September 27, 1990

Docket No. 50-482

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Mr. Bart D. Withers
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

Dear Mr. Withers:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 40 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. 68507)

The Commission has issued the enclosed Amendment No. 40 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 June 20, 1988, (ET 88-0084) and as supplemented on May 22, (WM 90-0059) June 8, (ET 90-0097) and August 1, 1990 (ET-90-0121).

The amendment revises the heatup, cooldown and cold overpressure mitigation system power-operated relief valve setpoint pressure/temperature (P/T) limits as required by 10 CFR Part 50, Appendix H, and Technical Specification 4.4.9.1.2. The P/T limits were developed based on the data of a surveillance capsule which was withdrawn from the Wolf Creek reactor during the first refueling outage. The P/T limits are based on the irradiation damage prediction methods of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," and are applicable for a period up to seven effective full power years.

A copy of our related Safety Evaluation and Notice of Issuance are enclosed.

Sincerely,

Original Signed By:

Douglas V. Pickett, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 40 to NPF-42
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

| OFC | PDIV-2/LA | PDIV-2/PM | OGC | PDIV-2/D | | | |
|---|---|---|---|---|---|---|---|
| NAME | EPeyton | DPickett: | PJehle | CGrimes | | | |
| DATE | 08/5/90 | 09/12/90 | 09/17/90 | 09/27/90 | | | |

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

# WOLF CREEK NUCLEAR OPERATING CORPORATION

## WOLF CREEK GENERATING STATION

## DOCKET NO. 50-482

## AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 40
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 June 20, 1988, and as supplemented on May 22, June 8, and August 1, 1990, 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.

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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 40 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Christopher I. Grimes

Christopher I. Grimes, Director
Project Directorate IV-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 27, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 40

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.

| REMOVE | INSERT |
|---|---|
| VIII | VIII |
| XVI | XVI |
| 3/4 4-29 | 3/4 4-29 |
| 3/4 4-30 | 3/4 4-30 |
| 3/4 4-31 | 3/4 4-31 |
| 3/4 4-36 | 3/4 4-36 |
| B 3/4 4-7 | B 3/4 4-7 |
| B 3/4 4-8 | B 3/4 4-8 |
| B 3/4 4-11 | B 3/4 4-11 |
| B 3/4 4-12 | B 3/4 4-12 |
| B 3/4 4-13 | B 3/4 4-13 |
| B 3/4 4-14 | B 3/4 4-14 |
| B 3/4 4-15 | B 3/4 4-15 |
| B 3/4 4-16 | -- |

| SECTION | | PAGE |
|---|---|---|

INSTRUMENTATION (Continued)

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 in accordance with the schedule in Table 4.4-5. 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 Deg F

