April 4, 1988

Dockets Nos. 50-325/324

Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

DISTRUBUTION LIST:

See Attached Page

Dear Mr. Utley:

SUBJECT: ISSUANCE OF AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2, REGARDING REACTOR VESSEL MATERIAL SURVEILLANCE (TAC NOS. 60978 AND 60979)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 117and 147 to Facility Operating License Nos. DPR-71 and DPR-62 for the Brunswick Steam Electric Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your submittals of March 5, 1986 and December 17, 1987.

The amendments change the Technical Specifications to revise Table 4.4.6.1.3-1 to require the schedule for the removal of the second and third surveillance capsules to be proposed after the results of the first capsule have been evaluated and add to Technical Specification 4.4.6.1.3 the requirement to calculate cumulative effective full power years at least once every 18 months to support the reactor vessel material surveillance test.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,
/S/

Ernest D. Sylvester, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 117 to License No. DPR-71
2. Amendment No. 147 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICIAL RECORD COPY

OGC concurred as marked.

PD21:DRPR	EMCB:NRR	PD21:DRPR EDS	PD21:DRPR
PAnderson	CCheng	ESylvester	EAdensam
2/25/88	2/2/88	3/22/88	3/28/88

Mr. E. E. Utley
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

cc:

Mr. P. W. Howe
Vice President
Brunswick Nuclear Project
Box 10429
Southport, North Carolina 28461

Mr. R. E. Jones, General Counsel
Carolina Power & Light Company
P. O. Box 1551
Raleigh, North Carolina 27602

Mr. Mark S. Calvert
Associate General Counsel
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Mr. Christopher Chappell, Chairman
Board of Commissioners
Post Office Box 249
Bolivia, North Carolina 28422

Mrs. Chrys Baggett
State Clearinghouse
Budget and Management
116 West Jones Street
Raleigh, North Carolina 27603

Resident Inspector
U. S. Nuclear Regulatory Commission
Star Route 1
Post Office Box 208
Southport, North Carolina 28461

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 2900
Atlanta, Georgia 30303

Mr. Dayne H. Brown, Chief
Radiation Protection Branch
Division of Facility Services
N. C. Department of Human Resources
701 Barbour Drive
Raleigh, North Carolina 27603-2008

Mr. C. R. Dietz
Plant General Manager
Brunswick Nuclear Project
Box 10429
Southport, North Carolina 28461

Mr. H. A. Cole
Special Deputy Attorney General
State of North Carolina
Post Office Box 629
Raleigh, North Carolina 27602

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
Post Office Box 29520
Raleigh, North Carolina 27626-0520

AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NO. DPR-71, Brunswick, UNIT 1
AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-62, Brunswick, UNIT 2

DISTRIBUTION:

Docket No. 50-325
Docket No. 50-324
NRC PDR
Local PDR
PD 21 r/f
S. Varga
G. Lainas
P. Anderson
E. Sylvester
OGC-B
D. Hagan
E. Jordan
J. Partlow
T. Barnhard (8)
Wanda Jones
E. Butcher
F. Litton
J. Hayes
ACRS (10)
GPA/PA
ARM/LFMB
C. Cheng

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117
License No. DPR-71

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

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated March 15, 1986 and December 17, 1987 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. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 117, 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 and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 4, 1988

LA:PD21:DRPR
PAnderson
2/25/88

EMTB/NRR
CCheng
3/3/88

PM:PD21:DRPR
ESylvester
3/9/88

OGC-B
3/31/88

D:PD21:DRPR
EAdensam
4/4/88

ATTACHMENT TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

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
3/4 4-13	3/4 4-13
3/4 4-17	3/4 4-17

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor vessel shell temperature and reactor vessel pressure shall be limited in accordance with the limit lines shown on (1) Figure 3.4.6.1-1 for heatup by non-nuclear means, cooldown following a nuclear shutdown, and low power PHYSICS TESTS; (2) Figure 3.4.6.1-2 for operations with a critical core other than low power PHYSICS TESTS or when the reactor vessel is vented; and (3) Figure 3.4.6.1-3 for inservice hydrostatic or leak testing, with:

a. A maximum heatup of 100°F in any one-hour period, and

b. A maximum cooldown of 100°F in any one-hour period.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the reactor coolant system; determine that the system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 The reactor vessel shell temperature and reactor vessel 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.

