Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

FPL

July 19, 2000

L-2000-149
10 CFR 50.90
10 CFR 50.92

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

Re: St. Lucie Unit 2
Docket No. 50-389
Proposed License Amendment
RCS Pressure/Temperature Limits

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating License NPF-16 for St. Lucie Unit 2 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 21.7 effective full power years (EFPY) of operation. The associated low temperature overpressure protection (LTOP) temperature limits, which are based on the pressure/temperature limits, will also be extended to 21.7 EFPY of operation.

This proposed license amendment is similar to St. Lucie Unit 1 Amendment 141 that extended the St. Lucie Unit 1 pressure/temperature limit curves from 15 EFPY to 23.6 EFPY. It is requested that the proposed amendment, if approved, be issued by March 31, 2001 to support continued plant operation. It is estimated St. Lucie Unit 2 will reach 15 EFPY in May 2001. Please issue the amendment to be effective on date of issuance and to be implemented within 60 days of receipt by FPL.

Attachment 1 is an evaluation of the proposed TS changes. Attachment 2 is the Determination of No Significant Hazards Consideration. Attachment 3 is a copy of the appropriate TS pages marked-up to show the proposed changes.

A001

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,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/GRM

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. William A. Passetti, Florida Department of Health and Rehabilitative Services

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

Rajiv S. Kundalkar 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.

Rajiv S. Kundalkar

Rajiv S. Kundalkar

STATE OF FLORIDA

COUNTY OF ST. LUCIE

Sworn to and subscribed before me

this 19 day of JULY, 2000
by Rajiv S. Kundalkar, who is personally known to me.

George R Madden

Name of Notary Public - State of Florida

GEORGE R. MADDEN
Notary Public - State of Florida
My Commission Expires Jun 17, 2004
Commission # CC931387

(Print, type or stamp Commissioned Name of Notary Public)

ATTACHMENT 1

SAFETY ANALYSIS

Introduction

Florida Power and Light Company (FPL) proposes to change the St. Lucie Unit 2 Technical Specifications (TS) for reactor coolant system (RCS) pressure/temperature (P/T) limits allowed for heat up and cool down of the RCS. The applicability of the existing specified limits will be extended from 15 effective full power years (EFPY) to 21.7 EFPY of operation based on new fluence information. The associated low temperature overpressure protection (LTOP), which is based on the pressure/temperature limits, will also be extended to 21.7 EFPY of operation.

Background

Pressure/temperature limits are developed to satisfy 10 CFR Part 50 Appendix A, Design Criterion 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 rapid failure, and of gross failure. 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 the probability of rapidly propagating fracture is minimized.

The requirements of 10 CFR 50 Appendix G *Fracture Toughness Requirements* describe the requirements for developing P/T limits and the basis for the limitations. The margins of safety against fracture provided by the P/T limits using the requirements of 10 CFR 50 Appendix G are equivalent to those recommended in ASME Section III, Appendix G (now Section XI, Appendix G). The method to predict the reactor vessel (RV) material irradiation damage is provided in Regulatory Guide 1.99 Revision 2, *Radiation Embrittlement of Reactor Vessel Materials*.

The current P/T limit curves were approved in 1990 by St. Lucie Unit 2 License Amendment 46[1] and expire at 15 EFPY. The period of applicability was based on projections of irradiation embrittlement for the reactor vessel beltline limiting materials. At that time, an assumption in the fluence analysis was that St. Lucie Unit 2 would be switching to 24-month fuel cycles which would result in a higher fluence (compared to 18-month cycles) over the same period of time or EFPY. The 24-month cycle plan was never implemented, and the accumulated fluence and corresponding embrittlement was much less in the 15 EFPY period.

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[1] NRC letter to FPL, St. Lucie Unit 2 - *Issuance of Amendment 46 Pressure/ Temperature Limit and Low Temperature Overpressure Protection, TAC No. 76016*, dated August 08, 1990.

Since that time, there have been minor material property data changes to the low copper, low nickel RV beltline welds, as a result of responses to NRC Generic Letter 92-01 *Reactor Vessel Structural Integrity* (L-97-223[2] and L-99-189[3]) but there have been no changes that effect the limiting plate materials. The calculation method[4] to determine projected RT_{ndt} or adjusted reference temperature (ART), as a result of irradiation embrittlement, has also remained the same since the last P/T limits analysis was submitted. Since the limiting material properties and the calculation methods have not changed, a new period of P/T limit curve applicability has been determined using the actual accumulated fluence to date and current future fluence projections for 18-month cycles. Using the new fluence prediction data, the existing P/T curves in the Technical Specifications can be extended with the same analyzed margin of safety.

