ATTACHMENT 1

Marked-up St. Lucie Unit 1 Technical Specifications Pages:

3/4 4-24 (plus replacement)
B 3/4 4-7 (plus insert)

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
1.	8 years
2.	16 years
3.	23 years
4.	30 years
5.	35 years
6.	40 years

Replace with attached

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

The heatup and cooldown limit curves (Figures 3.4-2a and 3.4-2b) 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°F/hr and for any cooldown rate of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the applicable service period.

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>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be calculated based upon the fluence. The heatup and cooldown limit curves shown on Figures 3.4-2a and 3.4-2b include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

~~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, 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 irratiation 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 ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.~~

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The pressure-temperature limit lines shown on Figures 3.4-2a and 3.4-2b 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 estimated to be 90°F. The Lowest Service Temperature limit line shown on Figures 3.4-2a and 3.4-2b is based upon this RT_{NDT} since Article NB-2332 of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F

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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 E185-82, reactor vessel material surveillance specimens installed near the inside wall of the reactor vessel in the core area. The capsules are scheduled for removal at times that correspond to key accumulated fluence levels within the vessel through the end of life. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, measured ΔRT_{NDT} for surveillance samples can be applied with confidence to the corresponding material in the reactor vessel wall. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

ATTACHMENT 2

SAFETY ANALYSIS

INTRODUCTION

Title 10 CFR 50 Appendix H requires reactor vessels constructed of ferritic materials to have their beltline regions monitored by a surveillance program. The St. Lucie Unit 1 program was designed to meet the requirements of ASTM E185-73, which was the current edition corresponding to the issue date of the American Society of Mechanical Engineers Code when the reactor vessel was purchased. Title 10 CFR 50 Appendix H requires capsules withdrawn after July 26, 1983, to meet the testing requirements of ASTM E185-82 to the extent practical. This amendment will update the St. Lucie Unit 1 surveillance program to the extent practical by revising the capsule removal schedule to reflect the requirements of ASTM E185-82.

St. Lucie Unit 1 has a program consisting of six surveillance capsules attached to the inner radius of the vessel in the beltline region. Samples of the beltline shell plate, weld and heat affected zone are included in the surveillance capsules.

DISCUSSION

Standard ASTM E185-82 recommends a minimum number of capsules for removal and testing based on the predicted transition temperature shift at the vessel inside surface. Based on the current fluence projections and predictions of transition temperature shift, St. Lucie Unit 1 will remove a minimum of 5 capsules for testing. The current schedule for removal of capsules from the vessel, which is in calendar years, has been in place since the initial licensing of St. Lucie Unit 1. Standard ASTM E185-82 provides a withdrawal schedule in terms of effective full power years (EFPY) or key fluence levels at 1/4 T and the inner diameter (ID) at end of life, whichever comes first. Due to the fact that the St. Lucie Unit 1 capsules are attached to the vessel wall, these capsules have relatively low lead factors. These low lead factors result in capsules removed in accordance with the ASTM E185-82 EFPY removal schedule not having the key fluence levels and corresponding shifts in transition temperature benchmarked against the predictive calculational methodology of Regulatory Guide 1.99, Revision 2.

To obtain the most meaningful results from the surveillance capsules, Florida Power & Light Company proposes to amend the St. Lucie Unit 1 surveillance capsule removal schedule to allow capsules to be removed at times when key fluence levels can be obtained to predict 1/4 T and ID end of life shifts in transition temperature. An additional enhancement of the new removal schedule is including the predicted fluences at the capsule removal times as well as the actual data from the 97° capsule test. The proposed schedule is in EFPY per ASTM E185-82 recommendations.

ATTACHMENT 3

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 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 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 current St. Lucie Unit 1 surveillance capsule withdrawal schedule is based upon the original licensing withdrawal schedule. Subsequently, ASTM E185-82 has been developed which recommends capsule removal numbers and intervals which are more representative of the reactor pressure vessel embrittlement at the 1/4 T and inner diameter (ID) locations. The proposed amendment will result in better predictions of reactor vessel material embrittlement in accordance with the requirements of 10 CFR 50 Appendix H. Additionally, the same number of surveillance capsules will be removed in the proposed removal schedule as are currently required to be removed. Accordingly, 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.

The proposed amendment revises the St. Lucie Unit 1 surveillance capsule withdrawal schedule to more accurately represent vessel embrittlement at the 1/4 T and ID locations. This improved schedule follows the recommendations of ASTM E185-82 and meets the requirements of 10 CFR 50 Appendix H. Additionally, the same number of surveillance capsules will be removed in the proposed removal schedule as are currently required to be removed. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

The revised surveillance capsule withdrawal schedule provides for more realistic determinations of reactor pressure vessel material embrittlement at the 1/4 T and ID locations. These determinations result in more refined evaluations of material transition temperature shifts to meet the requirements of 10 CFR 50 Appendix H. The more refined removal interval takes into account current neutron fluence projections which were not available at plant licensing 13 years ago. This results in a program meeting the recommendations of ASTM E185-82. Additionally, the same number of surveillance capsules will be removed in the proposed removal schedule as are currently required to be removed. Therefore, the proposed amendment does not involve a reduction in a margin of safety.

Based on the above, we have 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 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.