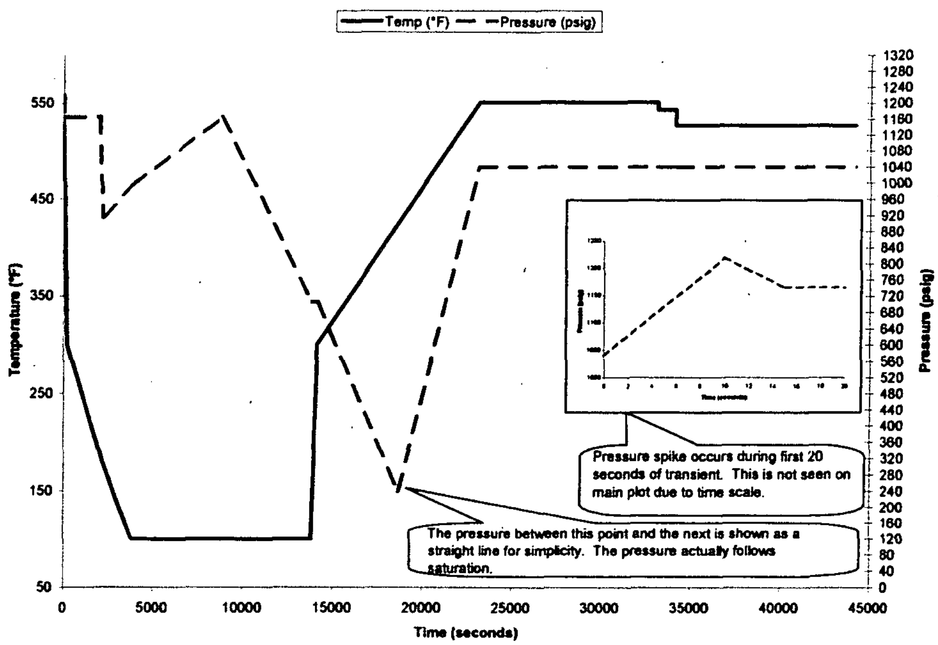

Figure C-9: Transient 11 for CRD Nozzle

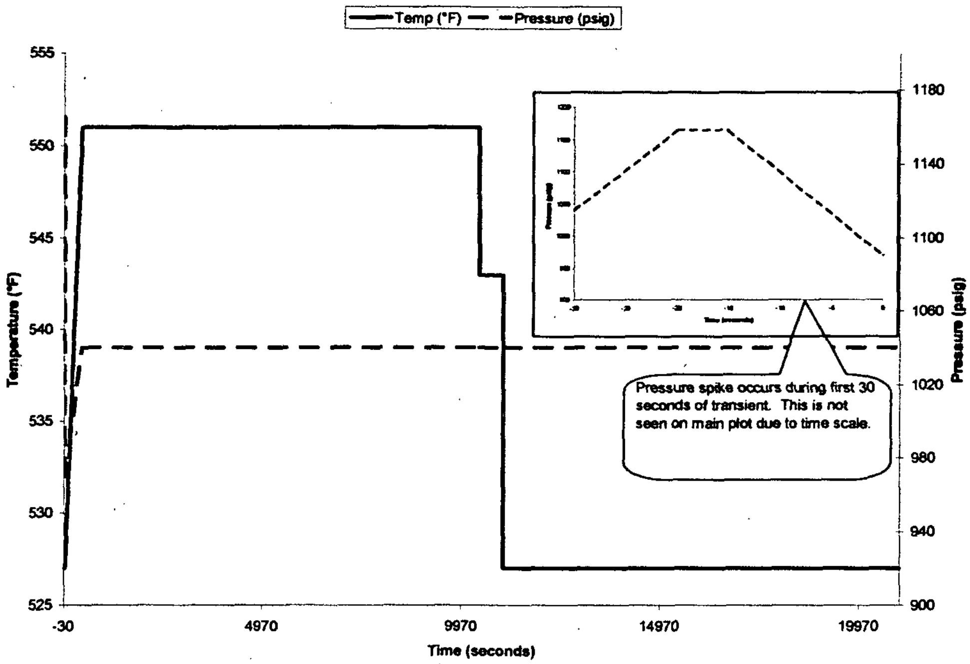

Figure C-10: Transient 12 for CRD Nozzle

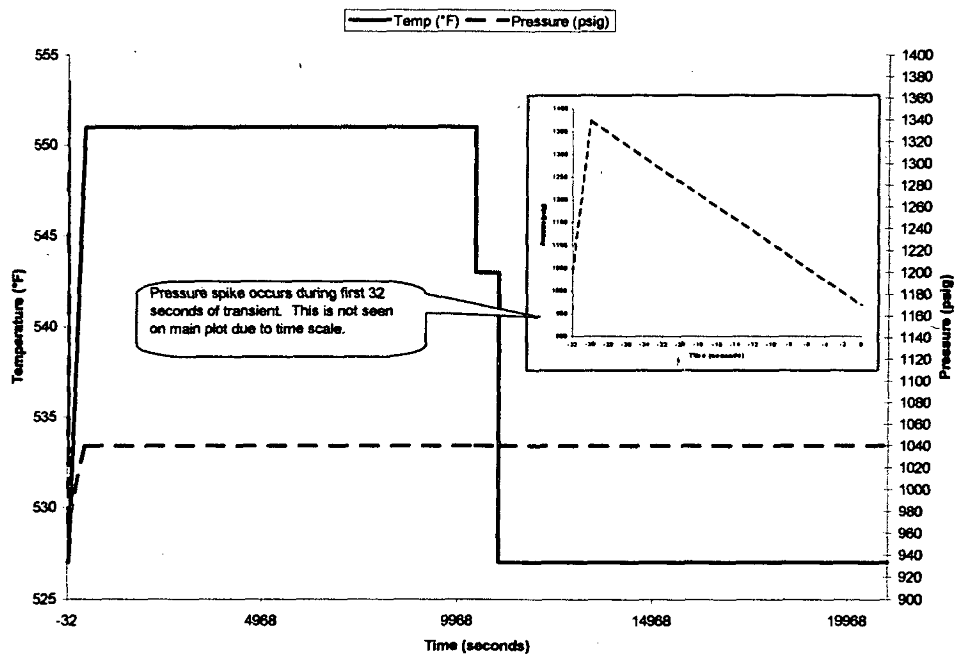

Figure C-11: Transient 13 for CRD Nozzle

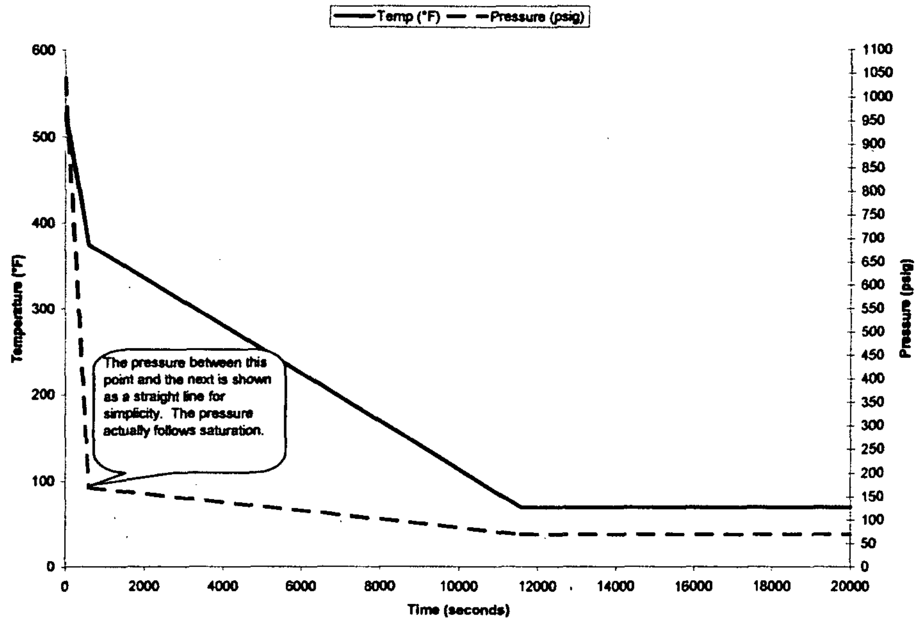

Figure C-12: Transient 14 for CRD Nozzle

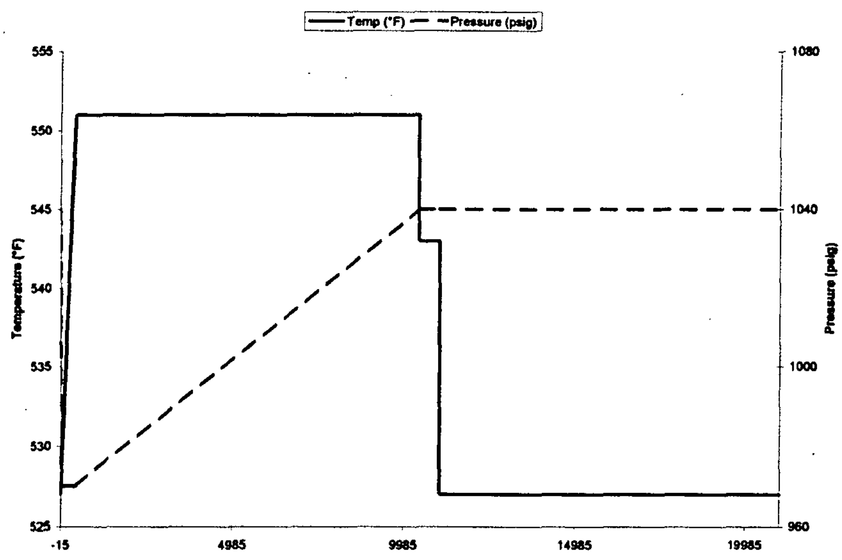

Figure C-13: Transient 15 for CRD Nozzle

RAI 4.3.1-1, Part C (iii)

(c). Justify how these 12 unidentified transients are characterized based on the analyzed P-T limit data used in SIA's updated cumulative usage factor calculations.

Response:

The unidentified transients were classified as Scram events (Event 15) by GE in the 1987 fatigue update. As discussed in the response to Part C (i) above, the previously unidentified events have been re-classified as shutdown and startup events based on a review of additional JAFNPP data. This is conservative because these