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

October 24, 1990

Posted
Amdt. 144 to DPR-37

Docket Nos. 50-280
and 50-281

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 AMENDMENTS RE: PRESSURIZER SAFETY VALVES (TAC NOS. 76913 AND 76914)

The Commission has issued the enclosed Amendment No. 148 to Facility Operating License No. DPR-32 and Amendment No. 144 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated May 22, 1990, as supplemented June 15, 1990.

These amendments delete Section 3.1.A.3.a from the TS, which required that at least one pressurizer safety valve be operable whenever the reactor vessel head is on the reactor vessel, except during hydrostatic testing. In addition, the amendments correct a typographical error in Section 3.7.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Bart C. Buckley

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. 148 to DPR-32
2. Amendment No. 144 to DPR-37
3. Safety Evaluation

cc w/enclosures:
See next page

October 24, 199

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and 50-281

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See attached sheet

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5000 Dominion Blvd.
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/s/

Bart C. Buckley, Senior Project Manager
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DATE	10/10/90	10/10/90	10/15/90	10/11/90	10/11/90	10/11/90	

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

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 148
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 May 22, 1990, as supplemented June 15, 1990, 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. 148, 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

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 24, 1990

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. 144
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 May 22, 1990, as supplemented June 15, 1990, 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. 144, 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

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 24, 1990

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 148 FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 144 FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS 3.1-3	TS 3.1-3
TS 3.1-4	TS 3.1-4
TS 3.1-5b	TS 3.1-5b
TS 3.7-7	TS 3.7-7
TS 3.7-18	TS 3.7-18

e. Reactor power shall not exceed 50% of rated power with only two pumps in operation unless the overtemperature ΔT trip setpoints have been changed in accordance with Section 2.3, after which power shall not exceed 60% with the inactive loop stop valves open and 65% with the inactive loop stop valves closed.

f. When all three pumps have been idle for > 15 minutes, the first pump shall not be started unless: (1) a bubble exists in the pressurizer or (2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

2. Steam Generator

A minimum of two steam generators in non-isolated loop shall be operable when the average reactor coolant temperature is greater than 350°F.

3. Pressurizer Safety Valves

a. Three valves shall be operable when the head is on the reactor vessel and 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.

b. Valve lift settings shall be maintained at 2485 psig ± 1 percent*

* For the remainder of Cycle 10 and Cycle 11 operation for both units, the valve lift settings shall be maintained at 2485 psig (+5,-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.

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The requirement for redundant coolant loops ensures the capability to remove core decay heat when the reactor coolant system average temperature is less than or equal to 350°F. Because of the low-low steam generator water level reactor trip, normal reactor criticality cannot be achieved without water in the steam generators in reactor coolant loops with open loop stop valves. The requirement for two operable steam generators, combined with the requirements of Specification 3.6, ensure adequate heat removal capabilities for reactor coolant system temperatures of greater than 350°F.

Each of the pressurizer safety valves is designed to relieve 295,000 lbs. per hr. of saturated steam at the valve setpoint. Two safety valves have a capacity greater than the maximum surge rate resulting from complete loss of load.[2]

The limitation specified in item 4 above on reactor coolant loop isolation will prevent an accidental isolation of all the loops which would eliminate the capability of dissipating core decay heat when the Reactor Coolant System is not connected to the Residual Heat Removal System.

Auxiliary Feedwater System Actuation

The automatic initiation of auxiliary feedwater flow to the steam generators by instruments identified in Table 3.7-2 ensures that the Reactor Coolant System Decay Heat can be removed following loss of main feedwater flow. This is consistent with the requirements of the "TMI-2 Lesson Learned Task Force Status Report," NUREG-0578, item 2.1.7.b.

Setting Limits

1. The high containment pressure limit is set at about 10% of design containment pressure. Initiation of Safety Injection protects against loss of coolant(2) or steam line break(3) accidents as discussed in the safety analysis.

2. The high-high containment pressure limit is set at about 23% of design containment pressure. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant(2) or steam line break accidents(3) as discussed in the safety analysis.

3. The pressurizer low pressure setpoint for safety injection acutation is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis.(2)

TABLE 3.7-4

ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

NO.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
1	High Containment Pressure (High Containment Pressure Signal)	a) Safety Injection b) Containment Vacuum Pump Trip c) High Press. Containment Iso. d) Safety Injection Contain. Iso. e) F.W. Line Isolation	≤ 5 psig
2	High High Containment Pressure (High High Containment Pressure Signals)	a) Containment Spray b) Recirculation Spray c) Steam Line Isolation d) High High Press. Contain. Iso.	≤ 10.3 psig
3	Pressurizer Low Low Pressure	a) Safety Injection b) Safety Injection Cont. Iso. c) Feedwater Line Isolation	≥ 1,700 psig
4	High Differential Pressure Between Steam Line and the Steam Line Header	a) Safety Injection b) Safety Injection Contain. Iso. c) F.W. Line Isolation	≤ 150 psig
5	High Steam Flow in 2/3 Steam Lines	a) Safety Injection b) Steam Line Isolation c) Safety Injection Contain. Iso. d) F.W. Line Isolation	≤ 40% (at zero load) of full steam flow ≤ 40% (at 20% load) of full steam flow ≤ 110% (at full load) of full steam flow
	Coincident with Low T_{avg} or		≥ 541°F T_{avg}
	Low Steam Line Pressure		≥ 500 psig steam line pressure

