Docket File

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

December 16, 1993

Docket No. STN 50-456

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III, Suite 500
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Farrar:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M88199)

The U. S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 43 to Facility Operating License No. NPF-72 for the Braidwood Station, Unit 1. The amendment is in response to your application dated November 12, 1993, as supplemented by letters dated November 18 and December 6, 1993.

The amendment changes the existing Technical Specifications (TS) by adding a footnote to TS 4.4.5.0 to address steam generator (SG) operability requirements. The change references an unscheduled inspection of the 1C SG which occurred due to a tube leak in that SG. The amendment was required because the circumstances of the inspection were not covered by existing TS 3/4.4.5. It will allow SG operability requirements to be satisfied until the next SG inservice inspection, scheduled to begin no later than March 5, 1994. Emergency treatment of this amendment was necessary to allow resumption of Unit 1 operation. Without this amendment, an inservice inspection of all Unit 1 SGs would have been required. However, because the remaining tubes in 1C SG showed no degradation similar to that which caused the tube leak, restarting Unit 1 without inspecting the remaining SGs involved minimal safety significance.

A Notice of Enforcement Discretion (NOED) was verbally granted on November 5, 1993, to allow plant operation for the period beginning November 5, 1993, until this emergency TS amendment could be issued. The NOED was formally documented in a letter from the NRC to Commonwealth Edison Company (CECo) dated November 24, 1993.

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A copy of the related Safety Evaluation is also enclosed. The notice of issuance and final determination of no significant hazards consideration and opportunity for a hearing will be included in the Commission's bi-weekly Federal Register Notice.

Sincerely,

Original signed by:

Ramin R. Assa, Acting Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 43 to NPF-72
2. Safety Evaluation

cc w/enclosures:
See next page

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. NPF-72

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated November 12, 1993, as supplemented by letters dated November 18 and December 6, 1993, 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-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 43 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee 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

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 16, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. NPF-72

DOCKET NO. STN 50-456

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised page is identified by amendment number and contains vertical lines indicating the area of change. The page marked with an asterisk is provided for convenience.

Remove Pages	Insert Pages
3/4 4-13	3/4 4-13
*3/4 4-14	*3/4 4-14

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.#

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

Unit 1 entered an unplanned outage (A1F26) on October 24, 1993, to repair a tube leak in the 1C Steam Generator. The tube leak was less than the reactor-to-secondary leakage limit of Specification 3.4.6.2c for one steam generator. The generator was determined to be OPERABLE following completion of the inspection plan detailed in Letter # SVP/93-063, S. M. Berg, Jr. (CECo) to J. Zwolinski (NRC), dated November 10, 1993. The generator shall be demonstrated OPERABLE in accordance with Specification 4.4.5.0 prior to the initial resumption of plant operation following the Unit 1 Cycle 4 Refuel Outage (A1R04).

SURVEILLANCE REQUIREMENTS (Continued)

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20% of wall thickness),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10% of wall thickness) further wall penetrations to be included in the above percentage calculations.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-72

COMMONWEALTH EDISON COMPANY

BRAIDWOOD STATION, UNIT 1

DOCKET NO. STN 50-456

1.0 INTRODUCTION

On October 24, 1993, Braidwood, Unit 1, was shut down due to a primary-to-secondary leak. The leak rate was below the technical specifications (TS) limit. After investigating the cause of the leak, the licensee identified a through-wall crack in one tube in the cold leg of 1C steam generator (SG). The tube degradation was characterized as an isolated manufacturing defect and the tube was plugged. However, during the inspection of the 1C SG the licensee discovered additional tube degradations at the support plates which would require the licensee to comply with the surveillance requirements of TS 3.4.5., Table 4.4-2.

In a conference call conducted on November 5, 1993, as documented by NRC letter of November 24, 1993, the staff granted a Notice of Enforcement Discretion (NOED) because existing TS 3/4.4.5 did not apply to the circumstances surrounding the inspection of the 1C SG. It was then concluded that a change in the TS would be required.

In a letter dated November 12, 1993, Commonwealth Edison Company (CECo, the licensee) requested a change to the TS to Facility Operating License No. NPF-72 for Braidwood Station, Unit 1. The proposed change would grant a one-time only deviation from Surveillance Requirement 4.4.5.0 by adding a footnote. The footnote reads as follows: "Unit 1 entered an unplanned outage (A1F26) on October 24, 1993, to repair a tube leak in the 1C SG. The tube leak was less than the reactor-to-secondary leakage limit of Specification 3.4.6.2c for one SG. The generator was determined to be OPERABLE following completion of the inspection plan detailed in Letter # SVP/93-063, S. M. Berg, Jr. (CECo) to J. Zwolinski (NRC), dated November 10, 1993. The generator shall be demonstrated OPERABLE in accordance with Specification 4.4.5.0 prior to the initial resumption of plant operation following the Unit 1 Cycle 4 Refuel Outage (A1R04)."

