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Docket Nos. 50-338
and 50-339

Posted
Amdt. 161 to NPF-4

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

Dear Mr. Stewart:

SUBJECT: NORTH ANNA UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ELIMINATION OF RESISTANCE TEMPERATURE DETECTORS AND SUBSTITUTION OF THERMOWELLS (TAC NOS. M82838 AND M82839)

The Commission has issued the enclosed Amendment Nos. 161 and 142 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2). The amendments revise the Technical Specifications (TS) in response to your letter dated December 20, 1991.

The amendments eliminate the use of the reactor coolant resistance temperature detectors bypass system and implement in its place the use of thermowells that extend into the main reactor coolant system piping.

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,

/s/

Leon B. Engle, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 161 to NPF-4
2. Amendment No. 142 to NPF-7
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDII-2	:PE:PDII-2	:PM:PDII-2	:SRXB	:OGC	:D:PDII-2
NAME	:D. Miller	:F. Rinaldi	:L. Engle	:R. Jones	: E. Bachmann	:H. Berkow
DATE	:4/22/92	: 4/23/92	: 4/23/92	: 4/23/92	: 5/1/92	: 4/27/92

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Virginia Electric & Power Company

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Units 1 and 2

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated December 20, 1991, 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 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 161 , 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 its date of issuance and shall be implemented prior to restart from the steam generator replacement project.

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: May 15, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 161

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
TS 3/4 3-10	TS 3/4 3-10
TS 3/4 3-28	TS 3/4 3-28

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 27.0*
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure -- Low-Low	
a. Safety Injection (ECCS)	$\leq$ 27.0*/13.0#
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
4. Differential Pressure Between Steam Lines-High	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
5. Steam Flow in Two Steam Lines - High Coincident with T_{avg}--Low-Low	
a. Safety Injection (ECCS)	$\leq$ 16.75#/26.75##
b. Reactor Trip (from SI)	$\leq$ 6.75
c. Feedwater Isolation	$\leq$ 11.75
d. Containment Isolation-Phase "A"	$\leq$ 21.75#/31.75##
e. Auxiliary Feedwater Pumps	$\leq$ 61.75
f. Essential Service Water System	Not Applicable
g. Steam Line Isolation	$\leq$ 11.75

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and	8%	>7%	Prevents reactor trip when any of the following occur: low flow, reactor coolant pump breakers open, under-voltage (RCP busses), under-frequency (RCP busses), pressurizer low pressure or pressurizer high level.
	2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8%	>7%	
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Allows reactor trip when any of the following occur: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Prevents reactor trip when any of the following occur: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	≤ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	≤ 0.5 seconds*
7. Overtemperature ΔT	5.75 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	≤ 2.0 seconds
10. Pressurizer Pressure--High	≤ 2.0 seconds
11. Pressurizer Water Level--High	≤ 2.0 seconds

*Neutron detectors are exempt from response time testing. Response of the neutron flux signal portion of the channel time shall be measured from detector output or input of first electronic component in channel.

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 142
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated December 20, 1991, 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 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 142 , 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 its date of issuance and shall be implemented prior to restart after the fall 1993 refueling outage.

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: May 15, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 142

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
TS 3/4 3-10	TS 3/4 3-10
TS 3/4 3-30	TS 3/4 3-30

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION	SETPOINT	ALLOWABLE VALUES	FUNCTION
P-7 (Cont'd)	3 of 4 Power range below setpoint and	8%	>7%	Prevents reactor trip when any of the following occur: low flow, reactor coolant pump breakers open, undervoltage (RCP busses), underfrequency (RCP busses), pressurizer low pressure or pressurizer high level.
	2 of 2 Turbine Impulse chamber pressure below setpoint (Power level decreasing)	8%	>7%	
P-8	2 of 4 Power range above setpoint (Power level increasing)	30%	<31%	Allows reactor trip when any of the following occur: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip.
	3 of 4 Power range below setpoint (Power level decreasing)	28%	>27%	Prevents reactor trip when any of the following occur: low flow in a single loop, a single reactor coolant pump breaker open, or a turbine trip.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	NOT APPLICABLE
2. Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3. Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4. Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5. Intermediate Range, Neutron Flux	NOT APPLICABLE
6. Source Range, Neutron Flux	$\leq$ 0.5 seconds*
7. Overtemperature ΔT	5.75 seconds*
8. Overpower ΔT	NOT APPLICABLE
9. Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10. Pressurizer Pressure--High	$\leq$ 2.0 seconds
11. Pressurizer Water Level--High	$\leq$ 2.0 seconds