events result in the maximum pressure and temperature range and the associated fatigue usage is higher than for the Scram event. Revised projections for all affected events are provided in Table C-2.

RAI 4.3.1-1, Part C (iii)

(iv). Clarify how your response to this part (Part C) factors into your response to Part A, particularly with respect to the recording the number of cycles for the transients defined in LRA Table 4.3-2 and using this data to project the 60 year cycles for the transients.

Response:

Due to the re-classification of events described in the responses in RAI 4.3.1-1, Part B (i) and Part C (i) above, the number of shutdown, startup and Scram transient events require modification from the values shown in JAFNPP LRA Table 4.3-2.

Note that the revised number of Scrams (Event # 15) has been changed because the SIA evaluation [10] accounted for the twelve unidentified events in the Year 3 number of Scram events, but not in the Year 26.5 number of Scram events. In addition, the re-classification of the six Scram events described in the response to Part B (i) further reduced the number of Scram events, and increased the number of startup and shutdown events.

The results of these changes are shown in Table C-2.

Table C-2: Updated Transient Event Projections for Startup, Scram & Shutdown Transients

Event Number	Design Transient	Revised Year 3 No. of Cycles	Revised Year 13 No. of Cycles	Revised Year 30.5 No. of Cycles	Revised Year 60 Projected No. of Cycles	Current Design Basis Analyzed No. of Cycles (4)
3	Startup (100°F/hr to 546°F at 100°F/hr)	30 (1)	68 (2)	132 (2)	242	233
	SCRAMs					
11	Loss of FW Pumps, MSIVs Close	0	4	5	11	12
12	Turbine Generator Trip, FW on, MSIVs stay open	2	9	10	12	12
13	Reactor Overpressure	0	0	0	0	1
14	Single Relief Valve Blowdown	1	1	1	1	2
15	All Other Scrams	20	48 (3)	51 (3)	57	64
19-23	**Shutdowns**	29 (1)	67 (2)	145 (2)	270	233

