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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • Richland, Washington 99352-0968

April 20, 1999
GO2-99-076

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21
REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION 3.4.11**

Reference: Letter GO2-97-077, dated April 24, 1997, JV Parrish (SS) to NRC, "Submittal of WNP-2 RPV Surveillance Materials Testing and Analysis Report"

In accordance with the Code of Federal Regulations, Title 10, Parts 2.101, 50.59 and 50.90, the Washington Public Power Supply System hereby submits a request for amendment to the WNP-2 Operating License. Specifically, the Supply System is requesting a revision to Technical Specification 3.4.11 to replace the existing reactor pressure temperature limit curves by June 30, 2000. The proposed amendment reflects shifts in the pressure temperature limit curves based on 10CFR50, Appendix G, reactor vessel material testing and evaluation.

In the referenced report submittal letter we committed to submit new pressure temperature limit curves using the results of the analyses which are detailed in the "WNP-2 RPV Surveillance Materials Testing and Analysis Report" pursuant to 10CFR50, Appendix H. The report documented testing associated with a surveillance capsule that was withdrawn from the reactor vessel as required by 10CFR50, Appendix H. Charpy V-Notch impact testing and uniaxial tensile testing were performed to establish the properties of the irradiated surveillance materials.

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The Charpy data for the irradiated specimens were compared to the corresponding unirradiated specimen test data. The results of the testing reflected 30 ft-lb shifts and changes in uppershell energy of the base plate and the weld material. These results are well within the values predicted by Regulatory Guide 1.99, Revision 2. The adjusted reference temperature values and upper shelf energy of the reactor vessel beltline materials are expected to remain within the limits of 10 CFR 50, Appendix G, for at least 32 effective full power years of reactor operation.

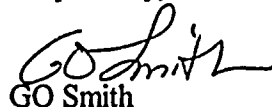
In addition to the results of the reactor pressure vessel surveillance material testing, the proposed changes to Technical Specification 3.4.11 reflect: 1) an alternate methodology for calculating changes in through-wall temperature; 2) incorporation of assessments of additional vessel sections; and 3) reflect current data from the ASME Code (Section XI, Appendix G, 1989).

Additional information has been attached to this letter to complete the Supply System's amendment request. Attachment 1 provides a detailed description and the basis for acceptability of the proposed changes. Attachment 2 describes an evaluation of the proposed changes in accordance with 10CFR50.92(c) and concludes they do not result in a significant hazards consideration. Attachment 3 provides the Environmental Assessment Applicability Review and notes that the proposed change meets the eligibility criteria for a categorical exclusion as set forth in 10CFR51.22(c)(9). Therefore, in accordance with 10CFR51.22(b), an environmental assessment of the change is not required. Attachment 4 summarizes the proposed changes and provides marked up pages of the Technical Specifications. Attachment 5 submits the typed Technical Specification pages as proposed by this amendment.

This request for an amendment has been reviewed and approved by the WNP-2 Plant Operations Committee and the Supply System Corporate Nuclear Safety Review Board. In accordance with 10CFR50.91, the state of Washington has been provided a copy of this letter.

Should you have any questions or desire additional information regarding this matter, please contact me or P. J. Inserra at (509) 377-4147.

Respectfully,



GO Smith

Vice President, Generation
Mail Drop 927M

Attachments

cc: EW Merschoff - NRC RIV
JS Cushing - NRC NRR
NRC Senior Resident Inspector - 927N

DJ Ross - EFSEC
PD Robinson - Winston & Strawn
DL Williams - BPA/1399

STATE OF WASHINGTON)
)
)
COUNTY OF BENTON)

Subject: Operating License NPF-21
Request for Amendment
Technical Specification 3.4.11

I, GO Smith, being duly sworn, subscribe to and say that I am the Vice President, Generation for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

DATE April 20, 1999

GO Smith
GO Smith
Vice President, Generation

On this date personally appeared before me GO Smith, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 20 day of April 1999

Ben Kask
Notary Public in and for the
STATE OF WASHINGTON

Residing at Kennewick

My Commission Expires 4/28/02



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Attachment 1

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Description of the Proposed Changes

Summary of Proposed Technical Specification Request

The Washington Public Power Supply System is requesting a revision to Technical Specification 3.4.11, "RCS Pressure and Temperature (PT) Limits," to revise the curves which set forth the pressure temperature limit lines. Specifically, existing Figures 3.4.11-1, 3.4.11-2 and 3.4.11-3 are being replaced with new pressure temperature limit curves.

This proposed revision changes the PT limits of 32 effective full power years (EFPY). The proposed PT limits were developed based on Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, Appendix G of the ASME Code, and Appendix G of 10 CFR 50.

The proposed change provides up-to-date PT limits for the operation of the reactor coolant system for heatup and cooldown during inservice leak and hydrostatic testing, non-nuclear heating and cooldown, and nuclear heating and cooldown.

Basis for Proposed Technical Specification Request

As required by 10 CFR 50, Appendix G, operating pressure and temperature limits are calculated, and implemented by plant procedure requirements, to ensure that fracture toughness requirements of the reactor pressure boundary are maintained. These requirements specify the vessel pressure temperature limits designed to prevent brittle fracture. Published methodology is used to derive the specific parameters of the curves. Furthermore, pursuant to 10 CFR 50, Appendix H, specimens of reactor vessel material are installed near the inside reactor vessel wall and are withdrawn in accordance with a schedule to provide data on the effects of radiation fluence and thermal environment on the vessel material.

The pressure temperature limits are adjusted, as necessary, to compensate for the shift in material transition temperature as indicated by tests on the withdrawn specimens. This ensures that the plant is operated in the ductility region of the vessel material.

During the Spring 1996 refueling outage, the surveillance capsule at the 300° azimuth location was removed at 7.2 EFPY (normalized full power of 3323 MWt) from the WNP-2 vessel. The capsule contained flux wires for neutron fluence measurement and Charpy and tensile test specimens for material property evaluations. Charpy V-Notch impact testing and uniaxial tensile testing were performed to establish the properties of the irradiated surveillance materials.

The irradiated Charpy data for the base plate, weld and heat affected zone specimens were compared to the corresponding unirradiated specimen test data to determine the shift in Charpy curves. Both the irradiated and unirradiated base plate data were of longitudinal Charpy orientation.

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The testing and analyses were performed by General Electric and the final report was submitted to the NRC in April 1997.¹ Although the current curves are bounding, we committed to provide to the NRC, within two years of the associated transmittal letter, new pressure temperature limit curves for WNP-2 using the results of the surveillance material testing and analysis report. For ease of reference, the conclusions made from the evaluation of surveillance test results and the related analyses are summarized as follows.

We concluded from the analyses that the 30 ft-lb shifts and changes in upper shelf energy were well within the values predicted by Regulatory Guide 1.99, Revision 2. From the surveillance test results it is clear that the 30 ft-lb shifts and the upper shelf energy of the base plate and the weld show little change after accumulating a peak irradiation fluence ($E > 1$ MeV) of $1.55E+17$ n/cm² (normalized full power of 3323 MWt because the reactor was at the uprated power level of 3486 MWt for Cycle 11 from June 6, 1995 to March 2, 1996).

The adjusted reference temperature and uppershell energy values are expected to remain within the limits of 10 CFR 50, Appendix G, for at least 32 EFPY of reactor operation. These values are $< 200^{\circ}\text{F}$ and > 50 ft-lb respectively.

These surveillance capsule test results were used to calculate the new Adjusted Reference Temperature (ART) for both the vessel beltline limiting plate material that is 9.625 inches minimum thickness, which includes a 0.125-inch thick clad, and the thinner plate of 6.625 inches minimum thickness, which also includes a 0.125-inch clad. The calculation for the thicker plate used 80 percent of the peak inner diameter fluence of $7.57E17$ n/cm² because only 14 inches of the plate extends into the active fuel area whereas peak fluence is found at 100 inches above the bottom of active fuel. Based on calculation, the new ARTs are as follows

- 9.625 inch section (9.5" + 0.125")

$$\text{ART} = 79.2 \text{ for } 1/4 \text{ t}$$

$$\text{ART} = 53.1 \text{ for } 3/4 \text{ t}$$

- 6.625 inch section (6.5" + 0.125") using the fluence of $7.57E17$ n/cm²

$$\text{ART} = 53.32 \text{ for } 1/4 \text{ t}$$

$$\text{ART} = 31.06 \text{ for } 3/4 \text{ t}$$

¹ Letter GO2-97-077, dated April 24, 1997, JV Parrish (SS) to NRC, "Submittal of WNP-2 RPV Surveillance Materials Testing and Analysis Report"

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An alternate methodology for calculating changes in through wall temperature was also used in the development of the updated pressure temperature curves. Previously approved Supply System submittals to the NRC used actual readings from external vessel metal thermocouples and internal vessel water temperatures to develop a linear equation for calculating ΔT through wall. The ΔT value is used to calculate the K_{It} (thermal stress intensity factor) value by using the curves in Appendix G of the ASME code.

The following heat transfer equation, based on a published methodology², was used instead of actual plant data:

$$\Delta T = \frac{t^2}{2 * \beta} * 100 \text{ degrees F/ hour}$$

ΔT = Differential temperature through wall,

t = vessel thickness

β = Thermal Diffusivity at temperature

The new curves also reflect the incorporation of a thicker section of the vessel (9.625 inches). Both the 9.625 inch thick section and the 6.625 inch thick section of the reactor vessel beltline region were evaluated. The 9.625 inch section was determined to be the limiting location based on the Regulatory Guide 1.99, Revision 2, evaluation and the referenced surveillance materials testing and analysis report.

The initial RT_{NDT} for the limiting plate (9.625 inches) is 28°F. For the 6.625 inch plate, the initial RT_{NDT} is -8°F. The initial RT_{NDT} for the reactor pressure vessel flange is 20°F.

In accordance with 10CFR50, Appendix G, the most recent version of the ASME Code was used in performing the analysis (Section XI, Appendix G, 1989).

² "Heat, Mass, and Momentum Transfer", by Warren M. Rohsenow, Professor of Mechanical Engineering Massachusetts Institute of Technology, and Harry Y. Choi, Associate Professor of Mechanical Engineering Tufts University, published by Prentice-Hall, Inc. (see page 96, equation 6.4)

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Evaluation of Significant Hazards Considerations

Summary of Proposed Change

The Washington Public Power Supply System is requesting a revision to Technical Specification 3.4.11, "RCS Pressure and Temperature (PT) Limits," to revise the curves which set forth the pressure temperature limit lines. Specifically, existing Figures 3.4-11-1, 3.4.11-2 and 3.4.11-3 are being replaced with new pressure temperature curves.

This proposed revision changes the PT limits of 32 effective full power years (EFPY). The proposed PT limits were developed based on Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, Appendix G of the ASME Code, and Appendix G of 10 CFR 50.

The proposed change provides up-to-date PT limits for the operation of the reactor coolant system during heatup, cooldown, criticality and hydrostatic testing. The proposed amendment reflects shifts in the pressure temperature limit curves to values which were calculated using a published methodology that was discussed with the NRC rather than previously approved plant data for through wall ΔT .

No Significant Hazards Consideration Determination

The Washington Public Power Supply System has evaluated the proposed change to the Technical Specifications using the criteria established in 10 CFR 50.92(c) and has determined that it does not represent a significant hazards consideration as described below:

- The operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The pressure temperature shift is well within the operating margins of plant equipment. Using the new non-nuclear and nuclear heating and cooldown curves, higher temperature values for corresponding pressures at temperatures which are closest to RT_{NDT} , further reduce the potential for brittle fracture.

The proposed 32 EFPY curves were developed using methodology that is consistent with the guidance provided in Regulatory Guide 1.99, Revision 2, Appendix G of the ASME Code and Appendix G of 10 CFR 50. This methodology is recognized by the NRC and industry as providing acceptable margin.

Therefore, operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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- The operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change has no impact on previously analyzed accidents or transients. The proposed change does not introduce any credible mechanisms for unacceptable radiation release nor does it require physical modification to the plant. The 32 EFPY curves are calculated using a published methodology that was discussed with the NRC.

The proposed change is also within any upper bound limit. The only impact on plant operation is that the plant will be operated with new pressure temperature limits derived from the proposed alternative calculational methodology in place of the previously approved model based on actual plant data.

Therefore, the operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

- The operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

The results of the testing reflected 30 ft-lb shifts and changes in uppershell energy of the base plate and the weld material. However, the results are well within the values predicted by Regulatory Guide 1.99, Revision 2. Furthermore, the adjusted reference temperature values and upper shelf energy of the reactor vessel beltline materials are expected to remain within the limits of 10 CFR 50, Appendix G, for at least 32 effective full power years of reactor operation.

For the non-nuclear and nuclear heating and cooldown curves (with a calculated through wall ΔT), lower temperatures which are closest to RT_{NDT} , have an increased margin of safety due to the higher required temperature values for a given pressure than is required by the current curve calculation methodology. Thus additional margin to brittle fracture is achieved for non-nuclear and nuclear heating.

Therefore, operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.



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Environmental Assessment Applicability Review

The Washington Public Power Supply System has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21.

The proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9) because the change requested does not pose a significant hazards consideration nor does it involve an increase in the amounts, or a change in the types, of any effluent that may be released off-site.

Furthermore, this proposed request does not involve an increase in individual or cumulative occupational exposure.