March 16, 1998

Mr. J. S. Keenan, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Unit No. 2
3581 West Entrance Road
Hartsville, South Carolina 29550

SUBJECT: ISSUANCE OF AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. DPR-23 REGARDING H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2 (TAC NO. MA0409)

Dear Mr. Keenan:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 178 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR). This amendment changes the HBR Technical Specifications (TS) in response to your request dated December 17, 1997, as supplemented by letters dated February 6, 1998 and March 12, 1998.

The amendment revises TS Section 5.6.5, "Core Operating Limits Report," and associated TS bases, to reflect approval of a new method for correlation of departure-from-nucleate boiling (DNB) parameters which is based upon DNB test data for high thermal performance (HTP) fuels. The approved method is contained in Siemens Power Corporation Topical Report, EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."

You are requested to inform the staff in writing when you have implemented the provisions of this amendment.

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

Sincerely,
Original signed by:
Joseph W. Shea, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 178 to DPR-23
2. Safety Evaluation

cc w/enclosures: See next page
* Previous Concurrence

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Plant, Unit No. 2

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AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

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

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 178
License No. DPR-23

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, 1997, as supplemented by letters dated February 6, 1998 and March 12, 1998, 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 3.B. of Facility Operating License No. DPR-23 is hereby amended to read as follows:

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B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 178 , are hereby incorporated in the license. Carolina Power & Light Company 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.

FOR THE NUCLEAR REGULATORY COMMISSION

P. T. Kuo, Acting 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: March 16, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 178

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 areas are indicated by marginal lines.

Remove Pages	Insert Pages
5.0-31	5.0-31
B 2.0-3	B 2.0-3
B 2.0-4	B 2.0-4
B 2.0-5	B 2.0-5
B 3.2-10	B 3.2-10
B 3.2-11	B 3.2-11
B 3.4-2	B 3.4-2

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, Richland WA 99352.

19. EMF-92-081(A), latest Revision and Supplements, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Siemens Power Corporation - Nuclear Division, Richland, WA 99352.

20. EMF-92-153(P)(A), Revision 0 and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, Richland WA 99352, March 7, 1994.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

a. Notification of a pending sample tendon test, along with detailed acceptance criteria, shall be submitted to the NRC at least two months prior to the actual test.

b. A report containing the sample tendon test evaluation shall be submitted to the NRC within six months of conducting the test.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum DNB ratio, or DNBR, during normal operational and anticipated transients, is restricted to the safety limit. A DNBR at the safety limit corresponds to a 95% probability, at a 95% confidence level, that DNB will not occur, and is chosen as an appropriate margin to DNB for all operating conditions. The DNBR safety limit is a conservative design value which is used as a basis for setting core safety limits. Based on rod bundle tests, no fuel damage is expected at this DNBR or greater. For the standard mixing vane fuel, the Siemens Power Corporation XNB correlation has a DNBR safety limit of 1.17 (Ref. 2) and for the high thermal performance fuel the Siemens HTP correlation has a DNBR safety limit of 1.141 (Ref. 3). The safety limit curves provided in Figure 2.1.1-1 remain valid using the Siemens HTP correlation.

The Reactor Trip System setpoints specified in Limiting Condition for Operations (LCO) 3.3.1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressurizer pressure, flow, core power distribution, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Overtemperature ΔT trip;

b. Overpower ΔT trip;

c. Power Range Neutron Flux trip; and

d. Main steam safety valves.

Maintaining the DNBR above the limit ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid and also ensures that the ΔT

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the Updated Final Safety Analysis Report (UFSAR), Ref. 4) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and reactor vessel inlet temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. Figure 2.1.1-1 shows the allowable power level decreasing with increasing reactor vessel inlet temperature at selected pressurizer pressures for constant flow (i.e., three loop operation, minimum flow 97.3×10^6 lbm/hr). The area where clad integrity is assured is below these lines. The temperature limits at low power are considerably more conservative than would be required if they were based on the minimum allowable DNB ratio, but are set to preclude bulk boiling at the vessel exit. The safety limit curves given in Figure 2.1.1-1 are for constant flow conditions. These curves would not be applicable in cases where total reactor coolant flow is less than 97.3×10^6 lbm/hr. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the UFSAR.

The SL is higher than the limit calculated when the Axial Flux Difference (AFD) is within the limits of the $F_t(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature and overpower ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Ref. 4).

(continued)

BASES (continued)

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

If SL 2.1.1 is violated, the requirement to restore compliance and go to MODE 3 places the unit in a safe condition and in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Proposed Appendix A, 32FR10213, July 11, 1967.
2. XN-NF-621(P)(A) Revision 1, "Exxon Nuclear DNB Correlation PWR Fuel Designs," Exxon Nuclear Company, September 1983.
3. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."
4. UFSAR, Sections 3.1, 4.4, 7.2, and 15.0.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to 1.141 using the Siemens Power Corporation's (SPC's) DNB correlation (i.e., HTP) and 1.17 using SPC's XNB

(continued)

BASES

BACKGROUND (continued)

correlation. All DNB limited transient events are assumed are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements. Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition (Ref. 1);

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. Fuel design limits required by HBRSEP Design Criteria (Ref. 3) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.141 using the HTP correlation or 1.17 using the XNB correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion of ≥ 1.17 for the Standard Mixing Vane fuel, and ≥ 1.141 for the High Thermal Performance fuel (Ref. 2). This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2205 psig and the RCS average temperature limit of 579.4°F correspond to analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables–pressurizer pressure, RCS average temperature, and RCS total flow rate–to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error of 2.6% based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators.

The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 178 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER & LIGHT COMPANY

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

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated December 17, 1997, as supplemented by letters dated February 6, 1998 and March 12, 1998, the Carolina Power & Light Company (the licensee) submitted a request for changes to the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR), Technical Specifications (TS). The requested changes revise TS Section 5.6.5, "Core Operating Limits Report," and associated TS bases, to reflect approval of a new method for correlation of departure from nucleate boiling (DNB) parameters based upon DNB test data for high thermal performance (HTP) fuels. The approved method is contained in Siemens Power Corporation Topical Report, EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel." The February 6 and March 12, 1998 submittals contained clarifying information and did not alter the initial no significant hazards consideration determination published in the Federal Register on January 28, 1998.

2.0 EVALUATION

In order to ensure that integrity of the fuel cladding is maintained, overheating of the fuel clad must be prevented under all operating conditions. It is known that by ensuring that regions of the core remain below the upper limit of the nucleate boiling heat transfer regime, fuel clad temperatures can be maintained at acceptable values. Operation beyond the upper limit of the nucleate boiling regime is termed departure from nucleate boiling. For pressurized water reactors, such as HBR, operational limits are developed to ensure that regions of the core do not achieve or proceed beyond DNB for all normal operations and expected transients. These safety limits are expressed in terms of a ratio of power needed to achieve DNB to actual core power. This ratio is known as the Departure from Nucleate Boiling Ratio (DNBR). Plant safety limits are expressed in term of the DNBR.

Appropriate safety limit (SL) DNBR values are calculated using analytical methods capable of predicting core conditions under various operating conditions. The approved analytical methods are listed in Section 5.6.5, "Core Operating Limits Report," in the HBR Technical Specifications. The licensee has proposed to change the analytical methods used to calculate the SL DNBR for one type of fuel currently in use and to revise Section 5.6.5 of the TS accordingly.

Specifically, the licensee currently references ANF-1224(P), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel" (ANFP) as the approved methodology for the correlation used to evaluate the DNBR safety limit for HTP fuel. For HTP fuel, the ANFP correlation gives a DNBR safety limit of 1.154. ANF-1224(P) is currently listed in TS 5.6.5 as an approved methodology for use in developing the periodic Core Operating Limits Report.

The licensee has proposed to use a new NRC-approved correlation described in EMF-92-153(P), "HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel." This correlation was submitted to the NRC by the Siemens Power Corporation in September 1992. Use of the new correlation results in a SL DNBR for HTP PWR fuels of 1.141.

As documented in its safety evaluation (SE) dated December 23, 1993, the NRC staff reviewed the topical report and approved it for use in license applications. The staff conditioned use of the correlation on the successful satisfaction of several conditions. Those conditions are that:

(1) the HTP critical heat flux correlation is applicable to fuels whose design characteristics fall within the values for certain parameters specified in Table 2 of the December 1993 SE.

(2) the application of the HTP correlation for DNB analysis is restricted to the operating conditions specified in Table 1 of the December 1993 SE.

In the amendment application, the licensee stated that both of these conditions are satisfied by application of the HTP correlation to the HTP fuel at HBR. In a supplemental submittal dated February 6, 1998, the licensee specified expected HBR values for each of the parameters listed in Table 1 and Table 2 of the December 1993 SE. The staff reviewed the values specified by the licensee and confirmed they fell within the allowable ranges specified in the December 1993 SE.

Although the EMF-92-153(P) correlation determined a new DNBR safety limit for HTP fuel, the safety limit curves in Figure 2.1.1-1 of the TS are unchanged. The curves in Figure 2.1.10-1 represent the loci of points of thermal power, reactor coolant system pressure and reactor vessel inlet temperature for which the minimum DNBR is not less than the safety limit as determined by an appropriate correlation. For HBR, the TS bases and Updated Final Safety Analysis Report present DNBR safety limits for HTP fuel assemblies and standard mixing vane fuel assemblies, both of which are used at HBR. The DNBR safety limit for standard mixing vane fuel is determined by a separate, approved, correlation and has a current value of 1.17. This value is more restrictive than either the existing or proposed safety limit for HTP fuel assemblies. As a result, the safety limit curves in Figure 2.1.1-1, which ensure that the most restrictive DNBR safety limit (in this case the standard mixing vane fuel safety limit) is not violated, remain unchanged.

Subsequent to the staff's December 1993 SE on EMF-92-153(P), the vendor issued EMF-92-153(P)(A) Revision 0 and Supplement 1 in March 1994 which reflected the approved version of the methodology and incorporated the staff's SE. The licensee proposed to revise TS Section 5.6.5 and Bases Sections 2.1.1, 3.2.2 and 3.4.1 to reference the topical report EMF-92-153(P)(A) Revision 0 and Supplement 1 and the new HTP DNBR safety limit value of 1.141. Use of NRC-approved methodology will ensure that values for cycle specific parameters are determined such that all applicable limits (e.g., nuclear limits, fuel thermal and mechanical limits, and transient analysis limits) of the safety analysis are met. The staff has determined that this topical report is appropriate for use at HBR. Therefore, the TS change is acceptable

The licensee also revised the Bases Section 2.1.1 to reflect reference to the correct methodology by which the DNBR limit for the standard mixing vane fuel is determined. This methodology, XN-NF-621(P)(A) Revision 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, September 1983, was included in the TS by amendment 141 which implemented the COLR on July 15, 1992. However, the reference to XN-NF-621 was inadvertently omitted from the Bases at that time. Thus, the licensee is now revising the Bases to delete reference to a previous methodology (XN-NF-711) and insert reference to XN-NF-621, which as described above, is already included in the TS. The staff finds this administrative correction acceptable.

Based on the licensee's demonstration that it met the conditions in the December 1993 SE for use of Topical Report EMF-92-153 in licensing applications, thus demonstrating an acceptable basis for plant safety limits, the staff concluded the licensee's proposed changes are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South 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 amendment also changes record keeping and reporting requirements. 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 (63 FR 4309). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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, (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: JShea
LKopp

Date: March 16, 1998