2.0 EVALUATION

TS 3/4.4.5 surveillance requirements for SG tube inspections are intended to detect and correct any tube damage or degradation that could lead to tube failure as a primary pressure boundary. The TS mandates inspections of the other SGs if the inspection results of one SG fall into severity category C-3. This follows the guidance of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes" which implements the General Design Criteria (GDC) related to reactor coolant pressure boundary integrity by reducing the probability and consequences of SG tube failures.

SG tube leakage is used as an indicator of tube integrity. The TS Bases state that the tube leak limit of 500 gallons per day (gpd) per SG ensures that tubes will have adequate margin to withstand loads imposed during normal operation and postulated events. This leakage limit is used to satisfy the guidance of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" pertaining to operational tube degradation margins.

An inspection of the 1C SG was conducted to determine the cause of the primary-to-secondary leakage detected on October 23, 1993. Based on visual and eddy current examination, the licensee attributed the leakage to a freespan crack in tube R49C76. Bobbin coil and motorized rotating pancake coil (MRPC) probe examination permitted characterization of the crack as being 1.3-inches in length (of which approximately 5/8-inch is through-wall). The crack is located in the tube freespan between the third and fourth antivibration bars (AVBs) on the cold leg side of the SG. The licensee reported that the MRPC examination indicated that the crack was superimposed on a much smaller "ridge" approximately 18 inches long. The licensee believes this ridge, which is located on the outside-diameter of the tube, is indicative of a scratch or a deposit. There was no detectable degradation at this location on tube R49C76 during the previous bobbin coil examination performed approximately twelve months earlier.

As a result of the tube leak, a 100-percent bobbin coil inspection of SG 1C was performed. This examination identified 17 additional freespan indications which were subsequently tested with the MRPC probe. Based on an analysis of the MRPC data, the licensee concluded that none of these 17 indications were similar in nature to the flaw in tube R49C76. Therefore, the licensee concluded that the flaw in tube R49C76 was an isolated incident and that there was no reason to believe that such a flaw would be identified in the other SG.

In addition to the 17 freespan indications, the licensee also identified 116 quantifiable tube support plate indications greater than 40-percent through-wall in SG 1C. These indications, located at the tube support plate elevations on the hot leg side of the SG, were not unexpected since similar degradation was observed during previous inservice inspections of the SG tubing. The licensee plugged all of the 116 tubes.

Several other non-quantifiable indications were also identified at the tube support plate elevations during the bobbin coil examination. Since the indications depth could not be quantified with the bobbin coil, the licensee evaluated the voltage-growth of these indications since the previous inspection. Since none of these non-quantifiable indications displayed an abnormal growth rate and would be further evaluated during the next scheduled SG inspection currently scheduled to begin in early March 1994, these indications were not plugged by the licensee.

The licensee reported that SG 1C has historically exhibited the most tube degradation when compared to the other three SGs at Unit 1. Furthermore, the licensee reported that the results of the inspections performed during this unplanned outage were consistent with the degradation expected since the last SG tube inservice inspection. As a result, the licensee concluded that the previous SG inservice inspection performed during the Unit 1 refueling outage provides sufficient assurance that the Unit 1 SGs could be safely operated until the next scheduled SG inspection. In addition, the licensee concluded that it was not necessary to accelerate the scheduled SG tube inservice inspections for the other SGs at this time.

At the staff's request, by letter dated November 18, 1993, the licensee provided additional data to support their claim that the SG 1C inspection results did not suggest a tube integrity concern for the remaining Unit 1 SGs. To assess the structural significance of the quantifiable and non-quantifiable indications at the tube support plates, the licensee provided the voltage growth and voltage amplitude of the indications detected at the tube support plate elevations. The voltages were obtained after calibrating the 550 kHz bobbin coil channel to read 4.0 volts peak-to-peak on the ASME calibration standard 20-percent flat bottom holes. This method is, for the most part, consistent with industry practice in the United States as it pertains to voltage-based plugging criterion voltage calibration. The voltage values provided by the licensee are summarized in the following table. The values cited are bounding values (e.g., the average growth rate of all tube support plate (TSP) indications did not exceed 0.55-volts).