Notes:

1. The number of cycles for Event 3 has been revised from 17 and for Event 19-23 has been revised from 16, which was used in the reference [2] analysis to include the 13 additional events listed in Table C-1.
2. The number of cycles has been revised to include the 13 additional events listed in Table C-1, and the 6 additional events described in the response to RAI 4.3.1-1, Part B(i). The number of Year 13 cycles has been revised from 49 and for Event 19-23 has been revised from 48. The number of Year 30.5 cycles has been revised from 113 and for Event 19-23 has been revised from 126.
3. The number of Year 13 cycles has been revised from 54 and the number of Year 30.5 cycles has been revised from 57 to exclude the 6 events described in the response to RAI 4.3.1-1, Part B(i).
4. From Table A-1 of SIA analysis.

Attachment 2

James A. FitzPatrick Nuclear Power Plant

License Renewal Application – Amendment 12

Update Previous RAI Responses:
3.5.2-4
3.6.2-1
4.2.6-1
Table 3.3.2-13
LRA Revisions associated with Class 1 Fatigue

The following information revises the previously submitted response in Attachment 1 of LRA Amendment 10.

RAI 3.5.2-4 Revised Response

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.

Revised 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.

Vent Line to Torus penetration bellows consist of two sections of two-ply stainless steel bellows. Type B LLRT testing consists of pressurizing the space between the two plies of each bellows section, and measuring leakage as inlet flow to this space. This effectively tests all of the surface area of each bellows section.

The rest of the penetration assembly, including the vent insert in the Torus shell and mounting plates connecting the bellows to the vent piping and vent insert, is carbon or stainless steel of welded construction. Type A ILRT testing includes pressurizing the assembly from the Torus airspace, and measuring leakage as inlet flow to the Containment. This effectively tests all of the surface area of the assembly except the two two-ply bellows sections. Therefore, the combination of Type A and Type B testing effectively tests the entire assembly.

As noted in the response to NRC audit question 200 (provided in JAFP 07-0048, dated April 6,2007), there is no history at JAF of exposure of this material to corrosive contamination; neither is there any history of corrosion or other degradation of the assembly.

There is no history of leakage of the bellows assemblies under Type A or Type B testing. Exposed inner (i.e., Torus side) surfaces of the assemblies are viewed during Type B testing and during other Torus internal inspections. There is no convenient method for inspecting the unexposed portions of the assemblies, and no perceived need to do so in light of the available history.

The following information supplements the previously submitted response to RAI 3.6.2-1 in Attachment 2 of LRA Amendment 9.

Add the following to the Periodic Surveillance and Maintenance Program:

A power factor or partial discharge test will be performed in accordance with industry standards. The initial test will be completed prior to the period of extended operation. The frequency of the test will be adjusted based on the initial test results; the test frequency shall be at least once every ten years.

LRA Table 3.6.2-1 is revised as shown below (strike-outs deleted, underlined text added).

Component Type	Intended Function	Material	Environment	AERM	AMP	NUREG -1801, Vol.2 Item	Table 1 Item	Notes
Oil-filled cable system - MECH (passive mechanical for SBO recovery)	Pressure boundary	Carbon steel, stainless steel, copper alloy, glass	Mineral oil (internal) Outdoor weather (external)	Loss of material	Oil analysis External surfaces monitoring			J
Oil-filled cable – ELEC (passive electrical for SBO recovery)	Conducts electricity	~~Insulation material – various~~ Organic polymers	Outdoor Weather, Soil Voltage Stress	~~None~~ Breakdown of insulation leading to electrical failure	Periodic Surveillance and Preventive Maintenance			J, 602

Notes for Table 3.6.2-1

602 – Based on vendor information this transmission cable is not subject to water treeing, since it is designed for continuously wet conditions. Industry and plant operating experience has not provided any information on failures of this type of cable. The only portion of the cables exposed to the environments (outdoor weather and soil) is the okolene (black polyethylene) outer jacket, which is over the lead sheath and serves as an anti-corrosion and moisture protection. These environments do not affect the oil impregnated paper insulation. However, breakdown of insulation (reduced insulation resistance) leading to electrical failure will be managed by the Periodic Surveillance and Maintenance Program.

