DCS-MS-016

Docket No. 50-339

OCT 19 1983

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Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 32 to Facility Operating License No. NPF-7 for the North Anna Power Station, Unit No. 2 (NA-2). The amendment revises the NA-2 Technical Specifications in response to your letters dated June 8, 1982 (Serial No. 327) and May 3, 1983 (Serial No. 327A) and in our discussions with you regarding this matter. The amendment is effective within 30 days of the date of issuance.

The amendment revises the average reactor coolant system temperature from 580.3°F to 582.8°F at the currently licensed thermal power level of 2775 MWt. The approved increase of 2.5°F in the average reactor coolant system temperature implements Phase I of your NA-1&2 Plant Upgrade Program.

A copy of the Safety Evaluation is enclosed. The notice of issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

Original signed by

Leon B. Engle, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 32 to NPF-7
2. Safety Evaluation

cc: See next page

Also Immediately before signing check for Petitions & Comments If any come back to OELD

8311080024 831019
PDR ADOCK 05000339
P PDR

Signed CMM 10/19/83

See note re SE

ORB#3:DL	ORB#3:DL	ORB#3:DL	AD:OR:DL	OELD
PMKreutzer	LEngle/pn	JRMiller	GCLainas	
10/4/83	10/7/83	10/11/83	10/11/83	10/12/83

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. NPF-7

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

   A. The applications for amendment by Virginia Electric and Power Company (the licensee) dated June 8, 1982 and May 3, 1983 comply 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. 32 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for Charles M Trammell

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 19, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 32 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.

Pages

2-8
2-9
2-10
3/4 2-16

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	$\geq$ 18% of narrow range instrument span-each steam generator	$\geq$ 17% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$\leq$ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 25% of narrow range instrument span--each steam generator	$\leq$ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pump Busses	$\geq$ 2905 volts-each bus	$\geq$ 2870 volts-each bus
16. Underfrequency-Reactor Coolant Pump Busses	$\geq$ 56.1 Hz - each bus	$\geq$ 56.0 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	$\geq$ 45 psig	$\geq$ 40 psig
B. Turbine Stop Valve Closure	$\geq$ 1% open	$\geq$ 0% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left[\frac{1+\tau_1 S}{1+\tau_2 S}\right](T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 582.8°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 25 secs, τ_2 = 4 secs.

S = Laplace transform operator (sec^{-1})

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.141	K_1 = ( )	K_1 = ( )
K_2 = 0.0128	K_2 = ( )	K_2 = ( )
K_3 = 0.000608	K_3 = ( )	K_3 = ( )

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 35 percent and + 7 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 35 percent, the ΔT trip setpoint shall be automatically reduced by 1.58 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 7 percent, the ΔT trip setpoint shall be automatically reduced by 1.24 percent of its value at RATED THERMAL POWER.

*Values dependent on NRC approval of ECCS evaluation for these operating conditions.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o[K_4 - K_5 \left[\frac{\tau_3 S}{1+\tau_3 S}\right] T - K_6 (T-T'') - f_2(\Delta I)]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T'' = Indicated T_{avg} at RATED THERMAL POWER $\leq 582.8°F$.

K_4 = 1.088

K_5 = 0.02/°F for increasing average temperature

K_5 = 0 for decreasing average temperatures

K_6 = 0.00119 for $T > T''$; $K_6 = 0$ for $T \leq T''$

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 secs.

S = Laplace transform operator (sec^{-1})

$f_2(\Delta I)$ = 0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent span.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops In Operation	LIMITS: 2 Loops In Operation** & Loop Stop Valves Open	LIMITS: 2 Loops In Operation** & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$587°F		
Pressurizer Pressure	$\geq$2205 psig*		
Reactor Coolant System Total Flow Rate	$\geq$278,400 gpm		

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

**Values dependent on NRC approval of ECCS evaluation for these conditions.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 32 TO FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NO. 2

DOCKET NO. 50-339

Introduction

By letter dated June 8, 1982 (Serial No. 327), the Virginia Electric and Power Company (the licensee) requested amendments to Facility Operating Licenses No. NPF-4 and No. NPF-7 for the North Anna Power Station, Units No. 1 and No. 2 (NA-1&2).

The licensee's amendment request would implement Phase I of a Plant Upgrade Program for NA-1&2. Phases I and II of the Upgrade Program consist of implementing a steam pressure increase to maximize the electrical output at the currently licensed thermal power level. Completion of Phases I and II would be followed by implementation of Phase III, a core thermal power uprating program.

