Technical Specification Change Request No. 28 (Appendix A)

Replace pages 3/4 4-29, B3/4 4-12 and B3/4 4-13 with the attached revised pages 3/4 4-29, B3/4 4-12 and B3/4 4-13.

Proposed Changes

Change Table 4.4-5 to show the removal of Crystal River Unit 3 Capsule B at 270 $\pm$10 EFPD. Also change the installation schedule of the remaining CR3 capsules to 270 $\pm$10 EFPD.

Change the Bases for Specification 3/4.4.9.1 to reflect the removal of CR3 Capsule B at 270 $\pm$10 EFPD. Also, change the Bases to correct typographical errors in the figure references.

Reason for Proposed Change

Crystal River Unit 3 Capsule B was scheduled to be removed at the end of the first fuel cycle. Since the temperature-pressure limit curves were prepared based upon the predicted impact properties of the reactor vessel at the end of the fifth effective full power year, it was predicted that no readjustment would be required to the curves until that time. Capsule B was to be used only to check that the calculations of ΔRT_{NDT} were as predicted after the first fuel cycle (at whatever the exposure happened to be).

The plant was shut down in March due to the failure of a burnable poison rod assembly. Subsequent inspection and repair resulted in the removal of the reactor vessel head and the core support assembly making the capsule holders readily accessible. Capsule B then could be removed if the data obtained from the specimens could give the results as would be obtained if the capsule were removed at the first refueling outage.

Appendix H of 10 CFR Part 50 requires the first capsule to be withdrawn when the ΔRT_{NDT} reaches 50°F or 1/4 of service life, whichever is earlier. For CR3, a 50°F shift in RT_{NDT} has already occurred. Therefore, withdrawal of Capsule B at 269 EFPD will address the requirements of Appendix H and will provide the necessary information to check the temperature-pressure limit curves.

The installation of CR3 Capsules A, C, E and F at 270 $\pm$10 EFPD of the first cycle will give these capsules approximately 200 EFPD of additional exposure beyond that already projected but should have no effect on their removal schedule. In fact, this additional exposure within the original removal schedule will allow the comparison of real and predicted ΔRT_{NDT} at greater than anticipated burnups.

Safety Analysis of Proposed Change

This change only revises the installation and removal schedule of the reactor vessel surveillance specimens. None of the requirements applicable to the safety analysis are diminished by the proposed changes and no unreviewed safety question is involved.

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TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

CR#3 Capsule	Installation	Removal
A	At 270 $\pm$10 EFPD of First Cycle	Standby
C	At 270 $\pm$10 EFPD of First Cycle	End of Ninth Cycle
E	At 270 $\pm$10 EFPD of First Cycle	Standby
B	Initial Fuel Load	At 270 $\pm$10 EFPD of First Cycle
D	Initial Fuel Load	End of Fifth Cycle
F	At 270 $\pm$10 EFPD of First Cycle	End of Fifth Cycle

All pressure-temperature limit curves are applicable up to the fifth effective full power year. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, and 3.4-3, and 3.4-4.

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

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 limitations imposed on pressurizer heatup and cooldown 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 inspection programs for ASME Code Class 1, 2 and 3 components, except steam generator tubes, ensure that the structural integrity of these componentns will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves 1) ensure OPERABILITY, 2) ensure that the valves are not stuck open during normal operation, and 3) demonstrate that the valves are fully open at the forces assumed in the safety analysis.

4.0 SPECIAL SURVEILLANCE, RESEARCH, OR STUDY ACTIVITIES

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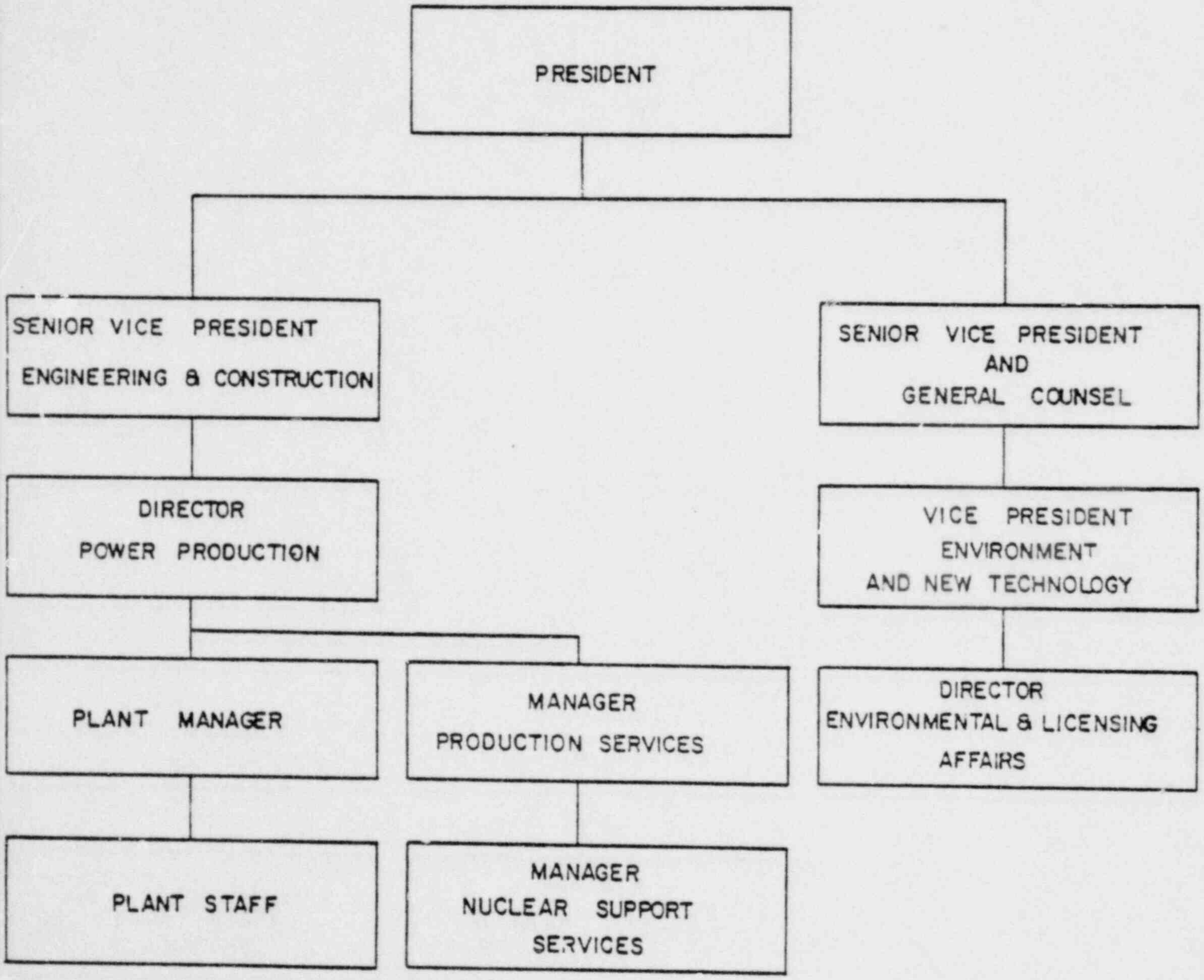

FIGURE 5.1-1 ORGANIZATION FOR IMPLEMENTING ENVIRONMENTAL TECHNICAL SPECIFICATIONS

with confidence to the adjacent section of the reactor vessel. The limit 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. Since the first surveillance program capsule will be withdrawn at 270 $\pm$1(.FPD and the limit curves were prepared based upon the predicted impact properties at the end of the fifth effective full power year, it is predic[illegible] [illegible]c no readjustment will be required to the limit curves for the fir[illegible] [illegible]ctive full power years. Adjustment may be required after the wit[illegible] f the second capsule.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The unirradiated impact properties and residual elements of the beltline region materials are listed in Bases Table 4-1. The adjusted reference temperature are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated. The predicted ΔRT_{NDT} are calculated using the respective neutron fluence and copper and phosphorus contents. Bases Figure 4-1 illustrates the calculated peak neutron fluence. at several locations through the reactor vessel beltline region wall and at the center of the surveillance capsules as a function of exposure time.

Bases Figure 4-2 illustrates the design curves for predicting the radiation induced ΔRT_{NDT} as a function of the material's copper and phosphorus content and neutron fluence. The adjusted RT_{NDT}'s are given for the 1/4T and 3/4T (T is wall thickness) vessel wall locations. The assumed RT_{NDT} of the closure head region and of the outlet nozzle steel forging is 60°F.

During cooldown at the higher temperatures, the limits are imposed by thermal and loading cycles on the steam generator tubes. These limits are the vertical segments of the limit lines on Figures 3.4-3 and 3.4-4, respectively. These limits will not require adjustments due to the neutron fluences.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.