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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.135 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO.135 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

Pursuant to 10 CFR 50.90 and 50.91 Virginia Electric and Power Company (VEPCO) proposed to amend Facility Operating Licenses Nos. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. By letter dated November 10, 1989, VEPCO proposed to revise the pressurizer safety valves' (PSVs) setpoint tolerance of Technical Specification 3.1.A.3.c from ± one percent to minus (-) one percent and plus (+) five percent for the remainder of Cycle 10 for Surry, Units 1 and 2 by replacing the current footnote.

These Technical Specification changes are required because of recent information indicating a potential shift in the pressurizer safety valve shift setpoint tolerance that may exceed the ± one percent value currently required by the Technical Specifications.

This change will maintain the reactor coolant system pressure below the 110 percent design limit specified in the Updated Final Safety Analysis Report (UFSAR).

2.0 DISCUSSION AND EVALUATION

The Surry Units 1 and 2 PSVs are installed downstream of loop seals which are filled with 300°F water. The lift setpoints of the PSVs on both units were set with steam. In October 1989, the licensee was informed by Westinghouse of a finding that the actual PSV lift setpoint could shift by 4 to 8 percent under environments different from that used to establish the setpoint. Since Unit 2 was shut down on October 13, 1989 to correct a leakage problem in the "B" PSV, the licensee decided to test the Unit 2 PSVs. When tested in a loop seal water environment, the results showed an increase of lift setpoint of +3.5 to +5 percent from the as-found setpoint established with steam. The licensee, therefore, performed a safety analysis whose results indicated that the reactor coolant system (RCS) pressure of the limiting overpressurization events would remain below the acceptance criterion of 2750 psia (110 percent design pressure) with lift pressures up to 5.4 percent above the setpoint pressure. In addition, the licensee proposed compensatory measures to maintain operability of at least one power-operated relief valve (PORV) and the anticipatory reactor trip on turbine trip circuitry. Based on the licensee's analysis and proposed compensatory actions, NRC granted relief from the existing Technical Specification in the form of discretionary enforcement until December 1, 1989 (NRC letter to VEPCO dated October 27, 1989).

The lift pressures of the Unit 2 PSVs were subsequently reset with loop-seal water to correspond to the actual installation environment. However, during reactor coolant system (RCS) pressure testing prior to return to service on November 6, 1989, the "C" PSV lifted prematurely at 2335 psig due to an apparent loss of loop seal water. In order to minimize the potential for challenges to the PSVs, which may result in failure of the valve to reseat, resulting in a small break loss of coolant accident, the licensee decided to reset the lift pressures for the Unit 2 PSVs with steam, consistent with Unit 1.

Considering the fact that the actual PSV lift pressure under a loop seal environment may be 3.5 to 5 percent higher than the setting established with steam, the licensee has performed a safety analysis for the relevant UFSAR transients including loss of load/turbine trip, locked rotor, main feedline break, loss of normal feedwater and rod ejection. In all cases the peak RCS pressure was found to be below the acceptance criterion of 2750 psia even if the PSV lift pressure are assumed to increase by 5.4 percent. Therefore, the TS change to allow the PSV setpoint tolerance increase to 5 percent would not result in the RCS pressure exceeding 110 percent of design pressure.

Since the "C" PSV on Unit 2 lifted at a pressure about 6 percent lower than the set pressure, contrary to the maximum of 5 percent shift found during the valve testing earlier, the licensee was requested to examine causes of the apparent discrepancy. In addition to indicating a RCS pressure control accuracy of 2.5 percent, the licensee attributed the discrepancy as due to (1) the slower pressurization rates in the RCS pressure test relative to the rapid pressurization rate in the valve setting testing, and (2) the leakage of a steam/water mixture through the valve seat resulting in uneven heating of the dissimilar material of the valve seat and body which is postulated to result in a earlier lifting. This explanation may have merit; however, the staff is unable to make a determination that the actual PSV lift setting will be within -5 percent of the valve setting. However, considering the fact that (1) earlier analysis showed that, even without PSVs, the maximum RCS pressure would remain below 2750 psia with operability of one PORV and the reactor trip on turbine trip circuitry, and (2) the licensee indicated that measures will be taken to ensure operability of at least one PORV and the anticipatory reactor trip on turbine trip, there is reasonable assurance that the 110 percent design pressure criterion will not be exceeded even if the actual PSV setpoint increased by more than 5 percent. We therefore conclude that the TS change request for the remainder of Cycle 10 is acceptable. However, because of the uncertainty in the actual PSV lift pressure, we require that the licensee maintain the measures discussed above as compensatory measures. VEPCO has committed to continue to work with the NRC, industry and Owners Group to determine and expedite a satisfactory resolution to this generic issue in order to support the end of Cycle 10 application of this Technical Specification change.

