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 Technical Specifications

   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

EV Imbro

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.

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* 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
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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.

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

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

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* 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.