September 7, 1988

Docket No. 50-338

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Mr. D. S. Cruden
Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Cruden:

SUBJECT: NORTH ANNA UNIT 1 - ISSUANCE OF AMENDMENT RE: AXIAL FLUX DIFFERENCE BANDS (TAC NO. 67994)

The Commission has issued the enclosed Amendment No. 105 to Facility Operating License No. NPF-4 for the North Anna Power Station, Unit No. 1 (NA-1). The amendment revises the Technical Specifications (TS) in response to your letter dated January 14, 1988.

This amendment changes the NA-1 TS to allow the widening of the axial flux difference bands from the current ±5% about a target value to +6% to -15% at 100% power and +20% to -28% at 50% power. The implementation of the relaxed power distribution control methodology is intended for application in the latter part of the NA-1 Fuel Cycle. The amendment is effective for forthcoming fuel cycles based on VEPCO's submittal of the NA-1 core surveillance report to the NRC on a cycle-by-cycle basis (Cycle 7, Cycle 8, etc.).

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

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 105 to NPF-4
2. Safety Evaluation

cc w/enclosures:
See next page

| LA:PDII-2 | PM:PDII-2 | D:PDII-2 | RSBX | OGC w/ noted revisions to SE; check review branch STATE bef. issuance |
|---|---|---|---|---|
| DMiller | LEngle:bg | HBerkow | WHodges | MYoung |
| 8/8/88 | 8/15/88 | 8/17/88 | 8/17/88 | 8/24/88 |

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PDR ADOCK 05000338
P PNU

Mr. D. S. Cruden
Virginia Electric & Power Company

North Anna Power Station
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. 105
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 January 14, 1988, 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. 105 , 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 issuance and shall be implemented within 14 days.

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: September 7, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 105

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.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

## INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

## 3/4.2 POWER DISTRIBUTION LIMITS

### AXIAL FLUX DIFFERENCE (AFD)

### LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the allowed operational space defined by Figure 3.2-1.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the Figure 3.2-1 limits,

   1. Either restore the indicated AFD to within the Figure 3.2-1 limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the Figure 3.2-1 limits.

LIMITING CONDITION FOR OPERATION (Continued)

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limit when at least 2 OPERABLE excore channels are indicating the AFD to be outside of the limit shown in Figure 3.2-1.

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Figure 3.2-1 is provided in the Core Surveillance Report as per Technical Specification 6.9.1.7

Figure 3.2-1 - Axial Flux Difference Limits as a Function of Rated Thermal Power

## HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

### LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.15}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.30] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower $\Delta T$ Trip Setpoint (value of $K_4$) has been reduced at least 1% (in $\Delta T$ span) for each 1% $F_Q(Z)$ exceeds the limit.

b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

## SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.15}{P \times N(z)} \times K(z) \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.15}{N(z) \times 0.5} \times K(z) \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainties, 2.15 is the $F_Q$ limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and N(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Core Surveillance Report as per Specification 6.9.1.7.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding the THERMAL POWER at which $F_Q(z)$ was last determined by 10% or more of RATED THERMAL POWER*, or

2. At least once per 31 effective full power days, whichever occurs first.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left( \frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

*During power escalation, the power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

1. $F_Q^M(z)$ shall be increased by 2% over that specified in 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 effective full power days until 2 successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left( \frac{F_Q^M(z)}{K(z)} \right) \quad \text{is not increasing.}$$

f. With the relationships specified in 4.2.2.2.c above not being satisfied:

1. Calculate the percent $F_Q(z)$ exceeds its limit by subtracting one from the measurement/limit ratio and multiplying by 100:

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left( \frac{F_Q^M(z)}{\frac{2.15}{P \times N(z)} \times (K(z))} \right) -1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \underset{\text{over } z}{\text{maximum}} \left( \frac{F_Q^M(z)}{\frac{2.15}{0.5 \times N(z)} \times (K(z))} \right) -1 \right\} \times 100 \text{ for } P < 0.5$$

2. Either of the following actions shall be taken:

   a. Power operation may continue provided the AFD limits of Figure 3.2-1 are reduced 1% AFD for each percent $F_Q(z)$ exceeded its limits, or

   b. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above.

g. The limits specified in 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 15 percent inclusive.

2. Upper core region 85 to 100 percent inclusive.

4.2.2.3 When $F_Q(z)$ is measured for reasons other then meeting the requirements of Specification 4.2.2.2, an overall measured $F_Q(z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

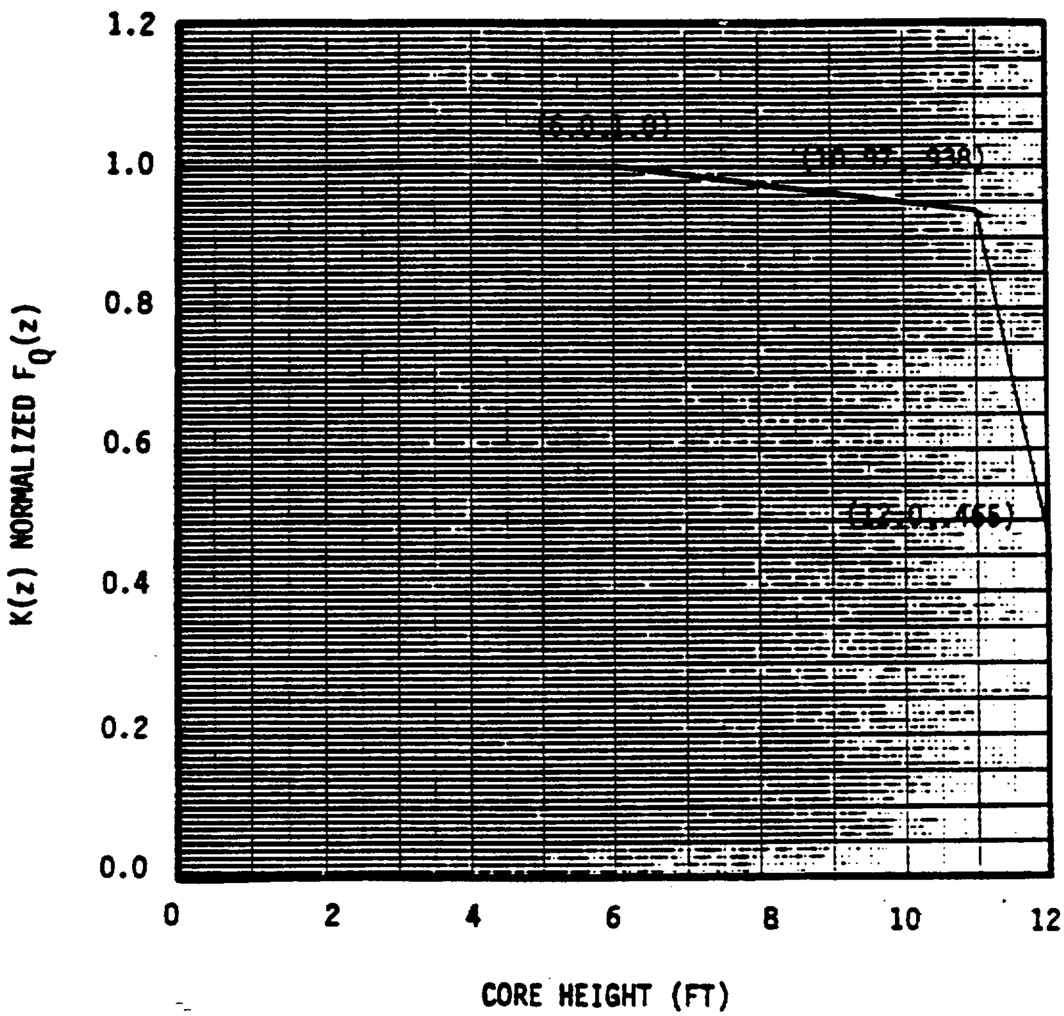


Figure 3.2-2 NORMALIZED $F_Q(z)$ AS A FUNCTION OF CORE HEIGHT

## TABLE 3.2-1

## DNB PARAMETERS

| PARAMETER | LIMITS | | |
|---|---|---|---|
| | 3 Loops in Operation | 2 Loops in Operation** & Loop Stop Valves Open | 2 Loops in Operation** & Isolated Loop Stop Valves Closed |
| Reactor Coolant System $T_{avg}$ | $\leq$ 591$^{o}$F | | |
| Pressurizer Pressure | $\geq$ 2205 psig* | | |
| Reactor Coolant System Total Flow Rate | $\geq$ 289,200 gpm | | |

---

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

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

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## SHUTDOWN MARGIN

### LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, initiate and continue boration at $\geq$ 10 gpm of at least 12,950 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor sub-critical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 10 gpm of at least 12,950 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

### SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod that is not fully inserted shall be demonstrated capable of full insertion when tripped from at least 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

## GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

### LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained $\leq$ 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

### SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be $\leq$ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

## 3/4.2 POWER DISTRIBUTION LIMITS

### BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core from going beyond the design limit DNBR during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

### 3/4 2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope, as given in Specification 3.2.2, is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the allowed $\Delta$I-power operating space and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

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## 3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS- $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than $\pm$ 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an $F_Q$ measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

The specified limit for $F^N_{\Delta H}$ contains a 4% error allowance. Normal operation will result in a measured $F^N_{\Delta H} \leq 1.49$. The 4% allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, effect $F^N_{\Delta H}$ more directly than $F_Q$,

b. although rod movement has a direct influence upon limiting $F_Q$ to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup physics tests can be compensated for in $F_Q$ by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset this reduction in the margin available between the safety analysis design DNBR values (1.57 and 1.59 for thimble and typical cells, respectively) and the limiting design DNBR values (1.39 for thimble cells and 1.42 for typical cells). The applicable value of rod bow penalties can be obtained from the FSAR.

The hot channel factor $F_Q^M(Z)$ is measured periodically and increased by a cycle and height dependent power factor, N(Z), to provide assurance that the limit on the hot channel factor, $F_Q(Z)$, is met. N(Z) accounts for the non-equilibrium effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The N(Z) function for normal operation is provided in the Core Surveillance Report per Specification 6.9.1.7.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on $F_Q$ is reinstated by reducing the power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. The two sets of 4 symmetric thimbles is a unique set of 8 detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11 and N-8.

## 3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. Measurement uncertainties must be accounted for during the periodic surveillance.

The 12 hour periodic surveillance of these parameters thru instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE PARTS MONITORING SYSTEM

OPERABILITY of the Loose Parts Monitoring System provides assurance that loose parts within the RCS will be detected. This capability is designed to ensure that loose parts will not collect and create undesirable flow blockages.

## 3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

## 3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

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CORE SURVEILLANCE REPORT

6.9.1.7 The N(Z) function for normal operation and the Axial Flux Difference limits (T.S. Figure 3.2-1) shall be provided to the NRC in accordance with the applicable provisions of 10 CFR 50.4 at least 60 days prior to cycle initial criticality unless otherwise approved by the Commission by letter. In the event that this information would be submitted at some other time during the core life, it shall be submitted 60 days prior to the date the information would become effective unless otherwise approved by the Commission by letter.

Any information needed to support N(Z) and/or the Axial Flux Difference limits will be by request from the NRC and need not be included in this report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.8 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate) with preoperational studies, operational controls, and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 4.12-1 and discussion of all analyses in which the LLD required by Table 4.12-3 was not achievable.

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

## SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.9 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report shall be submitted within 60 days after January 1 of each year. This report shall include an assessment of the radiation doses to the maximum exposed MEMBERS OF THE PUBLIC due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. Annual Meteorological data collected over the previous year shall be in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This meteorological data shall be retained in a file on site and shall be made available to the NRC upon request. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in the OFFSITE DOSE CALCULATION MANUAL (ODCM). Concurrent meteorological conditions or historical annual average atmospheric dispersion conditions shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

If the dose to the maximum exposed MEMBER OF THE PUBLIC due to the radioactive liquid and gaseous effluents from the station during the previous calendar year exceeds twice the limits of Specifications 3.11.1.2a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, the dose assessment shall include the contribution from direct radiation. The dose to the maximum exposed MEMBER OF THE PUBLIC shall show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operations.

The Radioactive Effluent Release Reports shall include a list of unplanned releases as required to be reported in Section 50.73 to 10 CFR Part 50 from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATED TO AMENDMENT NO. 105

## FACILITY OPERATING LICENSE NO. NPF-4

## VIRGINIA ELECTRIC AND POWER COMPANY

## OLD DOMINION ELECTRIC COOPERATIVE

## NORTH ANNA POWER STATION, UNIT NO. 1

## DOCKET NO. 50-338

### INTRODUCTION:

By letter dated January 14, 1988, the Virginia Electric and Power Company (the licensee) requested a change to the Technical Specifications (TS) for the North Anna Power Station, Unit No. 1 (NA-1). The licensee's proposed change is related to the "Relaxed Power Distribution Control Methodology" (RPDC). The methodology was described in topical report VEP-NE-1 submitted by the licensee for review on December 10, 1984. The staff has reviewed the report and concluded that it is acceptable. The proposed changes would allow the widening of the axial flux difference bands from the current $\pm 5\%$ about a target value to +6% to -15% at 100% power and +20% to -28% at 50% power. The implementation of the proposed changes is intended to be implemented during the latter part of the NA-1 Fuel Cycle No. 7. The proposed changes are effective for forthcoming fuel cycles (Cycle 7, Cycle 8, etc.) based on the licensee's submittal of the NA-1 core surveillance report on a cycle-by-cycle basis. An identical amendment was reviewed, approved and issued on April 14, 1986 for NA-2 (Amendment No. 64) and NA-2 has a design similar to NA-1.

### EVALUATION

The affected sections of the Technical Specifications are:

1. 3/4.2.1, B3/4.2.1 and 3.10.2: Replacement of Constant Axial Offset Control (CAOC) Axial Flux Difference Limits with RPDC Limits.

2. 3.2.2a: Deletion of the requirement to place the reactor in at least hot standby to reduce the overpower $\Delta T$ trip setpoint.

3. 3.2.2a.2, 3.2.6, 3/4.3.3.8, B3/4.2.6, B3/4.3.3.8, 6.9.1.7: Removal of all references to the Axial Power Distribution Monitoring System.

4. 4.2.2, B3/4.2, B3/4.2.3, 6.9.1.7: Replacement of $F_{xy}$ Surveillance Requirement with $F_Q$ Surveillance.

5. 6.9.1.7: Modification of the Core Surveillance Report.

Each of the proposed changes is discussed below:

TS 3/4.2.1, B3/4.2.1 and 3.10.2: Replacement of CAOC Axial Flux Difference Limits with RPDC Limits

In these TS sections, all references to the axial flux difference for the CAOC operating strategy would be deleted and replaced with the axial flux difference limits required in the RPDC methodology. In the action statement, the requirement to restore the axial flux difference to the indicated value within 15 minutes would be retained. If this requirement is not met, power must be reduced to less than 50% within 30 minutes. The new TS (Section 3.2.2) assures that the $F_Q$ will not exceed the specified limits, nor will the axial flux distribution fall outside the range ensuring adequate protection from the overtemperature and overpower ΔT. The special test exception of Section 3.10.2 would be removed, and thus, the axial flux difference limits would apply during the performance of physics tests. The TS is identical to the one proposed in VEP-NE-1 which has been approved and, therefore, is acceptable.

TS 3.2.2a: Deletion of Requirement to Place the Reactor in at Least Hot Standby to Reduce the Overpower ΔT Trip Setpoints

One of the action items in 3.2.2(a) requires reduction of the overpower ΔT trip setpoint by 1% for each 1% the $F_Q(Z)$ exceeds the limit. The requirement to place the reactor in hot standby in order to reduce the overpower ΔT trip setpoint would be deleted since the reduction can be performed one channel at a time while at power without exceeding specified limits. The deletion of the hot standby requirement is part of the proposed and approved TS in VEP-NE-1, hence, it is acceptable.

TS 3.2.2a.2, 3.2.6, 3/4.3.3.8, B3/4.2.6, B3/4.3.3.8, 6.9.1.7: Removal of all References to the Axial Power Distribution Monitoring System.

Under the RPDC operating methodology, the operating limits on axial offset are established to ensure that the $F_Q$ loss of coolant accident (LOCA) limit is not exceeded. The change of the axial flux difference envelope is now the essential variable which is subject to cycle-by-cycle analytic verification. The revised specifications would account for potential $F_Q$ violations which could occur under nonequilibrium conditions by narrowing the change of the axial flux difference. Therefore, the axial power distribution monitoring system would not be needed to maintain safety limits and could be eliminated. The axial power distribution monitoring has been eliminated from the proposed and approved specification in VEP-NE-1, and hence, this change is acceptable.

TS 4.2.2, B3/4.2, B3/4.2.3, 6.9.1.7: Replacement of $F_{xy}$ Surveillance Requirement with $F_Q$ Surveillance.

The revised specifications would require a direct measurement of $F_Q$ at least once per 31 effective full power days. The measured $F_Q$ would then be increased by the nonequilibrium factor N(Z) to account for power distribution transient during normal operation. Since the $F_Q$ is measured directly, the requirement for $F_{xy}$ surveillance would no longer be needed.

TS 6.9.1.7: Modification of the Core Surveillance Report.

As discussed above, the $F_Q$ surveillance requires the use of N(Z) as a cycle-specific multiplier to incorporate nonequilibrium effects. The core surveillance report provides this function on a cycle-by-cycle basis. This would replace the requirement to provide the $F_{xy}$ limit and the power level.

## SUMMARY

The staff has reviewed the information presented in the request for the NA-1 TS related to the adoption of the relaxed power distribution control methodology and intended for application in the last part of the NA-1 cycle 7. An identical amendment was issued on April 14, 1986 for NA-2. The methodology described in the report VEP-NE-1 has been reviewed and approved by the staff. The proposed Technical Specification changes are identical with those approved in report VEP-NE-1. The surveillance requirements have been adjusted to the new proposed specification. In addition, the licensee has performed cycle-specific analyses to ascertain that the $F_Q$ values are within the allowable limits for overtemperature overpower protection. Therefore, the proposed NA-1 TS changes are acceptable and can be applied to the latter part of the NA-1 cycle 7 and for forthcoming fuel cycles based on the licensee's submittal of the NA-1 core surveillance report to the NRC on a cycle-by-cycle basis.

## ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes surveillance requirements. The staff has determined that the amendment involves 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 published a proposed finding that the amendment involves no significant hazards considerations and there has been no public comment on such finding. 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 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: September 7, 1988

Principal Contributor:

L. Engle