$RT_{NDT}$ After 7 EFPY: 1/4T: 108.0 Deg F

3/4T: 99.47 Deg F

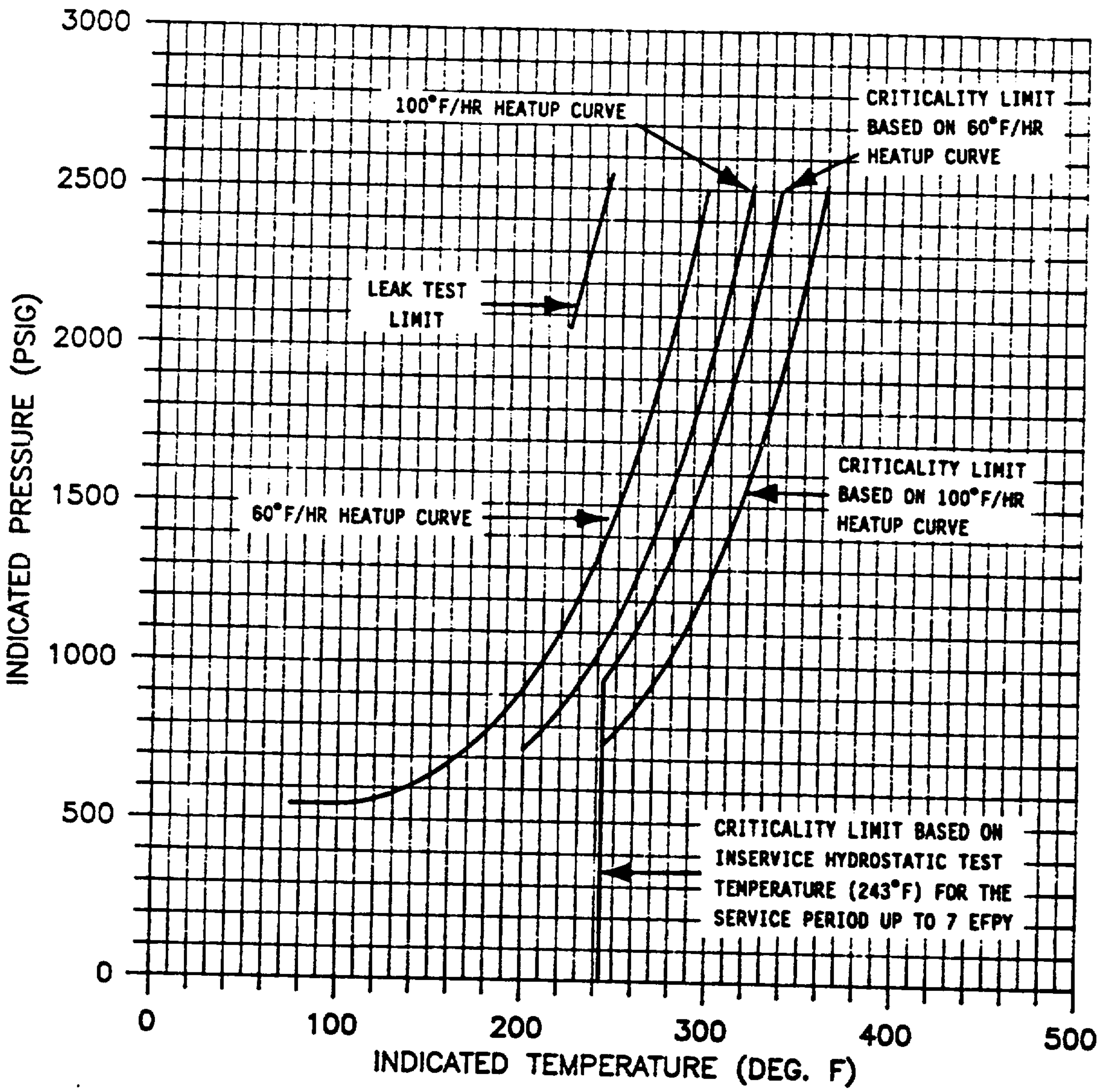


FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 7 EFPY

MATERIAL PROPERTY BASIS

Controlling Material: RV Lower Shell
Copper Content: 0.07 Weight %
Nickel Content: 0.62 Weight %
Initial $RT_{NDT}$ : 40 Deg F
$RT_{NDT}$ After 7 EFPY: 1/4T: 108.0 Deg F
3/4T: 99.47 Deg F

