ATTACHMENT 2

PROPOSED OPERATING LICENSE AND
TECHNICAL SPECIFICATIONS CHANGES

NORTH ANNA UNITS 1 AND 2

VIRGINIA ELECTRIC AND POWER COMPANY

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. Deleted.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Originally 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

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

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, 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. 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.

3/4.9.1 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 materials 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.

3/4.9.10 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.

I. Deleted.

J. This license is effective as of the date of issuance and shall expire at midnight on August 21, 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Originally Signed by
Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendices A & B

Date of Issuance: AUG 21 1980

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, 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.

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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.

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3/4.9.1 BORON CONCENTRATION

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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.

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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.

ATTACHMENT 3

SIGNIFICANT HAZARDS DETERMINATION

VIRGINIA ELECTRIC AND POWER COMPANY

SIGNIFICANT HAZARDS DETERMINATION

North Anna Technical Specifications section 3.9.4 requires that one of the containment personnel airlock doors be closed during core alterations or movement of irradiated fuel in containment. This requires cycling 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. There could be a large number of personnel in containment during refueling operations and it may take several cycles of the airlock to evacuate personnel from containment if a Fuel Handling Accident were to occur. The time required for these cycling operations would increase personnel doses. A change is being proposed to Technical Specifications section 3.9.4 to allow both doors to remain open during fuel movements or core alterations provided that one door is operable and an individual is available to close the airlock door after personnel are evacuated if a Fuel Handling Accident should occur. This would reduce the maintenance requirements for the airlock doors and the dose to personnel in containment in the event of a Fuel Handling Accident.

While reviewing the licensing basis for the Fuel Handling Accident, it was determined that Virginia Power's response the NRC Question 6.72 discussing conformance with the recommendations in Regulatory Guide 1.52, "Design , Testing, And Maintenance Criteria For Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," requires clarification. Specifically, Virginia Power indicated that the North Anna Auxiliary Building Ventilation System complies with position C.2.h of Regulatory Guide 1.52. This position states that the power supplies and electrical distribution systems should be designed in accordance with IEEE-308, "Criteria for Class I E Electrical Systems of Nuclear Power Generation Station." Contrary to this, the fans (HV-F-7A and 7B) in the Fuel Building exhaust system, which are considered by Regulatory Guide 1.52 to be part of the filtration system, are non-safety related and are powered from non-safety related power supplies. In addition, Virginia Power stated that the system complied with Position C.2.c, which states that all components of an engineered-safety-feature system should be designed to Seismic Category I if failure of the component would lead to release of fission products that would result in potential

offsite exposures comparable to 10 CFR 100 limits. While some portions of the Fuel Building Ventilation system required to support filtration of the exhaust are not Seismic Category I, exhaust filtration is not required to ensure that doses are less than 10 CFR 100 limits.

The dose consequences resulting from a Fuel Handling Accident for the North Anna Power Station have been reanalyzed to support the proposed Technical Specifications changes and to help clarify requirements for systems to mitigate a Fuel Handling Accident. The Fuel Handling Accident was evaluated without credit for iodine removal by the charcoal filtration system provided for exhaust gases from the fuel building or credit for isolation of the containment. The calculated dose consequences for the Fuel Handling Accident are within the applicable regulatory limits of GDC 19 and 10 CFR 100 and also meet the SRP 15.7.5 guideline of well within 10 CFR 100 limits (less than 25% of these limits). Although it has been determined that some of the responses to 6.72 must be clarified, the as-built plant configuration is consistent with the safety analysis and the revised Fuel Handling Accident analysis shows that all regulatory limits and NRC Standard Review Plan guidelines are met without credit for filtration or radionuclide retention in the fuel building or containment.

It has been determined that the proposed changes to the Technical Specifications, revision of the Fuel Handling Accident analysis and clarification of the Virginia Power responses to NRC Question 6.72 do not involve a significant hazards consideration as described in 10 CFR 50.92. The results of this determination can be stated as follows:

> There is no significant change in the probability or consequences of an accident previously evaluated. There are no system changes which would increase the probability of an accident occurring. Allowing both personnel airlock doors to remain open during core alterations or fuel movement inside containment will not have any impact on the probability of a Fuel Handling Accident either in containment or in the fuel building. The consequences of a Fuel Handling Accident have been investigated by performing a reanalysis with no credit for isolation or filtration by the Fuel Building or containment ventilation systems. The Exclusion

Area Boundary and Low Population Zone doses for a Fuel Handling Accident without credit for iodine filtration remain well within (<25%) of the NRC regulatory limits of 10 CFR 100. The predicted control room operator doses remain bounded by the limiting case for control room doses and within the regulatory limits of General Design Criterion 19. In addition, the action to clarify the responses to NRC question 6.72 will not increase the probability or consequences of the Fuel Handling Accident.

No new accident types or equipment malfunction scenarios are introduced as a result of the clarification to the Virginia Power response to 6.72 or as a result of these changes in analysis methods or the proposed Technical Specifications changes to allow both personnel airlock doors to remain open during core alterations or fuel movement inside containment. Therefore, there is no possibility of an accident of a different type than any previously evaluated in the North Anna UFSAR.

There is no significant reduction in the margin of safety. An evaluation of the Fuel Handling Accident doses at the EAB, the LPZ and to control room operators has been performed and it has been concluded that the acceptance criteria defined by GDC-19, 10 CFR 100, and the NRC Standard Review Plan will continue to be met.