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SUBJECT: Application for amend to license DPR-67,extending applicability of current RCS pressure/temp limits & maximum allowed RCS heatup & cooldown rates to 23.6 Effective Full Power YRs of operation.

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Florida Power & Light Company, P.O. Box 128, Fort Pierce, FL 34954-0128

FPL

May 17, 1995

L-95-139
10 CFR 50.90

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

Re: St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
RCS Pressure/Temperature Limits

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating License DPR-67 for St. Lucie Unit 1 by incorporating the attached Technical Specifications (TS) revisions. The amendment will extend the applicability of the current Reactor Coolant System (RCS) Pressure/Temperature Limits and maximum allowed RCS heatup and cooldown rates to 23.6 Effective Full Power Years of operation. In addition, administrative changes are proposed for TS 3.1.2.1 (Boration Systems Flow Paths-Shutdown) and TS 3.1.2.3 (Charging Pump-Shutdown) to clarify the conditions for which a High Pressure Safety Injection pump may be used.

It is requested that the proposed amendment, if approved, be issued by March 31, 1996.

Attachment 1 is an evaluation of the proposed TS changes. Attachment 2 is a table of St. Lucie Unit 1 Reactor Vessel Beltline Adjusted Reference Temperatures for Determining Pressure Temperature and LTOP Limits Curves. Attachment 3 is the "Determination of No Significant Hazards Consideration." Attachment 4 contains a copy of the appropriate TS pages marked-up to show the proposed changes.

The proposed amendment has been reviewed by the St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board. In accordance with 10 CFR 50.91 (b)(1), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

Please contact us if there are any questions about this submittal.

Very truly yours,

D A Sager

D. A. Sager
Vice President
St. Lucie Plant

230032

A001

DAS/RLD

Attachments

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC.

Senior Resident Inspector, USNRC, St. Lucie Plant.

Mr. W.A. Passetti, Florida Department of Health and Rehabilitative Services.

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

D. A. Sager being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant for the Nuclear Division of Florida Power & 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 A Sager

D. A. Sager

STATE OF FLORIDA

COUNTY OF St. Lucie

The foregoing instrument was acknowledged before

me this 17th day of May, 1995

by D.A. Sager, who is personally known to me and who did take an oath.

Alexis R Molenda

Alexis R. Molenda
Name of Notary Public

My Commission expires 10-29-95

Commission No. CC156175

NOTARY PUBLIC STATE OF FLORIDA
OFFICIAL NOTARY SEAL
ALEXIS R MOLENDA
COMMISSION NUMBER
CC 156175
MY COMMISSION EXP.
OCT. 29,1995

St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
RCS Pressure/Temperature Limits

ATTACHMENT 1

EVALUATION OF PROPOSED TS CHANGES

EVALUATION OF PROPOSED TS CHANGES

Introduction

Florida Power and Light Company (FPL) proposes to change the St. Lucie Unit 1 Technical Specifications (TS) for Reactor Coolant System (RCS) Pressure/Temperature Limits and maximum allowed RCS heatup and cooldown rates. The applicability of the existing specified limits will be extended from 15 Effective Full Power Years (EFPY) to 23.6 EFPY of operation. The associated Low Temperature Overpressure Protection (LTOP) requirements will likewise be extended. In addition, administrative changes are proposed for TS 3.1.2.1 and TS 3.1.2.3 to clarify the conditions required if a High Pressure Safety Injection (HPSI) pump is used to establish a boration flow path during operational Modes 5 and 6.

Description of Changes

Marked-up copies of the affected TS pages are contained in Attachment 4. A description of the proposed changes follows:

1. TS 3.4.9.1 currently provides the pressure and temperature limits in terms of Figures 3.4-2a, 3.4-2b, and 3.4-3 for the RCS (except the pressurizer) during heatup, cooldown, criticality, and inservice leak and hydrostatic testing for 15 EFPY. The proposed amendment would use these existing figures, as-is, and only revise the title by changing "15 EFPY" to "23.6 EFPY".