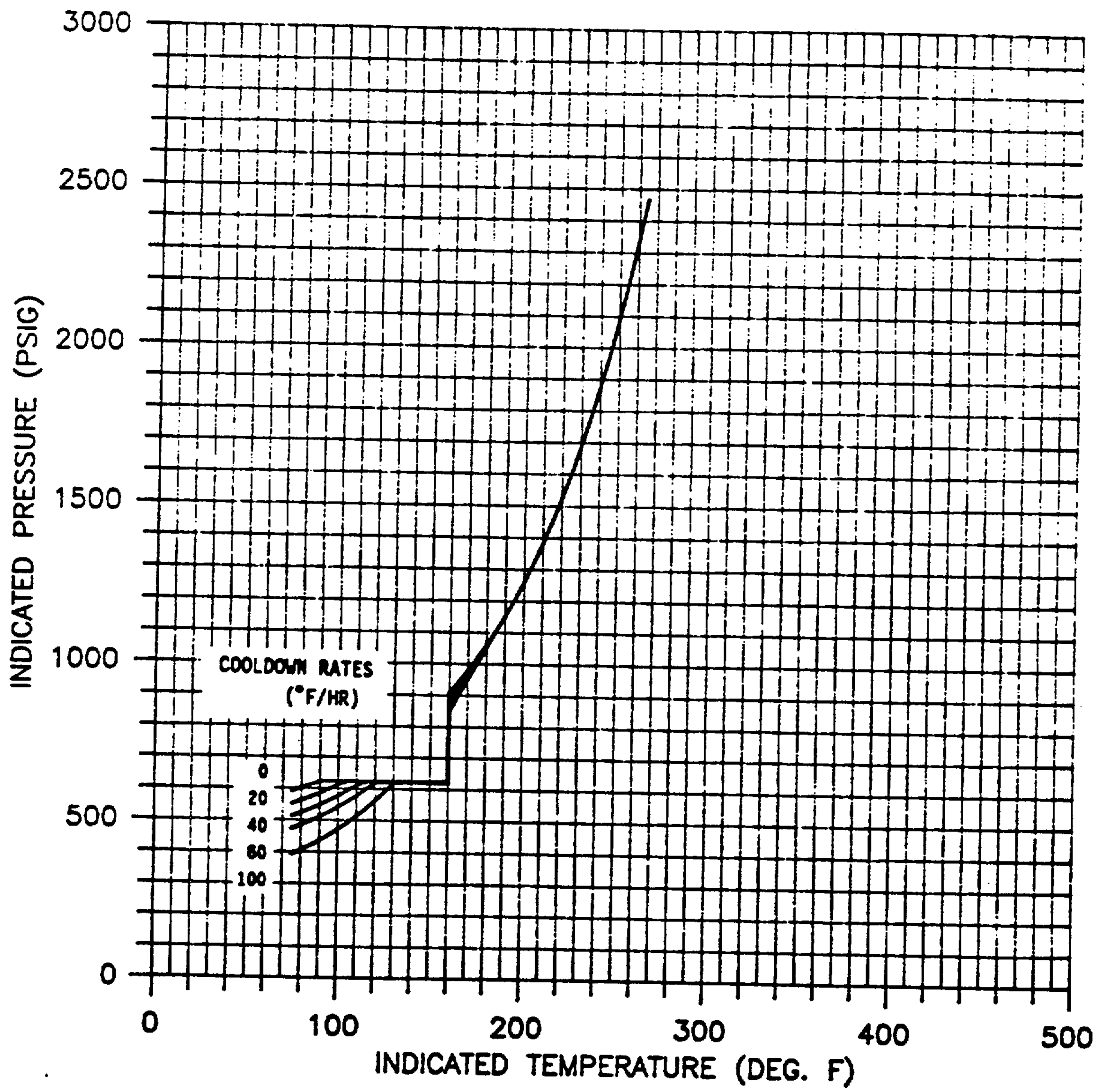


FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 7 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

| CAPSULE NUMBER | VESSEL LOCATION | LEAD FACTOR | WITHDRAWAL TIME (EFPY) |
|---|---|---|---|
| U | 58.5° | 4.00 | 1st Refueling |
| Y | 241° | 3.69 | 5 |
| V | 61° | 3.69 | 9 |
| X | 238.5° | 4.00 | 15 |
| W | 121.5° | 4.00 | Standby |
| Z | 301.5° | 4.00 | Standby |

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:

   1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve (RRSIV) 8701B is open with power to the valve operator removed, and

   2) By verifying at least once per 12 hours that RRSIV 8702B is open.

b. For RHR suction relief valve 8708A:

   1) By verifying at least once per 31 days that RRSIV 8702A is open with power to the valve operator removed, and

   2) By verifying at least once per 12 hours that RRSIV 8701A is open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

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*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

