VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

January 24, 1995

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC. 20555

Serial No.	95-016
NL&P/ETS:	R0
Docket Nos.	50-280 50-281
License Nos.	DPR-32 DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
PROPOSED TECHNICAL SPECIFICATIONS CHANGES
PRESSUIRZER SAFETY VALVE ACCEPTANCE CRITERIA

Pursuant to 10 CFR 50.90, the Virginia Electric and Power Company requests amendments, in the form of changes to the Technical Specifications, to Facility Operating License Nos. DPR-32 and DPR-37 for Surry Power Station Units 1 and 2. The proposed changes will modify the "as-found" test acceptance criterion for the pressurizer safety valves.

A discussion of the proposed Technical Specifications changes is provided in Attachment 1. The proposed Technical Specifications changes are provided in Attachment 2. It has been determined that the proposed Technical Specifications changes do not involve an unreviewed safety question as defined in 10 CFR 50.59 or a significant hazards consideration as defined in 10 CFR 50.92. The basis for our determination that these changes do not involve a significant hazards consideration is provided in Attachment 3. The proposed Technical Specifications changes have been reviewed and approved by the Station Nuclear Safety and Operating Committees and the Management Safety Review Committee.

Should you have any questions or require additional information, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachments

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Surry Power Station

Commissioner
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. P. O'Hanlon, who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 24TH day of January, 1995.

My Commission Expires: May 31, 1998.

Vicki L. Hull
Notary Public

(SEAL)

Attachment 1

Discussion of Changes

Surry Power Station

DISCUSSION OF PROPOSED CHANGES

INTRODUCTION

In order to eliminate unnecessary Technical Specification violations when testing the pressurizer safety valves, a change to the "as-found" acceptance criterion is being proposed. Since the "as-left" acceptance criterion remains unchanged, this change does not result in any reduction of operating or safety analysis margin. An evaluation of the affected accidents/transients has been performed which supports increasing the current Technical Specification 3.1.A.3.b pressurizer safety valve lift setpoint "as-found" acceptance criterion from ±1% to ±3%, while maintaining the "as-left" acceptance criterion at ±1%. As discussed below, analyses of the Loss of Load, Locked Rotor, and Rod Withdrawal transients demonstrate that increasing the as-found pressurizer safety valve lift setpoint acceptance criterion to ±3% does not result in transient pressures in excess of the overpressure safety limits for the primary or secondary coolant systems. These transient analyses were performed assuming an uprated core power level consistent with Virginia Electric and Power Company's Core Uprate License request (6). A Departure from Nucleate Boiling (DNB) assessment demonstrates that the increased "as-found" pressurizer safety valve lift setpoint acceptance criterion does not adversely impact the DNB results of any Surry UFSAR Chapter 14 transient analyses. The proposed TS change does not pose a significant safety hazard and does not represent an unreviewed safety question. The specific Technical Specification changes and a discussion detailing the analysis and associated results are provided.

BACKGROUND

Surry Technical Specification 3.1.A.3.b currently establishes a ±1% "as-found" and a ±1% "as-left" lift setpoint acceptance criterion for the pressurizer safety valves. Industry and Virginia Electric and Power Company experience indicates that operating effects coupled with the sensitivity of test results to variations in the test conditions have led to "as-found" test results being outside the current Technical Specification ±1% acceptance criterion. These results are considered a Technical Specification violation even though the accident analyses support actual operation at ±3% without

any safety impact. However, to maintain operational margin the "as-left" acceptance criterion remains at ±1% for the safety valves.

An "as-found" pressurizer safety valve lift setpoint acceptance criterion of ±3% is supported by the ANSI-OM-1 inservice performance testing standard. The proposed setpoint acceptance criterion increase does not change the nominal setpoint of the pressurizer safety valves. Only the allowable tolerance about the existing setpoint is to be changed.

SPECIFIC CHANGES

The Technical Specification 3.1.A.3.b pressurizer safety valve lift setpoint acceptance criterion for the "as-found" condition is being changed from ±1% to ±3%. This is the only Technical Specification affected by the proposed change to the pressurizer safety valve lift setpoint acceptance criterion.

SAFETY SIGNIFICANCE

An increase in the pressurizer safety valve lift setpoint tolerance affects the maximum pressure that will be attained in a system transient. Evaluation of the overall effect of changing the pressurizer safety valve setpoint acceptance criterion was accomplished by examining the effect of the changes on those transients which experience the most limiting pressure increases. These transients are the Complete Loss of External Electrical Load, the Locked Reactor Coolant Pump Rotor, and Rod Withdrawal events initiated from low power.

