UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 128 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated August 8, 1997, as supplemented by letters dated December 16, 1997, January 20, 1998, March 4, 1998, March 17, 1998, June 29, 1998, and July 28, 1998, Union Electric Company (UE), requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-30) for the Callaway Plant, Unit 1. The proposed amendment would revise TS 3.7-2 to specify that the lift setting tolerance for the main steam line safety valves (MSSV) is +3/-1 percent as-found and +/- 1 percent as-left. The amendment would also revise TS Table 2.2-1 to reduce the sensor error for the pressurizer pressure-high trip.

The December 16, 1997, January 20, 1998, March 4, 1998, March 17, 1998, June 29, 1998, and July 28, 1998, supplemental letters provided additional clarifying information that did not change the staff's original no significant hazards consideration determination that was published in the FEDERAL REGISTER on December 17, 1997 (62 FR 66144).

2.0 EVALUATION

MSSV Setpoint Change

Callaway has three pressurizer safety valves (PSVs) with nominal setpoints of 2485 psig and a total of 20 MSSVs (5 per steam line) with nominal setpoints of 1185 psig, 1197 psig, 1210 psig, 1222 psig, and 1234 psig. The function of the PSVs and MSSVs is to provide overpressure protection for the reactor coolant system (RCS) and main steam system (MSS) by limiting the pressure of each to within 110 percent of the system design pressure.

The licensee evaluated the transient and accident analyses in Chapter 15 of the Callaway Final Safety Analysis Report (FSAR) to assess the effects of the proposed changes. The licensee's evaluation of the non-LOCA analyses concluded that many of these analyses are not affected by the increase in the MSSV tolerance because the pressures reached never approach the values for the lift setpoints. The licensee determined that the rod withdrawal at power, reactor coolant pump locked rotor/shaft break, loss of normal feedwater/station blackout, feedwater line

9810150255 981002
PDR ADOCK 05000483
P PDR

break, partial loss of forced reactor coolant flow, complete loss of forced reactor coolant flow, inadvertent emergency core cooling system actuation at power, and rod ejection analyses could have possibly been impacted by the requested changes but, upon evaluation, further determined that the existing FSAR conclusions for these events remained valid.

In addition, the licensee determined that the turbine trip event is the bounding transient for overpressurization of the RCS and MSS. During a turbine trip event, the turbine valve closure interrupts the heat sink for the primary system consequently resulting in an overheating transient. This results in expansion of reactor coolant, insurge into the pressurizer, and pressurization of the RCS. In addition, the sudden closure of the turbine stop valves and sudden loss of steam load results in pressurization of the MSS. Upon such an event, the PSVs and MSSVs are designed to lift at predetermined setpoints in order to maintain the pressures of the RCS and MSS within acceptable limits. In addition to pressurization concerns, heating of the reactor coolant could also result in a reduction of the departure from nucleate boiling ratio (DNBR). The reactor protection system is designed with trips, namely the overtemperature delta-T trip for this case, to protect against violating the DNBR limit.

To demonstrate acceptability of the proposed changes, the licensee reanalyzed the turbine trip transient with the proposed tolerance (i.e., MSSVs were modeled to open at setpoint plus 3 percent). The turbine trip event was analyzed using the NRC approved LOFTRAN computer code and the conservative assumptions discussed below. A complete loss of steam load from full power without direct reactor trip was assumed for the analyses. The reactor was not tripped until conditions in the RCS resulted in a trip (i.e., no credit was taken for direct reactor trip on turbine trip). Reactor control was assumed to be in the manual mode to ensure that it does not automatically act to reduce the severity of the transient. No credit was taken for operation of the steam dumps or the steam generator power operated relief valves. The main feedwater flow was assumed to terminate at the time of the turbine trip with no credit taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient. Minimum reactivity feedback was conservatively assumed. This includes an assumption of 0 pcm/°F for the moderator coefficient and the least negative doppler coefficient.

In addition to the assumptions discussed above, which were common to both analyses, the licensee used the following case-specific assumptions for the RCS peak pressure and DNBR analyses.

