March 30, 1994

Docket Nos. 50-413
and 50-414

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Mr. David L. Rehn
Vice President, Catawba Site
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Dear Mr. Rehn:

SUBJECT: ISSUANCE OF AMENDMENTS - CATAWBA NUCLEAR STATION, UNIT 1 AND 2, RCS FLOW RATE MEASUREMENT (TAC NO. M88480 UNIT 1)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. **116** to Facility Operating License NPF-35 and Amendment No. **110** to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 10, 1994, as supplemented March 21, 1994. The changes in this amendment apply only to Unit 1 for the remainder of fuel cycle 8. Several TS pages are replicated in order to preserve the TS values for Unit 2. The proposed changes for Unit 2 and the proposal to change the uncertainty value on TS Figure 3.2-1 for both units are not acted on by the amendment. They will be addressed by further correspondence.

The amendments revise TS Table 2.2-1 and TS 4.2.5 to allow a change in the method for measuring reactor coolant system flowrate from the calorimetric heat balance method to a method based on a one-time calibration of the reactor coolant system cold leg elbow differential pressure taps.

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

Sincerely,
ORIGINAL SIGNED BY:
Robert E. Martin, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. **116** to NPF-35
2. Amendment No. **110** to NPF-52
3. Safety Evaluation

cc w/enclosures:
See next page

***SEE PREVIOUS CONCURRENCE**

OFFICE	PDII-3/LA	PDII-3/PM	OGC*	PDII-3/D	SRXB*
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DATE	3/30/94	3/30/94	3/28/94	3/30/94	3/28/94

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Catawba Nuclear Station

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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. **116**
License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees), dated January 10, 1994, as supplemented March 21, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-35 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. **116** , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: **March 30, 1994**

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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 110
License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated January 10, 1994, as supplemented March 21, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-52 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. **110** , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: **March 30, 1994**

ATTACHMENT TO LICENSE AMENDMENT NO. 116

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 110

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

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

Remove Pages	Insert Pages
2-A4	2-A4
B 2-6	B 2-6
3/4 2-14	3/4 2-14
B 3/4 2-4	B 3/4 2-4

TABLE 2.2.-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤109% of RTP*	≤110.9% of RTP*
b. Low Setpoint	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Intermediate Range, Neutron Flux	≤25% of RTP*	≤31% of RTP*
5. Source Range, Neutron Flux	≤10^5 cps	≤1.4 x 10^5 cps
6. Overtemperature ΔT	See Note 1	See Note 2
7. Overpower ΔT	See Note 3	See Note 4
8. Pressurizer Pressure-Low	≥1945 psig	≥1938 psig***
9. Pressurizer Pressure-High	≤2385 psig	≤2399 psig
10. Pressurizer Water Level-High	≤92% of instrument span	≤93.8% of instrument span
11. Reactor Coolant Flow-Low	≥91% of loop minimum measured flow**	≥89.7% of loop minimum measured flow**

*RTP = RATED THERMAL POWER

**Loop minimum measured flow = 95,500 gpm

***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, is automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection and prevents DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 91% for Unit 1 and 90% for Unit 2 of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 91% for Unit 1 and 90% for Unit 2 of nominal full loop flow. Conversely, on decreasing power between P-8 and P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

2. Within 24 hours of initially being within the region of prohibited operation specified on Figure 3.2-1, verify that the combination of THERMAL POWER and Reactor Coolant System total flow rate are restored to within the regions of restricted or permissible operation, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement*.

4.2.5.3 The Reactor Coolant System total flow rate shall be determined by precision heat balance measurement* at least once per 18 months.

*For Unit 1 Cycle 8 only, RCS flow shall be measured using cold leg elbow tap ΔPs, normalized to constants derived from averaged valid calorimetrics from previous cycles.

3/4.2.5 DNB PARAMETERS (Continued)

to maintain a design limit DNBR throughout each analyzed transient. As noted on Figure 3.2-1, Reactor Coolant System flow rate and THERMAL POWER may be "traded off" against one another (i.e., a low measured Reactor Coolant System flow rate is acceptable if the THERMAL POWER is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relationship defined on Figure 3.2-1 remains valid as long as the limits placed on the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$ in Specification 3.2.3 are maintained. The indicated T_{avg} value and the indicated pressurizer pressure value correspond to analytical limits of 594.8°F and 2205.3 psig respectively, with allowance for measurement uncertainty. When Reactor Coolant System flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-1 since a measurement error of 2.1% for Reactor Coolant System total flow rate has been allowed for in determination of the design DNBR value.

For Unit 1, the measurement error for Reactor Coolant System total flow rate is based upon the performance of past precision heat balances. Sets of elbow tap coefficients, as determined during these heat balances, were averaged for each elbow tap to provide a single set of elbow tap coefficients for use in calculating Reactor Coolant System flow. This set of coefficients establishes the calibration of the Reactor Coolant System flow rate indicators and becomes the set of elbow tap coefficients used for Reactor Coolant System flow measurement. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from these heat balances in a non-conservative manner. For Unit 2, the measurement error for Reactor Coolant System total flow rate is based upon performing a precision heat balance and using the result to calibrate the Reactor Coolant System flow rate indicators. Potential fouling of the feed-water venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi is included in Figure 3.2-1. Any fouling which might bias the Reactor Coolant System flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the Reactor Coolant System flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

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. Indication instrumentation measurement uncertainties are accounted for in the limits provided in Table 3.2-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 110 TO FACILITY OPERATING LICENSE NPF-52

DUKE POWER COMPANY, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated January 10, 1994, as supplemented March 21, 1994, Duke Power Company, et al. (the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would revise TS Table 2.2-1, TS 4.2.5, and the BASES to allow a change in the method for measuring reactor coolant system (RCS) flowrate from the calorimetric heat balance (CHB) method to a method based on a one-time normalization of the RCS cold leg elbow tap signals to constants derived from averaged valid calorimetrics from previous cycles. The March 21, 1994, letter provided clarifying information that did not change the initial scope of the January 10, 1994, application, and the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

In the past, RCS flowrate has been determined, in accordance with TS 4.2.5.2 and 4.2.5.3, by the CHB method each 18 months since the issuance of Amendment Nos. 22 and 3, dated June 28, 1983, for McGuire Units 1 and 2 and since initial operation of Catawba Units 1 and 2. This is done by: (1) determining the energy transferred to the plant's steam generators (secondary side), (2) correcting this value for RCS pump heat input and system energy losses, and (3) dividing the result by the primary side differential enthalpy from the RCS cold to hot leg. This yields a value of RCS (primary side) flowrate that is then used to determine compliance with the TS minimum measured required RCS flowrate on TS Figure 3.2-1 and the footnote on TS Table 2.2-1.

The licensee believes that the measurement uncertainty in the calorimetric heat balance method is dominated by the uncertainty in determining the hot leg enthalpy; specifically, the hot leg temperature. In recent fuel cycles of operation, the licensee believes that a phenomenon termed hot leg temperature streaming has become more pronounced. The licensee has attributed this as being largely due to the increased usage of low neutron leakage reactor cores. The use of lower leakage core designs results in a higher percentage of the core power being produced in the inner core regions. This leads to an increased temperature distribution within the hot leg due to incomplete mixing in the upper plenum and results in different temperature readings by one or more of the three hot leg temperature sensors (RTDs) and an indication of an average hot leg temperature that does not accurately reflect true bulk average hot leg coolant temperature. The hot leg coolant density is calculated from the temperature and impacts the calculated RCS flowrate. When reflected in the CHB method of determining RCS flowrate, the licensee believes that hot leg streaming results in calculation of a conservatively low RCS flowrate.

Previously, the indication from the cold leg elbow taps has been normalized each 18 months to result in an indication of RCS flow that is equivalent to that determined from the CHB. This indication is utilized by the reactor protection system (RPS) to trip the reactor on low RCS flowrate at the setpoint value specified in TS Table 2.2-1, Item 11. This normalization, each 18 months, has resulted in a data set of normalization factors (K values). The licensee believes that the trend of the non-normalized indications from the cold leg elbow taps over the life of the plant are more reflective of the expected actual trends in RCS flow, particularly for the last several fuel cycles, than the CHB results. Therefore, the licensee has proposed to determine a fixed value of the normalization factors, K, from the available plant lifetime data set, that would, henceforth, be applied to the cold leg elbow tap indication. The resulting indication of flow would then be used both as the flow input to the RPS and to determine compliance with TS Table 2.2-1, footnote **, on loop minimum measured flow and the TS Figure 3.2-1 required flow value to attain full power operation. The resulting data set of elbow tap coefficients are presented in the licensee's submittal of January 10, 1994.

The NRC staff met with the licensee on March 16, 1994, and requested that the licensee provide additional information. The licensee responded in its submittal dated March 21, 1994, as discussed below.

The licensee evaluated the effects on core pressure drop due to the transition to B&W fuel and found that a slight pressure decrease across the core would result. This would contribute to a small increase in RCS flow. The licensee evaluated the effects of steam generator tube plugging and sleeving on expected RCS flow and determined the expected contribution to an increase in flow resistance. It then calculated that the combination of steam generator and core changes during the EOC 7 refueling outage would result in a decrease in flow in each loop that ranged from 0.15% to 1.13%.

The licensee provided elbow tap differential pressure data from operating cycles 6 and 7. The staff evaluated these averaged/selected data and found sufficient stability to support use of elbow tap information in the manner proposed by the licensee for the remainder of cycle 8.

The staff requested the licensee to evaluate the most conservative assumption regarding the consequences of anticipated operational occurrences (AOOs) and accidents in the Final Safety Analysis Report, Section 15 events; namely, that the postulated conservatively low indication of RCS flow from the CHB is accurate, at a value of 379,285 gallons per minute (gpm) versus the minimum required value of 382,000 gpm. This would represent a flow deficit of 0.71%. In its March 21, 1994, submittal, the licensee evaluated the effects of this postulated flow deficit against the margins available for:

- Departure from Nucleate Boiling Ratio (DNBR) limited events,
- Secondary System Peak Pressure events,
- Primary System Peak Pressure events,
- Feedwater Line Break Long Term Core Cooling Analysis,
- Steam Generator Tube Rupture Dose Analysis,
- Loss of Coolant Accidents, and
- Boron Dilution Events

The licensee's evaluation concludes that margin exists for each event to account for a postulated 0.71% flow reduction and, accordingly, to allow Catawba, Unit 1, to operate at 100% power. In its summary in the March 21, 1994, submittal, the licensee cites a flow uncertainty margin of up to 0.27% flow based on the conclusion that the overall uncertainty on the measurement of RCS flow is 1.93% versus the 2.2% uncertainty now reflected on TS Figure 3.2-1. The staff has not accepted the lower value of 1.93% uncertainty since the components of this value attributable to flow temperature streaming have not been adjusted to reflect the increased uncertainty in hot leg RCS temperature observed in recent fuel cycles. This will be the subject of further review.

The licensee also identifies a DNB margin for the current Unit 1 cycle of 7.4% DNB for Mark BW fuel. The staff feels that the appropriate value is 4.6% which is the value identified in the licensee's application dated September 7, 1993, in support of the current fuel Cycle 8. The higher value of 7.4% is based on the results from a testing program sponsored by the licensee that has not been submitted to the NRC. Accordingly, those results have not been reviewed by the staff and are not relied on in this evaluation. These exclusions of certain specific licensee identified margins do not compromise the licensee's overall conclusions. This is based on the licensee's identification that the postulated 0.71% flow decrease translates to a 1.6% DNB penalty which is more than compensated for by the available 4.6% margin. The licensee has also determined that the remaining nine Westinghouse fuel assemblies have significant DNB margin based on the reduced value of their peaking factors in this fuel cycle.

The revised method for determining RCS flowrate was proposed by the licensee as being applicable to the Catawba Units 1 and 2 and McGuire Units 1 and 2 plants for the remainder of their life. While the NRC staff has concluded that the method is acceptable for a short period consistent with the remainder of fuel cycle 8 for Catawba, Unit 1, it is not prepared, at this time, to approve departure from the well-established CHB method for all four units for the remainder of their plant life. Therefore, the staff has restricted the approval given in this safety evaluation to the remainder of fuel cycle 8 for Catawba, Unit 1. The staff will continue to review the licensee's proposal with respect to the appropriate long term corrective action for the uncertainty attributable to hot leg streaming.

The licensee's revised method of determining RCS flowrate would provide an indicated flowrate of 389,533 gpm versus the value of 379,285 gpm provided by the CHB method. The staff concludes that, based on the issues discussed above, this provides adequate assurance that there will be at least the

382,000 gpm available to support continued Unit 1 operation at full power for the remainder of the current Cycle 8 fuel cycle.

Technical Specification Changes

The specific TS changes are as follows:

TS 4.2.5.2 and TS 4.2.5.3

The terms "calorimetric flow measurement" and "precision heat balance measurement" will be followed by an asterisk as proposed in the March 21, 1994, submittal to include the following notation:

> *For Unit 1 Cycle 8 only, RCS flow shall be measured using cold leg elbow tap ΔPs, normalized to constants derived from averaged valid calorimetrics from previous cycles.

This is the same in its effect as the licensee's January 10, 1994, submittal for modification of these TS except that the above provision limits reliance on the identified elbow tap normalization constants for Catawba, Unit 1, to the remainder of the present fuel Cycle 8.

TS Table 2.2-1

The values of the Reactor Coolant Flow-Low trip setpoint will be increased from 90% to 91% and the allowable value will be increased from 88.9% to 89.7% to account for an increase in the channel statistical allowance for the low flow trip signal attributable to the inclusion of allowances for elbow tap uncertainties since these will no longer be normalized out each 18 months by the CHB process.

TS Figure 3.2-1

As noted in the evaluation above, the licensee's proposal to lower the value of flow measurement uncertainty of 2.2% to 1.9% is not acted on by this amendment. This proposal will be the subject of further review.

BASES

The BASES to the TS were modified to reflect the above noted changes.

Certain previously existing TS pages have been replicated to preserve values for Unit 2 which are not changed by this amendment. On the basis of the above noted discussions, the staff concludes that these changes are acceptable for the remainder of Catawba, Unit 1, Cycle 8 operation.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements 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 NRC 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 is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 3743 dated January 26, 1994). 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.

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

Principal Contributors: H. Balukjian
W. Lyon
R. Martin

Date: **March 30, 1994**