The following information supplements the LRA FSAR section A.2.2.1.6, Reactor Vessel Axial Weld Failure Probability, with information previously submitted in response to RAI 4.2.6-1 in Attachment 1 of LRA Amendment 6.

Add the following paragraph to LRA FSAR section A.2.2.1.6:

A.2.2.1.6 Reactor Vessel Axial Weld Failure Probability

The BWRVIP-74 SER states it is acceptable to show that the mean RT_{NDT} of the limiting beltline axial weld at the end of the period of extended operation is less than the value given in Table 1 of the BWRVIP-74 SER. This value supports the axial weld failure probability and is based on the assumption of essentially 100% (> 90%) inspection of the axial welds in the beltline region. Due to various obstructions within the reactor vessel, JAFNPP is able to inspect approximately 88% of the axial welds in the beltline region. The NRC granted a relief request for less than 90% coverage. The projected 54 EFPY mean RT_{NDT} value for JAFNPP is well below the limiting mean RT_{NDT} of 114 °F. The 2% difference in the amount of inspected weld will not offset the 16.8 °F margin between the 97.2 °F mean RT_{NDT} for JAFNPP and the 114 °F mean RT_{NDT} used in the NRC SER for BWRVIP-74. Therefore, the axial weld failure probability will not exceed 5 x 10-6 per reactor operating year during the period of extended operation. As such, this TLAA has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

The following information revises the previously submitted response in Attachment 2 of LRA Amendment 11 for LRA Table 3.3.2-13.

Security Generator System Changes

LRA Table 3.3.2-13, Security Generator System Summary of Aging Management Evaluation, is revised changing Note E to Note A as shown below. (strikeouts deleted, underlined text added)

Componen Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG -1801, Vol.2 Item	Table 1 Item	Notes
Sight glass	Pressure boundary	Carbon steel	Lube oil (int)	Loss of material	Oil analysis	VII.H2-20 (AP-30)	3.3.1-14	~~E~~ A

Based on the information provided in Attachment 1 for LRA section 4.3.1, Class 1 Fatigue, the following revisions to the LRA have been completed.

LRA Section 4 Changes

Table 4.1-1, List of JAFNPP TLAA and Resolution, is revised as shown below. (strikeouts deleted, underlined text added)

TLAA Description	Resolution Option	Section
Class 1 fatigue	~~Analysis remains valid~~ ~~10 CFR 54.21(c)(1)(i)~~ ~~OR~~ Analysis projected 10 CFR 54.21(c)(1)(ii) OR Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.1
Effects of reactor water environment on fatigue life	~~Analysis projected~~ ~~10 CFR 54.21(c)(1)(ii)~~ ~~OR~~ ~~Analyses will be projected~~ ~~OR~~ Aging effect managed 10 CFR 54.21(c)(1)(iii)	4.3.3

Section 4.3.1.1, Reactor Vessel, is revised as follows. (strikeouts deleted, underlined text added)

The reactor pressure vessel was designed in accordance with ASME Section III. Fatigue analyses were performed as required based on an allowed number of transient cycles.

An evaluation of fatigue usage factors was performed in 2002 accounting for sixty years of operation. This analysis projected that all components of the vessel would have fatigue usage factors below 1.0. Not all reactor vessel components have fatigue usage factors. Fatigue analyses were originally performed for limiting components of the vessel, as listed in Table 4.3-1. Fatigue usage factors for other vessel components not listed in Table 4.3-1 are bounded by the most limiting location. <u>The Fatigue Monitoring Program will assure that the analyzed numbers of transients used in fatigue calculations are not exceeded during the period of extended operation.</u>

Therefore, the <u>effects of aging associated with</u> TLAA (fatigue analyses) for reactor pressure vessel fatigue ~~remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i)~~ <u>are managed per 10 CFR 54.21(c)(1)(iii)</u>.