RCS Peak Pressure:

- Initial power was assumed to be at 102 percent of rated thermal power.

- Initial RCS temperature was assumed to be nominal. Callaway-specific calculations have shown that a nominal temperature results in conservative peak RCS pressure calculation.

- Initial RCS pressure was assumed to be nominal minus 30 psi. This effectively delays the reactor trip on pressurizer pressure-high and results in a higher peak pressure.

- Initial pressurizer level was conservatively assumed to be nominal plus instrument uncertainty of 5 percent.

- In modeling of the PSVs, the licensee included the effects of loop seal clearing in accordance with the NRC approved methodology in WCAP-12910-P-A.

- In the peak RCS pressure analysis, the high pressurizer pressure reactor trip response time was assumed to be 1 second. This is a reduction from the 2 seconds assumed in the current analysis of record. With regard to this change, the licensee stated that it will revise FSAR Table 16.3-1 item 10 to incorporate the 1 second response time after issuance of this amendment. Additionally, the licensee stated that it will include the 1 second time in the surveillance requirement for the high pressurizer pressure reactor trip to assure the validity of the analysis assumption.

- No credit was taken for the effect of the pressurizer spray and power-operated relief valves since these would aid the safety valves in limiting the peak pressure.

DNBR:

- Initial core power, reactor coolant temperature, and reactor coolant pressure were assumed to be at the most limiting nominal values. The DNBR calculations were performed using the improved design procedure in which the uncertainties in the initial conditions are included in the DNBR limit value.

- The pressurizer spray and power-operated relief valves were assumed operable. These functions help maintain a lower pressure. Therefore, this assumption is bounding with respect to DNBR.

The DNBR case (i.e., primary pressure control available) is more limiting from a MSS peak pressure perspective due to a delay in reactor trip which results from the availability of pressure control. However, the DNBR analysis did not account for initial condition uncertainties. Therefore, the licensee performed sensitivity studies on the DNBR analysis with respect to the different initial parameters to evaluate the impact that such uncertainties would have on peak MSS pressure. The sensitivity studies concluded that the combined impact of modeling initial condition uncertainties and 0 percent steam generator tube plugging is on the order of 6 to 7 psi.

The licensee performed an evaluation of LOCA and non-LOCA transients with an additional 0.5 percent tolerance to account for test equipment uncertainty. The licensee stated that the 0.5 percent tolerance bounds the test equipment vendor's published equipment accuracy of 0.27 percent. The licensee's evaluation concluded that the RCS peak pressure was unaffected while the MSS peak pressure increased by less than 6 psi.

The licensee's analyses and sensitivity studies discussed above have shown that the acceptance criterion of 110 percent of system design pressure continue to be met for both the RCS and MSS with the positive increase from 1 percent to 3% percent for setpoint tolerances.

Peak pressures achieved in the analyses were 2741 psia for the RCS and 1308 psia for the MSS and were below the limits of 2748 psia and 1318.5 psia, respectively. The analyses further demonstrated that the minimum DNBR criteria continue to be met with the proposed changes. The minimum DNBR calculated was 2.201 which was above the limit of 1.7.

In addition, the licensee evaluated the effect of the proposed changes on the LOCA accidents. large break LOCA analyses result in a rapid depressurization of the RCS. As a result, the secondary side of the steam generators quickly become a heat source rather than a heat sink such that the MSSVs are not challenged. Therefore, the proposed changes have no effect on the large break LOCA analyses.

The current licensing basis small break LOCA analysis relies on the MSSVs to provide a significant path for RCS energy release until steam venting through the break occurs. The licensee computed a slight increase in the RCS pressure during this portion of the transient due to the higher secondary pressure which results from the increase in the MSSV tolerances. The licensee's analysis has shown that this also results in a 120°F increase in peak cladding temperature (PCT). When added to existing penalties, this results in a total PCT of 1626°F which is below the 10 CFR 50.46 limit of 2200°F. Therefore, the licensee concluded that the relaxed MSSV tolerance is acceptable with respect to small break LOCA.

