February 27, 1996

Mr. J. P. O'Hanlon
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

SUBJECT: NORTH ANNA UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISIONS TO THE FACILITY OPERATING LICENSES AND TECHNICAL SPECIFICATIONS TO ALLOW THE CONTAINMENT PERSONNEL AIRLOCK DOORS TO REMAIN OPEN DURING REFUELING OPERATIONS (TAC NOS. M94076 AND M94078)

Dear Mr. O'Hanlon:

The Commission has issued the enclosed Amendment Nos. 198 and 179 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 Facility Operating Licenses and the Technical Specifications (TS) in response to your letter dated October 17, 1995, as supplemented by facsimile on February 26, 1996.

The amendments revise TS 3.9.4 to allow both containment personnel airlock (PAL) doors to remain open during fuel movements or core alterations provided that one door is operable and a designated individual is available to close the airlock door if a fuel handling accident should occur. The amendments also revise the TS to provide clarification of the design and licensing bases for systems required to mitigate the consequences of a Fuel Handling Accident. Further, the Facility Operating Licenses are modified to delete license condition 2.G for Unit 1 and 2.I for Unit 2.

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

Sincerely, Original signed by
Richard J. Clark

Leon B. Engle, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 198 to NPF-4
2. Amendment No. 179 to NPF-7
3. Safety Evaluation

cc w/enclosures: See next page FILENAME - G:\NA94076X.AMD
*Previously concurred

OFFICE	LA:PDII-1	DRPE	BC:SCSB*	BC:PERB*	OGC*	D:PDII-1
NAME	Dunnington	RClark	CBerlinger	CMiller	EHoller	EImbro
DATE	2/27/96	02/27/96	02/13/96	02/13/96	02/22/96	2/27/96
COPY	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

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Mr. J. P. O'Hanlon
Virginia Electric & Power Company

North Anna Power Station
Units 1 and 2

cc:
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North Anna Power Station
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Mr. Al Belisle
U.S. Nuclear Regulatory Commission
101 Marietta Street N. W. Suite 2900
Atlanta, Georgia 30323-0199

DATED: February 27, 1996

AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. NPF-4-NORTH ANNA UNIT 1
AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. NPF-7-NORTH ANNA UNIT 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 198
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 October 17, 1995, as supplemented February 26, 1996, 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 the following paragraphs under Section 2 of Facility Operating License No. NPF-4 are hereby amended to read as follows:

   2.D.2 <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. **198** , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.G Deleted

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Eugene V. Imbro, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: (1) Page 7 of License No. NPF-4*
(2) Changes to the Technical Specifications

Date of Issuance: **February 27, 1996**

* Page 7 is attached, for convenience, for the composite license to reflect these changes.

ATTACHMENT TO LICENSE AMENDMENT NO. 198

FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

LICENSE

Remove Page	Insert Page
7	7

TECHNICAL SPECIFICATIONS

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

Remove Pages	Insert Pages
3/4 9-4	3/4 9-4
B 3/4 8-1	B 3/4 8-1
B 3/4 9-1	B 3/4 9-1
B 3/4 9-3	B 3/4 9-3

2.E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "North Anna Power Station Physical Security Plan," with revisions submitted through February 24, 1988; "North Anna Power Station Guard Training and Qualification Plan," with revisions submitted through May 14, 1987; and "North Anna Power Station Safeguards Contingency Plan," with revisions submitted through January 9, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

2.F. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittal dated November 6, 1986 (Serial No. 86-477A).

2.G. Deleted.

2.H This license is effective as of the date of issuance and shall expire at midnight on April 1, 2018.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
R. C. DeYoung, for

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:
1. Construction Related Items to be completed prior to Initial Criticality
2. Appendices A and B Technical Specification page changes
3. Figure 1
4. Table 1

Date of Issuance: APR 1 1978

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 150 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 150 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 150 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts.

b. A minimum of one door in each airlock is closed, * and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or
2. Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, ** or

b. Testing the Containment Purge and Exhaust isolation valves and system per the applicable portions of Specifications 4.6.3.1.2 and 4.9.9.

---

* Both doors of the containment personnel airlock may be open provided:

a. One personnel airlock door is OPERABLE (i.e., the door is capable of being closed and that an individual is designated to close the door), and

b1. There is at least 23 feet of water above the top of the reactor pressure vessel flange during movement of fuel assemblies within the containment, or

b2. There is at least 23 feet of water above the top of irradiated fuel assemblies within the reactor pressure vessel during CORE ALTERATIONS excluding movement of fuel assemblies.

