ATTACHMENT 2

TECHNICAL SPECIFICATION PAGES WITH PEN AND INK CHANGES

LICENSE APPLICATION 91-08, NLR-N91131
SPECIMEN REMOVAL SCHEDULE RELOCATION

The following Technical Specifications have been revised to reflect the proposed changes:

Technical Specification	Page
4.4.6.1.3	3/4 4-22
Table 4.4.6.1.3-1	3/4 4-24
Bases 3/4.4.6	B 3/4 4-6
Figure B 3/4.4.6-1	B 3/4 4-8

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR 50, Appendix H ~~in accordance with the schedule in Table 4.4.6.1.3-1~~. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1 based on the greater of the following criteria:

a. The actual shift in reference temperature for plate material from heat 5K3238-1 and weld metal 510-01205 as determined by Charpy impact test, or

b. The predicted shift in reference temperatures for plate material from heat 5K3025-1 as determined by Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials."

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

1. ≤ 100°F, at least once per 12 hours.

2. ≤ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR @ ¼ T	WITHDRAWAL TIME (EFPY)
1	30°	1.20	6
2	120°	1.20	15
3	300°	1.20	EOL

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and C', and A and A', 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 Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

UFSAR Section 5.3 and Appendix 5A

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in ~~Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50~~.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs 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.

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 1977 Edition and Addenda through Summer 1978.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed 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 NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

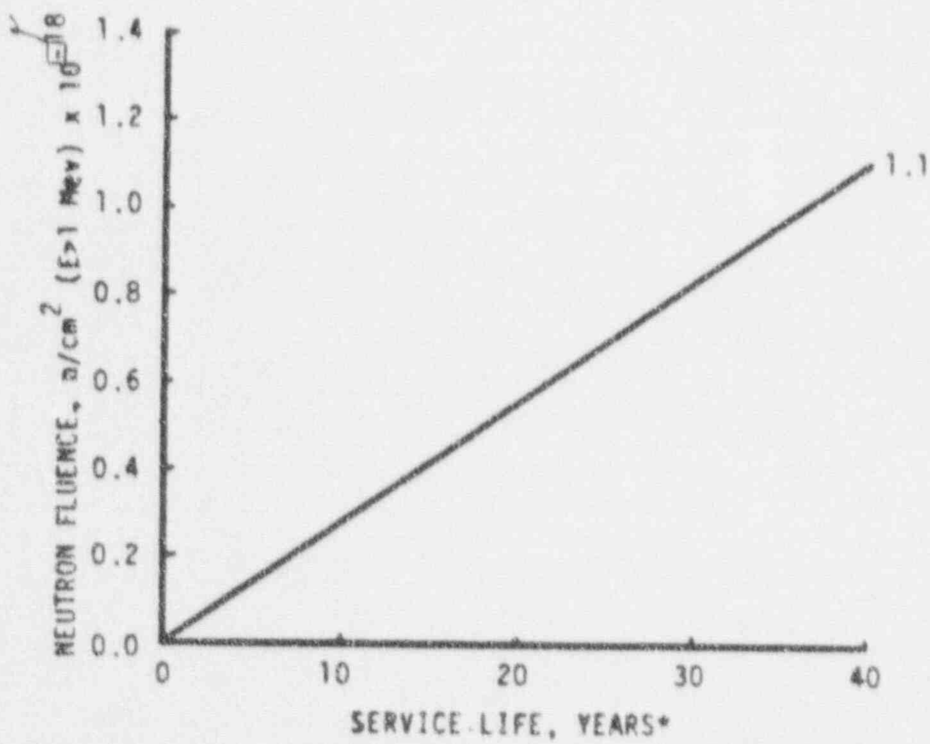

FIGURE B 3/4 4.6-1 FAST NEUTRON FLUENCE (E>1 Mev) AT 1/4 T AS A FUNCTION OF SERVICE LIFE*

Bases Figure B 3/4.4.6-1

* At 90% of RATED THERMAL POWER and 90% availabi