Overpressure protection is provided to keep the RCS pressure below the P/T limits after the initiation of assumed energy-addition and mass-addition transients, while operating at low temperatures, in accordance with Standard Review Plan Section 5.2.2, Revision 2. Since this evaluation will demonstrate that the existing P/T limit curves remain unchanged, the LTOP requirements which are based on the P/T limit curves will remain unchanged.

Description of Proposed Technical Specification Change

The current St. Lucie Unit 2 Technical Specification reactor coolant system (RCS) P/T limits are applicable up to 15 EFPY of operation. The existing LTOP analysis that is based upon these P/T limits is also applicable up to 15 EFPY.

The proposed extension to the P/T limits, which are based upon fluence predictions at 21.7 EFPY and the Limiting Conditions for Operation (LCO) ensure that all RCS components will be able to withstand the effects of cyclic loads due to system temperature and pressure changes without their functions or performance being impaired. These cyclic loads are introduced by normal load transients, reactor trips, startup, and shutdown operations. The LTOP system, provided by the power operated relief valves (PORVs) and also by the shutdown cooling system (SDCS) relief valves when the SDCS is operating, ensures RCS over pressurization below certain temperatures would be prevented, thus maintaining reactor coolant pressure boundary integrity.

The existing LTOP analysis for 15 EFPY has also been extended to 21.7 EFPY based upon extending the P/T limits. The LTOP analysis yields an LCO that constitutes LTOP alignments beyond 15 EFPY.

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[2] FPL letter to NRC, L-97-223, *St. Lucie Unit 1 and 2 Reactor Vessel Structural Integrity, Generic Letter 92-01, Revision 1, Updated Information*, dated August 28, 1997.

[3] FPL letter to NRC, L-99-189, *St. Lucie Unit 1 and 2 Requested Corrections to the NRC RVID2 Database, Generic Letter 92-01, Revision 1, Supplement 1*, dated August 26, 1999.

[4] NRC Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Revision 2, dated May 1988.

The proposed changes are as follows:

1. On INDEX pages XXI and XXII change ***15 EFPY*** to ***21.7 EFPY*** in the title of Figures 3.4-2, 3.4-3, 3.4-4, and delete reference to Figure B3/4.4-1. This is an administrative conforming change to the titles of the figures only.

2. LCO 3.4.9.1 currently provides the pressure and temperature limits in terms of Figures 3.4-2, 3.4-3, and 3.4-4 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 ***21.7 EFPY***.

3. LCO 3.4.9.3 currently provides for low temperature overpressure protection, when the RCS is not vented by at least 3.58 square inches, in terms of cold leg temperature as defined in Tables 3.4-3, and 3.4-4 during heatup and cooldown in modes 4, 5, and 6. The table defines an operating period of applicability as > ***6 EFPY*** and less than or equal to ***15 EFPY***. The proposed amendment would use these existing tables values, as-is, and only revise the operating period to be less than or equal to ***21.7 EFPY***.

4. Changed the TS 3/4 4.9 Bases as indicated to be consistent with this proposed amendment. The change to the Bases deletes the obsolete Figure B3/4.4-1 on page B3/4 4-10.

Basis And Justification Of Proposed Change

The analysis for the current P/T limits and LTOP requirements for 15 EFPY were provided with the proposed license amendment as FPL Letter L-90-43[5] and subsequently approved by the issuance of St. Lucie Unit 2 License Amendment 46. The extension is needed because the curves are nearing expiration in terms of EFPY. The existing 15 EFPY limit curves can be extended because the actual fluence to reach the limiting ART that the curves are based on, will not be reached until 21.7 EFPY. The P/T Limits and LTOP requirements are unchanged from the current 15 EFPY analysis that was approved in License Amendment 46 with the exception of the fluence projections and the period of applicability. The extended period of applicability for the curves and limits is due to new fluence projections which show that the accumulated fluence assumed in the 1990 analysis will not be achieved until 21.7 EFPY. The basis and justification to change the expiration of these P/T limits and LTOP requirements from 15 EFPY to 21.7 EFPY is provided below by reviewing the conclusions of each section of the analysis.

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

[5] FPL letter to NRC, L-90-43, *St. Lucie Unit 2 Proposed License Amendment, P-T Limits and LTOP Analysis*, dated February 07, 1990.

Vessel Code, Section III, Appendix G (1986). The current method is unchanged except for the following. In 1992, the ASME relocated Appendix G to Section XI, and the reference stress intensity, K_{Ir}, is now referred to as K_{Ia}. Also, in 1999 the NRC approved the 1995 through 1996 addenda of ASME Section XI for use. This version incorporated Code Case N-514 which permits the LTOP 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 for LTOP[6]. The existing LTOP analysis does not utilize this margin relaxation.

The irradiated material properties for the P/T limit curves are based upon the irradiation damage prediction methods of Regulatory Guide 1.99 Revision 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, fluence and margin, and is the primary material variable input that is considered in the P/T analysis. In 1998, the second surveillance capsule was evaluated as part of the St. Lucie Unit 2 reactor vessel surveillance program[7] allowing actual data to be considered in the determination of ART. The results of that capsule combined with the first indicate that the limiting plate material data was credible and a margin term reduction is available per R.G. 1.99 Revision 2. The new 21.7 EFPY period of applicability is inherently conservative because the full margin term of 34°F (plate) was used to determine ART for the period of applicability, with no reduction taken as a result of the credible surveillance data. The R.G. 1.99 Revision 2 methodology is still the current embrittlement prediction method used and the material inputs for the limiting beltline materials are unchanged with the exception of the projected fluence.

At the time that the current 15 EFPY analysis was prepared, the fluence projections for St. Lucie Unit 2 incorporated the higher flux associated with a pending 24 month fuel cycle plan and the conservatism associated with forward fluence projections. The resulting 15 EFPY maximum projected beltline fluence was 1.826E+19 n/cm^2.

Projected fluence is determined by the calculation of fluence to date and using recent core loading pattern data to project into the future. The model used for this calculation is benchmarked against actual measurements taken from surveillance capsule dosimetry data. The most recent fluence benchmarking with a surveillance capsule at St. Lucie Unit 2 occurred in 1998. NRC Draft RG 1053[8], supplies guidance for these calculations and was used in preparation of the calculation which forms the basis for the fluence used to determine the new period of applicability for the P/T limit curves. The peak vessel fluence is used in the ART projections since the St. Lucie Unit 2 limiting material is plate.

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[6] ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components - 1995 Edition through the 1996 Addenda. (Accepted for use by the NRC as published in the 64 Federal Register, Vol. 64, No. 183, Pages 51370-51400 on September 22, 1999.)

[7] Westinghouse Report WCAP-15040 *Analysis of Capsule 263° from the FPL St. Lucie Unit 2 Reactor Vessel Radiation Surveillance Program*, dated April, 1998.

[8] NRC Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Pressure Vessel Neutron Fluence," June 1996.

Since St. Lucie Unit 2 has maintained the lower flux 18 month cycles and actual fluence data has been collected (cycle 5-8) to replace conservative projections, it will take significantly longer than 15 EFPY to reach the limiting ART input values for the PT limit curves and LTOP analysis. The new fluence projection at the end of cycle (EOC) 12 with 15.421 EFPY is 1.27E+19 n/cm^2 with a projected fluence of 8.846E+17 n/cm^2 per EFPY thereafter. Using the new accumulated and projected fluence, the time to reach the fluence (and ART values) used as input into the 15 EFPY PT limit curves and LTOP set point analysis can be determined as follows:

F = Fluence
F at P/T curve expiration = 1.826E+19 n/cm^2
F @ EOC 12 = 1.27 E+19 n/cm^2 (15.421 EFPY)
F/EFPY = 8.846E+17

Determine remaining F to expiration of P/T Limit Curves:

(F @P/T Curve expiration)-(F @15.421 EFPY) =F remaining past EOC12

1.826E+19 n/cm2- 1.27E+19 n/cm2 = 0.556E+19 n/cm2

Convert F remaining (Past EOC 12) to EFPY:

(F remaining)/(F per EFPY) = F remaining in EFPY

0.556E+19 n/cm2 / 8.846E+17 n/cm2/EFPY = 6.285 EFPY

New EFPY to reach limiting fluence in the current PT limit Curves

15.421 EFPY + 6.285 EFPY = 21.706 EFPY

The result is that the fluence and ART values used in the 15 EFPY P/T limit curve and LTOP analysis will not be accumulated until 21.7 EFPY.

The original 15 EFPY projected values of ART for the beltline plate materials are provided in Table 1. The controlling material is Plate M-605-1 for this fluence level. The term *controlling* means having the highest ART for a given time and position within the vessel wall. The highest ARTs are then used to develop the P/T limits for the corresponding period. The weld materials are not shown below, since they have an extremely low copper and low nickel content and the nearest weld ART is more than 80°F lower than the controlling plate. There have been minor chemistry changes in the values for the beltline welds, as a result of the FPL responses to NRC GL 92-01. None of the changes make the welds more embrittled than the controlling plate for this period of applicability. Therefore the weld materials are clearly not a consideration for determination of P/T limits.

TABLE 1
Original PT/LTOP CURVE Material input data From FPL Letter L-90-43
(Controlling RT_{NDT} Values in Bold and Enlarged)

LOCATION	ID #	Cu%	Ni%	TABLE CF	INITIAL RTndt	MARGIN (RG Position 1.1)	15 EFPY PEAK FLEUNCE E19 n/cm^2	15 EFPY ART @ 1/4T	15 EFPY ART @ 3/4T
Lower shell plate (M4116-1)	B-8307-2	0.06	0.57	37.0	20 °F	34 °F	1.826	92 °F	81 °F
Lower shell plate (M4116-2)	A-3131-1	0.07	0.60	44.0	20 °F	34 °F	1.826	99 °F	86 °F
Lower shell plate (M4116-3)	A-3131-2	0.07	0.60	44.0	20 °F	34 °F	1.826	99 °F	86 °F
Int. shell plate (M-605-1)	A-8490-2	0.11	0.61	74.2	30 °F	34 °F	1.826	**140 °F**	**119 °F**
Int. shell plate (M-605-2)	B-3416-2	0.13	0.62	91.5	10 °F	34 °F	1.826	138 °F	111 °F
Int. shell plate (M-605-3)	A-8490-1	0.11	0.61	74.2	0 °F	34 °F	1.826	110 °F	89 °F

Since the current projected fluence at 21.7 EFPY results in the same fluence and limiting ART values 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 21.7 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.

LOWEST SERVICE TEMPERATURE, MINIMUM BOLT-UP TEMPERATURE, AND MINIMUM PRESSURE LIMITS

The P/T analysis for 15 EFPY also provided the limits for lowest service temperature, minimum bolt-up 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 plus 100°F per ASME Section III NB2332. The most limiting RT_{ndt} for the balance of the RCS is the reactor coolant system piping (+60°F). Accumulated plant operation does not effect this component's material properties; therefore the lowest service temperature remains the same.

The minimum bolt-up temperature is the minimum allowable temperature at pressures below the 20% of the pre-operational system hydrostatic test pressure that stresses can be applied to

the flange region. It is defined as the initial RT_{ndt} for the higher stressed region of the reactor vessel, plus any irradiation effects[9] (Section G-2222), and testing uncertainty. The maximum initial RT_{ndt} associated with the stressed region of the reactor vessel flange (which conservatively includes the upper shell plate adjacent to the flange ring) is 50 °F. For conservatism a minimum bolt-up temperature of 80 °F is utilized, which more than accounts for any measuring or testing uncertainty. The flange region fluence is greater than three orders of magnitude lower than the peak vessel fluence at the vessel beltline and therefore, there is no measurable irradiation effect on the flange region material properties. Therefore, the 80°F minimum bolt-up temperature is unchanged and provides sufficient margin over the measured flange region RT_{ndt} of 50 °F to account for any uncertainties or changes in flange material fracture toughness.

The minimum pressure limit is the break point between the minimum bolt-up 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 after accounting for pressure corrections and pump flow corrections. This value was not affected by accumulated plant operation.

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 93-58 *Non-conservatism in Low-Temperature Overpressure Protection for Pressurized Water Reactors*.

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) 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 RCPs in operation; above 200 °F, pressure drop is based on three RCPs in operation. This addresses the information notice concerns.

The lead/lag temperature differential between the vessel base metal and the RCS bulk fluid has been accounted for in the calculations based on the rate of heat up or cooldown.

Instrument uncertainties have also been factored into the LTOP set point for the power operated relief valves (PORV) to account for the relative instrument uncertainty between the pressure indication and actuation channels. These uncertainties were based on a 24-month cycle which has never occurred, so there is additional conservatism over the 18-month cycle schedule.

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[9] ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components - 1995 Edition through the 1996 Addenda.

Correction factors are not affected by neutron fluence, and, therefore, remain unchanged by the extension of applicability to 21.7 EFPY.

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 21.7 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 of ASME Section XI Appendix G. This code change permits the LTOP system to limit 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, Eq. 1). The NRC[10] has accepted this code change for use.

The current 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 per Appendix G of Section XI, (Article G-2215, Eq. 1) and do not use the ASME Code relaxation above.

