# WOLF CREEK
NUCLEAR OPERATING CORPORATION

Forrest T. Rhodes
Vice President
Engineering & Technical Services

May 14, 1991

ET 91-0075

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, D. C. 20555

Reference: 1) Letter dated June 25, 1990 from J. G. Partlow, NRC to all Pressurized Water Reactor Licensees and Construction Permit Holders
2) Letter ET 90-0190 dated December 21, 1990 from F. T. Rhodes, WCNOC to NRC

Subject: Docket No. 50-482: Revision to Technical Specification 3/4.4.4 - Relief Valves and 3.4.9.3 - Overpressure Protection System

Gentlemen:

The purpose of this letter is to transmit an application for amendment to Facility Operating License No. NPF-42 for Wolf Creek Generating Station (WCGS) Unit No. 1. This license amendment request proposes revising Technical Specification 3/4.4.4 and its associated Bases to modify the limiting conditions of operation of power-operated relief valves (PORVs) to follow the staff positions, with plant specific alternatives, as provided in Reference 1 and committed to in Reference 2. Additionally, this license amendment request proposes revising Technical Specification 3.4.9.3 to reflect the use of either the PORVs or the residual heat removal (RHR) suction relief valves for overpressure protection as provided in Reference 1 and committed to in Reference 2.

Attachment I provides a description of the amendment along with a Safety Evaluation. Attachment II provides the Significant Hazards Consideration Determination. Attachment III provides the Environmental Impact Determination. The proposed changes to the technical specifications are provided as Attachment IV.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State Official.

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P.O. Box 411 / Burlington, KS 66839 / Phone: (316) 364-8831

If you have any questions concerning this matter, please contact me or Mr. H. K. Chernoff of my staff.

Very truly yours,

Forrest T. Rhodes

Forrest T. Rhodes
Vice President
Engineering & Technical Services

FTR/jra

Attachments: I - Safety Evaluation
II - Significant Hazards Consideration Determination
III - Environmental Impact Determination
IV - Proposed Technical Specification Changes

cc: G. W. Allen (KDHE), w/a
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STATE OF KANSAS )
) SS
COUNTY OF COFFEY )

Forrest T. Rhodes, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering and Technical Services of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the content thereof; that he has executed that sa- for and on behalf of said Corporation with full power and authority to do and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By *Forrest T. Rhodes*
Forrest T. Rhodes
Vice President
Engineering & Technical Services

SUBSCRIBED and sworn to before me this *14* day of *May* , 1991.

MARLINE HEATHMAN
NOTARY PUBLIC
STATE OF KANSAS

*Marline Heathman*
Notary Public

Expiration Date *Aug 4, 1994*

# ATTACHMENT I

# SAFETY EVALUATION

# Safety Evaluation

## Proposed Change

The purpose of the proposed technical specification change is to revise Specification 3/4.4.4 on Relief Valves and its associated Bases and Specification 3.4.9.3 on Overpressure Protection System to address the recommendations of 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".

## Background

On June 25, 1990, the NRC issued Generic Letter 90-06 to advise pressurized water reactor licensees of the staff's position resulting from the resolution of Generic Issues (GIs) 70 and 94. On the basis of technical studies for GIs 70 and 94, the staff requested that to enhance safety, the actions in the Generic Letter (including changes to technical specifications) be taken by licensees that use or could use power-operated relief valves (PORVs) to perform safety-related functions.

Wolf Creek Nuclear Operating Corporation (WCNOC) participated with six other utilities to develop a common approach to Generic Letter 90-06. The plants involved in this effort were: Callaway, Vogtle, Commanche Peak, Millstone 3, Seabrook, Byron, Braidwood, and Wolf Creek Generating Station (WCGS). This group was formed due to the lack of specific guidance and a sample technical specification for plants that have the ability to use either the PORVs or the residual heat removal (RHR) suction relief valves for low-temperature overpressure protection. A joint effort was possible due to the similarity of plant types and existing technical specifications. All the plants are Westinghouse pressurized water reactors which utilize the PORVs and RHR suction relief valves for low-temperature overpressure protection. Enclosure B of the generic letter was reviewed by the group and a proposed technical specification developed that reflects the use of either the PORVs or the RHR suction relief valves.

## Evaluation

Technical Specification 3/4.4.4 and Associated Bases

An evaluation of the proposed changes to Technical Specification 3/4.4.4 is provided below:

1. The Limiting Condition for Operation (LCO) statement is being clarified by replacing "All" with "Both" as the WCGS design includes two PORVs.

2. ACTION statement a. is being revised to include the requirement to maintain power to closed block valves(s) because removal of power would render the block valve(s) inoperable and the requirements of ACTION d. would apply. Power is maintained to the block valve(s) so that it is operable and may be subsequently opened to allow the PORV to be used to control reactor coolant system pressure. Closure of the block valve(s) establishes reactor coolant pressure boundary integrity for a PORV that has excessive seat leakage. Reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event. However, the APPLICABILITY requirements to the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage so that maintenance can be performed on the PORVs to eliminate the seat leakage condition.

3. ACTION statement a., b., and c. are being changed to terminate the forced shutdown requirements with the plant being in HOT SHUTDOWN rather than COLD SHUTDOWN because the APPLICABILITY requirements of the LCO do not extend past the HOT STANDBY mode.

4. ACTION statement d. is being changed to establish remedial measures that are consistent with the function of the block valves. The prime importance for the capability to close the block valve is to isolate a stuck-open PORV. Therefore, if the block valve(s) cannot be restored to operable status within 1 hour, the remedial action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valve is inoperable. The time allowed to restore the block valve(s) to operable status is based upon the remedial action time limits for inoperable PORVs per ACTION statements b. and c. These actions are also consistent with the use of the PORVs to control reactor coolant system (RCS) pressure if the block valves are inoperable at a time when they have been closed to isolate PORVs that have excessive seat leakage. The modified ACTION statement does not specify closure of the block valves because such action would not likely be possible when the block valve is inoperable. Likewise, it does not specify either the closure of the PORV, because it would not likely be open, or the removal of power from the PORV. When the block valve is inoperable, placing the PORV in manual control is sufficient to preclude the potential for having a stuck-open PORV that could not be isolated because of an inoperable block valve. For the same reasons, reference is not made to ACTION statements b. and c. for the required remedial actions.

The change to Bases page B 3/4 4-2 adds a clarification for the PORVs operability. Technical Specification 3/4.4.4 requires that if one PORV is inoperable due to causes other than excessive seat leakage, within one hour the PORV must be restored to operable status or the associated block valve must be closed with the power removed. A PORV is considered inoperable if it is not capable of performing its specified function. As noted in the Bases revision, no credit for automatic PORV operation is taken in the USAR analysis for Modes 1, 2 and 3 transients, and the PORVs can be considered operable in either the manual or automatic mode. This clarification is added due to the potential situation where an automatic signal to the PORVs is inoperable, but the PORV is mechanically functional. Since the PORV is still mechanically functional, it would enhance safe operation to not close and remove power from the block valve, and allow the PORV to remain in a condition where it could easily be manually opened from the control room if required. This clarification is consistent with the operability requirements for the PORVs in Modes 1, 2 and 3.

In support of resolution of Generic Issue 70, Brookhaven National Laboratory performed a study[1] that estimated the risk reduction from improved PORV and block valve reliability. This study showed a small potential decrease in core melt probability due to increased PORV and block valve reliability. This was in part because the study did not include consideration of feed and bleed capability. In the course of resolution of Unresolved Safety Issue A-45 as reported in NUREG/CR-5230[2], the use of feed and bleed cooling on the primary system as an alternative measure to remove decay heat from the reactor core was explored in some detail. These studies in general support the concept of feed and bleed and indicate the probability of core melt is significantly reduced. Current technical specifications require the removal of power from the block valve(s) making it unlikely that feed and bleed could be initiated in a timely manner. The proposed changes to Technical Specifications 3/4.4.4 would require that with the block valve(s) closed (e.g., due to leaking PORVs) electric power be maintained to the block valve(s) so they can be readily opened from the control room. The increased capability for feed and bleed operation provides an increase in the overall protection of the public health and safety.

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[1] C. Hsu et al., "Estimation of Risk Reduction from Improved PORV Reliability in PWRs", Brookhaven National Laboratory, NUREG/CR-4999, BNL-NUREG-52101, Final Report, March 1988.

[2] D. M. Ericson, Jr., et al., "Shutdown Decay Heat Removal Analysis-Plant Case Studies and Special Issues: Summary Report," Sandia National Laboratories, NUREG/CR-5230, SAND88-2375, April 1989.

Technical Specification 3.4.9.3

A discussion of the proposed changes to Technical Specification 3.4.9.3 is provided below:

1. The LCO statement is being modified to require that at least two overpressure protection devices must be operable. That is, two PORVs or two RHR suction relief valves or one PORV and one RHR suction relief valve must be operable when cold overpressure protection is required. The NRC found acceptable the use of the RHR suction relief valves for low-temperature overpressure protection in NUREG-0881, Supplement No. 5, "Safety Evaluation Report related to the operation of Wolf Creek Generating Station, Unit No. 1". Analyses show that there is sufficient relief valve capacity to prevent exceeding 10 CFR Appendix G limits in the event of an inadvertent loss of letdown flow when either one charging pump or one safety injection pump is operating at full flow. The analyses also show that the RHR relief valves will prevent exceeding the Appendix G limits in the event of a reactor coolant pump start with the steam generator secondary temperature no more than 50$^{o}$F higher than reactor coolant system temperature. Additionally, the stipulations relating to depressurizing and venting of the RCS is being relocated from the APPLICABILITY statement and incorporated into the LCO statement.

2. ACTION statement a. is revised to clarify that it is only applicable in Modes 3 or 4.

3. ACTION statement b. is added to reduce the allowed outage time (AOT) for one of the two required PORV or RHR suction relief valve to 24 hours in Modes 5 or 6. The NRC has considered the conditions under which a low-temperature overpressure transient is most likely to occur. While low-temperature overpressure protection is required for all shutdown modes, the most vulnerable period of time was found to be Mode 5 with the reactor coolant temperature less than or equal to 200$^{o}$F, especially when water solid, based on the detailed evaluation of operating reactor experiences performed in support of GI 94. The staff concluded that the low-temperature overpressure protection system performs a safety-related function and inoperable overpressure protection equipment should be restored to an operable status in a shorter period of time. The current 7-day AOT is considered by the NRC to be too long under certain conditions. The NRC has concluded that the AOT should be reduced to 24 hours when operating in Modes 5 or 6 when the potential for an overpressure transient is highest.

WCNOC plans to implement a plant modification during the next refueling outage (currently scheduled to begin in September 1991) to remove the Autoclosure Interlock (ACI) function of the RHR suction isolation valves. As indicated in item 1 above, the RHR suction relief valves may be utilized as one of several means of protecting the RCS from overpressurization at low temperature conditions. During this mode of operation the ACI function can be detrimental, since a failure of one pressure transmitter can cause the closure of an isolation valve in both RHR suction lines. This would simultaneously isolate both RHR suction relief valves from the RCS and defeat their overpressure protection function. In order to prevent such a scenario, technical specifications require that power be removed from one isolation valve in each suction line. The modification being implemented enhances RHR system reliability and overpressure protection system availability by precluding spurious suction valve closures caused by potential malfunctions of the ACI circuit.

The proposed changes to Technical Specification 3.4.9.3 provide added flexibility for low-temperature overpressure protection which increases overpressure protection system availability. The combination of PORVs and RHR suction relief valves provides an equivalent level of overpressure protection with no degradation in the level of safety. Added assurance of overpressure protection system availability is provided by reducing the AOT for an inoperable PORV or RHR suction relief valve from 7 days to 24 hours in Modes 5 and 6. The increased availability of the overpressure protection system provides an increase in the overall protection of the public health and safety.

**Variances from the Recommendations of Generic Letter 90-06**

Technical Specification 3/4.4.4

Attachment A-1 to Generic Letter 90-06 proposed modified standard technical specifications for Combustion Engineering and Westinghouse plants with two PORVs. Provided below is a discussion of the variances between the WCNOC proposed technical specifications and those provided in Attachment A-1 to Generic Letter 90-06.

1. Surveillance Requirement 4.4.4.1a. required the testing of PORVs in HOT STANDBY or HOT SHUTDOWN in order to simulate the temperature and pressure environmental effects on PORVs. The PORVs are included in the NRC approved WCGS Inservice Testing (IST) program. The PORVs are full stroke tested on a COLD SHUTDOWN frequency with the block valve open in accordance with the WCGS IST program and technical specifications.

2. Surveillance Requirement 4.4.4.1b. was added to include testing of the mechanical and electrical aspects of control systems for air-operated PORVs. At WCGS there are no pneumatic components associated with the PORVs.

3. Surveillance requirement 4.4.4.3 was added to include the testing of an emergency power supply for the PORVs and block valves. As a consequence of the Three Mile Island action requirements to upgrade the operability of PORVs and block valves, an emergency power supply was provided where the valves are installed with non-safety-grade power sources. The WCGS PORVs and block valves were originally designed as safety-related components. Therefore, their normal power supplies are from Class 1E busses with no emergency power supply transfer required.

# ATTACHMENT II

## SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

## Significant Hazards Consideration Determination

This amendment application requests a change to Technical Specifications 3/4.4.4 its associated Bases and 3.4.9.3. Technical Specification 3/4.4.4 and its associated Bases and 3.4.9.3 are being revised to incorporate certain NRC staff positions resulting from the resolution of Generic Issue 70 and 94. The following sections discuss the proposed changes under the three standards of 10 CFR 50.92.

### Standard 1 - Involves a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed changes to Technical Specification 3/4.4.4 requires that with the block valve(s) closed, power be maintained to the valve(s) so they can be readily opened from the control room. This change would decrease the amount of time to initiate feed and bleed capabilities in the event an alternative measure to remove decay heat from the reactor core is necessary and thus be a benefit to plant safety. The proposed changes to Technical Specification 3.4.9.3 provides added flexibility and availability for providing low-temperature overpressure protection with no degradation in the level of plant safety. Therefore, the proposed changes to Technical Specifications 3/4.4.4, its associated Bases and 3.4.9.3 do not involve a significant increase in the probability or consequences of an accident previously evaluated.

### Standard 2 - Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes to Technical Specifications 3/4.4.4, its associated Bases and 3.4.9.3 do not create the possibility of a new or different kind of accident from any previously evaluated. No change to the design of the facility is being performed and the manner of plant operations is not significantly altered.

### Standard 3 - Involve a Significant Reduction in the Margin of Safety.

The proposed changes to Technical Specification 3/4.4.4, its associated Bases and 3.4.9.3 do not involve a significant reduction in the margin of safety. The proposed changes to Technical Specification 3/4.4.4 increase the reliability of the power-operated relief valves (PORVs) and block valves to perform their intended function. The proposed changes to Technical Specification 3.4.9.3 increases the flexibility and availability of the overpressure protection system to mitigate a low-temperature overpressurization event. The changes do not affect any technical specification margin of safety.

Based upon the above discussions it has been determined that the requested technical specification revision does not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new of different kind of accident or condition over previous evaluations; or involve a significant reduction in a margin of safety. The requested license amendment does not involve a significant hazards consideration.

# ATTACHMENT III

## ENVIRONMENTAL IMPACT DETERMINATION

**Environmental Impact Determination**

10 CFR 51.22(b) specifies the criteria for categorical exclusions from the requirement for a specific environmental assessment per 10 CFR 51.21. This amendment requests meets the criteria specified in 10 CFR 51.22(c)(9). Specific criteria contained in this section are discussed below.

**(i) the amendment involves no significant hazards consideration**

As demonstrated in the Significant Hazards Consideration Determination in Attachment II, the requested license amendment does not involve any significant hazards considerations.

**(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.**

The requested license amendment involves no change to the facility and does not significantly alter the manner of operation in a way which could cause an increase in the amounts of effluents or create new types of effluents.

**(iii) there is no significant increase in individual or cumulative occupational radiation exposure**

The proposed changes do not impact plant design features or operations that affect radiation protection, radioactive effluent processing, radioactive waste handling, or radiological environmental monitoring. The changes do not result in additional exposure by personnel nor affect levels of radiation present. The proposed changes do not result in significant individual or cumulative occupational radiation exposure.

Based on the above, it is concluded that there will be no impact on the environment resulting from this change and the change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to specific environmental assessment by the Commission.