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 144 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

By letter dated May 22, 1990, as supplemented June 15, 1990, Virginia Electric and Power Company (the licensee) proposed to amend Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station (SPS), Units 1 and 2. The proposed changes would delete Section 3.1.A.3.a from the SPS Units 1 and 2 Technical Specifications (TS). Currently, Section 3.1.A.3.a of the TS requires that at least one pressurizer safety valve be operable whenever the reactor vessel head is on the reactor vessel, except during hydrostatic testing. The proposed amendments would also correct a typographical error in Section 3.7.

2.0 EVALUATION

The pressurizer code safety valves are designed to prevent the reactor coolant system from being pressurized above the safety limit of 2750 psia. The relief capacity of the safety valves is adequate to relieve any overpressure condition which could occur during power operations until the low temperature overpressure protection (LTOP) system is activated at or below 350°F as specified in TS Section 3.1.G. Those transients described in the Surry Updated Final Safety Analysis Report (UFSAR) which experience pressures that challenge the pressurizer safety valves are: loss of normal feedwater, locked rotor, and loss of load transients. The limiting cases for these transients assume the reactor is initially at hot full power conditions. No credit is taken for the operability of the pressurizer safety valves to mitigate pressure transients when the reactor coolant temperature is at or below 350°F. The residual heat removal system is brought on line when the reactor coolant temperature is at 350°F to remove the decay heat and to bring the plant to cold shutdown condition. The TS and their Bases imply that the above-cited operability requirement (i.e., one operable pressurizer safety valve whenever the head is on the reactor vessel, except during hydrostatic tests) is intended to provide overpressure protection when the reactor coolant temperature is less than 350°F, the reactor is subcritical, and the reactor coolant system is connected to the residual heat removal (RHR) system. However, since safety valves are not needed to mitigate overpressure transients at or below 350°F, their operability is not needed. To protect the reactor coolant system at or below 350°F, the licensee will utilize the LTOP system.

The NRC staff previously reviewed the licensee's proposed LTOP system and certain changes to the SPS, Units 1 and 2 TS, which the NRC staff found to be acceptable. The LTOP system was installed and the TS pertaining to the reactor coolant system overpressure mitigation were issued by Amendments No. 56 and 55 for SPS, Units 1 and 2, respectively. Ferritic materials used as pressure retaining components to the reactor coolant system are less tough and could potentially fail in a brittle manner if subjected to high pressures at low temperatures. Since the pressurizer safety valves have a nominal setpoint value of 2485 psig, they would not provide the required overpressure mitigation capability when the reactor coolant temperature is less than 350°F with the head installed on the reactor vessel.

The LTOP system would provide the required overpressure protection during these latter circumstances. Moreover, as stated in Amendments Nos. 147 and 143, issued on October 24, 1990, for SPS, Units 1 and 2, LTOP is provided by TS controls on charging pump operability, and by reactor coolant system (RCS) vent paths through operable power operated relief valves (PORVs). These controls ensure anticipated mass or energy addition transients cannot result in excessive RCS pressurization in low temperature conditions. These above-cited amendments reduced the PORV setpoint from 435 to 385 psig.

The most limiting mass addition transient was analyzed assuming an inadvertent actuation of a charging pump. The present TS Section 3.1.G.1.b allows only one charging pump to be operable when the RCS temperature is less than or equal to 350°F, which is the maximum RCS temperature for which LTOP is required. The analysis was performed to determine the pressure overshoot past the LTOP setpoint such that the Appendix G curves are not exceeded during the transient. Separate analyses were performed for each unit since the pressure-temperature limits are different for each unit. However, for ease of operation, the more restrictive (lower limit) was selected for the proposed TS.

The heat input transient was analyzed assuming a 50°F temperature difference between the steam generator and the RCS. A reactor coolant pump startup in one loop was assumed to maximize the heat transfer effect. As was the case for the mass addition transient, the pressure overshoot was calculated such that the Appendix G pressure-temperature curves for each unit were not exceeded.

The staff concludes that since the pressurizer safety valves are not needed to mitigate overpressure transients at or below 350°F, their operability is not needed and thus, the proposed amendment is acceptable.

With respect to the typographical error, Section 3.7 of the TS stipulates incorrectly a containment High-High actuation setpoint of 25 psig instead of 25 psia or 10.3 psig. Therefore, Table 3-7.4 is being revised to incorporate the correct actuation setpoint of 10.3 psig. The Bases is also being changed to indicate the correct percent of the containment design pressure that the containment spray system is actuated. Therefore, the current number of 50 percent, which was incorrectly determined using the 25 psig value, is being changed to 23 percent based on the correct value of 25 psia.

3.0 SUMMARY

Based on the NRC staff review of the information, we conclude that the proposed amendments are acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the 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 previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet 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 these amendments.

5.0 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 24, 1990

Principal Contributor:
B. Buckley