DRAFT

Received 8/26/02

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

Re: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Response to NRC Request for Additional Information for Review of the
St. Lucie Units 1 and 2 License Renewal Application

By letters dated July 1, 2002 and July 18, 2002, the NRC requested additional information regarding the St. Lucie Units 1 and 2 License Renewal Application (LRA) Sections 2.0, 3.0, 4.0 and Appendix B. Attachment 1 to this letter contains FPL's response to the requests for additional information (RAIs) associated with the Time-Limited Aging Analyses (TLAAs) Section 4.0 of the LRA.

Should you have any further questions, please contact S. T. Hale at (772) 467-7430.

Very truly yours,

D. E. Jernigan
Vice President
St. Lucie Plant

DEJ/STH/hlo
Attachment (1)

Enclosure 8

St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389

Response to NRC Request for Additional Information Regarding the License Renewal Application, Section 4.0 - Time-Limited Aging Analyses.

STATE OF FLORIDA)
) ss
COUNTY OF ST. LUCIE)

D. E. Jernigan being first duly sworn, deposes and says:

That he is Vice President – St. Lucie of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

____
D. E. Jernigan

Subscribed and sworn to before me this

______ day of ________________, 2002.

____

____
Name of Notary Public (Type or Print)

D. E. Jernigan is personally known to me.

cc: U.S. Nuclear Regulatory Commission, Washington, D.C.
Chief, License Renewal and Standardization Branch
Project Manager – St. Lucie License Renewal
Project Manager - St. Lucie

U.S. Nuclear Regulatory Commission, Region II
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ST. LUCIE UNITS 1 AND 2
DOCKET NOS. 50-335 AND 50-389
ATTACHMENT 1
RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION
FOR REVIEW OF THE ST. LUCIE UNITS 1 AND 2
LICENSE RENEWAL APPLICATION

4.0 TIME-LIMITED AGING ANALYSES (TLAAs)

4.1 Identification Of TLAAs

RAI 4.1 - 1

Table 4.1-1 of the LRA does not identify pipe break postulation based on cumulative usage factor (CUF) as a TLAA. Section 3.6.2.2.1 of the St. Lucie Unit 2 UFSAR describes the criteria used to provide protection against pipe whip inside the containment. Part of the criteria specifies the postulation of pipe breaks at locations where the CUF exceeds 0.1. Although the fatigue usage factor calculation was identified as a TLAA, the pipe break criterion was not identified as a TLAA. However, the usage factor calculation used to identify postulated pipe break locations meets the definition of a TLAA as specified in 10 CFR 54.3 and, therefore, the staff considers the associated criteria for pipe break postulation a TLAA. Provide a description of the TLAA performed to address the pipe break criteria for St. Lucie Unit 2. Also identify any pipe break postulations based on CUF at St. Lucie Unit 1 and describe the TLAA performed for these locations. Indicate how these TLAAs meet the requirements of 10 CFR 54.21(c).

Table 4.1-1 of the LRA does not identify fatigue of the reactor coolant pump flywheel as a TLAA. Indicate whether fatigue crack growth calculations were performed for the St. Lucie Unit 1 and 2 reactor coolant pump flywheels. If fatigue crack growth calculations were performed for these pump flywheels, describe the TLAA evaluations and indicate how these TLAAs meet the requirements of 10 CFR 54.21(c).

FPL Response

As indicated in LRA Table 4.1-1 (page 4.1-5) and Subsection 4.3.1 (page 4.3-2), the fatigue analyses of ASME Boiler and Pressure Vessel Code (ASME), Section III, Class 1 components have been identified as a time-limited aging analyses (TLAA) as defined by 10 CFR 54.3. The Class 1 component fatigue analyses have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). In other words, the Class 1 component fatigue analysis results remain unchanged for the period of extended operation.

As described in Section 3.6.2.2.1 of the St. Lucie Unit 2 UFSAR, the current licensing basis (CLB) postulates failures in Class 1 piping at locations where the cumulative usage factor (CUF) obtained from the component fatigue analyses exceeds 0.1. No additional analyses beyond those performed for Class 1 component fatigue are required to determine these postulated pipe break locations. Accordingly, no additional TLAA evaluation has been performed, other than those associated with Class 1 component fatigue, to address pipe break criteria for St. Lucie Unit 2. In addition, since the current Class 1 component fatigue analyses remain valid for the period of extended operation, postulated pipe break locations also remain unchanged.

The CLB for St. Lucie Unit 1 does not explicitly use the CUF results from the Class 1 component fatigue analyses to determine postulated pipe break locations. As such, pipe break criteria for St. Lucie Unit 1 does not meet the definition of a TLAA as provided in 10 CFR 54.3.

Potential TLAAs associated with fatigue crack growth of reactor coolant pump (RCP) flywheels were specifically investigated during the license renewal process. The only CLB reference to RCP flywheel crack growth calculations was found in Section 5.5.5.3 of the St. Lucie Unit 1 UFSAR. As indicated in Unit 1 UFSAR Section 5.5.5.3, RCP flywheel crack growth calculations indicate that the number of starting cycles to cause a reasonably small crack to grow to critical size is more than 100,000. This represents in excess of 4.5 RCP starts per day over the 60-year license renewal period, which is orders of magnitude greater than the number of cycles expected during the life of the plant. As such, the crack growth calculation was determined not to be relevant in making a safety determination and did not meet the definition of a TLAA as defined in 10 CFR 54.3. An evaluation of RCP flywheel integrity for St. Lucie Unit 2 is provided in Section 5.4.1.4 of the Unit 2 UFSAR. The St. Lucie Unit 2 CLB (which includes this UFSAR section) did not identify or reference fatigue crack growth calculations for the RCP flywheels. Thus there are no TLAAs associated with the Unit 2 RCP flywheels.

4.3 Metal Fatigue

RAI 4.3 - 1

In Section 4.3.1 of the LRA, the applicant discusses its evaluation of the fatigue TLAA for ASME Class 1 components. The discussion indicates that based on its review of the plant's operating history, the applicant concluded that the number of cycles assumed in the design of the ASME Class 1 components is conservative and bounding for the period of extended operation. Section 3.9 of the UFSARs for St. Lucie, Units 1 and 2, provides a listing of transient design conditions and associated design cycles. Provide the following information for each transient described in the UFSARs:

(1) the current number of operating cycles and a description of the method used to determine the number and severity of the design transients from the plant's operating history

(2) the number of operating cycles estimated for 60 years of plant operation and a description of the method used to estimate the number of cycles at 60 years

(3) a comparison of the design transients listed in UFSAR with the transients monitored by the Fatigue Monitoring Program (FMP) as described in Section B3.2.7 of the LRA; an identification of any transients listed in the UFSAR that are not monitored by the FMP; and an explanation of why it is not necessary to monitor these transients

FPL Response

Item 1

Sections 3.9 and 5.2.1.2 of the St. Lucie Unit 1 UFSAR and Section 3.9 of the St. Lucie Unit 2 UFSAR contain a listing of the design transients used in the design of the various Reactor Coolant System (RCS) Class 1 components. These design transients have been consolidated into Tables 4.3-1.1 and 4.3-1.2 for St. Lucie Units 1 and 2, respectively. However, each of these design transients is not necessarily a significant contributor to the overall Class 1 component fatigue usage. As part of license renewal, a comprehensive review of each Reactor Coolant System Class 1 component fatigue analysis was performed to determine which design transients are a significant contributor to overall fatigue usage. A design transient was deemed to be significant if the transient contributed greater than 0.1 to the overall component cumulative usage factor (CUF).

FPL has implemented a Fatigue Monitoring Program (FMP) (LRA Appendix B Subsection 3.2.7 page B-37) at both St. Lucie Units 1 and 2 to fulfill plant Technical Specification requirements and to ensure that the significant "fatigue-sensitive" design transient counts are not exceeded during plant operation. A summary of the design transients included in the Fatigue Monitoring Program is provided in Tables 4.3-1.3 and 4.3-1.4 for St. Lucie Units 1 and 2, respectively. Note that some transients listed in these tables are not fatigue-sensitive, but they are included in the Fatigue Monitoring Program because of plant Technical Specification requirements. Also note that some fatigue-sensitive transients identified from the CUF screening process have been excluded from the Fatigue Monitoring Program due to large margins that are present with respect to actual cycle counts versus allowable cycle counts. For example, plant loading/unloading events are not monitored because the St. Lucie units are not load following plants, so these events rarely occur.

Cycle counting has been performed since the startup of each St. Lucie unit. This program counts the design transients identified in Tables 4.3-1.3 and 4.3-1.4 by recording the actual number and types of transients imposed on the RCS components, and ensures that the design transient limits are not exceeded. A comprehensive review of plant operating records was performed to validate that the transient counts included in the Fatigue Monitoring Program are accurate. This review concluded that the program accurately identifies and classifies plant design transients and provides an effective and consistent method for categorizing, counting, and tracking design transients. The current number of operating cycles (as of December 31, 2000) for each transient included in the Fatigue Monitoring Program is included in Tables 4.3-1.3 and 4.3-1.4.

As part of license renewal, design basis transient severities were compared to the actual transients experienced at St. Lucie Units 1 and 2. This review was performed to demonstrate that the original design transient assumptions are severe enough to bound all operating events. Typical plant design transients were reviewed as part of the evaluation. The results of the review concluded that the original design transient assumptions are severe enough to bound all operating events.

Item 2

The number of operating cycles estimated for 60 years of plant operation is also shown in Tables 4.3-1.3 and 4.3-1.4 for St. Lucie Units 1 and 2, respectively. Conservative linear cycle projections based on the plant startup date were used for all events, except where noted in the "Comments" column in each table, as follows:

$$N_{60} = [\ N_{2000} / (2000\text{-}Y_{startup})\ ] * (Y_{60} - Y_{startup})$$

where:	N60	=	projected number of events for 60 years
	N2000	=	number of events as of 12/31/2000
	Ystartup	=	year of plant startup
		=	1976 for St. Lucie Unit 1
		=	1982.67 for St. Lucie Unit 2
	Y60	=	60th year of plant operation
		=	2036 for St. Lucie Unit 1
		=	2043 for St. Lucie Unit 2

This projection method is conservative in that it includes "learning curve" effects of early plant operation, as opposed to trends established by the most recent years of plant operation. The results provided in Tables 4.3-1.3 and 4.3-1.4 indicate that all transient projections remain well within the number of occurrences assumed in the design analyses for all events.

Item 3

The design transients listed in Tables 4.3-1.1 and 4.3-1.2 are a compilation of all RCS Class 1 design transients included in the St. Lucie Unit 1 and 2 UFSARs. A comparison of these transients with those being monitored by the Fatigue Monitoring Program (Tables 4.3-1.3 and 4.3-1.4) indicates that some of the UFSAR transients are not monitored by the program.

An explanation of this difference is provided in the response to Item 1 above. As discussed, the Fatigue Monitoring Program only tracks those design transients that are a significant contributor to the overall component CUF. As such, it has been concluded that it is not necessary to track those design transients that are not a significant contributor to component fatigue.

