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

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APR 8 1980

MEMORANDUM FOR: Chairman Ahearne
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Hendrie
Commissioner Bradford

FROM: Harold R. Denton, Director, Office of Nuclear Reactor Regulation

THRU: William J. Dircks, Acting Executive Director for Operations (Signed) T. A. Rehm

SUBJECT: CONSIDERATION OF OL FOR NORTH ANNA UNIT NO. 2

In the Commission meeting of April 1, 1980 related to the issuance of a low power license for North Anna Unit 2, we advised the Commission that we had just received the applicant's evaluation of the inspection conducted on the six large diameter reactor vessel nozzles and our review of this evaluation was not completed. The purpose of these examinations was to determine if cracking existed under the stainless steel cladding in the nozzle base metal, and to characterize and evaluate any cracking found. The examination results of the North Anna Unit 2 reactor vessel nozzles show that there are a very large number of cracks in each of the six reactor vessel nozzles.

We have reviewed the information presented by the applicant and based on our review of this information, we have concluded that the flaws found in the North Anna Unit 2 reactor vessel nozzles are reheat cracks. Based on our earlier review of the reheat cracking phenomenon and the specific information provided for North Anna Unit 2, we conclude that the cracks found in the Unit 2 nozzles are within the acceptance standards established by Section XI of American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Compliance with Section XI of the Code provides adequate assurance that the reactor vessel has sufficient margin against flaw induced fracture. To provide added assurance that adequate margins continue to be maintained during service, we will require that the Unit 2 reactor vessel nozzles be inspected periodically during service and that results of the examinations be reported to the Commission. Our evaluation of this matter is presented in Enclosure 1.

The proposed April 2, 1980 operating license which was transmitted to the Commission for review was modified as discussed with Commissioner Bradford's staff. These changes are noted by a marginal double bar. The revised proposed operating license is presented in Enclosure 2.

Harold Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
As Stated

cc: SECY
OPE
OGC

ENCLOSURE 1

Reactor Vessel Nozzle Inspection

Ultrasonic examinations of the six reactor vessel nozzles have been completed at the North Anna Unit No. 2 clear plant. These examinations supplemented the preservice volumetric inspections required by Section XI of the ASME Code. The purpose of the supplemental examinations was to determine if cracking existed under the stainless steel cladding in the nozzle base metal, and to characterize and evaluate any cracking found.

There are two ways that underclad cracking can be produced. One is referred to as "reheat" underclad cracking, and is caused by high heat input during clad deposition. Variations in welding procedures, involving lower pre and post weld temperature control were developed partly to prevent underclad "reheat" cracking. It was recently discovered that this new procedure could cause "cold" cracking under the cladding, and that the depth of cracks produced by this second mechanism could be two to three times as deep as those produced by the "reheat" mechanism.

The presence of underclad reheat cracks was recognized several years ago, was previously evaluated by the staff, and was found acceptable based on extensive metalurgical and fracture mechanics analyses. The nature of the reheat cracks is described in detail in two Westinghouse reports (References 1 and 2). The staff safety evaluation for the reheat cracks was completed in April 1972 (Reference 3). We have reviewed the earlier Westinghouse report and our previous evaluation and find no new information to indicate that our earlier conclusions concerning the acceptability of reheat cracks should be modified.

Cold cracks can be produced in the base material immediately under the cladding at low temperatures (below 350°F) during cooling subsequent to welding and is associated with three factors: (1) the presence of hydrogen in the heat affected zone of the weldment, (2) a susceptible metallurgical structure, and (3) the presence of residual stresses. Cold cracking can be avoided by pre-heating and post-weld heating the component to permit any hydrogen generated in the welding process to diffuse out of the susceptible area while the material is above the temperature range where cold cracking will occur.

