Docket File

SEPTEMBER 1 4 1979

Docket No. 50-261

REGULATORY DOCKET FILE COPY

Mr. J. A. Jones, Senior Vice President
Carolina Power and Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 42 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications and is in response to your request dated December 22, 1977.

The amendment consists of additions to the Technical Specifications which incorporate the proposed low temperature overpressure protection system into the Limiting Conditions for Operation and Surveillance Requirements.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 42 to DPR-23
2. Safety Evaluation
3. Federal Register Notice

cc: w/enclosures
See next page

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

September 14, 1979

Docket No. 50-261

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Senior Vice President
Carolina Power and Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

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See next page

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

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. DPR-23

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

A. The application for amendment by Carolina Power and Light Company (the licensee) dated December 22, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and 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 changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-23 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 42 , are hereby incorporated in the license. The licensee 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: September 14, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 42

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment Number and contain vertical lines indicating the area of change.

Remove	Insert
3.1-1	3.1-1
3.1-4	3.1-4
3.1-5	3.1-5
3.1-6	3.1-6
3.1-7	3.1-7
3.1-8	3.1-8
3.3-5	3.3-5
4.1-6	4.1-6
4.1-10	4.1-10
6.9-8	6.9-8

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those Reactor Coolant System conditions which must be met to assure safe reactor operation.

Specification

3.1.1 Operational Components

3.1.1.1 Coolant Pumps

a. At least one reactor coolant pump or the Residual Heat Removal System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

b. When the reactor is critical, except for special low power tests during initial start-up testing, at least one reactor coolant pump shall be in operation.

c. Reactor power shall not exceed 10% rated power unless at least two reactor coolant pumps are in operation.

d. Reactor power shall not exceed 45% of rated power with only two pumps in operation.

e. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer or the steam generator temperature is no higher than 50°F higher than the temperature of the reactor coolant system.

3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2, and are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2. This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2 may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1 is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

d. The overpressure protection system shall be operable whenever the RCS temperature is below 350°F and not vented to the containment. One PORV may be inoperable for seven days. If the inoperable PORV has not been returned to service within 7 days, or if at any time both PORVs become inoperable, then one of the following actions should be completed within 12 hours:

   1. Cooldown and depressurize the RCS or
   2. Heatup the RCS to above 350°F.

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as required in Section 6.9.3.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

3.1.2.3 The pressurizer shall neither exceed a maximum heatup rate of 100°F/hr nor a cooldown rate of 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

3.1.2.4 Figures 3.1-1 and 3.1-2 shall be updated periodically in accordance with the following criteria and procedures before the calculated exposure of the vessel exceeds the exposure for which the figures apply.

a. At least 60 days before the end of the integrated power period for which Figures 3.1-1 and 3.1-2 apply, the limit lines on the figures shall be updated for a new integrated power period utilizing methods derived from the ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda, Non-Mandatory Appendix G. These limit lines shall reflect any changes in predicted vessel neutron fluence over the integrated power period or changes resulting from the irradiation specimen measurement program.

b. The results of the examinations of the irradiation specimens and the updated heatup and cooldown curves shall be reported to the Commission within 90 days of completion of the examinations.

<u>Basis</u>

The ability of the large steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as ASTM A302 Grade B parent material of the H. B. Robinson Unit No. 2 reactor pressure vessel are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility under certain conditions of irradiation. In pressure vessel material, the most serious mechanical property change is the reduction in the upper shelf impact strength. Accompanying the decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

A method for guarding against fast fracture in reactor pressure vessels has been presented in Appendix G, "Protection Against Non-Ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of: 1) the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or 2) the temperature 60°F less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The value of RT_{NDT}, and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel still can be monitored by a surveillance program such as the Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program[1] where a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. The increase in the Charpy V-notch 50 ft-lb temperature (Δ RT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + RT_{NDT}) is utilized to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods (2) derived from Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the I.D. position. The thermal gradients induced during cooldown tend to produce tensile stresses at the I.D. location and compressive stresses at the O.D. position. Thus, the I.D. flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The overpressure protection system consists of two operable pressurizer Power Operated Relief Valves (PORV's) connected to the station instrument air system, a backup nitrogen supply, and associated electronics.

References:

1. S.E. Yanichko, "Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program, "Westinghouse Nuclear Energy Systems - WCAP-7373 (January, 1970)

2. E. B. Norris, "Reactor Vessel Material Surveillance Program for H. B. Robinson Unit No. 2, Analysis of Capsule V,"Southwest Research Institute - Final Report SWRI Project No. 02-4397.

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated, and in addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures.

The safety injection pump power supply breakers must be racked out when the reactor coolant system temperature is below 350°F and the system is not vented to containment atmosphere.

3.3.2 Containment Cooling and Iodine Removal Systems

3.3.2.1 The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

a. The spray additive tank contains not less than 2505 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

b. Two containment spray pumps are operable.

c. Four fan cooler units are operable.

d. All essential features, including valves, controls, dampers, and piping associated with the above components are operable.

e. The system which automatically initiates the sodium hydroxide addition to the containment spray simultaneously to the actuation of the containment spray is operable.

TABLE 4.1-1 (Continued)

	Channel Description	Check	Calibrate	Test	Remarks
21.	Containment Sump Level	N.A.	R	N.A.	
22.	Turbine Trip Set Point**	N.A.	R	R	
23.	Accumulator Level and Pressure	S	R	N.A.	
24.	Steam Generator Pressure	S	R	M	
25.	Turbine First Stage Pressure	S	R	M	
26.	Emergency Plant Portable Survey Instruments	M	R	M	
27.	Logic Channel Testing	N.A.	N.A.	M(1)	(1) During hot shutdown and power operations. When periods of reactor cold shutdown and refueling extend this interval beyond one month, the test shall be performed prior to startup.
28.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
29.	4 Kv Frequency	N.A.	R	R	
30.	Control Rod Drive Trip Breakers	N.A.	N.A.	M	
31.	Overpressure Protection System	N.A.	R	M	

**Stop valve closure or low EH fluid pressure

S	-	Each Shift	M	-	Monthly
D	-	Daily	Q	-	Quarterly
W	-	Weekly	P	-	Prior to each startup if not done previous week
B/W	-	Every two weeks	R	-	Each Refueling Shutdown
A/R	-	After each refueling startup	N.A.	-	Not applicable

c.	Fuel Inspection	2.1	Upon completion of the inspection at second and third refueling outages
d.	Inservice Inspection	4.2	After five years of operation
e.	Containment Sample Tendon Surveillance	4.4	Upon completion of the inspection at 5 and 25 years of operation
f.	Post-operational Containment Structural Test	4.4	Upon completion of the test at 3 and 20 years of operation
g.	Fire Protection System	3.14	As specified by limiting condition for operation.
h.	Overpressure Protection System Operation	3.1.2.1e	Within 30 days of operation.

TABLE 4.1-3 (Continued)

		Check	Frequency	Maximum Time Between Tests
13.	Turbine Inspection	Visual, Magnaflux and Die Penetrant	Every five years	6 years
14.	Fans and Associated Charcoal and Absolute Filters for Control Room and Residual Heat Removal Compartments	Fans functioning. Charcoal and absolute filter efficiencies checked $\geq 99\%$ for Iodine and 0.3 Micron Particulate. DOP Test on absolute filters. Freon Test on Charcoal Filter Units	Each refueling shutdown	NA
15.	Isolation Seal Water System	Functioning	Each refueling shutdown	NA
16.	Overpressure Protection System	Functioning	Each refueling shutdown	NA

*NA - Not applicable

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

INTRODUCTION

By letter to the Carolina Power and Light Company (the licensee) dated August 11, 1976, the NRC requested an evaluation of the H. B. Robinson Steam Electric Plant, Unit No. 2, system designs to determine susceptibility to overpressurization events, an analysis of the possible events and proposed interim and permanent modifications of systems and procedures to reduce the likelihood and consequences of such events. By letter dated September 8, 1976 and subsequent letters (see references) the licensee submitted the information we requested including the administrative operating procedures, the proposed low temperature overpressure protection system (OPS), and Technical Specifications. The hardware includes sensors, actuating mechanisms, alarms and valves to prevent a reactor coolant system transient from exceeding the pressure and temperature limits of the Technical Specifications for Robinson 2, as required by Appendix G to Chapter 10, Code of Federal Regulations, Part 50 (10 CFR 50).

BACKGROUND

Over the last few years, incidents identified as pressure transients have occurred in pressurized water reactors. As used in this report "pressure transients" refers to events during which the temperature-pressure limits of the reactor vessel, as shown in the Technical Specifications, are exceeded. All of these incidents occurred at relatively low temperatures (less than 200°F) where the reactor vessel material toughness (resistance to brittle failure) is reduced from that which exists at normal operating temperatures.

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The "Technical Report on Reactor Vessel Pressure Transients" in NUREG-0138 (Reference 14) summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to resolve this issue by reducing the likelihood of future pressure transient events at operating reactors. A brief discussion is presented here.

Reactor vessels are constructed in accordance with the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at normal reactor operating pressure and temperature conditions. However, these steels are less tough if subjected to relatively high pressures at low temperatures. Thus, restrictions are placed on the pressure levels during startup and shutdown operations.

At operating (hot) temperatures, the pressures allowed by Appendix G limits are in excess of the setpoint of currently installed pressurizer code safety relief valves. However, most operating PWRs did not have automatic pressure relief device setpoints to prevent lower level pressure transients during cold conditions (startup and shutdown) from exceeding the Appendix G limits when the reactor coolant system (RCS) is water-solid and non-vented.

By letter dated August 11, 1976 (Reference 1), we requested that the licensee begin efforts to design and install plant systems to mitigate the consequences of pressure transients at low temperatures. We also requested that operating procedures be examined and administrative changes be made to prevent initiating overpressure events. We also concluded that interim administrative controls should be imposed to assure safe operation until the proposed overpressure mitigating hardware could be installed.

The licensee responded (References 2, 3 and 4) with preliminary information describing interim measures to prevent these transients along with some discussion of a proposed OPS hardware change. The change, which has been made, was to install a low pressure (400 psig) automatic actuation setpoint on the existing pressurizer power operated relief valves (PORV) for operation at all temperatures below 350°F whenever the RCS is water-solid and non-vented.

The licensee participated as a member of a Westinghouse user's group which was formed to support the analytical effort required to verify the adequacy of the OPS. Using input data supplied by the user's group, Westinghouse performed transient analyses (Reference 15) which were used as the bases for plant specific calculations.

We requested additional information concerning the proposed procedural and OPS hardware changes (References 5 and 6) and obtained licensee responses (References 7, 8 and 9). After subsequent telephone conversations and meetings, the licensee submitted plant specific analyses and further supporting information (References 11, 12, 17 and 18). Also, the licensee proposed supporting OPS Technical Specifications (Reference 13).

The purpose of this safety evaluation is to document the basis for our approval of the licensee's OPS and associated Technical Specifications.

The proposed overall approach to eliminating overpressure events incorporates administrative, procedural and hardware controls with reliance upon the plant operator for the principal line of defense. Preventive administrative and procedural measures include: (1) procedural precautions; (2) de-energization of components during cold shutdown; (3) avoidance of water solid reactor coolant system whenever possible; and (4) addition of an OPS which incorporates an automatically actuated setpoint low pressure relief using the existing power operated relief valves (PORV).

The design basis criteria applied in determining the acceptability of the electrical, instrumentation and control aspects of the added low temperature OPS are:

Operator Action: Except for enabling the OPS (as required when RCS temperature falls below 350°F), no credit can be taken for operator action until ten minutes after the operator is aware of a transient.

Single Failure Criterion: The low temperature OPS shall be designed to protect the reactor vessel given a single failure in addition to the event that initiated the pressure transient.

Testability: The OPS must be testable on a periodic basis to assure operability.

Seismic and IEEE Std-279 Criteria: The OPS should satisfy seismic Category I and IEEE Std-279 criteria. The basic objective is that the system not be vulnerable to a failure that could both initiate a pressure transient and disable the OPS. Such events as loss of instrument air and loss of offsite power must be considered.

EVALUATION

System Description

The licensee has adopted the "Reference Mitigating System" concept developed by Westinghouse and the user's group and has supplemented the actuation circuitry of the two existing pressurizer PORVs to provide a constant, automatic low pressure relief setpoint at 400 psig during startup and shutdown conditions. The new actuation circuitry uses multiple pressure sensors, power supplies and logic trains to improve reliability.

A keylock switch allows an operator to manually enable and disable the low pressure setpoint of each relief valve whenever the RCS temperature is below 350°F. Enabling alarms are provided which monitor the RCS temperature, the position of both keylock switches, and the position of the PORV upstream MOVs. Each alarm also serves as an "OPS actuation" alarm by annunicating if the associated OPS channel trips (i.e., P_{RCS}<400 psig). Separate enabling/actuation alarms are provided for the channels to maintain separation of redundant channels.

The existing PORVs are spring closed and air (or nitrogen, N_2) opened gate valves. Each PORV receives actuating air from the existing plant instrument air system or the newly added backup N_2 supply system. The single plant instrument air header supplies both PORVs. The backup N_2 supply system has two N_2 accumulators, one for each PORV, and a bank of 2000 psi N_2 bottles connected to a single header that supplies both PORVs. Each accumulator contains enough N_2 for approximately ten minutes of PORV operation (about 100 cycles) without operator action during the most limiting transient, assuming a loss of both the plant instrument air system and the backup 2000 psi N_2 bottles. Check valves and pressure regulators are arranged such that the primary source of actuating air is the plant instrument air system. Pressure instruments (four), monitor the supply of air and N_2 to each PORV, and low pressure alarms are installed in the control room to alert the operator to a malfunction in the system. We find the PORV normal and backup actuation air supply systems are acceptable.

Operator Action

Operator awareness of the overpressure transient will be by the low temperature overpressure transient alarm. No credit has been taken until 10 minutes later. We find this acceptable.

Single Failure

Redundant pressure protection channels are used to satisfy the single failure criterion. The design basis for the long-term mitigating system is such that a single failure is considered in addition to the single event that initiates the overpressurization incident. This single failure is assumed to apply to the sensing and actuation mechanisms and pressure relief devices.

Since redundant pressure and temperature sensors and associated electronics, two PORV's and isolation valves, and independent power sources are provided for the long-term overpressure mitigating system, we find that the long term mitigating system meets the single failure criterion and is, therefore, acceptable.

Testability

The program for testing the PORV's for low pressure protection system operability resulted in an NRC position that the control circuitry from pressure sensor to the valve solenoid should be tested prior to each heatup or cooldown and the PORV's should be stroked during each refueling. The testibility program for the Robinson 2 PORV's will be as follows:

a. Verification of upstream isolation valves functioning once per cold shutdown.

b. Performance of a Channel Functional test of the control circuitry from the pressure sensor to the valve solenoid to be conducted once per refueling outage.

c. Performance of a Channel Calibration of the pressurizer pressure sensors once per 18 months.

Verification of operability is possible prior to a water-solid system, low temperature operation by use of the remotely operated isolation valves, enable/disable switches, and normal electronics surveillance methodology.

Plant tests during cold shutdown could potentially result in RCS overpressurization above the minimum operating limit curves unless systems are properly aligned. These tests are the integrated emergency core cooling system test, the temperature loop calibrations for the letdown isolation detectors, and the periodic surveillance tests of the charging pumps.

Overpressurization resulting from these tests will be prevented through the use of special valve lineup procedures. These lineup procedures are permanent parts of the tests and are reviewed and approved by operations personnel and management prior to performance. During the lineup, the operator is required to verify each valve position after he confirms it to be properly aligned. These procedures allow verification of proper valve lineup prior to system test. We find these special procedures to be acceptable.

Testing requirements will be incorporated in the Technical Specifications as discussed in Section 4.0. From the EI&C standpoint, the OPS satisfies our testability criteria and is acceptable.

Seismic Design and IEEE Std-279 Criteria

IEEE Std-279 and seismic criteria were considered in the design of the OPS. The mechanical equipment of the OPS has been designed and installed to meet the Seismic Class I requirements of Robinson 2. The new electrical components used in the design of the OPS are identical in specification to the current reactor protection system instrumentation components.

Also, the additional electrical equipment will be installed so as not to compromise the seismic design of existing safety systems.

The control air supply from the air accumulators will be seismically designed. The existing PORV's will not be degraded by the OPS modification.

As a result of our review of the OPS modification, the licensee made some changes to bring it into compliance with IEEE Std-279. To assure adequate functional separation, the logic has been changed such that in the revised OPS design, each valve is associated with its own logic train and will not receive signals from the other train.

Although one area of the OPS design does not fully meet the IEEE Std-279 separation criteria, our evaluation shows that this is not a safety problem. Three temperature indicators in parallel are used to provide input to both trains through low-auctioneer devices. A short in one auctioneer circuit could however, prevent operation of the other auctioneer circuits. However, if the short occurs while the OPS is not enabled, i.e., is not required to be operable, then the audible alarm and annunciator are activated. If the OPS is enabled, then in addition to activation of the alarm and annunicator the PORV would open. Thus, failure would either be alarmed immediately for appropriate corrective action or if the OPS has been enabled, it would actuate a PORV to relieve pressure.

It is concluded that the long-term OPS of Robinson 2 has been appropriately designed for Class I seismic design conditions and also adequately satisfies the intent of IEEE Std-279. We find this design to be acceptable with the above described changes.

Appendix G Curves

The Robinson 2 Appendix G pressure versus temperature curves submitted by the licensee for purposes of overpressure transient analysis are for zero to 20 and 20 to 32 effective full power years. The licensee has used the maximum pressure corresponding to a zero degree per hour cooldown rate since most pressure transients occur during isothermal conditions. Margins of 60 psig and 10°F are included for possible instrument errors. The Appendix G pressure limit at 100°F according to these curves is about 500 psig. That is, at 100°F the PORV's would have to actuate if the reactor coolant exceeded that pressure. We find that this use of these curves is acceptable as a basis for OPS performance.

Electrical Controls

The licensee's design for the OPS is based on the use of existing PORV's located on the pressurizer. These valves may be operated manually by closing a switch or may be automatically activated by redundant channels of protective instrumentation which compare actual pressure and temperature with Appendix G limits. The PORV's are spring loaded closed and require air to open. Air is supplied from an air source controlled by solenoid valves. Each channel of the OPS operates in the manner described below.

Wide range temperature signals from all three RCS loops will be fed to a low-auctioneer device which will select the lowest loop temperature. The lowest loop temperature signal will then be fed to a function generator which will convert the input temperature signal into an output pressure signal conforming to that pressure allowed by the plant's 10 CFR 50 Appendix G curve at the input temperature. The converted temperature signal (allowable pressures) from the signal generator will then be compared with a plant wide-range pressure signal. (The plant wide-range pressure signals will be produced from newly installed pressure transmitters, one for each protection channel). When the measured pressure exceeds allowed pressure the comparator will trip and activate a relay whose contacts will in turn activate the primary PORV solenoid. An alarm circuit will annunciate the condition in the control room. One channel will operated PORV-A while the redundant channel will operate PORV-B.

By administrative control, a manual permissive switch will arm both channels when the plant is below 350°F. A non-redundant temperature comparator with control room annunciation will alert the operator when loop temperature drops below 350°F. During normal plant operation, the permissive switch will administratively disable both channels but not the temperature comparator. Plant operating procedures will require enabling the system before the temperature drops below 350°F. When the permissive switch is armed, both channels are available for automatic control of the PORV's. The low temperature annunciator will not light under normal conditions if the system is enabled prior to the temperature dropping below 350°F. With the system enabled below 350°F, the energizing of the annunciator indicates that the Appendix G limit curve has been exceeded and the PORV's will open. All pressure and temperature sensors are powered by a vital power source.

There is both audible and visual flashing annunciation of all alarms. Acknowledgment of an alarm silences the audible portion and changes the visual display to a non-flashing light. If a second (or third or fourth) alarm occurs after the preceding alarm has been acknowledged, it will also be annunciated by both the audible tone and flashing light. The steady annunciator panel light cannot be disabled until all alarms have been cleared.

Connections to the equipment racks are made on terminal strips in separate compartments. The cables are routed via separate cable trays. The lamps are housed in separate compartments in the annunciator panel. Circuit isolation between inputs to an annunciator is accomplished both by transformer circuitry and resistance isolation in each circuit; i.e., two isolating methods per input.

High Pressure Alarm

We required that a high pressure alarm be operable during low RCS temperature operations (below 350°F) to alert the operator of a pressure transient.

This capability has been acceptably provided as described above.

Isolation Valve Alarm

We required that the position of the isolation valves upstream of the PORV be wired into the overpressure protection alarm systems so that the alarm will not clear unless the OPS is enabled and the isolation valve is open. Means should be provided to insure proper alignment of the isolation valve during OPS operation. A description of the alarm system is provided below:

These isolation valves (MOV 535 and MOV 536) are wired into the OPS such that manual operation of the manual permissive switch discussed above will also result in the opening of the isolation valves. An open-close indication for each isolation valve is provided on the main control board by means of annunciators A1-16 and A1-24. (If closed, these annunciator lights cannot be cleared.)

We find this design to be acceptable.

Enable Alarm

We required that an alarm be activated as part of the plant cooldown process to insure that the PORV "Low" setpoint is activated before the RCS temperature is equal to or less than 350°F. A description of the alarm system is provided above. We find this design to be acceptable.

PORV Open Alarm

We required that an alarm be activated to alert the operators that a PORV is in the open position.

A description of the system is provided below.

Alarms A1-16 and A1-24 will annunciate when an overpressure event occurs. The same signals that activate the low temperature/high pressure alarms are also used (with isolation) to open the PORVs. PORV opening is indicated by alarm A3-8, "Pressurizer Protection, High Pressure". We find this design to be acceptable.

Pressure Transient Reporting and Recording Requirements

We require a 30-day licensing event report (LER) for pressure transients which cause the OPS to function, thereby indicating the occurrence of a serious pressure transient. In addition, pressure and temperature recording instrumentation are required to provide a permanent record of the pressure transient. The response time of the P/T recorders shall be compatible with pressure transients increasing at a rate of approximately 100 psig per second.

A wide range pressure recorder with a scale of 0-3000 psig is provided to record pressure transients. In addition, RCS cold leg and hot leg temperatures are recorded on two strip chart recorders. We find this to be acceptable.

Disabling of Non-Essential Components During Cold Shutdown

Our position requires the de-energizing of safety injection system (SIS) pumps and closure of SI header/discharge valves during cold shutdown operations. A description of the licensee's commitments follows.

Spurious SIS operation during cold shutdown will be averted by procedurally requiring the SI accumulator isolation valves to be closed and de-energized, and the high head SI pumps to be de-energized. This equipment will, however, be allowed limited, controlled operation for tests, filling of various systems etc. during cold shutdown, but only with approved procedures and valve lineups as discussed elsewhere in this report.

During cold shutdown conditions, there are only three cases where SIS pumps will not be required to be de-energized and the accumulators not isolated. (1) During refueling, periodic tests are required to meet Technical Specifications requirements. SIS injection into the RCS is prevented during these tests by having the SIS pumps header/discharge valves closed and the accumulators depressurized. (2) Testing after maintenance on the pump will also require operation. There will be similar safeguards as in (1) above for this situation. (3) The SIS pumps are also used to flood the refueling canal. This operation cannot create an overpressurization situation since the vessel head will have been unbolted and the RCS will thus be unpressurized.

Evaluation of Design Basis Events

The incidents that have occurred to date have been the result of operator errors or equipment failures. Two causes of pressure transient initiation can be identified: coolant mass increases resulting from charging pumps, safety injection pumps, safety injection accumulators; and heat buildup (which in a free system would cause thermal expansion) from external or internal heat sources such as the steam generators or reactor residual heat.

On Westinghouse designed plants, the most common cause of the overpressure transients to date has been isolation of the reactor coolant letdown path. Letdown during low pressure operations is through the reactor residual heat removal (RHR) system. Thus, isolation of RHR system can initiate a pressure transient if a charging (reactor makeup) pump is left running. Other transients have occurred with lower frequency. Those which have resulted in the most rapid pressure increases were selected by us for analysis. The most limiting mass input transient we have identified was caused by inadvertent startup of the largest safety injection pump. The most limiting thermal transient we have identified was the startup of a reactor coolant pump (RCP) with a $50^{O}F$ temperature difference between the reactor coolant in the reactor vessel and the reactor coolant in the steam generator associated with that RCP.

Based on the historical record of overpressure transients and the imposition of more effective administrative controls, we determined that the limiting events identified above and discussed below form an acceptable basis for analyses of the performance of the Robinson 2 OPS, including establishment of instrument setpoints.

Setpoint Analysis

The one loop version of the LOFTRAN (Reference WCAP 9707) code was used to perform the mass input analysis case. The four loop version was used for the heat input analysis case. Both versions require some input modeling and initialization changes. LOFTRAN is currently under review by the NRC and is judged to be an acceptable code for treating problems of this type.

The results of this analysis are provided in terms of PORV "setpoint" overshoot. (There is actually a family of setpoints corresponding to the Appendix G temperature pressure limit curves). The predicted maximum transient pressure is simply the sum of the overshoot magnitude and the "setpoint magnitude." The PORV "setpoint" is adjusted so that given the overshoot, the maximum transient pressures will always remain below those allowed by Appendix G limits.

The licensee used the following Robinson 2 plant characteristics to determine the maximum transient pressures reached for the design basis pressure transients:

SI Pump Flow Rate	80.5 lb/sec
RCS Volume	9343 ft^3
PORV Opening Time	2 sec
S G Heat Transfer Area	43,400 ft^2
Relief Valve Setpoint	400 psig

Westinghouse identified certain assumptions used in LOFTRAN that Westinghouse considers to be conservative, that is, overprediction of the peak RCS pressure in these design basis transients. These are listed below along with some plant parameters for Westinghouse assumed values in the generic analysis that the licensee has identified to be conservative relative to the actual Robinson 2 values.

- One PORV was assumed to fail.
- The RCS was assumed to be rigid with respect to metal expansion.
- No credit was taken for the reduction in reactor coolant bulk modulus at RCS temperatures above 100^{o}F (constant bulk modulus at all RCS temperatures).
- No credit was taken for the shrinkage effect caused by low temperature SI water added to higher temperature reactor coolant.
- The entire volume of water of the steam generator secondary was assumed available for heat transfer to the reactor coolant. In reality, the liquid immediately adjacent and above the tube bundle would be the primary source of energy in the transient.
- The overall steam generator heat tranfer coefficient, U, was assumed to be the free convective heat transfer coefficient of the secondary side water, h_{sec}. The forced convective heat transfer coefficient of the primary side reactor coolant, h_{pri}, and the tube metal resistance have been ignored thus resulting in a conservative (high) coefficient.

- The RCP startup time assumed in the heat input analysis was 9-10 seconds whereas the actual Robinson 2 RCP startup time (in a hot, 350°F, RCS) is 12-14 seconds.
- The SI pump startup time assumed in the mass input analysis was 1.64 seconds whereas the actual SI pump startup times is 3-4 seconds.
- The heat input analysis assumed a steam generator (SG) heat transfer area of 58,000 ft^2 whereas the actual Robinson 2 SG area is 43,400 ft^2.

We agree that these are conservative assumptions.

Mass Input Case

The inadvertent start of a safety injection pump with the plant in a cold shutdown condition was selected as the limiting mass input case.

Westinghouse provided the licensee with a series of curves based on the LOFTRAN analysis of a generic plant design which indicates overshoot of the PORV "setpoint" for this transient as a function of system volume, relief valve opening time and relief valve "setpoint" (Reference 15). This sensitivity analysis was then applied to the Robinson 2 parameters to obtain a conservative estimate of the overshoot. We find this method of analysis to be acceptable.

Using the Westinghouse methodology, the Robinson 2 PORV "setpoint" overshoot was determined to be 78 psi. At a temperature corresponding to 400 psig on the Appendix G temperature/pressure curve , the transient pressure would be 478 psig for the worst case mass input transient when the PORV was signaled to open (400 psig). Since the 32 EFPY Appendix G limit at temperatures above 100°F is above 500 psig, we concluded that OPS performance would be acceptable with a 400 psig low pressure relief valve setpoint.

Heat Input Case

Inadvertent startup of a reactor coolant pump with a reactor coolant (RC) to secondary coolant temperature differential across the steam generator of 50°F, and with the reactor coolant system (RCS) in a water-solid condition, was selected as the limiting heat input case. For this case, Westinghouse provided the licensee with a series of curves based on the LOFTRAN analysis of a generic plant design to determine the PORV "setpoint" overshoot as a function of RCS volume, steam generator heat transfer and initial RC temperature (Reference 15).

Also, Westinghouse supplied the licensee with supplemental calculations (Reference 16) for overpressure protection system performance appraisals. For this transient, the reference relief valve selected was assumed to have a total opening time of two seconds from the instant the signal to open is received (PORV "setpoint" readied) until the valve reached the full open position.

Using the supplemental calculations, the licensee calculated final pressures for the heat input transient for a fixed ΔT of 50°F and a range of initial RC temperatures. These values are given here:

RC Temperature	PORV "Setpoint"	Maximum Pressure	Appendix G Limit
100°F	400 psig	419.7 psig	500 psig
180°F	400 psig	445.7 psig	500 psig
250°F	400 psig	466.9 psig	500 psig

The Appendix G limits are not exceeded for the heat input cases over this range of RC temperatures.

Based on the above analyses of the limiting mass input and heat input cases which show a maximum pressure transient below that allowed by Appendix G limits we find that a PORV "setpoint" of 400 psig is acceptable.

Implementation Schedule

The licensee installed and tested the Robinson 2 overpressure protection system during the 1978 refueling outage (February to April) except for the low pressure alarm on the air and N_2 supply to the PORV's. This alarm was not installed during that outage due to equipment delivery schedules. However, pending installation of this alarm, the licensee has committed to monitor the four PORV actuating air supply pressure indicators frequently whenever the OPS is enabled.

The alarm will be installed at the first shutdown of sufficient duration after the necessary equipment is delivered. Since there are in essence three supplies of actuating air to each PORV (instrument air, accumulators, and 2000 psi bottles), we find the above delay in installation of the low pressure alarm to be acceptable.

Administrative Controls

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analysis provided by the licensee, a defense in depth approach has been adopted using procedural and administrative controls. Those specific conditions required to assure that the plant is operated within the bounds of the analysis are spelled out in Technical Specifications.

Procedures

A number of provisions to reduce the likelihood of pressure transients at low temperature conditions are contained in the Robinson 2 operating procedures. These procedures require that an acceptable RCS pressure/temperature profile be achieved prior to startup (and jogging) of a reactor coolant pump (RCP) with the RCS in a water-solid condition. In addition, the plant shutdown and cooldown procedures require the RCP's to be run until the pressurizer has been cooled to less than 200°F. The RCS temperature when the RCP's are tripped will actually be about 150°F, thus the possibility of a significant RCS temperature asymmetry existing when the RCP's are restarted is reduced.

Also, the licensee has modified plant procedures to restrict RCS water-solid operation to only those times when absolutely necessary.

The plant shutdown and cooldown procedures require the closure and removal of power from the safety injection accumulator isolation valves prior to the reduction of RCS pressure to below 1000 psig. Also, the procedures require that prior to the RCS temperature going below 350°F the power supply breakers to the three high pressure safety injection pumps be opened and the control power fuses to these breakers be removed. These breakers remain open whenever the plant is in a shutdown condition unless required by periodic tests (surveillance requirements) or post-maintenance operational tests. During these tests, the safety injection pump header discharge valves are required to be shut and other administrative safeguards are to be taken to prevent an overpressure transient.

We find the procedural and administrative controls described above to be necessary and acceptable. We have further determined that certain of these procedural and administrative controls should be included in the Technical Specifications. These are listed in the following section.

Technical Specifications

We requested the licensee to submit proposed Technical Specifications to ensure proper availability of the overpressure mitigating system and continued applicability of the RCS pressure transient analyses.

The licensee has submitted proposed Technical Specifications in its December 22, 1977 submittal (Reference 13). These specifications are summarized below:

1. Both PORV's must be operable whenever the RCS temperature is less than 350°F, except one PORV may be inoperable for seven days. If these conditions are not met, the RCS must be depressurized and vented (to atmosphere or to the pressurizer relief tank) within eight hours.

2. OPS operability requires that both PORVs have their low pressure "setpoint" (400 psig) selected and that their actuation logic (electronics, power supplies and alarms) be operable. Each PORV upstream isolation valve must be open and the backup air supplies charged.

3. Only one high head safety injection pump may be energized whenever the RCS temperature is at or below 350°F unless specifically required for Technical Specification surveillance tests. During these tests, the safety injection header discharge valves must be shut and de-energized.

4. A reactor coolant pump may be started (or jogged) in a stagnant, water-solid RCS only if the greatest SG/RCS ΔT has been verified to be less than 50°F.

5. The OPS must be tested on a periodic basis consistent with the need for its use. On RCS cooldowns, the OPS electronics should be tested prior to going below 350°F. The PORV and isolation valve operation should be verified every refueling outage.

The licensee further discussed the techniques used to determine the SG/RCS ΔT (for conformance with item 4 above) in its submittal (Reference 18). The procedure for RCS shutdown and cooldowns minimizes the possibility of a significant SG/RCS ΔT from developing by running an RCP as long as possible (discussed in Section 4.1). However, to ensure the ΔT is less than 50°F, the licensee will use the permanently installed RCS loop and RHR system instrumentation to determine the RCS temperature and will measure the SG shell side temperature with a portable pyrometer held to the steam generator outer surface (lagging removed) at an elevation about four feet above the tube sheet*. From these measurements, the largest ΔT is determined and the appropriate action taken (i.e., start the first RCP, establish a pressurizer steam bubble, or cool down the SG secondary water and remeasure temperatures).

We have reviewed the proposed Technical Specifications and conclude that they are consistent with or conservative in relation to the analyses and assumptions discussed above for the Setpoint Analysis, and are therefore acceptable.

Environmental Consideration

We have determined that this amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusions

The design of the Robinson 2 low temperature overpressure protection system in the areas of electrical, instrumentation and control (EI&C) is in accordance with those design criteria originally prescribed by the staff and later expanded during subsequent discussions with the licensee with the exceptions noted previously.

*The licensee has justified (Reference 18) the measurement at this height rather than the maximum water height (about 30 feet higher) with recent Westinghouse correspondence (attached to Reference 18) in which the vendor states that the SG is essentially isothermal about four hours after feed and RCS flow have been secured. Therefore, if the licensee wishes to start an RCP in a water-solid system before four hours after the last RCP has been stopped, SG shell temperature measurements should be made at a location that accounts for a possible non-isothermal steam generator.

We find the EI&C aspects of the proposed design acceptable, on the basis that: (1) the proposed overpressure protection system complies with IEEE Std-279-1971, and seismic criteria as identified in Section 2.0; (2) the system is redundant and satisfies the single failure criterion; (3) the system is testable on a periodic basis, and (4) the recommended Technical Specifications reduce the probability of overpressurization events to an acceptable level.

The administrative controls and hardware changes proposed by the licensee provide protection for Robinson 2 from pressure transients at low temperatures by reducing the probability of initiation of a transient and by limiting the pressure of such a transient to below the limits set by Appendix G. We find that the overpressure protection system meets the criteria established by the NRC and is acceptable as a long term solution to the problem of overpressure transients.

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment 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.

Date: September 14, 1979

REFERENCES

1. NRC (Reid) to Carolina Power and Light Company, (CP&L), (J. A. Jones) dated August 11, 1976.
2. CP&L (J. A. Jones) to NRC (Reid) dated September 7, 1976.
3. CP&L (E. E. Utley) to NRC (Reid) dated October 27, 1976.
4. CP&L (E. E. Utley) to NRC (Reid) dated December 16, 1976.
5. NRC (Reid) to CP&L (J. A. Jones) dated January 10, 1977.
6. NRC (Reid) to CP&L (J. A. Jones) dated February 15, 1977.
7. CP&L (E. E. Utley) to NRC (Reid) dated March 3, 1977.
8. CP&L (E. E. Utley) to NRC (Reid) dated March 3, 1977.
9. CP&L (E. E. Utley) to NRC (Reid) dated April 1, 1977.
10. CP&L (E. E. Utley) to NRC (Reid) dated July 28, 1977.
11. CP&L (B. J. Furr) to NRC (Reid) dated October 31, 1977.
12. CP&L (E. E. Utley) to NRC (Reid) dated December 15, 1977.
13. CP&L (E. E. Utley) to NRC (Reid) dated December 22, 1977.
14. "Staff Discussion of Fifteen Technical Issues listed in Attachment G November 3, 1976 Memorandum for Director NRR to NRR Staff". NUREG-0138, November 1976.
15. "Pressure Mitigating System Transient Analysis Results" prepared by Westinghouse for the Westinghouse user's group on reactor coolant system overpressurization, dated July 1977.
16. "Pressure Mitigating System Transient Analysis Results - Supplement to the July 1977 Report" prepared by Westinghouse for the Westinghouse user's group. This report submitted as an attachment to Reference 12.
17. CP&L (E. E. Utley) to NRC (Reid) dated January 25, 1978.
18. CP&L (E. E. Utley) to NRC (Schwencer) dated April 11, 1978.
19. CP&L (E. E. Utley) to NRC (Schwencer) dated May 16, 1978.
20. NRC Memorandum from R. L. Baer, RSB, DOR, to B. K. Grimes, AD for Engineering and Projects, DOR, "Safety Evaluation of the Overpressure Protection System Designed for Robinson Unit 2," April 25, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 42 to Facility Operating License No. DPR-23, issued to Carolina Power and Light Company, which revised Technical Specifications for operation of the H. B. Robinson Unit No. 2 (the facility) located in Darlington County, South Carolina. The amendment is effective as of its date of issuance.

The amendment consists of additions to the Technical Specifications which incorporate the proposed low temperature overpressure protection system into the Limiting Conditions for Operations and Surveillance Requirements.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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 December 22, 1977, (2) Amendment No. 42 to License No. DPR-23, and (3) the Commission's related Safety Evaluation. 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 Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina 29550. 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 Operating Reactors.

Dated at Bethesda, Maryland, this 14th day of September, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors