Docket No. 50-482

February 10, 1994

Mr. Neil S. Carns
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

Dear Mr. Carns:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M86642)

The Commission has issued the enclosed Amendment No. 71 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications in response to your application dated May 27, 1993.

The amendment revises the heatup, cooldown, and cold overpressure mitigation system power-operated relief valve setpoints pressure/temperature limits.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original Signed By

William D. Reckley, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.71 to NPF-42
2. Safety Evaluation

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. NPF-42

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Wolf Creek Generating Station (the facility) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated May 27, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-42 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 71 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C. Black

Suzanne C. Black, Director
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 10, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 71

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 60°F in any 1-hour period for indicated T_{avg} less than or equal to 200°F,

b. A maximum heatup of 100°F in any 1-hour period for indicated T_{avg} greater than 200°F,

c. A maximum cooldown of 100°F in any 1-hour period, and

d. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, and 3.4-4.

MATERIAL PROPERTY BASIS

Controlling Material:	RV Lower Shell
Copper Content:	0.07 Weight %
Nickel Content:	0.62 Weight %
Initial RT_{NDT}:	40 °F
Limiting ART after 13.6 EFPY:	1/4T, 89 °F
	3/4T, 79 °F

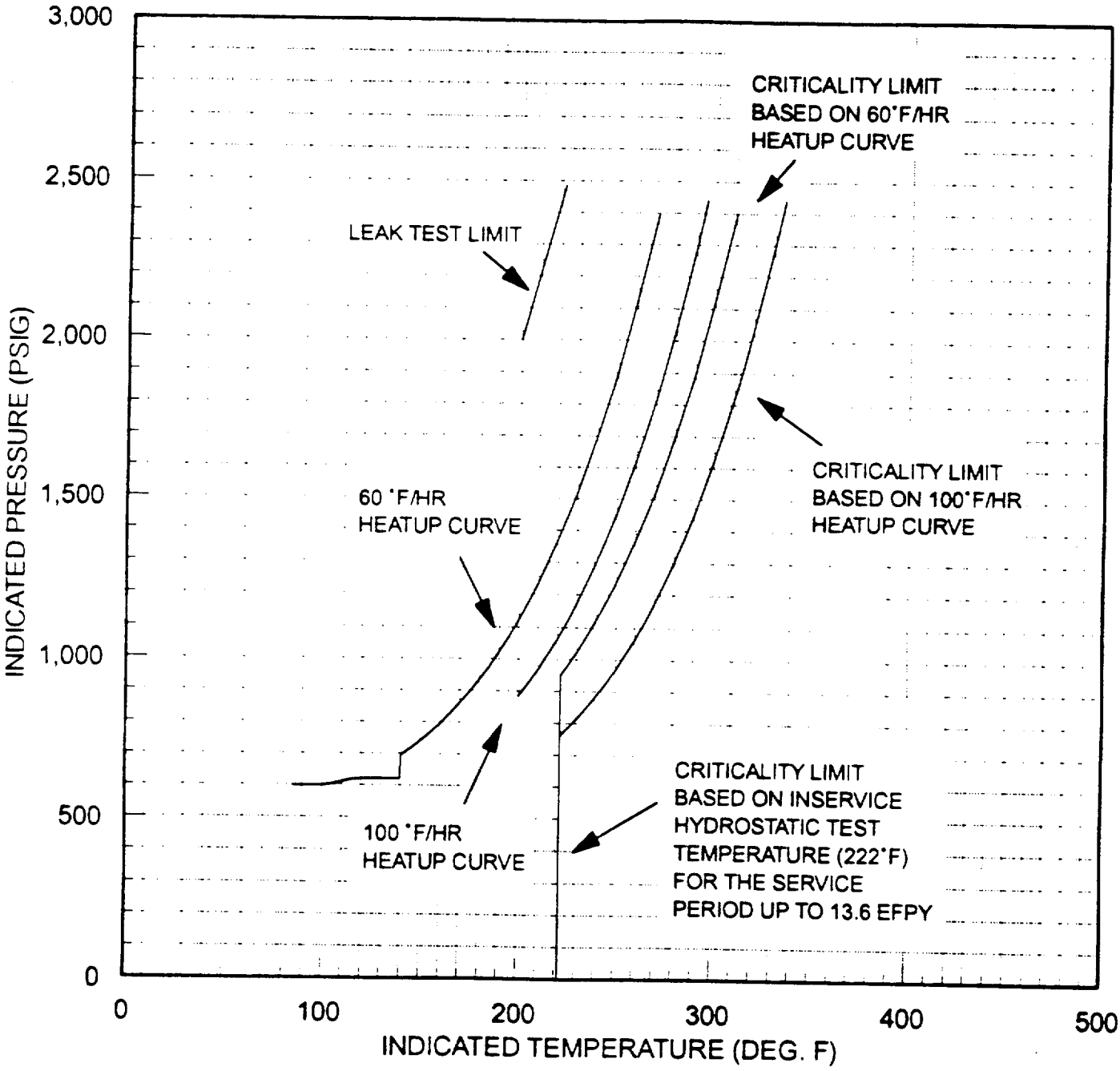

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 13.6 EFPY

MATERIAL PROPERTY BASIS

Controlling Material:	RV Lower Shell
Copper Content:	0.07 Weight %
Nickel Content:	0.62 Weight %
Initial RT_{NDT}:	40 °F
Limiting ART after 13.6 EFPY:	1/4T, 89 °F
	3/4T, 79 °F

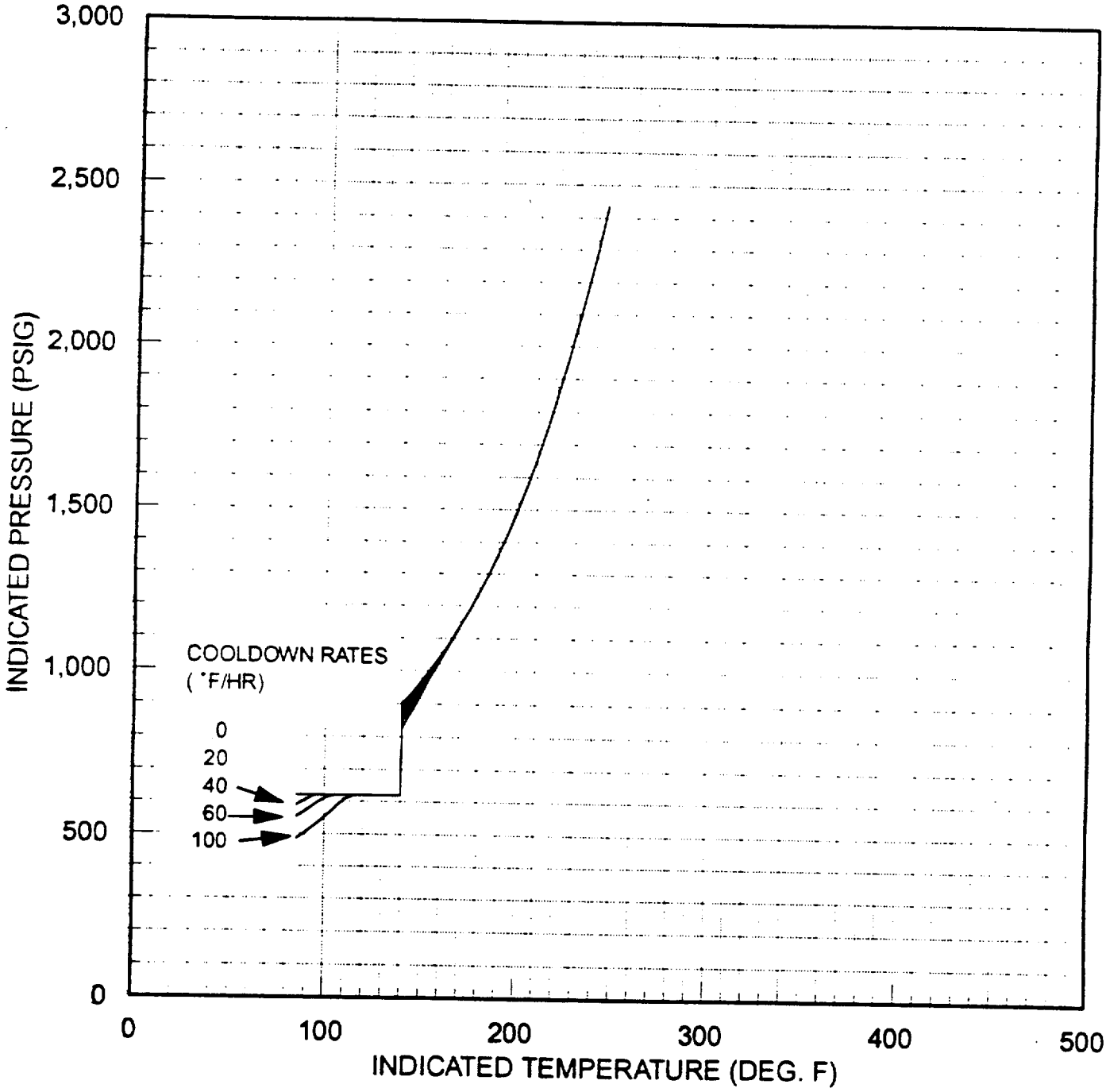

FIGURE 3.4-3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 13.6 EFPY

TABLE 4.4-5

(Deleted)

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 8708B:

   By verifying at least once per 72 hours that RHR RCS Suction Isolation Valves (RRSIVs) 8701B and 8702B are open.

b. For RHR suction relief valve 8708A:

   By verifying at least once per 72 hours that RRSIVs 8701A and 8702A are open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 With the RCS vented, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