2. Footnote "*" is appended to TS 3.1.2.1 and TS 3.1.2.3, applicable during Modes 5 and 6, to provide requirements that must be met if the boration flow path from the Refueling Water Tank (RWT) is established via a single HPSI pump. This footnote will be revised to more clearly indicate that the specified requirements apply to the condition when no charging pumps are operable.

3. Figure 3.1-1b specifies the maximum allowable heatup and cooldown rates for the case where a single HPSI pump is in operation pursuant to TS 3.1.2.1 and TS 3.1.2.3. The limit of applicability for this figure will be indicated by adding "23.6 EFPY" to the title.

Background

The reactor coolant pressure boundary is designed pursuant to 10 CFR Part 50, Appendix A, Design Criteria 14 and 31. These design criteria require that the reactor coolant pressure boundary be designed, fabricated, erected and tested in order to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The criteria also require that the reactor coolant pressure boundary be designed with sufficient margin to assure that, when stressed, the boundary behaves in a non-brittle manner, and that the probability of rapidly propagating fracture is minimized.

Pressure/Temperature (P/T) limits are developed to provide adequate safety margins for the prevention of fracture of the RCS coolant pressure boundary pursuant to 10 CFR 50, Appendix G, "Fracture Toughness Requirements." The ASME Code forms the basis for these requirements. The method to predict the reactor vessel material irradiation damage is provided in USNRC Regulatory Guide (RG) 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials".

The existing P/T limits were implemented in 1990, and are based on predictions for up to 15 EFPY of reactor operation. It is anticipated that PSL1 will achieve 15 EFPY of operation during the spring of 1996. FPL recalculated the predicted material radiation damage based on recent data obtained in response to NRC Generic Letter 92-01, "Reactor Vessel Structural Integrity." Using this new data and RG 1.99, Rev.2, it was determined that the existing material radiation damage prediction can be extended to 23.6 EFPY of operation.

Additional assurance that RCS P/T limits will not be exceeded in the lower temperature operating modes is provided by the Low Temperature Overpressure Protection (LTOP) system. Analyses for this system consider the initiation of assumed energy-addition and mass-addition transients while operating at low temperatures in accordance with Standard Review Plan 5.2.2, Revision 2.

Bases for Proposed TS Changes

The analysis for the P/T limits and LTOP requirements for 15 EFPY were provided with FPL letter L-89-408 (Reference 3), and subsequently approved by the issuance of Amendment No. 104 (Reference 1) to the facility operating license. The basis for determining that these limits can be extended to 23.6 EFPY is provided below by reviewing the conclusions of each section of the analysis.

1. EXTENDING THE APPLICABILITY OF THE P/T LIMIT CURVES

The P/T limit curve analysis for 15 EFPY was developed using the requirements of 10 CFR 50 Appendix G. The basic calculation method was referenced to ASME Boiler and Pressure Vessel Code, Section III, Appendix G (1986). The current method is unchanged except that in 1992 ASME relocated Appendix G to Section XI, and the reference stress intensity, K_{IR}, is now referred to as K_{Ia}.

The irradiated material properties for the P/T limit curves are based upon the irradiation damage prediction methods of RG 1.99, Rev. 2, which are used to calculate the adjusted reference temperature (ART) for all reactor vessel beltline materials. These ART predictions utilize initial material test properties, material chemistry, neutron fluence and margin, and is the primary material variable input that is considered in the P/T analysis. A higher ART value is more limiting for plant operation.

The ART is determined for the reactor vessel limiting beltline materials for the period of applicability for the P/T limit curves based on the predicted neutron fluence. These values are shown in the first portion of Table 1 (Attachment 2 of this submittal). The controlling values of ART used for the P/T limit curves and LTOP analysis at 15 EFPY were 191°F at the 1/4 Thickness (1/4T) location and 137°F at the 3/4 Thickness (3/4T) location (Ref. 3, Attachment 4, page 9). The controlling material was the lower shell longitudinal welds (3-203 A, B and C, material heat number 305424) for the 1/4T location and two of the lower shell plates (C-8-1 and C-8-2) for the 3/4T location. The term "controlling" means having the highest ART for a given time and position within the vessel wall. The highest ART's are then used to develop the P/T limits for the corresponding period. Note that in developing the P/T limits, a bounding material (SA 302 Grade B) is assumed.

RG 1.99, Rev. 2, predictions for ART are determined from the following formulas:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

Initial RT_{NDT} = Reference temperature (°F) for the un-irradiated material.

ΔRT_{NDT} = $(CF)f^{(0.28 - 0.10 \log f)}$

"CF"(°F)is the chemistry factor, a function of Copper and Nickel content of the weld or base material. "f" is the calculated neutron fluence (10^{19} n/cm^2, E > 1 MeV) at the depth into the vessel from the inside diameter wetted surface to the postulated flaw (surface, 1/4T and 3/4T). Projections of fluence are made for specific periods of time of operation (EFPY). The thickness T, without the cladding, is conservatively used to give a higher projected RT_{NDT}.

Margin = The quantity (°F) to be added to obtain a conservative upper bound value of ART.

= $2(\sigma_I^2 + \sigma_\Delta^2)^{1/2}$

σ_I = 0°F for measured test values, 17° for generic data
σ_Δ = 17°F for plate and 28°F for welds

New beltline weld material data became available as a result of the research needed to respond to NRC Generic Letter 92-01. This data is highlighted in the second section of Table 1. As described below, the plant specific data significantly reduces the values used in calculating the ART.

For the limiting lower longitudinal welds, the data included new chemistry values for copper and nickel, and an actual initial RT_{NDT}. This new data came from the Duquesne Light Co.

Beaver Valley Unit 1 reactor vessel surveillance program (Ref. 4), which has material identical to the St. Lucie Unit 1 lower longitudinal welds. As a result of using actual plant specific RT_{NDT} data, the overall margin term is reduced from 65.5 °F to 56 °F, because σ_I becomes 0 °F. The initial RT_{NDT} is lowered from -56 °F to -60 °F. The new chemistry values were used with existing St. Lucie 1 qualification data to get new average copper and nickel values (Ref. 4), which lowered the CF value from 200.2 to 191.7.

Based on the surveillance capsule data removed from the St. Lucie Unit 1 reactor vessel, the CF's for the limiting lower shell plates were calculated and provided to FPL by the NRC (Ref. 5). The new CF values were lowered, nominally from 108 to 80. Since plant specific data is used to calculate the CF, the margin term (σ_Δ) can be halved from 34 °F to 17 °F (RG 1.99, Rev. 2, Position 2.1). This data is highlighted in the second section of Table 1.

Using the new data, FPL calculated (Ref. 6) the neutron fluence that would result in values of RT_{NDT} equal to or less than the limiting ARTs determined in the 15 EFPY analysis. The new values of fluence correspond to 23.6 EFPY, as shown in the third section of Table 1. Note that the lower longitudinal welds are now the controlling material at both the 1/4T and 3/4T locations.

Since the 23.6 EFPY limiting RT_{NDT} values were equal to or less than those used in the 15 EFPY analysis, the current Technical Specification P/T limit curves and LTOP analysis would be applicable for a period not to exceed 23.6 EFPY with the same margin of safety as the previous 15 EFPY analysis. This equivalence covers the P/T limits for heatup, cooldown, hydrostatic test and core critical operation.

2. LOWEST SERVICE TEMPERATURE, MINIMUM BOLTUP TEMPERATURE, AND MINIMUM PRESSURE LIMITS

The P/T analysis for 15 EFPY calculated the limits for lowest service temperature, minimum boltup temperature, and minimum pressure limits for reference. These limits are not based on accumulated fluence at the reactor vessel beltline material, and remain unchanged.

The lowest service temperature is based on the most limiting RT_{NDT} for the balance of RCS components, which is the Reactor Coolant Pump (RCP). Accumulated plant operation does not effect this component's material properties, therefore the lowest service temperature remains the same.

The minimum boltup temperature is based on the higher stressed region of the reactor vessel, plus any effects for irradiation effects (Ref. 3) and testing uncertainty. The maximum initial RT_{NDT} associated with the stressed region of the reactor vessel flange is 30 °F. For conservatism a minimum boltup temperature of 80 °F is utilized, which more than accounts for any irradiation effects and testing uncertainty (Ref. 3). The fluence at the flange region is at least two orders of magnitude lower than the peak vessel fluence at the beltline materials (i.e., at fuel core midplane). Reference 9 indicates that the fluence at the top of the active core, which is less than 79 inches above the core midplane, decreases to 0.1 times the peak fluence at the core mid-plane. The flange is approximately 130 inches above the top of the core, resulting in a reduction from the peak fluence by at least two orders of magnitude. Therefore, the radiation affects are minimal, and the 50 °F margin is sufficient to account for any changes in flange material fracture toughness.

The minimum pressure limit is the break point between the minimum boltup temperature and the lowest service temperature, and is defined by the ASME Boiler and Pressure Vessel Code as 20% of the pre-operational hydrostatic pressure (Ref. 3). This value was not affected by accumulated plant operation.

3. CORRECTION FACTORS

Since the P/T limit curves and LTOP setpoints are based on coordinates of pressurizer pressure and indicated RCS fluid temperature, correction factors are included in the analysis to account for actual conditions at the limiting beltline materials. The P/T limits and LTOP analysis for 15 EFPY provided for these correction factors, which address the concerns of NRC Information Notice (IN) 93-58 "Non-conservatism in Low-Temperature Overpressure Protection for Pressurized Water Reactors" (Ref. 8).

The lead/lag temperature differential between the vessel base metal and the RCS bulk fluid has been accounted for.

Pressure correction factors were based upon: 1) the static head due to the elevation difference of the vessel wall adjacent to the active core and the pressurizer pressure instrument nozzle, and; 2) the pressure differential based on the number of reactor coolant pumps (RCP's) in operation. Actual pressure at the core region would be higher than at the RCS hot leg and the pressurizer by the amount of head loss due to RCP flow and the static head. Below 200 °F, flow induced pressure drop is based on two RCP's in operation; above 200 °F, pressure drop is based on three RCP's in operation. This addresses the concerns documented in IN 93-58.

Instrument uncertainties have not been included in the P/T curves. The uncertainties (errors) were determined to be insignificant relative to the conservatism of the margin terms included in the ASME Section III, Appendix G, method. NRC concurrence was documented in the 15 EFPY analysis. Correction factors are not affected by neutron fluence, and, therefore, remain unchanged by the extension of applicability to 23.6 EFPY.

4. LTOP ANALYSIS

The objective of the LTOP analysis for 15 EFPY was to preclude violation of the P/T limits during startup and shutdown conditions. The LTOP analysis remains unchanged by the applicability extension to 23.6 EFPY because the P/T limit curves are not being changed. Therefore it is not necessary to re-analyze or modify the LTOP system.

A relaxation of the LTOP requirements from ASME Code Case N-514 was incorporated in the 1993 Addenda. The code case permits the maximum pressure in the vessel to be 110% of the pressure determined to satisfy the P/T limits per Appendix G of Section XI, Article G-2215. The 15 EFPY LTOP analysis is inherently conservative in that the setpoints for power operated relief valves and administrative and operational controls protect the 100% value of the P/T limits.

5. AMENDING THE FOOTNOTE FOR USE OF HPSI AS A BORATION PATH

One result of the LTOP analysis for 15 EFPY was a requirement to limit the mass addition rate of a HPSI pump when it is used as a boration path during shutdown, with RCS pressure boundary integrity established. The method to limit the mass addition rate was to close two of the four HPSI header isolation valves. However, this requirement only applies when RCS pressure boundary integrity exists.

The syntax used in footnote "*" to TS 3.1.2.1 and TS 3.1.2.3 could result in an unnecessary restrictive interpretation of the requirements to close the HPSI header isolation valves. The proposed revision clarifies that closing two HPSI valves is only required under part (b) of the footnote, e.g., when no charging pumps are operable and RCS pressure boundary integrity exists.

Conclusion

The proposed period of applicability is based on projections of irradiation embrittlement for the reactor vessel beltline limiting materials. Since 1990, when the current P/T limit curves were approved, new material property data has become available to make better predictions of embrittlement for the limiting beltline materials. Using the new data, the period of applicability for the existing P/T limit curves and LTOP requirements can be extended from 15 EFPY to 23.6 EFPY with the same analyzed margin of safety.

REFERENCES:

1. USNRC letter to J.H. Goldberg, "St. Lucie Unit 1 - Issuance of Amendment Re: Pressure Temperature (P/T) Limits and Low Temperature Overpressure Protection (LTOP) Analysis (TAC No. 75386)", dated June 11, 1990. (Amendment 104)

2. (Not used)

3. FPL letter to USNRC, L-89-408, "St. Lucie Unit 1, Docket No. 50-335, Proposed License Amendment, P-T Limits and LTOP Analysis", dated December 5, 1989.

4. FPL letter to USNRC, L-93-286, "St. Lucie Unit 1 and 2, Docket No. 50-335 and 50-389, Generic Letter 92-01, Revision 1, Response to Request for Additional Information (RAI)", dated November 15, 1993.

5. USNRC Letter to J. H. Goldberg; Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity," St. Lucie Unit 1 and 2 (TAC NOS M83505 and M83506); May 26, 1994.

6. FPL calculation PSL-1FJM-95-004, Rev. 0, "St. Lucie Unit 1 Reactor Vessel Adjusted Reference Temperature and LTOP limits at 23.6 EFPY", dated 3/22/95.

7. USNRC Generic Letter 92-01 "Reactor Vessel Structural Integrity", Revision 1, Dated March 6, 1992.

8. USNRC Information Notice 93-58 "Non-conservatism in Low-Temperature Overpressure Protection for Pressurized Water Reactors", dated July 26, 1993.

9. CE Report TR-F-MCM-004 "FPL, PSL-1 Evaluation of Irradiated Capsule W97", dated December, 1983.

ATTACHMENT 2

TABLE 1: ST. LUCIE UNIT 1 REACTOR VESSEL BELTLINE ADJUSTED REFERENCE TEMPERATURES FOR DETERMINING PRESSURE TEMPERATURE AND LTOP LIMITS CURVES.

TABLE 1: ST LUCIE UNIT 1 REACTOR VESSEL BELTLINE ADJUSTED REFERENCE TEMPERATURES FOR DETERMINING PRESSURE TEMPERATURE AND LTOP LIMITS CURVES

15 YEAR PT/LTOP CURVES USING EXISTING 1989 ANALYSIS
(Original critical values bolded)

LOCATION	ID #	Cu%	Ni%	CALC. CF	TABLE CF	INITIAL RTndt	MARGIN	15 EFPY PEAK FLUENCE, E19 n/cm^2	1/4T FLUENCE	3/4T FLUENCE	15 YEAR ART @ 1/4T	15 YEAR ART @ 3/4T
Lower shell plate	C-8-1	0.15	0.56		107.8	20 °F	34 °F	2.05E+19	1.22E+19	4.34E+18	168 °F	**137 °F**
Lower shell plate	C-8-2	0.15	0.57		108.4	20 °F	34 °F	2.05E+19	1.22E+19	4.34E+18	168 °F	**137 °F**
Int to Lower girth welds	90136	0.23	0.11		109.8	-60 °F	56 °F	2.05E+19	1.22E+19	4.34E+18	112 °F	80 °F
Lower longitudinal welds	305424	0.30	0.64		200.2	-56 °F	65.5 °F	1.202E+19	7.16E+18	2.54E+18	**191 °F**	135 °F

NEW CHEMISTRY FACTORS & MARGINS BASED ON GL92-01 VALUES
(Changes resulting from new data submitted with GL92-01 and use of capsule data for plates bolded)