4.4.6.1.2 The reactor vessel shell temperature and reactor vessel pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-2 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor material irradiation surveillance specimens shall be removed and examined to determine changes in material properties at the intervals shown in Table 4.4.6.1.3-1. The results of these examinations shall be used to update Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1-3. The cumulative effective full power years shall be determined at least once per 18 months.

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM CAPSULE WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	WITHDRAWAL TIME[a] (EFPY)
3	300°	8
2	120°	(b)
1	30°	(b)

(a) The specimen shall be withdrawn during refueling outage immediately preceeding or following the specified withdrawal time.

(b) The schedule for removal of the second and third capsule shall be proposed after the results of the first capsule have been evaluated.

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. DPR-62

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

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated March 15, 1986 and December 17, 1987 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. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 147 , 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 and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

\S\

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 4, 1988

LA:PD21:DRPR	EMTB:NRR	PM:PD21:DRPR	OGC-B	D:PD21:DRPR
PAnderson	CCheng	ESylvester		EAdensam
3/2/88	3/3/88	3/9/88	3/31/88	4/4/88

ATTACHMENT TO LICENSE AMENDMENT NO. 147

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

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
3/4 4-13	3/4 4-13
3/4 4-17	3/4 4-17

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM CAPSULE WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	WITHDRAWAL TIME[a] (EFPY)
3	300°	10
2	120°	(b)
1	30°	(b)

(a) The specimen shall be withdrawn during refueling outage immediately preceeding or following the specified withdrawal time.

(b) The schedule for removal of the second and third capsule shall be proposed after the results of the first capsule have been evaluated.

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor vessel shell temperatue and reactor vessel pressure shall be limited in accordance with the limit lines shown on (1) Figure 3.4.6.1-1 for heatup by non-nuclear means, cooldown following a nuclear shutdown, and low power PHYSICS TESTS; (2) Figure 3.4.6.1-2 for operations with a critical core other than low power PHYSICS TESTS or when the reactor vessel is vented; and (3) Figure 3.4.6.1-3 for inservice hydrostatic or leak testing, with:

a. A maximum heatup of 100°F in any one-hour period, and

b. A maximum cooldown of 100°F in any one-hour period.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the reactor coolant system; determine that the system remains acceptable for continued operations, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 The reactor vessel shell temperature and reactor vessel 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.

4.4.6.1.2 The reactor vessel shell temperature and reactor vessel pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-2 within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality.

4.4.6.1.3 The reactor material irradiation surveillance specimens shall be removed and examined to determine changes in material properties at the intervals shown in Table 4.4.6.1.3-1. The results of these examinations shall be used to update Figures 3.4.6.1-1, 3.4.6.1-2, and 3.4.6.1.3. The cumulative effective full power years shall be determined at least once per 18 months.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO.147 TO FACILITY OPERATING LICENSE NO. DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

In letters to the NRC dated March 5, 1986 and December 17, 1987, Carolina Power & Light Company requested a revision to Table 4.4.6.1.3-1 of the Technical Specifications for Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick Units 1 and 2). The schedule for removing the reactor vessel material surveillance capsules is specified in Table 4.4.6.1.3-1. Carolina Power & Light Company (the licensee) proposed to change the schedule for removing the surveillance capsules. In the initial submittal of March 5, 1986, the licensee proposed an integrated schedule for Brunswick Units 1 and 2 whereby a capsule would be removed alternately from each reactor after 8, 10, 13 and 15 effective full power years (EFPY) of operation. This schedule was proposed in lieu of the schedule in the present technical specifications which calls for the removal of a capsule from each reactor after 10 and 30 years of operation. The licensee proposed to remove the first and third capsules from Brunswick Unit 1 after 8 and 13 years of EFPY operation and the second and fourth capsules from Brunswick Unit 2 after 10 and 15 years of EFPY of operation. Following discussions with the staff, the licensee modified their proposal in a December 17, 1987 submittal. In the latest submittal, the initial capsule would be removed from Brunswick Unit 1 after 8 effective full power years (EFPY) operation and the initial capsule would be removed from Brunswick Unit 2 after 10 EFPY. The schedule for the removal of each Unit's second and third capsule would be proposed after the results of the first capsule had been evaluated. In addition, the licensee proposed to insert, in paragraph 4.4.6.1.3 of the Surveillance Requirements in the Technical Specifications, a statement that "The cumulative effective full power years shall be determined at least once per 18 months."

The stated purpose of the proposed change is to revise the reactor vessel surveillance capsule removal schedule to achieve compliance with the provisions of Appendix H, 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," and ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." Paragraph II.B.1 of Appendix H, 10 CFR Part 50, requires that the test procedures and reporting requirements for the reactor vessel material surveillance program meet the requirements of ASTM E185-82 to the extent practical.

The capsule withdrawal schedule is to permit monitoring of the fracture toughness properties of the vessel materials to the end of life (EOL) of the reactor. Table 1 of ASTM E185-82 lists the recommended number of capsules and the withdrawal schedule for three ranges of shifts in the predicted nil-ductility transition temperature. Where the shift at the inside surface of the vessel is equal to or less than 100°F (37.8°C), the program would consist of three capsules which are withdrawn after 6 and 15 EFPY and at the EOL operation. The first capsule would be withdrawn either after 6 EFPY operation or after the predicted shift of all encapsulated material is about 50°F (18°C), whichever arrives first. Paragraph II.B.3 of Appendix H, 10 CFR Part 50, requires that the proposed capsule withdrawal schedule be submitted to the Director, Office of Nuclear Reactor Regulation, with technical justification for approval prior to implementation.

2.0 EVALUATION

The Safety Evaluation for the Brunswick Units 1 and 2, November 1973, Section 5.2.7, states that the reactor vessel materials surveillance program was acceptable in regard to the number of capsules, the number and type of specimens, and the retention of archive material. Detailed information on the encapsulated materials for the program are recorded in General Electric Company Reports, NEDO-24161 and NEDO-24157.

The limiting materials in the vessels are identified as beltline plates: plate 201 in Brunswick, Unit 1, containing 0.15% Cu, 0.54% Ni and 0.012% P, and plate 351 in Brunswick Unit 2, containing 0.19% Cu, 0.58% Ni and 0.013% P.

The reactor vessels were purchased prior to the issuance of Appendix G, 10 CFR Part 50, and the pressure boundary materials were qualified by drop weight test for the plates and Charpy V-Notch test for the weld metal. The requirements for the plates were NDT of 10°F or less and for the welds was a Charpy V-Notch energy of 30 ft-lb at 10°F. Full impact curves were not obtained on the pressure boundary materials, and the upper-shelf energy levels were not reported. The EOL (32EFPY) fluence at the surface was estimated at 1.98×10^{18} n/cm² and at the 1/4t position at 1.42×10^{18} n/cm² for both reactors. The values were calculated by Westinghouse Electric Corporation from dosimetry measurements.

The EOL transition temperature shift at the reactor vessel surface was calculated using methodology described in Regulatory Guide 1.99, Revisions 1 and 2. For Brunswick, Unit 1, the transition temperature shift was 57.8°F and 58.8°F, for Revisions 1 and 2 respectively; while for Brunswick, Unit 2, the transition temperature shift was 77.9°F and 77.0°F, respectively. These surface transition temperature shifts correspond to an adjusted reference temperature of 67.8°F and 102.8°F for Brunswick Unit 1 and 87.9°F and 121.0°F for Brunswick Unit 2.

The transition temperature shift at the 1/4t position was calculated for both reactors after 8 and 10 EFPY operation. The 1/4t calculated values using Regulatory Guide 1.99 Revisions 1 and 2 are tabulated as follows:

EFPY Operation	Fluence n/cm^2	Transition Temperature Shift, °F. Regulatory Guide 1.99, Revision 1		Regulatory Guide 1.99, Revision 2	
		Unit 1	Unit 2	Unit 1	Unit 2
8	2.54×10^{17}	20.6	27.8	21.4	28.0
10	3.18×10^{17}	23.1	31.1	24.6	32.2
13	4.13×10^{17}	26.3	35.5	27.8	36.4
15	4.76×10^{17}	28.3	38.1	29.9	39.2
EOL(32)	1.42×10^{18}	49.0	65.9	52.4	68.6

The fluence at the surveillance capsules in both reactors lags the fluence at the inside surface of the reactor vessels by a factor of 0.56. The estimated fluence values in the tabulation for the 1/4t positions are slightly less than the values currently estimated by the licensee for the capsules. The licensee estimates that the capsules will receive a fluence of 2.79×10^{17} n/cm^2 after 8 EFPY and 3.49×10^{17} n/cm^2 after 10 EFPY operation. ASTM E185-82 recommends that the surveillance capsules lead the fluence received at the surface of the reactor vessel by a factor from one to three.

The withdrawal of the first capsule in a surveillance program should be scheduled early in the reactor vessel life in order to verify the initial response predictions of the surveillance material to the actual thermal and radiation environment of the reactors. The removal of the first capsule from Brunswick Unit 1 after 8 EFPY and from Brunswick Unit 2 after 10 EFPY operation is expected to permit verification of the adequacy and conservatism of the reactor vessels' pressure/temperature operational limits. The first capsule withdrawal schedule complies with the requirements of ASTM E185-82 to the extent practical.

The requirements of Appendix H of 10 CFR Part 50 are such that after each capsule withdrawal, the test results must be the subject of a summary technical report to be submitted as specified in 10 CFR Part 50.4 within one year after capsule withdrawal unless an extension is granted by the Director, Office of Nuclear Reactor Regulation. The report must include the data required by ASTM E 185-82, as specified in paragraph II.B.1 of Appendix H, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions. If a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specifications must be provided with the report.

The withdrawal schedule for the second and third capsules in Table 1 of ASTM E185-82 is fluence related. The second capsule should be withdrawn when the accumulated fluence corresponds to the approximate EOL fluence at the reactor vessel inner wall location, or after 15 EFPY operation, whichever arrives first. After 15 EFPY operation in Brunswick Units 1 and 2, the surveillance capsule is estimated to receive approximately 25% of the EOL vessel surface fluence. Upon withdrawal, the third capsule should have received not less than the peak EOL inside surface fluence nor more than twice that value. As a result of the severe lag in capsule fluence in Brunswick Units 1 and 2, the licensee has proposed that the withdrawal schedule for the second and third capsules be deferred pending the results of the analysis and evaluation of the test specimens from the first capsules withdrawn from the respective reactors after 8 and 10 EFPY operation. The review of the test results from the initial capsule withdrawal will allow the licensee and the staff to determine when the second capsule withdrawal for Unit 1 and for Unit 2 is appropriate.

EVALUATION SUMMARY

The purpose of the materials surveillance program required by Appendix H, 10 CFR Part 50, is to monitor changes in the fracture toughness properties of ferritic materials in the beltline region of nuclear power reactors resulting from the neutron irradiation and thermal environment. The proposed changes to the Technical Specifications of Brunswick Units 1 and 2, revising the withdrawal schedule of surveillance capsules shown in Table 4.4.6.1.3-1, meet, to the extent practical, the provisions of Appendix H, 10 CFR Part 50, and ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

Further, the staff believes that the licensee's withdrawal schedule should be reviewed again after the fracture toughness data of materials from the first surveillance capsules from both Brunswick Unit 1 and 2 are known and analyzed. Consideration should be given to the dosimetry measurements made in each 18 month period with a determination of the fluences at the surveillance capsule and the wall of the reactor vessel.

3.0 ENVIRONMENTAL CONSIDERATIONS

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite; and that there should be no significant increase in individual or cumulative occupational radiation exposure.

The Commission has previously published a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet 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 amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was initially published in a May 21, 1986 Federal Register Notice (51 FR 18677), and then again on February 24, 1988 (53 FR 5487) and consulted with the State of North Carolina. No public comments were received, and the State of North Carolina did not have any comments.

The staff 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 the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Felix Litton

Dated: April 4, 1988