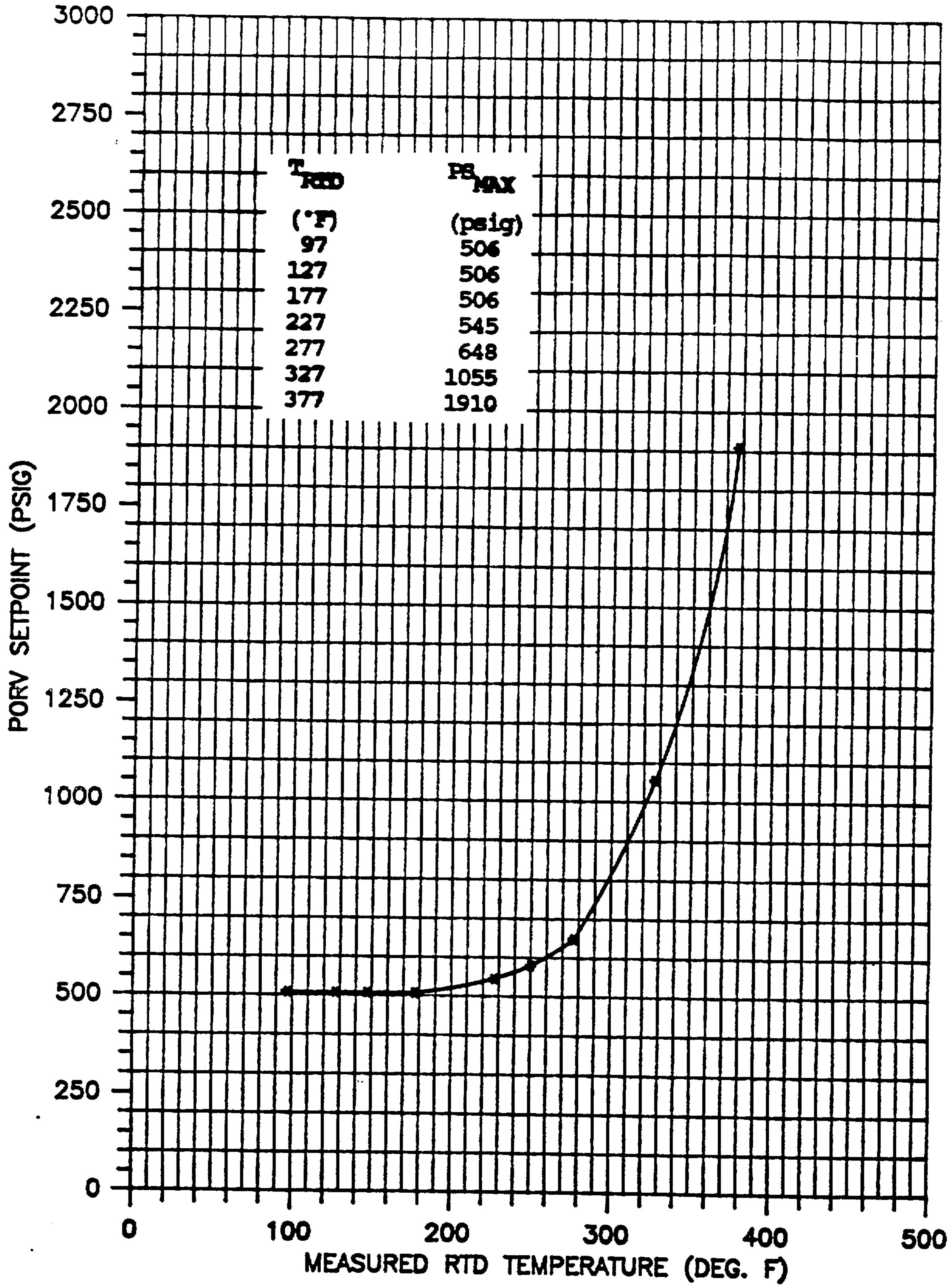


FIGURE 3.4-4

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 7 effective full power years (EFPY) of service life. The 7 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 $\Delta RT_{NDT}$ computed by either 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 7 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

PRESSURE/TEMPERATURE LIMITS (Continued)

Values of $\Delta RT_{NDT}$ 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 surveillance specimen withdrawal schedule is shown in Table 4.4-5. 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 $\Delta RT_{NDT}$ determined from the surveillance capsule exceeds the calculated $\Delta RT_{NDT}$ 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, $\Delta RT_{NDT}$, 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:

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

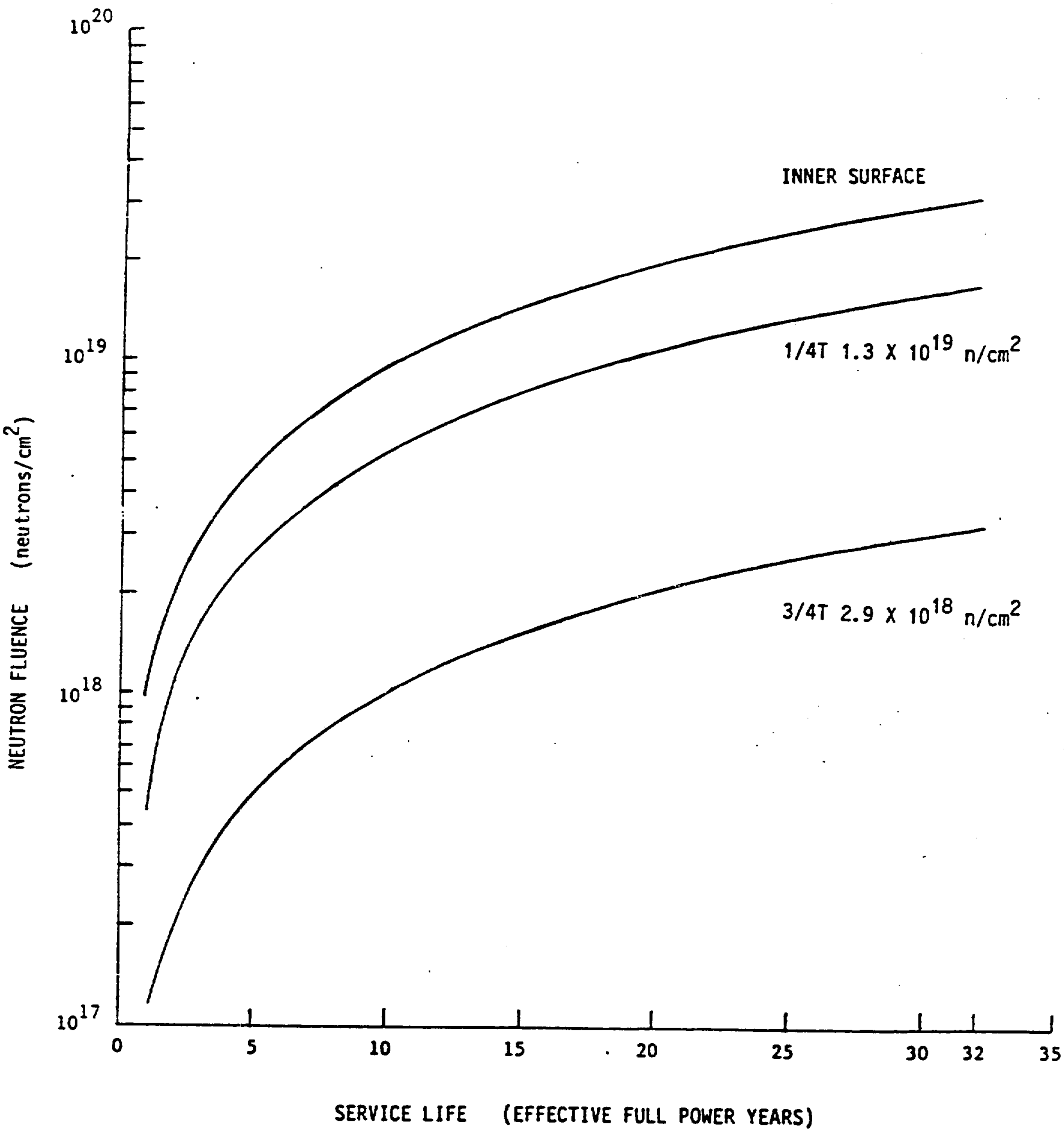


FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or two RHR suction relief valves, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 368°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water solid RCS.

