



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

July 14, 1994

Docket No. 50-397

Mr. J. V. Parrish (Mail Drop 1023)  
Assistant Managing Director, Operations  
Washington Public Power Supply System  
P. O. Box 968  
Richland, Washington 99352-0958

Dear Mr. Parrish:

SUBJECT: ISSUANCE OF AMENDMENT CORRECTION FOR THE WASHINGTON PUBLIC POWER  
SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 (TAC NO. M88839)

The Commission issued Amendment No. 122 on May 27, 1994, to the Facility Operating License No. NPF-21 for WPPSS Nuclear Project No. 2. The amendment changed the Technical Specifications (TS) in response to your February 17, 1994, application and May 13, 1994, clarification letter.

Through a telephone conference on July 1, 1994, you identified some administrative errors that had been made in issuing this amendment. The errors and corrections are as follows:

- Page xx(a) was issued with Figure 3.4.6.1.C, "PRESSURE/TEMPERATURE LIMITS FOR 8 EFPY TESTING AND NONNUCLEAR HEATING CURVES," listed as being on page 3/4 4-21b. Page xx(a) has been reissued with the figure listed as being on page 3/4 4-21a.
- Page 3/4 4-21b was issued as a single-sided TS; you agreed that this page should be renumbered as page 3/4 4-21a with page 3/4 4-22 as the overleaf.
- Page B 3/4 4-6 was issued with Bases Table B 3/4.4.6-1, "Reactor Vessel Toughness." You requested that Bases Table B 3/4.4.6-1 be deleted, the words "THIS PAGE IS BLANK INTENTIONALLY" put in the table's place, and the page reissued with overleaf page B 3/4 4-5.

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Mr. J. V. Parrish

- 2 -

July 14, 1994

Copies of the corrected pages are enclosed to replace the incorrect pages. We apologize for any inconvenience this might have caused you.

Sincerely,

ORIGINAL SIGNED BY BRIAN E. HOLIAN FOR:

James W. Clifford, Senior Project Manager  
Project Directorate IV-2  
Division of Reactor Projects III/IV  
Office of Nuclear Reactor Regulation

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Mr. J. V. Parrish

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July 14, 1994

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Sincerely,

ORIGINAL SIGNED BY BRIAN E. HOLIAN FOR:

James W. Clifford, Senior Project Manager  
Project Directorate IV-2  
Division of Reactor Projects III/IV  
Office of Nuclear Reactor Regulation

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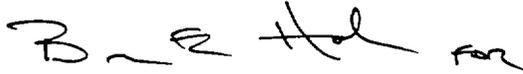
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Mr. J. V. Parrish

- 2 -

Copies of the corrected pages are enclosed to replace the incorrect pages. We apologize for any inconvenience this might have caused you.

Sincerely,

A handwritten signature in black ink, appearing to read "James W. Clifford". The signature is written in a cursive style with a large initial "J" and a long horizontal stroke at the end.

James W. Clifford, Senior Project Manager  
Project Directorate IV-2  
Division of Reactor Projects III/IV  
Office of Nuclear Reactor Regulation

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See next page

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(WNP-2)

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# WNP-2 PRESSURE/TEMPERATURE LIMITS FOR 8 EPFY TESTING AND NONNUCLEAR HEATING CURVES A' & B'

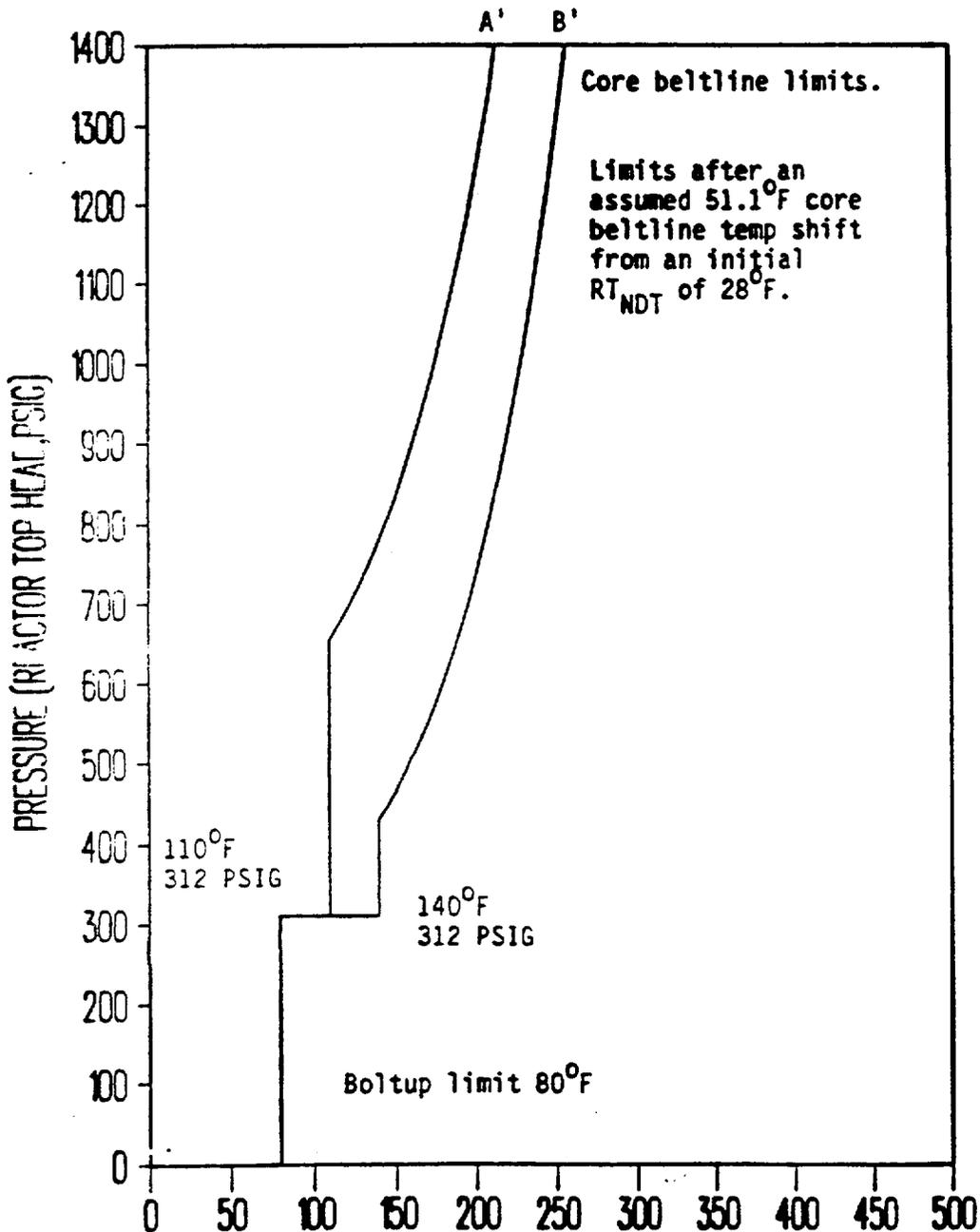


FIGURE 3.4.6.1.c

MINIMUM REACTOR VESSEL METAL TEMPERATURE  
TEMPERATURE F

BASES

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PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1 and 3.4.6.1.c 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.

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.

Access to permit inservice inspections of components of the reactor coolant system is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

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 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 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.