
Safety Evaluation Report

related to the operation of
**North Anna Power Station,
Unit 2**

Docket No. 50-339
Virginia Electric and Power Company
Supplement No. 11

August 1980

**Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission**



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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On June 4, 1976, the Nuclear Regulatory Commission (Commission) issued its Safety Evaluation Report regarding the application by the Virginia Electric and Power Company (VEPCO, licensee) for licenses to operate the North Anna Power Station, Units 1 and 2. The Safety Evaluation Report was supplemented by Supplement Nos. 1 through 10 which documented the resolution of several outstanding issues in further support of the licensing activities.

On November 26, 1977, Facility Operating License NPF-4 was issued for North Anna Power Station, Unit 1. The license permitted fuel to be loaded into Unit 1 and was subsequently amended [Amendment No. 3] on April 1, 1978 to permit Unit 1 to operate at 100 percent power.

On April 10, 1980, the Commission issued Supplement No. 10 to the Safety Evaluation Report related to the issuance of an operating license for North Anna Power Station Unit 2. This action permitted VEPCO to load fuel and to achieve criticality, and to operate Unit 2 at power levels not to exceed five percent of full power; i.e., low power operation in accordance with requirements developed from the lessons learned from the TMI-2 accident. The Advisory Committee on Reactor Safeguards has requested that VEPCO discuss this program and other matters at an informational meeting on August 7, 1980.

Subsequently on July 3, 1980, authorization was granted by the NRC to VEPCO to conduct its special low power test program. The program consisted of conducting seven special tests involving conditions for natural circulation heat removal. Training was provided for the operators during the conduct of the tests. Further details of the program and the results are given in Section 22.2 Item I.G.1 of this report.

The purpose of this supplement is to update our Safety Evaluation Report (and Supplements No. 1 through No. 10) by providing (1) our evaluation of additional information submitted by the licensee since the issuance of Supplement No. 10 to the Safety Evaluation Report, (2) our evaluation and status of the Non-TMI-2 outstanding issues identified in Section 1.10 of Part I of Supplement No. 10, (3) our evaluation of TMI-2 requirements which must be completed prior to the issuance of a full power operating license, (4) our evaluation of dated requirements which the licensee must implement by the dates identified in NUREG-0694, "TMI-Related Requirements for New Operating Licenses" and (5) our evaluation of additional information for those sections of the Safety Evaluation Report where further discussion or changes are in order.

Our review of TMI-2 requirements is based on the Commission policy statement issued on June 16, 1980, regarding the requirements to be met for current operating license applications. The requirements are derived from NRC's Action Plan (NUREG-0660) and are found in NUREG-0694, "TMI-Related Requirements for New Operating Licenses." The North Anna Power Station Unit 2 was measured against the NRC regulations as augmented by these requirements.

Each of the following sections of this supplement is numbered the same as the corresponding sections of the Safety Evaluation Report and its supplements except that Section 22.0 will address TMI-2 requirements and Section 23.0 will present our conclusions.

Each section is supplementary to and not in lieu of the discussion in the Safety Evaluation Report and the supplements thereto, except where specifically noted. Appendix A is a continuation of the chronology of any principal actions related to the processing of the application. Appendix B is an evaluation related to emergency preparedness and Appendix C is an errata to Supplement No. 10 to the Safety Evaluation Report.

On the basis of staff review, we conclude that the North Anna Unit 2 facility may be operated safely at full power in accordance with the facility Technical Specifications without undue risk to the health and safety of the general public. Licensing action would be subject to successful completion of the Emergency Planning drill as discussed in Section 22.2 (Item III.A.1.1) in mid August, 1980.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.7 Seismic Design

In a letter dated June 18, 1980, VEPCO proposed a change in the Technical Specifications whereby fifteen additional snubbers will be added to the piping of one of the safety injection systems. In a similar system in the North Anna Unit 1 plant, VEPCO has reinforced a number of rigid restraints and has proposed the removal of one snubber. These two systems, although designed identically, will differ significantly in the as-built conditions as a result of these changes.

The safety of the piping systems in North Anna Unit 2 has heretofore been based on the design of the same systems in North Anna Unit 1. The proposed changes may introduce significant differences in the two systems so that conclusions resulting from the evaluations in Unit 1 may not necessarily apply to Unit 2. For this reason, the proposal by VEPCO in their letter of June 6, 1980 that Unit 1 results be extended without further justification for all Unit 2 systems was not accepted by the staff. We therefore requested additional information in our letter of June 26, 1980 regarding the safety significance of the proposed changes.

In response to this request VEPCO has submitted the following information for staff evaluation:

1. A list of North Anna Unit 2 safety-related piping systems for which the model, analysis and results for the same system in Unit 1 are not directly applicable, together with a commitment to resolve any current seismic analysis issues for these systems.
2. Detailed design and construction information for the safety injection system where the fifteen snubbers will be added, including a seismic reanalysis using envelope response spectra. This information will be used by us for an independent safety evaluation of this system.
3. A commitment to resolve any current seismic analysis issues applicable to the Unit 2 systems which are the same as installed in Unit 1.

Based on our review of the results of the safety injection system analysis and preliminary results of Unit 1 system and Unit 2 systems analyzed by VEPCO, we have concluded that the existing system margin is sufficient to permit operation at full power during the completion of the analysis. The license will be conditioned to require VEPCO to provide the staff with its completed seismic analysis regarding this matter within six months of the issuance of the Unit 2 full power operating license. If upon completion of our review we determine that design modifications are necessary to meet Criterion 2 of the General Design Criteria, we will require that they be implemented at the North Anna Power Station, Unit 2.

3.10 Seismic and Environmental Qualification of Seismic Category I Instrumentation and Electrical Equipment

3.10.3 Environmental Qualification of Westinghouse and Balance-of-Plant Seismic Category I Instrumentation and Electrical Equipment

In December 1979 the staff issued guidance for the environmental qualification of safety related electrical equipment (NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"). By letter dated February 19, 1980 the staff requested that VEPCO review the environmental qualification documentation for each item of safety-related electrical equipment which could be exposed to a harsh environment so as to identify the degree to which the associated environmental qualification program complies with the staff's position as described in this NUREG. Further, where there are deviations, we requested the applicant to provide the basis for concluding that the associated environmental qualification program demonstrates that each item in question is environmentally qualified for its service conditions. In response to this request, VEPCO provided an environmental qualification submittal on June 20, 1980 which provides the results of their review. The results of this review essentially confirms our previous conclusion as stated in the safety evaluation report, in that, the associated electrical equipment is adequately qualified for its expected service environments with the exception of 57 apparent deficiencies.

For 48 of the items which were identified as deficient, VEPCO has received from the manufacturers, certificates of conformance, telex or other letters stating that the equipment is qualified to the levels stated. However, sufficient test data from the manufacturer to verify their statements have not been provided. VEPCO has stated that these test data which are expected to be confirmatory will be provided from all manufacturers by November 1, 1980. Any deficiencies will be corrected promptly in accord with the Commission May 23, 1980 Order.

For the remaining nine items identified, either the applicant provided justification for increased power level operation or the item in question was previously addressed by the staff in Supplement Number 10 to the Safety Evaluation Report. We find these actions, that is, VEPCO's review, additional documentation providing the stated justifications, and previous evaluations provided by the staff to be an adequate bases for the operation of this unit at full power pending completion of ongoing actions discussed below.

The Commissioner's Memorandum and Order dated May 23, 1980 directs the staff to complete its review of environmental qualification including the publication of the Safety Evaluation Reports by February 1, 1981 for all operating reactors. Also, this order directs that by no later than June 30, 1982, all electrical equipment in operating reactors subject to this review be in compliance with NUREG-0588 or Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors. Accordingly, the staff intends to complete the environmental qualification review in accordance with these stated dates.

By letter dated July 28, 1980, VEPCO has stated that interim plant operation at full power is acceptable pending completion of the ongoing environmental qualification program. The basis for their conclusion includes such factors as the short-term operating period until the program is completed, some environmental qualifications have already been performed, and that actual conditions which may exist for an accident could be much less than those specified for the qualification program. The staff agrees to these aspects and accepts them as a basis for initial plant operation.

4.0 REACTOR

4.2 Fuel Mechanical Design

In Section 4.2 of Supplement No. 10 to the North Anna Power Station Safety Evaluation Report, we stated that for operation at five percent of full power the restriction for PAD-3.3 is not significant and the analysis as presently docketed is acceptable. We further indicated that we will complete our review of the Westinghouse evaluation (and the application of the revised model) prior to authorizing operation at full power.

The new Westinghouse code was approved with four restrictions as described in our safety evaluation of February 9, 1979 (Letter from J. Stolz, NRC to T. Anderson, Westinghouse). Three of those restrictions deal with numerical limits and have been complied with. The fourth restriction relates to use of the PAD-3.3 code for the analysis of fission gas release from uranium dioxide (UO_2) for power increasing conditions during normal operation. This restriction applies to the safety analysis of North Anna Unit 2. However, Westinghouse has stated that this restriction does not adversely affect the results of the safety analyses performed for North Anna Unit 2. Although we believe that this is essentially correct for the planned operation of the plant, Westinghouse has prepared and submitted a detailed evaluation of this restriction in WCAP-8720, Addendum 1.

At this time, we have not completed our review of the Westinghouse evaluation of this restriction. However, our review has progressed to the point where the following conclusions can be made.

1. The Westinghouse evaluation of our restriction on the use of the PAD-3.3 code supports their earlier statement that the restriction does not adversely affect the results of the safety analyses performed for North Anna.
2. We continue to believe that this result is essentially correct and anticipate some additional information from Westinghouse to confirm this conclusion.
3. Because the restriction pertains to the release of fission gases from the fuel, any change in our conclusions would not have significant impact at low burnup, when the fission gas inventory in the fuel is low.

At this time we can therefore state that for first cycle operation at full power, the restriction for PAD-3.3 is not significant and the analysis as presently docketed is acceptable with regard to Criterion 10 of the General Design Criteria of Appendix A to 10 CFR 50. We anticipate a timely completion of our review of the Westinghouse evaluation prior to operation at the extended burnup.

5.0 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.11 Inservice Inspection of Pressure Isolation Valves

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated Reactor Coolant System (RCS pressure). Also included are those systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems causing an inter-system LOCA. Periodic leak testing of pressure isolation valves shall be performed after all disturbances to the valve are complete. The licensee has categorized their pressure isolation valves as Category A or AC. These categorizations meet our requirements and we find them acceptable. Pressure isolation valves are required to be Category A or AC and to meet the appropriate valve leak rate test requirements of IWV-3420 of Section XI of the ASME Code except as discussed below. The allowable leakage rate shall either not exceed 1.0 gallon per minute (GPM) for each valve or at leak rate stated in the technical specifications.

VEPCO has committed to meet the allowable leak rate limit of 1.0 GPM for all pressure isolation valves except for the RHR system. For the RHR isolation valves VEPCO has committed to meet a leak rate of 5.0 GPM at low pressure along with installation of leak test connections which will be provided at their next refueling. The staff finds these commitments acceptable and will condition the license to reflect this requirement. The Limiting Conditions for Operation (LCO) will be added to the Technical Specifications which will require corrective action, i.e., shutdown a system isolation when the above leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, will be provided in the Technical Specifications.

The RHR system isolation valves are included in their testing program and are categorized as Category B. We consider these valves to perform a pressure isolation function and VEPCO has agreed to categorize these valves as Category A and leak rate test them to the above requirements. We will include these valves in Table 3.4-1 of the Technical Specifications.

VEPCO has stated that all containment isolation valves (regardless of their pressure isolation function) would be leak tested in accordance with Appendix J of 10 CFR 50 in lieu of IWV-3420, Section XI. We have concluded that the leak test procedures and requirements of Appendix J are the governing criteria for containment isolation

valves when they perform only a containment isolation function. In those cases where a valve performs both a pressure isolation function and a containment isolation function, both Section XI and Appendix J should be complied with unless justification is given for alternative testing. VEPCO has stated that valves which perform a containment isolation/pressure isolation function will meet the above requirements.

We conclude that VEPCO's commitments to periodic leak testing of pressure isolation valves between the reactor coolant system and low pressure systems will provide reasonable assurance that the design pressure of the low pressure systems will not be exceeded, and thus reduces the probability of an intersystem LOCA from occurring. This matter is directly a consideration for Criterion 55 of the General Design Criteria of Appendix A to 10 CFR 50.

5.4 Component and Subsystem Design

5.4.3 Residual Heat Removal System

In Supplement Number 10 to the Safety Evaluation Report, we stated that further confirmatory documentation was necessary on the capability of the residual heat removal system (RHR) to meet our Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal Systems."

Four processes are involved in taking the plant from hot standby to cold shutdown conditions. These are: (1) removal of residual heat and stored energy; (2) circulation of the reactor coolant; (3) makeup and boration of the reactor coolant to the cold shutdown boron concentration; and (4) depressurization. With loss of offsite power, the reactor coolant pumps, main condenser and the main feedwater pumps are unavailable. Heat removal and coolant circulation under natural circulation conditions is then controlled by use of the steam generator atmospheric dump valves and the auxiliary feedwater system.

The three air-operated atmospheric dump valves at North Anna Unit 2 (one per steam generator) are seismic Category 1. Air is supplied from plant instrument air system and is automatically backed up by the service air system. Electrical power is obtained from separate channels of uninterruptable safety-grade power from independent station batteries. The most limiting single failure would be the loss of one main steam line dump valve. The valves could be operated by manual action (outside of containment) to correct for this single failure. Since this is a control function, VEPCO has performed tests which confirm the feasibility of this type of manual action. Mechanical failure could prevent opening of a single dump valve. Manual action to correct for this failure would involve opening the decay heat release line which effectively bypasses each dump valve. Alternatively, manual action could be taken to close an upstream isolation valve and replace or repair the dump valve.

The water supply to the auxiliary feedwater system is provided initially from the seismic Category 1 condensate storage tank which has a minimum reserve of 110,000 gallons. This supply is backed up by the Seismic Category 1 Service Water System.

The supply is transferred manually to either of these systems via fully qualified admission valves to maintain adequate net positive suction head at the auxiliary feedwater pumps.

During a normal plant cooldown from hot standby conditions, the chemical and volume control system letdown line from the reactor coolant system (RCS) would be used during both the initial boration to the required boron shutdown concentration and while the RCS inventory is controlled during the cooldown. Loss of the nonseismic air supply results in loss of letdown due to air-operated valves failing closed in the letdown line. Under these conditions, boration without letdown could still be accomplished using safety-grade equipment. Borated water (12 weight percent boric acid) could be supplied to the suction of the centrifugal charging pumps from one of the three boric acid tanks using one of the four boric acid transfer pumps. The tanks, pumps, and associated piping are seismic Category 1. The capacity of one

boric acid tank is sufficient to provide boration to the required shutdown concentration. Makeup above that provided by the boric acid tanks is obtained from the refueling water storage tank. Borated water from the centrifugal charging pumps can be supplied to the RCS via the normal charging, and reactor coolant pump seal injection flow paths or via the boron injection tank path. The effect of valve failures due either to loss of air supply or postulated single failure is mitigated either by manual actions to correct the failure or use of an alternate injection path.

Calculations, based on our review of VEPCO's injection of borated water with 12 weight percent of boric acid, indicate that the available volume in the pressurizer steam space is greater than that needed to achieve a cold shutdown boron concentration in the RCS without taking credit for letdown or contraction of the primary coolant in cooldown. In addition, the available volume for borated water injection without letdown which results from the contraction of the primary coolant is much larger than that required to cool and, hence, depressurize the pressurizer to 425 pounds per square inch gauge by injection of borated water through the pressurizer spray. This pressure must be reached to permit shutdown cooling with the RHR system.

Under natural circulation conditions the normal supply for the pressurizer spray from the cold legs of two coolant loops is lost. In this case, the pressurizer spray can be supplied by flow from the centrifugal charging pumps through a line branching off from the charging line of the chemical and volume control system (CVCS). This supply could be lost by a single failure involving either closing of a single valve in the supply line or opening of one of several valves in lines connected to the supply line. If manual actions to correct for such failures were not successful, a backup method of depressurization would involve opening either of the two seismic Category 1 power-operated relief valves on the pressurizer, which discharges to the pressurizer relief tank. Concern was expressed that the pressurizer relief tank might not be designed for continuous operation and may not have safety-grade equipment to provide for intermittent operation. Hence, these actions might result in rupture of the tank rupture disc and a release to containment.

The RHR system is fully contained in the containment building and it has not been shown to be qualified for operation under the high humidity moderate temperature environment which was postulated to result from bursting of the rupture disc. VEPCO has indicated that depressurization using the power operated relief valves (PORV's) was only intended for use in the event that the auxiliary spray line was irreparably lost. Past experience with rupture of pressurizer relief tank (PRT) rupture discs was cited in which the resultant environment did not adversely affect the performance of a similarly designed RHR system. VEPCO has inserted cautionary guidance to the operator in his procedure for depressurization using the PORV to avoid rupture of the PRT disc. Conveniently located instrumentation was identified to monitor RCS pressure and PRT status while manipulating the PORV. Using the Surry simulator, VEPCO has demonstrated that the depressurization can be performed expeditiously without rupture of the PRT disc. We will require that this demonstration be supplemented by a confirmatory test of this backup depressurization capability (to the conditions at RHR cut-in), using the same shutdown procedure at the North Anna plant prior to startup

after first refueling. The license will be conditioned to require VEPCO to demonstrate depressurization capability prior to startup following the first refueling outage.

Branch Technical Position RSB 5-1 requires that a natural circulation test with supporting analysis be conducted to demonstrate the ability to cool down and depressurize the plant and to demonstrate that boron mixing is sufficient under such circumstances. Comparison with performance of previously tested plants of similar design may be substituted for these tests, if justified. VEPCO plans to reference tests to be conducted at Diablo Canyon to meet this requirement. VEPCO reviewed differences between Diablo Canyon and North Anna which might affect boron mixing under natural circulation.

VEPCO's comparisons of system and upper head region characteristics for North Anna Unit 2 and Diablo Canyon suggest that the results of the Diablo Canyon test and supporting analysis should satisfy the BTP RSB 5-1 requirement. However, we plan to defer reaching a conclusion on this matter until the Diablo Canyon results have been reviewed. If the Diablo Canyon tests are not completed or do not provide satisfactory results, VEPCO will be required to submit such tests results applicable to North Anna Unit 2 prior to startup following the first refueling. The license will be conditioned to require VEPCO to provide results from the Diablo Canyon test regarding the above matter or submit test results applicable to North Anna Unit 2 prior to startup following the first refueling outage.

This testing is not necessary for first cycle operation of North Anna. The major purpose of the natural circulation test is to obtain information on the time needed to take the plant from hot standby to the cut-in point of the RHR system under conditions such as extended loss of offsite power when the RCS pumps are not available. It would be preferable to run this test after the first reload when the decay heat is relatively large. This would result in more meaningful test data and testing under conditions more representative of those occurring over the 40-year plant life. For North Anna, operation at full power is acceptable because the seismic condensate storage tank supply is backed up by the seismic Category 1 service water system. With this supply to the auxiliary feedwater system, the plant could be maintained at hot standby or slowly brought to the cut-in point of the RHR system.

On the basis of our review we have determined that the North Anna Unit 2 residual heat removal system meets Branch Technical Position RSB 5-1, which is based upon Criterion 34 of the General Design Criteria of Appendix A to 10 CFR Part 50, and therefore is acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System (ECCS)

6.3.6 Post-Loss-of-Coolant Accident (LOCA) Sump Debris

6.3.6.1 Introduction

We have reviewed the design of the emergency core cooling system containment sump screen and have determined that additional protection from core blockage due to containment debris entrained in the recirculating coolant need not be provided. We had previously concluded that the low power operation program could safely proceed while additional information was gathered and positions were developed. Since the issuance of Supplement No. 10 to the Safety Evaluation Report, we have rereviewed the overall issue of debris in the emergency core cooling system recirculation system and our evaluation is discussed in the following paragraphs.

6.3.6.2 Housekeeping

We have evaluated housekeeping requirements within containment to preclude debris from non-loss-of-coolant accident sources, e.g., maintenance and inspection activities.

The North Anna Power Station, Unit 2 quality assurance program establishes written guidelines for assuring that good housekeeping practices are followed during maintenance. The North Anna Unit 2 Technical Specifications include surveillance requirements which are implemented pursuant to written procedures. The requirements include inspections to verify that no loose debris which could be transported to the sump remains in the containment, periodic inspections of the containment sump suction inlets to ensure that they are not blocked by debris, and inspection of the sump components (trash racks, screens, etc.) to verify that structural distress or corrosion is not present.

The North Anna Unit 2 Technical Specifications and surveillance requirements adequately address control of loose debris in the Unit 2 containment. The Office of Inspection and Enforcement will monitor the compliance of the licensee with the Technical Specification requirements.

We find the housekeeping provisions for the North Anna Power Station, Unit 2 to be acceptable.

6.3.6.3 Small Debris

We have considered materials which might be capable of being transported to the sump which would have a tendency to form particles small enough to pass through the fine screens in the sump.

Virtually all of the piping insulation in the containment, particularly in the lower containment regions, is of the metal-mirrored type. This material is not expected to float or to form small particles as a result of pipe whip or jet impingement.

The seven types of insulation materials used inside containment were analyzed by VEPCO for their properties, location, and quantity. The most plausible source of debris was identified as the steam generator cubicles. Studies were conducted by VEPCO to determine potential paths to the containment sump. Particle settling analyses showed that only a small percentage of this debris with a specific gravity greater than 1.0 would reach the sump. Buoyant debris would float on the water surface above the screen. Less plausible sources (pressurizer cubicle, reactor cavity, pressurizer spray line area) have a smaller amount of potential debris material with a lesser likelihood of reaching the sump.

VEPCO has verified that sand or similar material is not used in the containment for purposes such as subcompartment blowout or sand filled tanks or sandbags for the reactor cavity annulus biological shielding.

With regard to other potential sources of debris (i.e., paint chips or other degraded material), periodic surveillance inspections are provided to detect occurrences of degraded materials.

Based on the above conditions, we conclude that the North Anna Unit 2 design avoids the use of materials in the containment which would be likely to produce small-sized debris in significant quantities.

6.3.6.4 Larger Debris

We have considered the use of materials which would have the potential to block the containment sump screens if transported to the screens as a result of an accident. The present design of the containment sump has been modeled in one-third scale and successfully tested under conditions of up to fifty percent screen blockage.

Virtually all of the piping insulation in the containment and particularly in the lower containment regions is of the metal-mirrored type. Based on the observations made during our site visit, we believe it unlikely that a significant quantity of metal-mirror insulation debris would be transported to the sump. This belief is based largely on the great number of obstructions in the form of piping of varying sizes, pipe hangers, snubbers, pipe support members, structural steel, platforms, cabling, motors, and stairways, to the passage of a material like metal-mirror insulation to the sump.

6.3.6.5 Emergency Core Cooling System Status

We have reviewed the adequacy of the information available to the control room operator to monitor the low pressure injection system status during recirculation cooling. We conclude that sufficient information (e.g., flow rate, pump motor

current, pump suction pressure, and pump discharge pressure) is available to the operator to detect low pressure injection system performance degradation. Emergency operating procedures require the maintenance of a "Post LOCA Log" for the purpose of monitoring low pressure injection system performance, and this log is complemented by reference information (pump curves and decay heat curves) required to be available which is used to determine low pressure injection system performance. Low pressure injection system reliability has been emphasized at North Anna Unit 2 in preoperational tests and in operator training. The North Anna Unit 2 operators are specifically instructed in recognition and mitigation of LPI performance degradation. The North Anna Unit 2 loss-of-coolant accident emergency operating procedures also include guidance to alert the operator of the symptoms of inadequate core cooling.

Based on VEPCO's past experience with emergency core cooling system pump reliability, operator training which addresses emergency core cooling system performance degradation, and procedures to monitor emergency core cooling system performance, we find the above measures acceptable to monitor emergency core cooling system performance during the recirculation mode at North Anna Unit 2.

6.3.6.6 Summary

Based on the considerations noted above with respect to housekeeping requirements, the avoidance of materials likely to form small-sized debris, the lack of an apparent mechanism for blockage of more than the previously tested value of fifty percent of the screen area by larger debris, and the ability to monitor and control the low pressure injection system status, we conclude that the present design of North Anna Unit 2 provides reasonable assurance that the post-loss-of-coolant accident recirculation of core coolant will not be impaired by debris, meets the requirements of 10 CFR 50.46 and Criterion 35 of the General Design Criteria, given in 10 CFR 50 Appendix A, and is therefore acceptable.

6.5 Containment Pressure Boundary Fracture Toughness

The fracture toughness of the ferritic materials that constitute the containment pressure boundary of the North Anna, Unit 2 nuclear plant was reviewed to assess compliance with Criterion 51 of the General Design Criteria (GDC-51), "Fracture Prevention of Containment Pressure Boundary." The North Anna Unit 2 containment is a load-bearing reinforced concrete structure with a thin steel liner on the inside surface which is designed to serve as a membrane providing leak tightness. The fracture toughness requirements of GDC-51 apply to those ferritic steel parts of the containment pressure boundary which are not supported by concrete and are thus load-bearing. These materials are typically applied in containment penetrations such as the equipment hatch, personnel airlock, and pipe system penetrations.

The applicant has stated in the FSAR that the ASME Code Section III, 1968 Edition, was applied in the fabrication of the equipment hatch, while the piping system penetrations were fabricated in accordance with the Nuclear Power Piping Code, USAS B 31.7-1969. Compliance with the requirements of the ASME Code for the ferritic steel parts of the containment pressure boundary satisfies the requirements of GDC-51.

In late 1979, we reviewed the ASME Code fracture toughness requirements for metal containment and other safety-related components. Based on our review, we have determined that past and current fracture toughness requirements contained in the ASME Code for some parts of the containment pressure boundary can be significantly less stringent than those currently contained in the Code for other safety-related equipment and may not ensure compliance with GDC-51 for all areas of the containment pressure boundary. We have initiated a study designed to review fracture toughness criteria for containment pressure boundary materials for the purpose of defining those fracture toughness criteria which most appropriately address the requirements of GDC-51. When this study is completed late in 1980, the fracture toughness criteria previously applied to containment pressure boundary materials will be re-evaluated to ensure that under operating, maintenance, testing, and postulated accident conditions (1) ferritic materials behave in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized.

For North Anna, the limiting material is the ring forging used in the hot pipe penetrations. Based on our evaluation at the limiting material temperature condition, we conclude that the limiting material will be maintained at no less than 60°F above the nil ductility temperature. Based upon this information on North Anna Unit 2, the containment meets the requirements of GDC-51.

7.0 INSTRUMENTATION AND CONTROL

7.2 Reactor Trip System

7.2.4 Anticipated Transients Without Scram (ATWS)

In Section 7.2.4 of Part I of Supplement No. 10 to the Safety Evaluation Report, we stated that the plant can be safely operated at low power prior to completion of our review regarding anticipated transients without scram because of the expected plant response to relevant anticipated transients without scram events at power levels not exceeding five percent.

In a pressurizer water reactor, the anticipated transients which require prompt action to shut down the reactor in order to avoid plant damage and possible offsite effects can be classified in two groups: those that isolate the reactor from the heat sink and those that do not. (A list of these transients is included in Appendix IV of Volume II of NUREG-0460, April 1978). In general, the consequences of both of these types of events are an increase in reactor power or system pressure, or both. In Section 6.3 of NUREG-0460, Volume I, potentially unacceptable consequences of anticipated transients without scram events for pressurized water reactors of designs like North Anna are indicated to include (1) pressure rises that could threaten the integrity of the reactor coolant pressure boundary, (2) loss of core cooling, and (3) leakage of radioactive material from the facility.

In NUREG-0460, we concluded that for plants which fall within the envelope of the Westinghouse generic anticipated transient without scram analyses, the anticipated transient without scram acceptance criteria will not be violated if the actuation circuitry of turbine trip and auxiliary feedwater systems which are relied upon to mitigate anticipated transient without scram consequences are sufficiently reliable and are separate and diverse from the reactor protection system. Additionally, the functionality of valves required for long-term cooling following the postulated anticipated transient without scram events has to be demonstrated.

The NRC's Regulatory Requirements Review Committee has completed its review and concurred with our approach described in Volume 3 of NUREG-0460 insofar as it applies to North Anna Unit 2. We issued requests for the industry to supply generic analyses to confirm the anticipated transient without scram mitigation capability described in Volume 3 of NUREG-0460. The staff evaluation of these reports was published as NUREG-0460, Volume 4, in March 1980.

We plan to present our recommendations on anticipated transients without scram to the Commission in August 1980.

The Commission would determine required modifications to resolve anticipated transient without scram concerns as well as the required schedule for implementation of such modifications. North Anna Unit 2 would, of course, be subject to the Commission decision on this matter. The following discusses the bases for operation of North Anna Unit 2 at power levels up to full power while final resolution of anticipated transients without scram is before the Commission.

In NUREG-0460, Volume 3, we state: "The staff has maintained since 1973 (for example, see pages 69 and 70 of WASH-1270) and reaffirms today that the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is no undue risk to the public from ATWS. This conclusion is based on engineering judgement in view of: (a) the estimated arrival rate of anticipated transients with potentially severe consequences in the event of scram failure; (b) the favorable operating experience with current scram systems; and (c) the limited number of operating reactors."

In view of these considerations and our expectation that the necessary plant modifications will be implemented in one to four years following Commission decision on anticipated transients without scram, we have generally concluded that pressurized water plants can continue to operate because the risk from anticipated transient without scram events in this time period is acceptably small. As a prudent course, in order to further reduce the risk from anticipated transients without scram events during the interim period before completing any plant modifications determined by the Commission to be necessary, we have required that the following steps be taken:

- (1) Emergency procedures be developed to train operators to recognize an anticipated transient without scram event, including consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicators, pressurizer relief valve and safety valve indicators, and any other alarms annunciated in the control room with emphasis on alarms not processed through the electrical portion of the reactor scram system.
- (2) Operators be trained to take actions in the event of an anticipated transient without scram, including consideration of manually scrambling the reactor by using the manual scram button, prompt actuation of the auxiliary feedwater system to assure delivery of the full capacity of this system, and initiation of turbine trip. The operator should also be trained to initiate boration by actuation of the high pressure safety injection system to bring the plant to a safe shutdown condition.

We consider these procedural requirements an acceptable basis for interim operation of the North Anna Unit 2 plant in accordance with Criteria 10, 15 and 20 through 29 of 10 CFR 50 Appendix A, General Design Criteria based on our understanding of the plant response to postulated anticipated transient without scram events.

In response to our requirements on operator training and emergency procedures, VEPCO submitted on January 10, 1980, emergency operating procedures for the postulated anticipated transient without scram (ATWS) events.

Following review of the procedures by us, VEPCO revised the procedures to accommodate staff comments. The revised procedures were reviewed as part of the staff review of emergency procedures addressed in Section I.C.1 of this supplement. The instructions provided in the procedure for Reactor Trip permit the operator to diagnose an ATWS event and take appropriate actions required for minimizing its effects and bringing the plant to a safe shutdown condition.

The instructions include the descriptions of the automatic responses of the plant as well as the operator's actions taken immediately after he diagnoses ATWS and later when he attempts to bring the plant to a cold shutdown condition.

7.2.4.1 Generic Task Action Plan

Item A-9 Anticipated Transients Without Scram (ATWS)

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reactor (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired, then an "anticipated transient without scram," or ATWS, would have occurred.

The anticipated transient without scram issue and the requirements that must be met by the applicant prior to operation of North Anna Unit 2 are discussed in Section 7.2.4 above. The requirements set forth are for the interim period pending completion of Task A-9 and implementation of additional requirements if found to be necessary.

As stated in Section 7.2.4 above, VEPCO has submitted procedures for anticipated transients without scram. These procedures were reviewed as part of our review of the VEPCO emergency procedures discussed in Section I.C.1 of this supplement. We have concluded that the emergency procedures adequately address ATWS mitigating actions for operation at full power.

7.9 Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation

In Section 7.9 of Part I of Supplement No. 10 to the Safety Evaluation Report, we stated that on November 30, 1979, the Office of Inspection and Enforcement issued IE Bulletin 79-27 "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation" to all power reactor facilities with an operating license and to those nearing licensing. This bulletin outlined actions to be taken to address

control system malfunctions and significant loss of information to the control room operator as a potential consequence of the loss of Class IE and non-class IE buses supplying power to these plant systems. Further, IE Information Notice 80-10, issued on March 7, 1980, provided information relating to the Crystal River Unit 3 event of February 26, 1980, in which a significant loss of information to the operator resulted from a loss of power to a portion of the plant instrumentation system. We also indicated that our review of this matter for full power operation was not yet completed.

As a result of these concerns for operating plants, VEPCO conducted a thorough review of electrical one-line diagrams, flow diagrams, and procedures used by the plant operators to achieve safe hot and cold shutdown conditions. Based on these reviews, additional procedures have been prepared and implemented to ensure satisfactory capability for hot or cold shutdown upon loss of any single electrical bus. In addition, certain design modifications were identified to further improve the ability of the operators to evaluate plant status, particularly through bus reassignment of certain displays. We will require that the modifications be implemented at the next scheduled outage of sufficient duration, or within six months of issuance of a full power license, whichever comes first. Accordingly, the license will be conditioned to reflect this action. In addition, as stated above VEPCO in the interim has developed procedures to deal with such events.

Based on our review we find that the response to the Bulletin meets Criterion 13 of the General Design Criteria and is acceptable.

7.10 Engineered Safety Features (ESF) Reset Controls

On March 13, 1980, the Office of Inspection and Enforcement issued Bulletin 80-06 "Engineered Safety Feature (ESF) Reset Controls" to address the concern that the use of reset pushbuttons alone could permit certain engineered safety feature system components to revert to the normal state following safety system actuation.

As a result of these concerns for operating plants, VEPCO conducted detailed drawing reviews, including schematic level review where appropriate. Reset actions were also tested during the North Anna Unit 2 Preoperational Test Program after this concern was discovered. The review and testing showed that a number of changes were required to prevent certain safety related equipment from changing position during override or reset of a safety actuation signal. These modifications have been or will be completed prior to full power operation and our Office of Inspection and Enforcement will verify that the modifications have been completed. Each modification was tested at the time of implementation which completes the reset testing required by the Bulletin. To further verify proper reset action, all reset actions will be tested again during the first refueling outage. We will condition the license to reflect this action.

Based on our review, we find that the response to the Bulletin meets Criterion 23 of the General Design Criteria and is acceptable.

8.0 ELECTRICAL POWER SYSTEMS

8.3 Onsite Power System

8.3.2 Diesel Generator Reliability

In Section 8.3.2 of Supplement No. 10 to the Safety Evaluation Report we indicated that the applicant had been requested to review the diesel generator design with regard to the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." We also requested information concerning the design of the fuel oil storage and transfer system. In a letter dated January 31, 1980, VEPCO indicated how they meet or will meet the recommendations of NUREG/CR-0660 and our concerns regarding fuel oil storage and transfer system.

We have reviewed the information provided by VEPCO and have determined that conformance to the recommendations is as follows:

<u>Recommendation</u>	<u>Conformance</u>
(1) Moisture in Air Start System	Yes
(2) Dust and Dirt in Diesel Generator Room	Yes
(3) Turbocharger Gear Drive Problem	Not Applicable
(4) Personnel Training	Partial
(5) Automatic Prelube	No
(6) Testing, Test Loading and Preventative Maintenance	Partial
(7) Improve Identification of Root Cause of Failures	Yes
(8) Diesel Generator Ventilation and Combustion Air Systems	Yes
(9) Fuel Storage and Handling	Partial
(10) High Temperature Insulation	*

*Explicit conformance is considered unnecessary by the staff in view of the equivalent reliability provided by the design, margin and qualification testing requirements that are normally applied to emergency standby diesel generators.

<u>Recommendation</u>	<u>Conformance</u>
(11) Engine Cooling Water Temperature Control	Yes
(12) Concrete Dust Control	Yes
(13) Vibration of Instruments and Controls	Partial

On the basis of our review we have concluded that there is sufficient assurance of diesel generator reliability to warrant unrestricted plant operation through the first refueling period. However, to assure long term reliability of the diesel generator installations we require that the following design and procedural modifications be implemented prior to the startup following the first refueling.

(1) Personnel Training:

Preventative maintenance, minor repairs, and trouble shooting for the emergency diesel generators is performed by the plant's electrical and mechanical maintenance personnel, but no specific training concerning diesel generator maintenance and trouble shooting is being provided for these personnel. VEPCO states that any maintenance, except routine minor repairs and adjustments, is performed under the direction of the manufacturer's representative. We find this unacceptable. We require that a complete formal training program be implemented for all the mechanical and electrical maintenance and quality control personnel, including supervisors, who will be responsible for the maintenance and availability of the diesel generators. The depth and quality of this training program shall be at least equivalent to that of training programs normally conducted by major diesel engine manufacturers.

(2) Automatic Prelube:

The lubrication system for the diesel at North Anna includes manually started ac prelube pumps which are used only during non-emergency manual starts. VEPCO stated that "Fairbanks Morse recognizes that dry starts will be required in emergency conditions." Dry starting the diesel generators under emergency conditions results in momentary lack of lubrication at the various moving parts which can eventually lead to failures with resultant equipment unavailability. We require that these pumps shall be used for all modes of diesel engine starting. The objective is to improve the availability of this equipment on demand. The pumps shall be powered from a reliable dc power supply and installed in the system to operate in parallel with the engine driven lube oil pump. In an automatic or manual start, the prelube pump should operate only during the engine cranking cycle or until a satisfactory lube oil pressure is established in the engine main lube oil distribution header. The prelube pump should also be provided with manual start.

(3) Test Loading:

The diesel generator manufacturer (Fairbanks Morse) of the North Anna Diesel Generators recommends loading the engine to about 50 to 75 percent of full load for one hour after eight hours of unloaded operation. VEPCO states that they are modifying their operating procedures to insure "that the engine is loaded up prior to securing the unit, after extended no-load operation." VEPCO did not define "extended no-load operation." We require that the operating procedures be modified to require loading the engine up to 50 to 75 percent of full load for one hour after eight hours of continuous no load operation.

(4) Fuel Storage and Handling:

(a) Diesel Generator Day Tank: The overflow line from the diesel generator day tank, as presently designed, permits fuel oil to be spilled to the ground whenever the tank is overfilled or in the event of level controller malfunction. We find this design unacceptable because the capacity of the seven day fuel oil storage tank could be compromised and because a ground oil spill is a fire hazard that could endanger the availability of the diesel generators. We require the day tank overflow line to be rerouted to return excess fuel to the seven day fuel oil storage tank.

(b) Seven Day Fuel Oil Storage Tank: The Final Safety Analysis Report description and piping and instrument diagrams indicate that the seven day diesel generator fuel oil storage tank is not provided with an alternate means for filling it in the event of an emergency. Presently fuel to these tanks is replenished from the above ground, non-seismically designed bulk fuel storage system and, in the event of a design basis earthquake, fuel cannot be introduced directly into the seven day storage tank from tanker truck or other means. This design is not acceptable. We require that each seven day fuel oil storage tank be provided with a seismic - Category 1, tornado missile, and flood protected emergency fill line. Each fill line shall have a shut-off valve, a strainer, and a truck fill connection consisting of a hose coupling with cap and chain.

(5) Vibration of Instruments and Controls:

VEPCO stated that there are three diesel generator control cubicles in the diesel generator room, one panel is floor mounted while the others are engine skid mounted. It also stated that vibration induced failures of the skid mounted panels and control equipment are unlikely, since the diesel generators have been operated over a period of three years without vibration induced failures in the control equipment, and the equipment has been seismically qualified by static analysis and tests. We disagree with this statement. A three year operating period without vibration induced failures in the engine skid mounted control equipment is not satisfactory justification that vibration induced failures will not occur over the lifetime of the equipment. In addition seismic qualification of the panels and control equipment does not qualify the equipment for continuous operation under severe vibrational stresses, unless the skid mounted panels and equipment have been specifically developed, tested, and qualified for these conditions. We require that VEPCO either provide test

results and results of analyses which qualify the engine skid mounted control cubicles for the severe vibrational stress that will be encountered during engine operation, or floor mount the skid mounted panels and control equipment presently furnished with the diesel generators.

As stated above, we have concluded that there is sufficient assurance of diesel generator reliability to warrant unrestricted plant operation through the first refueling period. To assure long-term reliability, the operating license will be conditioned to require VEPCO to implement the above design and procedural modifications prior to the startup following the first refueling outage.

The present diesel generator design meets the requirements of Criteria 17 and 21 of the General Design Criteria of Appendix A to 10 CFR 50. Upon completion of the above changes and modifications, the design of the diesel generator and its auxiliary systems will also be in conformance with recommendations of NUREG CR/0660 for enhancement of diesel generator reliability, and the related NRC guidelines and criteria. We therefore conclude that this will provide reasonable assurance of diesel generator reliability through the design life of the plant.

9.0 AUXILIARY SYSTEMS

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

The staff has received VEPCO's proposed fire protection program and fire hazards analysis. The fire protection program was reviewed against the guidelines of Appendix A to Branch Technical Position APCS 9.5-1, supplemental staff guidelines dated June 14, 1977, and applicable NFPA standards. The staff concluded that the fire protection program meets the requirements of Criterion 3 of the General Design Criteria of Appendix A to 10 CFR 50, and is therefore acceptable for full power operation.

By letter dated June 30, 1980, the licensee informed us that three modifications, of which one is an alternate shutdown system would not be implemented by November 1, 1980. However, subsequently, the licensee has agreed in a letter dated July 25, 1980 to complete the alternate shutdown system by April 1981, and the other two modifications by November 1, 1980. The licensee will have to shut down to complete these modifications.

On April 23, 1980, the Commission approved a proposed rule concerning fire protection. The proposed rule and its Appendix R to 10 CFR 50 were developed to establish the minimum acceptable fire protection requirements necessary to resolve certain areas of concern in contest between the staff and licensees of plants operating prior to January 1, 1979. On May 23, 1980, the Commission issued a Memorandum and Order (CLI-80-21) which states that: "The combination of the guidance contained in Appendix A to BTP 9.5-1 and the requirements set forth in this proposed rule define the essential elements for an acceptable fire protection program at nuclear power plants docketed for Construction Permit prior to July 1, 1976, for demonstration of compliance with General Design Criterion 3 of Appendix A to 10 CFR Part 50." (p. 19) In the event that the rule, when it becomes an effective rule, has provisions which apply to North Anna Unit 2, such provision will be implemented in accordance with the rule.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.7 Turbine Missiles

In Supplement No. 10 to the Safety Evaluation Report, we stated that the Unit 2 Technical Specifications will require that VEPCO conduct a preservice inspection of the turbine.

During November 1979, we became aware of a problem of stress corrosion cracking in Westinghouse turbines. Meetings were held with Westinghouse to ascertain the probable extent and severity of the problem. Westinghouse was recommending early inspection of turbines that had long operating times and particularly those machines with discs of marginal material properties or history of secondary water chemistry problems. Since then, inspections have been performed on about eighteen more Westinghouse turbines, with indications of cracking, some severe, found in most of them. Investigations are continuing.

In accordance with Unit 2 Technical Specifications, VEPCO performed the preservice inspection and submitted a report of the inspection including the material properties of the Low Pressure Turbine discs, as well as the calculations of critical crack sizes and predicted crack growth rates. The method used by VEPCO to predict crack growth rates is based on evaluating all of the cracks found to date in Westinghouse turbines, past history of similar turbine disc cracking, and results of laboratory tests. This prediction method takes into account two main parameters; the yield strength (and stress) of the disc, and the temperature of the disc at the bore area where the cracks of concern are occurring. The higher the yield strength of the material and the higher the temperature, the faster the crack growth rate will be.

We have evaluated the data and calculation submitted by VEPCO and, in addition, performed our own calculations for crack growth and critical crack size. We conclude that North Anna 2 may be safely operated. For continued assurance of turbine integrity, the low pressure turbine discs shall be inspected during the second refueling outage. The license will be conditioned to require VEPCO to inspect the turbine prior to startup following the second refueling outage.

13.0 CONDUCT FOR OPERATIONS

13.2 Training Programs

In Section 13.2 of Part I of Supplement No. 10 to the Safety Evaluation Report, we stated that VEPCO must assure that it has a sufficient number of licensed personnel for full power operation. VEPCO has expanded their training program to include areas of thermodynamics, fluid mechanics, and heat transfer. These subjects have also been included in the requalification program. During May, 1980, NRC additional examinations were administered. The following complement of licensed operators are available for operation of North Anna Unit 1 and 2 at this time.

<u>No.</u>	<u>Type of License</u>
14	Unit 1 and 2 Senior Operator Licenses
7	Unit 1 and 2 Operator Licenses
9	Unit 1 Senior Operator Licenses
8	Unit 1 Operator Licenses

Additional examinations were administered during the first week of July, and we expect our review of these examinations to be completed in late July.

In Section 6.0 of the Technical Specifications, the minimum shift crew composition for operation at full power for North Anna Units 1 and 2 is described and is presented in Table 13.1 of this report. Prior to the issuance of a full power operating license, we will determine the number of licensed operators VEPCO has available to operate North Anna Units 1 and 2 and will assure that VEPCO has a sufficient number of licensed operators to meet the requirements specified in the Technical Specifications and 10 CFR 50.57(a)(4).

TABLE 13.1
MINIMUM SHIFT CREW COMPOSITION[#]

Condition of Unit 2 - No Fuel in Unit 1

LICENSE CATEGORY	APPLICABLE MODES†	
	1, 2, 3 & 4	5 & 6
SOL	2	1*
OL	2	1
Non-Licensed	2	1

Condition of Unit 2 - Unit 1 in MODES 1, 2, 3 or 4

LICENSE CATEGORY	APPLICABLE MODES†	
	1, 2, 3 & 4	5 & 6
SOL**	2	2*
OL***	3	3
Non-Licensed	3	3

Condition of Unit 2 - Unit 1 in MODES 5 or 6

LICENSE CATEGORY	APPLICABLE MODES†	
	1, 2, 3 & 4	5 & 6
SOL**	2	1*
OL***	3	3
Non-Licensed	3	3

* Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS after the initial fuel loading.

** Each unit will be supervised by a shift supervisor who is a licensed SRO on that unit; this may be a single individual if he is suitably licensed. The second senior operator licensed for each unit must be stationed in the control room area at all times when the unit is in operating modes 1 through 4; this also could be a single individual if he is appropriately licensed.

*** A reactor operator licensed for each unit must be at the controls of that unit at all times when fuel is in the reactor. Also, a relief reactor operator licensed for each unit must be available on-shift. This could be a single individual if he is licensed for both units.

Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

† Mode 1 Power Operation
 Mode 2 Startup
 Mode 3 Hot Standby
 Mode 4 Hot Shutdown
 Mode 5 Cold Shutdown
 Mode 6 Refueling

15.0 ACCIDENT ANALYSES

15.1 Normal Operations and Anticipated Operational Transients

In Section 15.1 of Part I of Supplement No. 10 to the Safety Evaluation Report, we stated that we required that the applicant commit to provide prompt responses to additional information requirements regarding our review of Westinghouse transient analysis codes dealing with steam line and feedline break accidents.

The plant response analyses for postulated steam line and feedwater line breaks were evaluated with the use of the MARVEL computer program Westinghouse Topical Report (WCAP-8844). MARVEL is a systems code designed to model transients which do not result in primary side two-phase conditions. The primary system is treated homogeneously. The MARVEL computer program is presently under review by the NRC staff. Due to some simplified assumptions used in the development of the code, the staff requested confirmation of the steam line break and feedwater line break analyses with a more detailed model, as documented in Westinghouse Topical Reports WCAP-9226 "Reactor Core Response to Excessive Secondary Steam Releases," WCAP-9230 "Report on the Consequences of a Postulated Main Feedline Rupture," and WCAP-9236 "NOTRUMP - A Nodal Transient Steam Generator and General Network Code." VEPCO has committed to provide the confirmatory analyses using this more detailed model and to respond to resulting questions in an acceptable time frame. A commitment to support an audit analysis, if required by us, was also provided. The license will be conditioned to require VEPCO to provide the confirmatory analysis and support a confirmatory analysis.

The analytical methods used for postulated transients and accidents are normally reviewed on a generic basis. Our review at this time indicates that there is reasonable assurance that the conclusions based on the Final Safety Analysis Report analyses will not be appreciably altered by the completion of the analytical methods review. If the final approval of the methods indicates revisions to the analyses are required, the licensee will be required to implement the results of such changes.

Based on previous acceptable analyses for Westinghouse plants, on a comparison with other industry models, on independent staff audit calculations, and on previous startup testing experience, we conclude that, with the exceptions noted above, the analytical methods used for North Anna Power Station, Unit 2 are acceptable for the issuance of a full power operating license and will assure conformance with the dose guidelines given in 10 CFR Part 100 for various postulated accidents.

17.0 QUALITY ASSURANCE

Our review of the quality assurance program description for the operations phase for the North Anna Unit No. 2 has verified that the criteria of Appendix B to 10 CFR Part 50 have been adequately addressed in Section 17.2 of the FSAR through Amendment 67. This determination of acceptability included a review of the list of safety-related structures, systems, and components (Q-list) to which the quality assurance program applies. The review of the Q-list was performed by using a revised procedure that involves our technical review branches and significantly enhances the staff's confidence in the acceptability of the Q-list. This review resulted in the identification of several differences between the current Q-list and our requirements. By letter, dated July 2, 1980, and July 22, 1980, VEPCO has resolved these differences to our satisfaction. We therefore find the Q-list to be acceptable for full power operation.

22.0 TMI-2 REQUIREMENTS

22.1 Introduction

In a letter dated June 26, 1980, we advised all applicants for construction permits and operating licenses of the Commission's policy statement regarding the requirements to be met for current operating license application. The requirements are derived from NRC's Action Plan (NUREG-0660) and are found in NUREG-0694, "TMI-Related Requirements for New Operating Licenses." These requirements are deemed to be necessary and sufficient for responding to the TMI-2 accident and current operating license applications should be measured against the NRC regulations as augmented by these requirements.

The requirements discussed in NUREG-0694 were listed in four categories: 1) Those required for fuel loading and low power testing requirements; 2) those required for full-power operation; 3) those requiring internal NRC action; and 4) those required to be implemented by a certain date.

Since requirements for fuel loading and low power testing were addressed in Part II of Supplement No. 10 to the North Anna Power Station Unit 2 Safety Evaluation Report, this supplement only addresses full power requirements (including NRC actions) and those requirements to be met as given in item (4) above.

Each applicable full power requirement (including NRC actions) and dated requirement is discussed below and follows the numbering sequence utilized in NUREG-0694.

22.2 FULL POWER REQUIREMENTS

I. Operational Safety

I.B.1 Management for Operations

I.B.1.1 Organization and Management Criteria

POSITION

Corporate management of the utility-owner of a nuclear power plant shall be sufficiently involved in the operational phase activities, including plant modifications, to assure a continual understanding of plant conditions and safety considerations. Corporate management shall establish safety standards for the operation and maintenance of the nuclear power plant. To these ends, each utility-owner shall establish an organization, parts of which shall be located onsite, to: perform independent review and audits of plant activities; provide technical support to the plant staff for maintenance, modifications, operational problems, and operational analysis; and aid in the establishment of programmatic requirements for plant activities.

The licensee shall establish an integrated organizational arrangement to provide for the overall management of nuclear power plant operations. This organization shall provide for clear management control and effective lines of authority and communication between the organizational units involved in the management, technical support, and operation of the nuclear unit.

The key characteristics of a typical organization arrangement are:

Integration of all necessary functional responsibilities under a single responsible head.

The assignment of responsibility for the safe operation of the nuclear power plant(s) to an upper level executive position.

DISCUSSION AND CONCLUSIONS

In Section I.B.1.1 of Part II of Supplement No. 10 to the Safety Evaluation Report, we stated that we found the VEPCO organization in regard to its capability to operate North Anna Nuclear Power Station Unit 2 to be acceptable for full power operation. However, we also indicated that the applicant's procedures regarding offsite technical support to the plant staff, in the event of an emergency were acceptable for operation at power levels not exceeding five percent.

Subsequent to the issuance of Supplement No. 10 to the safety evaluation report, we have completed our review of the procedures regarding offsite technical

support to the plant staff in the event of an emergency, for full power operation. On the basis of our review, we have determined that these procedures meet our requirement for full power operation and, therefore, are acceptable.

I.C. Procedures

I.C.1 Short-Term Effort

Analysis and Procedure Modification (2.1.9 - NUREG-0578)

POSITION

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;
2. Inadequate core cooling; and
3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required)
loss-of-coolant accident (LOCA) with forced flow, LOCA without forced flow.
2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.2.b of NUREG-0578).

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each Final Safety Analysis Report (FSAR). The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of

function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similar, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncover for some transients, and computer calculations could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncover, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the nuclear steam supply system (NSSS) vendors will be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

DISCUSSION AND CONCLUSIONS

I. Introduction

In Section I.C.1, Part II of Supplement No. 10 of the Safety Evaluation Report for North Anna Unit 2 and in our Safety Evaluation Report attached to Amendment 1 of License NPF-7 we stated that prior to operation above 5% power we would observe a simulation of selected North Anna Unit 2 emergency procedures conducted by North Anna personnel and a walk-through of at least one emergency procedure in the North Anna Unit 2 control room. The objective was to

verify that the emergency procedures adequately addressed successful mitigation of accidents and transients.

On July 2, and 3, 1980, a team of NRC personnel observed North Anna Unit 2 personnel participating in a simulation of several transients and accidents on the VEPCO simulator at the Surry Power Station. The transients and accidents included loss-of-coolant accidents (LOCA) in a range of break sizes, steam generator tube rupture, loss of main feedwater both from loss of main feedwater pumps and from a large feedwater line break inside containment, and recovery from inadequate core cooling. Some transients and accidents were run more than once and equipment failures such as loss of offsite power and failure of one emergency diesel generator, failure of scram breakers to open (ATWS), and failure of individual components in emergency core cooling systems (ECCS) and auxiliary feedwater system (AFW) were included in the simulated events. We observed the operators' actions and discussed their actions and the procedures with the operators as they performed the procedures and after each event.

On July 10, 1980, the team observed a walk-through of a small break LOCA in the North Anna Unit 2 control room. During the first hour of the event, we required that all actions be taken by the minimum shift crew allowed by plant Technical Specifications. After the first hour, the shift supervisor was permitted to request extra personnel as he deemed necessary. The Technical Support Center was activated after the first hour and manned by a minimum of personnel to carry on communications that were required to implement the emergency procedures. The Emergency Plan Implementation Procedures required to be performed by the Control Room Operators were carried out but onsite and offsite emergency response was simulated. Emergency preparedness is addressed in Item III.A.1.1 of this section.

The emergency procedures that VEPCO provided for our review had also been revised to reflect the analysis of small break loss-of-coolant accidents and inadequate core cooling in accordance with license requirements 2.D(6)a and Task Action Plan (NUREG-0660) Item I.C.1. The emergency procedures and power ascension program had been reviewed by the NSS Supplier, Westinghouse, and changes recommended by Westinghouse had been incorporated in compliance with Task Action Plan Item I.C.7.(a).

II. EVALUATION OF EMERGENCY PROCEDURES

A few procedural deficiencies were identified to VEPCO personnel during the simulation of transients and accidents at the Surry Simulator. During the plant walk-through at North Anna, we verified that VEPCO personnel had implemented the necessary changes. The revised procedures have been approved by the safety committee and the North Anna Unit 2 operators have reviewed the procedures and have been briefed on the changes and the bases for the changes.

Based on our review of the emergency procedures and our observation of the procedures being implemented in the simulator and in the plant walk-through, we

have concluded that the North Anna Unit 2 emergency procedures are adequate for operation at power levels up to 100 percent. We have concluded that the actions called for in Task Action Plan Items I.C.1.a(1) and (2), I.C.7(a), and I.C.8 have been adequately completed. Future actions addressed by Task Action Plan Items I.C.1.a(3) and I.C.9 may require future revisions to the emergency procedures. If necessary, these revisions will be identified in the long term program stipulated in Item I.C.9.

I.C.7 NSSS Vendor Review of Procedures Position

POSITION

Obtain NSSS vendor review of power ascension test and emergency procedures to further verify their adequacy.

This requirement must be met before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

Our evaluation of this matter is addressed in Item I.C.1 of this section.

I.C.8 Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants

POSITION

Correct emergency procedures as necessary based on the NRC audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of ac power, steam-line break or steam-generator tube rupture).

This action will be completed prior to issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

Our evaluation of this matter is addressed in Item I.C.1 of this section.

I.D.1 Control Room Design

In Section IV of Part II of Supplement No. 10 to the Safety Evaluation Report for North Anna Unit 2, we identified a number of corrective actions which we believed were necessary to improve operator effectiveness during emergency operations. VEPCO was required to implement several of these actions prior to achieving initial criticality. Accordingly, the operating license for North Anna Unit 2 was conditioned to reflect these actions. Our evaluation regarding this matter was presented in our letter of June 12, 1980, which permitted VEPCO to operate North Anna Unit 2 in Mode 2 to perform zero power physics tests. VEPCO was also required to implement additional actions prior to operation at power levels exceeding five percent of full power.

The Office of Inspection and Enforcement will audit the measures implemented by VEPCO to meet the corrective actions required prior to escalation beyond five percent of full rated power. The audit will verify implementation of the human factors improvements made to the control room which will serve to substantially improve the operator's ability to take effective control actions under stressful conditions. The corrective actions that VEPCO was required to implement and which will be verified by us prior to full power operation are:

1. Improve operator accessibility to Core Cooling Monitor displays.
2. Correct deficiencies associated with the strip chart recorders.
3. Install equipment for testing lamps on safeguards panels and establish a mechanism for testing other lamps important to safety.
4. Review all emergency and abnormal operating procedures and correct deficiencies. Perform sufficient procedure walkthroughs to ensure that all operators are familiar with and understand these procedures.
5. Correct violations of design convention.
6. Correct deficiencies in operator procedures for utilizing plant computer outputs.
7. Correct operational problems associated with the Hagan controllers.
8. Procure sufficient emergency air packs to supply all operators required to be in the control room during emergencies. Ensure that sufficient replacement air is available when needed. Train all operators to use the air packs.
9. Install sufficient protective guards to prevent inadvertent operation of J-handle switches located in the control room.
10. Assess control room staffing requirements during emergency operation.

11. Correct the control room noise problem.

In addition, the Office of Inspection and Enforcement will verify VEPCO's implementation of corrective administrative actions to improve general maintenance in the control room.

VEPCO will make a temporary installation of meter banding to display normal operating ranges prior to escalations beyond five percent of full power. VEPCO will make such installation permanent on each meter face for each loop when the loops are periodically calibrated, except that all engineered safety features system meters and all meters with safety significance shall be completed prior to the startup following the first refueling. With reference to item 3 above, VEPCO is not required to commit to purchasing and installing as soon as possible, data recording and logging equipment in the control room. However, we will require that VEPCO along with all other operating reactor licensees evaluate the benefits of installing such equipment in the control room to correct deficiencies associated with the trending of important parameters on strip chart recorders in use at most nuclear power plants, as part of their one-year control room design review.

CONCLUSION

Subject to verification by the Office of Inspection and Enforcement of the adequacy of the corrective actions taken by VEPCO to meet the human factors requirements resulting from our control room design review and stated herein and in Supplement No. 10 to the Safety Evaluation Report, we conclude that Unit 2 can be safely operated at full power.

POSITION

The TMI Task Action Plan states that applicants for operating licenses will perform a set of low power tests to increase the capability of shift crews and ensure training in plant evolutions and off-normal events. Near-term operating license facilities will be required to develop and implement intensified exercises during the low power testing programs. This may involve the repetition of startup tests on different shifts for training purposes.

DISCUSSION AND CONCLUSIONS

By letters of April 2 and April 21, Virginia Electric and Power Company (VEPCO) submitted draft procedures for conducting nine tests. The April 21 letter also transmitted the safety evaluation and training program for the tests. In a letter dated April 29, 1980, Westinghouse stated concerns with repeating two of the proposed tests, Startup from Stagnant Conditions and Boron Mixing and Cooldown, at plants other than Sequoyah. By letter dated June 5, 1980, VEPCO requested deletion of these two tests from its low power test program; however, a test similar to one of the tests would be performed at the end of the startup test program using decay heat. On June 13, 1980, VEPCO submitted test procedures that had been approved by their safety committee for the seven remaining tests (these seven tests were combined into four procedures). On June 18, 1980, VEPCO submitted changes to the test procedures that had also been approved by the North Anna safety committee. The special low power test program was reviewed and approval to conduct the tests was granted in Amendment 1 to Facility License No. NPF-7, North Anna Power Station, Unit 2 dated July 3, 1980.

The special low power test program, as approved by the staff was conducted at North Anna Unit 2 starting on July 3, 1980. NRC staff representatives were present to observe each test the first time it was performed. In addition, an NRC Resident Inspector was present for all tests. Special Tests 2-ST-6 (Cooldown Capability of the CVCS) and 2-ST-11 (Effect of Steam Generator Secondary Side Isolation on Natural Circulation) were each performed once. Test 2-ST-8 (Natural Circulation Verification) was performed five times and 2-ST-9 (Natural Circulation with Loss of Offsite Power and Loss of Offsite and Onsite AC Power) was performed four times. We have concluded that VEPCO satisfied the requirement for operator training by having every licensed operator participate in at least one test and observe two or more.

In letters dated July 17 and 22, 1980, VEPCO submitted the results of the special low power test program. Our evaluation is based on the preliminary test results and the observations made by the staff during conduct of the tests. Preliminary data have been reviewed which together with direct observation of the tests makes it possible to determine if the staff requirements have been fully met.

A summary of the first performance of each test follows.

1. Test ST-6 Cooldown Capability of the CVCS

This test was conducted at zero power with one reactor coolant pump running and all three steam generators isolated. Heat was removed from the primary system using the charging and letdown systems.

The maximum charging and letdown flow rate of approximately 120 gpm resulted in a Primary Coolant System temperature decrease of about 2°F per hour. The minimum flow rate of approximately 40 gpm resulted in a temperature increase of approximately 2.5°F per hour. The plant responded as expected during this test and all test objectives were met.

2. Test ST-8 Natural Circulation Verification

With the reactor at 3% power and heat being removed with all three steam generators, all three reactor coolant pumps were tripped and natural circulation was established. Primary system pressure increased to 2310 psig where one PORV lifted. The PORV reseated and the pressure was controlled using auxiliary spray. The primary coolant system stabilized at approximately 30°ΔT.

Following this portion of the test, a core flux map was run in natural circulation for comparison with the zero power flux map taken before the test.

In stable natural circulation, the average core exit thermocouple reading was 581°F. The average of the three hot leg RTDs was 580.5°F. A map of core exit thermocouple readings taken in natural circulation indicated that the core flow distribution did not change. Location of the high and low thermocouple readings stayed the same. There was a 6°F difference in high and low readings in natural circulation and a 3°F difference with forced flow.

The second part of Test ST-8 was to demonstrate that RCS saturation margin can be maintained without pressurizer heaters. With Natural circulation established, all pressurizer heaters were turned off. Both auxiliary and main spray valves were closed to ensure that spraying would not influence pressure drop. The depressurization rate over a 2-hour period was approximately 38 psi per hour. RCS temperature decreased at about 5°F per hour and RCS subcooling margin decreased by about 6°F per hour.

The last part of Test ST-8 was to determine the effect of decreased subcooling margin on natural circulation and the effects of charging and steam flows on saturation margin.

System pressure was decreased using pressurizer spray to a subcooling margin of approximately 30°F. Saturation margin could not be substantially increased by changing steam dump rate alone. Charging flow or pressurizer heaters were needed to increase saturation margins.

The plant responded as expected during this test. Lifting of the PORV was a normal system response with the specified test conditions.

3. Test ST-9 Natural Circulation with Loss of Offsite Power and Simulated Loss of All Offsite and Onsite AC Power

This test was conducted at 1% reactor power. To simulate loss of offsite power, the following actions were taken:

- A. Motor-driven auxiliary feedwater control valves were closed.
- B. Steam dump controllers were placed in manual control.
- C. Pressurizer backup heater groups 2 and 5 and control heaters group 3 were tripped and locked out.
- D. All three reactor coolant pumps and the operating main feedwater pump were simultaneously tripped.

Approximately eight minutes after initiation of the test, a PORV lifted at 2305 psig. Pressure was controlled using atmospheric steam dumps. The PORV reseated and the pressure remained below the PORV setpoint. Auxiliary feedwater was initiated normally and stable natural circulation was achieved with cold leg temperature of 547°F and ΔT of 22°F.

To simulate loss of all ac, both motor-driven auxiliary feed pumps were turned off. The feed pump house exhaust fans were also turned off.

This test deviated from a real loss of all ac in that the steam turbine-driven auxiliary feed pump was aligned to feed all three steam generators. The normal alignment at North Anna is to have each of the three auxiliary feed pumps feeding one steam generator. If all ac were lost, only one steam generator would receive feedwater. This realignment did not preclude meeting test objectives.

Operators were sent to the auxiliary feed pump house to realign the feedwater system to feed all three steam generators and to control feed flow by manually adjusting control valve position at the direction of the control room operator.

Steam generator levels were initially 40% to 45%. Levels dropped to about 35% and stabilized. By manually controlling flow, levels were maintained between 35% and 43% throughout the test. Stable natural circulation was established and maintained at approximately 22° ΔT .

The plant responded as expected during this test. Lifting of the PORV was a normal system response with the specified test conditions.

4. Test ST-11 Effect of Steam Generator Secondary Side Isolation on Natural Circulation

This test was initiated by tripping all three reactor coolant pumps with the reactor maintained at 1% power. After stable natural circulation was achieved with a ΔT of approximately 21°F, the B steam generator was isolated.

Following isolation of the B steam generator, ΔT in the isolated loop slowly decreased to about 9°F in one hour. ΔT in the other loops was at 27° and 25°. At this time it was suspected that some leakage was occurring around the feedwater isolation valve and steam isolation valve. Operators were dispatched to tighten the feedwater bypass valve and the other isolation valves. The test then continued approximately 4 hours at which time the isolated loop had a ΔT of 3°F and was stable.

The plant responded as expected during this test.

VEPCO is currently evaluating the test results for incorporation into the Surry Power Station Simulator. After completion of this evaluation, VEPCO will provide a report to us describing changes made to the simulator model as the result of the tests.

It is concluded that the special low power tests conducted at North Anna satisfy all requirements of Item I.G.1 of the Task Action Plan. This conclusion is based on the following:

1. All licensed operators received adequate training during the program by participating in at least one test and observing at least two or more.
2. Meaningful technical information was obtained on plant response to a variety of abnormal conditions.
3. At all times during the tests, the plant was under complete control and responded as predicted.
4. Acceptance criteria for each test as specified in the test procedures were met.
5. The operational safety criteria discussed in our "Safety Evaluation Report attached to Amendment No. 1 to Facility License No. NPF-7, North Anna Power Station, Unit 2," dated July 3, 1980, were not exceeded in any of the lists and all test parameters remained well within safety margins.

II. Siting and Design

II.B.1 Reactor Coolant System Vents

POSITION

Provide a description of the design of reactor coolant system and reactor vessel head high point vents that are remotely operable from the control room and supporting analyses. This requirement shall be met before issuance of a full-power license. See letter of September 27 and November 9, 1979.

DISCUSSION AND CONCLUSIONS

By letter dated January 10, 1980, and as supplemented by letters dated July 10 and July 17, 1980, VEPCO, provided their conceptual design for the TMI task action plan requirement to install reactor coolant system vents (Item II.B.1). VEPCO has designed the vent system to be remotely controlled and monitored. VEPCO has committed that the design is to be safety grade, seismically qualified, and single failure proof. Finally, VEPCO has stated that the system design is to be such that a break in the vent line is within the capability of one charging pump makeup, and is therefore, smaller than the definition of the smallest LOCA.

Our preliminary review of this information indicates that this conceptual design adequately addresses the requirements of our November 9, 1979 letter on vents. However, a detailed evaluation of the design has not been completed. Some areas that will require further detail are vent system qualification to operate under accident conditions, system testability to satisfy the requirements of IEEE 279, piping design, procedural guidelines and analyses.

Specifically, the criteria for venting initiation and termination have not been addressed. These guidelines for vent operation will address adequate core cooling and the potential for producing combustible mixtures in the containment. They must also provide methods and tests and/or analyses to assure adequate heat removal through the U-tubes of the steam generator. The guidelines are currently under development in a generic effort by VEPCO's NSSS.

VEPCO and Westinghouse have concluded that the vent system should not be operated without an indication of vessel water level (vessel water level is required per the TMI Task Action Plan, Item II.F.2). Components for the installation of vessel level instrumentation can not be obtained before mid-summer 1981. This timing is very close to the refueling outage and VEPCO has committed that both the vent and vessel level systems will be installed at that outage. (NRC staff review of the vessel level system has concluded that the delay of installation beyond January 1, 1981 deadline is acceptable given system installation during late 1981, consistent with scheduled or forced plant outage. Procedural guidelines and bases shall be submitted before January 1, 1981.)

VEPCO has estimated the required outage time for installation as one month for the vent system and two months for the vessel level system. The earliest possible installation dates are approximately early 1981 for vents and mid-1981 for level. Currently, refueling is planned to begin in late 1981, and the licensee has stated that this is the most appropriate time to install both vent and level systems.

The reasoning that vent operation has a link to the vessel level indication is that venting should proceed only with a reliable means of determining both the location of non-combustibles (e.g., reactor vessel head) and when to terminate the venting. The venting operation should be controlled and monitored to assure no resultant (or additional) core damage due to loss of inventory. Therefore, to assure core cooling Westinghouse has concluded that a direct, reliable indication of vessel level is needed to conduct the venting operation.

While VEPCO and Westinghouse have indicated that the vessel level instrumentation would be needed under all foreseen scenarios to operate the RCS vents, they did not preclude the potential for other scenarios where venting without vessel level may be desirable. However, it's our judgment that the additional several months extension would not significantly affect reactor safety.

We concur with VEPCO's and Westinghouse's conclusion that reactor vessel level is important in the initiation and control of venting. However, we will require that procedural guidelines and analytical bases be submitted to us by January 1, 1981 and that the vent system be installed and functional during late 1981.

On these bases we conclude that the applicant has provided an acceptable description of the vent conceptual design per the Task Action Plan full power requirements, but that further detailed review will be necessary as outlined above.

II.B.2 Plant Shielding

POSITION

Provide (1) a radiation and shielding design review that identifies the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by radiation during operations following an accident resulting in a degraded core, and (2) a description of the types of corrective actions needed to assure adequate access to vital areas and protection of safety equipment.

This requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.6.b, and letters of September 27 and November 9, 1979).

DISCUSSIONS AND CONCLUSIONS

The plant shielding design report was reviewed to evaluate the ability to operate essential systems required after a loss-of-coolant accident (LOCA) with significant core damage. The systems designed to function after an accident included: the high head safety injection (HHSI), portions of the chemical volume control system (CVCS) and safety injection system (SI), the low head safety injection (LHSI), recirculation spray system, sample system, containment atmosphere cleanup (hydrogen recombiner) system, and the auxiliary building sump and drain lines.

The remainder of the CVCS was excluded because it is isolated and because its use in a post-accident situation would be unacceptable. The residual heat removal (RHR) system was not considered because all piping in this system is located inside the containment.

Calculation of source terms and estimated dose rates used for shielding design are based on Stone and Webster computer codes "Activity-2" and "Radioisotope", Regulatory Guide 1.4, "Assumption Used For Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident For Pressurized Water Reactors", and TID-14844 "Calculation of Distance Factors for Power and Test Reactor Site." The licensee has produced "Radiation" zone maps to be used as an administrative guide in the control of access and reduction of personnel exposure during the course of an accident. All vital areas which require continuous or frequent occupancy in order to control, monitor, and evaluate the accident were identified. These areas include the control room, technical support center, the counting lab/health physics area, the operational support center, and security control center. Limited access is needed to such places as emergency power supplies and sampling stations.

The need for modification in 8 areas was identified. (See Section 22.3 Item II.B.2) Our evaluation of environmental qualification of equipment regarding this matter is discussed in Section 3.10.3 of this supplement. In Section 3.10.3 we concluded that in regard to environmental qualification of equipment, operation at full power is acceptable.

On the basis of our review, we conclude that the radiation and plant shielding design described by VEPCO meets our position in NUREG-0578 and is, therefore, acceptable for full power operation.

II.B.3 Postaccident Sampling

POSITION

Provide (1) a design and operational review of the capability to promptly obtain and perform radioisotopic and chemical analyses of reactor coolant and containment atmosphere samples under degraded core accident conditions without excessive exposure, (2) a description of the types of corrective actions needed to provide this capability, and (3) procedures for obtaining and analyzing these samples with the existing equipment.

This requirement shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.8a, and letters of September 27 and November 9, 1979.

DISCUSSION AND CONCLUSIONS

The licensee has provided the staff with copies of his interim procedures for post-accident sampling and analysis of reactor coolant and of contaminated atmosphere. These procedures establish preparations, actions, techniques, and instructions for the safe procurement, handling, and analysis of potentially highly radioactive samples, such as would be encountered following a reactor accident involving core damage. The staff has reviewed these procedures and has found them to be acceptable until installation of an improved sampling system is complete.

The licensee has also provided the staff with a proposed design for an improved sampling system. The proposed design is consistent with the criteria of NUREG-0578 for postaccident sampling systems. In accordance with the implementation schedule given in NUREG-0578, the staff review of this system will be performed after installation of the sampling system is completed.

II.B.4 Training for Mitigating Core Damage

POSITION

Complete the training of all operating personnel 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.

A. Incore Instrumentation

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

B. 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.

C. 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.

D. 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.

E. 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 (overranged 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.

F. Gas Generation

1. Methods of H_2 generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of non-condensibles.
2. H_2 flammability and explosive limit; sources of O_2 in containment or Reactor Coolant System.

DISCUSSIONS AND CONCLUSIONS

VEPCO is currently conducting a training program that meets all the requirements as stated above. This training program will also become a part of the requalification program for VEPCO personnel.

Attendance is required for all personnel in the Operations Department at North Anna Power Station, Unit 2. This includes licensed operators, licensed senior operators and non-licensed operators. Personnel identified by the Emergency Plan as qualified to become Emergency Directors are required to attend. All Shift Technical Advisors and Nuclear Training Coordinators are also required to attend: an examination will be given to all personnel attending the program, and any person scoring less than 80% will be required to review the material and be reexamined until a grade of 80% is achieved.

In a letter dated July 24, 1980, VEPCO has stated that all Unit 2 licensed operators have received training for degraded core conditions.

Based on the foregoing, we have concluded that the VEPCO training and requalification program meets our requirements for training personnel in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

II.B.7 Analysis of Hydrogen Control (Containment Inerting)

II.B.8 Rulemaking Proceeding on Degraded Core Accidents

POSITION

Reach a decision on the immediate requirements, if any, for hydrogen control in small containments and apply, as appropriate, to new OIs pending completion of the degraded core rulemaking in II.B.8 of the Action Plan.

Issue an advance notice of rulemaking or requirements for design and other features for accidents involving severely damaged cores.

These actions shall be completed before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

The accident at Three Mile Island, Unit 2 resulted in a severely damaged core accompanied by the generation and release to containment of hydrogen in excess of those limits allowed in current regulations. This accident highlighted the difficulties associated with mitigating the consequences of an accident more severe than the current design basis accidents. As a consequence the TMI Action Plan (NUREG-0660), at item II.B.8, calls for a rulemaking proceeding on consideration of degraded or melted cores in safety reviews to solicit comments. Additionally, the TMI Action Plan at item II.B.7 discusses analysis of hydrogen control and the need for inerting small containments.

The staff action on item II.B.7 was completed with issuance of the Commission papers (SECY-80-107, -80-107A and -80-107B) which discussed the technical basis for: 1) the staff position on interim hydrogen control requirements (inerting) for small containments; and 2) continued operation and licensing of nuclear power plants pending the rulemaking proceeding. With regard to North Anna Unit 2, the staff position on sub-atmospheric containments is that inerting is not required as an interim action and that continued operation and licensing of sub-atmospheric containment plants is justified using the current design basis, pending the rulemaking proceeding.

The first steps in the resolution of item II.B.8 will be the issuance of an advance notices of rulemaking and the issuance of an Interim Rule. The advance notices has been drafted and is under staff review. The Interim Rule has also been prepared and is expected to be ready for Commission consideration by July 30, 1980. The Interim Rule, in summary, addresses the following areas:

1. Requires inerting of all BWR Mark I and Mark II containments.
2. Requires all other plants to evaluate the effects of large amounts of hydrogen generation and to propose and assess mitigation techniques for control of hydrogen.
3. Codifies various Lessons Learned items to reduce the likelihood of degraded core accidents.

In addition to the effects related to the rulemaking, the staff has requested that a research program be initiated to investigate the effects of degraded/melted core accidents for generic LWR plant designs, and to investigate various safety systems to reduce the effects of such accidents. Additionally, the staff will seek assistance to evaluate the effectiveness of distributed ignition sources within containment on an expedited basis; i.e., within 3 months. The staff will, however, evaluate a spectrum of mitigation techniques to control hydrogen and reduce the impact of severely degraded core accidents as part of the safety research program discussed above.

Projected Completion Date

We estimate the end date of the rulemaking proceeding to be about 1983. However, the projected end date for all the internal NRC actions identified above is September 1, 1980.

II.E.1.1 Auxiliary Feedwater System Reliability Evaluation

POSITION

- (1) Provide a simplified auxiliary feedwater system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFWS failure following a main feedwater transient, with particular emphasis on potential failures resulting from human errors, common causes single point vulnerability, and test and maintenance outage.
- (2) Provide an evaluation of the AFWS using the acceptance criteria of Standard Review Plan Section 10.4.9.
- (3) Describe the design basis accident and transients and corresponding acceptance criteria for the AFWS.
- (4) Based on the analyses performed modify the AFWS, as necessary.

These requirements shall be met before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

I. Introduction and Background

In Section II.K.3 of Part II of Supplement 10 to the North Anna Power Station, Unit 2 Safety Evaluation Report, we stated that "as part of its generic review of small break LOCA's and feedwater transients in Westinghouse-designed operating plants, the NRC's Bulletins and Orders Task Force (B&OTF) performed a review of the North Anna Unit 1 auxiliary feedwater system. The B&OTF generic review is described in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of Coolant Accidents in Westinghouse-Designed Operating Plants."

"By letter dated September 28, 1979, D. Eisenhut to W. L. Proffitt, the NRC transmitted requirements for the North Anna Unit 1 auxiliary feedwater system resulting from the above mentioned review to VEPCO. VEPCO provided its response to these requirements in its November 2, 1979 letter, C. M. Stillings to Harold R. Denton. Our review of VEPCO's response is currently in progress."

"Since the North Anna Unit 2 auxiliary feedwater system is essentially identical to that at North Anna Unit 1, this evaluation is also applicable to North Anna Unit 2. Completion of the auxiliary feedwater system reliability analysis and appropriate system modifications is classified as a requirement for full power operation for near term operating license applications in Appendix A of the NRC TMI-2 Action Plan (NUREG-0660) and is not necessary for low power testing. Hence, we will report the results of the implementation of the B&OTF auxiliary feedwater system requirements in another supplement to this Safety Evaluation Report prior to full power operation of North Anna Unit 2."

Since the two plants are essentially identical, the results of the reliability study of Unit 1 are also considered applicable to Unit 2 for full power operation. In Appendix III of NUREG-0611 we presented our generic evaluation of AFW systems of Westinghouse designed operating plants. In Appendix X of NUREG-0611, we also addressed our evaluation of operating plant specific AFW system designs.

Our evaluation of the North Anna Power Station Unit 1 AFW design is presented in Appendix X - Item 12 of NUREG-0611. In our evaluation we identified short term and long term recommendation that we required to be implemented in the North Anna Unit 1 AFW system. As a result of our review of AFW systems at Babcock & Wilcox - designed operating plants, we identified additional short-term recommendations to be considered for applicability to the North Anna AFW system design.

Short-term recommendations identified in our evaluation were (1) Recommendation GS-4 which is related to emergency procedures, (2) Recommendation GS-6 which is related to auxiliary flow path availability and (3) Recommendation GS-7 which is related to verification that the automatic start AFW system signals and associated circuitry are safety grade.

As stated above, the additional short-term recommendations identified in our review of AFW systems at Babcock & Wilcox designed operating plants for consideration in the North Anna AFW design were related to (1) primary AFW water source to low level alarm, (2) AFW pump endurance test, (3) indication of AFW flow to steam generators and (4) AFW system availability during periodic surveillance testing.

One long-term recommendation GL-5 related to upgrading AFW system automatic initiation signals and circuits to meet safety-grade requirements was also identified.

The following plant generic recommendations discussed in NUREG-0611 did not apply to North Anna Unit 1; GS-1, GS-2, GS-3, GS-5, GL-1, GL-2, GL-3 and GL-4. The basis for not applying these recommendations can be found in our evaluation of the North Anna Unit 1 AFW system (NUREG-0611 - Appendix X (Item 12)).

Subsequent to the issuance of Supplement No. 10, VEPCO has provided additional information regarding the implementation of our requirements related to the auxiliary feedwater system.

The following paragraphs present the results of our evaluation of the information provided by VEPCO to meet our requirements in accordance with NUREG-0611.

II. Implementation of Our Recommendations Identified in NUREG-0611 - Appendix 10 - Item 12

A. Short Term Recommendations

1. Recommendation GS-4 - Emergency procedures for transferring to alternative sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators,

when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- (1) The case in which the primary water supply is not initially available. The procedures for this case should include any operation actions required to protect the AFW system pumps against self-damage before water flow is initiated.
- (2) The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

In a letter dated November 2, 1979, the licensee stated that procedures for abnormal occurrences will be developed to inform the operator when and in what order the transfer to alternate sources should take place. These procedures will cover cases when the Emergency Condensate Storage Tank (ECST) is not initially available and cases when the ECST water supply is being depleted. Draft emergency procedures for loss of feedwater were furnished in a letter dated May 1, 1980, by VEPCO. These procedures include and discuss feedwater transfer from alternate sources. We conclude these procedures are acceptable and our Office of Inspection and Enforcement will verify that these procedures are in place prior to operation at power level exceeding five percent.

2. Recommendations GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform period testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

Regarding the procedures, VEPCO in a letter dated November 2, 1979, stated that the procedures will be modified to require an additional operator to independently verify that the AFW system valves are properly aligned. Present procedures already require one operator to verify that AFW system valves are properly aligned. Procedures will be modified prior to power levels above five percent and our Office

of Inspection and Enforcement will verify that the procedures were modified in accordance with our requirements prior to operation at power levels above five percent.

Regarding the modifications to the Technical Specifications the licensee has proposed and we have evaluated Specification 4.7.1.2c to the surveillance requirements for the auxiliary feedwater system. Our safety evaluation for this modification on Unit 1 was performed previously, and is included herein for continuity.

By letter dated September 28, 1979, from D. Eisenhut to VEPCO the B&O Task Force recommended certain modifications and procedural changes to the North Anna Unit 1 auxiliary feedwater (AFW) system and its supporting systems. These recommendations resulted from a reliability study of the North Anna AFW system. Among these recommendations was a requirement that following an extended cold shutdown, a flow test should be performed to verify the normal flow path from the primary water source to the steam generators.

As a result, the licensee proposed to add Specification 4.7.1.2c to the surveillance requirements for the auxiliary feedwater system. Proposed Specification 4.7.1.2c will require that prior to entry into Mode 3 (hot standby) from Mode 4 (hot shutdown) following operation in Mode 5 (cold shutdown) each auxiliary feedwater pump will be started and deliver water to its associated steam generator. Since this proposed Specification is in accordance with the B&O Task Force recommendation and will verify that the AFW flow path from the primary AFW water supply to the steam generator(s) is available prior to power operation, we find it acceptable.

The licensee also proposed a specification change to the surveillance requirements for the turbine driven AFW pump. This change is based on plant operating experience at the North Anna Unit 1 plant. The original specification required the turbine driven pump to be tested at a main steam pressure greater than 835 psig. This was to allow entry into Mode 3 without violating a limiting condition for operation; namely, that the turbine driven AFW pump must be operable during all phases of Modes 1, 2 & 3. However, during Mode 1 (power operation) at higher power levels, 835 psig was not available and the pump could not be tested in accordance with the surveillance requirements. The licensee proposed to delete the steam pressure requirement such that the turbine driven AFW pump can be tested during all power levels of Mode 1. The deletion of the minimum steam pressure necessitates an exemption to the action statement for the operability of the steam driven pump such that heatup into Mode 3 can be accomplished without violating the limiting conditions for operation. The surveillance test will now be performed when heating up in Mode 3 as soon as sufficient steam pressure is available to achieve AFW pump flow and discharge pressure specified in the Technical Specification surveillance requirements.

Since these changes result in more flexibility for testing the turbine driven AFW pump including testing in Mode 1 without diminishing safety, (when the pumps would be needed most if called upon) we conclude that the proposed changes are acceptable.

Based on our review as described above, we conclude that the proposed changes are in accordance with the B&O Task Force recommendations and are not less limiting than the Standard Technical Specifications. We, therefore, conclude that proposed Technical Specification change No. 26 to the North Anna Unit 1 Specification is acceptable."

Subsequent to the issuance of our safety evaluation regarding this matter, we determined that a change to the Technical Specification was necessary. This change is as follows: Prior to entry into the Mode 3 following cold shutdown, performance of a flow test of each auxiliary feedwater pump to verify the normal flow path from the condensate storage tank through the pump to its associated steam generator is required.

With respect to North Anna Unit 2, we have required and implemented a comparable modification to the Unit 2 Technical Specifications.

3. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.
 - (1) The design should provide for the automatic initiation of the auxiliary feedwater system flow.
 - (2) The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
 - (3) Testability of the initiation signals and circuits shall be a feature of the design.
 - (4) The initiation signals and circuits should be powered from the emergency buses.
 - (5) Manual capability to initiate the auxiliary feedwater system from the control should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

- (6) The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- (7) The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

In a letter dated November 2, 1979, the licensee stated that the automatic start AFW signals and associated circuitry are safety grade. The AFW system is initiated automatically by a safety injection signal, a loss of offsite power, or a low-low steam generator level. These actuation signals are testable and these signals are the system actuations on which the FSAR Chapter 15 accident analysis is based.

The AFW system is also automatically initiated on loss of the main feedwater pumps in anticipation of low steam generator level. This anticipatory actuation is not testable during normal operation. All initiation signals and circuits are designed to prevent a single failure from causing loss of the AFW system. On this basis, we conclude that this recommendation has been satisfactorily met.

B. Additional Short Term Recommendations

1. Primary AFW Water Source Low Level Alarm - Plants which do not have level indication and alarm for the primary water source may not provide the operator with sufficient information to properly operate the AFW system.

Recommendation - The licensee should provide redundant level indication and low level alarms in the control room for the AFW system primary water supply, to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

VEPCO responded in a letter dated November 2, 1979, that the feedwater system provides the operator with indication of steam generator levels, ECST level, auxiliary feedwater pump suction pressure, auxiliary feedwater pump discharge pressures, and auxiliary feedwater pump flow. All of these indications are powered by an emergency power supply. The control switches for the remote control valves are located in Main Control Room (MCR) in the same general area of the feedwater system indicators. This arrangement allows one operator to easily monitor the system indicators while controlling AFW to the steam generators. The level transmitter for ECST provides indication in the MCR and at

the Auxiliary Shutdown Panel. There is also a ECST level recorder in the MCR. A local indicator is also provided in the AFW pumphouse. These two recorders or indicators receive signals from one transmitter. This does not meet our single failure criterion. The licensee advised us by letter dated May 19, 1980 that redundant level indicators will be provided for both units and will be implemented by January 1, 1981 in accordance with our requirements. On this basis we find the commitment acceptable.

2. AFW Pump Endurance Test - Since it may be necessary to rely on the AFW system to remove decay heat for extended periods of time, it should be demonstrated that the AFW pumps have the capability for continuous operation over an extended time period without failure.

Recommendation - (This recommendation has been revised from the original recommendation in NUREG-0611). The licensee should perform a 48 hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 48 hour pump run, the pumps should be shutdown and cooled down & then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

The license responded in a letter dated February 22, 1980 that the endurance test has not been performed on the AFW pumps. Since then a test report for the Motor driven pumps in North Anna Unit 2 has been received. (VEPCO letter dated July 11, 1980.) We have reviewed the test results and have concluded that they are acceptable. VEPCO has indicated and we concur, that the steam turbine-driven pump, cannot be tested until after unit startup, when steam will be available.

The licensee by letter dated February 26, 1980 had agreed to perform the recommended endurance test for Unit 2 and has tested the motor driven pumps as stated above. VEPCO's letter of July 11, 1980 reaffirms the commitment to test the turbine driven pump when steam is available. Therefore, we find the licensee's commitment acceptable.

3. Indication of AFW Flow to the Steam Generators - Indication of AFW flow to the steam generators is considered important to the manual regulation of AFW flow to maintain the required steam generator water level. This concern is identical to Item 2.1.7.b of NUREG-0578.

Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578 and NUREG-0194. NUREG-0694 states that the requirements by met by January 1, 1981.

- (1) Safety-grade indication of AFW flow to each steam generator should be provided in the control room.
- (2) The AFW flow instrument channels should be prepared from the emergency buses consistent with satisfying the emergency power diversity requirements for the AFW system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

The licensee provided a response to item 1 in a letter dated November 2, 1979, that the AFW system signals and circuits are presently designed to meet safety-grade requirements. The licensee's response to item 2 above is that modifications are presently underway to upgrade the safety-grade indications of AFW flow from semi-vital bus power to vital bus power. VEPCO has indicated that implementations of this modification will be completed by January 1, 1981.

We have not completed our review regarding this matter. However, we intend to complete our review of the design as to whether VEPCO's design meets safety grade requirements in time to allow VEPCO to implement any design modifications by the January 1, 1981 date.

4. AFW System Availability During Periodic Surveillance Testing - Some plants require local manual realignment of valves to conduct periodic pump surveillance tests on one AFW system train. When such plants are in this test mode and there is only one remaining AFW system train available to respond to a demand for initiation of AFW system operation, the AFW system redundancy and ability to withstand a single failure are lost.

Recommendation - Licensees with plants which require local manual realignment of valves to conduct period tests on one AFW system train and which have only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system from the test mode to its operational alignment.

VEPCO responded in a letter dated November 2, 1979, that periodic testing does not require local manual realignment of valves. Also, there are three AFW trains available.

On this basis, we conclude that VEPCO's response to this recommendation is acceptable.

C. Long Term Recommendations

1. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

The licensee responded in a letter November 2, 1979, that the AFW system automatic initiation signals and circuits are presently designed to meet safety-grade requirements. As stated in Section II.E.1.2 (Auxiliary Feedwater Initiation) of Part II of Supplement No. 10 to the Safety Evaluation Report we concluded that the North Anna Unit 2 AFW initiation circuitry meets NUREG-0578 requirements. On this basis we consider this matter resolved.

D. Additional Recommendations

In Enclosure 2 of our letter of September 28, 1979, we requested VEPCO to provide the "Design Basis for Auxiliary Feedwater Flow Requirements." VEPCO provided a response for North Anna, Unit 2 in a letter dated July 10, 1980.

We have evaluated the information provided by VEPCO which addresses the transient events stated in Enclosure 2 of our September 28, 1979 letter. In its submittal VEPCO stated that the flow requirement was 680 gpm to at least two steam generators following a loss of main flow event (including station blackout). The auxiliary feedwater system must have a capacity to supply 340 gpm to one steam generator following a main feedwater line break. This assumes 340 gpm (1 minute after trip) and 680 gpm (after 30 minutes). Each motor driven AFW pump is rated at 350 gpm and the turbine driven pump is rated at 700 gpm. The seismic designed condensate storage tank is normally filled with 110,000 gallons of water and other sources of water are available.

Based on the pump configuration of Unit 2, we conclude that the licensee's design meets its design basis for flow requirements and is therefore acceptable.

E. Conclusions

On the basis of the above considerations we have concluded that the North Anna Unit 2 auxiliary feedwater system meets our full power requirements

as delineated in NUREG-0611-Appendix X Item 12, and therefore, is acceptable.

II.E.3.1 Emergency Power For Pressurizer Heaters

POSITION

Install the capability to supply from emergency power buses a sufficient number of pressurizer heaters and associated controls to establish and maintain natural circulation in hot standby conditions. The requirement shall be met before issuance of a full-power license. (See NUREG-0578, Section 2.1.1, and letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSIONS

The Westinghouse Owner's Group analysis has determined that to maintain natural circulation in a three loop plant with a pressurizer volume of 1400 cubic feet, a heater of 125 kw capacity should be available within one hour. Two backup heater groups rated at 270 kw and their associated controls are energized from redundant emergency buses which are capable of being fed from either offsite power or onsite diesel generators (D-G). The Class IE interfaces for motive and control power are protected by safety grade circuit breakers.

The pressurizer heaters are not automatically loaded on to the bus following the occurrence of a safety injection (SI) actuation signal or loss of offsite power. The continuous rating of the diesel generator indicates that following automatic sequence loading of emergency loads there is insufficient D-G capacity to allow loading of the pressurizer heaters without first load shedding selected loads. Procedures are in force to instruct the operator in load shedding sequences and in use of pressurizer heaters to establish and maintain natural circulation.

The licensee has satisfied the short term Lessons Learned requirements for pressurizer heaters.

II.E.4.2 Containment Isolation Dependability

POSITION

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.
2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to NRC.
3. All non-essential systems shall be automatically isolated by the containment isolation signal.
4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

Clarification:

1. Provide diverse containment isolation signals that satisfy safety-grade requirements.
2. Identify essential and non-essential systems and provide results to NRC.
3. Non-essential systems should be automatically isolated by containment isolation signals.
4. Resetting of containment signals shall not result in the automatic loss of containment isolation.

DISCUSSION AND CONCLUSIONS

The containment isolation system is designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, is provided to assure that no signal active failure will result in the loss of containment integrity.

VEPCO has categorized all systems penetrating containment as being either essential or non-essential. All non-essential systems having automatic containment isolation valves, and not required for an orderly reactor shutdown or to maintain containment atmospheric conditions, are closed by a Phase A containment isolation signal. The operator will have the option of manually resetting the actuation signal and taking

deliberate action to open the isolation valves of certain non-essential systems if post-LOCA conditions warrant their use.

The essential systems are divided into two categories (levels) which are based on their ability to mitigate the severity of various types of accidents. The Level 1 essential systems are defined as the Engineered Safety Features (ESF) and Containment Depressurization systems required to operate after a LOCA. The Level 2 essential systems are defined as those required to maintain the operation of critical systems and functions such as containment heat removal and, therefore, remain unisolated from the containment until a design bases LOCA is indicated (Phase B isolation) or the systems are no longer required. Diversity is not required for the Phase B isolation signal which initiates upon sensing containment high high pressure.

Our review of the containment isolation systems includes verification that there is diversity of parameters sensed for the initiation of containment isolation, as called for by Standard Review Plan Section 6.2.4, "Containment Isolation System." The Phase A containment isolation system design meet this requirement. The parameters sensed for the initiation of containment isolation include high containment pressure, high differential pressure between main steam lines, pressurizer low pressure and high main steam line flow with either low steam line pressure or low low average temperature.

All containment isolation valves (CIVs) in non-essential systems that were originally designed to close upon receipt of an automatic isolation signal meet the Lessons Learned position of diversity. A diverse safety injection signal is provided for these valves, with the exception of the main steam isolation valves (MSIVs). However, diverse parameters are also sensed to initiate MSIV closure.

The North Anna design precludes automatic reopening of containment isolation valves upon reset of the isolation signal. On the basis of our review we have determined that North Anna Unit 2 meets all the requirements of TMI-2 Action Plan Item II.E.4.2. Containment Isolation Dependability. Therefore, we conclude that the isolation dependability of the containment is acceptable.

II.K.3 Final Recommendations of B&O Task Force (Item C.3.3)

POSITION

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in an annual report.

This requirement shall be met before issuance of a full-power license.

DISCUSSION AND CONCLUSIONS

In a letter dated June 30, 1980, VEPCO has indicated that they will prepare a Technical Specification to ensure that all failures or challenges of the PORVs or safety valves are identified, recorded, and promptly reported to the NRC. The Technical Specification also requires annual documentation of all PORV or safety valve challenges.

On the basis that these requirements are specified in the Technical Specifications, we consider this matter resolved.

III. Emergency Preparations and Radiation Protection

III.A.1.1 Upgrade Emergency Preparedness

POSITION

- a. Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified after May 13, 1980 based on public comments) except that only a description of an completion schedule for the means for providing prompt notification to the population (App. 3), the staffing for emergencies in addition to that already required (Table B.1), and an upgraded meteorological program (App. 2) need be provided. NRC will give substantial weight to FEMA findings on offsite plans in judging the adequacy against NUREG-0654.
- b. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations.

DISCUSSION AND CONCLUSIONS

We have reviewed the applicant's revised emergency plan and find that it is in substantial compliance with NUREG-0654 and meets the requirements of 10 CFR 50, Appendix E.

The basis for this finding is summarized in our Emergency Preparedness Evaluation Report and is presented in Appendix B to this report.

The Federal Emergency Management Agency is reviewing the State and local emergency plans. However we have not received their findings and determinations. Until we receive their recommendations we cannot determine whether VEPCO meets the requirements for a full - power license. Prior to authorizing full power operation, we will also require a successful emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations. This text is presently expected to take place in the middle of August 1980.

POSITION

Reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels, measure actual leak rate and establish a program to maintain leakage at as-low-as-practical levels and monitor leak rates.

This requirement shall be met before issuance of a full-power license. See NUREG-0578, Section 2.1.6a and letters of September 27 and November 9, 1979.

DISCUSSION AND CONCLUSIONS

The licensee, prior to issuance for full power license, is required to institute a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious accident or transient to as-low-as-practical levels. Initially, the licensee should measure and report actual leak rates and establish a preventive maintenance program for the monitoring and minimizing the leaks.

The licensee described his program for reducing leakage and reported the results of leak tests on the waste gas system on February 8, 1980. Tests of the liquid systems were reported in a letter of June 9, 1980.

The staff has reviewed the proposed leak reduction monitoring program, together with the initial test results provided, and finds the licensee's program to be acceptable.

III.D.2.4 Offsite Dose Measurements

POSITION

The NRC will place approximately 50 thermoluminescent dosimeters (TLDs) around the site in coordination with the applicant and State environmental monitoring program.

This action shall be completed prior to issuance of a full-power license.

DISCUSSION AND CONCLUSION

Our Office of Inspection and Enforcement has advised us that thermoluminescent dosimeters have been placed around the North Anna site in accordance with IE Manual Chapter 1420.

III.D.3.4 Control Room Habitability

POSITION

Identify and evaluate potential hazards in the vicinity of the site as described in SRP Sections 2.2.1, 2.2.2, and 2.2.3. Confirm that operators in the control room are adequately protected from these hazards and the release of radioactive gases as described in SRP Section 6.4, and if necessary, provide the schedule for modifications to achieve compliance with SRP Section 6.4.

This requirement shall be met by issuance of a full-power license.

DISCUSSION AND CONCLUSION

The North Anna Station Units 1 and 2 have a common control room. In Supplement No. 1 to the North Anna Safety Evaluation Report (NUREG-0053) issued on June 30, 1976, we found, in Section 6.4, that the North Anna control room is adequately protected against the potential consequences from postulated accidents involving airborne radioactivity and against an accident release of chlorine stored on the site. Subsequently, the "NRC Action Plan Developed as a Result of the TMI-2 Accident" (NUREG-0660 published in May 1980) included an item III.D.3.4 which is directly concerned with control room habitability. This was reaffirmed by the Commission as an item requiring resolution prior to a full power license in NUREG-0694, "TMI-Related Requirements for New Operating Licenses" issued in June, 1980. In addition, during the process associated with the review of the application for a low power license, the battery room ventilation system at North Anna was identified as a subject requiring additional staff attention. Our evaluation of these matters is addressed below:

(1) "Control Room Habitability"

In a letter dated May 7, 1980, VEPCO was advised by the NRC of "Five Additional TMI-2 Related Requirements to Operating Reactors", the fifth item of which required a response to Task Action Plan Item III.D.3.4 "Control Room Habitability". In a letter dated June 6, 1980, VEPCO committed to meet the provisions of III.D.3.4 for North Anna.

Additional clarification regarding the implementation of III.D.3.4 was provided in NUREG-0694 "TMI - Related Requirements for New Operating Licenses", which was approved by the Commission on June 16, 1980, as stated in our Position.

We informed VEPCO of the Commission action during a plant inspection on June 26, 1980. In a letter dated July 7, 1980, VEPCO informed us that they have reviewed the control room habitability at North Anna in accordance with Standard Review Plan sections 2.2.1, 2.2.2, 2.2.3 and 6.4, and Regulatory Guides 1.78 and 1.95 and that they "... conclude that the control room meets the specifications and guidance in these SRP sections and Regulatory Guides, and therefore no modifications are required." This action by VEPCO satisfies the requirements of NUREG-0694 for a full power license.

(2) Battery Room Ventilation System

Condition 2.D.(7) of license No. NPF-7 for low power operation of North Anna Unit 2 required that "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." In a letter dated June 10, 1980, VEPCO forwarded an evaluation of the ventilation system which included an analysis of the hydrogen mixing and dilution features of various modes of battery operation. VEPCO determined that it will take approximately fifty days for the control room in the isolation configuration to reach a two percent hydrogen concentration, which is well below the burning concentration of about four percent. VEPCO also evaluated the potential for contamination of the control environment with sulfuric acid vapor resulting from the postulated rupture of a battery case in accordance with NUREG-0570 "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release."

VEPCO concluded that neither hydrogen generated by the batteries nor a loss of battery electrolyte will pose a safety hazard or control room habitability problem.

We have reviewed this evaluation. In addition, on June 26, 1980 we inspected the control room ventilation complex at North Anna Power Station. As a result of their evaluations and inspections, the staff concludes that the battery room ventilation system is acceptable for a full power license.

(3) Conclusion

In summary, as a result of the staff's previous review of the North Anna control room habitability systems, of the staff's plant inspection and of the review of the additional information provided by VEPCO, the staff concludes and reaffirms the previous determination that the control room at the North Anna Station meets the current NRC requirements for a full power license.

IV. Practices and Procedures

IV.F.1 Power Ascension Test

POSITION

The Office of Inspection and Enforcement should increase scrutiny of the power ascension test program to prevent any compromising of safety in view of the proposed expansion of startup test programs and the economic incentives to achieve the already delayed commercial operation of new plants.

This action shall be taken during startup and power ascension test program.

DISCUSSION AND CONCLUSIONS

The licensee's power ascension test program is defined by Section 14.0 and Table 14.1-2 of the Units 1 and 2 Final Safety Analysis Report. Portions of tests on all shifts will be witnessed by the resident inspectors, with assistance by IE Region II inspectors as necessary. We believe that this inspection will assure that VEPCO is not compromising safety during this power ascension testing program.

With respect to TMI-2 dated requirements, we state in NUREG-0694 that "Experience with implementation of the dated requirements on operating reactors is indicating to NRR that the January 1, 1981 deadline may be too tight in some cases to allow reasonable time for completion of the work required. This experience may prove to be the case for some of the dated requirements for NTOLs. The staff would intend to allow case-by-case exceptions to the deadlines if good cause is shown. The dated requirements are not preconditions for licensing of new plants. That is, if a completion deadline falls later than the operating license date for a new plant, then that requirement need not be met by the newly licensed plant until the completion deadline. If in the future a completion deadline falls before an operating license issuance date, then that requirement is a prerequisite for the new operating license, except when a good cause is shown for exception."

Among the factors the staff will consider in its determination of whether good cause has been shown for exceptions are problems associated with the specification, development, procurement, delivery, and installation of components and other factors beyond the control of the applicant.

In a letter dated July 7, 1980, VEPCO submitted a mid-year status of design and installation of Category B (dated requirement items identified in NUREG-0694) modifications and the proposed schedule for implementation of modifications at North Anna. VEPCO indicates that installation dates for some of the Category B items are after the NUREG-0694 specified implementation date of January 1, 1981. VEPCO further indicates that one of the reasons for these delays is that the limited number of vendors who can supply equipment qualified to the requirements of NUREG-0578, "Lessons Learned Task Force Short-Term Recommendations," are attempting to provide timely delivery to the entire industry. However, their limited production capabilities have produced particular material delivery problems for individual utilities. As indicated above, the staff has determined that implementation delays caused by problems associated with procurement and delivery of components can provide a sufficient basis for finding that good cause for delaying implementation has been established. The factors applicable to each of the dated items and our conclusions concerning the acceptability of these factors are addressed in our discussion of each of the dated items.

A meeting was held on July 23, 1980, with VEPCO, Public Service Electric and Gas Company, and the Tennessee Valley Authority to discuss the dated requirements and the bases for any exceptions that would be required to meet the implementation dates specified in NUREG-0694.

If good cause is established on certain items, an exception may be granted. Good cause was defined above as establishing that the applicant has made reasonable effort to complete the dated requirements and could not do so due to circumstances beyond his control such as those discussed above.

We also require that the applicant demonstrate that extending the implementation date will not cause any significant risks to the health and safety of the public.

The following section presents an evaluation of each of the dated requirement items including justification for extending the implementation dates where required.

There are 15 dated requirements that should be met. VEPCO will meet all of these requirements except for five for which good cause has been shown which supports the staff decision to allow an extension to the dates given in NUREG-0694. These are summarized below:

<u>Item</u>	<u>Title</u>	<u>Date (0694)</u>	<u>Date (VEPCO)</u>
II.B.1	Reactor Coolant System Vents	Jan. 1, 1981	12/81
II.B.3	Post-Accident Sampling	Jan. 1, 1981	4/81
II.F.1(c)	Hydrogen Monitor	Jan. 1, 1981	4/81
II.F.1(e)	Noble Gas Effluent Monitor	Jan. 1, 1981	7/81
II.F.2	Reactor Coolant Vessel Water Level	Jan. 1, 1981	12/81

The extension beyond January 1, 1981 for the above is based upon several factors that support good cause; i.e., procurement and installation. Backup capability in the form of either alternate hardware or procedures are available for short-term operations.

I. Operational Safety

I.A.1 Operating Personnel and Staffing

I.A.1.1 Shift Technical Advisor

POSITION

The Shift Technical Advisor shall have a technical education, which is taught at the college level and is equivalent to about 60 semester hours in basic subjects of engineering and science, and specific training in the design, function, arrangement and operation of plant systems and in the expected response of the plant and instruments to normal operation, transients, and accidents including multiple failures of equipment and operator errors. This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.2.1b, and letters of September 27, and November 9, 1979.)

DISCUSSION AND CONCLUSIONS

VEPCO has committed to provide an onshift technical advisor (STA). VEPCO intends to meet this commitment by increasing shift staffing to include an additional licensed Senior Reactor Operator (SRO) or an experienced engineer who is a member of the site Safety Engineering Staff. This additional staffing began on January 1, 1980. A complete description of the STA training program is included in Supplement No. 10 of the North Anna Safety Evaluation Report, Part II, Section I.A.1.1.

Based on our review of the material submitted, we have concluded that Vepco has met this requirement. Qualified STA's will serve on shift to perform an accident assessment role. In addition, they will provide a communication link between the shift and the individual(s) performing the operating experience assessment function. The STA's will undergo annual requalification training.

I.A.2.1 Immediate Upgrading of Operator and Senior Operator Training and Qualification

POSITION

Applicants for SRO license shall have 4 years of responsible power plant experience, of which at least 2 years shall be nuclear power plant experience (including 6 months at the specific plant) and no more than 2 years shall be academic or related technical training.

Certifications that operator license applicants have learned to operate the controls shall be signed by the highest level of corporate management for plant operation. These requirements shall be met on or after May 1, 1980. (See letter of March 28, 1980).

Revise training program to include training in heat transfer, fluid flow, thermodynamics, and plant transients. This requirement shall be met by August 1, 1980. (See letter of March 28, 1980).

DISCUSSION AND CONCLUSION

Each individual that is currently licensed at the North Anna Power Station meets the experience requirements of the staff position. Current applications being submitted are signed by the VEPCO Manager - Nuclear Operations and Maintenance.

VEPCO has submitted revised training programs that included training in areas required by the staff position.

We conclude that VEPCO has satisfied the requirements of this position.

I.A.2.3 Administration of Training Programs for Licensed Operators

POSITION

Training instructors who teach systems, integrated responses, transient and simulator courses shall successfully complete an SRO examination and instructors shall attend appropriate retraining programs that address, as a minimum, current operating history, problems and changes to procedures and administrative limitations. In the event an instructor is a licensed SRO, his retraining shall be the SRO requalification program. Applications for SRO licenses shall be submitted by August 1, 1980 and retraining programs shall be initiated by May 1, 1980. (See letter of March 28, 1980)

DISCUSSION AND CONCLUSION

There is currently one licensed SRO on the training staff at North Anna. Three instructors have applied for senior licenses and will be examined during July 1980. All licensed personnel and nuclear training coordinators at the North Anna plant are required to participate in the requalification program. Based on the foregoing, we have concluded that VEPCO has complied with the requirements of this position.

I.A.3.1 Revise Scope and Criteria for Licensing Exams

POSITION

Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations.

Contents of the licensed operator requalification program shall be modified to include instruction in heat transfer fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

These requirements shall be met by May 1, 1980. (See letter of March 28, 1980).

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

This requirement shall apply to all annual requalification examinations conducted after March 28, 1980. (See March 28, 1980 letter.)

Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operation shall be walked through and evaluated by a member of the training staff. An appropriate simulator may be used to satisfy the requirements for control manipulations.

This requirement shall be met by August 1, 1980. (See March 28, 1980 letter.)

DISCUSSION AND CONCLUSION

Current applications for licenses from North Anna personnel include consent for informing facility management of the examination results. In letters dated June 27, 1980, VEPCO submitted their outline of the training in heat transfer, fluid flow, thermodynamics and mitigation of accidents for their initial training and requalification program. Also included is the revised examination criteria for accelerated training consistent with the new passing grade for issuance of licenses.

The details of the modifications to the requalification program covering training in specific reactivity control manipulation for startup, normal, abnormal and emergency operations were submitted by VEPCO on July 15, 1980. We have reviewed the information provided by VEPCO and have determined that it meets the requirements identified in our letter of March 28, 1980, which addressed qualifications of Reactor Operators.

Based on the information submitted by VEPCO, we conclude that they have satisfied all the requirements of this position and consider this matter resolved.

I.C.1 Short-Term Accident Analysis and Procedure Revision

Analyze the design basis transients and accidents including single active failures and considering additional equipment failures and operator errors to identify appropriate and inappropriate operator actions. Based on these analyses, revise, as necessary, emergency procedures and training.

This requirement was intended to be completed in early 1980; however, some difficulty in completing this requirement has been experienced. Clarification of the scope and revision of the schedule are being developed and will be issued by July 1980. It is expected that this requirement will be coupled with Task I.C.9., Long-term Upgrading of Procedures. See NUREG-0578, Section 2.1.3b and 2.1.9, and letters of September 27 and November 9, 1979.

DISCUSSION AND CONCLUSIONS

Our evaluation of this matter is addressed in Section 22.2, Item I.C.1, of this supplement.

II. Siting and Design

II.B.1 Reactor Coolant System Vents

POSITION

Install reactor coolant system and reactor vessel head high-point vents that are remotely operable from the control room.

This requirement shall be met before January 1, 1981. (See Enclosure 4 to letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSION

The staff's review of VEPCO's response to this position is included in the full power requirement, reactor coolant system vents, Section 22.2, Item II.B.1 of this supplement.

Projected Completion Date

On the basis given in Section 22.2, Item II.B.1 of this supplement, the NRC staff has concluded that the delay of installation beyond the January 1, 1981 deadline is acceptable.

II.B.2 Plant Shielding

POSITION

Complete modification to assure adequate access to vital areas and protection of safety equipment following an accident resulting in a degraded core.

This requirement shall be met by January 1, 1981. (See NUREG-0578, Section 2.1.6b and letters of September 27, and November 9, 1979.)

DISCUSSION AND CONCLUSIONS

Our evaluation of the radiation and plant shielding report which is required prior to full-power operation is presented in Section 22.2, Item II.B.2 of this report.

In a letter dated April 1, 1980, Vepco stated that certain areas, including the technical support center and the counting room had to be shielded and that the security boundary would be moved as necessary after an accident. In a letter dated July 7, 1980, Vepco specified that "essential areas would be shielded without further identifying 'essential.'" They also identified the technical support center, counting room, and security center as areas requiring continuous occupancy. Therefore, we require that the technical support center, counting room, and security control center be designated as "essential areas" and that they be shielded to allow the continuous occupancy that Vepco stated was necessary.

PROJECTED COMPLETION DATE

Vepco has committed to make the necessary modifications by January 1, 1981, but has stated that completion of the work is dependent on delivery of valves and temporary relief from technical specifications. In a meeting on July 23, 1980, (letter dated July 25, 1980) VEPCO stated that in order to meet the January 1, 1981 date, they would need relief from technical specifications that require two operable hydrogen recombiners so that they can take one recombiner at a time out of service for modification. VEPCO stated that they should be able to complete modifications by January 1, 1981 and would, therefore, meet the NUREG-0694 requirement. Based on a preliminary assesement of the relief required, the staff believes that such relief is warranted. Therefore, the staff finds that this item has been acceptably resolved.

POSITION

Complete corrective actions needed to provide the capability to promptly obtain and perform radioisotopic and chemical analysis of reactor coolant and containment atmosphere samples under degraded-core conditions without excessive exposure. This requirement shall be met by January 1, 1981.

DISCUSSION AND CONCLUSION

In a letter dated July 7, 1980, VEPCO provided the staff with the preliminary design of a post-accident sampling system which conforms to the design criteria established by the NRC staff.

VEPCO's installation schedule for North Anna, Unit No. 2, anticipates component delivery in December 1980, with installation projected for the first refueling outage in late 1981. Completion of system installation is being delayed by extended procurement delivery date for special system isolation valves. Otherwise, installation could have been complete in December 1980. Until the improved system can be installed, the licensee will continue to use the interim procedure discussed in Section 22.2, II.B.3 for sampling and analysis.

The staff has reviewed VEPCO's submittals on this item. VEPCO has committed to procure and install equipment and to implement the relevant procedures for operation of the equipment necessary to comply with the staff's criteria, as set forth in NUREG-0578, in the letter of November 9, 1979, and in NUREG-0694. The staff finds the described equipment and procedures to be in compliance with these criteria. The staff further finds that the dates scheduled by VEPCO for completion of actions show reasonable effort and intent on the part of VEPCO to comply with the staff's projected completion dates and are therefore acceptable, provided that implementation proceeds at the first scheduled or forced plant outage of sufficient duration for installation.

PROJECTED COMPLETION DATE

VEPCO projects late 1981 as the date for installation of the equipment items necessary for safe operation of the improved post-accident sampling system. The staff requires that implementation should be completed at the first scheduled or forced plant outage of sufficient duration for installation.

II.D.1 Relief and Safety Valve Test Requirements

POSITION

Complete tests to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

This requirement shall be met by July 1, 1981. (See NUREG-0578), Section 2.1.2, and letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSIONS

VEPCO has stated that they are actively pursuing a joint effort with other members of the utility industry which will develop requirements for a generic test facility and program for RCS relief and safety valve prototypical testing. This involves subscription to and participation in a program developed and managed by the Electric Power Research Institute (EPRI). As stated in Supplement No. 10 to the North Anna Safety Evaluation Report, we believe that there is adequate assurance at this point that the NUREG-0578 requirement regarding performance verification of RCS relief and safety valves will be met satisfactorily for the North Anna 2 unit. We conclude that, pending satisfactory results from the ongoing test program, this requirement places no restrictions on North Anna 2 operation through full power.

II.E.1.2 Auxiliary Feedwater Initiation and Indication

(a) Initiation

POSITION

Upgrade as necessary, automatic initiation of the auxiliary feedwater system to safety-grade quality.

This requirement shall be met by January 1, 1981.

DISCUSSION AND CONCLUSIONS

The staff's review of VEPCO's response to the auxiliary feedwater initiation requirement is included in the full power requirement II.E.1.1, Auxiliary Feedwater System Reliability Evaluation, Section 22.2, Item II.E.1.1 Paragraph II A-3 of this supplement.

(b) Indication

POSITION

Upgrade, as necessary, the indication of auxiliary feedwater flow to each steam generator to safety grade quality.

DISCUSSION AND CONCLUSIONS

The staff's review of VEPCO's response to the indication of auxiliary feedwater flow to each steam generator to safety grade quality is included in the full power requirement II.E.1.1, Auxiliary Feedwater System Reliability Evaluation, Section 22.2, Item II.E.1.1 Paragraph II B-3 of this supplement.

II.E.4.1 Containment Dedicated Penetrations

POSITION

Install a containment isolation system for external recombiners or purge systems for post-accident combustible gas control, if used, that is dedicated to that service only and meets the single-failure criterion. This requirement shall be met before January 1, 1981.

DISCUSSION AND CONSLUSION

The North Anna design uses redundant external hydrogen recombiners shared between Units 1 and 2. The hydrogen recombiner line takes suction from the same penetration used for the suction of the containment vacuum pumps, the hydrogen purge lines and the hydrogen analyzer. Each of these lines has the suction intake downstream of two containment isolation valves located outside of containment. Since radioactive gases could be flowing through these pipes during the post-accident mode, these systems become extensions of containment. Therefore, we have required that adequate provisions be installed for containment isolation.

VEPCO has committed to install redundant, remote-manual actuated valves in series to isolate the containment vacuum pumps from the combustible gas control system. This provides a single failure proof design to isolate the containment vacuum pumps, thus dedicating the penetration to the combustible gas control system.

The backup hydrogen purge system is presently isolated from the hydrogen analyzers and recombiners by an administratively locked closed valve. This system is not operated during normal plant operations. Its use would only be contemplated post-accident, and then only if there is need to use the containment hydrogen recombiners and both of them fail. VEPCO is currently examining the radiological consequences of personnel manually opening this valve with a substantial radiation source in the containment building. If the analysis shows that personnel exposure is too high, VEPCO intends to install remote manual operators to actuate this valve. If remote manual actuation is considered necessary, the staff will require redundant valves in series receiving redundant power supplies so that a spurious electrical signal could not open the recombiner system to the plant's vent stack.

VEPCO has committed to convert the manual valves in the hydrogen recombiner piping to remote manual actuation. This is in response to evaluating the personnel exposures that might occur if these valves required manual opening.

The licensee is also evaluating the radiological consequences of personnel opening the administratively locked closed valves of the hydrogen analyzers. This evaluation may conclude that these valves should be administratively locked open. Since these valves and the hydrogen analyzer piping constitutes a closed system outside of containment, the staff will not object to opening these valves.

The discharge line from the hydrogen recombiner shares the same penetration with the discharge line from the hydrogen analyzer. Containment isolation is provided by a check valve inside containment and a remote manual valve outside containment. The combined hydrogen recombiner suction and discharge line is sized such that the flow requirements for the use of the combustible gas control system are satisfied. VEPCO has committed to complete all plant modifications by January 1, 1981.

VEPCO has committed to comply with the staff's position on containment dedicated penetrations. The conceptual design and target implementation schedule satisfy our requirements for this item. Therefore, we conclude that the applicant's response to date concerning this item is acceptable and that it is in compliance with this portion of NUREG-0694.

PROJECTED COMPLETION DATE

VEPCO has committed to meet all plant modifications by January 1, 1981, provided unforeseen delays in equipment deliveries are not excessive.

II.F.1 Additional Accident Monitoring Instrumentation

POSITION

Install continuous indication in the control room of the following parameters:

- a. Containment pressure from minus 5 psig to three times the design pressure of concrete containments and four times the design pressure of steel containments;
- b. Containment water level in PWRs from (1) the bottom to the top of the containment sump, and (2) the bottom of the containment to a level equivalent to 600,000 gallons of water;
- c. Containment atmosphere hydrogen concentration from 0 to 10 volume percent;
- d. Containment radiation up to 108 Rad/hr;
- e. Noble gas effluent from each apotential release point from normal concentrations to 105 mCi/cc (Xe-133).

Provide capability to continuously sample and perform onsite analysis of the radio-nuclide and particulate effluent samples.

This instrumentation shall meet the qualification, redundancy, testability, and other design requirements of the proposed revision to Regulatory Guide 1.97.

This requirement shall be met by January 1, 1981.

DISCUSSION AND CONCLUSION

In order to monitor the containment pressure and meet the NUREG-0578 and Regulatory Guide 1.97 requirements, two separate transmitters, indicators and a two-pen recorder will be installed. The system will be capable of measuring containment pressure from 0 to 180 psia. The transmitters will be located outside of the containment and tap into two existing pressure sensing lines. The transmitter to be installed at North Anna will be qualified for 2.2×10^8 rads total dose. The indicators and recorder will be located in the control room. Material delivery is scheduled for December 1980. The conceptual design and target implementation schedule satisfy our requirements for this item. Therefore, we conclude that VEPCO's response to date concerning this item is acceptable and that it is in compliance with this portion of NUREG-0694.

VEPCO has committed to provide narrow range instruments and wide range instruments to measure containment water level. The narrow range instruments cover the range from the bottom to the top of the containment sump while the wide range instruments cover from the bottom of containment to the elevation equivalent to a 600,000 gallon capacity. The wide range instruments satisfy the provisions of Regulatory Guide 1.97 and the narrow range instruments meet the provisions of Regulatory Guide 1.89. The containment water level indication meets the staff's requirements and is therefore acceptable.

In a telephone conversation on July 23, 1980, VEPCO verified that they previously committed to install a hydrogen indicator capable of measuring hydrogen concentrations between 0 and 10%. The hydrogen indicators will be installed by March 1981, VEPCO intends to use containment gas samples to monitor the hydrogen concentrations in the interim two month period. In a letter dated July 25, 1980, VEPCO committed to connect a gas chromatograph to a containment sampling line if required. This system will be capable of measuring hydrogen concentrations of 0 to 10%. Installation of the system is scheduled to be completed by January 1, 1981. We conclude that monitoring hydrogen in containment by analyzing gas samples is acceptable until March 1, 1981, when hydrogen indicators capable of measuring hydrogen concentrations up to 10% will be installed.

VEPCO has stated that containment radiation monitors that meet the criteria specified in our letter of October 30, 1979, which specifies a maximum range of 10^7 R/hr as an acceptable alternative to 10^8 rads/hr, are to be shipped from Victoreen in August and November 1980 and the control panel is to be shipped in December 1980. In a letter dated July 25, 1980, VEPCO stated that a class IE recorder has been located onsite and that the monitors would be installed by January 1, 1981.

In a letter dated April 1, 1980, VEPCO specified criteria for interim monitoring equipment for measurement of radioactivity in releases from steam safety valves and atmospheric steam dump valves. The licensee is studying means to improve the low range sensitivity of these monitors and to account for the low energy spectrum after an accident. Until such a system can be developed, the existing monitors will be used for releases from steam safety valves and atmospheric steam dump valves, together with sampling and analysis of secondary system fluids and offsite radiation monitoring. The staff considers that, pending possible eventual development of alternative monitoring systems, that the equipment proposed for interim monitoring meets the intent of our criteria for installation of permanent equipment to meet the January 1, 1981 date and is satisfactory.

In a letter dated July 7, 1980, VEPCO stated that a noble gas effluent monitor will be installed on the process vent and ventilation vents at North Anna Unit 2. The stated range of the monitor meets the staff criteria established in NUREG-0578. The letter further states that radioiodine and particulate releases will be determined by filtration and adsorption samples collected using the noble gas monitor as a collection device; analyses for radioiodine and particulates will be performed at an onsite location. VEPCO has stated that, on vendor proposals, a noble gas effluent monitoring system meeting our stated requirements cannot be delivered until mid-1981. At that time, the equipment will be installed. It is noted that a reactor outage is not required for installation of the proposed system. Until the equipment can be installed and placed in operation, VEPCO will continue to use the interim procedures and equipment provided to meet the interim monitoring criteria specified in the clarification letter of November 9, 1979, and reviewed in SER Supplement No. 10. VEPCO has committed to procure and install equipment and to implement the relevant procedures for operation of the equipment necessary to comply with the staff's criteria, as set forth in NUREG-0578, in the letter of November 9, 1979, and in NUREG-0694. The staff finds

the described equipment and procedures to be in compliance with these criteria. The staff further finds that the dates scheduled by VEPCO for completion of actions show reasonable effort and intent on the part of VEPCO to comply with the staff's projected completion dates and are therefore acceptable.

PROJECTED COMPLETION DATE

The hydrogen indicators at the North Anna, Unit 2 plant will be installed by April 1, 1981. The hydrogen sampling system to be used in the interim will be installed by January 1, 1981.

The licensee projects July 1981 as the date for installation of the noble gas effluent monitoring equipment. Since radioiodine and particulate samples will also be provided by the noble gas monitor, the projected end date will also be mid-1981 for this item. Until that time, VEPCO will continue to use interim monitoring procedures and equipment.

All other accident monitoring instrumentation will meet the required installation date of January 1, 1981, provided unforeseen delays in equipment deliveries are not excessive.

II.F.2 Inadequate Core Cooling Instruments

POSITION

Install, if required, additional instruments or controls needed to supplement installed equipment in order to provide unambiguous, easy-to-interpret indication of inadequate core cooling.

This requirement shall be met by January 1, 1981. (See NUREG-0578), Section 2.1.3b, and letters of September 27 and November 9, 1979.)

DISCUSSION AND CONCLUSION

In NUREG-0053, SER Supplement No. 10, Part II, Item II.F.2, the staff concluded that VEPCO had accomplished the necessary actions and commitments such that Item II.F.2 placed no restrictions on full power operation for North Anna 2. This conclusion was based in part on a commitment by the applicant to complete installation of a level measurement system for detection of inadequate core cooling conditions prior to January 1, 1981.

VEPCO, in the submittal "Lessons Learned Short-Term Requirements Surry Units 1 and 2 and North Anna Units 1 and 2," dated July 7, 1980, now indicates that installation cannot be completed on schedule due to the need for further development of the ΔP vessel level measurement system and due to material delivery problems. The staff has been monitoring the progress of other applicants and licensees in meeting the schedule requirements of II.F.2, and has had meetings with suppliers of various level measurement systems to review the design and development progress and the equipment procurement that the applicant is making a good faith effort to install a system as early as feasible. Therefore, we find the North Anna Unit 2 compliance with Item II.F.2 to be acceptable for full power operation. However, we will require that the procedure guidelines for use of the proposed equipment, the analysis used in developing these procedures, an updated schedule giving the development and procurement status, and any available test data be submitted for staff review by January 1, 1981, consistent with scheduled or forced plant outages, and that in service testing, calibration, and implementation proceed on a schedule acceptable to the staff.

In the interim, as stated in Section 22.2, Item I.C.1 of this supplement, VEPCO has procedures related to the recovery from inadequate core cooling which we find acceptable. These procedures include the use of control room instrumentation (reactor coolant system pressure, reactor coolant system temperature ex-core thermocouples, and pressurizer level) to provide the necessary information to aid the operator in his evaluation of events.

III. Emergency Preparations and Radiation Protection

III.A.1.2 Upgrade Emergency Support Facilities

POSITION

Provide radiation monitoring and ventilation systems, including particulate and charcoal filters, and otherwise increase the radiation protection to the onsite technical support center to assure that personnel in the center will not receive doses in excess of 5 rem to the whole body or 30 rem to the thyroid for the duration of the accident. Provide direct display of plant safety system parameters and call up display of radiological parameters.

For the near-site emergency operations facility, provide shielding against direct radiation, ventilation isolation capability, dedicated communications with the onsite technical support center and direct display of radiological and meteorological parameters.

This requirement shall be met by January 1, 1981, although the safety parameter information requirements will be staged over a longer period of time. (See NUREG-0578, Section 2.2.2b and 2.2.2c and letters of September 27, and November 9, 1979 and April 25, 1980.

DISCUSSION AND CONCLUSIONS

The requirements stated above have been revised. This revision has been approved by the Commission. The licensee will be required to meet the requirements of NUREG-0696, "Functional Criteria for Emergency Response Facilities" to be published for comment in July or August 1980. NUREG-0696 provides the details needed to design and implement a Technical Support Center (TSC) and Emergency Operations Facility (EOF). A revised schedule for implementation of a total requirements package is also under development.

The Emergency Preparedness Evaluation Report (Appendix B to this Supplement) describes the Technical Support Center, Operations Support Center, and Emergency Operations Facility established on an interim basis. Therefore, we conclude as a result of our review that these facilities are adequate for full power operation.

III.D.3.3 Inplant Radiation Monitoring

POSITION

Provide the equipment, training, and procedures to accurately measure the radioiodine concentration in areas within the plant where plant personnel may be present during an accident.

This requirement shall be met before January 1, 1981. See NUREG-0578, Section 2.1.8.C and letters of September 27 and November 9, 1979.

DISCUSSION AND CONCLUSION

VEPCO has a portable monitoring system available which have the capability of accurately monitoring iodine in the presence of noble gases via the use of silver zeolite sampling cartridges and the multi-channel Analyzer (or Single-Channel Analyzer). To ensure timely analysis of the cartridges in an emergency, a dedicated single channel analyzer has been purchased for use in air monitoring. The required procedures are in effect. The current analysis system (MCA) is located in a concrete walled room for shielding purposes. Also the Ge (Li) detectors are set in thick steel cabinet to reduce background. Should these existing facilities not be available, samples will be transported to VEPCO's Surry Station for analysis.

The requirement and procedures described by VEPCO meet our position in NUREG-0578 and NUREG-0660 and are, therefore, acceptable.

23.0 CONCLUSIONS

23.0 Based on our evaluation of the application as set forth in our Safety Evaluation Report issued on June 4, 1976 and Supplement Nos. 1 through 10 and our evaluation as set forth in this supplement, we conclude, that subject to resolving matters related to emergency preparedness as discussed in Section 22.2 Item III.A.1.1, the operating license can be issued to allow power operations at full rated power (2775 megawatts thermal) subject to license conditions which will require further Commission approval and license amendments before the stated condition can be removed.

We conclude that the construction of the facility has been completed in accordance with the requirements of Section 50.57(a)(1) of 10 CFR Part 50, and that construction of the facility has been monitored in accordance with the inspection program of the Commission's staff.

Subsequent to the issuance of the operating license for full rated power for the North Anna Power Station, Unit 2, the facility may then be operated only in accordance with the Commission's regulations and the conditions of the operating license under the continuing surveillance of the Commission's staff.

We conclude that the activities authorized by the license can be conducted without endangering the health and safety of the public, and we reaffirm our conclusions as stated in our Safety Evaluation Report and its supplements.

APPENDIX A

Continuation of Chronology of Radiological Review

March 7, 1980	Letter to applicant requesting additional information on Residual Heat Removal System.
March 10, 1980	Letter to applicant concerning an interim upgrade of NRC emergency planning regulations.
March 10, 1980	Letter to applicant concerning followup actions resulting from our reviews regarding the TMI accident.
March 11, 1980	Letter to applicant concerning a Change of Submittal for Evacuation Time Estimates.
March 11, 1980	Letter to applicant concerning Independent Field Audit of Installed Electric Systems and Equipment.
March 12, 1980	Letter from applicant concerning Training Criteria for Mitigating Core Damage.
March 13, 1980	Letter to applicant concerning Baseline Hydraulic Data.
March 13, 1980	Representatives from VEPCO & NRC met at Louisa, Virginia to discuss matters related to the Unit 2 Steam Generators.
March 13, 1980	Letter to applicant concerning Potential Design Deficiencies in Bypass, Override, and Reset Circuits of Engineered Safety Features.
March 14, 1980	Letter from applicant concerning Operations Staffing.
March 17, 1980	Letter from applicant concerning North Anna Unit 2 Reactor Vessel Nozzle Cladding.
March 17, 1980	Letter from applicant concerning habitability of the control room following a postulated release of hazardous chemicals.
March 17, 1980	Letter from applicant transmitting procedures for emergency conditions.
March 19, 1980	Letter from applicant concerning a special test program for North Anna, Unit 2.
March 20, 1980	Letter from applicant transmitting responses to NRC letter of March 7, 1980 requesting additional information pertaining to achieving cold shutdown using safety grade equipment.
March 20, 1980	Representatives from NRC & VEPCO met in Bethesda, Maryland to discuss matters related to the operation of North Anna, Unit 2. (Summary issued 3/26/80).
March 26, 1980	Letter from applicant concerning Safety Engineering and Operating Experience Evaluation.
March 26, 1980	Letter from applicant concerning Shift Technical Advisor Training.

March 28, 1980	Representatives from VEPCO and NRC met in Bethesda, Maryland to discuss matters related to the North Anna Unit 2 Control Room Design. (Summary issued April 11, 1980.)
March 28, 1980	Letter from applicant transmitting procedures for pressurizer Safety Valve or PORV open and Loss of Reactor Coolant System Pressure.
April 2, 1980	Letter from applicant transmitting procedures for the special test program for North Anna 2.
April 3, 1980	Letter from applicant concerning North Anna Unit 2 Technical Specifications Surveillance Requirement 4.4.9.3.1.d.
April 3, 1980	Letter from applicant concerning procedural guidance for the shift supervisor.
April 7, 1980	Letter from applicant concerning Reactor Vessel Nozzle Inspection Report.
April 10, 1980	Letter to applicant requesting additional information (4.0 Reactor).
April 11, 1980	Letter to applicant issuing fuel loading and low power testing license for North Anna Unit 2. License NPF-7, Federal Register Notice, Amendment 7 to Indemnity Agreement B-80 attached.
April 11, 1980	Letter to applicant concerning Completion of Items 7.1(1) and 7.1(2) as Delineated in Appendix A to N. A. Unit 2 Technical Specifications - License - NPF-7.
April 15, 1980	Letter to applicant concerning Emergency Plan for North Anna.
April 15, 1980	Letter from applicant concerning Change in Review Procedures for Equipment Qualification Documentation for North Anna, Unit 2.
April 15, 1980	Letter from applicant concerning Potential Design Deficiencies in Bypass, Override and Reset Circuits of Engineered Safety Features.
April 18, 1980	Letter from applicant concerning a test program for the pressurizer power operated relief valves.
April 21, 1980	Letter to applicant concerning Masonry walls.
April 21, 1980	Letter from applicant concerning North Anna 2 special test program.
April 24, 1980	Letter to applicant concerning a page change to Appendix A Technical Specifications issued with NPF-7.
April 24, 1980	Letter to applicant transmitting copies of Supplement No. 10 to the Safety Evaluation Report for North Anna.
April 25, 1980	Letter to applicant concerning the completion of Items 7.2(1) and 7.2(2) of Appendix A.
April 25, 1980	Letter to applicant concerning Clarification of NRC Requirements for Emergency Response Facilities at Each Site.
April 29, 1980	Letter from applicant concerning the development of the revised Emergency Plan.

April 29, 1980	Westinghouse letter on North Anna Docket transmitting proprietary and non-proprietary versions of information previously submitted to support the Westinghouse position on guide tube wear characteristics.
May 5, 1980	Letter to applicant requesting additional information - Floodplain Management.
May 7, 1980	Letter to applicant concerning five additional TMI-2 related requirements to Operating Reactors.
May 9, 1980	Letter from applicant concerning test procedure 1-PT-112.
May 9, 1980	Letter from applicant concerning Inservice Testing Program.
May 9, 1980	Letter from applicant concerning a procedure change to the periodic test to collect a sample from the drains under the service water pump house.
May 16, 1980	Letter from applicant proposing an amendment to NPF-4 & NPF-7 which would clarify the meaning of the OPERABLE.
May 19, 1980	Letter to applicant concerning NUREG-0577, Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Collant Pump Supports."
May 23, 1980	Letter from applicant concerning Floodplain Management.
May 27, 1980	Letter from applicant concerning Inservice Testing Program.
May 28, 1980	Letter from applicant advising they will respond to DPM's request for a Guard Training and Qualification Plan by July 14, 1980.
May 28, 1980	Representatives from NRC & VEPCO meet in Bethesda, Maryland to discuss matters related to environmental qualifications of Class IE electrical equipment and instrumentation. (Summary issued June 2, 1980.)
May 30, 1980	Representatives from VEPCO & NRC meet in Bethesda, Maryland to discuss matters related to the North Anna Power Station Emergency Plan. (Summary issued June 2, 1980.)
May 30, 1980	Letter from applicant transmitting Emergency Procedures for North Anna 2, revised to reflect recent analyses and NSSS vendor recommendations.
May 31, 1980	Letter to applicant requesting additional information on Containment Sump.
May 31, 1980	Letter to applicant requesting additional information - NUREG-0578 - Item 2.1.8.b.
June 5, 1980	Letter to Westinghouse on North Anna Docket withholding from public disclosure AW-80-22 - Westinghouse supplemental information concerning guide tube wear.
June 5, 1980	Letter from applicant concerning Special Low Power Tests.
June 5, 1980	Representatives from VEPCO and NRC visit the North Anna Unit 2 site to discuss matters related to the use of masonry walls. (Summary issued 6/24/80.)

June 9, 1980	Letter from applicant concerning Bulletin and Orders Task Force Final Recommendations.
June 9, 1980	Letter from applicant submitting the Monthly Operating Report for the month of May 1980.
June 9, 1980	Letter from applicant concerning Lessons Learned Short Term Requirement 2.1.6.a., North Anna 2.
June 10, 1980	Letter from applicant concerning an evaluation of the battery room ventilation system which exhausts into one end of the control room.
June 10, 1980	Letter from applicant concerning lamp test circuits.
June 12, 1980	Letter from applicant concerning Full Power License Issuance.
June 12, 1980	Letter to applicant concerning Resolution of Items 7.3(1) through 7.3(24) as Delineated in Appendix A of the North Anna Unit 2 Technical Specifications and License Conditions D.5(k) and D.5(o) and D.5(v) - Operating License NPF-7.
June 13, 1980	Letter from applicant concerning Special Low Power Tests.
June 16, 1980	Letter from applicant concerning Additional Information Required - The Review of the North Anna Power Station Unit 2 FSAR.
June 17, 1980	Letter from applicant concerning lamp test circuits.
June 18, 1980	Letter to applicant requesting additional information - Branch Technical Position RSB3-1.
June 20, 1980	Letter from applicant concerning electrical equipment qualification review, NUREG-0588.
June 23, 1980	Letter from applicant concerning response to request for additional information for the Final Safety Analysis.
June 24, 1980	Letter from applicant concerning Auxiliary Feedwater System.
June 24, 1980	Letter from applicant concerning Low Pressure Turbine Disc Inspection Report.
June 24, 1980	Letter from applicant concerning procedures for conducting the special low power test program at North Anna 2.
June 25, 1980	Letter to applicant requesting additional information on Quality Assurance.
June 27, 1980	Letter from applicant requesting that Table 3.7-4 to Technical Specifications be revised.
June 30, 1980	Letter from applicant concerning NUREG-0660 Full Power and 1980 Dated Requirements, Unit 2.
June 30, 1980	Letter to applicant advising of regional meetings for applicants and vendors.

July 1, 1980	Letter from applicant concerning Supplement No. 10 to the SER issued by NRC.
July 2, 1980	Letter to applicant requesting additional information related to snubbers.
July 2, 1980	Letter from applicant concerning Quality Assurance.
July 3, 1980	Letter to applicant issuing Amendment No. 1 to NPF-7 for low power test program.
July 7, 1980	Representatives from NRC & VEPCO meet in Louisa, Virginia to discuss North Anna Unit 2 operating procedures.
July 7, 1980	Letter from applicant concerning review of control room habitability.
July 7, 1980	Letter from applicant concerning Lessons Learned Short Term Requirements for Surry and North Anna.
July 8, 1980	Letter from applicant concerning Power Ascension Procedures.
July 9, 1980	Letter from applicant concerning Water Hammer Demonstration Test Safety Analysis and Technical Specifications Exemption Requests - North Anna 2.
July 10, 1980	Letter from applicant pertaining to achieving cold shutdown using safety grade equipment.
July 10, 1980	Letter from applicant concerning Additional TMI - Related Dated Requirements.
July 10, 1980	Letter from applicant concerning Auxiliary Feedwater System Requirements.
July 10, 1980	Letter from applicant concerning Reactor Coolant System Vents.
July 11, 1980	Letter from applicant proposing Technical Specifications Change and Amendment to Operating License NPF-7.
July 11, 1980	Letter from applicant concerning emergency planning.
July 11, 1980	Letter from applicant concerning Auxiliary Feedwater Pump Test Results.
July 14, 1980	Letter from applicant responding to NUREG-0660.
July 14, 1980	Letter from applicant concerning Residual Heat Removal.

APPENDIX B

EMERGENCY PREPAREDNESS EVALUATION REPORT

INTRODUCTION

The Virginia Electric and Power Company (hereinafter referred to as the Licensee, The Company, VEPCO) filed with the Nuclear Regulatory Commission a revision to the North Anna Power Station Emergency Plan dated May 1, 1980, as amended (hereinafter referred to as the Plan). The Commission's staff conducted a review of this Plan. The staff's review also included a site visit to the facility and a public meeting during the week of October 15, 1979.

The Plan was reviewed against the criteria of the sixteen operator Planning Objectives in Part II of the "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," (For Interim Use and Comment) NUREG-0654.

As a result of public comments, staff comments and development of the final rule on emergency planning, NUREG-0654 will be revised. The Plan will be reviewed against the revised criteria and a supplement to this report will provide our review results and conclusions.

This Emergency Preparedness Evaluation Report lists each objective in order followed by a summary of applicable portions of the Emergency Plan as they apply principally to the operator Planning Objectives. The final section of this report provides our review results and conclusions.

At a later date an appendix will be added to this report describing the findings and determinations of the Federal Emergency Management Agency on the State and local emergency response plans.

EVALUATION

A. Assignment of Responsibility (Organization Control)

Planning Objective

To assure that primary responsibilities for emergency response in nuclear facility operator, State and local organizations within the Emergency Planning Zones have been assigned, that the emergency responsibilities of the various supporting organizations have been specifically established, and that each principal response organization is staffed to respond and to augment its initial response on a continuous basis.

Emergency Plan

The Shift Supervisor for each unit of the North Anna Power Station is initially designated as the Emergency Director. When an abnormal condition arises it is his responsibility to determine if the abnormality meets any of the emergency classifications specified in the plan and to implement the Plan, if necessary. There is 24 hours a day communication capability between the Station and Federal, State, and local response organizations to ensure rapid transmittal of accurate notification information and emergency assessment data.

Responsibility for overall performance of the emergency response organization is vested in the Station Manager, who is empowered to implement company policy with regard to operation of the North Anna Power Station. Qualified members of the station staff who report directly have been assigned specific responsibilities for the major elements of emergency response.

Updated written agreements have been executed with appropriate agencies and organizations.

B. Onsite Emergency Organization

Planning Objective

To assure that on-shift facility operator responsibilities for emergency response are unambiguously defined, that adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, and timely augmentation of response capabilities is available, and that the interfaces among various onsite response activities and offsite support and response activities are specified.

Emergency Plan

The Shift Supervisor on duty is designated as the Emergency Director until relieved by a senior member of the plant staff. A six-step line of succession has been established commencing with the Station Manager and descending in order by station staff members reporting directly to him. The authorities and responsibilities of the Emergency Director have been clearly specified, including those that cannot be delegated. The Emergency Director can immediately and unilaterally declare an emergency and make offsite notifications.

Station staff emergency assignments have been made and the relationship between the emergency organization and normal staff complement are shown in the Plan. Positions and/or titles and qualifications of shift and plant staff personnel both on and offsite who are assigned major emergency functional duties are listed. Minimum shift manning requirements are in the plan. The augmentation time is one hour. The augmentation time is acceptable on an interim basis. The minimum shift manning does not meet the objectives to Table B-1 of NUREG-0654. A description of minimum shift manning meeting the objectives of Table B-1 of NUREG-0654 and an acceptable proposed schedule for meeting those objectives has been provided.

Corporate management personnel who will augment the plant staff and their duties and responsibilities have been established; a long-term emergency organization framework is in place, headed by the Executive Vice President-Power. Interfaces between and among the company corporate staff, station staff, governmental and private sector organizations and technical and/or engineering contractor groups have been specified along with services to be provided.

C. Emergency Response Support and Resources

Planning Objective

To assure that arrangements for requesting and effectively using assistance resources have been made, that arrangements for State and local staffing of the operator's Emergency Operations Facility have been made, and that organizations capable of augmenting the planned response have been identified.

Emergency Plan

Arrangements for requesting and utilizing outside resources have been made including authority to request implementation of the Department of Energy Radiological Assistance Plan and The Interagency Radiological Assistance Plan, as well as requesting assistance from the reactor vendor and the architect/engineer by either the Emergency Director or Recovery Manager. In addition, administrative and technical personnel plus radiation monitoring and protective equipment from the VEPCO Surry Power Station are available as well as services from several offsite radiological laboratories with response times of about one hour to 3 1/2 hours. The Emergency Operations Facility will be activated for the more serious emergency classifications having or potentially having environmental consequences (Alert, Site Emergency, General Emergency). The facility will accommodate representatives from Federal, State and local government agencies, as well as representatives from contractor and other support groups. It will be the central data collection point for providing information needed by primary response agencies for implementation of protective actions.

D. Emergency Classification System

Planning Objective

To assure that a standard emergency classification and action level scheme is in use by the nuclear facility operator, including facility system and effluent parameters; and to assure that State and local response organizations, will rely on information provided by facility for determinations of initial offsite response measures.

Emergency Plan

Four standard emergency classes (i.e. Notification of Unusual Event, Alert, Site and General Emergency) have been established. Reactor system values and parameters for each classification are discussed. Initiating conditions for each class form the basis for establishment of specific instrumentation readings which, if exceeded, initiate the emergency class. The current Virginia Radiological Emergency Response Plan (RERP) and the local counties emergency classification system define only two emergency levels, "Yellow" (Site Emergency) and "Red" (General Emergency). When the State Office of Emergency and Energy Services (OEES) is notified of either a "Yellow" (Site Emergency) or "Red" (General Emergency) they will notify the Department of Health, who will implement their response procedures. The Emergency Director will recommend protective actions according to the guidance of NUREG-0654, using Form 6.1, "Report of Radiological Emergency."

The State is presently revising and updating its RERP to address the interim guidance contained in NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," January 1980. Shortly, the State will be assisting political subdivisions located within a 10-mile radius of the two nuclear power stations in Virginia in updating their RERPs to meet the new criteria. The revised State and local government Plans will also include the revised emergency classes. In the interim the Station will report the emergency classification in both

systems to insure compatability. This measure is acceptable on an interim basis. We will await the FEMA finding on this matter.

E. Notification Methods and Procedures

Planning Objective

To assure that procedures have been established for notification, by the facility, of State and local response organizations and for notification of emergency personnel by all response organizations; to assure that the content of initial and followup messages to response organizations and the public have been established; and to assure that means to provide early warning and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

Emergency Plan

Procedures have been established for notification of State and local response organizations in case of emergency. The Emergency Director has been given authority and responsibility for making prompt notification to these agencies. In addition, the Emergency Director is empowered to implement the activation procedures set forth in the Commonwealth of Virginia Radiological Emergency Response Plan or recommend activation of local county plans. A General Emergency ("Red") or Site Emergency ("Yellow") require immediate appropriate county(s) notification prior to State notification. The type and amount of information to be reported to State and County(s) officials have been predetermined in conformance with NUREG-0654 recommendations and is shown on the "Report of Radiological Emergency" Form along with completion and use instructions. A procedure for message verification is included in the form. Contents of initial and followup public and response organization messages ranging from "Local - No Action Necessary" to "State -Sheltering or Evacuation" have been prepared and are part of the station, State and County(s) Emergency Plans. A warning system has been proposed consisting of a combination of sirens and tone alert radios for installation in Louisa, Orange, Spotsylvania, Caroline and Hanover Counties. The design objective of the system is to alert 100% of the population within 5 miles of the site and

90% of the population from 5-10 miles from the site within 15 minutes. The design objective for the remaining 10% of the public within 10 miles of the site is notification within 45 minutes after notification of local officials. The inservice date of the warning system will be in accordance with the final rule on this matter. An acceptable proposed schedule for ordering, receiving, constructing, testing and full service has been provided.

F. Emergency Communications

Planning Objective

To assure that provisions exist for prompt communications among principal response organizations, to emergency personnel and to the public.

Emergency Plan

The station communication system is designed to provide secure, redundant and diverse communications to all essential onsite and offsite locations during normal operations and under accident conditions. Communication systems are designed to preclude the failure of one component impairing reliability of the entire system. Within-station systems are comprised of a 5 channel public address system, UHF and VHF two-way radio systems, a private branch (PBX) exchange and a sound powered telephone system. Offsite systems are comprised of both listed, unlisted and leased telephone lines, a microwave system and UHF and VHF two-way radio systems capable of reaching both the Louisa and Spotsylvania County sheriff's department. Two separate commercial telephone lines are dedicated to NRC communications.

These telephones plus other unlisted telephones are located in plant areas manned 24 hours a day. The Emergency Director will in emergency situations communicate directly with the State dispatcher, the dispatchers at each of the five surrounding counties, the NRC duty officer and if required the DOE (IRAP) duty officer. These offices are manned 24 hours a day.

Communications between the Control Room and the Technical Support Center, Operations Support Center and Emergency Operations Facility, (i.e., telephone and radio) are available. Tests of the systems are held once per quarter for telephone systems and once per month for radio systems.

G. Public Information

Planning Objective

To assure that accurate and timely information is provided to the public on how they will be notified and what their initial actions should be; to assure that the principal points of contact with the news media for dissemination of information (including physical location or locations) are established in advance; and to establish procedures for coordinated dissemination of information to the public.

Emergency Plan

The Emergency Operations Facility will serve as the principal point of interaction between station, governmental authorities and corporate management for exchange of information. Informational news releases will be coordinated with the Corporate Public Relations Department. The Executive Manager - Licensing and Quality Assurance is the Company designated spokesperson responsible for news release approval and for briefings with the news media. Briefings will be conducted at the public news center located on the second floor of the Mineral Volunteer Fire Department building.

The Licensee will conduct annual programs to keep the news media serving the population in the 10 mile Emergency Planning Zone acquainted with emergency plans, effects of radiation and points of contact for release of public information in an emergency. The Licensee will also provide periodic dissemination of educational information to the public on, but not limited to, emergency preparedness, protective actions in case of emergency radiation, sheltering and evacuation routes. Coordination with State and local authorities has been established to ensure the public including the permanent and transient population within 10 mile Emergency

Planning Zone are informed on an annual basis by utility bill inserts, notification in telephone books, newspaper ads and postings. A statistical sample of residents within an approximate 10 mile radius will be taken annually to assess the public's awareness of the prompt notification system and availability of information of what to do in case of an emergency.

Emergency Facilities and Equipment

Planning Objective

To assure that adequate emergency facilities and equipment to support the emergency response are provided.

Emergency Plan

Emergency facilities needed to support an emergency response have been provided including a Technical Support Center, Emergency Operations Facility and Operations Support Center. Each will be activated for an Alert or higher emergency classification. The Technical Support Center has been established in the Records Building adjacent to the Station inside the protected area. The Technical Support Center contains a complete set of controlled drawings, technical manuals and other records. The Emergency Operations Facility is currently located at the North Anna Visitors Center and will be utilized to evaluate and coordinate emergency and re-entry/ recovery operations on a continuing basis by the Licensee, Federal and State officials. It will also be the center for receipt and analysis of field monitoring information submitted by field teams. The facility is located within one mile from the station. The alternate Emergency Operations Facility is located at the Louisa County Courthouse complex located approximately nine miles from the site.

The Operations Support Center (assembly area) is located in the station's cafeteria and will be the assembly point for unassigned personnel. Equipment and supplies are normally stored in the Health Physics Office and in the Warehouse and will be transferred to the Operations Support Center as needed.

Detailed requirements are being developed on data transmission, staffing and physical facilities for the various emergency centers. The current proposals are adequate for initial full power operation.

Stored equipment is inspected and inventoried each quarter and replaced if in need of calibration or repair. Sufficient equipment exists to ensure a minimum inventory in case of replacement delay. Portable monitoring instruments are calibrated quarterly and count room instruments are source checked daily and calibrated annually. An appendix lists the equipment in emergency kits. However, no field monitoring or detection instruments are included in the list. These and other portable monitoring instruments that would be used in an emergency are stored in the Health Physics area.

Onsite monitoring systems and instrumentation used to initiate emergency measures and/or provide continuing assessment are identified. They are a meteorology system with wind speed and direction and temperatures capability; seismic instrumentation to measure ground acceleration levels; installed process radiation monitors to measure upward deviations in radiation levels in process lines that actually or potentially contain radioactive effluents; installed area radiation monitors to measure upward deviations in radiation levels in specific locations in the station; fire and smoke detection instruments placed in strategic plant locations; portable dose rate and radiation detection instruments and laboratory counting and analysis facilities.

The meteorology equipment at the site meets the criteria of Regulatory Guide 1.23, "Onsite Meteorological Programs," dated February 17, 1972. The licensee has provided a description of and an acceptable completion schedule for an upgraded meteorological program (NUREG-0654, Appendix 2).

Provisions for offsite monitoring equipment have been made. Meteorology, seismic data and respiratory protection equipment, portable detection

instruments and count room equipment can be obtained from the VEPCO Surry Power Station. The Virginia State Health Department is equipping a mobile laboratory with radioassay capability to respond to radiation emergencies. It will be equipped with a radio assigned to VEPCO's frequency. Offsite meteorological data can be obtained from the National Weather Service, Federal Aviation Administration or the offsite VEPCO meteorology station. Distance to these facilities from the Station ranges from 43 to 65 miles.

I. Accident Assessment
Planning Objectives

To assure the adequacy of methods, systems and equipment for assessing and monitoring actual or potential offsite consequences or a radiological emergency condition.

Emergency Plan

The Plan divides assessment actions into eight discrete areas of interest as opposed to being grouped under specific emergency classifications, since certain assessment actions will be required prior to classification of an emergency. The areas of interest are: Natural Phenomena, Personnel Hazards, Station Condition, Offsite Condition, Onsite Radiological, Offsite Radiological, Post Accident Sampling and Offsite Monitoring Teams. Assessment actions for each area of interest are listed.

Procedures contain system and radiological effluent parameter values characteristic of a spectrum of off-normal conditions and accidents. These parameter values and other information are tabulated to initiating conditions for each of the Emergency Classes. Specific setpoints and alarms both audio and visual in the control room alert the operator.

The installed Radiation Monitoring System consists of process and area monitors that read out and record in the Control Room. The process system continuously monitors selected lines actually or potentially

containing radioactive effluents. The high range monitors on the Process Vent, Ventilation Vent and Main Steamline can be used to determine the source term. A range of 10,000 R/hr allows conversion to Curies released using methods contained in the Emergency Plan Implementing Procedures.

Emergency Plan Implementing Procedures (EPIPs) provide the methodology for determining the magnitude of a release by three separate and independent methods: (1) using data or samples continuously obtained by the onsite Radiation Monitoring System, (2) using known inventory data for the system(s) affected and (3) obtaining offsite data from air samplers or dosimeters which are continuously in place, or taking radiation surveys and appropriate samples, and using this data to calculate releases.

J. Protective Response
Planning Objectives

To assure that a range of protective actions is available for the plume exposure pathway for emergency workers and the public, guidelines for the choice of protective actions during an emergency, consistent with federal guidance, are developed and in use, and that protective actions for the ingestion exposure pathway appropriate to the locale have been developed.

Emergency Plan

An area within 5000 feet of the North Anna 3 and 4 units is owned by the Licensee and is defined as the Exclusion Area. During normal operations VEPCO employees, contractor personnel, site visitors and members of the general public may be in this area. When an emergency requiring evacuation is declared, measures will be taken in cooperation with local and State agencies to evacuate persons in the Exclusion Area including boaters on the exclusion portion of Lake Anna. However, persons considered as transients on VEPCO owned property will always be evacuated when an Alert or higher classification emergency is declared. VEPCO employees considered as nonessential may also be evacuated if the

projected radiation dose to the majority of affected personnel will exceed 1 Rem whole body or 5 Rem thyroid due to inhalation.

The recommended actions (i.e. sheltering or evacuation) made to the State for a Site (State-Yellow) or General (State-Red) Emergency will be based on current meteorological data and projected whole body and/or thyroid dose, factored against the protection afforded by dwellings in the plume exposure pathway.

It is estimated that the primary sector and the two buffer sectors can currently be alerted to the emergency and given instructions within two hours and within 15 minutes once the early warning system is installed. Population distribution and evacuation time estimates for zones within a 2, 5 and 10 mile radius have been compiled and are included in the Plan. Evacuation routes for station employees being evacuated are described in and shown on an area map that is part of the Plan. Evacuation routes (maps) for the general public are contained in individual County Emergency Response Plans.

K. Radiological Exposure Control

Planning Objectives

To assure that means for controlling radiological exposures, in an emergency, are established for emergency workers and the affected population.

Emergency Plan

Emergency response personnel may receive radiation exposure in excess of the limits imposed by 10 CFR 20 when authorized by the Emergency Director. Emergency Plan Implementing Procedures contain emergency guidelines for whole body and thyroid dose consistent with EPA Emergency Worker and Life Saving Activity Protective Actions Guides.

The station will provide and distribute self-reading and accumulative type dosimeters to personnel involved in emergency onsite response

regardless of company affiliation. Dose records for workers will be maintained and checked on a 24-hour per day basis throughout the emergency.

Onsite contamination control procedures for personnel, equipment and access control are in place. Decontamination of personnel and equipment is required when the contamination level equals or exceeds 1000 dpm/100cm². Criteria for permitting return of contamination areas and their contents to normal use are stated in the appropriate contamination control procedures.

The station will supply clothing and decontamination materials particularly with respect to radioiodine skin contamination to onsite personnel required to relocate and who routinely leave the site.

L. Medical and Public Health Support

Planning Objectives

To assure that arrangements are made for medical services for contaminated individuals.

Emergency Plan

VEPCO has made arrangements with the Medical College of Virginia, Virginia Commonwealth University to provide medical assistance to site personnel injured or exposed to radiation and/or radioactive material. The Medical College has set up a special area of the hospital for treatment with appropriate health physics functions. VEPCO does not have arrangements for a backup hospital in the local area. Based on the quality of the facilities at the Medical College of Virginia, we find the arrangement acceptable.

The Station has a first aid facility that contains the normal complement of first aid supplies and equipment necessary to treat injuries not involving hospitalization or medical services. First aid team members have at least a basic first aid certificate and many have had multi-media

training. Arrangements have been made with volunteer rescue squads in the Counties of Louisa and Spotsylvania to transport personnel to the Medical College of Virginia if necessary.

M. Recovery and Reentry Planning and Postaccident Operations

Planning Objective

To assure that general plans for recovery and reentry are developed.

Emergency Plan

The Corporate Emergency Response Plan requires that a Recovery Center be established at VEPCO's corporate headquarters. The Recovery Team will consist of experienced company management headed by the Executive Vice President-Power and supervisory personnel who have the authority to assure the best available use of Company resources to assist in rapid recovery. The corporate plan is consistent with the Atomic Industrial Forum's recommended Recovery Organization.

Any decision by the Company's part to relax protective measures must be based on a comprehensive review of station system parameters and reached in and by a meeting of the Station Manager (Emergency Director) and Manager Nuclear Operations (Recovery Manager). The decision must be concurred in by the Executive Vice-President Power (Corporate Response Manager). Notification of the decision and any resulting changes to the corporate or station response to the recovery will be maintained between Federal, State and local entities. Conditions considered appropriate for consideration of relaxation of protective measures are: station parameters of operation no longer indicate a potential or actual emergency; release of radioactivity from the station is controllable, no longer exceeds permissible levels and presents no public danger; the station is capable of sustaining itself in a long term shut down condition and station entry and clean up is possible without workers receiving in excess of their permissible exposure.

The Recovery Manager will notify onsite agencies representatives, the Station Manager, Federal agencies (e.g., NRC and DOE) and State and local county(s) Emergency Operations Centers of the decision to initiate recovery operations and of any resulting changes to the Station organization structure.

N. Exercises and Drills

Planning Objective

To assure that periodic exercises are conducted to evaluate major portions of emergency response capabilities, that the results of exercises form the basis for corrective action for identified deficiencies and that periodic drills are conducted to develop and maintain key skills.

Emergency Plan

A combined exercise involving Station, State and local personnel will be held annually. The scenario for the exercise will be mutually agreed to and rotated each year to ensure that all major elements of the Emergency Plan are tested over a five year period. At least once every six years an exercise will be scheduled for each of the off-shifts.

Observers from Federal, State and local governments will be invited to participate and/or critique all exercises of Emergency Preparedness. A formal critique of the exercise will be held as soon as possible after the exercise. The critique will be sent to participating organizations.

The Station Manager is responsible for ensuring that deficiencies disclosed in the critique are addressed and appropriate corrective action is taken. The Manager Quality Assurance is responsible for corrective action at the Corporate level.

Drills based on Site or General Emergency Conditions will be held at predetermined frequencies for response components (e.g. fire, medical, communications, Health Physics etc.) to ensure maximum effectiveness of

the plan. Appropriate offsite agencies will participate in or observe the drill(s) where applicable. Emergency Plan Implementing Procedures will be utilized to ensure adequacy of an overall response to the scenario. Performance criteria will be established for all levels of participation. Audit personnel will be stationed so as to observe response to and adequacy of the plan. Simulated emergency conditions will be reviewed and approved by the Station Nuclear Safety and Operations Committee prior to the drill. This committee is also responsible for critique review.

0. Radiological Emergency Response Training

Planning Objective

To assure that radiological emergency response training is provided to those who may be called upon to assist in an emergency.

Emergency Plan

Company personnel involved in emergency response will receive training appropriate to their functions, authority and role during an emergency. The training will be documented and annually assessed for suitability. Initial training programs will be separate from retraining programs. The programs will cover the specifics of individual assignments as well as interfaces with other response actions. The training will be conducted in formal fashion with individual tests at the end of training to determine each persons qualifications. The training plan will include practical drills as applicable, so each individual can demonstrate his ability to perform his emergency functions. During the drills on-the-spot correction of erroneous performance will be made with demonstrations of proper performance given to the individual(s) by the instructor. Offsite agencies who potentially will be called upon to participate in the plan have concurred with the responsibilities assigned their agency in the Plan by executing a Letter of Agreement with the Company. Each agreement agency has a copy of the Emergency Plan and will receive revisions with an acknowledgement receipt request to ensure that the Plan

is kept updated. Meetings with agreement agencies will be held biannually to discuss the plan and review agency status.

Local support service groups such as volunteer fire and rescue squads will be given formalized training in nuclear station response and what conditions may be encountered on an annual basis by the State and/or VEPCO personnel.

The Company provides formal training for individuals responsible for Emergency Planning in the form of available courses offered by other than Company agencies.

P. Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans

Planning Objective

To assure that responsibilities for plan development, review and distribution of emergency plans are established and that planners are properly trained.

Emergency Plan

The Emergency Plan and Emergency Plan Implementing Procedures are formally reviewed annually for adequacy and applicability by the Station Nuclear Safety and Operating Committee. Revisions to the Plan and/or EPIP's are also reviewed and approved by this committee. Approved revisions are forwarded to those on a "Controlled Distribution" list with a receipt acknowledgement request to ensure that holder's copy is maintained in current status. Quality Assurance personnel also periodically audit those on the distribution list and ensure their copy of the Plan is properly updated.

The Executive Manager of Licensing and Quality Assurance has overall authority and responsibility for Emergency Planning at the Corporate level. The Station has an Emergency Planning Coordinator whose responsibilities include updating of Emergency Plans and EPIP's and coordination

of these plans with other response organizations, distribution of revisions and obtaining and re-negotiating Letters of Agreement. Re-negotiated letters will be distributed as revisions to the Plan.

CONCLUSION

Based on our review of the Plan we have concluded that the Plan meets the Planning Objectives as applicable to the licensee (operator) of the "Criteria for Preparation and Evaluation and Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (For Interim Use and Comment)", NUREG-0654.

After receiving the findings and determinations made by the Federal Emergency Management Agency on the State and local emergency response plans, a supplement to this report will provide the staff's overall conclusions on the status of emergency preparedness for the North Anna Power Station and related Emergency Planning Zones.

APPENDIX C

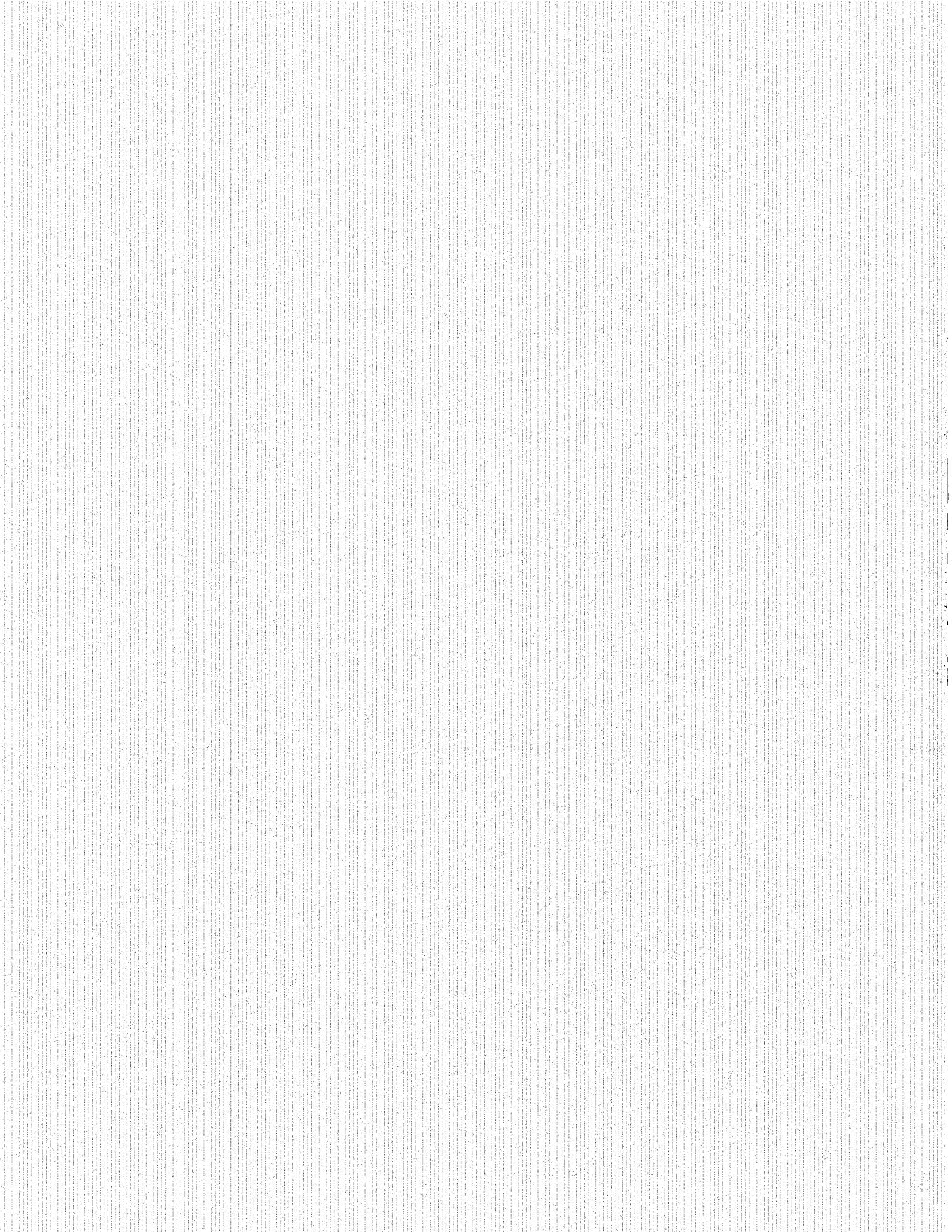
ERRATA TO SUPPLEMENT NO. 10
TO THE SAFETY EVALUATION REPORT
NORTH ANNA POWER STATION UNIT 2

<u>Page</u>	<u>Line</u>
B-12	3 through 9

Delete "In Section 5.2 of this supplement, we indicated....
we conclude that the exemption for this area of noncompliance
to Appendix G of 10 CFR Part 50 is justified."

This reflects the present staff position that an exemption is
not required for North Anna Unit 2.

NRC FORM 335 (7-77)		U.S. NUCLEAR REGULATORY COMMISSION BIBLIOGRAPHIC DATA SHEET		1. REPORT NUMBER (Assigned by DDC) NUREG-0053 Supplement No. 11	
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Safety Evaluation Report related to the operation of North Anna Power Station, Unit 2, Virginia Electric and Power Company, Supplement No. 11				2. (Leave blank)	
7. AUTHOR(S)				3. RECIPIENT'S ACCESSION NO.	
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) U. S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation Washington, D.C. 20555				5. DATE REPORT COMPLETED MONTH YEAR August 1980	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Same as 9, above.				DATE REPORT ISSUED MONTH YEAR August 1980	
				6. (Leave blank)	
				8. (Leave blank)	
				10. PROJECT/TASK/WORK UNIT NO.	
				11. CONTRACT NO.	
13. TYPE OF REPORT Safety Evaluation Report Supplement		PERIOD COVERED (Inclusive dates) April 1980 - August 1980			
15. SUPPLEMENTARY NOTES Docket No. 50-339				14. (Leave blank)	
16. ABSTRACT (200 words or less) <p>On June 4, 1976, the Nuclear Regulatory Commission issued its Safety Evaluation regarding the application for licenses to operate the North Anna Power Station, Units 1 & 2. The application was filed by Virginia Electric and Power Company. Supplement No. 1 to the Safety Evaluation Report was issued on June 30, 1976; Supplement No. 2 was issued on August 2, 1976; Supplement No. 3 was issued on September 15, 1976; Supplement No. 4 was issued on December 8, 1976; Supplement No. 5 was issued on December 29, 1976; Supplement No. 6 was issued on February 2, 1977; Supplement No. 7 was issued on August 18, 1977; Supplement No. 8 was issued on December 14, 1977; Supplement No. 9 was issued on March 31, 1978, and Supplement No. 10 was issued on April 10, 1980. Supplements 1 through 9 documented the resolution of several outstanding items. Supplement No. 10 addresses the requirements for fuel loading and conducting low power testing of North Anna Unit 2. This supplement, No. 11 addresses the requirements which must be completed prior to the issuance of a full power operating license for Unit 2.</p>					
17. KEY WORDS AND DOCUMENT ANALYSIS			17a. DESCRIPTORS		
17b. IDENTIFIERS/OPEN-ENDED TERMS					
18. AVAILABILITY STATEMENT Unlimited		19. SECURITY CLASS (This report) Unclassified		21. NO. OF PAGES	
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NUREG-0053
SUPP. NO. 11

SER RELATED TO THE OPERATION OF NORTH ANNA POWER STATION, UNIT 2

AUGUST 1980