RHR RCS suction isolation valves 8701A and B are interlocked with an "A" train wide range pressure transmitter and valves 8702A and B are interlocked with a "B" train wide range pressure transmitter. Removing power from valves 8701B and 8702A, prevents a single failure from inadvertently isolating both RHR suction relief valves while maintaining RHR isolation capability for both RHR flow paths.

In addition to opening RCS vents to meet the requirement of Specification 3.4.9.3c., it is acceptable to remove a pressurizer Code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS.

COLD OVERPRESSURE

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for: (1) process and instrumentation uncertainties; and (2) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of both Safety Injection pumps and all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary coolant temperature is more than 50°F above reactor coolant temperature. Exceptions to these requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centrifugal charging pump OPERABLE and no Safety Injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration and the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and Safety Injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic Safety Injection actuation signals except Containment Pressure - High are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of Safety Injection (one centrifugal charging pump, and one Safety Injection pump). For temperatures above 325°F, an overpressure event occurring as a result of starting two pumps can be successfully mitigated by operation of both PORVs without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

Although COMS is required to be OPERABLE when RCS temperature is less than 368°F, operation with all centrifugal charging pumps and both Safety Injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F. Should an inadvertent Safety Injection occur above 350°F, a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F two RCPs and all pressure safety valves are required to be OPERABLE. Operation of an

COLD OVERPRESSURE (Continued)

RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering instrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in accordance with Appendix G limitations. Should an overpressure event occur in these conditions, the presurizer safety valves provide acceptable and redundant overpressure protection.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are 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 Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

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, 1974 Edition and Addenda through Summer 1975.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 40 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 June 20, 1988, and as supplemented by letters dated May 22, June 8, and August 1, 1990, the Wolf Creek Nuclear Operating Corporation (the licensee) proposed changes to the Technical Specifications (TS) (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station regarding heatup, cooldown and cold overpressure mitigation system power-operated relief valve (PORV) setpoint pressure/temperature (P/T) limits as required by 10 CFR Part 50, Appendix H. The current P/T limits were developed during the initial licensing process using results of the theoretical neutron transport analysis. The proposed P/T limits were developed based on the data obtained from the actual surveillance capsule which was withdrawn from the Wolf Creek Generating Station (WCGS) reactor during the first refueling outage. Based on the revised P/T limits, new PORV setpoints were derived for the cold overpressure mitigation system. The new data for both the pressure/temperature limits and the PORV setpoints are applicable for seven effective full power years (EFPY). As of September 6, 1990, the Wolf Creek facility had accumulated 3.86 EFPY of operation.

## 2.0 DISCUSSION

### 2.1 Pressure/Temperature (P/T) Limits

#### (a) Background

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H to 10 CFR Part 50; the ASTM Standards and ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P/T limits are among the limiting conditions of operation in the TS for all commercial nuclear plants in the U.S. Appendices G and H to 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance. These must be considered in setting P/T limits. An acceptable method in constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, to test the beltline materials in the surveillance capsules in accordance with Appendix H to 10 CFR Part 50. Appendix H, in turn, refers to the ASTM Standards. These tests define the condition of vessel embrittlement at the time of capsule withdrawal in terms of the increase in the reference temperature ($RT_{NDT}$). Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted $RT_{NDT}$ and upper shelf energy (USE). A method that is acceptable to the NRC staff is described in Regulatory Guide 1.99, Rev. 2.

Appendix H to 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, requires that the capsules be installed in the vessel before startup and that they contain test specimens that are made from plate, weld, and heat-affected-zone materials of the reactor beltline.

(b) Evaluation

The licensee removed the first surveillance capsule, U, from the reactor vessel after 1.08 EFPY of operation. The surveillance specimens in capsule U were tested and analyzed by Westinghouse Corporation. The results were published in the Westinghouse report, "Analysis of Capsule U From the Wolf Creek Nuclear Operating Corporation - Wolf Creek Reactor Vessel Radiation Surveillance Program," WCAP-11553. In accordance with Appendix H to 10 CFR Part 50, the licensee submitted the report on November 4, 1987, a year after the capsule withdrawal.