The search for underclad cold cracking was initiated when information received from Westinghouse disclosed potential "cold" cracking under the cladding in vessel nozzles that had been fabricated in Europe using the low temperature cladding process. This low temperature procedure is not commonly used in the United States. The processes that are of concern are characterized by a lack of sufficient heat input to the nozzle before and after the clad layers are deposited. As this safety evaluation later discusses, the cladding procedure used for the North Anna Unit No. 2 reactor vessel nozzles is not expected to produce cold cracking. Nevertheless, because there was limited experience with the cold cracking phenomenon in the United States, and some uncertainty as to the cladding process used, the ultrasonic examinations were performed to provide additional information regarding the condition of the North Anna Unit No. 2 nozzles.

The examination results show that there are a very large number of cracks in each of the six reactor vessel nozzles. Because there are a large number of cracks, each crack could not be evaluated during the examination. Consequently, only a sample of the cracks were examined in detail and only a qualitative description of the nature of the cracking is available. The cracks are reported to be aligned in rows that form circumferential bands around the nozzle circumference. Within the circumferential bands the cracks are tightly spaced and have their lengths oriented parallel to the axial length of the nozzle. These circumferential bands of axially oriented cracks are spaced about 1-1/2 to 2 inches apart and extend along the entire length of the nozzles. Measurements by ultrasonic methods indicate the cracks are typically 3/8 inches long with the maximum reported length being 1/2 inch. Limitations on the examination methods preclude an acceptable measurement of crack depth into the base metal of the nozzle.

We have reviewed the ultrasonic examination results obtained for the North Anna Unit No. 2 reactor vessel nozzles and have performed an independent evaluation to determine if the cracks are acceptable for service. Our evaluation includes an assessment of the examination results, a determination of the likely cause of the cracking, an estimate of the depth of the cracks, and an assessment of the safety significance of the cracks.

The cracking pattern reported to exist in the North Anna Unit No. 2 reactor vessel nozzles is not typical of the underclad cold cracking associated with inadequate pre and post clad temperature. Instead, the reported cracking pattern is characteristic of underclad reheat cracking that results in local material degradation due to excessive heat input to the nozzle base metal during the deposition of the cladding. However, the large number of reheat underclad cracks would likely obscure small cracks that may have been produced by other cracking mechanisms.

Although the extensive reheat underclad cracking found in the North Anna nozzles could be expected to mask detection of cold underclad cracking by the ultrasonic testing method we believe that the welding procedures and temperature controls specified would have precluded the formation of any significant degree of cold cracking. The process used to clad the North Anna nozzles utilized the automatic gas metal arc process. A relatively high heat input was used during the application of the cladding and is associated with the observed reheat cracks. Prior to welding the nozzle was preheated to 250°F and then given a post weld soak at 400°F for two hours prior to any cool down to ambient temperatures. A post weld heat treatment at 1150°F was performed as a final treatment. Not only does the use of the automatic gas metal welding process minimize the generation of hydrogen in the material, but the preheat and post-weld heat treatments specified would be expected to dissipate any hydrogen that was formed; thus eliminating the major source of cold cracking.

Based on our evaluation of the inspection results and the pre and post clad thermal treatments, we have concluded that the flaws in the North Anna Unit No. 2 reactor vessel nozzles are reheat cracks. Based on our earlier review (Reference 3) of the reheat cracking phenomenon and the specific information provided for North Anna Unit No. 2 we conclude that the cracks in the North

Anna Unit No. 2 nozzles likely are not greater than 1/8 inch deep and 1/2 inch long and are within the acceptance standards established by Section XI of the ASME Code. Compliance with Section XI of the ASME Code provides adequate assurance that the reactor vessel has sufficient margin against flaw induced fracture. To provide added assurance that adequate margins continue to be maintained during service we will require that the North Anna Unit No. 2 nozzles be inspected periodically during service and that the results of the examinations be reported to the Commission. Prior to conducting the inservice examinations, we will require VEPCO to demonstrate to the staff that the examination techniques will allow reliable detection and evaluation of individual cracks, should they grow larger than the acceptance standards contained in Section XI of the ASME Code. We require that this information be supplied to us within five years.

