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

DO NOT REMOVE

November 16, 1989

Docket Nos. 50-280
and 50-281

Posted
Amdt. 135 to DPR-37

Mr. W. L. Stewart
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Stewart:

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENT (TAC NOS. 75273 And 75274)

The Commission has issued the enclosed Amendment No. 135 to Facility Operating License No. DPR-32 and Amendment No. 135 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively, in response to your application dated November 10, 1989.

The change revises the pressurizer safety valves' ± one percent setpoint tolerance of Technical Specification 3.1.A.3.c to minus (-) one percent and plus (+) five percent for the remainder of Cycle 10 for both Surry Units 1 and 2. This change was requested because of the potential that the Surry safety valve lift pressure may exceed the current ± one percent of setpoint tolerance required by the existing Technical Specifications due to setpoint testing methodology. The revised safety valve setpoint tolerance is encompassed by the data obtained from recent testing of Surry Unit 2 safety valves and, based on your re-analysis, the reactor coolant system pressure will remain below the 110 percent design overpressure limit for applicable system transients. The NRC staff has reviewed the bases for these changes and agrees that, based on the re-analysis, the reactor coolant pressure will not exceed the design limits specified in the Updated Final Safety Analysis Report (UFSAR). Because of the uncertainty in the actual safety valve lift pressure, we require that the measures you committed to take regarding the operability of the power operated relief valves and the reactor trip on the turbine trip circuitry, to further protect against overpressurization, be maintained as compensatory measures. In addition, we acknowledge your commitment to continue to work with the NRC, industry and Owners Groups 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.

Upon identification of this generic issue and, based on your re-analysis and evaluation of the data you had obtained on the Surry Unit 2 safety valves on October 19, 1989, you requested and were granted a discretionary enforcement to permit continued operation of Surry Unit 1 and further evaluation of this issue. The discretionary enforcement for Surry Unit 1 expires on December 1, 1989. Surry Unit 2 is currently in a maintenance outage and is scheduled to restart on November 23, 1989. Your letter of November 10, 1989 requested that these amendments be processed on an emergency basis so that Surry Unit 1 would not have to shut down on December 1, 1989, the end of the discretionary period, and that Surry Unit 2 would be able to restart on schedule. Insufficient time exists for the Commission's usual 30-day notice period without shutting down Surry Unit 1 and preventing restart of Surry Unit 2.

A copy of the staff's Safety Evaluation is also enclosed. The enclosed Notice of Issuance and final determination of no significant hazards considerations and opportunity for a hearing will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Bart C. Buckley, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 135 to DPR-32
2. Amendment No. 135 to DPR-37
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

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Mr. W. L. Stewart
Virginia Electric and Power Company

Surry Power Station

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 135
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated November 10, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 135 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Gus C. Lainas, Assistant Director
for Region II Reactors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1989

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.135
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated November 10, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 135 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Gus C. Lainas, Assistant Director
for Region II Rectors
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 16, 1989

ATTACHMENT TO LICENSE AMENDMENT

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

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

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows: replace the following page of Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

Remove Page	Insert Page
3.1-4	3.1-4

b. Three valves shall be operable when the reactor coolant average temperature is greater than 350°F, the reactor is critical, or the Reactor Coolant System is not connected to the Residual Heat Removal System.

c. Valve lift settings shall be maintained at 2485 psig $^{+1}_{-1}$ percent.*

4. Reactor Coolant Loops

Loop stop valves shall not be closed in more than one loop unless the Reactor Coolant System is connected to the Residual Heat Removal System and the Residual Heat Removal System is operable.

5. Pressurizer

a. The reactor shall be maintained subcritical by at least 1% until the steam bubble is established and the necessary sprays and at least 125 KW of heaters are operable.

b. With the pressurizer inoperable due to inoperable pressurizer heaters, restore the inoperable heaters within 72 hours or be in at least hot shutdown within 6 hours and the reactor coolant system temperature and pressure less than 350°F and 450 psig, respectively, within the following 12 hours.

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* For the remainder of Cycle 10 operation for both units the valve lift settings shall be maintained at 2485 psig $^{+5}_{-1}$ percent.

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 these amendments be treated on an emergency basis because, unless approved, Surry Unit 1 would be required to shut down upon expiration of a discretionary enforcement period on December 1, 1989 and Surry Unit 2 would be prevented from restart, currently scheduled for November 23, 1989. As a result of recent information of a generic nature, on a shift in the setpoint tolerance of pressurizer safety valves due to setpoint testing methodology, there is a potential that the setpoint tolerance of the currently operating Surry Unit 1 safety valves may exceed the ±1 percent value required by the 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 enforcement 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 of 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 emergency provisions in this instance. Accordingly, the Commission has determined that there are emergency circumstances warranting prompt approval.

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:
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