October 11, 1989

Docket No. 50-400

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. 13 TO FACILITY OPERATING LICENSE NO. NPF-63 - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, REGARDING ADMINISTRATIVE CHANGES FOR CORRECTION AND CLARIFICATION (TAC NO. 68340)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 13 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment changes the Technical Specifications to make corrections and clarifications to several sections in response to your request dated December 17, 1987, as supplemented September 25, 1989. These changes are discussed in detail in the enclosed Safety Evaluation.

A Notice of Issuance will be included in the Commission's regular Bi-weekly Federal Register notice.

Sincerely,

/S/

Richard A. Becker, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 13 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

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OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	: OGC	:	:
NAME	:PAnderson	:RBecker	:EAdensam	:MYoung	:	:
DATE	:09/26/89	:09/27/89	:09/28/89	: 9/29/89	:	:

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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. 13
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 December 17, 1987, as supplemented September 25, 1989, 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. 13, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

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: October 11, 1989

OFC	LA:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR			
NAME	PAnderson	BBecker	MYoung	EAdensam			
DATE	9/26/89	9/27/89	9/29/89	10/3/89			

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ATTACHMENT TO LICENSE AMENDMENT NO. 13

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.

Remove Pages	Insert Pages
v	v
viii	viii
ix	ix
x	x
xi	xi
2-7	2-7
3/4 3-9	3/4 3-9
3/4 3-37	3/4 3-37
3/4 3-66	3/4 3-66
3/4 3-79	3/4 3-79
3/4 4-4	3/4 4-4
3/4 4-17	3/4 4-17
3/4 4-34	3/4 4-34
3/4 4-37	3/4 4-37
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
3/4 7-20	3/4 7-20
3/4 7-24	3/4 7-24
3/4 8-21	3/4 8-21
3/4 8-40	3/4 8-40
B 3/4 2-6	B 3/4 2-6
6-20	6-20
6-24	6-24

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

3/4.6 CONTAINMENT SYSTEMS

3/4.7 PLANT SYSTEMS

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1 (\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.09;
K_2	=	0.0182/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 20 s, τ_5 = 4 s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

TABLE 3.3-2 REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

PAGE 3/4 3-10 HAS BEEN DELETED.

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

PAGES 3/4 3-38 THROUGH 3/4 3-40 HAVE BEEN DELETED.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Required Number of Channels shown in Table 3.3-10, except for the pressurizer safety valve position indicator or the sub-cooling margin monitor, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; or

b. With the number of OPERABLE accident monitoring instrumentation channels, except the radiation monitors, the pressurizer safety valve position indicator, or the sub-cooling margin monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; or

c. With the number of OPERABLE channels for the radiation monitors, the pressurizer safety valve position indicator*, or the sub-cooling margin monitor#, less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within the next 14 days, that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.

d. The provisions of Specification 3.0.4 are not applicable.

* The alternate method shall be a check of safety valve piping temperatures and evaluation to determine position.

The alternate method shall be the initiation of the backup method as required by Specification 6.8.4.d.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Lines				
1) Treated Laundry and Hot Shower Tanks Discharge Monitor	D	P	R(3)	Q(1)
2) Waste Monitor Tanks and Waste Evaporator Condensate Tanks Discharge Monitor	D	P	R(3)	Q(1)
3) Secondary Waste Sample Tank Discharge Monitor	D	P, M(5)	R(3)	Q(1)
b. Turbine Building Floor Drains Effluent Line	D	M	R(3)	Q(1)
c. Outdoor Tank Area Drain Transfer Pump Monitor	D	M	R(3)	Q(1)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Normal Service Water System Return From the Waste Processing Building to the Circulating Water System	D	M	R(3)	Q(2)

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. RHR Loop A, or

e. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 335°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Tables 4.4-2A and B.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected,

2. Location and percent of wall-thickness penetration for each indication of an imperfection, and

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted from Technical Specifications. Refer to plant procedure PLP-106, Technical Specification Equipment List Program.

TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

CONTAINMENT SYSTEMS

CONTAINMENT ISOLATION VALVES

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each normal, preentry purge makeup and exhaust, and containment vacuum relief valve actuates to its isolation position, and

d. Verifying that, on a Safety Injection "S" test signal, each containment isolation valve receiving an "S" signal actuates to its isolation position, and

e. Verifying that, on a Main Steam Isolation test signal, each main steam isolation valve actuates to its isolation position, and

f. Verifying that, on a Main Feedwater Isolation test signal, each feedwater isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1 CONTAINMENT ISOLATION VALVES

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

PAGES 3/4 6-17 THROUGH 3/4 6-29 HAVE BEEN DELETED.

The inservice inspection program for SNUBBERS is deleted from Technical Specifications and is controlled in plant procedure PLP-106, Technical Specification Equipment List Program.

PAGES 3/4 7-21 THROUGH 3/4 7-23 HAVE BEEN DELETED.

FIGURE 4.7-1 SAMPLE PLAN(2) FOR SNUBBER FUNCTIONAL TEST

This figure is deleted from Technical Specifications and is controlled in plant procedure PLP-106, Technical Specification Equipment List Program.

TABLE 3.8-1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

PAGES 3/4 8-22 THROUGH 3/4 8-38B HAVE BEEN DELETED.

TABLE 3.8-2 MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

This table is deleted from Technical Specifications.

The information in this table is controlled by plant procedure PLP-106, Technical Specification Equipment List Program.

PAGES 3/4 8-41 THROUGH 3/4 8-43 HAVE BEEN DELETED.

QUADRANT POWER TILT RATIO (Continued)

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The preferred sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. If other locations must be used, a special report to the NRC should be submitted within 30 days in accordance with 10CFR50.4.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are compared to analytical limits of 592.6°F and 2205 psig, respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC in accordance with 10CFR50.4.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or

*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

PEAKING FACTOR LIMIT REPORT

6.9.1.6 The W(Z) Functions for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be established for each reload core and implemented prior to use.

The methodology used to generate the W(Z) functions for RAOC and Base Load operation and the value for APL^{ND} shall be those previously reviewed and approved by the NRC.* If changes to these methods are deemed necessary, they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the W(Z) function for RAOC and Base Load operation and the value for APL^{ND} (as required) shall be provided to the NRC in accordance with 10 CFR 50.4 within 30 days after each cycle initial criticality.

Any information needed to support W(Z), $W(Z)_{BL}$, and APL^{ND} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10CFR50.4 within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

* WCAP-10216, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification."

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY, et al.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated December 17, 1987, as supplemented September 25, 1989, the Carolina Power & Light Company submitted a request for changes to the Shearon Harris Nuclear Power Plant, Unit 1, (Harris) Technical Specifications (TS) which would make corrections or clarifications to several sections. The September 25, 1989, letter provided clarifying information that did not alter the actions noticed, or change the initial determination of no significant hazards consideration as published in the Federal Register (54 FR 37042). Specifically, the following eight corrections and clarifications would be made: (1) establish consistency of nomenclature when a table or figure is deleted from the TS index; (2) insert omitted brackets to equation in Note 1 of Table 2.2-1 used to calculate the overtemperature trip set points; (3) clarify report submission timing for consistency between TS 3.3.3.6 Action c, 6.9.2, 3.3.3.4a and 3.3.3.9.a; (4) correct spelling of radioactivity in Item 1, Table 4.3-8; (5) improve consistency in nomenclature by deleting brackets enclosing train identifiers for residual heat removal loops in TS 3.4.1.3.d and 3.4.1.3.e; (6) delete references to deleted Table 3.6-1 from TS 4.6.3.2 and 4.6.3.3; (7) revise reference for special reporting in Surveillance Requirement 4.4.5.5.c, bases for Section 3/4.2.4 and TS 6.9.1, and 6.9.2 to refer to TS 6.9.2 or 10 CFR 50.4, as appropriate, for consistency; and (8) add additional information to TS sections where the specifications were relocated to plant procedure PLP-106. These changes are discussed in the following paragraphs.

2.0 EVALUATION

The requested changes are all administrative in nature and, as such, do not affect equipment, system or plant safety. For example, change (2) inserting omitted brackets to the equation in Note 1 of Table 2.2-1 and (4) correct the spelling of radioactivity are corrections and are, therefore, acceptable. Changes (1), (3), (5) and (7) are for consistency throughout the TS, which improves clarity and reduces the potential for misinterpretation, and are, therefore, acceptable. Section 6.9.1.6 was included as a change as part of the request for revised reporting requirements in the licensee's December 17, 1987 submittal. However, the reporting revision for Section 6.9.1.6 was included in Amendment No. 7

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issued August 16, 1988 and, therefore, is not included in this amendment. Change (6), removal of references to a table which has been removed from the TS, reduces the potential for confusion and increases clarity and is, therefore, acceptable.

A major revision in the TS between the low power and full power licenses occurred for Harris. This modification was part of the TS improvement efforts and resulted in a revision and reissue of the Harris TS, relocating certain surveillances and technical data related to valves, snubbers, materials and containment penetration conductor overcurrent protective devices to the Harris plant procedure, PLP-106, Technical Specification Equipment List Program. The requested changes describe more thoroughly where the relocated material may be found, are clarifications and are, therefore, acceptable. This change affects Tables 3.3-2, 3.3-5, 4.4-5, 3.6-1, 3.8-1, and 3.8-2; Surveillance Requirement 4.4.9.2.2; pages 3/4 7-20 through 3/4 7-23; and Figure 4.7-1.

In summary, the staff finds the requested changes to be administrative in nature. These changes are corrections, amplifications and clarifications and, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and changes surveillance requirements. The 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 this amendment involves no significant hazards consideration, and there has been no public comment on such findings. Accordingly, this 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 this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 37042) on September 6, 1989, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The NRC staff has concluded, based on the consideration 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 the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Richard A. Becker

Date: October 11, 1989

AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

Docket File
NRC PDR
Local PDR
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