References

1. Westinghouse Nuclear Energy Systems Report WCAP-7673-L, "Reactor Vessels Weld Cladding-Base Metal Interaction," by T.R. Mager, E. Landerman, and C.J. Kubit, April, 1971 (NES Proprietary Class 2).

2. Westinghouse Nuclear Energy Systems Report WCAP-7673-L Addendum 1, "Reactor Vessels Weld Cladding-Base Metal Interaction", August, 1971 (NES Proprietary Class 2).

3. U.S. AEC, Tennessee Valley Authority, Sequoyah Nuclear Plant, Unit 1, Docket No. 50-237, Safety Report for Sequoyah Unit One - Cladding Cracks, Memorandum R.R. Maccary to Richard C. DeYoung, April 10, 1972.

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

LICENSE FOR FUEL LOADING AND LOW POWER TESTING

License No. NPF-7

1. The Nuclear Regulatory Commission (the Commission) having found that:

   A. The application for license filed by the Virginia Electric and Power Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the North Anna Power Station, Unit No. 2 (facility) has been substantially completed in conformity with Construction Permit No. CPPR-78 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility requires an exemption from certain requirements of Appendix J to 10 CFR Part 50. This exemption is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 10. This exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. The exemption is, therefore, hereby granted. With the granting of this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

   D. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   E. There is reasonable assurance: (i) that the activities authorized by this license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission;

F. Virginia Electric and Power Company is technically and financially qualified to engage in the activities authorized by this license in accordance with the regulations of the Commission;

G. The Virginia Electric and Power Company has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements", of the Commission's regulations;

H. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

I. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of License No. NPF-7 subject to a condition for protection of the environment set forth herein is in accordance with 10 CFR Part 51, of the Commission's regulations and all applicable requirements have been satisfied; and

J. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30, 40, and 70, including 10 CFR Sections 30.33, 40.32, 70.23 and 70.31.

2. Pursuant to approval by the Nuclear Regulatory Commission at a meeting on April , 1980 License No. NPF-7 is hereby issued to the Virginia Electric and Power Company to read as follows:

A. This license applies to the North Anna Power Station, Unit No. 2, a pressurized water nuclear reactor and associated equipment (the facility), owned by the Virginia Electric and Power Company (VEPCO). The facility is located near Mineral in Louisa County, Virginia and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 17 through 68) and the Environmental Report as supplemented and amended (Supplements 1 through 4).

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses VEPCO:

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use, and operate the facility at the designated location in Louisa County, Virginia, in accordance with the limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. VEPCO is authorized to perform steam generator moisture carryover studies at the North Anna Power Station. These studies involve the use of an aqueous tracer solution of two (2) curies of sodium-24. VEPCO's personnel will be in charge of conducting these studies and be knowledgeable in the procedures. VEPCO will impose personnel exposure limits, posting, and survey requirements in conformance with those in 10 CFR Part 20 to minimize personnel exposure and contamination during the studies. Radiological controls will be established in the areas of the chemical feed, feedwater, steam, condensate and sampling systems where the presence of the radioactive tracer is expected to warrant such controls. VEPCO will take special precautions to minimize radiation exposure and contamination during both the handling of the radioactive tracer prior to injection and the taking of system samples following injection of the tracer. VEPCO will insure that all regulatory requirements for liquid discharge are met during disposal of all sampling effluents and when reestablishing continuous blowdown from the steam generators after completion of the studies.

D. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

VEPCO is authorized to (a) load fuel, (b) proceed to initial criticality, (c) perform startup testing at zero power in Operational Mode 2, and (d) after prior written approval by the Director of Nuclear Reactor Regulation, operate the facility for testing at reactor core power levels not in excess of 139 megawatts thermal (five percent of rated power).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B attached hereto are hereby incorporated into this license. VEPCO shall operate the facility in accordance with the Technical Specifications.