LOCATION	ID #	Cu%	Ni%	CALC. CF	TABLE CF	INITIAL RTndt	MARGIN
Lower shell plate	C-8-1	0.15	0.56	**79.42**		20 °F	**17 °F**
Lower shell plate	C-8-2	0.15	0.57	**79.82**		20 °F	**17 °F**
Int to Lower girth welds	90136	0.23	0.11	**84.80**		-60 °F	**28 °F**
Lower longitudinal welds	305424	**0.28**	**0.63**		**191.7**	**-60 °F**	**56 °F**

EQUIVALENT FLUENCE (23.6 EFPY) AT TIME OF REACHING RT/NDT VALUES
USED IN 15 EFPY CURVES WITH GL 92-01 CHEMISTRY FACTORS/MARGINS (Ref. 7.6)
(New critical values bolded)

LOCATION	ID #	Cu%	Ni%	CALC. CF	TABLE CF	INITIAL RTndt	MARGIN	23.6 EFPY PEAK FLUENCE, E19 n/cm^2	1/4T FLUENCE	3/4T FLUENCE	23.6 EFPY ART @ 1/4T	23.6 EFPY ART @ 3/4T
Lower shell plate	C-8-1	0.15	0.56	79.42		20 °F	17 °F	2.82E+19	1.68E+19	5.97E+18	128 °F	105 °F
Lower shell plate	C-8-2	0.15	0.57	79.82		20 °F	17 °F	2.82E+19	1.68E+19	5.97E+18	128 °F	105 °F
Int to Lower girth welds	90136	0.23	0.11	84.80		-60 °F	28 °F	2.82E+19	1.68E+19	5.97E+18	65 °F	41 °F
Lower longitudinal welds	305424	0.28	0.63		191.7	-60 °F	56 °F	1.783E+19	1.063E+19	3.78E+18	**191 °F**	**136 °F**

St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
<u>RCS Pressure/Temperature Limits</u>

ATTACHMENT 3

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Pursuant to 10CFR50.92, a determination may be made that a proposed license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The pressure-temperature (P/T) limit curves in the Technical Specifications are conservatively generated in accordance with the fracture toughness requirements of 10 CFR 50 Appendix G as supplemented by the ASME Code Section XI, Appendix G recommendations. The RT_{NDT} values are based on Regulatory Guide 1.99, Revision 2, shift prediction and attenuation formula. Analyses of reactor vessel material irradiation surveillance specimens are used to verify the validity of the fluence predictions and the P/T limit curves. Use of these curves in conjunction with the surveillance specimen program ensures that the reactor coolant pressure boundary will behave in a non-brittle manner and that the possibility of rapidly propagating fracture is minimized. Based on the use of plant specific material data, analysis has demonstrated that the current P/T limit curves will remain conservative for up to 23.6 EFPY.

In conjunction with extending the applicability of the existing P/T limit curves, the low temperature overpressure protection (LTOP) analysis for 15 EFPY is also extended. The LTOP analysis confirms that the current setpoints for the power-operated relief valves (PORVs) will provide the appropriate overpressure protection at low Reactor Coolant System (RCS) temperatures. Because the P/T limit curves have not changed, the existing LTOP values have not changed, which include the PORV setpoints, heatup and cooldown rates, and disabling of non-essential components.

The proposed amendment does not change the configuration or operation of the plant, and assurance is provided that reactor vessel integrity will be maintained. Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

By applying plant specific data in the determination of critical vessel material limits, the applicability of the existing pressure temperature limits and LTOP requirements can be extended. There is no change in the configuration or operation of the facility as a result of the proposed amendment. The amendment does not involve the addition of new equipment or the modification of existing equipment, nor does it alter the design of St. Lucie plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Analysis has demonstrated that the fracture toughness requirements of 10 CFR 50 Appendix G are satisfied and that conservative operating restrictions are maintained for the purpose of low temperature overpressure protection. The P/T limit curves will provide assurance that the RCS pressure boundary will behave in a ductile manner and that the probability of a rapidly propagating fracture is minimized. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the discussion presented above and on the supporting Evaluation of Proposed TS Changes, FPL has concluded that this proposed license amendment involves no significant hazards consideration.