*Neutron detectors are exempt from response time testing. Response of the neutron flux signal portion of the channel time shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. Safety Injection (ECCS)	Not Applicable
Feedwater Isolation	Not Applicable
Reactor Trip (SI)	Not Applicable
Containment Isolation-Phase "A"	Not Applicable
Auxiliary Feedwater Pumps	Not Applicable
Essential Service Water System	Not Applicable
Containment Air Recirculation Fan	Not Applicable
b. Containment Spray	Not Applicable
Containment Isolation-Phase "B"	Not Applicable
c. Containment Isolation-Phase "A"	Not Applicable
d. Steam Line Isolation	Not Applicable
2. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}/28.0^{(3)}$
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Essential Service Water System	Not Applicable
3. Pressurizer Pressure--Low-Low	
a. Safety Injection (ECCS)	$\leq 27.0^{(1)}/13.0^{(2)}$
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	$\leq 18.0^{(2)}$
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Essential Service Water System	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Differential Pressure Between Steam Lines – High	
a. Safety Injection (ECCS)	≤ 13.0(2)/23.0(3)
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation – Phase "A"	≤ 18.0(2)/28.0(3)
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Essential Service Water System	Not Applicable
5. Steam Flow in Two Steam Lines – High Coincident with Tavg – Low-Low	
a. Safety Injection (ECCS)	≤ 16.75(2)/26.75(3)
b. Reactor Trip (from SI)	≤ 6.75
c. Feedwater Isolation	≤ 11.75
d. Containment Isolation – Phase "A"	≤ 21.75(2)/31.75(3)
e. Auxiliary Feedwater Pumps	≤ 61.75
f. Essential Service Water System	Not Applicable
g. Steam Line Isolation	≤ 11.75
6. Steam Flow in Two Steam Lines – High Coincident with Steam Line Pressure – Low	
a. Safety Injection (ECCS)	≤ 13.0(2)/23.0(3)
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation – Phase "A"	≤ 18.0(2)/28.0(3)
e. Auxiliary Feedwater Pumps	≤ 60.0
f. Essential Service Water System	Not Applicable
g. Steam Line Isolation	≤ 8.0

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 161 AND 142 TO

FACILITY OPERATING LICENSE NOS. NPF-4 AND NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

NORTH ANNA POWER STATION, UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated December 20, 1991, the Virginia Electric and Power Company (the licensee) requested revisions to Technical Specifications (TS) 3.3.1.1 (Table 3.3-2) and 3.3.2.1 (Table 3.3-5) of Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units 1 and 2, respectively. The proposed revisions eliminate the use of the reactor coolant resistance temperature detectors (RTD) bypass system and implement in its place the use of thermowells that extend into the main reactor coolant system (RCS) piping. This proposed change increases the response time associated with RCS temperature measurement, but eliminates drawbacks associated with leakages, flow reductions, and man-rem exposures. The licensee plans to implement this design change during the upcoming steam generator replacement outage commencing in January 1993 for Unit 1 and the Cycle 9 refueling outage in fall 1993 for Unit 2. Also, the licensee proposes the deletion of an expired footnote on Table 3.3-2.

2.0 BACKGROUND

The reactors at the North Anna Power Station have three cold leg inlets and three hot leg outlets. Temperature measurements are made at each leg in order to determine the average temperature and the differential temperature between hot and cold legs. The current method of measuring the hot and cold leg reactor coolant temperatures utilizes RTD electrical devices. Three mixing scoops are located in each hot leg, 120 degrees apart, to provide a representative sample. Each scoop has five orifices which sample the flow from the hot leg. The flow is then piped to a manifold where a direct immersion RTD measures the hot leg loop temperature. The cold leg temperature is measured in a similar manner with piping to a separate bypass manifold, except that no scoops are used. The RTD bypass system consists of about

240 feet of piping, 21 associated valves, hangers which include 30 snubbers, 6 sets of flanges and 6 RTD manifolds. The current RTD bypass system has had two major drawbacks: a lack of reliability, which has caused forced shutdowns and flow reductions, and high man-rem exposure. The forced plant shutdowns were due to leakages from valve packings or mechanical joints, and flow reductions resulted from valve problems. The RTD bypass piping has contributed to the man-rem exposure as a result of crud trapped by the valves and socket-welded pipes. The proposed modification will eliminate the RTD bypass loop and introduce thermowells that extend into the RCS piping.

3.0 EVALUATION

The licensee has identified that the proposed TS modification will increase the total response time as determined by the surveillance requirements, due to the increase of the sensor thermal response time of a system that utilizes thermowells. However, the accident analysis results, as presented in the Updated Final Safety Analysis Report (UFSAR) Chapter 15, will not change. The increase in thermal response time is offset by a reduction in transport and thermal delay time associated with the elimination of the bypass piping. Further, the licensee has committed to perform post-modification testing to demonstrate that the total channel response time remains within the assumptions specified in the UFSAR.

The proposed thermowells will house Weed Instrument Company, Inc. dual element RTDs. Also, each spare RTD element will be wired to the 7300 Process Protection System cabinets so that switchover to the spare element can be done at the racks. In addition, each of the three hot leg RTDs per loop will be wired to an RTD amplifier card and the three signals will be averaged to produce one hot leg temperature signal.

The response time of the proposed thermowells is 1.75 seconds slower than the existing RTD system. This factor includes a 10 percent error allowance. Therefore, the response time for overtemperature (Table 3.3-2) and for low-low average temperature (Table 3.3-5) would be increased by 1.75 seconds. Table 3.3-2 lists the maximum time to automatically trip the reactor when the power, computed from the RCS temperature, exceeds the limit. Table 3.3-5 lists the maximum times to automatically start certain engineered safety features when two of the three steam lines have excessive flow and the average reactor coolant temperature is below its low-low limit. The elimination of the existing RTD bypass loop would reduce the thermal lag and travel time from 2.0 seconds to 0.25 seconds. This response time reduction will compensate for the previously stated increase of 1.75 seconds, resulting in no change in total system response time.

The following tabulation shows the respective gains and losses in specific and total response time for the two systems.

RESPONSE TIME PARAMETERS FOR RCS TEMPERATURE MEASUREMENT

Parameters	Bypass System	Thermowell System
Response Time/Thermal Lag	3.0	4.75
Loop/Scoop Transient & Thermal Lag	2.0	0.25
Electronics/Electrical Delays	1.0	1.00
Total System Response	6.0	6.0

The temperature input to the engineered safety features (high steam line flow coincident with low-low RCS Tavg) will be impacted in the same way as the reactor trip input. TS allowable response time will be increased by 1.75 seconds to reflect the increase in thermal response time and the elimination of the bypass loop transport effect ensures that the total system response time remains unchanged. Further, the proposed deletion of the expired footnote in Table 3.3-2, related to response time testing, is considered an administrative action.

4.0 SUMMARY

The staff has reviewed the licensee's proposed revisions to TS Sections 3.3.1.1 and 3.3.2.1 and has found that the accident analysis results of UFSAR Chapter 15 would not be exceeded by the proposed change. Therefore, the staff finds the proposed revisions to be acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

6.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to 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 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 the amendments involve no significant hazards consideration and there has been no pubic comment on such finding (57 FR 11117). Accordingly, the 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 the amendments.

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

Principal Contributor: F. Rinaldi

Date: May 15, 1992