Table 4.3-1.1
St. Lucie Unit 1 UFSAR Design Transients

Transient Description	Number of Cycles
Normal Conditions Transients:	
Plant Heatup	500
Plant Cooldown	500
Pressurizer Heatup	500
Pressurizer Cooldown	500
Plant Loading, 5%/min.	15,000
Plant Unloading, 5%/min.	15,000
10% Step Load Increase	2,000
10% Step Load Decrease	2,000
Normal Plant Variations, +/- 100 psi, +/- 6°F	10^6
Primary Coolant Pump Starting/Stopping	4,000
Purification	1,000
Low Volume Control and Makeup	2,000
Boric Acid Dilution	8,000
Cold Feed Following Hot Standby	15,000
Actuation of Main or Auxiliary Spray	500
Low Pressure Safety Injection, 40°F Water into 300°F Cold Leg	500
Opening of Safety Injection Return Line Valves	2,000
Initiation of Shutdown Cooling	500
Upset Condition Transients:	
Turbine Trip (Loss of Load)	40
Loss of Offsite Power (Loss of RCS Flow)	40
Reactor Trip	400
Inadvertent Auxiliary Spray Cycle	16
Loss of Charging Flow	200
Loss of Letdown Flow	50
Regenerative Heat Exchanger Isolation Long Term	80
Regenerative Heat Exchanger Isolation Short Term	40
Emergency Condition Transients:	
Loss of Secondary Pressure	5
Loss of Feedwater Flow	8
High Pressure Safety Injection, 40°F Water into 550°F Cold Leg	5
Test Condition Transients:	
Primary System Hydrostatic Test, 3125 psia	10
Primary System Leak Test, 2250 psia	200
Secondary System Hydrostatic Test, 1250 psia	10
Secondary System Leak Test, 1000 psia	200

Table 4.3-1.2
St. Lucie Unit 2 UFSAR Design Transients

Transient Description	Number of Cycles
Normal Condition Transients:	
Plant Heatup	500
Plant Cooldown	500
Plant Loading, 5%/min.	15,000
Plant Unloading, 5%/min.	15,000
10% Step Load Increase	2,000
10% Step Load Decrease	2,000
Normal Plant Variations, +/- 100 psi, +/- 6°F	10^6
Purification and Boron Dilution	24,000
Upset Condition Transients:	
Turbine Trip (Loss of Load)	40
Loss of Offsite Power (Loss of RCS Flow)	40
Reactor Trip	400
Operating Basis Earthquake	200
Loss of Charging Flow	20
Loss of Letdown Flow	50
Isolation Check Valve Leaks	40
Emergency Condition Transients:	
Loss of Secondary Pressure	5
Test Condition Transients:	
Primary System Hydrostatic Test, 3125 psia	10
Primary System Leak Test, 2250 psia	200

Table 4.3-1.3
St. Lucie Unit 1 Design Transients Included in Fatigue Monitoring Program

Transient	Design Cycles	Cycle Counts as of 12/31/00	60-Year Projection	Margin	Comments
Reactor Trip	400	46	115	71%	Fatigue-sensitive transient.
Plant Heatup	500	57	143	72%	Fatigue-sensitive transient.
Plant Cooldown	500	56	143	72%	Fatigue-sensitive transient.
Pressurizer Heatup	500	57	143	72%	Not a fatigue-sensitive transient, but included in FMP to be consistent with Unit 2.
Pressurizer Cooldown	500	56	143	72%	Not a fatigue-sensitive transient, but included in FMP to be consistent with Unit 2.
Primary Hydrostatic Test	10	1	3	75%	Fatigue-sensitive transient.
Secondary Hydrostatic Test	10	4	10	0%	Not a fatigue-sensitive transient, but included in Unit 1 Technical Specifications.
Primary Leak Test	200	45	113	44%	Not a fatigue-sensitive transient, but included in Unit 1 Technical Specifications.
Secondary Leak Test	200	1	3	99%	Not a fatigue-sensitive transient, but included in Unit 1 Technical Specifications.
Loss of Secondary Pressure	5	0	1	80%	Fatigue-sensitive transient. Assume 1 cycle occurs in 60-year life.
Pressurizer Spray	1,500	147	675	55%	Fatigue-sensitive transient (see Note 1).
Inadvertent Auxiliary Spray	16	3	8	53%	Not a fatigue-sensitive transient, but included in Unit 1 Technical Specifications.
Loss of Offsite Power (Loss of RCS Flow)	40	0	1	98%	Fatigue-sensitive transient. Assume 1 cycle occurs in 60-year life
Loss of Load	40	3	8	81%	Fatigue-sensitive transient.

Note: 1. Projection is based on recent cyclic trends versus linear projection. The number of cycles for this event was increased from the original number reported in the UFSAR based on additional plant-specific analysis of the pressurizer spray line.

Table 4.3-1.4
St. Lucie Unit 2 Design Transients Included in Fatigue Monitoring Program

Transient	Design Cycles	Cycle Counts as of 12/31/00	60-Year Projection	Margin	Comments
Reactor Trip	400	18	63	84%	Fatigue-sensitive transient. Assume 1 event/year since no additional events have occurred since 1996.
Plant Heatup	500	30	104	79%	Fatigue-sensitive transient.
Plant Cooldown	500	29	104	79%	Fatigue-sensitive transient.
Pressurizer Heatup	500	30	104	79%	Not a fatigue-sensitive transient, but included in Unit 2 Technical Specifications.
Pressurizer Cooldown	500	29	104	79%	Not a fatigue-sensitive transient, but included in Unit 2 Technical Specifications.
Primary Hydrostatic Test	10	1	3	65%	Not a fatigue-sensitive transient, but included in Unit 2 Technical Specifications.
Primary Leak Test	200	2	7	97%	Fatigue-sensitive transient.
Loss of Secondary Pressure	5	0	1	80%	Fatigue-sensitive transient. Assume 1 cycle occurs in 60-year life
Pressurizer Spray	1,500	108	509	66%	Fatigue-sensitive transient (see Note 1).
Loss of Offsite Power (Loss of RCS Flow)	40	0	1	98%	Not a fatigue-sensitive transient, but included in Unit 2 Technical Specifications. Assume 1 cycle occurs in 60-year life
Loss of Load	40	1	3	91%	Not a fatigue-sensitive transient, but included in Unit 2 Technical Specifications.

Note: 1. Projection is based on recent cyclic trends versus linear projection.

RAI 4.3 - 2

In Section 4.3.1 of the LRA, the applicant indicates that the pressurizer surge lines were reanalyzed in response to NRC Bulletin 88-11, "Pressurizer Surge Line Stratification." Identify whether calculations that meet the definition of a TLAA were performed in response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems." Describe the actions that will be taken to address NRC Bulletin 88-08 throughout the period of extended operation.

FPL Response

Review of St. Lucie Units 1 and 2 documentation and correspondence regarding NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems", identified no calculations that meet the definition of a TLAA as defined in 10 CFR 54.3. A review of piping systems for St. Lucie Units 1 and 2 in accordance with NRC Bulletin 88-08 determined that there are no unisolable sections of piping connected to the Reactor Coolant System that can be subjected to excessive thermal stresses from temperature stratification or temperature oscillations. As documented in a letter from NRC to FPL [Reference: J. A. Morris (NRC) letter to J. H. Goldberg (FPL), St. Lucie Units 1 and 2 – NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to the Reactor Coolant Systems" (TAC Nos. 69691 and 696920), September 16, 1991], FPL was advised that the requirements of NRC Bulletin 88-08 have been met and no further action is required. As such, there are no additional actions to be taken during the period of extended operation to address the considerations of NRC Bulletin 88-08.

RAI 4.3 - 3

In Section 4.3.3 of the LRA, the applicant discusses its evaluation of the impact of the reactor water environment on the fatigue life of components. The discussion references the fatigue-sensitive component locations for an older vintage Combustion Engineering plant identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The LRA indicates that these fatigue-sensitive component locations were evaluated for St. Lucie, Units 1 and 2. The LRA also indicates that the later environmental fatigue correlations contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," were considered in the evaluation. Provide the results of the usage factor evaluation for each of the six component locations listed in NUREG/CR-6260.

FPL Response

For St. Lucie Units 1 and 2, detailed environmental fatigue calculations were performed for each of the components identified in NUREG-6260 for the older vintage Combustion Engineering (CE) plant. The six fatigue-sensitive component locations chosen for the early-vintage CE pressurized water reactor (PWR) calculations were: (1) the reactor pressure vessel (RPV) shell and lower head, (2) the RPV inlet and outlet nozzles, (3) the pressurizer surge line elbow, (4) the reactor coolant system (RCS) piping charging system nozzle, (5) the RCS piping safety injection nozzle, and (6) the shutdown cooling system Class 1 piping. Counting the RPV inlet and outlet nozzles as separate locations, seven different component locations were evaluated for each unit.

The St. Lucie calculations were performed using the appropriate methodology contained in NUREG/CR-6583 for carbon/low alloy steel material, or NUREG/CR-5704 for stainless steel material, as appropriate. These calculations, along with the original design basis calculations, are summarized in Table 4.3-3.1. The environmental adjustments to the cumulative usage factor (CUF) results shown in Table 4.3-3.1 are considered to be very conservative, and are applicable for 60 years of plant operation.

Based on the results shown in Table 4.3-3.1, all candidate locations for environmental fatigue effects, except for the following locations, are acceptable for 60 years of operation (i.e., the cumulative usage factor is less than the allowable value of 1.0):

- St. Lucie Unit 1 safety injection nozzle
- St. Lucie Unit 1 pressurizer surge line
- St. Lucie Unit 2 pressurizer surge line

As shown in Table 4.3-3.1, the St. Lucie Unit 1 safety injection nozzle possesses a CUF value of 2.3 when environmental affects of fatigue (EAF) effects are considered. Further evaluation is expected to yield acceptable results due to the conservatism in the existing analysis. The most significant conservatism is the treatment of stress ranges resulting from a radial thermal gradient (treated as peak rather than secondary). A similar evaluation performed for the St. Lucie Unit 1 charging inlet nozzle resulted in a reduction in CUF of a factor of five. Similarly, refined evaluation that removes the conservatism for the safety injection nozzle will drop the CUF value to well below the design CUF value of 0.15, thereby demonstrating an acceptable CUF value when EAF effects are considered. Based on these considerations, the St. Lucie Unit 1 safety injection nozzle is considered to be acceptable for period of extended operation.

As shown in Table 4.3-3.1, the maximum CUF for the surge line elbow for both St. Lucie units was calculated to be above 1.0 when environmental effects were considered. Based on this and the refined nature of the existing evaluations, the surge lines for Units 1 and 2 are candidate components for additional inspection considerations during the license renewal period. A description of the aging management program proposed for the pressurizer surge lines is provided in LRA Subsection 4.3.3 (page 4.3-5).

Table 4.3-3.1
Summary of St. Lucie Environmental Fatigue Calculations

No.	Component	Design Cumulative Usage Factor	Environmental F_{en} Multiplier	Environmental Cumulative Usage Factor	Allowable Value
Unit 1					
1	Outlet Nozzle	0.0788	2.04	0.1607	1.0
2	Inlet Nozzle	0.0496	2.41	0.1198	1.0
3	Vessel Shell and Bottom Head	0.0031	1.77	0.0055	1.0
4	Charging Inlet Nozzle	0.1404	1.64	0.2297	1.0
5	Safety Injection Nozzle	0.1539	14.80	2.2787	1.0
6	Surge Line Elbow	0.9370	7.79	7.2998	1.0
7	Shutdown Cooling Piping	0.5612	1.65	0.9266	1.0
Unit 2					
1	Outlet Nozzle	0.3775	2.34	0.8825	1.0
2	Inlet Nozzle	0.2285	2.15	0.4909	1.0
3	Vessel Shell and Bottom Head	0.0017	2.37	0.0039	1.0
4	Charging Inlet Nozzle	0.0577	2.55	0.1468	1.0
5	Safety Injection Nozzle	0.0644	14.87	0.9569	1.0
6	Surge Line Elbow	0.9370	7.75	7.2603	1.0
7	Shutdown Cooling Piping	0.0485	15.35	0.7451	1.0

4.4 EQ Of Electrical Equipment

RAI 4.4 - 1

In Section 4.4 of the LRA, the applicant indicates that environmental qualification (EQ) acceptance criteria for temperature is the component's maximum required operating temperature. If the maximum operating temperature of a component for normal plant conditions is equal to or less then the temperature to which the component was qualified by test, the component is considered qualified. With a component's normal operating temperature equal to the temperature to which it was tested to demonstrate EQ, explain how temperature margin (or other conditions or attributes of the Arrhenius method) has been utilized to account for uncertainties of the Arrhenius method.

Explain how margin has been maintained to account for uncertainties of the Arrhenius method. Describe the margins built into the qualification process that will remain in the qualification process after re-analysis for 60 years. Explain why these remaining margins can be considered sufficient to address the uncertainties of the Arrhenius method for establishing qualified life.

FPL Response

The maximum operating temperatures referred to in the LRA are the 104°F design ambient for outside the Containments, and the 120°F design ambient (Unit 1) and 115°F design ambient (Unit 2) inside the Containments used to calculate the qualified life of Environmentally Qualified (EQ) components. LRA Section 4.4 (page 4.4-3) includes further details of Containment temperature monitoring including the location of temperature detectors in the Containments. Section 4.4 also indicates that EQ components are assumed to be exposed to continuous design ambient temperatures (104°F, 120°F, or 115°F, as appropriate), and that the evaluation does not credit lower temperatures due to seasonal/daily temperature changes or temperature changes associated with unit shutdown. These seasonal and shutdown reductions in temperature are more than adequate to account for the uncertainties of the Arrhenius Methodology when considering that the EQ components are assumed to be exposed to continuous design ambient temperature conditions. As an additional conservatism, continuous self-heating is also added to the design ambient temperatures.

For areas outside the Containments, LRA Section 4.4 (page 4.4-3) demonstrates that uncertainties of the Arrhenius Methodology are more than accounted for by the large difference between the mean ambient temperature of 72.5°F to 75°F and the assumed continuous exposure temperature of 104°F design ambient. Similar to the Containments, continuous self-heating is also added to the design ambient temperature of 104°F.

RAI 4.4 - 2

Explain and clarify how the electro-mechanical components of a normally energized continuous duty motor are maintained qualified for 40 years and 60 years of continuous operation.

FPL Response

The motors considered continuous duty in the Environmental Qualification Program (LRA Appendix B Subsection 3.2.6 page B-36) are the Units 1 and 2 containment fan cooler motors (LRA Subsections 4.4.1.47 page 4.4-54 and 4.4.1.36 page 4.4-43, respectively), the Units 1 and 2 charging pump motors (LRA Subsection 4.4.1.46 page 4.4-53), and certain Unit 2 ventilation fan motors (LRA Subsection 4.4.1.50 page 4.4-57). The qualification of the electro-mechanical components of these motors is maintained through a combination of maintenance required by the conditions in the test report (e.g., periodic replacement of seals because they were only aged for ten years prior to qualification testing), and maintenance recommended by the vendor (e.g., overhaul a motor after 25,000 hours of operation or every 5 years whichever comes first). The frequency of maintenance for these components are normally governed by the maintenance requirements of the vendor rather than by any restrictions that are required by the EQ test report.

4.5 Containment And Penetration Fatigue Analysis

RAI 4.5 - 1

In Section 4.5.1of the LRA, the applicant[s] states that the containment vessels are designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The LRA indicates that the design criteria provide assurance that the specified leak rate will not be exceeded under the design-basis accident conditions. Discuss how the design criteria applied to the steel vessels provide this assurance.

FPL Response

The containment vessels are designed in accordance with the applicable ASME Boiler and Pressure Vessel Code (ASME) Section III code requirements. The code requires that the containment vessels be designed to withstand the applicable design basis loading conditions (including normal operating and accident conditions). Specifically, the St. Lucie Unit 1 containment vessel is designed to meet the requirements of ASME Section III 1968, Article 4, Subsection N-415, titled “Analysis for Cyclic Operation.” Likewise, the Unit 2 containment vessel is designed to meet the requirements of ASME Section III 1971, Subsection NB-3222.4, “Analysis for Cyclic Operation.” By satisfying the subject code requirements, cracking due to fatigue (cyclic operation) is precluded by design. Compliance with the leakage design criteria is verified through periodic testing in accordance with the ASME Section XI, Subsection IWE Inservice Inspection Program as described in LRA Appendix B Subsection 3.2.2.2 (page B-26). Therefore, containment integrity is assured.

<u>RAI 4.5 - 2</u>

In Section 4.5.2 of the LRA, the applicant states that containment penetration bellows are specified to withstand a lifetime total of 7,000 cycles of expansion and compression attributed to maximum operating thermal expansion, and 200 cycles of other effects.

(1) Show that the specified cycles bound the period of extended operation.

(2) For Type I and Type III containment penetrations, describe the methods used to provide assurance that the penetration bellows will withstand these specified cycles under the corresponding thermal expansion and other loads for the extended period of operation.

<u>FPL Response</u>

<u>Item 1</u>

As described in LRA Subsection 4.5.2 (page 4.5-2), the piping systems associated with Type I and Type III penetration bellows have been evaluated in LRA Subsections 4.3.1 and 4.3.2 (pages 4.3-2 and 4.3-4 respectively), and found acceptable for the period of extended operation. LRA Subsection 4.3.1 describes the methods used to confirm that the existing design cycles for ASME Boiler and Pressure Vessel Code (ASME) Section III, Class 1 components are conservative and bounding for the period of extended operation. As indicated in the response to RAI 4.5-3, four St. Lucie Unit 1 containment penetrations are associated with Safety Injection piping designed to Class 1 requirements. Accordingly, the cycles that these piping components are subjected to are monitored as part of the Fatigue Monitoring Program. As indicated in the response to RAI 4.3.1, Table 4.3-1.3, the 7000 thermal expansion cycles assumed in the design of the containment penetration bellows bounds the total number of thermal cycles assumed for the Class 1 Safety Injection piping.

As indicated in the response to RAI 4.5-3, the remainder of the St. Lucie Unit 1 and 2 containment penetrations are associated with piping systems designed to ASME Section III, Class 2 requirements. As described in LRA Subsection 4.3.2 (page 4.3-4), these piping systems were originally designed for 7000 full temperature thermal cycles, which is consistent with the 7000 thermal cycles considered in the design of the containment penetration bellows. A rigorous evaluation of all piping systems associated with these containment penetrations, including a review of plant operating procedures and practices, concluded that these piping systems will not exceed 7000 equivalent full temperature thermal cycles during the period of extended operation.

The 200 cycles of "other effects" assumed in the design of all containment penetration bellows represents seismic and differential settlement cycles. A review of plant operation to date also concluded that these 200 cycles conservatively bounds the expected number of seismic and differential settlement cycles that could occur during the period of extended operation.

<u>Item 2</u>
As described above and in the response to RAI 4.5-3, the methods used to provide assurance that the penetration bellows will withstand these specified cycles under the corresponding thermal expansion and other loads for the extended period of operation include the Fatigue Monitoring Program and the inherent margin in the design of the containment penetration bellows as compared to actual plant operating experience.

RAI 4.5 - 3

State whether the containment penetration bellows are included within the scope of the St. Lucie Fatigue Monitoring Program, referred to in Sections 4.3.1 and B.3.2.7 of the LRA. If not, provide justification for not including these components in the program.

FPL Response

As described in LRA Appendix B, Subsection 3.2.7 (page B-37), the scope of the Fatigue Monitoring Program is associated with Reactor Coolant System (RCS) Class 1 components (reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and Class 1 RCS piping). Based on a review of the St. Lucie Units 1 and 2 RCS designs, the only containment penetrations associated with Class 1 (Quality Group A) piping are those associated with Unit 1 Safety Injection (reference containment penetrations 36, 37, 38, and 39 on License Renewal Boundary Drawing 1-SI-03). Accordingly, the Class 1 piping associated with these Unit 1 containment penetrations is included in the Fatigue Monitoring Program.

The remainder of the St. Lucie Unit 1 and 2 containment penetrations are associated with piping systems designed to ASME Section III, Class 2 requirements. As described in LRA Subsection 4.3.2 (page 4.3-4), these piping systems were originally designed for 7000 full temperature thermal cycles, which is consistent with the 7000 thermal cycles considered in the design of the containment penetration bellows. As concluded in LRA Subsection 4.3.2, these piping systems will not exceed 7000 equivalent full temperature thermal cycles during the period of extended operation. No confirmatory program is required for monitoring thermal cycles of ASME Section III, Class 2 components.

Note that the containment penetrations associated with the Unit 1 and 2 Reactor Coolant System hot leg sample lines are classified as Type II penetrations. As indicated in Subsection 4.5.2 of the LRA (page 4.5-2), these penetrations are not required to accommodate thermal movements and as such, monitoring of thermal cycles is not required.

4.6.1 Leak-Before-Break for Reactor Coolant System Piping

RAI 4.6.1 - 1

As a result of the V.C. Summer event, in which primary water stress corrosion cracking (PWSCC) was identified in an Inconel 82/182 main coolant loop-to-reactor pressure vessel weld, the NRC staff is concerned about the impact of PWSCC on licensees' leak-before-break (LBB) evaluations. NUREG-1061, Volume 3, which addresses the general methodology accepted by the NRC staff for demonstrating LBB behavior, stipulates that no active degradation mechanism (more specifically, none which would undermine the assumptions made elsewhere in the LBB analysis) may be present in a line that is under consideration for LBB approval. Draft Standard Review Plan Section 3.6.3, suggests that lines with potentially active degradation mechanisms may be considered for LBB approval provided that two mitigating actions or programs are in place to address the potential active degradation mechanism. Given this background:

- Identify the welds in the reactor coolant pressure boundary piping approved for LBB, which contain Inconel 82/182 material that is exposed to the reactor coolant system environment.
- Evaluate the impact of the V.C. Summer PWSCC issue on the St. Lucie LBB assessment for lines that contain welds manufactured from Inconel 82/182 material.
- Identify what actions will be taken during the period of extended operation to ensure that the potential for PWSCC in Inconel 82/182 lines does not undermine the assumptions of the St. Lucie LBB analyses.

FPL Response

First Bullet

The Leak-Before-Break (LBB) analysis for St. Lucie Units 1 and 2 was performed by Combustion Engineering (LRA Reference 4.6-1). The scope of the LBB analysis was limited to the Reactor Coolant System (RCS) primary loop piping. For St. Lucie Units 1 and 2, the only bimetallic weld joints containing Inconel 82/182 material are located at the transition from the carbon steel primary loop piping to cast stainless steel safe ends. These cast stainless steel safe ends are provided for field welding of the piping to the suction and discharge of the four (4) stainless steel reactor coolant pumps on each unit. Accordingly, there are a total of eight (8) Inconel 82/182 welds per unit exposed to the RCS environment that are included in the scope of the LBB analysis for St. Lucie Units 1 and 2.

Second Bullet

There are significant differences between St. Lucie Units 1 and 2 and V. C. Summer with respect to the potential for primary water stress corrosion cracking (PWSCC) of primary coolant loop Inconel 82/182 weld material. The V. C. Summer event involved a through-wall crack in the "A" hot leg pipe to reactor vessel dissimilar metal weld. This particular weld presented problems during construction. The root pass and approximately 30% of the initial weld passes exhibited defects, and a decision was made to bridge the imperfect weld, grind it out, and reweld the joint from the inside of the pipe. The investigation of the V. C. Summer event identified two root causes. First, extensive repairs during completion of the original "A" hot leg nozzle-to-pipe weld (weld repairs and grinding performed during construction) were the only source available to provide the high stresses required to produce PWSCC. Second, the applicable welding codes, standards, and welding processes utilized at V. C. Summer did not recognize or require consideration of the high residual stresses caused by multiple weld repairs and the associated grinding (Reference: NRC Letter to Mr. Stephen A. Byrne, South Carolina Electric & Gas Co., "Virgil C. Summer Nuclear Station-NRC Special Inspection Report No. 50-395/00-08, Exercise of Enforcement Discretion," March 15, 2001).

There are a number of unique features of the St. Lucie bimetallic weld joints that distinguish them from the V. C. Summer "A" hot leg pipe to reactor vessel dissimilar metal weld. These unique features include the following:

1. All of the St. Lucie bimetallic weld joints are shop welds.
2. All of the Inconel 82/182 weld buttering was applied to the carbon steel piping prior to post-weld heat treatment (PWHT) of the piping.
3. The weld between the safe end and the buttered pipe was performed in the shop after final PWHT.
4. Full non-destructive examination (NDE) of the buttering and weld joint was performed in the shop.
5. St. Lucie primary coolant piping field welds join stainless to stainless (P8 to P8) or carbon steel to carbon steel (P1 to P1) materials. There are no bimetallic weld joints designated as field welds in the primary coolant piping for either unit.

These fabrication differences significantly reduce the residual stresses within the St. Lucie Inconel 82/182 weld material as compared to the welds at issue at V. C. Summer and reduce the susceptibility of the St. Lucie Inconel 82/182 welds to PWSCC.

Another important consideration is the temperature that the Inconel 82/182 weld material is exposed to during plant operation. PWSCC is temperature sensitive; the higher the temperature, the more susceptible the material is to PWSCC. The V. C. Summer hot leg piping is normally exposed to temperatures of 622°F (Reference: Figure 5.3-1, Virgil C. Summer Nuclear Station UFSAR, Amendment 96-02, July 1996). This is significantly higher than the St. Lucie Units 1 and 2 cold leg temperatures, which are approximately 550°F. Accordingly, the temperature at which the St. Lucie Inconel 82/182 primary loop piping weld material is exposed to is approximately 72°F lower than that of V. C. Summer. This difference further reduces the susceptibility of the St. Lucie Inconel 82/182 welds to PWSCC as compared to V. C. Summer.

To date, a total of 39 non-destructive examinations (surface and volumetric) have performed at St. Lucie Units 1 and 2 on the primary loop piping Inconel 82/182 welds. All welds have been examined at least twice. Examinations were performed in accordance with the applicable edition of the ASME Code and no unacceptable flaws have been detected.

In addition, industry studies performed subsequent to the V. C. Summer event demonstrated that there is a large tolerance for axially oriented flaws in Alloy 82/182 weld material. This results from the fact that axial cracks in Alloy 82/182 welds will arrest when they reach the carbon steel or stainless steel piping materials. This maximum postulated crack length is much less than the critical axial crack length. In addition, calculations demonstrate that there is a large tolerance for circumferential cracks that propagate through-wall over a relatively short arc length.

In summary, the differences in fabrication and operating temperature between the V. C. Summer "A" hot leg pipe to reactor vessel dissimilar metal weld and the St. Lucie primary coolant loop Inconel 82/182 weld material, in addition to St. Lucie NDE results to date and industry studies, provide reasonable assurance that the V.C. Summer event has no impact on the St. Lucie Units 1 and 2 primary loop piping LBB analysis.

Third Bullet

As indicated in LRA Table 3.1-1 (page 3.1-42), the Alloy 600 Inspection Program, ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program and the Chemistry Control Program (LRA Appendix B Subsections 3.2.1, 3.2.2.1, and 3.2.5 pages B-22, B-25, and B-32, respectively) provide reasonable assurance that PWSCC is managed and that the intended function of the Inconel 82/182 weld material is maintained consistent with the St. Lucie Units 1 and 2 CLBs for the period of extended operation. Based on the information provided above, no additional actions are required at this time to ensure that the potential for PWSCC in Inconel 82/182 weld material does not undermine the assumptions of the St. Lucie LBB analyses.

4.6.3 Unit 1 Core Support Barrel Repairs

RAI 4.6.3 - 1

Provide a detailed description of the fatigue analysis of the core support barrel middle cylinder with the expandable plugs, including the design thermal transients and cycles. Confirm that the fatigue evaluation meets the ASME Section III Class 1 fatigue criteria for the life of the plant.

FPL Response

As described in LRA Subsection 4.6.3 (page 4.6-3), the St. Lucie Unit 1 reactor vessel internals core support barrel (CSB) middle cylinder fatigue analysis was identified as a TLAA in accordance with the definition provided in 10 CFR 54.3. The CSB middle cylinder fatigue analysis was revised to confirm that the repaired CSB meets all the applicable design requirements for an increase in plant operating life of 60 years.

The fatigue methodology developed for the CSB repairs performed in 1983 was followed for license renewal. This fatigue methodology employs a conservative method for combining component stresses to obtain stress intensities for the various cyclical loading conditions and conservatively applies the same stress concentration factor to all of the stress combinations. Reactor vessel internals design limits are specified in Section 4.2.2.1.2 of the St. Lucie Unit 1 UFSAR. Accordingly, the allowable stress values for core support structures are not greater than those given in the May 1972 draft of Section III of the ASME Boiler and Pressure Vessel Code, Subsection NG, including Appendix F, “Rules for Evaluation of Faulted Conditions”.

Plant design transients and cycles utilized in the fatigue analysis are defined in Section 5.2.1.2 of the St. Lucie Unit 1 UFSAR. These design transients are specifically intended for use in the fatigue analysis of Reactor Coolant System Class 1 components, but are also considered to be applicable to reactor vessel internals components. In the fatigue evaluation of the CSB middle cylinder, the full 40-year design transient set has been conservatively applied. No reduction in design cycles was credited for those cycles that occurred prior to the CSB damage in 1983. As discussed in Subsection 4.3.1 of the LRA, the 40-year design cycles bound the extended period of operation. The CSB middle cylinder fatigue analysis results in a limiting cumulative usage factor of 0.58, which is below the allowable value of 1.0.

The CSB middle cylinder fatigue analysis is available for review at the St. Lucie site.

RAI 4.6.3 - 2

Provide the source and basis for the data and information that was used to assess irradiation induced relaxation of the plug preload, which is expected to occur in the core support barrel expandable plugs at the end of 60 years of reactor operation.

FPL Response

As described in LRA Subsection 4.6.3 (page 4.6-3), the acceptance criteria for the St. Lucie Unit 1 reactor vessel internals CSB expandable plugs preload based on irradiation induced stress relaxation was identified as a TLAA in accordance with the definition provided in 10 CFR 54.3. The CSB plug preload analysis was revised for increased, 60-year end-of-life (EOL), fluence as an irradiation-induced relaxation input.

The CSB repair plugs were installed at the end of Cycle 5 (EO5) as part of the overall St. Lucie Unit 1 CSB repair effort. This effort was undertaken to repair damage incurred following a failure of the thermal shield support system and subsequent removal of the thermal shield assembly. CSB damage consisted of through-wall cracks and thermal shield support lug tear-out areas. The through-wall cracks were arrested with crack arrestor holes, and the tear-out areas were sealed with patches. The function of the repair plugs is to seal the crack arrestor holes and attach the repair patches to the CSB. The repair plugs are of an expandable design that allows the plugs to be preloaded against the CSB. This preload is required to provide proper seating of the plugs/patches, and to prevent movement of the plugs due to hydraulic drag loads. Verification of the plug design was originally performed in 1984 and included an evaluation of plug design preload. The evaluation of plug design preload verified that the design preload was sufficient to accommodate normal operating hydraulic loads and thermal deflections for the for the original 40-year operating life of the plant.

The design of the plugs allows for the preload to be quantified by measuring deflection of the plug flange, which acts against the outside diameter of the CSB. Plug flange deflection was measured following installation at the end of Cycle 5 (EO5), and again at the end of Cycle 6 (EO6). These as-measured deflections were evaluated against minimum deflection requirements, and were determined to be acceptable. Minimum deflection requirements account for the applied hydraulic drag forces, relative thermal expansion effects, and irradiation-induced relaxation of preload over the operating life of the plugs. As part of the 1997 St. Lucie Unit 1 steam generator replacement effort, the reactor coolant flow rate was increased which increased the hydraulic drag forces on the plugs. The impact of these increased hydraulic drag loads on plug design was evaluated in 1997.

In support of license renewal, the preload analysis was revised to re-calculate the minimum plug flange deflection requirements using revised fluence and irradiation-induced relaxation input. As-measured deflections are then evaluated against these revised minimum requirements. Changes to the original methodology were made to eliminate excess conservatism. For example, the revised CSB fluence input assumed for license renewal is more detailed permitting a more accurate calculation of expected plug fluences. In addition, the fuel management schemes in use since the original CSB repair reduce temperatures and temperature gradients in the CSB relative to the originally assumed fuel management scheme. Relative thermal expansions between the repair plugs and the CSB utilized in the original analysis are therefore bounding for license renewal.

In accordance with the original evaluation of plug flange deflection measurements, actual measured plug flange deflection must be greater than or equal to the minimum required values. Satisfaction of this criterion demonstrates that the plugs have sufficient preload to perform their intended function over the 60-year operating life of the plant. In all cases, actual plug flange deflection measurements exceed the minimum required values. The revised analysis concludes that the CSB repair plugs will perform their intended function for the extended period of operation.

It should be emphasized that even if plug preload were completely lost, the plug would still be captured in the CSB by the plug flange. During plant operation, the differential pressure across the CSB would load the plug flange against the outside diameter of the CSB, and the plug would still perform its function of limiting reactor coolant bypass flow. An assessment of plug vibration under this condition concluded that excessive plug/CSB wear would not occur.

The CSB plug preload analysis is available for review at the St. Lucie site.

RAI 4.6.3 - 3

Provide a detailed description of the core support barrel plug preload analysis based on irradiation induced stress relaxation, showing that the expandable plugs will continue to perform their function given the predicted fluence, operating temperature, operating hydraulic loads, and thermal deflections for the period of extended operation.

FPL Response

See response to RAI 4.6.3-2.

4.6.4 Alloy 600 Instrument Nozzle Repairs

RAI 4.6.4 - 1

Consistent with the staff's safety evaluation dated February 8, 2002, on Combustion Engineering Owners Group (CEOG) Topical Report No. CE NPSD-1198-P, Revision 00, perform a plant-specific general corrosion rate analysis calculation for the bounding half-nozzle repair implemented at St. Lucie Units 1 and 2. Provide a discussion or evidence which demonstrates that the general corrosion rate analysis calculation provided in CEOG Topical Report No. CE NPSE-1198-P, Revision 00, is bounding relative to the plant-specific analysis.

FPL Response

Pending receipt of input from Westinghouse, FPL will provide the draft response prior to September 20, 2002.

RAI 4.6.4 - 2

Consistent with the staff's safety evaluation dated February 8, 2002, on CEOG Topical Report No. CE NPSD-1198-P, Revision 00, justify the conclusion in the topical report that existing flaws in ASME Class 1 nozzle Alloy 182 weldments will not grow into the adjacent ferritic pipes or vessels during the extended periods of operation. Review the reactor coolant system chemistry history over the last two operating cycles for the St. Lucie Units 1 and 2. Confirm that a sufficient hydrogen over-pressure for the reactor coolant system has been implemented at the facilities and that the ingress of dissolved elemental oxygen, halide, and sulfate into the reactor coolant over this period was adequately managed and controlled (i.e., minimized to acceptable levels).

FPL Response

Pending receipt of input from Westinghouse, FPL will provide the draft response prior to September 30, 2002.