** If both doors of the containment personnel airlock are open pursuant to Specification 3.9.4.b above, one door shall be verified to be capable of being closed at the above surveillance frequency.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration of 2300 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon a lack of containment pressurization potential while in the REFUELING MODE.

OPERABILITY of the containment airlock door requires that the door is capable of being closed, that the door is unblocked and no cables or hoses are being run through the airlock, and that a designated individual is continuously available to close the airlock door. This individual must be stationed near the airlock.

3/4.9.5 COMMUNICATIONS

The requirement for communication capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.8.1 and 3/4.8.2 A.C. and D.C. POWER SOURCES AND DISTRIBUTION

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least one of each of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The ACTION requirements specified in Modes 5 and 6 address the condition where sufficient power is unavailable to recover from postulated events (i.e., fuel handling accident). Implementation of the ACTION requirements shall not preclude completion of actions to establish a safe conservative plant condition. Completion of the requirements will prevent the occurrence of postulated events for which mitigating actions would be required.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods, 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and 3) sufficient power is available for systems necessary to recover from postulated events in these MODES, e.g., the control room emergency ventilation system fans during a fuel handling accident.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants" Revision 1, August 1977, as modified by Amendment No. 83 issued August 22, 1986.

The Surveillance Requirements for demonstrating the OPERABILITY of the Emergency Diesel Generator batteries and the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance, Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std. 450-1980, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," as modified by Amendment No. 97 issued March 25, 1988.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies; 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped, 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

After the reactor has shutdown and entered into MODE 3 for at least 100 hours, a minimum RHR system flow rate of 2000 gpm in MODE 6 is permitted, provided there is sufficient decay heat removal to maintain the RCS temperature less than or equal to 140°F. Since the decay heat power production rate decreases with time after reactor shutdown, the requirements for RHR system decay heat removal also decrease. Adequate decay heat removal is provided as long as the reactor has been shutdown for at least 100 hours after entry into MODE 3 and RHR flow is sufficient to maintain the RCS temperature less than or equal to 140°F. The reduced flow rate provides additional margin to vortexing at the RHR pump suction while in Mid Loop Operation. During a reduction in reactor coolant system boron concentration the Specification 3.1.1.3.1 requirement to maintain a 3000 gpm flow rate provides sufficient coolant circulation to minimize the effect of a boron dilution incident and to prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND SPENT FUEL PIT

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

The minimum water level for movement of fuel assemblies (23 feet above the vessel flange) assures that sufficient water depth is maintained above fuel elements being moved to or from the vessel. With the upper internals in place, fuel assemblies and control rods cannot be removed from the vessel. Operations involving the lifting of control rods with the vessel upper internals in place may proceed with less than 23 feet of water above the vessel flange provided that 23 feet of water is maintained above all irradiated fuel assemblies within the reactor vessel.

3/4.9.12 FUEL BUILDING VENTILATION SYSTEM

The limitations on the fuel building ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the auxiliary building HEPA and charcoal filter assemblies prior to discharge to the atmosphere. The Fuel Handling Accident analysis does not require filtration of the fuel building exhaust in order to meet the analysis criteria. However, the OPERABILITY of this system and the resulting iodine removal capacity provide additional conservatism compared with the assumptions of the accident analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 179
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 October 17, 1995, as supplemented February 26, 1996, 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 the following paragraphs under Section 2 of Facility Operating License No. NPF-7 are hereby amended to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 179 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.I Deleted

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

EV Imbro

Eugene V. Imbro, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments: (1) Page 13 of License No. NPF-7*
(2) Changes to the Technical Specifications

Date of Issuance: **February 27, 1996**

* Page 13 is attached, for convenience, for the composite license to reflect these changes.