Categories	Average Growth Rate	Maximum Growth Rate	Average Signal Amplitude	Maximum Signal Amplitude
All TSP indications	0.55		1.00	4.75
Non-quantifiable distorted TSP indications	0.35	1.35	0.75	2.0
Quantifiable ($\geq$40% through-wall) TSP indications	1.10	4.0	1.60	4.75

The licensee reviewed the results of the SG 1C inspection to ensure that there was not a tube integrity concern for the remaining Unit 1 SGs. The three types of indications identified in SG 1C during the forced outage were freespan cracking, quantifiable indications at the tube support plate elevations, and non-quantifiable indications at the tube support plate elevations. As mentioned previously, the licensee believes that the flaw identified in the tube which leaked was an isolated incident and that there was no reason to believe that a flaw similar in nature to that would be identified if the other SGs were inspected. Furthermore, with respect to the quantifiable and non-quantifiable indications, the licensee stated that the indication growth rate and the signal amplitudes of these indications were found to be consistent with the values observed from the previous inspection (the average growth rate observed during the prior cycle for all Unit 1 SGs was 0.90 volts after 17 months of operation while the average growth rate observed in SG 1C during this unplanned outage was 0.55 volts after 12 months of operation). The licensee also stated that since the next scheduled inservice inspection is only three and one-half months away, tubes with distorted indications in the 1C SG did not need to be plugged and a high level of confidence in all SG tube integrity exists.

The staff has reviewed the data provided by the licensee. The staff notes that the maximum voltage amplitude reported for the indications with depth calls greater than 40-percent through-wall was 4.75 volts. This voltage amplitude was achieved after only 12 months of operation. Although the licensee has not implemented voltage-based SG tube plugging criteria, the staff notes that a 4.75-volt signal for axially oriented outside-diameter stress corrosion cracking (ODSCC) at the tube support plate elevations corresponds to a burst pressure of approximately 3600 psi when evaluated at the lower 95-percent lower tolerance limit of the burst pressure versus voltage correlation. Regulatory Guide 1.121 states, in part, that the margin of safety against tube rupture under normal operating conditions should not be less than three at any tube location where defects have been detected. For Braidwood, Unit 1, this pressure would correspond to approximately 3800 psi. As a result, a tube with a 4.75-volt signal due to axially oriented ODSCC at the tube support plate elevations may not withstand the loads postulated in Regulatory Guide 1.121. However, a tube with such a signal amplitude would be expected to withstand the loads for a postulated main steam line break.

The staff notes that without inspecting the other SGs it is difficult to assess the growth rates and/or the signal amplitudes for defects within those SGs. The staff estimates that it is possible that tube support plate indications similar to those found in SG 1C may be found in the other three SGs. Therefore, tubes in the other three SGs may have voltages of the magnitude observed in SG 1C. Assuming a 4.75-volt indication exists in the other SGs and taking into consideration additional growth until the next inspection, the staff expects such a tube to have adequate margin to preclude SG tube rupture during a postulated main steam line break.

In addition, the licensee has taken compensatory measures to ensure that future leakage can be readily detected and quickly mitigated. The most notable of these is the administrative requirement for plant shut down if primary-to-secondary leakage exceeds 150 gpd, or if the leak rate increases by more than 25 gpd in a one hour period.

Furthermore, the licensee committed to making changes to leak response procedures by reinforcing monitoring actions, such as an increased chemistry sampling frequency from every 4 hours to hourly for known primary-to-secondary leakage below the administrative limit. Chemistry results are then used to determine leakage trends. Use of radiation monitor indications in the control room, and portable N-16 monitors are included to help ascertain the leakage trend. Radiation monitor setpoints for the main steam lines (MSL) and steam jet air ejectors (SJAE) have been reduced to increase leakage detectability. The MSL setpoints were reduced to a point just high enough above their background readings to preclude false alarms. Lowered SJAE alarm setpoints were established using the reading seen during the approximately 300 gpd leak, subtracting background levels, and then extrapolating to an expected reading for a 150 gpd leak rate. The alert setpoint is roughly three times the background as determined from operating experience.

Reducing the operational leakage limit from 500 gpd to 150 gpd and instituting a leakage trend limit of 25 gpd, furnishes added confidence in the margin of safety for the tubes to withstand loads imposed during normal operation or postulated accidents. Enhanced leakage monitoring measures, along with operator training, should enhance operator response to a tube leak event and effectively limit any resulting release of radioactive material. These measures address the requirements of the GDC by reducing the potential for a tube failure and mitigating the consequences of a tube failure if it occurs. The staff considers the compensatory measures taken, especially the reduced administrative leakage limits and the enhanced leakage monitoring strategy, to adequately address SG operability for an interim period until the next refueling outage scheduled to start no later than March 5, 1994.

3.0 EMERGENCY CIRCUMSTANCES

By letter dated November 12, 1993, the licensee requested that this amendment be treated as an emergency since inspection of the remaining three SGs of Unit 1 would have delayed plant startup. The licensee further stated in its submittal that the need for this Emergency Amendment could not have been avoided since existing TS 3/4.4.5 did not address the tube leak discovered on October 23, 1993.

With regard to the timeliness of the licensee's submittal, a conference call was conducted on November 5, 1993, between the licensee and the NRC to discuss the results of the 1C SG tube inspection. Enforcement Discretion was verbally granted at this time. A formal request for enforcement discretion was submitted in a letter by the licensee dated November 10, 1993, and was officially granted by the NRC in a Notice of Enforcement Discretion (NOED) dated November 24, 1993.

Accordingly, pursuant to 10 CFR 50.91(a)(5), the Commission has determined that there are emergency circumstances warranting prompt approval of the proposed changes.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of that facility in accordance with the amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

This amendment has been evaluated against the standards in 10 CFR 50.92. It does not involve a significant hazards consideration because:

1. The change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

A bobbin coil eddy current inspection of all tubes in the 1C SG was conducted. None of the indications detected in this test were similar to the flaw detected in tube R49C76. Additionally, rotating pancake coil (RPC) eddy current testing of tubes in the vicinity of the leak, and RPC testing of 17 other tubes with indications identified during the bobbin coil testing, showed no indications similar to that of tube R49C76. Based on these considerations, the flaw in tube R49C76 was determined to be an isolated event.

Data from the last SG tube inservice inspection provided sufficient assurance that the Unit 1 SGs may be safely operated until the next refueling outage. Furthermore, the 1C SG has historically exhibited the most tube degradation, and the degradation identified in the tubes of the 1C SG during this inspection was consistent with that expected since the last SG inservice inspection. Therefore, not inspecting the other SGs of Unit 1 at this time does not significantly increase the probability or consequences of an accident previously evaluated.

The bobbin coil inspection also identified 116 tube support plate indications of through-wall degradation greater than 40-percent. All 116 of the tubes with these indications were plugged. Based on the results of previous inspections, this degradation was not unexpected. If tube support plate indications of degradation greater than 40-percent through-wall are assumed to exist in the remaining SGs, and additional growth until the next inspection is considered, tubes with these indications should have sufficient margin against rupture from loads imposed during normal operation or in the event of a main steamline break.

Finally, the licensee has implemented the compensatory actions previously described in Section 2.0 of this document. These include reduction of the operational leakage limit from the TS limit of 500 gpd to an administrative limit of 150 gpd, and enhanced leakage monitoring methods including additional radiation monitoring and lowered radiation monitor setpoints. These actions should enhance operator response to a tube leak event. Additionally, these measures provide added confidence that the potential for a tube rupture is reduced and that the consequences of a rupture can be mitigated should one occur.

Therefore, based on the considerations above, there is no significant increase in the probability or consequences of an accident previously evaluated as a result of this proposed change.

2. The change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

No significant or adverse changes to the plant design basis would be introduced by implementation of the proposed amendment. Any accident resulting from potential tube degradation would be bounded by the existing analysis of a SG tube rupture.

Therefore, the proposed amendment does not create a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in margin of safety.

The flaw in tube R49C76 was determined to be an isolated event. Bobbin coil eddy current testing of all tubes in the 1C SG detected no indications similar to that of tube R49C76. RPC eddy current testing of tubes in the vicinity of the leak, and RPC testing of 17 additional tubes with indications identified from the bobbin coil testing, also showed no indications similar to the flaw in tube R49C76.

Results from the recent 1C SG tube inspection are consistent with those expected since the last inservice inspection. Furthermore, data from the last inservice inspection provided added confidence that the Unit 1 SGs can be safely operated until the next refueling outage. Assuming the tube support plate indications identified in the 1C SG are present in the remaining SGs, and additional growth until the next inspection is taken into account, tubes with these indications should have sufficient margin against rupture from normal operational loads or in the event of a main steam line break.

The compensatory measures described previously, namely the reduction of the operational leakage limit and enhanced leak monitoring procedures, provide for enhanced operator response in the event of a tube leak. Additionally, the compensatory measures provide added assurance that the potential for a tube rupture is reduced and that the consequences can be mitigated should a rupture occur. The procedure for dealing with suspected leaks and for mitigating the consequences of a tube rupture are not compromised by the proposed amendment.

Based on these considerations, the proposed amendment would not involve a significant reduction in a margin of safety.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of this amendment. The state official had no comment.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has made a final no significant hazards consideration finding with respect to this amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) 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 Contributors: K. Karwoski
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H. Dawson
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Date: December 16, 1993