(3) Secondary Water Chemistry

VEPCO shall perform a secondary water chemistry program to inhibit steam generator tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;

b. Identification of the procedures used to quantify parameters that are critical to control points;

c. Identification of process sampling points;

d. Procedure for the recording and management of data;

e. Procedures defining corrective actions for off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the data and the sequence and timing of administrative events required to initiate corrective action.

g. Monitor the condensate at the effluent on the condensate pump. When condenser leakage is confirmed repair or plug the leak in accordance with Branch Technical Position MTEB 5-3.

(4) Initial Test Program

VEPCO shall conduct the post-fuel-loading initial test program that has been reviewed and approved by the Commission at the time of issuance of this license without making any major modification of this program. Major modifications are deemed to involve unreviewed safety questions under 10 CFR Section 50.59 and are defined as:

a. Elimination of any test identified in Section 14 of the Final Safety Analysis Report as essential.

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of the Final Safety Analysis Report as essential.

c. Performance of any test at a power level different from there described.

d. Failure to complete any tests included in the described program (planned or scheduled for power levels up to the authorized power level).

(5) Fuel Load and Zero Power (i.e., power required to perform physics testing) Testing Conditions

The following conditions shall be completed to the satisfaction of the Commission. Unless otherwise noted, these conditions shall be completed prior to fuel loading. The following conditions are related to matters specified in the TMI Action Plan, Near Term Operating License (NTOL) Requirements, dated February 6, 1980, and applicable to fuel load and zero power testing. Each of the following conditions references the appropriate section of Part II of Supplement No. 10 to the Safety Evaluation Report (NUREG-0053 Suppl. No. 10) for the North Anna Power Station, Unit 2 and follows the numbering sequence utilized in the February 6, 1980 NTOL Requirements list.

a. Shift Technical Advisor (I.A.1.1)

VEPCO shall provide on each shift a Shift Technical Advisor (STA) whose principal duty shall be to act as an advisor to the shift supervisor primarily in assessing accident/transient occurrences.

During 1980, at least one Senior Reactor Operator (SRO) or an experienced degreed engineer who is a member of the site Safety Engineering Staff shall be designated as the Shift Technical Advisor.

The STA shall interface with the Safety Engineering Staff whose function is to perform operating experience assessments, for the purpose of disseminating operating experience to the other members of the plant staff.

At no time shall the SRO/STA, who is under the functional supervision of the Shift Supervisor, assume command or control function nor function as the Shift Supervisor or Assistant Supervisor.

b. Shift Supervisor Duties (I.A.1.2) and Shift Personnel Responsibilities (I.C.3)

VEPCO shall issue a management directive, signed by the highest level of corporate management, which emphasizes the assignment of primary management responsibility to the Shift Supervisor. This directive shall be reissued on an annual basis.

The responsibility of the Shift Supervisor shall be that defined in VEPCO Administrative Procedure ADM-1.0, "Station Organization and Responsibility."

The shift supervisor shall remain in the control room at all times when either unit is operating in Modes 1, 2, 3 or 4, except that he may be allowed to be absent provided an individual, other than the shift technical advisor, who possesses a valid SRO license assumes the control room command function during his absence. The individual who assumes the control room command function must remain in the control room until the shift supervisor returns and reassumes the command function.

The shift supervisor shall only be relieved by an individual who possesses a valid SRO license. Individuals who do not possess a valid SRO license, including members of station management, shall not relieve the shift supervisor, nor shall they direct the licensed activities of licensed operators.

c. Shift Manning (I.A.1.3)

The shift crew composition for the operation of North Anna, Unit 2, in conjunction with Unit 1 shall be in accordance with Table 6.2.1 of the Technical Specifications. An additional requirement is that one senior licensed operator, licensed on North Anna, Unit 2 shall be stationed in the Unit 2 control room at all times during Unit 2 operation in Modes 1 through 4.

VEPCO shall have administrative procedures to assure that qualified individuals to man the operational shifts are readily available in the event of an abnormal or emergency situation. These administrative procedures shall include provisions which limit the amount of overtime worked by licensed operators.

The need for a licensed operator to exceed the limits on overtime shall be infrequent. The limits on overtime work hours are:

(1) An individual should not be permitted to work more than 12 hours straight (not including shift turnover time).

(2) There should be at least a 12-hour break between all work periods (shift turnover time is included in this 12-hour break).

(3) An individual should not work more than 72 hours in any 7-day period.

(4) An individual should not work more than 14 consecutive days without having 2 consecutive days off.

However, for those circumstances which arise requiring deviation from the above, such deviation may be authorized by the plant manager or high levels of the VEPCO management in accordance with established procedures and with appropriate documentation of the cause.

d. <u>Revised Scope and Criteria for Licensing Examinations (I.A.3.1)</u>

VEPCO shall train all shift personnel (licensed and non-licensed) at North Anna Power Station, Units 1 and 2 in all the TMI related design changes which have been incorporated in the station.

e. <u>Organization and Management Criteria (I.B.1.1)</u>

With respect to offsite technical support to the plant staff in the event of an emergency, VEPCO shall develop procedures delineating the responsibility and authority of corporate office personnel in providing technical support.

The shift supervisor for Unit 2 shall report to an individual holding a senior operator license on Unit 2 (as required by Technical Specification 6.2) at all times during Unit 2 operation in Modes 1 through 6.

f. Safety Engineering Group and Onsite Evaluation Capability, and Dissemination of Operating Experience I.B.1.2 (I.B.1.4 and I.C.5)

VEPCO shall establish an on-site Safety Engineering Staff which shall be technically responsible to off-site management. The plant Safety Engineering Staff shall combine the review functions of engineering assessment, evaluation and dissemination of plant operating experience, and functions of the Shift Technical Advisor. The group shall consist of at least seven engineers, four of whom shall be designated Shift Technical Advisor. The group shall functionally report to the station Superintendent of Technical Services and technically offsite to the Director, Safety Evaluation and Control.

The plant Safety Engineering Staff is required to function only on the day shift, Monday through Friday. The technical disciplines represented in the group shall be electrical, mechanical, radiation protection and nuclear. The duties and responsibilities of this group shall be as follows:

1. Be cognizant of the scope and intent of the test program

2. Be cognizant of the tests being conducted

3. Be familiar with the operation of a pressurized water type reactor

4. Provide daily status reports to the Director, Safety Evaluation and Control at the corporate office.

VEPCO shall establish an onsite capability to evaluate operating history of North Anna Unit 2, in conjunction with that of North Anna Unit 1, and also plants of similar design.

g. Shift Relief and Turnover Procedures (I.C.2)

VEPCO shall implement the shift and relief turnover procedures in accordance with VEPCO Administrative Procedure ADM-29.3--"Conduct of Operations."

VEPCO shall establish a system to evaluate the effectiveness of the turnover procedures.

h. Control Room Access (I.C.4)

VEPCO shall implement control room access procedures in accordance with ADM-6.0, "Control Room Access."

i. Degraded Core - Training (II.B.4)

VEPCO shall initiate preparation of a program to ensure that all operating personnel are trained in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program shall include the following topics:

Incore Instrumentation

1. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.

2. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

Excore Nuclear Instrumentation (NIS)

1. Use of NIS for determination of void formation; void location basis for NIS response as a function of core temperatures and density changes.

Vital Instrumentation

1. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual vs. indicated level).

2. Alternative methods for measuring flows, pressures, levels, and temperatures.

   a. Determination of pressurizer level if all level transmitters fail.

   b. Determination of letdown flow with a clogged filter (low flow).

   c. Determination of other Reactor Coolant System parameters if the primary method of measurement has failed.

Primary Chemistry

1. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak tight systems.

2. Expected isotopic breakdown for core damage; for clad damage.

3. Corrosion effects of extended immersion in primary water; time to failure.

Radiation Monitoring

1. Response of Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overanged detector); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

2. Methods of determining dose rate inside containment from measurements taken outside containment.

Gas Generation

1. Methods of H_2 generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of noncondensibles.

2. H_2 flammability and explosive limit; sources of O_2 in containment or Reactor Coolant System.

j. Relief and Safety Valve Test (II.D.2)

VEPCO shall carry out a testing program to qualify the relief and safety valves under expected operating conditions for design basis transients and accidents in accordance with the requirements and schedule provided in NUREG-0578, Section 2.1.2, as clarified in NRC letter to operating license applicants dated November 9, 1979.

k. Relief and Safety Valve Position (II.D.5)

VEPCO shall provide reactor system relief and safety valves with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe. The indication shall comply with the requirements contained in NUREG-0578, Section 2.1.3.a, as clarified in NRC letter to operating license applicants dated November 9, 1979. Installation and calibration of control room indicators shall be completed prior to operation in Mode 2.

l. Auxiliary Feedwater Initiation and Indication (II.E.1.2)

VEPCO shall provide for automatic initiation of the auxiliary feedwater system and shall provide for indication, in the control room, of auxiliary feedwater flow to each steam generator. These requirements shall comply with NUREG-0578, Sections 2.1.7.a and 2.1.7.b, as clarified in NRC letter to operating license applicants dated November 9, 1979.

m. Inadequate Core Cooling - Subcooling Meter (II.F.2)

VEPCO shall provide a subcooling meter to provide on-line indication of coolant saturation condition. The meter shall comply with the requirements of NUREG-0578, Section 2.1.3.b, as clarified in NRC letter to operating license applicants dated November 9, 1979.

n. Inadequate Core Cooling - Additional Instrumentation (II.F.2)

VEPCO shall provide a design of additional instruments to provide an unambiguous indication of inadequate core cooling. This requirement shall comply with NUREG-0578, Section 2.1.3.b, as clarified in NRC letter to operating license applicants dated November 9, 1979.

o. Emergency Power for Pressurizer Equipment (II.G)

VEPCO shall provide emergency power for the power-operated relief valves (PORVs), the PORV block valves and pressurizer level instrument channels. This requirement shall comply with NUREG-0578, Section 2.1.1, sub-section 3.2, as clarified in NRC letter to operating license applicants dated November 9, 1979. Calibration and checkout will be completed prior to operation in Mode 2.

p. IE Bulletins on Measures to Mitigate Small Break LOCAs and Loss of Feedwater Accidents (II.K.1)

VEPCO shall review the operating procedures and operator training related to the measures to mitigate small-break LOCAs and loss-of-feedwater transients. These requirements shall comply with Items 1 through 13 of IE Bulletin 79-06A, as modified by IE Bulletin 79-06C. The requirements of Item 7(b) of IE Bulletin 79-06A have been superseded by the requirements set out in Section 2.D.6.a of this license.

q. Improve Licensee Facilities for Responding to Emergencies - Onsite Technical Support Center (III.A.1.2(a))

VEPCO shall establish a technical support center to meet the January 1, 1980 requirements of NUREG-0578, Section 2.2.2.b, as clarified by NRC letter to operating license applicants dated November 9, 1979.

r. Improve Licensee Facilities for Responding to Emergencies - Onsite Operational Support Center (III.A.1.2(b))

VEPCO shall provide an onsite operational support center to meet the requirements of NUREG-0578, Section 2.2.2.c, as clarified by NRC letter to operating license applicants dated November 9, 1979.

s. Emergency Preparedness of State and Local Governments - Near-Term Actions (III.B and III.B.1)

During the period of this license, VEPCO shall maintain in effect an emergency plan that meets:

(1) Regulatory requirements of 10 CFR Part 50, Appendix E

(2) Regulatory Position Statement in Regulatory Guide 1.101 (March 1977)

(3) The Essential Planning Elements in NUREG-75/111 and Supplement 1 thereto defined by NRR as significant for fuel load and low power testing.

t. In-Plant Radiation Monitoring - Partial (III.D.3.3)