ATTACHMENT TO LICENSE AMENDMENT NO. 179

FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

LICENSE

Remove Page	Insert Page
13	13

TECHNICAL SPECIFICATIONS

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

Remove Pages	Insert Pages
3/4 9-4	3/4 9-4
B 3/4 8-1	B 3/4 8-1
B 3/4 9-1	B 3/4 9-1
B 3/4 9-3	B 3/4 9-3

2.E. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "North Anna Power Station Physical Security Plan," with revisions submitted through February 24, 1988; "North Anna Power Station Guard Training and Qualification Plan," with revisions submitted through May 14, 1987; and "North Anna Power Station Safeguards Contingency Plan," with revisions submitted through January 9, 1987. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

2.F. The design of the reactor coolant pump and steam generator supports may be revised in accordance with the licensee's submittal dated November 6, 1986 (Serial No. 86-477A).

G. If VEPCO plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Power Station, the NRC shall be notified in writing regardless of whether the change affects the amount of radioactivity in the effluents.

H. VEPCO shall report any violations of the requirements contained in Section 2, Items C.(3) through C.(21), E, F and G of this license within 24 hours by telephone and confirmed by telegram, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate, no later than the first working day following the violation, with a written followup report within 14 days.

2.I. Deleted.

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 150 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 150 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 150 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, * and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange. or manual valve, or
2. Be capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Containment Purge and Exhaust isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, ** or

---

* Both doors of the containment personnel airlock may be open provided:

a. One personnel airlock door is OPERABLE (i.e., the door is capable of being closed and that an individual is designated to close the door), and

b1 There is at least 23 feet of water above the top of the reactor pressure vessel flange during movement of fuel assemblies within the containment, or

b2. There is at least 23 feet of water above the top of irradiated fuel assemblies within the reactor pressure vessel during CORE ALTERATIONS excluding movement of fuel assemblies.

** If both doors of the containment personnel airlock are open pursuant to Specification 3.9.4.b above, one door shall be verified to be capable of being closed at the above surveillance frequency.

3/4.8.1 and 3/4.8.2 A.C. and D.C. POWER SOURCES AND DISTRIBUTION

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least one of each of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The ACTION requirements specified in Modes 5 and 6 address the condition where sufficient power is unavailable to recover from postulated events (i.e., fuel handling accident). Implementation of the ACTION requirements shall not preclude completion of actions to establish a safe conservative plant condition. Completion of the requirements will prevent the occurrence of postulated events for which mitigating actions would be required.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods, 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status, and 3) sufficient power is available for systems necessary to recover from postulated events in these MODES, e.g., the control room emergency ventilation system fans during a fuel handling accident.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guide 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants" Revision 1, August 1977, as modified by Amendment No. 48 issued August 22, 1986.

The Surveillance Requirements for demonstrating the OPERABILITY of the Emergency Diesel Generator batteries and the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance, Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std. 450-1980, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," as modified by Amendment No. 84 issued March 25, 1988.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% $\Delta k/k$ conservative allowance for uncertainties. Similarly, the boron concentration of 2300 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon a lack of containment pressurization potential while in the REFUELING MODE.

OPERABILITY of the containment airlock door requires that the door is capable of being closed, that the door is unblocked and no cables or hoses are being run through the airlock, and that a designated individual is continuously available to close the airlock door. This individual must be stationed near the airlock.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL PIT

The restriction on movement of loads in excess of the nominal weight that of a fuel and control rod assemblies and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped, 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

After the reactor has shutdown and entered into MODE 3 for at least 100 hours, a minimum RHR system flow rate of 2000 gpm in MODE 6 is permitted, provided there is sufficient decay heat removal to maintain the RCS temperature less than or equal to 140°F. Since the decay heat power production rate decreases with time after reactor shutdown, the requirements for RHR system decay heat removal also decrease. Adequate decay heat removal is provided as long as the reactor has been shutdown for at least 100 hours after entry into MODE 3 and RHR flow is sufficient to maintain the RCS temperature less than or equal to 140°F. The reduced flow rate provides additional margin to vortexing at the RHR pump suction while in Mid Loop Operation. During a reduction in reactor coolant system boron concentration the Specification 3.1.1.3.1 requirement to maintain a 3000 gpm flow rate provides sufficient coolant circulation to minimize the effect of a boron dilution incident and to prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND SPENT FUEL PIT

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

The minimum water level for movement of fuel assemblies (23 feet above the vessel flange) assures that sufficient water depth is maintained above fuel elements being moved to or from the vessel. With the upper internals in place, fuel assemblies and control rods cannot be removed from the vessel. Operations involving the lifting of control rods with the vessel upper internals in place may proceed with less than 23 feet of water above the vessel flange provided that 23 feet of water is maintained above all irradiated fuel assemblies within the reactor vessel.

3/4.9.12 FUEL BUILDING VENTILATION SYSTEM

The limitations on the fuel building ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the auxiliary building HEPA and charcoal filter assemblies prior to discharge to the atmosphere. The Fuel Handling Accident analysis does not require filtration of the fuel building exhaust in order to meet the analysis criteria. However, the OPERABILITY of this system and the resulting iodine removal capacity provide additional conservatism compared with the assumptions of the accident analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 198 AND 179 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 October 17, 1995, as supplemented by facsimile on February 26, 1996, Virginia Electric and Power Company (the licensee) proposed an amendment to the North Anna Units 1 and 2 (NA-1&2) Technical Specifications (TS). Specifically, the proposed amendment requested changes to TS section 3.9.4, Containment Building Penetrations, to allow both containment personnel airlock (PAL) doors to remain open during fuel movements or core alterations provided that one door is operable, the door is unblocked and a designated individual is continuously available to close the airlock door after personnel are evacuated if a Fuel Handling Accident (FHA) should occur. Furthermore, section 3.9.4 would be revised to allow both PAL doors to be open only if: (1) at least 23 feet of water is above the top of the reactor pressure vessel (RPV) flange during movement of fuel assemblies within the containment; and (2) at least 23 feet of water is above the top of irradiated fuel assemblies within the reactor pressure vessel during core alterations excluding the movement of fuel assemblies.

The licensee has also proposed changes to the TS bases to clarify the emergency power system requirements relative to mitigation of the consequences of an FHA. Specifically, TS bases section 3/4.8.1 and 3/4.8.2 are being changed to clarify that one train of A.C. and D.C. busses must be available during fuel movement to ensure that the Control Room emergency ventilation system is operable in the event of an FHA. In addition, proposed changes to the facility operating licenses are requested to delete License Condition 2.G for Unit 1 and License Condition 2.I for Unit 2. These license conditions reference the analyses for limiting doses to the control room operators. The February 26, 1996 facsimile was within the scope of the original no significant hazards consideration determination.

2.0 EVALUATION

2.1 Deletion of License Conditions 2.G for Unit 1 and 2.I for Unit 2

On February 28, 1990, the NRC issued Amendment Nos. 126 and 110 to Facility

Operating License Nos. NPF-4 and NPF-7 for North Anna Units 1 and 2, respectively, in response to the licensee's application of March 1, 1989 and December 22, 1989. The amendments added the following sentence as License Condition 2.G for Unit 1 and 2.I for Unit 2:

> The limiting dose to the control room operators shall be revised in accordance with the licensee's submittals dated March 1, 1989 (Serial No. 89-022) and December 22, 1989 (Serial No. 89-022A).

The licensee subsequently incorporated the revised control room dose analyses in revision 23 (October 1993) of the Updated Final Safety Analysis Report (UFSAR), revising the sections on Habitability Systems Functional Design (6.4.1), Control Room Availability (7.7.1.13), the design of the Main Control Room air-conditioning, heating, cooling and ventilation systems (9.4.1) and the Main Control Room Shielding (12.1.2.10).

The license amendment was proposed to resolve an unreviewed safety question associated with control room ingress and egress after an accident. In this case, the original control room dose analysis had not accounted for the impact on control room doses of air infiltration resulting from control room personnel ingress and egress after an accident. The licensee subsequently reevaluated control room doses for certain accidents including the impact of air infiltration to the control room from multiple ingress and egress. The unreviewed safety question associated with this analysis was resolved by amending the Facility Operating Licenses to reference the revised control room dose analysis submittal since the analysis is not an item covered in the TS.

The proposed changes to the TS to allow both PAL doors to remain open during refueling operations is supported by a new FHA analysis including a revised assessment of control room doses. Based on the results of this new FHA analysis, the limiting doses to control room operators still result from the main steam line break and steam generator tube rupture as described in the facility operating licenses. However, because these limiting doses are now described in the UFSAR and the NRC Safety Evaluation Report supporting Amendment Nos. 126 and 110, facility operating license conditions 2.G for Unit 1 and 2.I for Unit 2 are no longer necessary. Therefore, deletion of License Condition 2.G for North Anna Unit 1 and 2.I for North Anna Unit 2 is acceptable.

