JULY 2, 1980

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

VIRGINIA ELECTRIC AND POWER COMPANY

SPECIAL LOW POWER TEST PROGRAM

FOR

NORTH ANNA POWER STATION, UNIT NO. 2

DOCKET NO. 50-339

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1.0 SPECIAL LOW POWER TEST PROGRAM

1.1 Introduction

In Section I.G of Part II of Supplement No. 10 to the Safety Evaluation Report for North Anna Power Station, Unit No. 2 we indicated that one of the activities proposed was to conduct a series of natural circulation tests at power levels up to five percent of normal full power. The proposed test program was described in letters of February 8, 1980 and March 19, 1980, from Mr. Stallings to Mr. Varga.

The low power test program proposed by VEPCO consisted of nine tests, eight of which involve natural circulation in the reactor coolant system at low power conditions, but at normal, or nearly normal, operating pressures and temperatures.

The specific tests proposed by VEPCO were:

1. Natural circulation test;
2. Natural circulation with a simulated loss of offsite power;
3. Natural circulation with loss of pressurizer heaters;
4. Effect of secondary side isolation on natural circulation;
5. Natural circulation at reduced pressures;
6. Cooldown capability of the charging and letdown system;
7. Simulated loss of all onsite and offsite ac power;

8. Establishment of natural circulation from stagnant conditions;
9. Forced circulation cooldown (Part A) and boron mixing and cooldown (Part B)

The proposed low power test program for VEPCO was reviewed by the staff using the following five criteria:

1. The tests should provide meaningful technical information beyond that obtained in the normal startup test program.

2. The tests should provide supplemental operator training.

3. The tests should not pose an undue risk to the public.

4. The risk of damage to the nuclear plant during the test program should be low.

5. The radiation levels that will exist after the low power test program is completed (including that from crud deposits) must not preclude implementation of requirements stemming from the NRR Lessons Learned Task Force, Kemeny Commission, Rogovin Commission or Task Action Plan.

In a letter to the staff dated April 29, 1980, Westinghouse expressed concern with the conduct of two of the proposed tests (Test No. 8 "Establishment of natural circulation from stagnant conditions" and Test 9B "Boron mixing and cooldown") at plants other than Sequoyah. The reasons for their concern were: (1) special conditions required to conduct the tests and (2) little benefit is to be derived from repeating the test since plant behavior should not be plant specific, whereas the difficulty of performing the test remains the same.

By letter dated June 5, 1980, VEPCO requested deletion of Tests 8, 9A and 9B. Subsequently, Test 9A was incorporated into Test 4.

VEPCO also stated that in lieu of performing test 9B during the low power test program they would perform a similar test using decay heat instead of performing it with the reactor critical. This test would be performed in conjunction with a planned test to demonstrate cold shutdown. Use of decay heat eliminates many of the special conditions required for test 9B, thus reducing the risks associated with performing this test.

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 in four procedures to take advantage of established initial conditions. On June 18, 1980, VEPCO submitted the safety analysis and technical specification exceptions necessary to conduct these tests. They also requested an amendment to the operating license to reflect the technical specification exceptions and indicated that Westinghouse has reviewed and approved the safety analysis and technical specification exceptions. On June 24, 1980, VEPCO submitted changes to the test procedures that had also been approved by the safety committee.

The purpose of this safety evaluation is to present the results of the staff review of the proposed special low power test program since approval by the staff is necessary for the conduct of the program.

2.0 SAFETY ASSESSMENT

Tests 1, 3 and 5 listed in Section 1.1 (Natural circulation, Natural circulation with loss of pressurizer heaters, and Natural circulation at reduced pressure) have been combined and designated as ST-8; tests 2 and 7 (Natural circulation with a simulated loss of offsite ac power and Simulated Loss of all onsite and Offsite ac Power) have been combined into a single test designated at ST-9. Test 9A has been incorporated into Test 4 designated as Test ST-11 (Effect of steam generator secondary side isolation on natural circulation). Test 6 (Cooldown capability of the charging and letdown system) is designated as ST-6.

Sections 3.0, 4.0 and 5.0 of this evaluation address (1) VEPCO's request to delete tests 8 and 9A and 9B, (2) combining the tests, and (3) the test procedures.

Sections 6.0, 7.0, and 8.0 of this evaluation address (1) exceptions to the technical specifications, (2) operational safety criteria and (3) safety evaluation.

3.0 DELETION OF TEST 8, AND MODIFICATION OF TESTS 9A AND 9B

The desirability of conducting test 8 "Establishment of natural circulation from stagnant conditions, test 9A "Forced circulation cooldown" and test 9B "Boron mixing and cooldown" has been discussed with the NSSS vendor, Westinghouse, and with VEPCO. As a result of these discussions, VEPCO in a letter dated June 5, 1980, has requested that these tests be deleted or modified from the special test program. VEPCO stated that there is a significantly higher risk associated with performance of tests 8 and 9B as compared with the other tests because of the special test conditions required. VEPCO also stated that Westinghouse agrees with this concern. Since the purpose of Test 9A was to provide calibration data for reactor power measurements over a range of cold leg coolant temperatures it was to be conducted as a prerequisite to test 9B. By combining test 9A with test 4 sufficient data will be obtained for conducting the test program.

We have considered the VEPCO request to delete tests 8 and 9B and have concluded that test 8 can be deleted and a similar test to 9B may be performed using decay heat near the end of the startup test program for Unit No. 2 for the following reasons: (1) there is a greater risk involved in operating the plant under the conditions described in the tests, (2) there appears to be little benefit to be derived from conducting these tests at more than one plant. (The plant response to this test should not be plant specific and Westinghouse and TVA have agreed to make the data collected from Sequoyah available to other applicants for training purposes.), (3) the Sequoyah operators have received special training in performing these tests, thus minimizing the risk at Sequoyah, (4) since it will take approximately six months for these test results to be fed back into simulator training programs for other plants, the relative schedules of the near term operating license applicants is considered insignificant, and (5) VEPCO will conduct a test to demonstrate boron mixing and cooldown capability on natural circulation (similar to test 9B) at the end of its startup test program. At that time there will be sufficient decay heat to perform the test with the reactor subcritical. The same training benefits will be derived as if the test were performed as part of the low power test program because the test procedure will be close to operating conditions and relieves the operator of maintaining the reactor critical during test.

4.0 COMBINING TESTS

We have reviewed the VEPCO proposal to combine tests and have concluded that combining the tests will not compromise the test objectives with regard to training. Each of the first seven tests and test 9A originally proposed are addressed discreetly in the four combined tests. The principle reasons for combining the tests are to take advantage of established initial conditions (e.g., reactor coolant pumps tripped and main feedwater isolated). The changes will eliminate the time that would have been required to re-establish the initial conditions and could reduce the chance for operator error by not having to restart each test all over again. These changes will not affect the overall test results.

5.0 REVIEW OF THE TEST PROCEDURES

Westinghouse has reviewed the revised, combined test procedures and provided comments which VEPCO has incorporated. The staff has reviewed the test procedures using the following criteria:

1. The tests should provide meaningful technical information beyond that obtained in the normal startup test program.
2. The tests should provide supplemental operator training.
3. The tests should not pose an undue risk to the health and safety of the public.
4. The risk of damage of the facility during the test program should be low.
5. The radiation levels that will exist after the low power test program is completed (including that from crud deposits) must not preclude implementation of requirements from the NRR Lessons Learned Task Force, Kemeny Commission, Rogovin Commission or Task Action Plan.

We have reviewed the procedures for the low power tests and conclude that they are acceptable based on the above criteria. However, the simulated loss of onsite and offsite ac power (portion of ST-9) does not fully meet criteria 1 and 2. This test will provide information on decay heat removal with the steam driven auxiliary feedwater pump but using reactor power in lieu of decay heat. The auxiliary feedwater system configuration for this test will not be the same as the configuration which would exist in the event of a real loss of all ac power.

The normal lineup of the auxiliary feedwater system at North Anna Unit No. 2 consists of two motor driven and one steam turbine driven auxiliary feedwater pumps each providing auxiliary feedwater to one of the three steam generators. In the event of loss of both onsite and offsite ac power, only the steam turbine driven pump would be available and consequently only one steam generator would receive auxiliary feedwater. There is some concern that flow maldistribution in the core may occur and could result in power anomalies when the reactor is used as the heat source. Consequently, VEPCO would prefer and we agree, not to conduct the test with only one steam generator removing heat while simulating decay heat with reactor power. The test procedure specifies that operators will proceed to the auxiliary feedwater pumphouse and using sound power telephones, manually realign the auxiliary feedwater system to distribute the feedwater to all three steam generators and will manually control feedwater addition to each steam generator. The operators in the control room will monitor steam generator levels and give instructions to the operators in the auxiliary feedwater pumphouse. Although this procedure does not simulate an actual loss of all ac power, it will provide (1) some plant information on the capabilities of the auxiliary feedwater system, (2) operator experience in manually throttling flow and (3) experience in training the operators to coordinate critical system realignments and control at remote locations of the plant.

Based on our review of the test procedures, we conclude that the special low power test program can be safely conducted as proposed at North Anna Power Station Unit No. 2. We will witness selected portions of the special test as necessary to ensure that the safety precautions and acceptance criteria are met.

6.0 EXCEPTIONS TO TECHNICAL SPECIFICATIONS

Exceptions to a number of technical specification requirements for North Anna Unit No. 2 will be made during the low power test program. Some exceptions are required because of operation with a critical reactor under conditions outside of the range allowed in the Technical Specifications (e.g. natural circulation conditions and low coolant temperatures and pressure). Other exceptions are required because some systems normally required to be operable will be rendered temporarily inoperable as part of the test program (e.g. simulated loss of offsite power and simulated loss of all ac power.) The exceptions required are listed in Table 6.1 for each of tests in the Special Lower Power Test Program and are discussed below.

6.1 Exceptions Involving Reactor Trip and Safety Injection (SI)

The exceptions involving reactor trip and safety injection (T.S. 2.2.1, 3.3.1, 3.3.2) are:

a. The Over-Temperature and Over-Power ΔT trip functions are based on reactor coolant system (RCS) hot and cold leg temperatures obtained from resistance temperature detectors (RTD's) which are located in bypass manifolds. Under natural circulation conditions, the very low expected flows in the bypass manifolds could result in spurious readings and inadvertent trips. Therefore, these trip functions will be bypassed. During the Special Low Power Test Program, the protection functions of these automatic trips will be performed by operator actions based on limiting values of system parameters and automatic trip at reduced neutron flux setpoints.

b. The setpoint for reactor trip on steam generator low level, which has a normal setting of 21% of the narrow range span will be reduced to 5% of the narrow range span. This reduction will be made to prevent inadvertent scrams for tests where it may be difficult to maintain the margin between the normal operating level and the normal setpoint. This trip provides margins for maintaining the secondary side heat sink. The low decay heat resulting from the low power levels during the test program permits reduction in the level setpoint.

c. Automatic safety injection will be blocked to prevent inadvertent safety injection at the low coolant flow rates expected in the test program. Manual safety injection initiation will be operable. In addition, any safety injection signal will provide a reactor trip and control room indication/alarm. For tests 3 and 5, the low pressurizer pressure safety injection signal which would cause reactor trip, is blocked to allow operation at low pressures. During this period of operation, the pressurizer power operated relief block valve will be closed to remove the major credible source of inadvertent depressurization.

d. Secondary pressure trip protection will be modified in several ways. The safety injection signal resulting from high steam line flow in two main steam lines coincident with either low-low Tavg or low steam line pressure in two main steam lines will be modified by (a) blocking the low-low Tavg input and (b) setting the high steam line flow setpoint to zero flow (i.e., bistable in tripped position). Reactor trip and main steam isolation valve (MSIV) isolation will then be actuated by low steam line pressure signals in any two steam lines to protect against steam line breaks downstream of the steam line check valves. For test 4 the setpoint for low steam line pressure will be reduced from the normal value of 600 psig to about 500 psig to permit operation at primary coolant temperatures down to about 550°F.

The reactor trip resulting from the SI signal caused by high differential pressure between steam lines will be disabled. This signal gives the normal protection against large steam line ruptures upstream of the steam line check valves. Manual action based on the operational safety criteria will be used for such breaks.

TABLE 6.1

EXCEPTIONS TO TECHNICAL SPECIFICATIONS FOR LOW POWER TEST PROGRAM

TECHNICAL SPECIFICATION		TEST 1	2	3	4	5	6	7
2.1.1	Core Safety Limits	X	X	X	X	X	X	X
2.2.1	Various Reactor Trips							
	Overtemperature ΔT	X	X	X	X	X		X
	Overpower ΔT	X	X	X	X	X		X
	Steam Generator Level	X	X	X	X	X	X	X
3.1.1.4	Moderator Temperature Coefficient				X			
3.1.1.5	Minimum Temperature for Criticality				X			
3.3.1	Various Reactor Trips							
	Overtemperature ΔT	X	X	X	X	X		X
	Overpower ΔT	X	X	X	X	X		X
	Steam Generator Level	X	X	X	X	X	X	X
3.3.2	Safety Injection - All automatic functions	X	X	X	X	X	X	X
3.4.4	Pressurizer			X		X		X
3.7.1.2	Auxiliary Feedwater		X				X	X
3.10.3	Special Test Exception Physics Tests				X			

X -- Exceptions Required

6.2 Other Exceptions to Technical Specifications

a. T.S. 2.1.1, "Reactor Core Safety Limits," gives limits to the average reactor coolant temperature in terms of reactor power, RCS pressure and number of operable loops. For the natural circulation tests, this specification cannot be met simply because no reactor coolant (RC) pumps would be running. However, the intent of the specifications with respect to clad temperature limits will be met by the planned operational limits on core exit temperature, average coolant temperature, loop ΔT and subcooling margin.

b. T.S. 3.1.1.4, "Moderator Temperature Coefficient," limits the moderator temperature coefficient of reactivity to zero or negative values. During some tests, this coefficient may be slightly positive. However, the isothermal temperature coefficient is expected to be zero to slightly negative. The effect of moderator temperature coefficient of reactivity was considered in the safety analysis.

c. The minimum temperature for criticality is limited to 541^{o}F by T.S. 3.1.1.5,"Minimum Temperature for Criticality," and to 531^{o}F by T.S. 3.10.3, "Special Test Exceptions - Physics Tests. During Test 4 it is expected that the average reactor coolant temperature will drop below these limits. VEPCO has stated that operation with the average reactor coolant temperatures as low as 500^{o}F is acceptable assuming that:

1. Control Bank D is inserted no deeper than 100 steps withdrawn and,

2. The Power Range Neutron Flux low setpoint and Intermediate Range Neutron Flux reactor trip setpoint are reduced from 25% rated thermal power (RTP) to 7% RTP.

These restrictions reduce the consequences of transients involving individual rod withdrawl or rod bank withdrawal by limiting reactivity insertion rates from inadvertent individual rod withdrawal or rod bank withdrawal, providing sufficient shutdown margins, maintaining the moderate temperature coefficient at near zero values and limiting the maximum power during power excursions.

The trip setpoint of 7% RTP is based on a coolant temperature in the reactor vessel downcomer region of about 545^{o}F. Operation at a lower coolant temperature in the downcomer region results in a reduced output of the ex-core detectors for a given core power. Hence, for operation at lower coolant temperatures, reactor trip would occur at powers higher than 7% RTP. This effect was included in the safety analysis by using a conservative estimate of 1% reduction in the ex-core detector reading per ^{o}F. Prior to the start of test 4, a special test will be run to assure that the actual decrease in the ex-core detector reading is less than that used in the safety analyses.

T.S. 3.4.4 requires operability of the pressurizer. In tests 2, 3, 5, and 7 the pressurizer heaters will either be turned off or rendered inoperable as the result of loss of power. This mode of operation is found acceptable because pressure control can still be maintained by use of the auxiliary spray and pressurizer level control.

T.S. 3.7.1 requires operability of at least three independent steam generator auxiliary feedwater pumps. During two tests simulating loss of offsite power and total loss of ac power, the auxiliary feedwater system will be rendered partially inoperable (motor driven pumps). The low decay heat allows sufficient time ($\sim$ 1/2 hour) for plant personnel to return ac power and regain steam generator level.

7.0 OPERATIONAL SAFETY CRITERIA

As the result of a safety evaluation of the Low Power Test Program at North Anna Unit 2, VEPCO has specified a set of operational safety criteria for test conditions (see Table 7.1) and for conditions requiring prompt operator initiation of reactor trip or safety injection or termination of test. The safety criteria include:

a. limits on maximum core exit temperature, maximum loop ΔT for any loop, maximum coolant average temperature, and minimum subcooling. These limits and operator actions are provided to ensure adequate margin to the saturation temperature and adequate core cooling.

b. limits on the minimum steam generator water level to provide a sufficient secondary side heat sink.

c. limits on the minimum pressurizer water level for heater coverage and pressure control.

d. limits on maximum insertion of control band D to minimize consequences of inadvertent rod withdrawal and maintain a small moderator temperature coefficient while providing sufficient margin for shutdown.

e. limits on the Power Range Neutron Flux low setpoint and Intermediate Range Neutron Flux reactor trip setpoint to limit maximum power to low values following possible uncontrolled power increases.

f. limits on containment pressure and unplanned or unexplained changes in pressurizer water level and pressure.

TABLE 7.1

OPERATIONAL SAFETY CRITERIA

1. Guidelines for All Tests

a) Primary System Sub-cooling (T_{sat} Margin)	> 20^{o}F
b) Steam Generator Water Level	> 33% Narrow Range Span
c) Pressurizer Water Level	
(1) With RCPs running	> 22% Span
(2) Natural Circulation	$\geq$ Value when RCPs tripped
d) Loop ΔT	$\leq$ 65^{o}F
e) T_{avg}	$\leq$ 580^{o}F
f) Core Exit Temperature (highest)	$\leq$ 610^{o}F
g) Power Range Neutron Flux Low Setpoint and Intermediate Range Neutron Flux Reactor Trip Setpoints	$\leq$ 7% RTP
h) Control Bank D	100 steps withdrawn or higher

2. Reactor Trip and Test Termination must occur if any of the following conditions are met:

a) Primary System Sub-cooling (T_{sat} Margin)	$\leq$ 15^{o}F
b) Steam Generator Water Level	< 5% Narrow Range Span or equivalent Wide Range Level
c) NIS Power Range, 2 channels	> 10% RTP
d) Pressurizer Water Level	< 17% Span or an unexplained decrease of more than 5% not concurrent with a T_{avg} change
e) Any Loop ΔT	> 65^{o}F
f) T_{avg}	> 580^{o}F
g) Core Exit Temperature (highest)	> 610^{o}F
i) Uncontrolled rod motion	

TABLE 7.1 (Continued)

3. Safety Injection must be manually initiated if any of the following conditions are met:

a) Primary System Sub-cooling (T_{sat} Margin)	$\leq 10^{o}F$
b) Steam Generator Water Level	< 0% Narrow Range Span or equivalent wide range level
c) Containment Pressure	$\geq$ 17 psia
d) Pressurizer Water Level	< 10% Span or an unexplained decrease of more than 10% not concurrent with a T_{avg} change.
e) Pressurizer Pressure	Decreases by 200 psi or more in an unplanned or unexplained manner.

The staff has been concerned with uncertainties in the core ΔT and RCS subcooling measurements under natural circulation flow conditions. These uncertainties are the result of uncertainties in the core exit thermocouple and loop resistance temperature detector readings. The North Anna subcooling meters use input from four hot leg RTD's and twenty core exit thermocouples. For North Anna the concerns involve principally (a) possible stratification in the hot and cold leg piping, (b) thermowell heat loss effects and (c) long time constants for the hot and cold leg temperature measurements since the resistance temperature detectors are inserted in thermowells which have good thermal contact with the RCS piping. Uncertainties in the temperature measurements are difficult to predict since local flow and temperature patterns under natural circulation conditions are unknown. VEPCO has stated that the results of Test 1 will be reviewed to determine the behavior of these temperature detectors. The objective of this review, which will be completed prior to the start of the remaining natural circulation tests, is to evaluate the adequacy of these measurements under natural conditions with respect to the specified core ΔT and RCS subcooling limits.

Since of the two North Anna subcooling meters uses the highest of two RTD's and ten core exit thermocouples, the uncertainties associated with the hot leg RTD's should not compromise the safety of these tests.

8.0 SAFETY EVALUATION

8.1 Introduction

VEPCO submitted the results of a study of the safety effects of the special conditions of the Low Power Test Program, including the exceptions to the technical specifications, which lead to operating conditions that are outside the bounds of conditions assumed in the Final Safety Analysis Report (FSAR). The effects of these conditions on the Condition II, III, and IV events treated in Chapter 15 of the FSAR were evaluated.

Condition II events, at worst, shall result in a reactor trip with the plant being capable of return to operation. Condition II events shall not propagate to cause a more serious Condition III or IV event and are not expected to result in fuel rod failure or reactor coolant system over-pressurization;

Condition III events are very infrequent faults which will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude immediate resumption of operation. For infrequent incidents, the plant should be designed to limit the release of radioactive material to assure that doses to persons offsite are limited to values which are a small fraction of 10 CFR Part 100 guideline values. A Condition III event shall not generate a Condition IV event or result in loss of function of the reactor coolant system or containment barriers;

Condition IV events are limiting design bases accidents which are not expected to occur, but are postulated because their consequences include a potential for the release of significant amounts of radioactive material.

System design for Condition IV events will prevent a fission product release to the environment which would result in an undue risk to the health and safety of the public in excess of limits established in 10 CFR Part 100. A Condition IV event is not to cause a consequential loss of required function of systems needed to mitigate the consequences of the accident, such as the emergency core cooling system and the containment.

The results of the analyses of Condition II, III and IV events are categorized in Table 8.1 according to the following evaluation bases.

ANALYSIS OF TEST	RESULTS OF ANALYSIS
Bounded by FSAR analysis results	1
Reanalysis shows fuel clad integrity is maintained	2
Operator action is required for protection	3
Probability of occurrence reduced by restrictions on operating conditions	4
Probability of occurrence reduced by short-testing period only	5

Table 8.2 lists those events for which a qualitative evaluation is sufficient to conclude that the consequences of the event for the low power test program are bounded by the FSAR results.

TABLE 8.1

SUMMARY OF SAFETY EVALUATION

TRANSIENT TEST:	1	2	3	4	5	6	7
RCCA Bank With., Subcritical	2, 4	2,4	2,4	2,4	2,4	1	2,4
RCCA Bank With., at power	4	4	4	4	4	1	4
RCCA Misalignment	1	1	1	1	1	1	1
Boron Dilution	1	1	1	1	1	1	1
Partial Loss of Flow	1	1	1	1	1	1	1
Start Inactive Loop	1	1	1	1	1	1	1
Loss of Load	1	1	1	1	1	1	1
Loss of Feedwater	1	1	1	1	1	1	3
Loss Offsite Power	1	1	1	1	1	1	3
Excessive Feedwater	2	2	2	2	2	1	2
Excessive Load	2	2	2	2	2	1	2
RCS Depressurization	1	1	4	1	4	1	1
Steam Depressurization	1	1	1	1	1	1	1
Spurious Safety Injection	1	1	1	1	1	1	1
Small LOCA	3	3	3	3	3	3	3
Small Secondary Breaks	2,3	2,3	2,3	2,3	2,3	1	2,3
Single RCCA Withdrawal	4	4	4	4	4	1	4
Misloaded Fuel Assembly	1	1	1	1	1	1	1
Complete Loss of Flow	1	1	1	1	1	1	1
Waste Gas Decay Tank Brk.	1	1	1	1	1	1	1
Major LOCA	3	3	3	3	3	3	3
Major Secondary Break	2,3	2,3	2,3	2,3	2,3	1	2,3
S/G Tube Rupture	1	1	1	1	1	1	1
RCP Locked Rotor	1	1	1	1	1	1	1
Fuel Handling	1	1	1	1	1	1	1
Ruptured CRDM	3,5	3,5	3,5	3,5	3,5	1	3,5

TABLE 8.2

EVENTS BOUNDED BY FSAR RESULTS

EVENT	REASON WHY CONSEQUENCES BOUNDED BY FSAR
RCCA Misalignment	Decrease in power caused by dropped rod cluster control assembly (RCCA). No increase in probability or consequences caused by test condition.
Uncontrolled Boron Dilution	Low setpoint for neutron flux scram (7%) Control rods not inserted to insertion limit Constant operator monitoring during tests.
Partial Loss of Coolant Flow	Low power level
Startup of Inactive Reactor Coolant Loop	Small moderator reactivity coefficients. Low power level during test. Low setpoint for neutron flux scram.
Loss of Offsite Power to Station Auxiliaries (Station blackout)	Low power level. Trip on low-low steam generator water level. Low decay heat.
Loss of Normal Feedwater	Low power level. Trip on low-low steam generator water level. Low decay heat.
Loss of Load and/or Turbine trip	Low power level. Turbine not operating

TABLE 3.2 (Continued)

EVENT	REASON WHY CONSEQUENCES BOUNDED BY FSAR
Excessive Load Increase Incident	Turbine not operating. Load control limited to single steam dump valve or relief valves.
Spurious Operation of Safety Injection System	Actuation of safety injection by any source except manual action disabled during tests.
Accidental Depressurization Of Main Steam System	For FSAR analysis where transient starts at hot shutdown with worst RCCA stuck out of core, safety injection prevents return to criticality. For tests, reactor remains subcritical down to room temperature without safety injection.
Misloaded Fuel Assembly	Low power level
Complete Loss of Flow	Low power level
Waste Gas Decay Tank Rupture	Low fission product inventory
Single Reactor Coolant Pump Locked rates	Low power level
Fuel Handling Accidents	Accident independent of low power test program conditions or low fission product inventory.
Rod withdrawal from subcritical condition	Test procedures require that RC pumps will be operating before rods withdrawn from subcritical condition.
Steam Generator Tube Rupture	Low radioactivity level in primary and secondary systems.

8.2 Cooldown Transients

Cooldown transients considered in the FSAR included (a) excessive increase in load, (b) accidental depressurization of the main steam system, (c) small secondary system breaks, (d) excessive heat removal due to feedwater system malfunctions, and (e) major secondary system breaks. With the exception of some types of breaks in the main steam lines, the consequences of these transients during the test program should be minor because of the low power levels, low neutron flux trip and small moderator temperature coefficient of reactivity.

The turbine will not be used during the tests and load control will be limited to operation of a single steam dump valve or the relief valves. A load increase or small steam pipe break equivalent to the opening of a single steam pressure relief valve, dump valve or safety valve would cause a small (~4% RTP), increase in reactor power, assuming the bounding negative value of the moderator temperature coefficient for the beginning of life (Cycle 1).

Consequences of the event, Excessive Heat Removal Due to Feedwater System Malfunctions, are reduced during the test program because the main feedwater control valves will not be used when the reactor is at power or critical. With flow restricted to the main feedwater bypass valve or auxiliary feedwater system, the maximum flow rate is about 15% of normal flow.

Analysis of the above types of transients indicates that the departure from nucleate boiling (DNB) criterion of the FSAR is met.

Automatic reactor trip and steam line isolation following postulated large steam line breaks which result in uniform depressurization of all loops is provided by low pressure signals from any two steam lines (normally requires coincident high steamline flow signal setpoint set to zero flow). An example is a double-ended break in a main steamline outside of the check and isolation valves. An analysis of this event indicated reactor trip about 15 seconds after the break and no power excursion. The reactor remained subcritical after the trip.

For large steam line breaks upstream of one of the steamline check valves, automatic reactor trip normally would result from the SI signal on high differential pressure between steam lines. However, this signal will be disabled for all tests. Isolation of the broken line for this case is provided by the non-return (floating disc type) valves which require no initiating signal. Reactor trip would be required by operator action based on the operational safety criteria discussed previously. Reactor trip could also occur at the Power Range Neutron Flux low setpoint. However, since the nuclear instrumentation system (NIS) detectors are not completely qualified for steamline break conditions, this flux trip might be delayed or prevented. An analysis of this event, assuming trip on the neutron flux signal, was made for an initial power of 1% RTP, one steam generator isolated and a double-ended break upstream of the steam venturi. The results indicated a reactor trip at about 104 seconds into the transient with a maximum core heat flux of about 5% of the full power value. Transients for which credit was not taken for the neutron flux trip were not analyzed. Since the Evaluation of such transients based upon calculations could lead to fuel damage, VEPCO provided a conservative estimate of the two-hour dose at the site boundary to bound the consequences of this event. The source term inside containment, obtained using the conservative assumptions discussed in Section 8.5 was corrected for the reduction in dose due to containment. The results of this analysis show that the calculated two-hour site boundary thyroid dose would be 9.2 rem.

For steam line breaks outside of containment, automatic protection is available and the accident is bounded by the FSAR results because of the low fission products inventory and is acceptable to the Staff. For steam line breaks inside of containment, corrective operation actions are needed. Close operator supervision during the tests and corrective actions based on the operational safety criteria should be sufficient to prevent significant clad damage. In addition, the bounding dose analysis performed for the postulated accident, which assumed 100% clad failure and other conservatisms, indicate that the offsite dose would be acceptably small.

The consequences of a main feedline rupture would be bounded in the cooldown direction by those for a major break in a main steamline break. Because of low operating power levels and decay heat, the heatup aspects of a feedline rupture are bounded by the FSAR results.

8.3 Loss of Coolant Accidents (LOCA)

The probability of occurrence of a break in the reactor coolant pressure boundary during the Low Power Test Program is very low because of the short time period involved (i.e. about 2-3 weeks). As the result of the low power level and short operating history, the magnitude of clad temperature transients for a LOCA event during the Low Power Test Program would be significantly less than that for the FSAR event because of low decay heat and stored energy in the fuel. In addition, the off-site dose consequences are reduced because of the low fission product inventory.

The system inventory and normal charging flow can provide short-term cooling for very small breaks. VEPCO has estimated that for a postulated 2 inch break, the time to uncover the core would be at least one hour if there were no safety injection. For major breaks in the reactor coolant pressure boundary, the applicant has stated that, even without automatic safety injection, there is sufficient cooling water available to prevent overheating of the fuel rod cladding in the short-term. For a large break the system inventory and cold leg accumulators will have removed sufficient energy to have filled the reactor vessel to the bottom of the nozzles. After system depressurization the water in the reactor vessel is sufficient to keep the core covered for more than one hour.

As the result of the low initial power levels of the test program, the decay heat which must be removed by the ECCS and the corresponding fuel rod surface heat fluxes are very low. For example, assuming reactor operation at 5% power for 1 year prior to the LOCA, the decay heat at one hour after the LOCA would be only 2.5 MW. At this time the maximum fuel rod surface heat flux would be less than 500 Btw/hr-ft^2 and the water needed to be added to the vessel to match boiloff would be about 20 gpm. Because of the limited core operating history prior to and during the Special Low Power Test Program, the actual decay heat load and corresponding surface heat fluxes and coolant in makeup requirements should be much less than the above values.

The staff concludes that the above times are sufficient for the operator to take manual action to initiate safety injection and align the system for long-term cooling.

8.4 Rod Withdrawal and Ejection

8.4.1 Uncontrolled Rod Cluster Control Assembly Rod Withdrawal at Power

Analyses of uncontrolled rod withdrawal were performed assuming natural circulation, starting power of 1% and 5% of full power, and with all steam isolation valves open or two of those closed. A range of reactivity insertion rates up to the maximum for two banks moving was assumed for cases with all steam lines open, and up to the maximum for one bank moving for the cases with steam lines isolated. Both maximum and minimum bounds on reactivity coefficients were investigated. Reactor trip was initiated at 10% nuclear power. These assumptions conservatively bound the test conditions.

The analyses performed show that the rod bank withdrawal at power is a mild transient. Because of the absence of the full complement of normal reactor trips, difficulty of calculating core hydraulic behavior under test conditions, and the paucity of DNB data in the low flow-high pressure regime of the tests, the potential for DNB has not been precluded in the applicant's analysis.

On the basis of the small amount of data and extrapolation of other data, the applicant concludes that DNB is not expected for any rod withdrawal event. We have reviewed the data presented by Westinghouse and additional data by Babcock and Wilcox and data from Bowring. Based on our review of the data we conclude that, at the low flow rates associated with natural circulation, the critical heat flux will be caused by an annular film dryout rather than by a disturbance in a bubbly surface layer, as is usually the case with DNB. In addition, we conclude that, at the low flow rates associated with natural circulation, annular film dryout will not occur until the fluid quality reaches the 80% to 100% range. It appears very unlikely that the fluid quality would approach this range for any of the rod withdrawal events.

Assuming that DNB occurs, however, VEPCO has performed analyses of the clad temperature for the RCCA bank withdrawal at power. The high power range neutron flux trip setpoint is 7% for the test program. To allow for calorimetric errors and normal system errors a trip setpoint is assumed to occur at 10% power. For the worst case, which assumes a low initial downcomer coolant temperature, a trip was assumed to occur at 20% power. The analyses show that the peak clad temperature would be well below 1800^{0}F. In fact, the peak clad temperature would be expected to be approximately 1200^{0}F. We agree that these results indicate a clad temperature excursion resulting in fuel damage is not likely to occur, even if DNB is assumed.

In addition, the bounding dose analyses performed for a hypothetical accident involving 100% clad failure and other conservatisms indicate that the offsite doses would be acceptably small. These analyses therefore include three levels of conservatism and the results are acceptable.

8.4.2 Single Rod Cluster Control Assembly Withdrawal at Power

This accident was not analyzed by the licensee. Although the FSAR analysis is not bounding for the test condition of natural circulation, the low probability of this accident, and the extra surveillance of the operator for uncontrolled control rod motion, power, and hot leg temperature are considered sufficient to eliminate the need for consideration of the consequences of this accident.

In addition, the bounding dose analyses performed for a hypothetical accident involving 100% clad failure and other conservatisms indicate that the calculated offsite doses would be acceptably small even if such an unlikely event were to occur.

8.4.3 Rupture of a Control Rod Drive Mechanism (CRDM)

Limitation of operation of the reactor with control rod withdrawn (Bank D only inserted, to 100 steps withdrawn) make an ejected rod worth less than the delayed neutron fraction, which would result in a transient which is relatively mild compared to those analyzed in the FSAR. We agree with the licensee's conclusion that the consequences are not considered severe enough to warrant analysis of the transient.

In addition, the bounding dose analyses performed for a hypothetical accident involving 100% clad failure and other conservatisms indicate that the off-site doses would be acceptably small.

8.5 Dose Analysis

VEPCO presented the results of calculations of the two hour site boundary doses resulting from a hypothetical accident during the Low Power Test Program which would bound the consequences of Condition II type transients analyzed in the FSAR. The analysis was based on an accident with coincident loss of condenser vacuum which did not involve a break in the primary coolant pressure boundary. The assumptions made in the analysis include:

139 Mwt (5% power)

1.0 micro curie per gram dose-equivalent I-131 RCS activity (technical specification limit) 500 gallons per day (gpd) steam generator leak in each SG (technical specification limit) 100% clad damage and gap activity release

10% iodine/noble gas in gap space

100 DF in steam generators

500 iodine spike factor over steady state

509,000 lb. atmospheric steam dump over 2 hours

1.7 x 10-3 sec/m^3 x/Q percentile value

The results of the analysis show that the two hour site boundary doses would be 5 rem thyroid, 0.9 rem total body and 0.4 rem to the skin.

The staff did not make independent calculations of the dose values because it believes VEPCO's calculated doses are conservative for the following reasons:

1) 100% of the fuel clad is assumed to fail.

   This assumption is conservative for the evaluation performed during a safety review. Typical values for cladding failure are about 10 to 20 percent.

2) Equilibrium radionuclide inventories for operation at 5% power were used to estimate the amount of activity in the core.

   This assumption would be conservative for the expected intermittent and shorter-term operation of the reactor prior to and during the North Anna low power tests.

3) Maximum technical specification values for the primary coolant concentration of iodine plus an iodine spike as a result of the accident.

   This assumption is in addition to the already assumed source of 100% cladding failure and therefore definitely maximizes the amount of iodine available for release or leakage to the secondary system.

4) Condenser vacuum is lost.

   This assumption is normally made for accidents occurring at 100% power. Since the nuclear station is attached to the electrical grid and presumably supplies a significant portion of the base load, a transient resulting in a turbine trip could cause the grid to become unstable with an increased potential for losing the electrical supply. During the low power tests the North Anna Station will not be supplying any power to the grid. Should the nuclear unit have a station transient, offsite power will probably continue as normal and condenser vacuum would not be lost.

5) Maximum technical specification steam generator tube leakage is assumed.

   Since there is always the possibility that even new tubes are defective, it is not possible to exclude steam generator tube leakage entirely. However, past experience suggests that new steam generator tubes do not leak at the technical specification limit. Therefore, a 1 gallon per minute (gpm) leak rate would be conservative for the new steam generators.

6) Meteorology is conservative.

The value for the short term diffusion coefficient ($X/Q=1.7x10^{-3}$ sec/m^3) is larger than the value used by the staff ($X/Q=4.2x10^{-4}$ sec/m^3 - Safety Evaluation Report value) for the consequence estimates contained in the staff safety evaluation report. This adds conservatism to the calculation of the dose estimates.

9.0 ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not authorize a change in effluent types, total amounts or an increase in design power level of 2900 MWt. The test program will not result in any environmental impacts other than those evaluated in the Staff's Final Environmental Statement since the test program is encompassed by the overall activity evaluated in the Final Environmental Statement.

10.0 CONCLUSIONS

The Low Power Test Program for North Anna Unit 2 involves seven tests at low power levels conducted over a short period of time and with a very low fission product inventory.

On the basis of the above considerations, the proposed operational safety criteria and the safety evaluations which include the effects of the exceptions to the Technical Specifications and operation under natural circulation conditions, the staff concludes that the Low Power Test Program will not result in undue risk to public health and safety and is acceptable.

Therefore, we have concluded based on the considerations discussed above, that: (1) it does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. Also, we reaffirm our conclusions as otherwise stated in our Safety Evaluation and its Supplements.

11.0 EMERGENCY OPERATING PROCEDURES

In addition to our requirement that the special low power test program be approved prior to operation above zero power, we stated in Section 1.C.1 of Part II of Supplement No. 10 to the North Anna Power Station, Unit No. 2 Safety Evaluation Report that VEPCO must also revise to our satisfaction emergency operating procedures related to the small break loss-of-coolant accident and inadequate core cooling.

In a letter dated May 30, 1980, VEPCO provided copies of emergency procedures that had been revised to reflect the analysis of small break loss-of-coolant accidents and inadequate core cooling in accordance with license condition 2D(6)a. and Task Action Plan (NUREG-0660) item I.C.1. The emergency procedures submitted by VEPCO have been reviewed by the NSSS supplier, Westinghouse Electric Corporation, and changes recommended by Westinghouse have been incorporated in compliance with Task Action Plan item I.C.7(a).

The staff has reviewed VEPCO's emergency procedures and has recommended some changes to VEPCO. VEPCO has made the recommended changes and is continuing with safety committee approval of the changes and operator training. The staff will observe a simulation of the emergency conditions conducted by North Anna Unit No. 2 personnel and a walk-through of at least one emergency procedure in the North Anna Unit No. 2 control room. We have concluded that the emergency procedures are adequate to support operation up to 5% power for training during low power testing. Prior to operation above 5% power we will evaluate the results of the procedure walk-throughs and ensure that the licensee has made any necessary procedural changes.