

June 23, 1993

Docket No. 50-482

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. 63 TO FACILITY
OPERATING LICENSE NO. NPF-42 (TAC NOS. M77395 AND M77471)

The Commission has issued the enclosed Amendment No. 63 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 14, 1991, as supplemented by letters dated October 2, 1992, and April 28, 1993.

The amendment revises Technical Specifications 3/4.4.4 related to power-operated relief valves and 3/4.9.3 related to low temperature overpressure protection. These changes were requested as part of your response to Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)."

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. 63 to NPF-42
2. Safety Evaluation

cc w/enclosures:
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Date	6/3/93	6/3/93	6/10/93	6/16/93	6/22/93	

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June 23, 1993

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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. 63
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 14, 1991 and supplemented by letters dated October 2, 1992 and April 28, 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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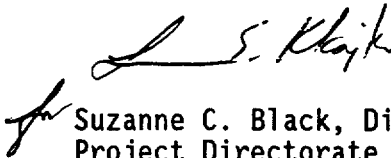
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-42 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 63, 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


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: June 23, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 63

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

3/4 4-10
3/4 4-34
3/4 4-35
B 3/4 4-2

INSERT

3/4 4-10
3/4 4-34
3/4 4-35
B 3/4 4-2

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with at least two groups of backup pressurizer heaters each having a capacity of at least 150 kW and a water level of less than or equal to 92% (1657 cubic feet).

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With one group of backup pressurizer heaters inoperable, restore at least two groups of backup heaters to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water level shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

- a. With one or both PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

*With all RCS cold leg temperatures above 368°F.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 200°F in any 1-hour period, and
- c. A maximum spray water temperature differential of 583°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following groups of two overpressure protection devices shall be OPERABLE when the Reactor Coolant System (RCS) is not depressurized through a 2 square inch or larger vent:

- a. Two residual heat removal (RHR) suction relief valves with Setpoints of 450 psig \pm 3%, or
- b. Two power-operated relief valves (PORV) with Setpoints which do not exceed the limit established in Figure 3.4-4, or
- c. One RHR suction relief valve and one PORV with Setpoints as prescribed above.

APPLICABILITY: MODE 3 when the temperature of any RCS cold leg is less than or equal to 368°F, MODE 4, MODE 5, and MODE 6 when the head is on the Reactor Vessel.

ACTION:

- a. With one of the two required overpressure protection devices inoperable in MODE 3 or 4, restore two overpressure protection devices to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.
- b. With one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 2 square inch vent within the next 8 hours.
- c. With both of the two required overpressure protection devices inoperable, complete depressurization and venting of the RCS through at least a 2 square inch vent within 8 hours.
- d. In the event either the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.
- e. The provisions of Specification 3.0.4 are not applicable.

REACTOR COOLANT SYSTEM

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.