VEPCO shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration throughout the plant under accident conditions.

u. Communications (III.A.3.3)

VEPCO shall install and maintain direct dedicated telephone lines between the North Anna plant control room, Site Technical Support Center and NRC Incident Response Center in Bethesda, Maryland.

v. Control Room Design (IV Item 1)

Prior to operation in Mode 2, VEPCO shall implement the following corrective actions related to the control room:

(1) Add labeling and instrumentation demarcation to correct problems specifically identified by ESSEX (letter dated March 27, 1980). Assess the need for other improvements in the area of labeling and demarcation and modify labeling and demarcate controls and displays as appropriate

(2) Color code annunicator windows as an aid in identifying high priority alarms

(3) Review, and improve where necessary, administrative controls used to ensure proper safety injection status

(4) Correct problems associated with general maintenance and implement measures to prevent their recurrence

(5) Institute controls to limit control room access

(6) Test the adequacy of emergency lighting.

(7) Establish procedures to limit use and sound intensity of the public address system to plant related matters.

(6) Conditions on Operations Beyond Zero Power Testing

The following conditions shall be completed to the satisfaction of the Commission prior to proceeding beyond zero power testing. The following conditions are related to matters specified in the TMI Action Plan, Near Term Operating License (NTOL) Requirements dated February 6, 1980, and applicable to operation beyond zero power testing. Each of the following conditions references the appropriate section of Part II of Supplement No. 10 to the Safety Evaluation Report (NUREG-0053 Suppl. No. 10) for the North Anna Power Station, Units 1 and 2 and follows the numbering sequence utilized in the February 6, 1980 NTOL Requirements list.

a. Short-Term Effort-Analysis and Procedure Modifications (I.C.1)

VEPCO shall revise its emergency operating instructions for dealing with small break LOCAs and inadequate core cooling based on its analysis of these events and the vendor guidelines derived from these analyses. These requirements supersede the requirements specified in Item 7(b) of IE Bulletin 79-06A.

b. NSSS Vendor Review of Low Power Test Procedures (I.C.7)

VEPCO's low power test procedures shall be reviewed by the nuclear steam supply system vendor, Westinghouse, and documentation of the review submitted to NRC.

c. Low Power Test Program (I.G.1)

As set forth in Section I.G.1, VEPCO shall obtain staff approval of a low power test program.

d Additional Accident Monitoring Instrumentation (II.F.1)

VEPCO shall comply with the requirements contained in NUREG-0578, Section 2.1.8.b as clarified in NRC letter to operating license applicants dated November 9, 1979.

(7) Battery Room Ventilation System

Within 90 days from date of issuance of this license, VEPCO shall provide for Commission review an evaluation and assessment of the industrial and safety hazards associated with the battery room ventilation system which exhausts air into one end of the control room. Proposed corrective actions including design modifications, if any, and an implementation schedule shall be included in this evaluation.

E. VEPCO shall report any violations of the requirements contained in Sections 2.D.(5) and (6) of this license in accordance with the prompt reporting requirements of Section 6.9.1.8 of Appendix A, which is attached to and is a part of this license.

F. VEPCO shall maintain and fully implement the physical security plan entitled, "Security Program, North Anna Power Station, Units 1 and 2," dated May 25, 1977, as amended, November 30, 1977, and revised September 25, 1978, October 25, 1978, January 12, 1979, March 23, 1979, April 30, 1979, and December 1, 1979, and as amended in accordance with the provisions of 10 CFR 50.54(p).

Pursuant to 10 CFR Section 2.790(d), the security plan is being withheld from public disclosure because it is deemed to be commercial or financial information within the meaning of 10 CFR Section 9.5(a)(4) and subject to disclosure only in accordance with 10 CFR Section 9.12.

G. This license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, the licensee will prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated, in the Final Environmental Statement or any addendum thereto, VEPCO shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation.

H. This license is effective as of the date of issuance and shall expire one year after that date.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendices A and B Technical
Specifications

Date of Issuance: