

WOLF CREEK

NUCLEAR OPERATING CORPORATION

Forrest T. Rhodes
Vice President
Engineering & Technical Services

July 24, 1991

ET 91-0115

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

Reference: Letter ET 91-0036 dated March 5, 1991 from
F. T. Rhodes, WCNO, to the NRC
Subject: Docket No. 50-482: Proposed Revision To Technical
Specification 4.4.9.3.2 and 4.5.2.d to Allow Removal of
RHR Suction Valve Autoclosure Interlocks

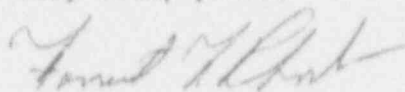
Gentlemen:

The reference submitted proposed revisions to the Wolf Creek Generating Station (WCGS) Technical Specifications. The proposed revisions allow the removal of the autoclosure interlock feature from the residual heat removal suction valves.

The attachment to this letter provides supplemental information concerning this proposed amendment. This information was requested during telephone conversations between Mr. S. G. Wideman, Wolf Creek Nuclear Operating Corporation, and Mr. H. I. Abelson, NRC. This additional information does not alter the conclusions of the safety evaluation, environmental impact determination or significant hazards consideration determination provided with the original amendment request.

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

Very truly yours,



Forrest T. Rhodes
Vice President
Engineering & Technical Services

FTR/jra

Attachment

cc: L. L. Gundrum (NRC), w/a
A. T. Howell (NRC), w/a
R. D. Martin (NRC), w/a
D. V. Pickett (NRC), w/a

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Supplemental Information

Augmented Responses to Selected NRC SER Items¹

Item 2: Valve position indication to the alarm must be provided from stem-mounted limit switches (SMLSS) and power to the SMLSS must not be affected by power lockout of the valve [justified by WCAP-11736].

Response: As noted in the amendment request, the valve position alarms will be provided by limit switches located internal to the motor operators. The alarm circuit will, however, use a set of limit-switch contacts which are separate from the limit switch contacts which provide valve position indication on the main control boards. The alarm circuit also will use a separate power supply from the normal valve control and position indication circuits. This will ensure that the valve position alarm circuit is not affected by removal of power from the valve motor operators. This design provides a sufficiently diverse means of valve position indication to the control room operators. The use of these internal limit switches has no impact on the PRA results presented in WCAP-11736² as the same failure probability is used for both types of switches (Table B-5 of WCAP-11736).

Item 3: The procedural improvements described in WCAP-11736 should be implemented. Procedures themselves are plant specific.

Response: Consistent with WCNOC practices for implementation of plant modifications, plant procedures have been reviewed to identify changes necessary to implement this modification. This review included the Wolf Creek Generating Station (WCGS) procedures corresponding to the general procedures listed in Section 9.0 of WCAP-11736. Planned procedure changes include revision of appropriate alarm response procedures. The alarm response procedure will direct the plant operators to take necessary actions to close an open Residual Heat Removal (RHR) suction valve(s) if they are not closed following alarm actuation during normal startup operations. If valve closure is not possible, the operators will be instructed to halt pressurization and return the plant to safe shutdown conditions. Surveillance procedures also will be revised to provide for appropriate testing to ensure the RHR suction valve alarms remain functional.

¹ Letter dated August 8, 1989, from Ashok Thadani, NRC, to Roger A. Newton, Westinghouse Owners Group.

² WCAP-11736, "Residual Heat Removal System Autoclosure Interlock Removal Report for the Westinghouse Owners Group," N. B. Closky, K. J. King, M. M. McHale, and C. A. Marmo, October, 1989.

Item 4: Where feasible, power should be removed from the RHR suction valves prior to their being leak-checked. [plant specific].

Response: Technical Specification 3/4.4.6.2, "Reactor Coolant System Leakage - Operational Leakage," contains the requirements for leakage testing for the RHR suction isolation valves. This specification allows leakage testing of the valves when RCS pressure is above 150 psig. The measured leakage is then adjusted to account for difference between the test pressure and normal operating pressure of reactor coolant system. Further restrictions on this testing are not desirable. Postponement of this leakage testing until after the valves are closed and power removed during startup would add critical path time to plant outages. In addition, should leakage exist, it is desirable to detect such leakage during earlier stages of an outage when corrective measures can be taken in a timely manner. Assurance of proper valve position prior to startup is confirmed by use of valve position indications and administrative controls.

Consistent with Technical Specification 3/4.4.6.2, valve leakage testing is normally performed during Mode 4 or 5. Technical Specification 3.4.1.3 and 3.4.1.4 establish the minimum operability requirements for the RHR system during these modes. These specifications require that at least one RHR Loop be in service at all times during Modes 4 and 5. Therefore, the valve leakage testing required by Technical Specification 3/4.4.6.2 is performed on one train at a time, with the valves returned to service following testing. There is no safety benefit to removing power from the valves during this evolution. Rather, power removal, and subsequent restoration, would unnecessarily add to the complexity and duration of this testing. Should a problem occur in the operating RHR loop during testing of the valves in the redundant loop, restoration of power to these valves could delay the restoration of RHR flow. The availability of the RHR suction relief valves to assist in the mitigating an RCS pressure transient could also be decreased as a result of the increased test duration.

Additional Basis for Decreased Frequency of Valve Position Verification

The proposed amendment request changes the frequency of valve position verification required by Technical Specification 4.4.9.3.2 from once per twelve (12) hours to once per seventy two (72) hours. As noted in the evaluation submitted in support of the proposed change, this revision provides a consistent surveillance interval with that provided in Technical Specification 4.4.9.3.1.c for position verification of the power operated relief valves (PORVs). It also should be noted that this proposed change follows the recommendations of WCAP-11736 and is consistent with the analysis performed for the WCGS reference plant (Callaway). The NRC has previously reviewed and approved WCAP-11736 for use as a reference in plant specific amendment requests.