Capsule U contained Charpy impact specimens and tensile specimens that were made from base metal, weld metal, and heat affected zone (HAZ) metal. The base metal specimens were made from the lower shell plate, R2508-3; the weld metal specimens were made from the girth weld located between the intermediate shell plate and lower shell plate. The weld was made by the submerged arc weld method using 3/16-inch Mil B-4 filler wire, heat number 90146 and the weld flux was Linde 124, lot number 108. The HAZ specimens were made from the HAZ metal of plate R2508-3. The specimens were tested in accordance with Appendices G and H to 10 CFR Part 50 and ASTM E185-82. Specifically, the Charpy impact tests were performed per ASTM E23-82 and tension tests were performed per ASTM E8-83 and E21-79. The Charpy impact tests of the R2508-3 plate showed that as a result of neutron fluence 3.39E18 $n/cm^2$, the increase in $RT_{NDT}$ was 30°F and the decrease in the USE was 3 ft-lb. The tests showed that the R2508-3 plate has the highest increase in $RT_{NDT}$; therefore, it was designated as the limiting (controlling) material.

The licensee used the method in Regulatory Guide 1.99, Rev. 2, to calculate an adjusted $RT_{NDT}$ of 108°F for the R2508-3 plate at 7 EFPY and 1/4T location (T is the reactor vessel thickness). The staff performed a similar calculation and verified the licensee's $RT_{NDT}$ value to be correct (see Table 1). Substituting the $RT_{NDT}$ of 108°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, criticality, and hydrotest are acceptable.

TABLE 1

The NRC Staff Calculated Adjusted Reference Temperature For The Limiting Reactor Beltline Material at Wolf Creek Unit 1.

| | |
|---|---|
| Material: | lower shell plate A553B, Class 1 |
| Code No.: | R2508-3 |
| Copper Content: | 0.07% |
| Nickel Content: | 0.62% |
| Initial $RT_{NDT}$: | 40°F |
| Neutron Fluence $n/cm^2$ | |
| at 7 EFPY | 6.825E18 |
| at 32 EFPY | 3.12E19 |
| Increase in $RT_{NDT}$ at 7 EFPY | 33°F |
| Adjusted $RT_{NDT}$ at 7 EFPY | 107°F (Licensee Calculated 108°F) |

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes P/T limits on the reactor vessel closure flanges. Section IV.2 of Appendix G states that when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the $RT_{NDT}$ 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 $RT_{NDT}$ of 20°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 the predicted USE at end-of-life to be above 50 ft-lb. At 1.08 EFPY, the measured USE is 92 ft-lb for the intermediate to low shell weld metal. This is an 8 percent reduction from the unirradiated value of 100 ft-lb and is limiting. Using the method in Regulatory Guide 1.99, Rev. 2, the staff predicted that the USE of the weld metal at end-of-life will still be above 50 ft-lb and thus the projected USE of the limiting metal at end-of-life satisfies Section IV.B of Appendix G.

The staff concludes that the proposed WCGS P/T limits on the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 7 EFPY because the limits conform to requirements of Appendices G and H to 10 CFR Part 50. The licensee also conforms to Generic Letter 88-11 because it used the method in Regulatory Guide 1.99, Rev. 2, to calculate the reference temperature, $RT_{NDT}$. The proposed P/T limits may be incorporated into the WCGS TS.

## 2.2 Low Temperature Overpressure Protection

### (a) Background

Low temperature overpressure protection (LTOP) is provided by the PORVs on the pressurizer. These PORVs are set at pressures low enough to prevent violation of the 10 CFR Part 50 Appendix G heatup and cooldown curves should a RCS pressure transient occur during low temperature operations. The licensee, in its June 20, 1988, submittal, identified the most limiting overpressure transients analyzed to determine the PORV setpoints for LTOP. The PORV setpoint limits have been set by two design criteria. These are the limiting transients for mass addition and heat addition. In response to an NRC request for additional information (letter dated August 21, 1989, from D. V. Pickett, NRC to B. D. Withers, WCNOC), the licensee provided additional description of the methodology used for determining the PORV setpoints and revised Figures 3.4-2, 3.4-3, and 3.4-4 for the Wolf Creek TS.