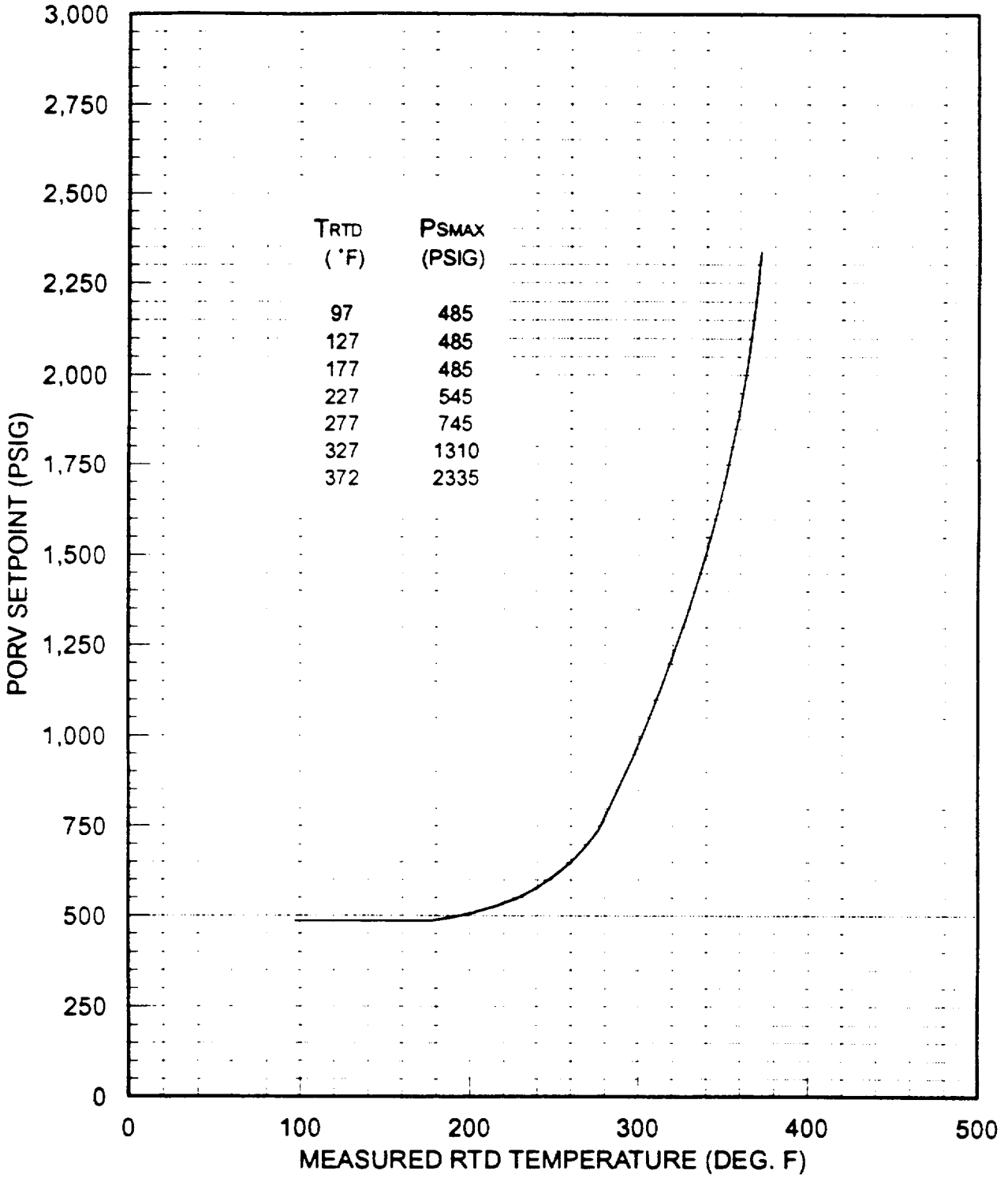

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT

FOR THE COLD OVERPRESSURE MITIGATION SYSTEM

PRESSURE/TEMPERATURE LIMITS (Continued)

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below.

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 583°F.

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 13.6 effective full power years (EFPY) of service life. The 13.6 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 13.6 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

PRESSURE/TEMPERATURE LIMITS (Continued)

Values of ΔRT_{NDT} determined in this manner may be used until the results of the next scheduled capsule from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

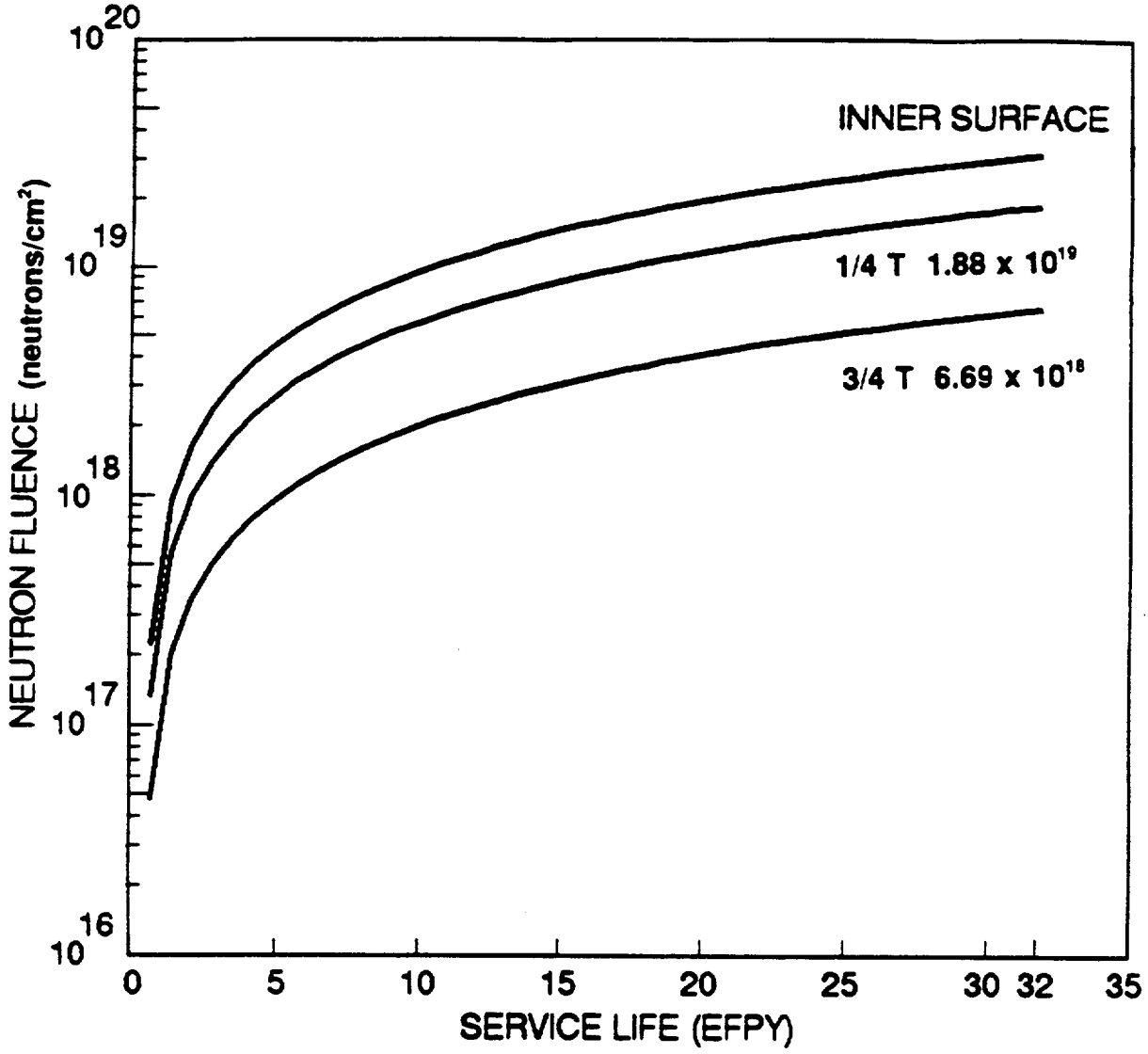

FIGURE B 3/4.4-1
FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF EFFECTIVE FULL POWER LIFE

COOLDOWN (Continued)

during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated May 27, 1993, Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station. The proposed changes would revise the heatup, cooldown, and cold overpressure mitigation system power-operated relief valve setpoint pressure/temperature limits. The revised limits reflect the analysis of the most recently withdrawn surveillance capsule associated with the reactor vessel radiation surveillance program (10 CFR Part 50, Appendix H). The revised limits bound operation through 13.6 Effective Full Power Years (EFPY).

2.0 BACKGROUND

To evaluate the P-T limits, the staff uses the following NRC regulations and guidance: 10 CFR 50.36(c)(2); Appendices G and H of 10 CFR Part 50; Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations;" Regulatory Guide (RG) 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials;" and Standard Review Plan (SRP) Section 5.3.2, "Pressure Temperature Limits."

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide technical specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the technical specifications. The P-T limits are among the limiting conditions of operation in the technical specifications for all commercial nuclear plants in the United States.

Appendix G to 10 CFR Part 50 requires that "...pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code..." Appendix G also imposes requirements on the minimum temperature for criticality, the closure head flange, and hydrostatic pressure tests or leak tests.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to monitor embrittlement of reactor vessel materials. The program includes capsules that contain test specimens made from plate, weld, and heat-

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affected-zone (HAZ) materials of the reactor beltline. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and be removed from the reactor vessel periodically for testing. The test results may be used in calculating P-T limits.

Generic Letter 88-11 requires that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the adjusted reference temperature (ART) as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

SRP 5.3.2 describes a step-by-step calculation of the P-T limits that is based on methodology specified in Appendix G to the ASME Code, Section III.

3.0 EVALUATION

The licensee calculated the ART for each beltline material in the Wolf Creek reactor vessel in accordance with RG 1.99, Rev. 2. The licensee determined that, at 13.6 EFPY, lower shell plate, R2508-3, is the limiting material. The chemistry for the plate is 0.07 percent copper and 0.62 percent nickel with an initial RT_{ndt} of 40°F. For plate R-2508-3, the licensee calculated the ARTs of 89°F at the 1/4T location and 79°F at the 3/4T location based on Regulatory Position C.2 of RG 1.99. The staff has identified the same plate as the limiting material and confirmed the licensee's ARTs to be correct. Substituting the ARTs of 89°F and 79°F into equations in SRP 5.3.2, the staff verified that the proposed P-T limits for heatup, cooldown, criticality, and leak test meet the requirements in Paragraphs IV.A.2 & IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent 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°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature, provided by the licensee, of 20°F, the staff has determined that the proposed P-T limits have included this requirement.

The licensee has removed surveillance capsules U and Y from the Wolf Creek reactor vessel and has performed required tests per ASTM E-185. The test results of both capsules are published in reports by Westinghouse (Refs. 1 and 2). The licensee used the surveillance test data from both capsules in the P-T limits calculation. The staff has determined that the licensee followed the recommended procedure in RG 1.99, Rev. 2.

The staff concludes that the proposed P-T limits for heatup, cooldown, leak test, and criticality are valid through 13.6 EFPY because the limits conform to the requirements of Appendix G of 10 CFR Part 50 and Generic Letter 88-11. Hence, the proposed P-T limits may be incorporated in the Wolf Creek Technical Specifications.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 36449). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. "Analysis of Capsule U from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Corporation, WCAP-11553, August 1987.

2. "Analysis of Capsule Y from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Corporation, WCAP-13365, April 1993.

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