2.2 Revision to TS 3.9.4

Airlocks

TS 3.9.4 currently requires that one of the containment PAL doors be closed during core alterations or movement of irradiated fuel in containment which results in cycling of the personnel airlock doors for each containment entry. Frequent containment entries are required while core alterations or fuel movement is in progress and the resulting heavy use of the personnel airlock produces wear and high maintenance requirements.

Therefore, the licensee proposes to revise TS 3.9.4 to allow the opening of both PAL doors at the same time during fuel movement or core alterations as long as certain conditions are met. This would reduce airlock door wear and would facilitate personnel access. When both airlock doors are open, a footnote to TS 3.9.4 will ensure that there is 23 feet of water above the fuel assemblies to provide iodine scrubbing in the event of an accident. TS 3.9.3 will ensure a minimum 150-hour period between unit shutdown and initiation of fuel movement; the dose analyses by the licensee and staff conservatively include only 100 hours of radioactive decay.

The footnote to 3.9.4b proposed by the licensee stated that both doors of the containment personnel airlock may be open provided that one personnel airlock door is OPERABLE. However, OPERABLE was not defined in the footnote. Instead, the licensee defined OPERABILITY of the containment airlock door in the BASES for 3/4.9.4. Of the conditions proposed by the licensee in the BASES, only two were relied on in the staff's evaluation of this license amendment; namely, that the door would be capable of being closed and that the licensee designate some individual to close the airlock door in the unlikely event that there was a fuel handling accident resulting in a significant release of radioactivity. The staff's position on "capable of being closed" is exactly what it says. One of the doors has to be able to be closed; it does not have to be sealed or meet the leakage criteria in TS 3.6.1.3/SR 4.6.1.3 that have to be met in Modes 1, 2, 3 and 4. The licensee has proposed that the individual designated to close the airlock door must be stationed near the airlock. This is acceptable but beyond what the staff has approved for other licensees. The staff's position is that there should be an individual, who, in addition to his normal duties, also is responsible for making sure one of the personnel airlock doors is closed when the last person is out of containment. The individual should not be outside the protective area but neither does the person have to remain near the airlock. In the BASES for 3/4 9.4, the licensee also proposed a condition that the airlock door is unblocked and no cables or hoses are being run through the airlock. This is acceptable but also beyond what the staff has approved in other licensee amendments. Licensees normally place guards over the door seals to protect the seals from being damaged by persons bringing tools and equipment through the airlock. This is not considered as blocking the door as long as they are removable in a reasonable time.

As noted above, two of the definitions of OPERABILITY for the airlock doors were relevant to the staff finding the proposed amendment acceptable. It was the staff's position that the term "OPERABLE" should be defined to reflect these two conditions, even though they would be in the BASES. By facsimile of February 26, 1996, the licensee amended the application to amplify footnote "a" to define OPERABLE as meaning the door is capable of being closed and that an individual is designated to close the door. With this clarification, the proposed TS changes are acceptable.

2.3 Fuel Handling Accident Evaluation

During core alterations, the most severe radiological consequences result from an FHA. The FHA is a postulated event that involves damage to irradiated fuel. FHAs include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The TS requirements associated with refueling are intended to ensure that the release of fission product radioactivity, subsequent to an FHA, results in doses that are "well within" the guideline values specified in 10 CFR Part 100. Standard Review Plan (SRP) Section 15.7.4, Rev. 1, defines "well within" 10 CFR Part 100 to be 25% or less of the 10 CFR Part 100 values, i.e., 6 rem to the whole body and 75 rem to the thyroid.

Regulatory Guide (RG) 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," provides acceptable assumptions that may be used in evaluating the radiological consequences of an FHA. The licensee's FHA analysis is consistent with the assumptions of RG 1.25.

Control room habitability following an FHA must also be considered using the dose criteria in 10 CFR Part 50, Appendix A, General Design Criteria 19 (GDC 19). During core alterations or fuel movements at North Anna, direct communication is established between fuel handling personnel in containment and the control room. Upon verbal notification of an FHA with the potential for radionuclide release or upon receipt of a high radiation signal from the containment radiation monitors, the control room will be manually isolated, and the bottled air supply initiated. The licensee has stated that it is estimated that up to a 2-minute delay can occur between detection of a high radiation level and isolation of the control room. However, the transit time for any released activity from the radiation detection point is expected to exceed 2 minutes. Therefore, control room isolation is modeled as occurring at the start of the accident. The licensee's analysis assumes the control room is supplied with bottled air for 1 hour after the start of an FHA and then with filtered air at a flow rate of 1000 cubic feet per minute (cfm) with an iodine filtration efficiency of 95% for organic and inorganic iodine through the remainder of the 30-day dose calculation period. No credit is taken for operation of fan/filter units to provide recirculation of the control room air. The fan/filter unit which supplies the 1000 cfm of filtered intake is supplied by emergency power to ensure that GDC 19 limits are met. The licensee found that the dose consequences calculated from the analyses met the applicable dose acceptance criteria.

The staff's analysis used the accident source term given in RG 1.4, assumptions contained in RG 1.25, and the review procedures specified in SRP Section 15.7.4. The staff assumed an instantaneous puff release of noble gases and radioiodine from the gap of the broken fuel rods as gas bubbles pass up through the 23 feet of water covering the fuel. All airborne radioactivity reaching the containment atmosphere is exhausted within 2 hours into the environment. All radioactive material in the fuel rod gap is assumed to have decayed for a period of 100 hours. The staff assumed an entire fuel assembly of 264 rods is damaged by the FHA.

The staff computed the control room operator doses and the doses for the exclusion area boundary (EAB), using the NRC computer code HABIT, Version 1.1., Table 1, which presents the doses calculated by the staff and the applicable acceptance criteria. Table 2 presents the staff's calculational assumptions. The offsite and control room doses are within the acceptance criteria given in SRP Section 15.7.4 and GDC 19.

The staff's dose calculation was based on the assumption that all of the radioactive material released to the containment escapes the containment within 2 hours. However, the staff has historically required plant technical specifications to maintain containment closure during core alterations and fuel handling as a defense-in-depth measure to further limit releases. Recently the staff has allowed changes to plant technical specifications to keep both doors to a containment air lock open during core alterations and fuel handling with provisions in place to close one door quickly, thereby reestablishing containment closure. The provisions described in this safety evaluation provide reasonable assurance that containment closure as a defense-in-depth measure can be reestablished quickly to limit releases to much lower than assumed in the dose calculations.

The staff has reviewed the licensee's analysis and has performed an independent assessment of the radiological consequences resulting from a fuel handling accident during refueling operations with the containment air locks open. The staff concludes that the radiological consequences associated with this accident are within the acceptance criteria set forth in 10 CFR Part 100 and the control room operator dose criteria specified in GDC-19 of Appendix A to 10 CFR Part 50 and are acceptable. Based on these reasons, the staff finds the proposed changes to TS 3.9.4 acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The 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 or change a surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent 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 (61 FR 187). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of 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.

Attachment: Tables 1 and 2

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Date: **February 27, 1996**

TABLE 1

CALCULATED RADIOLOGICAL CONSEQUENCES
(rem)

Exclusion Area Boundary	Dose	SRP 15.7.4 Guidelines
Whole Body	0.01	6
Thyroid	44	75
Control Room Operator	**Dose**	**GDC-19 Guidelines**
Whole Body	< 1	5
Thyroid	21	Equivalent to 5 rem whole body*

* Guideline doses provided in Standard Review Plan Section 6.4 define the dose-equivalent as 30 rem to the thyroid.

TABLE 2

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES

Input Parameters	Quantity
Power Level, Mwt	2900
Number of Fuel Rods Damaged	264
Total Number of Rods	41,448
Shut time, hours	100
Power Peaking Factor	1.65
Fission Product Release Duration	2 hours
Release Fraction	
Iodine	10%
Noble Gas	10%
Krypton Gas	30%
Iodine Form	
Elemental	75%
Organic	25%

Core Fission Product Inventories per TID-14844

Receptor Point Variables

Exclusion Area Boundary**

Atmospheric Relative Concentration, X/Q (sec/m^3) 0-2 hours (NUREG-0053, Supplement No.6)	4.2×10^{-4}

Control Room

Atmospheric Relative Concentration, X/Q (sec/m^3)	6.0×10^{-3}
Control Room Volume, cubic feet	23.0×10^{4}
Maximum Infiltration Rate, ft^3/min	1000
Iodine Protection Factor	30
Geometry Factor	18.1