In the analyses described herein, the pressurizer safety valves were assumed to open in accordance with the References (4) and (5) pressurizer safety valve model, hereafter termed the Westinghouse model. To support the proposed pressurizer safety valve lift setpoint acceptance criterion increase, the pressurizer safety valves were assumed to begin opening at a pressure 3% above the nominal lift setpoint. For pressurizer safety valves installed on a loop seal, the Westinghouse model requires application of an additional 1% "medium shift" to account for the effects of setting the valves on steam while installing them on a water-filled loop seal. However, because the loop seals have been removed from Surry, it was not necessary to apply the additional 1% "medium shift". In addition, the Westinghouse model requires

application of a 1.31 second delay to the opening of the valve to simulate purging of the loop seal. Pressurizer safety valve installation without loop seal was simulated by reducing the delay 0.2 seconds. Lastly, the time required for the pressurizer safety valve to "pop" completely open was simulated by application of an additional 0.1% to the assumed 3% lift setpoint tolerance. The pressurizer safety valves are assumed to close at a pressure 3% below the setpoint pressure (3% blowdown).

Accident analyses previously submitted to support an uprated core power level at Surry Units 1 and 2 (6) assumed a ±1% pressurizer safety valve lift setpoint acceptance criterion, and pressurizer safety valve installation on a water-filled loop seal. Because the impact on peak transient pressure of increasing the as-found pressurizer safety valve lift setpoint acceptance criterion from ±1% to ±3% is more than offset by draining the pressurizer safety valve loop seals, the pressure transient results presented in the Surry core uprating package (6) remain bounding.

It is highly unlikely that each safety valve's setpoint will be simultaneously at one extreme of the tolerance band. Therefore, the safety valve operation assumed in the analyses described herein is very conservative. Because an increased low-end acceptance criterion potentially reduces the system pressure experienced at the point of minimum departure from nucleate boiling ratio (DNBR), the effect of an increased pressurizer safety valve setpoint acceptance criterion on the DNBR results of affected transients was evaluated. As well, the proposed changes were evaluated in light of their impact on operational margins. Transient analyses were performed with the RETRAN system transient analysis code (1), (2).

LOSS OF LOAD

The Loss of Load event is characterized by a rapid reduction in steam flow from the steam generator and a resultant rapid rise in secondary pressures. Consequently, primary side temperatures and pressures increase. The transient is terminated either by a direct reactor trip or in the limiting case by the high pressurizer pressure trip. The transient has been shown to be non-limiting with respect to core thermal margins .

The Loss of Load analysis was performed to establish that a Loss of Load event would not result in primary side pressures beyond the limit of 2750 psia nor secondary side pressures beyond the limit of 1210 psia when the pressurizer safety valve lift setpoint

acceptance criterion is increased to ±3%. The following assumptions were made in this analysis:

1. The loss of load is a 100% loss of load with no condenser dumps or power operated relief valves (PORVs) available.
2. The transient is initiated from 102% of an uprated core power level (2546 MWt).
3. Main feedwater is isolated at the time of reactor trip.
4. A least negative Doppler temperature coefficient is assumed.
5. Pressurizer sprays are disabled.
6. Supports a moderator temperature coefficient (MTC) core design limit of +6.0 pcm/°F from 0% to 50% power and a linearly decreasing limit to 0.0 pcm/°F at 100% power.
7. No credit is taken for a direct reactor trip on turbine trip.
8. No credit is taken for automatic rod control.
9. A pressurizer safety valve lift setpoint acceptance criterion of 3% is simulated in accordance with the Reference (4) Westinghouse model.

The maximum primary side (cold leg) pressure was determined to be 2680 psia, which is well below the overpressure safety limit (110% of design pressure) of 2750 psia. The maximum secondary side (steam generator pressure was determined to be 1187 psia, which is well below the overpressure safety limit of 1210 psia.

LOCKED ROTOR ANALYSIS

This analysis was performed in order to determine if an increased pressurizer safety valve lift setpoint acceptance criterion would result in an overpressurization of either the primary or secondary side during a postulated Locked Rotor transient. The following assumptions were used in this analysis:

1. Initial reactor power is 102% of an uprated core power level (2545 MWt).
2. Initial average core temperature is nominal T(avg) + 4°F.
3. Initial pressurizer pressure is 2280 psia, nominal pressure + 30 psi .
4. Pressurizer sprays do not function.
5. Pressurizer power operated relief valves do not function.
6. Condenser steam dump PORVs never open.
7. Atmospheric steam dump PORVs never open.
8. A locked rotor is the initiating event.
9. Reactor trip occurs on low Reactor Coolant System flow.

10. Coolant flow is divided into 50% through the core and 50% bypass to maximize coolant expansion in the core region.
11. A least negative Doppler temperature coefficient.
12. Supports a moderator temperature coefficient (MTC) core design limit of +6.0 pcm/°F from 0% to 50% power and a linearly decreasing limit to 0.0 pcm/°F at 100% power.
13. Minimum trip reactivity.

The RETRAN transient analysis of the Locked Rotor event with a ±3% pressurizer safety valve setpoint acceptance criterion rendered a peak primary (cold leg) pressure of 2677 psia and a peak secondary (steam generator) pressure of 1164 psia. These values are below the primary and secondary pressure safety limits of 2750 psia and 1210 psia, respectively.

