REGULATORY DOCKET FILE COPY

AUG 22 1980

Docket No.: 50-311

Mr. R. L. Mittl, General Manager
Licensing and Environment
Engineering and Construction Department
Public Service Electric and Gas Company
80 Park Place
Newark, New Jersey 07101

Dear Mr. Mittl:

SUBJECT: ISSUANCE OF AMENDMENT NO. 2 TO LICENSE NO. DPR-75
SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

The Nuclear Regulatory Commission (the Commission) has issued Amendment No. 2 to License DPR-75 (Enclosure 1) in accordance with your letter, dated August 7, 1980 requesting changes to the Salem Nuclear Generating Station, Unit No. 2 Appendix A Technical Specifications for an interim period to permit you to perform the special low power test program required by Conditions 2.C(6)b and c of License No. DPR-75. In an attachment to your letter of August 7, 1980, you also provided a safety evaluation to support performing the low power test program. Operating procedures for conducting this program were also provided in an attachment to your letter dated August 7, 1980.

We have reviewed the above information and have concluded that these changes to the Technical Specifications for conducting low power testing and PSE&G's test procedures for low power testing are acceptable and can be performed without posing an undue risk to the public. Our Safety Evaluation regarding this matter is presented in Enclosure 2.

This amendment authorizes the Public Service Electric and Gas Company to conduct the special low power test program as defined in the Safety Evaluation and in Appendix A Technical Specification 8.16. Enclosure No. 3 is a copy of the Federal Register Notice of Issuance of Amendment No. 2 to License No. DPR-75.

The Commission staff is reviewing emergency operating procedures with respect to Condition 2.C(6)a for small break loss-of-coolant accident and inadequate core cooling. We have concluded that the presently available emergency operating procedures are acceptable for operation at power levels not exceeding five percent. Our evaluation of this matter is presented in Enclosure 2.

Our Office of Inspection and Enforcement has advised us that matters related to Condition 2.C(6)d of License DPR-75 related to operation above zero power have been satisfactorily completed. Therefore, we consider these matters resolved.

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CP

OFFICE ▶						
SURNAME ▶	8009150106					
DATE ▶						

Based on the above we have determined that items required to be completed prior to conducting the low power test program for Salem Nuclear Generating Station, Unit 2 have been satisfactorily resolved, and therefore, operation at power levels not exceeding five percent is permitted.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 2 to DPR-75 with Technical Specification page change.
2. Safety Evaluation for Special Low Power Test Program
3. Federal Register Notice

cc w/enclosures: See next page

DL:LB#3
Schwencer
8/21/80

OELD
J. Moore
8/21/80

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

AUG 22 1980

Docket No.: 50-311

Mr. R. L. Mittl, General Manager
Licensing and Environment
Engineering and Construction Department
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80 Park Place
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Based on the above we have determined that items required to be completed prior to conducting the low power test program for Salem Nuclear Generating Station, Unit 2 have been satisfactorily resolved, and therefore, operation at power levels not exceeding five percent is permitted.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 2 to DPR-75 with Technical Specification page change.
2. Safety Evaluation for Special Low Power Test Program
3. Federal Register Notice

cc w/enclosures: See next page

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New Jersey Board of Public Utilities
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 22 1980

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY AND
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO LICENSE

Amendment No. 2
License No. DPR-75

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

   A. The application for amendment by Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated August 7, 1980 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by the addition of Section 8.16 to Appendix A of the Technical Specifications. This addition permits Public Service Electric and Gas Company to perform the special low power test program identified in Conditions 2.C(6)b and c of License No. DPR-75. This license is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 2, are hereby incorporated into the license. PSE&G shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

H. R. Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Page 8-3 to the Technical
Specifications (Appendix A)

Date of Issuance: AUG 22 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 2

LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

Page

8-3

8.11 Prior to startup following the first regularly scheduled refueling outage, the licensees shall install a second level of under-voltage protection for the emergency buses.

8.12 Prior to startup following the first regularly scheduled refueling outage, the licensees shall add a fuse in series with the primary device of each one of 12 circuits fed from 230 volt ac motor control centers to provide backup protection for reactor containment electrical penetrations. Each fuse shall be located in an independent compartment in the control center of the present primary device.

8.13 Prior to startup following the first regularly scheduled refueling outage, the licensees shall submit for our approval the results, which are applicable to Salem Unit 2, of tests to study mixing of added borated water and cooldown under natural circulation conditions.

8.14 Prior to exceeding 50 percent power, the licensees shall complete the preoperational testing of the remaining three of six circulators to be tested in the main condenser for the circulating water system.

8.15 The licensees shall also report for the Salem facility any information reported for the Hope Creek facility relating to circumstances which suggest that the risk from flammable gas clouds (resulting from river traffic accidents on the Delaware River) may increase to unacceptable levels.

8.16 For the conducting of the low power test program only, the licensees have been granted an exception from the requirements of those Technical Specifications identified in Table 6.1 of our Safety Evaluation dated August 22 1980, related to the Special Low Power Test Program.

AUG 22 1980

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

SPECIAL LOW POWER TEST PROGRAM

FOR

SALEM GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

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1.0 INTRODUCTION

In Section 1.G of Part II of Supplement No. 4 to the Safety Evaluation Report for Salem Nuclear Generating 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 PSE&G letters of February 8, 1980 and March 31, 1980.

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

The specific tests proposed by PSE&G 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 PSE&G 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 11, 1980, the NRC staff advised PSE&G that Test 8 may be deleted if training for each operator is provided by conducting a simulation of the event on a simulator. PSE&G was also advised in the June 11, 1980 letter that Test 9B may be modified and deferred until completion of the power ascension program and manufacturer's acceptance test. We require that in lieu of performing test 9B during the low power test program, PSE&G perform a similar test using decay heat instead of performing it with the reactor critical. Use of decay heat eliminates many of the special conditions required for test 9B, thus reducing the risks associated with performing this test. Test 9A is required to provide recalibration of the nuclear instrumentation to compensate for the lowered primary system temperature. Test 9A is incorporated into the low power test program by use of caution statements in the test procedures. The caution statements stress the need to adjust excore NIS calibrations to compensate for temperature changes in the downcomer.

On August 7, 1980, PSE&G submitted test procedures for the seven remaining tests. This submittal also included the safety analysis and technical specification exceptions necessary to conduct these tests. PSE&G 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 procedures.

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

During our review of Virginia Electric and Power Company's (VEPCO) low power test program which was conducted at the North Anna Power Station, Unit No. 2, 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" was discussed with the NSSS vendor, Westinghouse, and with

VEPCO. As a result of these discussions, VEPCO, in a letter dated June 5, 1980, requested that these tests be modified or deleted from the special test program. VEPCO stated that there was 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 agreed 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. VEPCO proposed combining test 9A with test 4 so that sufficient data was obtained for conducting the test program.

We considered the VEPCO request to delete tests 8 and 9B and concluded that test 8 could be deleted and a similar test to 9B could be performed using decay heat near the end of the startup test program for North Anna 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 sub-critical. The same training benefits will be derived as if the tests 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.

We believe that the justification for deletion of test 8 and deferral of test 9B at North Anna, Unit 2 also applies to Salem, Unit 2. We informed PSE&G of our decision on this matter in a letter dated June 11, 1980. We require that in lieu of performing test 9B during the low power test program, PSE&G perform a similar test using decay heat instead of performing it with the reactor critical. This test should be performed as part of PSE&G'S normal startup test program.

The tests described above have recently been completed at both the Sequoyah Unit 1 and North Anna Unit 2 facilities. The special low power testing programs at both facilities have satisfied all NRC requirements. The results provided meaningful information on plant response, demonstrated natural circulation heat removal capability, provided base line data for the specific plant characteristics, and provided supplemental training for the operating crews. We expect similar results for Salem Unit 2.

3.0 REVIEW OF THE TEST PROCEDURES

Westinghouse reviewed the test procedures and provided comments which PSE&G incorporated. The staff 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 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 reviewed the procedures for the low power tests and provided comments to PSE&G. These comments were resolved in a meeting on August 6, 1980, with PSE&G and Westinghouse representatives. Revised test procedures were submitted in an attachment to a letter from PSE&G dated August 7, 1980. Our comments were appropriately incorporated in the revised procedures. The only significant difference between the Sequoyah 1 and North Anna 2 programs and the Salem 2 proposed program is that the simulated loss of all onsite and offsite power (Test 7), will be performed at Salem using heat from the reactor coolant system pumps to simulate decay heat; the test performed on Sequoyah Unit 1 and North Anna Unit 2 used fission heat to simulate decay heat. The NRC staff discussed the use of pump heat as a heat source for Test 7 on March 10 and March 13, 1980, and on March 17, 1980 issued a letter (Olan Parr to Mr. R. L. Mittl, PSE&G) stating tentative agreement that use of RCS pump heat to simulate decay heat is acceptable provided the test: (1) adequately simulate system thermodynamic response; and (2) minimum AC power is used. Our review of the procedure for Test 7 indicates that the test will result in a reasonable simulation of plant response to a loss of all AC for the purpose of operator training and that the use of the required AC to operate RCS pumps and essential RCS pump support functions will not preclude meeting objectives for the test. The staff believes that sufficient plant specific information on natural circulation will be obtained during the other seven special tests.

Based on our review of the Salem Unit 2 test procedures, we have concluded that the special low power test program will meet all the stated test objectives and can be safely performed at Salem Unit 2. NRC representatives will witness selected parts of the special tests as necessary to ensure that the safety precautions and acceptance criteria are met.

4.0 EXCEPTIONS TO TECHNICAL SPECIFICATIONS

Exceptions to a number of technical specification requirements for Salem 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 4.1 for each of the tests in the Special Low Power Test Program and are discussed below.

4.1 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 functiojs 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 17 percent of the narrow range span will be reduced to 5 percent 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 downsteam of the steam line check valves.

The reactor trip resulting from the SI signal caused by high differential pressure between steam lines will be disabled.

TABLE 4.1

EXCEPTIONS TO TECHNICAL SPECIFICATIONS FOR LOW POWER TEST PROGRAM

		TEST						
	TECHNICAL SPECIFICATION	1	2	3	4	5	6	7
2.1.1	Core Safety Limits	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
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
3.3.2	Safety Injection - All automatic functions	X	X	X	X	X		X
3.4.4	Pressurizer			X		X		X
3.7.1.2	Auxiliary Feedwater		X					X
3.8.1.1	AC Power Sources		X					X
3.8.2.1	AC Onsite Power Distribution System		X					X
3.8.2.3	DC Distribution System		X					X
3.10.3	Special Test Exception Physics Tests				X			

X -- Exceptions Required

4.2 Other Exceptions To Technical Specifications

a. T.S. 2.1.1, "Reactor Core Safety Limits", gives limits to the average 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 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°F by T.S. 3.1.1.5, "Minimum Temperature for Criticality", and to 531°F by T.S. 3.10.3, "Special Test Exceptions - Physics Tests. During Test 4 it is expecte' that the average reactor coolant temperature will drop below these limits. Westinghouse has stated that operation with the average reactor coolant temperatures as low as 485°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 percent thermal power (RTP) to 7 percent RTP.

   These restrictions reduce the consequences of transients involving individual rod withdrawal 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 tempertaure coefficient at near zero values and limiting the maximum power during power excursions.

   The trip setpoint of 7 percent RTP is based on a coolant temperature in the reactor vessel downcomer region of about 545°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 percent RTP. This effect was included in the safety analysis by using a conservative estimate of 1 percent reduction in the excore detector reading per °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.

It should be noted that the tests at Sequoyah and North Anna indicated that the actual reduction in the ex-core detector reading is less than 1/2 percent per °F.

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.

d. T.S. 3.7.1.2 "Auxiliary Feedwater" requires at least three steam generator auxiliary feedwater pumps be operable in modes 1, 2 and 3.

Tests numbers 7, "Loss of all onsite and offsite AC" requires that both electric driven auxiliary feedwater pumps be electrically isolated from their power source. This is acceptable because the test will be conducted using only pump heat and with the reactor sub-critical. In the event the electrically driven auxiliary feed pumps are needed, electric power can be restored to them very quickly by closing the supply breaker.

e. T.S. 3.8.1.1 "AC Power Sources", 3.8.2.1 "AC Onsite Power Distribution System" and 3.8.2.3 "DC Distribution System" specify the minimum A.C. electrical power sources, A.C. electrical bases and D.C. bus trains required for operation in modes 1, 2, 3 and 4. During the conduct of test 7 "Simulated loss of all onsite and offsite AC" all AC power sources, including emergency diesel generators, certain AC buses and all three battery chargers will be electrically isolated. During test number 2, "Simulated loss of offsite power", the offsite feed breakers will be opened.

This is acceptable because of the low power levels involved and because all power can be restored quickly if needed by closing the feed breakers and/or starting the diesel generators.

5.0 OPERATIONAL SAFETY CRITERIA

As the result of a safety evaluation of the Low Power Test Program at Salem Unit 2, a set of operational safety criteria have been specified for test conditions (see Table 5.1) 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 5.1

OPERATIONAL SAFETY CRITERIA

1. Guidelines for All Tests

a)	Primary System Sub-Cooling (T_{sat} Margin)	> 20°F
b)	Steam Generator Water Level	> 30% Narrow Range Span
c)	Pressurizer Water Level	
	(1) With RCPs running	> 22% Span
	(2) Natural Circulation	≥ Value when RCPs tripped
d)	Loop ΔT	≤ 65°F
e)	T_{avg}	≤ 580°F
f)	Core Exit Temperature (highest)	≤ 610°F
g)	Power Range Neutron Flux Low Setpoint and Intermediate Range Neutron Flux Reactor Trip Setpoints	≤ 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)	≤ 15°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°F
f)	T_{avg}	> 578°F
g)	Core Exit Temperature (highest)	> 610°F
i)	Uncontrolled rod motion	

Table 5.1 (Continued)

j) Control Bank D less than 100 steps withdrawn

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°F
b)	Steam Generator Water Level	< 0% Narrow Range Span or equivalent wide range level
c)	Containment Pressure	$\geq$ 4.7 psig
d)	Pressurizer Water Level	< 10% Span or an unexplained decrease of more than 10% not concurrent with a T change.
e)	Pressurizer Pressure	Decreases by 200 psi or more in an unplanned or unexplained manner.

The staff had 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.

However, after performance of the Special Low Power Test Programs at North Anna and Sequoyah, Westinghouse has concluded that the use of core exit thermocouples and wide range loop RTDs are acceptable for determination of the margin to saturation temperature under natural circulation flow conditions. The average core exit thermocouple temperature agreed with the average of the wide range loop RTD measurements of hot leg temperature to within 1°F for both plants.

6.0 SAFETY EVALUATION

6.1 Introduction

PSE&G 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 the containment.

The results of the analyses of Condition II, III and IV events are categorized in Table 6.1 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 6.2 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 6.1

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	1
RCCA Bank With., at power	4	4	4	4	4	1	1
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	1
Loss Offsite Power	1	1	1	1	1	1	1
Excessive Feedwater	2	2	2	2	2	1	1
Excessive Load	2	2	2	2	2	1	1
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	1
Single RCCA Withdrawal	4	4	4	4	4	1	1
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	1
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	1

TABLE 6.2

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 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.
Excessive Load Increase Incident	Turbine not operating. Load control limited to single steam dump valve or relief valves.
Spurious Operation of Safety 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.

TABLE 6.2 (Continued)

EVENT	REASON WHY CONSEQUENCES BOUNDED BY FSAR
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 Sub-critical 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.

6.2 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. 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 percent 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 required coincident high steamline flow signal setpoint set to zero flow). An example is a double-ended break in a main steamline downsteam of the flow restrictor with all steamline isolation valves initially open. 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.

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.

6.3 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 offsite 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. Westinghouse has estimated that for a postulated 2 inch break, the time to uncover the core would be at least a 6000 seconds, 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 percent 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 BTU/hr-ft 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.

6.4 Rod Withdrawal and Ejection

6.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition

Operation of the reactor without coolant pumps, and in some cases, a slightly positive moderator temperature reactivity coefficient, tends to make the consequences of rod cluster control assembly (RCCA) bank withdrawal worse than with the operating conditions assumed in the FSAR. For this reason the operating procedures require that following any reactor trip at least one of the reactor coolant pumps will be restarted and the reactor boron concentration adjusted so that the reactor will not go critical with less than 100 steps withdrawal of bank D.

An analysis was performed by Westinghouse for uncontrolled RCCS bank withdrawal using the FSAR methods but with conservative assumptions for the conditions of Test No. 8. These are:

1. Reactor Coolant flow was 0.1 percent of nominal.

2. Control rod incremental worth and total worth were upper bound values for the D bank 100 steps withdrawn.

3. Moderator temperature reactivity coefficient was an upper bound (positive) for any core average temperature at or above 485°F.

4. The lower bound for that delayed neutron fraction for the beginning of life for cycle 1 was used.

5. Reactor trip was initiated at 10 percent of full power.

6. DNB was assumed to occur instantaneously at the hot spot, at the beginning of the transient.

The Westinghouse analysis indicates that the clad temperature would not exceed 1300°F, even when a very low heat transfer coefficient of 2 BTU/hr-ft^2 -°F was used. We agree that clad failure is unlikely at this temperature.

In addition, the bounding dose analyses performed for a hypothetical accident involving 100 percent clad damage and other conservatisms indicate that the offsite doses would be acceptably small. These analyses therefore include several degrees of conservatism and are acceptable.

6.4.2 Uncontrolled Rod Cluster Control Assembly Rod Withdrawal at Power

Analyses of uncontrolled rod withdrawal were performed assuming natural circulation, starting power of 1 percent and 5 percent 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 percent 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 percent to 100 percent 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, PSE&G has performed analyses of the clad temperature for the RCCA bank withdrawal at power. The high power range neutron flux trip setpoint is 7 percent for the test program. To allow for calorimetric errors and normal system errors a trip setpoint is assumed to occur at 10 percent power. In fact, the peak clad temperature would be expected to be approximately 1300°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 percent 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.

6.4.3 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 percent clad failure and other conservatisms indicate that the calculated offsite doses would be acceptably small even if such an unlikely event were to occur.

6.4.4 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 percent clad failure and other conservatisms indicate that the off-site doses would be acceptably small.

6.5 Dose Analysis

PSE&G 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:

170 Mwt (5 percent power)

1.0 micro curie per gram dose-equivalent 1.131 RCS activity (technical specification limit)
500 gallons per day (gpd) steam leak in each SG (technical specification limit)
100 percent clad damage and gap activity release

10 percent 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 PSE&G's calculated doses are conservative for the following reasons:

1) 100 percent 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 percent 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 percent 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 occuring at 100 percent 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 Salem Plant 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 techincal 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 wuld be conservative for the new steam generators.

6) Meteorology is conservative.

The value for the short term diffusion coefficient ($X/Q=1.7 \times 10^{-3}$ sec/m^3) is larger than the value used by the staff ($X/Q=4.2 \times 10^{-4}$ sec/m^3 - Safety Evaluation Report value) for the consequences estimates contained in the staff safety evaluation report. This adds conservatism to the calculation of the does estimates.

7.0 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. 4 to the Salem Nuclear Generating Station, Unit No. 2 Safety Evaluation Report that PSE&G must also revise to our satisfaction emergency operating procedures related to the small break loss-of-coolant accident and inadequate core cooling.

PSE&G is revising the emergency procedures to reflect the analysis of small break loss-of-coolant accidents and inadequate core cooling in accordance with license condition 2C(6)2. and Task Action Plan (NUREG-0660) item 1.C.1. They are incorporating changes suggested by the NRC and by the NSSS supplier, Westinghouse Electric Corporation. PSE&G will obtain their safety committee's approval of the changes, implement all necessary changes and train their operators accordingly. The staff will observe a walk-through of at least one emergency procedure in the Unit 2 control room prior to operation at greater than 5 percent power. The NRC will also observe the Salem operators perform these emergency procedures on a simulator.

We have concluded that based on the low levels of residual heat in the reactor core that will result from operation below 5 percent power, complete implementation of these procedures will not be necessary for this low power operation and that the present emergency procedures are adequate to support operation up to 5 percent power.

8.0 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 3423 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.

9.0 CONCLUSIONS

The Low Power Test Program for Salem Nuclear Generating Station, Unit 2 involves seven tests at low power levels conducted over a short period of time and with a very low fission product inventory. Similar test have been conducted at both the Sequoyah, Unit 1 and North Anna, Unit 2 facilities.

On the basis of the above considerations, the proposed operational safety criteria and the safety evaluations which include the effets 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.

Dated: AUG 22 1980

AUG 22 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-311

PUBLIC SERVICE ELECTRIC AND GAS COMPANY,
PHILADELPHIA ELECTRIC COMPANY,
DELMARVA POWER AND LIGHT COMPANY, AND
ATLANTIC CITY ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 2 to License No. DPR-75, issued to Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees), which revised Technical Specifications for operation of the Salem Nuclear Generating Station, Unit No. 2 (the facility) located in Salem County, New Jersey. The amendment is effective as of the date of issuance.

The amendment permits Public Service Electric and Gas Company to conduct the special low power test program as described in our related Safety Evaluation concerning a Special Low Power Test Program.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. The activity authorized by the amendment is encompassed by the overall action involving the proposed issuance of an operating license for which prior public notice was issued in the Federal Register on October 20, 1972 (37 F.R. 22637).

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated August 7, 1980, (2) Amendment No. 2 to License No. DPR-75, and (3) the Commission's related Safety Evaluation concerning a Special Low Power Test Program. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Salem Free Public Library, 112 West Broadway, Salem, New Jersey. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 22ND day of AUGUST, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Acting Chief
Licensing Branch No. 3
Division of Licensing