

September 19, 1991

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Mr. Lynn W. Eury  
Executive Vice President  
Power Supply  
Carolina Power & Light Company  
Post Office Box 1551  
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE  
NO. NPF-63 REGARDING PRESSURIZER POWER OPERATED RELIEF VALVES  
AND BLOCK VALVES - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1,  
(TAC NO. 80506)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 27 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment consists of changes to the Technical Specifications (TS) in response to your request dated May 15, 1991.

The amendment which includes changes to TS 3/4.4.4, Relief Valves, and 3/4.4.9.4, Overpressure Protection Systems, addresses commitments made by the licensee in response to NRC Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' pursuant to 10 CFR 50.54(f)."

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's regular bi-weekly Federal Register notice.

Sincerely,

15/

Brenda L. Mozafari, Project Manager  
Project Directorate II-1  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

## Enclosures:

1. Amendment No. 27 to NPF-63
2. Safety Evaluation

cc w/enclosures:  
See next page

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Carolina Power & Light Company

Shearon Harris

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AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

Docket File

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27  
License No. NPF-63


1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Carolina Power & Light Company, (the licensee), dated May 15, 1991, 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.
2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 27, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

  
Elinor G. Adensam, Director  
Project Directorate II-1  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: September 19, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 27

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

| <u>Remove Pages</u> | <u>Insert Pages</u> |
|---------------------|---------------------|
| 3/4 4-11            | 3/4 4-11            |
| 3/4 4-12            | 3/4 4-12            |
| 3/4 4-40            | 3/4 4-40            |
| 3/4 4-42            | 3/4 4-42            |
| B 3/4 4-2           | B 3/4 4-2           |
| B 3/4 4-2a          | B 3/4 4-2a          |
| B 3/4 4-26b         | B 3/4 4-2b          |

## REACTOR COOLANT SYSTEM

### 3/4.4.4 RELIEF VALVES

#### LIMITING CONDITION FOR OPERATION

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3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

#### ACTION:

- a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and
  1. With only one safety grade PORV OPERABLE, restore at least a total of two safety grade PORVs to OPERABLE status within the following 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or
  2. With no safety grade PORVs OPERABLE, restore at least one safety grade PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable safety grade PORV or be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.
- c. With one or more block valve(s) inoperable, within 1 hour:
  - (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and
  - (2) apply the ACTION b., above, as appropriate, for the isolated PORV(s).
- d. The provisions of Specification 3.0.4 are not applicable.

## REACTOR COOLANT SYSTEM

### RELIEF VALVES

#### SURVEILLANCE REQUIREMENTS

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4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

- a. Performing a CHANNEL CALIBRATION of the actuation instrumentation, and
- b. Operating the valve through one complete cycle of full travel during MODES 3 or 4, prior to going below 325°F.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

4.4.4.3 The accumulator for the safety-related PORVs shall be demonstrated OPERABLE at least once per 18 months by isolating the normal air and nitrogen supplies and operating the valves through a complete cycle of full travel.



## REACTOR COOLANT SYSTEM

### OVERPRESSURE PROTECTION SYSTEMS

#### SURVEILLANCE REQUIREMENTS (Continued)

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entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and
- c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.4.2 The RCS vent(s) shall be verified to be open at least once per 12 hours\* when the vent(s) is being used for overpressure protection.

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\* Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

## REACTOR COOLANT SYSTEM

### BASES

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#### SAFETY VALVES (Continued)

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no reactor trip until the second Reactor Trip System trip setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shut-down and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

#### 3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

#### 3/4.4.4 RELIEF VALVES

In MODES 1, 2, and 3 the power-operated relief valves (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary and manual RCS pressure control for mitigation of a steam generator tube rupture (SGTR) are the safety-related functions of the PORVs in MODES 1, 2, and 3. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORVs to perform manual RCS pressure control for mitigation of a SGTR accident is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function in MODES 1, 2, and 3. The automatic pressure control function limits the number of challenges to the safety valves, but the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be OPERABLE.

## REACTOR COOLANT SYSTEM

### BASES

#### RELIEF VALVES (Continued)

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Operation with the block valves opened is preferred. This allows the PORVs to perform automatic RCS pressure relief should the RCS pressure actuation setpoint be reached. However, operation with the block valve closed to isolate PORV seat leakage is permissible since automatic RCS pressure relief is not a safety-related function of the PORVs.

The OPERABILITY of the PORVs and block valves in MODES 1, 2, and 3 is based on their being capable of performing the following functions:

1. Maintaining the RCS pressure boundary,
2. Manual control of PORVs to control RCS pressure as required for SGTR mitigation,
3. Manual closing of a block valve to isolate a stuck open PORV,
4. Manual closing of a block valve to isolate a PORV with excessive seat leakage, and
5. Manual opening of a block valve to unblock an isolated PORV to allow it to be used to control RCS pressure for SGTR mitigation.

The non-safety PORV and block valve are used only as a backup to the two redundant safety grade PORVs and block valves to control RCS pressure for accident mitigation. Therefore, continued operation with the non-safety PORV unavailable for RCS pressure control is allowed as long as the block valve or PORV can be closed to maintain the RCS pressure boundary.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their safety functions. Surveillance Requirements 4.4.4.1 and 4.4.4.3 address the PORVs and Surveillance Requirement 4.4.4.2 addresses the block valves.

Surveillance Requirement 4.4.4.1.a provides assurance the actuation instrumentation for automatic PORV actuation is calibrated such that the automatic PORV actuation signal is within the required pressure range even though automatic actuation capability of the PORV is not necessary for the PORV to be OPERABLE in MODES 1, 2, and 3.

Surveillance Requirement 4.4.4.1.b provides assurance the PORV is capable of opening and closing. The associated block valve should be closed prior to stroke testing a PORV to preclude depressurization of the RCS. This test will be done in MODES 3 or 4, before the PORV is required for overpressure protection in TS 3.4.9.4.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

On June 25, 1990, the staff issued Generic Letter 90-06, "Resolution Of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' pursuant to 10 CFR 50.54(f)." The generic letter represented the technical resolution of the above-mentioned generic issues.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's Technical Specifications (TS) were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock & Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 90, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

By letter dated May 15, 1991, Carolina Power & Light Company proposed changes to the Shearon Harris Nuclear Power Plant, Unit 1 (Harris), Technical Specifications (TS) in response to Generic Letter 90-06.

## 2.0 EVALUATIONS

### A. Evaluation For Generic Issue 70

The actions proposed by the NRC staff to improve the reliability of PORVs and block valves represent a substantial increase in overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants."

The TS changes proposed in this submittal revise TS 3/4.4.4, Relief Valves, and TS 3/4.4.9.4, Overpressure Protection Systems, in accordance with the recommendations of Generic Letter 90-06, concerning PORVs and their associated block valves.

The proposed changes to TS 3/4.4.4 are as follows:

1. Revise the mode applicability to MODES 1, 2 and 3 to be consistent with the guidance of Generic Letter 90-06.
2. Revise the Limiting Conditions for Operation (LCO) Action Statement wording to more clearly specify that power be maintained to a block valve which is closed due to its associated PORV being inoperable due to excessive seat leakage.
3. Revise the shutdown requirement from COLD SHUTDOWN to HOT SHUTDOWN to be consistent with the mode applicability requirements.
4. Wording changes necessary to reflect the plant design which includes two safety grade PORVs and one non-safety grade PORV.
5. Eliminate the option of continued operation for periods of greater than 72 hours with an inoperable safety grade PORV if the PORV is inoperable for reasons other than excessive seat leakage.
6. Clarify the 18-month Surveillance Requirement for the PORVs to clearly identify the surveillance test requirements, including the requirement for testing in MODES 3 or 4 prior to cooldown below 325 degrees Fahrenheit.
7. Revise the Surveillance Requirements for the backup Air/Nitrogen supply which operates the PORVs to clearly indicate that his backup supply is the accumulator.
8. Revise the associated TS Bases to reflect the proposed changes and to better define the basis for operability of the PORVs and block valves.

The proposed changes to the Harris TS included in the licensee's letter of May 15, 1991, are consistent with the intent of the staff's generic letter. In the recently completed refueling outage, two of the three PORVs' manual control function and associated circuits and power supplies were upgraded to safety-related. Also the LCO related to the PORV and block valves is being changed to incorporate the requirement that plant operation is not permitted for periods of more than 72 hours if either safety-related PORV or its associated block valve is inoperable for reasons other than excessive seat leakage.

The staff has reviewed the licensee's proposed modifications to the Harris TS. Since the proposed modifications are consistent with the staff's position previously stated in the generic letter and found to be justified in the above mentioned regulatory analysis, the staff finds the proposed modifications to be acceptable.

#### B. Evaluation For Generic Issue 94

The actions proposed by the NRC staff to improve the availability of the low-temperature overpressure protection (LTOP) system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

The proposed change to TS 3/4.4.9.4 is as follows:

Add a new Action Statement that reduces the allowed outage time from 7 days to 24 hours for an inoperable PORV in MODES 5 or 6.

The proposed changes to the Harris TS included in the licensee's letter of May 15, 1991, are consistent with that proposed in the staff's generic letter. The proposed modification to the TS involves plant operation in modes 5 or 6 with an inoperable LTOP channel. The licensee has adopted the staff position in that operations under such conditions are limited to 24 hours before restoring the LTOP channel to operable status.

The staff has reviewed the licensee's proposed modifications to the Harris TS. Since the proposed modifications are consistent with the staff's position as previously stated in the generic letter and justified in the above-mentioned regulatory analysis, the staff finds the proposed modifications to be acceptable.

### 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

#### 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 29270-June 26, 1991). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

#### 5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Robert Kirkwood (Generic Issue 70)  
Edward Throm (Generic Issue 94)  
Brenda L. Mozafari

Date: September 19, 1991