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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated March 31, 1989, Northern States Power Company (the licensee) requested an amendment to the Technical Specifications appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment would:

(1) revise the reactor vessel pressure vs. temperature (P/T) curves for consistency with Revision 2 of Regulatory Guide (RG) 1.99;

(2) add requirements for augmented inservice inspection (ISI) of piping susceptible to intergranular stress corrosion cracking (IGSCC); and

(3) revise the requirements for the periodic Type A containment integrated leak rate test (CILRT) to permit the use of the mass point test method.

A discussion of the proposed changes and the NRC staff's evaluation and findings relative to each are addressed in Section 2 of this Safety Evaluation.

2.0 DISCUSSION AND EVALUATION

2.1.1 Revised Pressure/Temperature Limits

Pressure/temperature limits are included in facility technical specifications for the purpose of precluding conditions conducive to brittle fracture of reactor coolant system (RCS) materials. The fracture toughness of RCS materials is a function of the material chemistry and decreases as irradiation accumulates. Revision 2 of Regulatory Guide 1.99, issued May 1988, specifies a more accurate and more conservative means than previously used for predicting the effects of irradiation damage on RCS materials. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations" requested licensees of operating reactors to reanalyze their P/T limits using the revised criteria. In response to Generic Letter 88-11, the licensee applied to revise the P/T limits in the Monticello Nuclear Generating Plant Technical Specifications, Section 3.6. The NRC staff evaluated the proposed changes using the following NRC regulations and guidance: Appendices

G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11. Appendices G and H of 10 CFR Part 50 define specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. SRP Section 5.3.2 describes an acceptable method for constructing the P/T limits. Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline. These tests define the extent of vessel embrittlement at the time of surveillance specimen capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE).

All surveillance capsules contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal. The licensee has removed one surveillance capsule. The results from capsule 117C 3991 G-1 were reported in a Battelle-Columbus Laboratories Report BCL-585-84-2, Revision 1.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 140 degrees F. for the limiting plate material (I-15) at 32 effective full power years (EFPY) at 1/4T (T=reactor vessel thickness) in the Monticello beltline. The ART was calculated using Section 1 of RG 1.99, Rev. 2, because only one surveillance capsule has been withdrawn from the Monticello reactor pressure vessel. The NRC staff performed a similar calculation and verified the licensee's ART to be conservative (see Table 1). Substituting the ART of 140 degrees F. into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 degrees F for normal operation and by 90 degrees F. for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 10 degrees F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Based on data from surveillance capsule 117C 3991 G-1 withdrawn at 7.63 EFPY, the lowest measured irradiated Charpy USE of the material tested is 109 ft-lb or the beltline plate I-15. To estimate the USE at 32 EFPY, the staff subtracted the fluence to which the surveillance capsule was exposed, 2.9E17 n/cm^2, from the fluence at 32 EFPY, 5.1E18 n/cm^2. The staff then referred to Figure 2 of RG 1.99, Rev. 2, for the predicted decrease in USE and calculated that the USE, at 32 EFPY, for the I-15 beltline plate material would be 85 ft-lb. This value is greater than 50 ft-lb and, therefore, is acceptable.

In addition to revising the P/T limit curves to meet RG 1.99, Rev. 2 criteria, the licensee also proposes that, during pressure tests, the reactor vessel bottom head temperature be monitored separately from the beltline region. This would facilitate pressure testing by reducing the amount of non-nuclear heatup required for pressure testing. In order for this to be acceptable, it is necessary that operators have the capability to monitor the beltline region temperature separately. The licensee has advised the staff that this capability is provided by redundant resistance temperature detectors (RTDs).

The staff concludes that the proposed P/T limits for the RCS for heatup, cooldown, leak test, and criticality are valid through 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Monticello Technical Specifications.

2.1.2 Augmented Inservice Inspection for IGSCC

Generic Letter 88-01 requested licensees to describe their plans for replacement, inspection, repair and leakage detection of piping susceptible to intergranular stress corrosion cracking (ISGCC). Among the items specified to be included in the licensees' responses to Generic Letter 88-01 is an application to change the Technical Specifications to include a requirement that the Inservice Inspection Program for piping covered by the scope of the Generic Letter be in conformance with the staff positions on schedule, methods and personnel, and sample expansion. The licensee's March 31, 1989 application proposed to invoke NUREG-0313, Revision 2, as a reference for the augmented ISI requirements. In a letter dated September 27, 1989, the licensee revised the application to cite Generic Letter 88-01 as the reference for the new augmented ISI requirements. The use of Generic letter 88-01 as the reference is consistent with the Model Technical Specifications provided in Generic Letter 88-01 and is acceptable.

(Note: The NRC staff will issue a separate evaluation of the licensee's complete response to Generic Letter 88-01 in the near future. This change serves only to implement that portion of the generic letter relating to Technical Specification requirements for augmented ISI.)

2.1.3 CILRT Test Method

The proposed amendment would change Technical Specification 4.7.A.2.b to delete the requirement that the test method for the CILRT be in accordance with the 1972 revision of ANSI N45.4. ANSI N45.4-1972 specifies use of either the total time or point-to-point method of containment integrated leak rate testing. An exemption was issued to the licensee on October 21, 1988 to permit use of the superior "mass point" method pending revision of 10 CFR Part 50, Appendix J.

Appendix J has since been revised and now permits use of the mass point method (when used for a period of at least 24 hours) as an alternative to the total time and point-to-point methods specified by ANSI N45.4-1972. The amendment would bring the Technical Specifications into consistency with the revised 10 CFR Part 50, Appendix J and is, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on September 28, 1989. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: November 2, 1989

TABLE 1

The NRC Staff Calculated Adjusted Reference Temperature for the Limiting Reactor Beltline Material at Monticello Nuclear Generating Plant

Limiting Beltline Material		Plate material
Code No.		I-15
Copper Content		0.17%
Nickel Content		0.58%
Initial Reference Temperature		14 degrees F
Reactor Vessel Beltline Thickness (in.)		5.06
Chemistry Factor Used in Calculation		125.3
Neutron Fluence (n/cm^2) at 32 EFPY		
	At ID	0.51E19
	At 1/4T	0.38E19
	At 3/4T	0.21E19
Fluence Factor		
	At ID	0.812
	At 1/4T	0.732
	At 3/4T	0.575
Margin		34 at 1/4T
ART at 1/4T at 32 EFPY:		140 degrees F.