The licensee evaluated the effect of the proposed changes on post-LOCA hot leg recirculation switch over time; LOCA hydraulic forcing functions acting on the vessel, internals, and loop; and post-LOCA long-term core cooling and concluded that the changes have no effect on the first two and an insignificant effect on the third. The licensee also evaluated the effect on the steam generator tube rupture accident and concluded that the increase in the setpoint tolerance will actually reduce the calculated break flow and offsite doses.

In conclusion, the staff has reviewed the proposed revision to the TS to increase the setpoint tolerance for the MSSVs from ±1% to +3/-1% as-found and ±1% as-left and has found these changes to be acceptable as evaluated above. In addition, maintaining an as-left tolerance of ±1% provides reasonable assurance that the valves will not drift outside of the proposed tolerance, and is therefore acceptable.

<u>Pressurizer Pressure-High Trip Sensor Error Reduction</u>

In its August 8, 1997, submittal, the licensee stated the following:

> "The revisions to Table 2.2-1 are made to assess the impact of reducing the high pressurizer pressure safety limit (SAL) from the current value of 2445 psig to 2410 psig. This SAL reduction ensures acceptable accident analysis results are obtained to accommodate a relaxation of the main steam safety valve setpoint tolerance to +3/-1 percent. These changes recognize that older generation Barton transmitters, subject to excessive negative drift, are no longer used in this application. This negative drift accounted for 4.25 percent of the 4.96 percent of span Z term. Deletion of this drift term requires that the S term be increased by 1 percent of span to account for a typical drift allowance. Thus, Z and S are revised to 0.71 percent of span and 2.0 percent span,

respectively. The reduced SAL directly results in a reduced total allowance (TA), i.e. since it reflects the difference between the SAL and the nominal trip setpoint [(2410 psig-2385 psig)/(800 span/100 percent span) = 3.125 percent span].

The decreased TA in conjunction with the increased S term results in a reduced allowable value of 2393 psig."

The pressurizer pressure setpoint changes are reflected in the markup of TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," item 10, Pressurizer Pressure - High, where the TA changed from 7.5 to 3.125, the Sensor Error Z changed from 4.96 to 0.71, and Sensor Error S changed from 1.0 to 2.0. The factors in determining the Z and S factors were not initially discussed or justified by the licensee. By letter dated January 20, 1998, in response to a staff request, the licensee indicated the information was submitted to support TS Amendment 57, dated September 20, 1990, and that except for the sensor drift (SD) and environmental allowance (EA) terms, the requested setpoint terms had not changed. By letters dated March 4 and 17, 1998, the licensee provided the values for the various parameters used in the Westinghouse analysis.

The setpoint equation, which had been used by Union Electric Company in the determination of the setpoints, came from Westinghouse, and is considered proprietary to Westinghouse. Using this equation, the staff calculated that the new value for channel statistical allowance (CSA) is 2.81 percent of span, which is consistent with the licensee's calculated CSA. The value for Z calculated by the staff is 0.71 percent of span, which also agrees with the licensee's value as stated in the original submittal. In addition, the value for S calculated by the staff is 2.0 percent, which agrees with the licensee's value as stated in the original submittal. Finally, with a TA of 3.125, a margin of 0.315 percent exists, since the CSA value is 2.81 percent. Since the CSA is less than the TA, the CSA is acceptable.

Based on the above, the staff concludes that the licensee's proposed changes to TS Table 2.2-1 to revise item 10, Pressurizer Pressure-High value for (1) TA from 7.5 to 3.125, (2) Sensor Error Z from 4.96 to 0.71, (3) Sensor Error S from 1.0 to 2.0, and (4) allowable value from less than or equal to 2400 psig to less than or equal to 2393 psig, are consistent with the approved Westinghouse setpoint methodology, and are therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri 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 the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change 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 (62 FR 66144). 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 Contributors: M. Shuaibi
G. Hammer
P. Loeser

Date: October 2, 1998