VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 12, 1989

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No. 89-304
NO/ETS
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
TECHNICAL SPECIFICATION CHANGE REQUEST
INSERVICE INSPECTION AND TESTING
SUPPLEMENTAL INFORMATION

By letter dated April 11,1988, we submitted proposed changes to the Technical Specifications to meet the new inservice inspection and testing requirements for nuclear components. In an April 11, 1989, telephone conference call with your staff additional information was requested to complete your review of our Inservice Inspection Technical Specification request of April 11, 1988. This letter provides the requested information. Attachment 1 is the revised Technical Specification pages. Page 4.2-4 reinstates, per your request, the requirement to visually inspect the Safety Injection System piping in the valve pit every refueling outage and Page 4.2-5 items 1.3 and 1.4 have been renumbered to accommodate the above requirement. Attachment 2 is a summary of the engineering evaluation that supports the $\pm 3\%$ "as-found" acceptance criterion for Main Steam Safety Valve testing.

This supplemental change has been reviewed and approved by the Station Nuclear Safety and Operating Committee and the Safety Evaluation Staff. The change does not impact the previous safety evaluation nor the significant hazards consideration determinations.

If you have further questions please call.

Very truly yours,

W. L. Stewart
Senior Vice President - Power

Attachments

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cc: U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N. W.
Suite 2900
Atlanta, Georgia 30323

Mr. W. E. Holland
NRC Senior Resident Inspector
Surry Power Station

Commissioner
Department of Health
Room 400
109 Governor Street
Richmond, Virginia 23219

ATTACHMENT I
SUPPLEMENTAL
TECHNICAL SPECIFICATION REQUEST
INSERVICE INSPECTION AND TESTING REQUIREMENTS

TABLE 4.2-1

SECTION A. MISCELLANEOUS INSPECTIONS

Item No.	Required Examination Area	Required Examination Methods	Tentative Inspection During 10-Year Interval	Remarks
1.1	Materials Irradiation Surveillance	Tensile and Charpy V notch (wedge open loading) and dosimetry as necessary to insure surveillance	Capsules shall be removed and examined after 10 years. (See Notes 1 and 2)	Capsule #1 = First refueling Capsule #2 = At five years Capsule #3 = At 10 years Capsule #4 = At 20 years Capsule #5-8 = Are spares for complementary or duplicate testing.
1.2	Low Head SIS piping located in valve pit	Visual	Non-applicable	This pipe shall be visually inspected at each refueling shutdown.

Note 1: 1 year corresponds to 1 year effective full power operation.

Note 2: The results obtained from these examinations shall be used to update Figure 3.1-1 as required.

TABLE 4.2-1

SECTION A. MISCELLANEOUS INSPECTIONS

Item No.	Required Examination Area	Required Examination Methods	Tentative Inspection During 10-Year Interval	Remarks
1.3	Primary Pump Flywheel	See remarks	See remarks	Examination to be conducted in accordance with regulatory position C.4.b of regulatory guide 1.14 Rev. 1, August 1975
1.4	Low Pressure Turbine Rotor	Visual and Magnetic Particle or Dye Penetrant	100% of blades every 5 years	None

SECTION B. SENSITIZED STAINLESS STEEL

Item No.	Required Examination Area	Required Examination Methods	Tentative Inspection During 10-Year Interval	Remarks
2.1.1	Circumferential and longitudinal pipe welds and branch pipe connections larger than 4 inches in diameter	Visual and Volumetric	By the end of the interval, a cumulative 75% of the circumferential welds in the piping system would have been examined, including one foot on any longitudinal weld on either side of the butt welds	A minimum of 5% of the welds will be examined every 1-2/3 years (generally each normal refueling outage). See Transcript of Hearing (pp. 303--34) and Initial Decision (p.7, p.10)
2.1.2	Circumferential and longitudinal pipe welds and branch pipe connections	Visual	By the end of the interval a cumulative 100% of the welds and pipe branch connections would be examined a minimum of three times	A minimum of 50% of the welds will be examined every 1-2/3 years (generally, each normal refueling outage). See Transcript of Hearing (pp. 303-304) and Initial Decision (p.7, p.10)

ATTACHMENT 2
SUMMARY OF ENGINEERING EVALUATION
$\pm$ 3% ACCEPTANCE CRITERION

SAFETY EVALUATION SUMMARY

An evaluation of the impact of an as-found Main Steam Safety Valve (MSSV) setpoint tolerance of +/- 3% on the UFSAR accident analysis has been performed for Surry Power Station. The results show that there is no adverse impact on the accident analysis results in that appropriate design limits continue to be met and no unreviewed safety question as defined in 10 CFR 50.59 is created. The results of this evaluation are discussed in more detail below.

