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DATED: June 7, 1991

AMENDMENT NO. 146 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1
AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
L. Engle
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
G. Hill (8), P-137
Wanda Jones, P-130A
C. Grimes, 11/F/23
T. Huang, 8-E-23
ACRS (10)
GPA/PA
OC/LFMB
PD Plant-specific file [Gray File]
M. Sinkule, R-II
Others as required

cc: Plant Service list

Posted
Amdt. 130 to NPF-7
(See Correction letter of
6-25-91)

Docket Nos. 50-338
and 50-339

Posted
Amdt. 130 to NPF-7

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: CORE OPERATING LIMITS REPORT (TAC NOS. 76828 AND 76829)

The Commission has issued the enclosed Amendment Nos. 146 and 130 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 March 29, 1990, as supplemented May 8, August 8, and October 30, 1990.

The amendments modify TS having cycle-specific parameters limits by replacing the value of those limits with a reference to a Core Operating Limits Report (COLR). Inclusion of your proposed boron concentration in the COLR is unacceptable at this time. We are currently sponsoring research effort to determine if the floor value of the boron concentration should be retained or eliminated. Our review of this proposed TS change in conjunction with the NA-1&2 lead plant submittal for the new standard TS will continue. The enclosed Notice of Denial has been forwarded to the Office of the Federal Register for publication.

In addition, the licensee previously submitted a core surveillance report prior to the last refueling outage in accordance with the previous TS. Therefore, the licensee need not submit a COLR to the NRC for restart from this previous refueling outage.

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,
(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. 146 to NPF-4
2. Amendment No. 130 to NPF-7
3. Safety Evaluation
4. Notice of Denial

cc w/enclosures: See next page

*SEE PREVIOUS CONCURRENCE

OFC	LA:PD22	:PM:PD22	:D:PD22	:OGC*	:	:
NAME	:DMiller	:LEngle/kdj	:HBerkow	:MYoung	:	:
DATE	:6/7/91	:6/7/91	:6/7/91	:5/31/91	:	:

OFFICIAL RECORD COPY
Document Name: NA 76828/29

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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. 146
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 March 29, 1990, as supplemented May 8, August 8, and October 30, 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 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. 146 , 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Rajender Auluck

for 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: June 7, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 146

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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The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, expressed in % of RATED THERMAL POWER between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

1.6.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1,

1.6.2 All equipment hatches are closed and sealed,

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7. Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment. The system is composed of the waste gas decay tanks, regenerative heat exchanger, waste gas charcoal filters, process vent blowers, waste gas surge tanks and waste gas diaphram compressor.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postman who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

OFFSITE DOSE CALCULATION MANUAL

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specifications 6.9.1.8 and 6.9.1.9.

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SLAVE RELAY TEST

1.30 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.33 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.34 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.35 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY where access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.36 A VENTILATION EXHAUST TREATMENT SYSTEM is the system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.37 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

BORON DILUTION

VALVE POSITION

LIMITING CONDITION FOR OPERATION

3.1.1.3.2 The following valves shall be locked, sealed or otherwise secured in the closed position except during planned boron dilution or makeup activities

a. 1-CH-217 or

b. 1-CH-220, 1-CH-241, FCV-1114B and FCV-1113B.

APPLICABILITY: MODES 3, 4, 5, and 6

ACTION:

With the above valves not locked, sealed or otherwise secured in the closed position:

a. In MODES 3 and 4 be in COLD SHUTDOWN within 30 hours

b. In MODES 5 and 6 suspend all operations involving positive reactivity changes or CORE ALTERATIONS and lock, seal or otherwise secure the valves in the closed position within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.3.2 The above listed valves shall be verified to be locked, sealed or otherwise secured in the closed position within 15 minutes after a planned boron dilution or makeup activity.

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be $\leq 0.6 \times 10^{-4}$ $\Delta k/k/°F$ below 70 percent RATED THERMAL POWER and $\leq 0.0 \times 10^{-4}$ $\Delta k/k/°F$ at or above 70 percent RATED THERMAL POWER.