The licensee's June 8, 1982 request would implement Phase I by revising the NA-2 Technical Specifications to allow operation with a Reactor Coolant System (RCS) average temperature (T_{av}) of 582.8 degrees Fahrenheit (°F) as opposed to the currently approved RCS T_{av} of 580.3°F. This 2.5°F increase in T_{av} will provide an increase in the secondary side steam pressure of 18 pounds per square inch (psi) resulting in a higher secondary cycle thermal efficiency and a 2 Megawatt electrical (MWe) increase in electrical output.

On October 4, 1982, Phase I of the Plant Upgrade Program was implemented at NA-1 with the issuance of Amendment No. 42 to Facility Operating License No. NPF-4. Although our Safety Evaluation supporting Amendment No. 42 stated that we found the Phase II Upgrade to be applicable to both NA-1&2, the issuance of a identical amendment for NA-2 was held in abeyance until the licensee could implement secondary steam line support modifications to support the 2.5°F uprating for NA-2.

By letter dated May 3, 1983 the licensee stated that the secondary steam line support modification had been completed at NA-2 to support the NA-2 Phase I Upgrade Program. Therefore, we are issuing the Phase I Upgrade for NA-2 at this time.

Due to the passage of time since first approved for NA-1&2 and specifically implemented for NA-1 on October 4, 1982, we are restating our safety evaluation as originally provided for NA-1&2 to support the Phase I Upgrade for NA-2 at this time. Our original discussion and evaluation in addition to our comments on the NA-2 secondary steam line support modifications is provided below.

Discussion

The licensee has provided safety evaluations in order to provide a technical basis that the proposed increase in the RCS T_{av} does not involve any unreviewed safety question in accordance with 10 CFR Part 50.59. The safety evaluations included the scope of the Nuclear Steam Supply System (NSSS), the Balance of Plant (BOP), and the Turbine-Generator System.

Section 15.1.2.2 of the NA-1&2 Final Safety Analysis Report (FSAR) indicates that the original design bases for the accident analyses included a 2.5°F additional allowance on temperature. The additional allowance, without invalidating any accident analysis, calls for steady state operation at nominal average temperatures up to 2.5°F greater than the design value of 580.3°F. All accident analyses were performed at either the design RCS T_{av} of 580.3°F plus 6.5°F (586.8°F) or at 580.3°F -4°F, whichever is more conservative.

An uncertainty of plus or minus (±) 4°F is required to envelope temperature and control uncertainties. Therefore, the existing FSAR analysis is adequate for operation at 582.8°F +4°F. For transients postulated to initiate at "No Load" conditions, the docketed temperature of 540°F remains unchanged. In summary, the docketed NA-1&2 FSAR accident analyses envelopes NSSS full power operations at 2785 Megawatts thermal (MWt) with a RCS T_{av} of 582.8°F.

All the TS data are appropriate for an RCS T_{av} of 582.8°F except for the overtemperature and overpower ΔT setpoints and minor changes incorporating the higher RCS T_{av}. The calculation of the currently licensed overpower and overtemperature ΔT setpoints and associated constants was based on a nominal RCS average temperature of 580.3°F at 2775 MWt. The licensee has performed analyses to determine the overpower and overtemperature ΔT setpoints for an RCS average temperature of 582.8°F. Also, the licensee has performed confirmatory analyses to verify that the revised constants and resulting setpoints are appropriate and provide adequate protection against Departure from Nucleate Boiling (DNB). The new setpoints and associated changes will be incorporated in the utility Precautions, Limitations and Setpoints (PLS) document and plant procedures.

Evaluation

We have reviewed the NA-1&2 FSAR and the licensee submittal justifying a 2.5°F increase in the RCS T_{av}. From our review we have determined that the increase is within the limits assumed in the docketed FSAR accident and transient analyses and, therefore, is acceptable. Thus, we find full power operation at the currently licensed thermal power level (2775 MWt) with an average RCS temperature of 582.8°F to be acceptable. Also, we have reviewed the TS changes associated with the NA-1&2 Phase I Upgrade Program and we find these changes acceptable.

By letter dated May 3, 1983, the licensee stated that the main steam line monoball support modifications which were found necessary to support the 2.5°D uprating at NA-2 had been completed. We requested that Region II inspection verify the completion of these modifications at NA-2. Verification for completion of these modifications is so stated in Inspection Report 50-339/8-11 dated July 1, 1983.

Therefore, based on all of the above, we find implementation of the 2.5°F increase in T_{av} (Phase I Upgrade) to be acceptable for NA-2.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. This amendment increases the efficiency of NA-2 to produce slightly greater available electrical power without changing the authorized core thermal power. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

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

Date: October 19, 1983

Principal contributor:
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