Conclusion

The proposed license amendment will extend the effectiveness of the current St. Lucie Unit 2 Technical Specification pressure/temperature (P/T) limit curves from 15 to 21.7 effective full power years (EFPY). The low temperature overpressure protection (LTOP) requirements, which are based on the P/T limits, would also be extended to 21.7 EFPY. The P/T limits and LTOP requirements are unchanged from the current 15 EFPY analysis. The extended period of applicability for the curves and limits is due to new fluence projections which show that the accumulated fluence assumed in the 1990 analysis will not be achieved until 21.7 EFPY. The previous fluence projections were conservatively based on a higher flux 24-month cycle operating schedule, which was never implemented. Using the new fluence data, the period of applicability for the existing P/T limit curves and LTOP requirements can be extended from 15 EFPY to 21.7 EFPY with the same analyzed margin of safety.

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[10] ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components - 1995 Edition through the 1996 Addenda.

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Florida Power and Light Company (FPL) proposes to change the St. Lucie Unit 2 Technical Specifications (TS) for reactor coolant system (RCS) pressure/temperature (P/T) limits allowed for heat up and cool down of the RCS. The applicability of the existing specified limits will be extended from 15 effective full power years (EFPY) to 21.7 EFPY of operation based on new fluence information. The associated low temperature overpressure protection (LTOP), which are based on the pressure/temperature limits, will also be extended to 21.7 EFPY of operation.

The standards used to arrive at a determination that a request for amendment involves no significant hazards consideration are included in the Commission's regulations, 10 CFR 50.92, which states that no significant hazards considerations are involved if the 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 below:

(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 adjusted reference temperature (ART) values are based on the Regulatory Guide 1.99, Revision 2 shift prediction and attenuation formula and have been validated by a credible reactor vessel surveillance program. There are no changes to the limit curve, only a change in the period of applicability based on more recent fluence predictions. Based on the current fluence projections, analysis has demonstrated that the current P/T limit curves will remain conservative for up to 21.7 EFPY.

In conjunction with extending the effectiveness 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 (PORV) will provide the appropriate overpressure protection at low RCS temperatures. Because the P/T limit curves have not changed, the existing LTOP values have not changed, this includes the PORV setpoints.

The P/T limit curves and LTOP analysis have not changed; therefore, the proposed amendment does not represent a change in the configuration or operation of the plant. The results of the existing LTOP analysis have not changed, and the limiting pressures for given temperatures will not be exceeded for the postulated transients. Therefore, assurance is provided that reactor vessel integrity will be maintained. Thus, the proposed amendment does not involve an increase in the probability or consequences of accidents 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.

The requirements for P/T limit curves and LTOP have been in place since the beginning of plant operation. The only changes in these curves are the extension of the period of applicability (EFPY), which is based on new fluence data and the operating time (EFPY) required to reach the same limiting fluence used for the current 15 EFPY P/T curves. Since there is no change in the configuration or operation of the facility as a result of the proposed amendment, the proposed amendment does 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 ductile manner and that the probability of a rapidly propagating fracture is minimized. Therefore, operation in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Conclusion

Based on the above discussion and the analysis performed, FPL has determined that the amendment request does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new and different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

Environmental Impact Consideration Determination

The proposed license amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendment involves no significant hazards consideration and therefore, meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendment.

ATTACHMENT 3

St. Lucie Unit 2 Marked-up Technical Specification Page

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Delete

FIGURE 3.4-2 21.7
ST. LUCIE-2 P/T LIMITS, ~~15~~ EFPY
HEATUP AND CORE CRITICAL

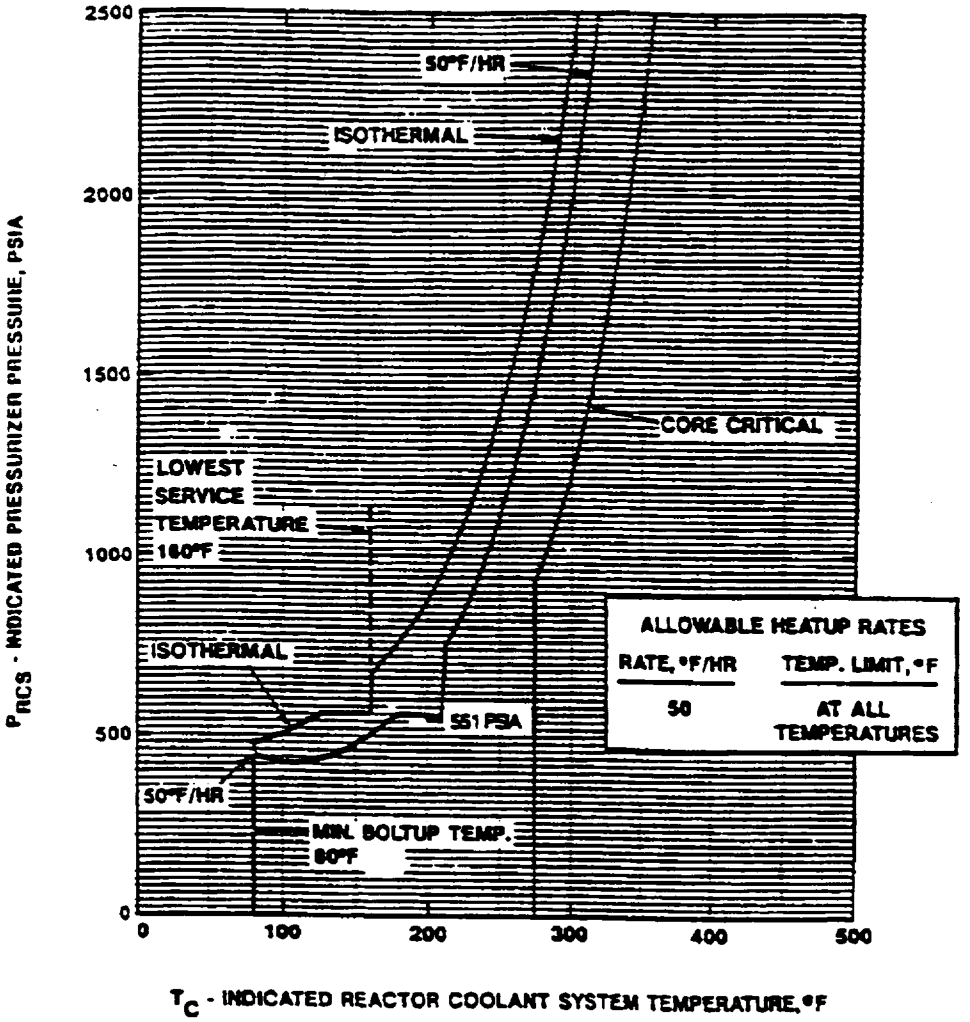

FIGURE 3.4-3
ST. LUCIE-2 P/T LIMITS, ~~1X~~ 21.7 EFPY
COOLDOWN AND INSERVICE-TEST

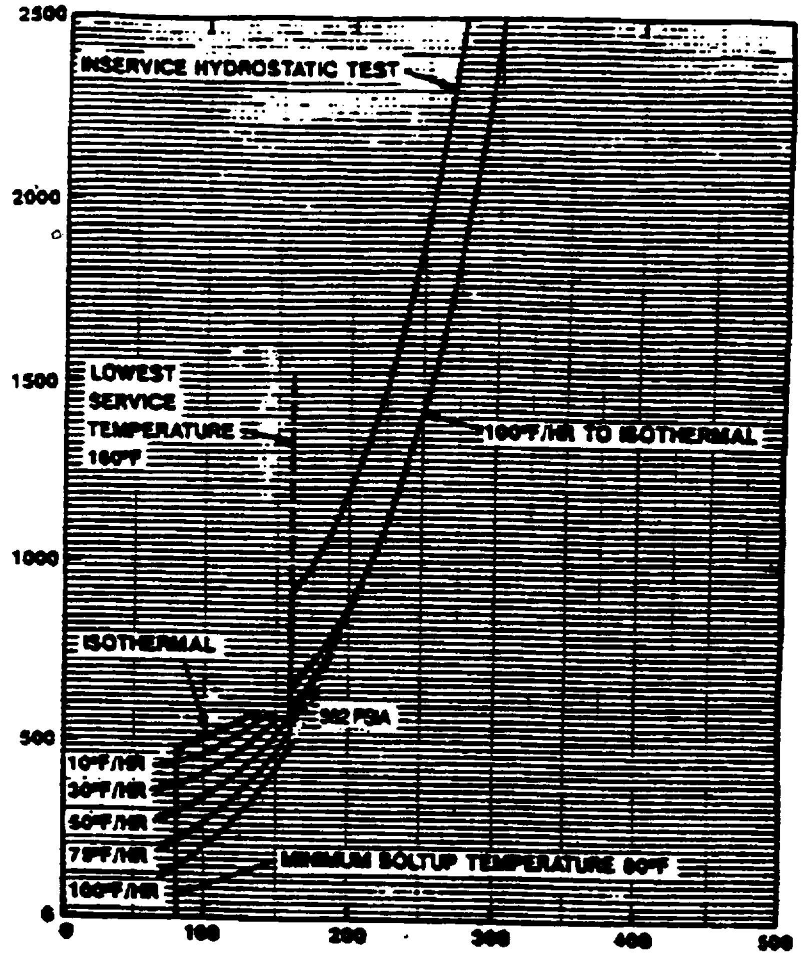

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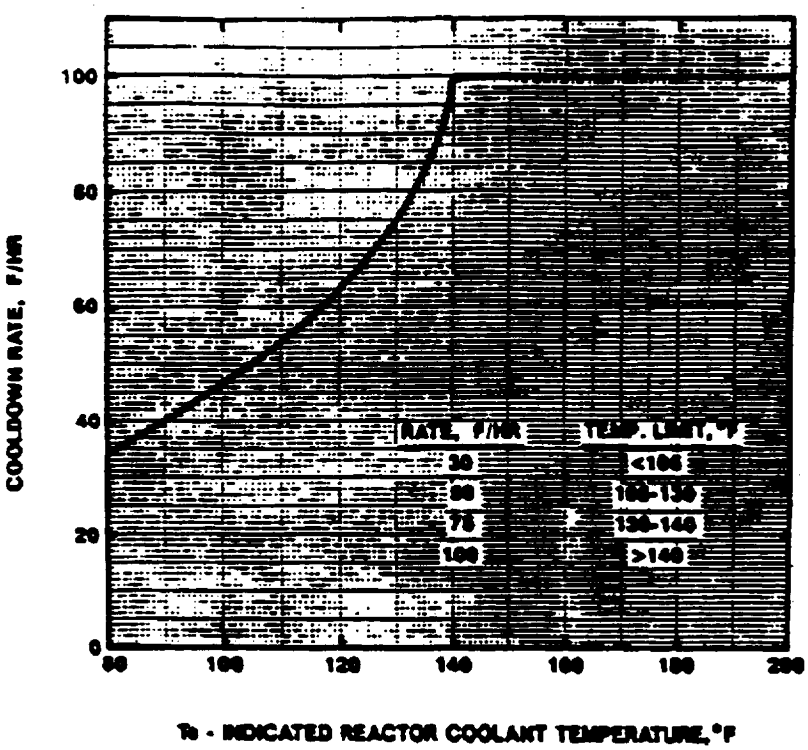
FIGURE 3.4-4
ST. LUCIE-2 P/T LIMITS, 21.7 EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



NOTE: A MAXIMUM COOLDOWN RATE OF 100 F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 140°F

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, EFPY	Cold Leg Temperature, F° During Heatup	Cold Leg Temperature, F° During Cooldown
~~6 < operating period ≤ 15~~ ≤ 21.7	≤ 247	≤ 230

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	T_{cold}, F° During Heatup	T_{cold}, F° During Cooldown
~~6 < operating period ≤ 15~~ ≤ 21.7	165	165

REACTOR COOLANT SYSTEM

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the FSAR.. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron indication damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50 degress F per hour or cooldown rate of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at ~~15~~ 21.7 EFPY, and they include adjustments for pressure differences between the reactor vessel beltline and pressurizer instrument taps.

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The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature can be predicted using a) the initial RT_{NDT}, b) the fluence (E greater than 1 MeV), including appropriate adjustments for neutron attenuation and neutron energy spectrum variations through the wall thickness, c) the copper and nickel contents of the material, and d) the transition temperature shift ~~from the curve shown in Figure B 3/4.4-1~~ as recommended by Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or other approved method. The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 include predicted adjustments for this shift in RT_{NDT} at ~~15~~ 21.7 EFPY.

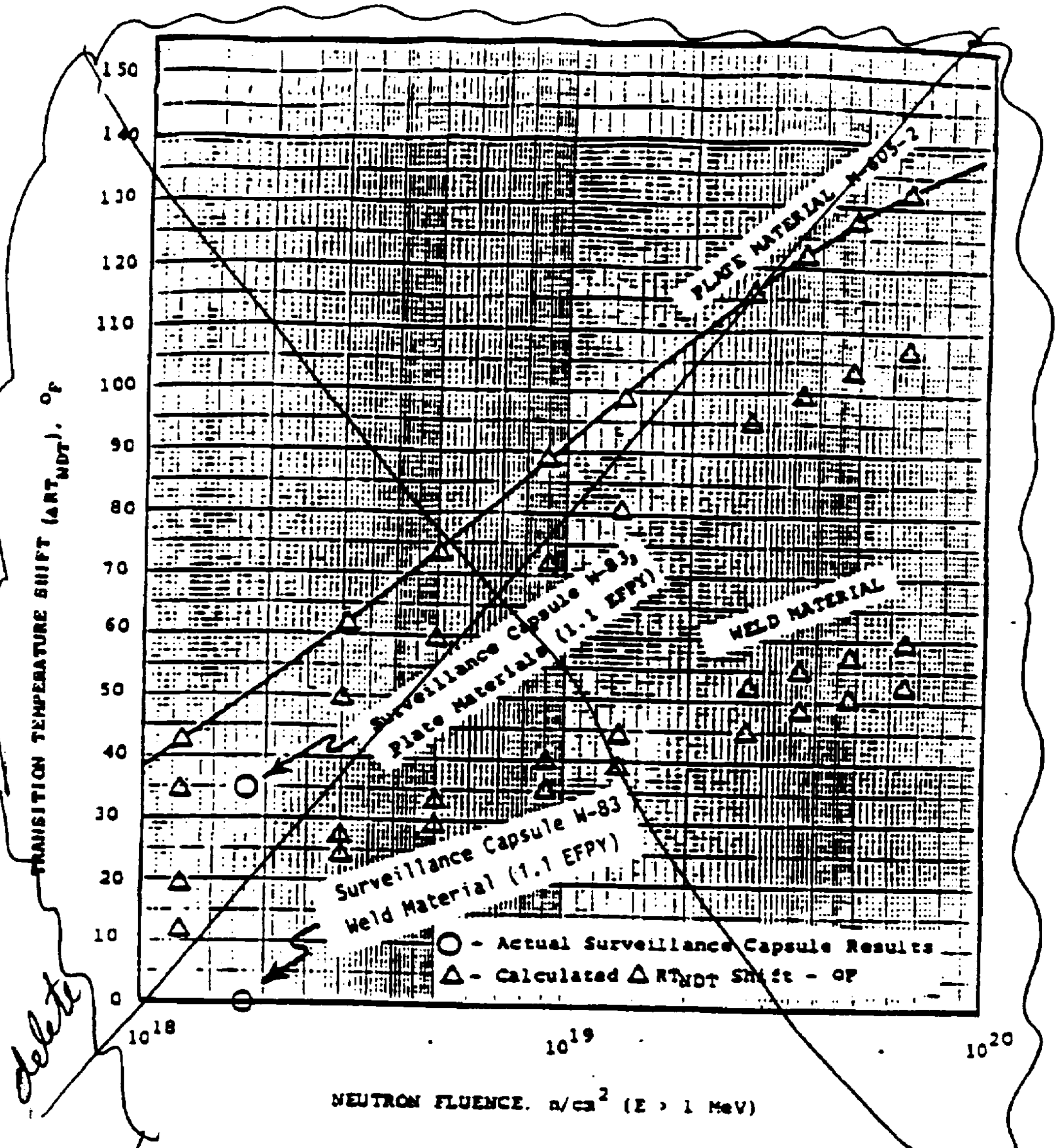

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Figure B 3/4.4-1

Reference Temperature (RT_{NDT}) Increases as a function of fast (E > 1 MeV) neutron fluence (550°F irradiation) for Reactor Vessel Beltline Materials

REACTOR COOLANT SYSTEM

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The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 and 10 CFR 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 60°F. The Lowest Service Temperature limit line shown on Figures 3.4-2, 3.4-3 and 3.4-4 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, two SDCRVs or an RCS vent opening of greater than 3.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold leg temperatures are less than or equal to the LTOP temperatures. The Low Temperature Overpressure Protection System has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) a safety injection actuation in a water-solid RCS with the pressurizer heaters energized or (2) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 40°F above the RCS cold leg temperatures with the pressurizer water-solid.

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The actual shift in RT_{NDT} of the vessel materials will be benchmarked periodically during operation, by removing and evaluating, in accordance with 10CFR50 Appendix H and ASTM E185, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and the vessel inside radius are essentially identical, the measured transition temperature shift in RT_{NDT} for a set of material samples can be compared to the predictions of RT_{NDT} that were used for preparations of the pressure/temperature limits curves. If the measured delta RT_{NDT} values from the surveillance capsule are not conservatively within the measurement uncertainty of the prediction method, then heat up and cooldown curves must be re-evaluated.