3.0 SUMMARY

The staff has reviewed the licensee's request for an emergency TS change to increase the PSV lift setpoint tolerance from +1 percent to +5 percent for the remainder of Cycle 10 operation for both Surry Units 1 and 2. Based on the licensee's safety analysis and its intended measures to ensure operability of at least one PORV and the reactor trip on turbine trip circuitry, we have found the TS change request acceptable.

The staff is currently evaluating the PSV setting problem on a generic basis. The outcome of the staff generic evaluation for a long-term solution will also apply to Surry Units 1 and 2.

4.0 EMERGENCY CIRCUMSTANCES

In its November 10, 1989 letter, VEPCO requested that t[illegible] be treated on an emergency basis because, unless appr[illegible] would be required to shut down upon expiration of [illegible] enforcement period on December 1, 1989 and Surry Unit 2 [illegible] restart, currently scheduled for November 23, 198[illegible] information of a generic nature, on a shift in the [illegible] pressurizer safety valves due to setpoint testing [illegible]ology, [illegible] potential that the setpoint tolerance of the curren[illegible] Unit 1 safety valves may exceed the ±1 percent value required [illegible] Technical Specifications. On October 19, 1989, VEPCO requested and was granted a discretionary enforcement to permit continued operation and to further evaluate this generic issue. This discretionary enforecment will expire on December 1, 1989. As previously stated, on November 6, 1989 during RCS pressure testing a Unit 2 PSV lifted prematurely at 2335 psig. As a result of this premature lifting of the PSV, VEPCO elected to have all three of the Unit 2 PSVs tested and reset using steam. Subsequently, based on additional data obtained from testing of the Surry Unit 2 safety valves and re-analysis, VEPCO submitted the subject proposed amendment dated November 10, 1989 stating that the proposed change would not result in reactor coolant system pressure exceeding the 110 percent design limit specified in the UFSAR. Moreover, VEPCO stated that additional measures would be taken by monitoring the operability of the power operated relief valves and the anticipatory reactor trip on turbine trip circuitry. Thus, unless these amendments are promptly authorized, Unit 1 would be required to shut down on December 1, 1989 and restart of Unit 2 would be delayed beyond the current scheduled date [illegible] November 23, 1989.

In accordance with 10 CFR 50.91(a)(5), VEPCO has explained that it could not have avoided this emergency situation since this generic concern was only recently identified. Thus, the NRC staff does not believe that VEPCO has abused the emergen[illegible]visions in this instance. Accordingly, the Commission has determ[illegible] are emergency circumstances warranting prompt approv[illegible]

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may take a final determination that a license amendment involves no significant hazards considerations if operation of the facility, in accordance with the proposed changes would not:

1. Involve a significant increase in the probability or consequences of any accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

This amendment has been evaluated against the standards in 10 CFR 50.92. It does not involve a significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability of occurrence or consequences of any accident or malfunction of equipment which is important to safety and which has been evaluated in the UFSAR. The proposed change effectively recognizes the potential shift in lift setpoint due to testing methodology. As such, the setpoint shift being positive, the probability of a safety valve challenge may be reduced. The consequences of such a challenge are unaffected as the UFSAR analysis remains bounding within the proposed setpoint tolerance. In addition, the Units 1 and 2 valve setpoint shift is expected to be in the same range as the Unit 2 valve test results (+3.5 percent to +5 percent) and therefore no increase in the consequences of any accident or malfunction of equipment important to safety is expected.

2. Create the possibility of a new or different type of accident from those previously evaluated in the safety analysis report. No modifications are being made to the pressurizer safety valves for either unit at this time. Potential installation of temporary strap-on temperature instrumentation has no operational impact on valve performance. Capping of loop seal drains is being performed only to ensure that the loop seals are not lost due to leakage through the drains and hence has no impact on the intended design of the safety valves. With the setpoint change expected to be in the same range as the Unit 2 valve test results, there is no new or different kind of accident or accident precursors expected. The additional measures being implemented are only being used to further ensure that the system pressure will remain below 2750 psia (110 percent of design pressure) during any analyzed transient or operating condition.

3. Involve a significant reduction in the margin of safety. Plant operations are not being changed. Although accident analysis assumptions have been modified to assume an initial 5.4 percent shift in

pressurizer safety valve lift pressure, there is no reduction in the margin of safety since the 110 percent design pressure is not exceeded in any accident evaluated in the UFSAR. For valve setpoint tolerance consistent with setpoint shift experienced during testing, the accident analysis remains bounding.

Accordingly, the Commission has determined that this amendment involves no significant hazards considerations.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Commonwealth of Virginia representative was contacted and had no comments regarding issuance of this amendment.

7.0 ENVIRONMENTAL CONSIDERATION

This 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 staff has determined that these amendments involve 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 made a final no significant hazards consideration finding with respect to this amendment. 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 amendments.

8.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 16, 1989

Principal Contributors:
Y. Hsii
B. Buckley