APPLICABILITY: Beginning of Cycle (BOC) Limit - MODES 1 and 2* only #
End of Cycle (EOC) Limit - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOC limit specified in the CORE OPERATING LIMITS REPORT:

1. Establish and maintain control rod withdrawal limits sufficient to restore the MTC to within its limit within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. With the MTC more negative than the EOC limit specified in the CORE OPERATING LIMITS REPORT, be in HOT SHUTDOWN within 12 hours.

---

*With $K_{eff} \geq 1.0$

#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOC limit specified in the CORE OPERATING LIMITS REPORT, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the CORE OPERATING LIMITS REPORT (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicated the MTC is more negative than the 300 ppm surveillance limit, the MTC shall be remeasured, and compared to the EOC MTC limit specified in the CORE OPERATING LIMITS REPORT, at least once per 14 EFPD during the remainder of the fuel cycle.(1)

---

(1) Once the equilibrium boron concentration (all rods withdrawn, RATED THERMAL POWER condition) is 60 ppm or less, further measurement of the MTC in accordance with 4.1.1.4.b may be suspended providing that the measured MTC at an equilibrium boron concentration of ≤ 60 ppm is less negative than the 60 ppm surveillance limit specified in the CORE OPERATING LIMITS REPORT.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be $\leq$ 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq 500°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to:

   1. $\leq$ 66% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open, or

   2. $\leq$ 71% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the CORE OPERATING LIMITS REPORT, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Restore the rod to within the insertion limit specified in the CORE OPERATING LIMITS REPORT, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the CORE OPERATING LIMITS REPORT

a. Within 15 minutes prior to initial control rod bank withdrawal during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With $K_{eff} \geq 1.0$

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the insertion limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group step counter demand position using the insertion limits specified in the CORE OPERATING LIMITS REPORT, or

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify either the individual rod positions (indicated positions) or the group step counter demand position of each rod group to be within the insertion limits at least once per 4 hours.

---

*See Special Test Exceptions 3.10.2 and 3.10.3.

#With $K_{eff} \geq 1.0$.

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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 limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the CORE OPERATING LIMITS REPORT,

   1. Either restore the indicated AFD to within the 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 limits specified in the CORE OPERATING LIMITS REPORT.

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 limits specified in the CORE OPERATING LIMITS REPORT.

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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:

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

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

where CFQ = the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT,

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}, \text{ and}$$

K(Z) = the normalized F_Q limit as a function of core height specified in the CORE OPERATING LIMITS REPORT.

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 ΔT Trip Setpoint (value of K_4) has been reduced at least 1% (in ΔT 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 limits.

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:

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

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

where $F_Q^M(z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, and N(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is specified in the CORE OPERATING LIMITS 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

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

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:

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

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

2. Either of the following actions shall be taken:

a. Power operation may continue provided the AFD limits of Specification 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 than 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.

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NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq CFDH\ [1 + PFDH\ (1-P)]$$

where CFDH = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT,

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

PFDH = The Power Factor Multiplier for $F^N_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT, and

$F^N_{\Delta H}$ = measured value of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.77% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} <200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal. A 1.77% $\Delta k/k$ shutdown margin provides adequate protection for the boron dilution accident.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9957 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed for this parameter in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value was obtained by incrementally correcting the MTC used in the FSAR analyses to nominal operating conditions. These corrections involved adding the incremental change in the MTC associated with a core condition of Bank D inserted to an all rods withdrawn condition and an incremental change in MTC to account for measurement uncertainty at RATED THERMAL POWER conditions. These corrections result in the End of Cycle (EOC) MTC limit. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the EOC MTC limit.

Once the equilibrium boron concentration falls below about 60 ppm, dilution operations take an extended amount of time and reliable MTC measurements become more difficult to obtain due to the potential for fluctuating core conditions over the test interval. For this reason, MTC measurements may be suspended provided the measured MTC value at an equilibrium full power boron concentration ≤ 60 ppm is less negative than the 60 ppm surveillance limit. The difference between this value and the EOC-MTC limit conservatively bounds the maximum credible change in MTC between the 60 ppm equilibrium boron concentration (all rods withdrawn, RATED THERMAL POWER conditions) and the licensed end-of-cycle, including the effect of boron concentration, burnup, and end-of-cycle coastdown.

The surveillance requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, and 3) the P-12 interlock is above its setpoint, and 4) compliance with Appendix G to 10 CFR Part 50 (see Bases 3/4.4.9).

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

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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 ± 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 $F^N_{\Delta H}$ less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT. 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 (1.46 for Virginia Electric and Power Company statistical methods) and the limiting design DNBR value (1.26 for Virginia Electric and Power Company statistical methods). A discussion of the rod bow penalty is presented in the FSAR.

The hot channel factor $F^M_Q(Z)$ 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 specified in the CORE OPERATING LIMITS 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 have been analytically demonstrated to be adequate to maintain a minimum DNBR greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin.

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.

CORE OPERATING LIMITS REPORT

6.9.1.7.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOC and EOC limits, and 300 ppm and 60 ppm surveillance limits for Specification 3/4.1.1.4,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference limits for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, K(Z), N(Z) for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier, for Specification 3/4.2.3.

6.9.1.7.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as identified in 6.9.1.7.e.

6.9.1.7.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.7.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.1.7.e REFERENCES

1. VEP-FRD-42, Rev. 1-A, "Reload Nuclear Design Methodology," September 1986.

   (Methodology for LCO 3.1.1.4 - Moderator Temperature Coefficient, LCO 3.1.3.5 - Shutdown Bank Insertion Limit, LCO 3.1.3.6 - Control Bank Insertion Limits, LCO 3.2.2 - Heat Flux Hot Channel Factor, LCO 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

Correction letter 66 6-25-91

2a. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL - 1981 VERSION", February 1982 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2b. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS - SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL", JULY, 1986, (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2c. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March 1987 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2d. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2e. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code", August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology", June 1987.

(Methodology for LCO 3.2.3, Nuclear Enthalpy Rise Hot Channel Factor.)

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code", July 1990.

(Methodology for LCO 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor).

4. VEP-NE-1-A, "Vepco Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," March 1986.

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor and LCO 3.2.1 - Axial Flux Difference.)

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
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 March 29, 1990, as supplemented May 8, August 8, and October 30, 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 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. 130 , 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Rajender Auluck

for 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: June 7, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 130

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.

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1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, expressed in % of RATED THERMAL POWER between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

1.6.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1,

1.6.2 All equipment hatches are closed and sealed,

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.6.5 The sealing mechanism associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.7 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.7. Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$- AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment. The system is composed of the waste gas decay tanks, regenerative heat exchanger, waste gas charcoal filters, process vent blowers, waste gas surge tanks and waste gas diaphram compressor.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postman who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

OFFSITE DOSE CALCULATION MANUAL

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specifications 6.9.1.8 and 6.9.1.9.

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2893 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SLAVE RELAY TEST

1.30 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOURCE CHECK

1.31 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.33 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.34 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.35 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY where access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.36 A VENTILATION EXHAUST TREATMENT SYSTEM is the system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.37 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be $\leq 0.6 \times 10^{-4}$ $\Delta k/k/°F$ below 70 percent RATED THERMAL POWER and $\leq 0.0 \times 10^{-4}$ $\Delta k/k/°F$ at or above 70 percent RATED THERMAL POWER.

APPLICABILITY: Beginning of Cycle (BOC) Limit - MODES 1 and 2* only #
End of Cycle (EOC) Limit - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the BOC limit specified in the CORE OPERATING LIMITS REPORT, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within its limit within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. With the MTC more negative than the EOC limit specified in the CORE OPERATING LIMITS REPORT, be in HOT SHUTDOWN within 12 hours.

*With $K_{eff} \geq 1.0$

#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOC limit specified in the CORE OPERATING LIMITS REPORT, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the CORE OPERATING LIMITS REPORT (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicated the MTC is more negative than the 300 ppm surveillance limit, the MTC shall be remeasured, and compared to the EOC MTC limit specified in the CORE OPERATING LIMITS REPORT, at least once per 14 EFPD during the remainder of the fuel cycle.(1)

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(1) Once the equilibrium boron concentration (all rods withdrawn, RATED THERMAL POWER condition) is 60 ppm or less, further measurement of the MTC in accordance with 4.1.1.4.b may be suspended providing that the measured MTC at an equilibrium boron concentration of ≤ 60 ppm is less negative than the 60 ppm surveillance limit specified in the CORE OPERATING LIMITS REPORT.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to:

   1. Less than or equal to 66% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open, or

   2. Less than or equal to 71% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the CORE OPERATING LIMITS REPORT, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Restore the rod to within the insertion limit specified in the CORE OPERATING LIMITS REPORT, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the CORE OPERATING LIMITS REPORT

a. Within 15 minutes prior to initial control rod bank withdrawal during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.0

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the insertion limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group step counter demand position using the insertion limits specified in the CORE OPERATING LIMITS REPORT, or

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify either the individual rod positions (indicated positions) or the group step counter demand position of each rod group to be within the insertion limits at least once per 4 hours.

* See Special Test Exceptions 3.10.2 and 3.10.3.

\# With $K_{eff} \geq 1.0$

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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 limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the CORE OPERATING LIMITS REPORT,

   1. Either restore the indicated AFD to within the 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 limits specified in the CORE OPERATING LIMITS REPORT.

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.

AXIAL FLUX DIFFERENCE (AFD)

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.2 The indicated AFD shall be considered outside of its limit when at least two OPERABLE excore channels are indicating the AFD to be outside of the limits specified in the CORE OPERATING LIMITS REPORT.

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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:

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

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

where CFQ = the F_Q limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT,

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}, \text{ and}$$

K(Z) = the normalized F_Q limit as a function of core height specified in the CORE OPERATING LIMITS REPORT.

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 ΔT Trip Setpoints (value of K_4) have been reduced at least 1% (in ΔT 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.

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:

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

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

where $F_Q^M(z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, and N(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is specified in the CORE OPERATING LIMITS 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

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

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:

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

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

2. Either of the following actions shall be taken:

a. Power operation may continue provided the AFD limits of Specification 3.2.1 are reduced 1% AFD for each percent $F_Q(Z)$ exceeded its limit, 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 than 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.

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NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq CFDH\ [1 + PFDH\ (1-P)]$$

where CFDH = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the CORE OPERATING LIMITS REPORT,

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

PFDH = The Power Factor Multiplier for $F^N_{\Delta H}$ specified in the CORE OPERATING LIMITS REPORT, and

$F^N_{\Delta H}$ = measured value of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map.

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to

ACTION Continued

exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.77% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 GPM, as provided by either one RCP or one RHR pump as required by Specification 3.4.1.1, provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 9957 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control. The requirement that certain valves remain closed at all times except during planned boron dilution or makeup, activities provides assurance that an inadvertent boron dilution will not occur.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed for this parameter in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value was obtained by incrementally correcting the MTC used in the FSAR analyses to nominal operating conditions. These corrections involved adding the incremental change in the MTC associated with a core condition of Bank D inserted to an all rods withdrawn condition and an incremental change in MTC to account for measurement uncertainty at RATED THERMAL POWER conditions. These corrections result in the End of Cycle (EOC) MTC limit. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the EOC MTC limit.

Once the equilibrium boron concentration falls below about 60 ppm, dilution operations take an extended amount of time and reliable MTC measurements become more difficult to obtain due to the potential for fluctuating core conditions over the test interval. For this reason, MTC measurements may be suspended provided the measured MTC value at an equilibrium full power boron concentration ≤ 60 ppm is less negative than the 60 ppm surveillance limit. The difference between this value and the EOC-MTC limit conservatively bounds the maximum credible change in MTC between the 60 ppm equilibrium boron concentration (all rods withdrawn, RATED THERMAL POWER conditions) and the licensed end-of-cycle, including the effect of boron concentration, burnup, and end-of-cycle coastdown.

The surveillance requirements for measurement of the MTC at the beginning and near the end of each fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, and 3) the P-12 interlock is above its setpoint, 4) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 5) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

When $F^N_{\Delta H}$ is measured, 4% is the appropriate experimental error allowance for a full core map taken with the incore detection system. The specified limit for $F^N_{\Delta H}$ contains a 4% error allowance. Normal operation will result in a measured $F^N_{\Delta H}$ less than or equal to the limit specified in the CORE OPERATING LIMITS REPORT. 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 value (1.46 for Virginia Electric and Power Company statistical methods) and the limiting design DNBR value (1.26 for Virginia Electric and Power Company statistical methods). A discussion of the rod bow penalty is presented in the FSAR.

The hot channel factor $F^M_Q(Z)$ 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 specified in the CORE OPERATING LIMITS 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 movable 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 have been analytically demonstrated to be 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.

CORE OPERATING LIMITS REPORT

6.9.1.7.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOC and EOC limits, and 300 ppm and 60 ppm surveillance limits for Specification 3/4.1.1.4,
2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,
3. Control Bank Insertion Limits for Specification 3/4.1.3.6,
4. Axial Flux Difference limits for Specification 3/4.2.1,
5. Heat Flux Hot Channel Factor, K(Z), N(Z) for Specification 3/4.2.2, and
6. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier, for Specification 3/4.2.3.

6.9.1.7.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as identified in 6.9.1.7.e.

6.9.1.7.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.7.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.9.1.7.e REFERENCES

1. VEP-FRD-42, Rev. 1-A, "Reload Nuclear Design Methodology," September 1986.

   (Methodology for LCO 3.1.1.4 - Moderator Temperature Coefficient, LCO 3.1.3.5 - Shutdown Bank Insertion Limit, LCO 3.1.3.6 - Control Bank Insertion Limits, LCO 3.2.2 - Heat Flux Hot Channel Factor, LCO 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

Correction letter 06 6-25-91

2a. WCAP-9220-P-A. Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL - 1981 VERSION", February 1982 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2b. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS - SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL", JULY, 1986, (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2c. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March 1987 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2d. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2e. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code", August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology", June 1987.

(Methodology for LCO 3.2.3, Nuclear Enthalpy Rise Hot Channel Factor.)

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code", July 1990.

(Methodology for LCO 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor).

4. VEP-NE-1-A, "Vepco Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," March 1986.

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor and LCO 3.2.1 - Axial Flux Difference.)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 146 AND 130 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 March 29, 1990, as supplemented by letters dated May 8, August 8, and October 30, 1990, the Virginia Electric and Power Company (the licensee) proposed changes to the Technical Specifications (TS) for the North Anna Power Station Units 1 and 2 (NA-1&2). The proposed changes would modify TS having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of the NA-1&2 TS. Guidance on the proposed changes was developed by the NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988.

The May 8, August 8, and October 30, 1990 letters provided revised TS pages to correct editorial errors in the original submittal, minor changes in format as requested by the staff, and clarified the methodologies to be used in maintaining safety limits. These changes did not change the proposed no significant hazards consideration determination.

2.0 DISCUSSION

The licensee's proposed changes to the NA-1&2 TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definitions section of the TS is modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC-approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following TS were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) TS 3.1.1.4 and Surveillance Requirement (SR) 4.1.1.4

The moderator temperature coefficient (MTC) limits for this specification and for this surveillance requirement are specified in the COLR.

(b) TS 3.1.3.5 and SR 4.1.3.5

The shutdown bank insertion limit for this specification and for this surveillance requirement is specified in the COLR.

(c) TS 3.1.3.6

The control bank insertion limits for this specification are specified in the COLR.

(d) TS 3.2.1 and SR 4.2.1

The axial flux difference (AFD) limits for this specification and for this surveillance requirement are specified in the COLR.

(e) TS 3.2.2 and SR 4.2.2

The heat flux hot channel factor F_Q limit at rated thermal power, the normalized F_Q limit as a function of core height K(Z), and the height dependent power factor N(Z) for this specification and for this surveillance requirement are specified in the COLR.

(f) TS 3.2.3 and SR 4.2.3

The nuclear enthalpy rise hot channel factor limit at rated thermal power and the power factor multiplier for this specification and surveillance requirement are specified in the COLR.

The Bases of affected TS have been modified by the licensee to include appropriate references to the COLR. Based on our review, the NRC staff concludes that the changes to these Bases are acceptable.

(3) With respect to the proposed inclusion of boron concentration in the COLR, this is a cycle-specific parameter meeting the intent of Generic Letter 88-16 only if the staff agrees to the change in TS format by eliminating the fixed minimum value of boron concentration. The staff is currently sponsoring a research effort to determine if the floor value should be retained (and probably increased) or eliminated. Until that effort is complete, the staff's position is to retain the fixed minimum value of boron concentration. Therefore, the NRC staff finds the proposed

inclusion of TS 3.9.1 in the COLR to be unacceptable at this time. Our review of this proposed TS change in conjunction with the NA-1&2 lead plant submittal for the new Standard Technical Specifications will continue.

(4) TS 6.9.1.9 is revised to add the COLR to the reporting requirements of the Administrative Controls section of the TS. This TS requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this TS requires that the values of these limits be established using NRC-approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies and the affected TS are as follows:

(a) VEP-FRD-42, Rev. 1-A, "Reload Nuclear Design Methodology," September 1986.

(Methodology for LCO 3.1.1.4 - Moderator Temperature Coefficient, LCO 3.1.3.5 - Shutdown Bank Insertion Limit, LCO 3.1.3.6 - Control Bank Insertion Limits, LCO 3.2.2 - Heat Flux Hot Channel Factor, LCO 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and LCO 3.9.1 - Boron Concentration).

(b) WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

(c) WCAP-9561-P-A, Add. 3, Rev. 1 "BART A-1: A Computer Code for the Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling in W ECCS Evaluation Model," July 1986, (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

(d) WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

(e) WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

(f) WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

(g) VEP-NE-2-A, "Statistical DNBR Evaluation Methodology," June 1987.

(Methodology for LCO 3.2.3, Nuclear Enthalpy Rise Hot Channel Factor).

(h) VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code," July 1990.

(Methodology for LCO 3.2.3, Nuclear Enthalpy Rise Hot Channel Factor).

(i) VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," March 1986.

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor and LCO 3.2.1 - Axial Flux Difference).

Finally, the TS requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in the TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC-approved methodologies, the NRC staff concludes that these changes are administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes, excluding the boron concentration limit TS, are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR, excluding the boron concentration limit TS, are acceptable.

3.0 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.

4.0 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, change surveillance requirements, and change reporting and recordkeeping requirements. 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 public comment on such finding (55 FR 18416). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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

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: T. Huang

Date: June 7, 1990

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NOS. 50-338 AND 50-339

NOTICE OF DENIAL OF PORTION OF APPLICATION FOR

AMENDMENTS TO FACILITY OPERATING LICENSES

AND OPPORTUNITY FOR HEARING

The U. S. Nuclear Regulatory Commission has denied a portion of an amendment request by Virginia Electric and Power Company (the licensee), for amendments to Facility Operating License Nos. NPF-4 and NPF-7, issued to the licensee for operation of the North Anna Power Station, Unit Nos. 1 and 2, located in Louisa County, Virginia. Notice of Consideration of Issuance of these amendments was published in the FEDERAL REGISTER on May 2, 1990 (55 FR 18416).

The purpose of the licensee's amendment request was to revise the Technical Specifications (TS) by replacing the value of the cycle-specific parameter limits with a reference to the Core Operating Limits Report (COLR).

The NRC staff has concluded that the portion of the licensee's request regarding the inclusion of boron concentration in the COLR cannot be granted. The licensee was notified of the Commission's denial of the proposed change by letter dated June 7, 1991 .

By July 15, 1991 , the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D.C., 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C., by the above date. A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to Michael W. Maupin, Esq., Hunton and Williams, P.O. Box 355, Richmond, Virginia 23212, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendments dated March 29, 1990, as supplemented May 8, August 8, and October 30, 1990, and (2) the Commission's letter to the licensee dated June 7, 1991 .

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. and at the Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901. A copy of Item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, Attention: Document Control Desk.

Dated at Rockville, Maryland, this 7th day of June 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Rajender Auluck

Rajender Auluck, Acting Director
Project Directorate II-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation