Attachment 1

St. Lucie Units 1 Marked-Up Technical Specification Pages

Pages

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SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties~~, at the intervals shown in Table 4.4-5.~~ The results of these examinations shall be used to update Figures 3.4-2a, 3.4-2b and 3.4-3.

DELETE AND REPLACE WITH

AS REQUIRED BY 10 CFR 50 APPENDIX H.

DELETE

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

Specimen Location on Vessel Wall	Lead Factor[2]	Approximate Removal Schedule (EFPY)	Predicted Fluence (n/cm^2)
97°[1]	1.54	4.67	5.5×10^{18}
104°	1.02	10	8.78×10^{18}
284°	1.02	18	1.58×10^{19}[3]
263°	1.54	21	2.78×10^{19}
277°	1.54	32	4.24×10^{19}
83°	1.54	Standby	---

NOTES

1) Information for this capsule is actual
2) Ratio of capsule fluence divided by the fluence at the controlling weld
3) Approximate end of life 1/4T fluence

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.

DELETE

~~The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.~~

The limiations 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.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection program for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. This program is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Winter 1972.

Attachment 2

St. Lucie Unit 2 Marked-Up Technical Specification Pages

Pages

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INDEX

LIST OF TABLES (Continued)

REPLACE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, ~~at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5.~~ The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

DELETE AND REPLACE WITH

AS REQUIRED BY 10 CFR 50 APPENDIX H.

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	83°	≤ 1.5	1.0
2	97°	≤ 1.5	24.0
3	104°	≤ 1.5	STANDBY
4	263°	≤ 1.5	12.0
5	277°	≤ 1.5	STANDBY
6	284°	≤ 1.5	STANDBY

DELETE

ADD 50 DELETE

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-~~73~~ and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. ~~The surveillance specimen withdrawal schedule is shown in Table 4.4-5.~~ DELETE 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 lead factors shown in Table 4.4-5 are the ratio of neutron flux at the surveillance capsule to that at the reactor inside surface.~~ DELETE

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.

Attachment 3

Safety Analysis

Introduction

This change is proposed to revise the St. Lucie Units 1 and 2 Technical Specifications to remove Table 4.4-5, Reactor Vessel Material Surveillance Program Withdrawal Schedule, and any references to the table from the Technical Specifications. The appropriate reactor vessel material withdrawal schedules have already been incorporated in Table 5.4-3 of the Unit 1 FUSAR and Table 5.3-9 of the Unit 2 FUSAR.

Discussion

In accordance with Generic Letter 91-01, the proposed change to the St. Lucie Units 1 and 2 Technical Specifications revises the Reactor Coolant System Section 3/4.4.9, Pressure/Temperature Limits, by removing Table 4.4-5 and any references to the table from the Technical Specifications.

Appendix H Section II.B.3 of 10 CFR Part 50, states, that: "A proposed withdrawal schedule must be submitted with a technical justification as specified in 10 CFR 50.4. The proposed schedule must be approved prior to implementation." Having this schedule in the Technical Specifications duplicates the control on changes to this schedule that has been previously established in 10 CFR 50 Appendix H.

The limiting conditions for operation (LCO) for the Reactor Coolant System include operating limits on pressure and temperature that are defined in Figures 3.4-2a, 3.4-2b and 3.4-3 of the St. Lucie Unit 1 Technical Specifications and Figures 3.4-2, 3.4-3, 3.4-4 of the St. Lucie Unit 2 Technical Specifications. They provide an acceptable region for operation during heatup, cooldown, criticality, and inservice leak and hydrostatic testing. The surveillance requirement associated with this LCO addresses the frequency of verifying that operation is within the specified limits during these operating conditions. Also included is an additional surveillance requirement that states, "The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2a, 3.4-2b and 3.4-3 for the St. Lucie Unit 1 Technical Specifications and Figures 3.4-2, 3.4-3, and 3.4-4 for the St. Lucie Unit 2 Technical Specifications." The proposed change would remove Table 4.4-5, and any references to the table from the Technical Specifications. Because the surveillance

requirement specifies that the results of these examinations shall be used to update Figures 3.4-2a, 3.4-2b and 3.4-3 for St. Lucie Unit 1 and 3.4-2, 3.4-3 and 3.4-4 for St. Lucie Unit 2 for the pressure and temperature limits this requirement will be retained.

St. Lucie Units 1 and 2 Technical Specification Bases Section 3/4.4.9, Pressure/Temperature Limits, gives a detailed description of the bases for this LCO and the related surveillance requirements. The Standard Technical Specification (STS) bases references Table 4.4-5 which provides the schedule for surveillance specimen withdrawal. This Bases Section provides considerable background information on the use of the data gathered from material specimens. This background information clearly defines the objective of this data as it relates to 10 CFR 50 Appendix H and the American Society of Mechanical Engineers (ASME) Code. Deletion of the reference to Table 4.4-5 does not affect the content of this section.

<u>Conclusion</u>

The reactor vessel material withdrawal schedules have already been incorporated into the Unit 1 and 2 FUSAR. Removing them from the Unit 1 and 2 Technical Specifications will not result in any loss of clarity or control over the regulatory requirements of 10 CFR 50 Appendix H.

Attachment 4

Determination of No Significant Hazards Consideration

The standards used to arrive at a determination that a request for amendment involves no significant hazards consideration are included in the Commissions regulation, 10 CFR 50.92, which state 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 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 proposed amendment change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the regulatory requirement of 10 CFR 50 Appendix H will remain in effect in the Technical Specifications. Removing Table 4.4-5, and any references to it, will not result in any loss of regulatory control because changes to this schedule are controlled by the requirements of 10 CFR 50 Appendix H.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The use of this modified specification cannot create the possibility of a new or different kind of accident from any previously evaluated because as previously stated in Appendix H Section II.B.3 of 10 CFR 50, the licensee must have a withdrawal schedule approved by the NRC prior to implementation. By removing Table 4.4-5, and any references to that table, FPL will only eliminate duplication of a requirement that it already adheres to in 10 CFR 50 Appendix H.

(3) Use of the modified specification would not involve significant reduction in a margin of safety.

By removing Table 4.4-5 the margin of safety would not be compromised because the surveillance requirement still requires surveillance specimens to be removed and examined, to determine changes in material properties, at intervals required by 10 CFR 50 Appendix H. In addition the results of

these examinations shall be used to update the figures for the pressure and temperature operating limits required by the Technical Specifications.

Based on the above, we have determined that the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or 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.