ROD WITHDRAWAL EVENTS

The Loss of Load and Locked Rotor events have historically been considered the limiting Reactor Coolant System (RCS) overpressurization events. However, recent reanalyses of the Rod Withdrawal at Power (RWAP) and Rod Withdrawal from Subcritical (RWSC) events revealed that these events may result in significant pressurization of the RCS, particularly those cases initiated from low power. Therefore, the results of these accident analyses are reviewed here for completeness.

The limiting case was initiated from 12% power and assumed a bounding reactivity insertion rate, a 3% pressurizer safety valve lift setpoint tolerance, a drained loop seal (0.2 second opening delay), and a least negative Doppler temperature coefficient. This case resulted in a maximum RCS pressure of 2697 psia. Similarly, a RWSC case which assumed a bounding reactivity insertion rate, a 3% pressurizer safety valve lift setpoint tolerance, a drained loop seal (0.2 second opening delay), and a least negative Doppler temperature coefficient resulted in a maximum RCS pressure of 2643 psia. Both of these results are well below 110% of the RCS design pressure and are, therefore, acceptable.

DNB CONSIDERATIONS

Because an increased low end tolerance potentially reduces the system pressure experienced at the point of minimum Departure from Nucleate Boiling Ratio (DNBR), the effect of a ±3% pressurizer safety valve setpoint acceptance criterion on the DNBR results of affected transients was evaluated by examining the Surry UFSAR Chapter 14 safety analysis results.

Of the affected transients, only the DNBR results of the Locked Rotor event are potentially adversely affected by the increased low end tolerance. An examination of the Locked Rotor event analysis DNBR results revealed that the minimum DNBR statepoint pressure is well below the proposed low-end pressurizer safety valve lift setpoint (2500 - 75 = 2425 psia, which is the nominal setpoint minus tolerance). Therefore, the setpoint acceptance criterion increase presents no concerns regarding the DNBR results of the Locked Rotor transient.

OPERATIONAL MARGIN CONSIDERATIONS

The proposed allowable as-found lift setpoint acceptance criterion has been chosen such that an inadvertent opening of the safety valves during normal operation will not occur. (The allowable as-left setpoint acceptance criterion will remain at ±1%.) To illustrate, the nominal high primary pressure trip setpoint is 2400 psia with a maximum uncertainty of 16 psi. The nominal setpoint plus uncertainty is, therefore, 2416 psia. Because the nominal pressurizer safety valve lift setpoint minus 3% tolerance is 2425 psia, a reactor trip would occur before an inadvertent opening of the pressurizer safety valves would occur. It may be concluded that the proposed setpoint acceptance criterion change does not present any operational considerations.

SUMMARY AND CONCLUSIONS

An evaluation has been performed which justifies increasing the current Technical Specification pressurizer safety valve lift setpoint acceptance criterion from ±1% as-found and ±1% as-left to ±3% as-found and ±1% as-left. The proposed Technical Specification change does not pose a significant safety hazard and does not represent an unreviewed safety question. Loss of Load, Locked Rotor, and Rod Withdrawal transient analyses demonstrate that increasing the pressurizer safety valve lift setpoint

acceptance criterion to ±3% does not result in a transient pressure in excess of the overpressure safety limits for the primary and secondary sides. Further, the increased setpoint acceptance criterion does not adversely impact the DNBR results of any Surry UFSAR Chapter 14 transient analyses. In summary, each pertinent safety criterion was evaluated for an increased safety valve setpoint acceptance criterion and all were found to be acceptable.

REFERENCES

(1) N. A. Smith: "VEPCO Reactor System Transient Analysis using the RETRAN Computer Code," VEP-FRD1-41A (May, 1985).

(2) Letter from W. L. Stewart (Virginia Power) to H. R. Denton (NRC), "Surry and North Anna Power Stations Reactor System Transient Analyses," Serial No. 85-753, dated November 19, 1985 (RETRAN02 MOD003).

(3) Letter from W.L. Stewart (VA Power) to H.R. Denton (USNRC), "Additional Information Related to NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves," NRC Letter No. 86094, dated February 28, 1986.

(4) "Pressurizer Safety Valve Set Pressure Shift," Westinghouse Owners Group Project MUHP2351, WCAP-12910, dated March, 1991.

(5) Letter from J. E. Richardson (USNRC) to T. E. Herrman, Chairman, Pressurizer Safety Valve Working Committee, Westinghouse Owners Group, "Acceptance For Referencing Of Licensing Topical Report WCAP-12910, 'Pressurizer Safety Valve Set Pressure Shift'", February 19, 1993.

(6) Letter from J. P. O'Hanlon to USNRC, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Proposed Technical Specification Changes to Accommodate Core Uprating," Serial No. 94509, dated August 30, 1994.