The impact of the MSSV tolerance for the design heat load rejection transient (loss of load accident) was examined first. A RETRAN sensitivity study was performed to examine the effects of modeling the 3% as-found tolerance with respect to the current licensing analyses. The case examined was a 100% load rejection from hot full power with no credit for condenser dumps or atmospheric relief valves. BOC (minimum temperature and maximum power feedback) conditions were assumed. This will be the limiting overpressure event from the standpoint of impact of MSSV setpoints.

The results showed that the proposed main steam safety valve tolerance had no impact on the peak reactor coolant system pressure. The secondary side pressure increased only slightly (less than 0.5%) and was still well within 110% of the system design pressure.

While the locked rotor event can give higher pressures on the primary side, the overpressure condition is driven by a rapid expansion of RCS water in the core combined with a loss of heat transfer to the secondary in the affected loop. MSSV setpoints would thus not be expected to significantly impact the primary side results for a locked rotor event. The effect on the secondary side pressure was examined by rerunning the current licensing analysis with the relaxed MSSV tolerances. The results showed that the secondary steam pressures remain within the limit of 110% of design pressure.

An evaluation was also performed to assess whether the proposed setpoint tolerance could affect the capability to deliver the auxiliary feedwater (AFW) flow required for core decay heat removal following a loss of normal feedwater (LONF). The results showed that in the event of a LONF, the decay heat removal requirements would not drive secondary pressures high enough to reduce AFW flow below the design requirement assumed in the safety analysis. Further, the potential increase in post-trip RCS temperature following a LONF is only a few degrees F and will have no impact on the conclusions of the UFSAR analysis.

A review of the effects of the MSSV setpoint tolerance on the effectiveness of the overtemperature/overpower delta-T protection setpoints was performed. It was concluded that the +/-3% tolerance will have no impact on the ability of the current protection setpoints to protect against DNB, vessel exit boiling or high fuel centerline temperatures.

The effects of the 3% setpoint tolerance on the low side (as-found 3% below design) on the safety analysis are bounded by the assumption that the condenser and atmospheric dump valves function normally. If the lowest setpoint safety valve (1100 psia) were to open 3% low, or 1067 psia, this is 17 psi above the nominal setpoint for the atmospheric steam dumps. The capacity of an atmospheric dump valve is approximately the same (about 3% higher) as the capacity of the lowest setpoint safety valve. The relaxed tolerances will not significantly increase the potential for post-trip actuation of the MSSV's. This potential will remain low due to the normal availability of the condenser steam dump system, the availablity of the atmospheric PORVs, the unlikelihood of the MSSV lift point being at the bottom of its tolerance band, and the margin between the actuation pressures for these components.

In conclusion, the current practice of using a +/- 3% as-found tolerance on the Main Steam Safety Valve (MSSV) as-found pressure relief setpoint does not create an unreviewed safety question as defined in 10 CFR 50.59. Specifically:

1) Use of the 3% tolerance does not increase the probability of occurrence of any accident previously evaluated in the UFSAR, since no new accident initiators result from the MSSV pressure relief setpoints. Also, the consequences of the UFSAR accidents are not increased since primary and secondary

pressures remain within appropriate design limits (i.e., less than the 110% overpressure limit) during the postulated accidents.

2) The proposed tolerances will not create the possibility of an accident or malfunction of a different type than previously evaluated. No design change to the valves is involved. The proposed Technical Specification change merely establishes a realistic and workable tolerance on the as-found relief setpoints for these valves. An evaluation has shown that relief setpoints as much as 3% above the nominal setpoint will not compromise the ability of the auxiliary feedwater system to deliver flow to the secondary side under normal or accident conditions.

3) The margin of safety as defined in the basis for any technical specification is not reduced, since the results of the accident analyses remain within their appropriate acceptance criteria.