(b) Evaluation

The present WCGS TSs 3.1.2.3 and 3.5.3 require one charging pump to be operable for boron injection or recirculation cooling in Modes 4, 5, and 6. For LTOP considerations, the most limiting mass addition transient was analyzed assuming letdown isolation with one centrifugal charging pump in operation with maximum charging flow. This analysis is typically performed to determine the pressure overshoot past the LTOP setpoint such that the Appendix G curves are not exceeded during the transient.

The heat input transient was analyzed assuming a 50°F temperature difference between the hotter secondary side of a steam generator and the RCS cold leg. A reactor coolant pump startup in one loop was assumed to maximize the heat transfer effect. As was the case for the mass addition transient, the pressure overshoot is calculated such that the Appendix G pressure-temperature curves for WCGS are not exceeded.

By letter dated May 22, 1990, the licensee provided supplementary information related to the thermal-hydraulic analyses used to support the proposed TS changes. The revised LTOP setpoints identified in the amendment request were derived using the same methodology employed in the development of the original LTOP setpoints currently described in the plant's Updated Safety Analysis Report. The original methodology is documented in the Westinghouse Owners' Group (WOG) report dated July 1977 and its supplement of September 1977. The WOG methodology used the LOFTRAN computer code to generate the PORV setpoint overshoot for a bounding envelope of mass addition rates. The plant specific PORV setpoints and overshoot were then determined with plant specific parameters and updated algorithms applicable to WCGS.

Based on the results of the most limiting LTOP transient, the licensee proposed TS PORV setpoints that are no higher than 1910 psig when the RCS average temperature is equal to 377°F. The PORV setpoints are progressively reduced at lower RCS temperatures to maintain the margin of safety required by Appendix G considerations. The combined effect of the new pressure-temperature limits is reflected in a modification to Figure 3.4-4 in the WCGS TS. The new Figure 3.4-4 presents the maximum allowed PORV setpoints as a function of RCS temperature and includes margin for possible instrument error. The modifications are based on analyses using approved methodology and are acceptable.

A proposed change to TS 3.4.9.1 will administratively restrict the RCS heatup rate to less than or equal to 60°F/hr for an indicated RCS average temperature less than or equal to 200°F. This change will maintain the margin of safety associated with 10 CFR Part 50, Appendix G requirements and is acceptable.

The licensee's proposed changes in indicated temperature and PORV setpoints in the TSs are consistent with the above discussed COMS alignment temperatures and the heatup and cooldown rates identified by the updated Figures 3.4-2, 3.4-3, and 3.4-4 in TS 3.4.9. The staff finds that they are reasonably conservative and are acceptable. The staff review and approval applies to the revised Figures and Bases pages provided in the licensee's May 22, 1990, and June 6, 1990, submittals.

Based on the evaluation above, the staff concludes that the licensee's proposed TS and their associated bases are acceptable to support the updated pressure-temperature limits applicable for a period up to seven EFPY.

## 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 14, 1990 (55 FR 37989).

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

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: September 27, 1990

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7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

WOLF CREEK NUCLEAR OPERATING CORPORATION

DOCKET NO. 50-482

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The Nuclear Regulatory Commission (the Commission) has issued Amendment No. 40 to Operating License No. NPF-42 issued to Wolf Creek Nuclear Operating Corporation, which revised the Technical Specifications for operation of the Wolf Creek Generating Station, located in Coffey County, Kansas.

The amendment is effective as of the date of issuance.

The amendment revised the heatup, cooldown and cold overpressure mitigation system power-operated relief valve setpoint pressure/temperature (P/T) limits as required by 10 CFR Part 50, Appendix H and Technical Specification 4.4.9.1.2.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on August 19, 1988 (53 FR 31777). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action, see (1) the application for amendment dated June 20, 1988, as supplemented on May 22, June 8 and August 1, 1990, (2) Amendment No. 40 to License No. NPF-42, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C., and at the Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects III, IV, V and Special Projects.

Dated at Rockville, Maryland, this 27th day of September 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V Pickett

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