Section 4.3.1.2, Reactor Vessel Internals, is revised as follows. (strikeouts deleted, underlined text added)

A fatigue evaluation was also performed on the tie rod assemblies installed as part of the core shroud repair. The maximum CUF for the tie rod components is 0.0575 for the spring rod based on 120 startups/shutdowns. The ~~current~~ <u>projected</u> number of startups <u>and shutdowns is 242 and 270 respectively</u>~~/shutdowns~~ ~~allowed~~ for 60 years of operation ~~is 233~~. Therefore, a conservative projection of the fatigue usage of the tie rods for 60 years of operation would be (~~233~~<u>270</u>/120) x 0.0575, which equals a CUF of ~~0.110~~<u>0.13</u>.

Table 4.3-2, Projected Cycles, is revised to replace the 60 year cycle projections as follows.

Transient 3, Startup – Replace 216 with 242.
Transient 11, Loss of FW Pumps, MSIVs Close – Replace 10 with 11.
Transient 15, All Other Scrams, Replace 62 with 57.
Transients 19-23, Shutdowns, Replace 244 with 270.

Table 4.3-2, Projected Cycles, is revised adding the following footnote for transients 13, 17, and 18. (underlined text added)

<u>[2]Although zero events are anticipated, the analyzed number of design basis cycles remains unchanged.</u>

Section 4.3.3, Effects of Reactor Water Environment on Fatigue Life, is revised as follows (strikeouts deleted, underlined text added)

The effects of environmental-assisted thermal fatigue for the limiting locations identified in NUREG-6260 have been evaluated. ~~Depending on the option chosen, which may vary by component, this TLAA will be projected through the period of extended operation or the~~ The effects of environmentally assisted fatigue will be managed per 10 CFR 54.21(c)(1)(iii). ~~For those locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10CFR54.21(c)(1)(ii).~~

LRA Appendix A Changes

Section A.2.1.24, Periodic Surveillance and Preventive Maintenance, is revised to add the following item to the list of components inspected by the program. (underlined text added)

- Internal surfaces of carbon steel components in the floor and roof drainage system

Section A.2.2.2.1, Class 1 Metal Fatigue, is revised to add the following to the last sentence of this section. (strikeouts deleted, underlined text added)

> Consequently, the TLAA (fatigue analyses) based on those transients ~~will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i), or~~ are projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii)~~,~~ or the aging effect is managed per 10 CFR 54.21(c)(1)(iii).

Section A.2.2.2.3, Environmental Effects on Fatigue, is revised as follows. (strikeouts deleted, underlined text added)

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations identified in NUREG/CR-6260. ~~For the locations with CUFs less than 1.0, the TLAA has been projected through the period of extended operation per 10 CFR 54.21(c)(1)(ii).~~ Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. For these locations, prior to the period of extended operation, JAFNPP will (1) refine the fatigue analysis to lower the predicted CUF to less than 1.0; (2) manage fatigue at the affected locations with an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or (3) repair or replace the affected locations. ~~Depending on the option chosen, which may vary by component, this TLAA will be projected through the period of extended operation or the~~ The effects of environmentally assisted fatigue will be managed per 10 CFR 54.21(c)(1)(iii).

LRA Appendix B Changes

Section B.1.20, Oil Analysis, is revised to clarify the description of underground oil filled cables (strikeouts deleted, underlined text added)

Attributes Affected	Enhancements
1. Scope of Program	The Oil Analysis Program guidance documents will be enhanced to periodically sample lubricating oil in ~~the underground oil filled cable,~~ the security generator and the fire pump diesel, as well as the oil internal to underground oil filled cables.

Section B.1.27.2, Structures Monitoring, is revised to clarify the physical location of lubrite surfaces (strikeouts deleted, underlined text added)

Attributes Affected	Enhancements
4. Detection of Aging Effects	Guidance for performing periodic inspections to confirm the absence of aging effects for lubrite surfaces in the ~~torus~~ drywell radial beam seats will be added to the Structures Monitoring Program procedure.