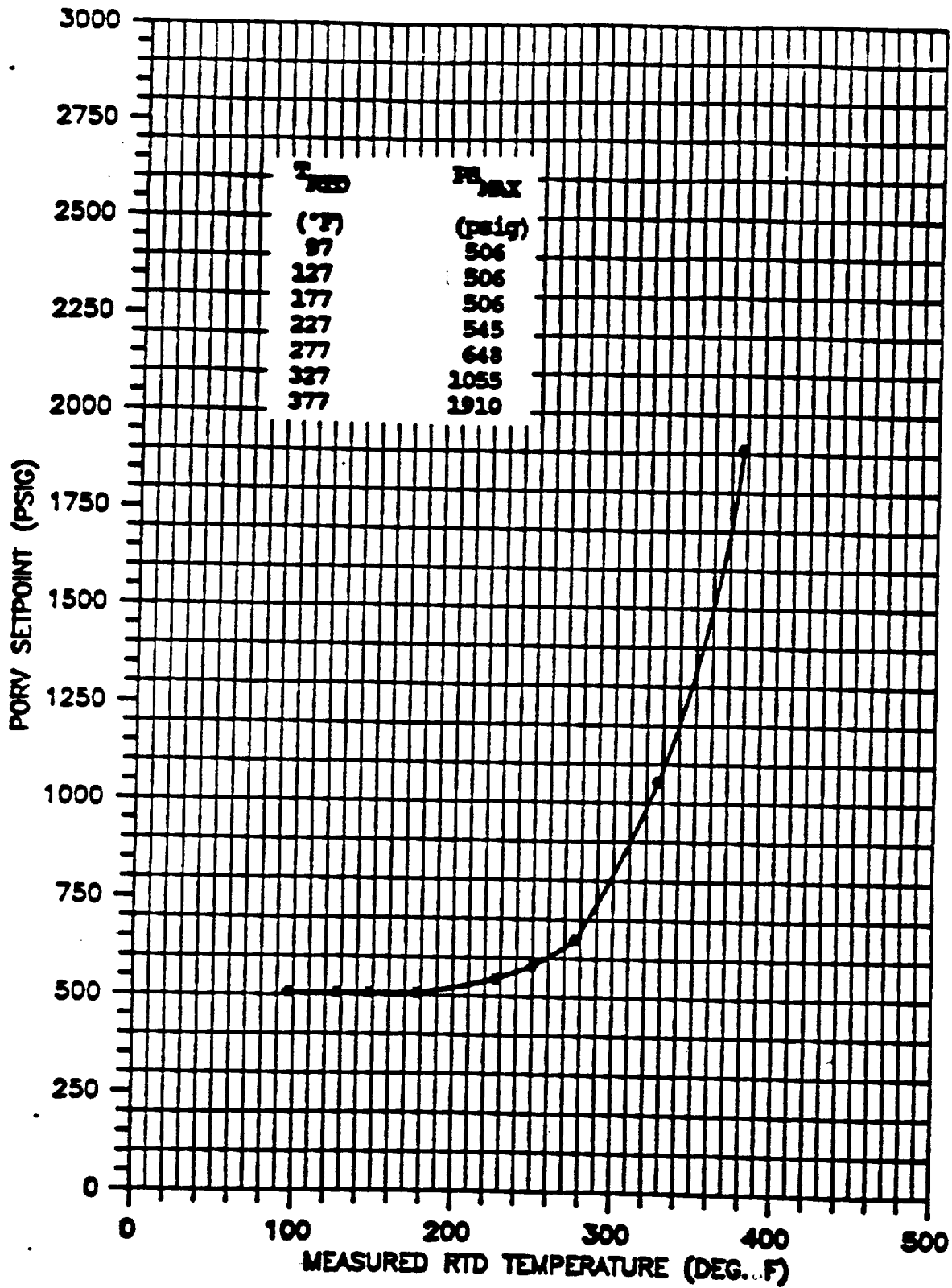


FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT
FOR THE COLD OVERPRESSURE MITIGATION SYSTEM

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the safety analysis limit DNBR (1.32) during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat even in the event of a bank withdrawal accident; however, single failure considerations require that three loops be OPERABLE. A single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented; i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5 are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

3/4.4 REACTOR COOLANT SYSTEM

BASES

SAFETY VALVES (Continued)

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The PORVs are equipped with automatic actuation circuitry and manual control capability. Because no credit for automatic PORV operation is taken in the USAR analyses for MODE 1, 2 and 3 transients, the PORVs are considered OPERABLE in either the manual or automatic mode. The automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operator action.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO.63 TO FACILITY OPERATING LICENSE NO. NPF-42
WOLF CREEK NUCLEAR OPERATING CORPORATION
DOCKET NO. 50-482
WOLF CREEK GENERATING STATION

1.0 INTRODUCTION

On June 25, 1990, the staff issued Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." The GL represented the technical resolution of the above mentioned generic issues.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluations of the reliability of power operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to plant's technical specifications were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock and Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 90, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The GL discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

By letter dated May 14, 1991, the Wolf Creek Nuclear Operating Corporation (the licensee) proposed changes to the Wolf Creek Generating Station's technical specifications in response to GL 90-06. Supplemental letters dated October 21, 1992 and April 28, 1993, provided additional information regarding PORV surveillance practices, but did not propose additional changes to the technical specifications.

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2.0 EVALUATION

2.1 Evaluation of Generic Issue 70

The actions proposed by the staff to improve the reliability of PORVs and block valves represent a substantial increase in overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The proposed changes included:

1. Include PORVs and block valves within the scope of an operational quality assurance program.
2. Include PORVs and block valves within the scope of the ASME inservice testing program. However, stroke testing of PORVs should be performed in Modes 3 or 4 and in all cases prior to establishing conditions where the PORVs are used for low-temperature overpressure protection.
3. Include technical specifications reflecting the staff positions provided in the GL (including maintaining power to block valves which have been closed due to a leaking PORV and requiring PORV testing in Modes 3 or 4).

The technical findings and the regulatory analysis related to Generic Issue 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants."

The licensee's December 21, 1990, response to GL 90-06 confirmed that the PORV and block valves were classified as safety-related at Wolf Creek and therefore were included in the quality assurance program. The PORVs and block valves were included in the NRC approved inservice testing (IST) program and the block valves were included in the motor-operated valve program developed in response to GL 89-10, "Safety Related Motor-Operated Valve (MOV) Testing and Surveillance."

The licensee's letter dated May 14, 1991, included proposed technical specification changes that were consistent with the examples provided in GL 90-06 with the exception of Surveillance Requirement 4.4.4.1. GL 90-06 requested that stroke testing of the PORV be performed in Modes 3 or 4 in order to simulate the conditions which might exist if the PORVs were actually called upon to provide a safety function. The licensee's proposed changes included continued reference to Technical Specification 4.0.5 which would allow testing in Mode 5 in accordance with the IST program.

In the letter dated April 28, 1993, the licensee committed to perform a full stroke test of the PORVs during Mode 4 prior to entering Mode 3 (ascending) every time the plant is placed in Mode 5 for greater than 72 hours (but not to exceed quarterly testing). Testing in Mode 4 was stated to provide advantages related to reduced decay heat loads compared to testing performed immediately

following power operation. The surveillance test would be considered to remain valid for up to 18 months and would maintain the valves as operable prior to establishing conditions where the PORVs are used for low temperature overpressure protection.

The staff has reviewed the licensee's proposed modifications to the Wolf Creek technical specifications and the April 28, 1993, commitment related to PORV surveillances. Since the proposed changes are consistent with the staff's positions provided in the GL and found to be justified in the aforementioned regulatory analysis, the staff finds the proposed modifications, inclusive of the April 28, 1993 commitment, to be acceptable.

2.2 Evaluation of Generic Issue 94

The actions proposed by the NRC staff to improve the availability of the low-temperature overpressure protection (LTOP) system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

In response to the GL, the licensee's May 14, 1991, submittal included proposed revisions to the Wolf Creek technical specifications. The proposed changes deviated from the example technical specification in GL 90-06 in that the licensee maintained the utilization of residual heat removal (RHR) system relief valves as an overpressure protection device as allowed by the existing Technical Specification 3.9.3. The proposed change introduces additional flexibility by having a combination of a PORV and a RHR relief valve serve as acceptable overpressure protection. The staff has reviewed the proposed limiting conditions for operation and associated surveillance requirements for the PORVs and RHR relief valves in their overpressure protection function and determined that the proposed changes satisfy the intent of the GL by increasing the availability of overpressure protection. The staff finds the proposed changes to be acceptable.

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 with the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant changes 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 (56 FR 37595). 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.

Principal Contributor: Robert Kirkwood (Generic Issue 70)
Edward Throm (Generic Issue 94)
William Reckley, Project Manager

Date: June 23, 1993