August 27, 1996

Mr. B. Ralph Sylvia
Executive Vice President
and Chief Nuclear Officer
Niagara Mohawk Power Corporation
Generation Business Group D-2
300 Erie Boulevard West
Syracuse, NY 13202

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT 2 (TAC NO. M95083)

Dear Mr. Sylvia:

The Commission has issued the enclosed Amendment No. 75 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit 2 (NMP-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 15, 1996, as supplemented July 18, 1996.

The amendment revises TS 4.6.2.1 "Containment Systems - Depressurization Systems - Suppression Pool" to extend the time interval for performing the containment drywell-to-suppression chamber bypass leakage tests consistent with schedules for containment integrated leak rate testing under Option B to 10 CFR Part 50, Appendix J.

As I discussed with Mr. David Baker of your licensing staff August 14, 1996, we have included a minor editorial correction to your proposed new TS 4.6.2.1.f. The correction relocates the phrase "verifying that" so as to precede both of the subsequent subparagraphs. Mr. Baker confirms that the correction is consistent with your original intent.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly <u>Federal Register</u> notice.

Sincerely,
/S/
Darl S. Hood, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-410

DFO1 1/1

Enclosures: 1. Amendment No. 75 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

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DATED: August 27, 1996

AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-69-NINE MILE POINT UNIT 2

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

August 27, 1996

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and Chief Nuclear Officer
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Darl S. Hood

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Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures:
1. Amendment No. 75 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

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NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 75
License No. NPF-69

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated March 15, 1996, as supplemented July 18, 1996, 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 1;

   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-69 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 75 are hereby incorporated into this license. Niagara Mohawk Power Corporation 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Jocelyn A. Mitchell, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 27, 1996

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 6-17	3/4 6-17*
3/4 6-18	3/4 6-18

* Overleaf page (for TS with two-side copies--text not changed)

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression pool shall be demonstrated OPERABLE:

a. By verifying the suppression pool water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression pool average water temperature to be less than or equal to 90°F, except:

   1. During testing that adds heat to the suppression pool verify the suppression pool average water temperature to be less than or equal to 105°F at least once per 5 minutes.

   2. When suppression pool average water temperature is greater than or equal to 90°F, verify at least once per hour that:

      a) Suppression pool average water temperature is less than or equal to 110°F, and

      b) THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 90°F for more than 24 hours.

   3. Following a scram with suppression pool average water temperature greater than 90°F, verify suppression pool average water temperature to be less than or equal to 120°F at least once per 30 minutes.

c. By verifying at least 20 suppression pool water temperature instrumentation channels* OPERABLE by performance of a:

   1. CHANNEL CHECK at least once per 24 hours,
   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
   3. CHANNEL CALIBRATION** at least once per 18 months,

   with the water high temperature alarm setpoints $\leq$ 90°F for 10 of the temperature instruments and $\leq$ 110°F for 10 of the temperature instruments.

---

* At least one pair in each of 10 suppression pool sectors with the alarm setpoint alternating between adjacent sectors.

** Calibration excludes sensors; sensors comparisons shall be made in lieu of calibration.

SURVEILLANCE REQUIREMENTS

4.6.2.1 (Continued)

d. At least once per 18 months by conducting a visual inspection of the exposed accessible interior and exterior surfaces of the suppression chamber.*

e. At least every outage requiring the performance of a Containment Integrated Leak Rate Test, as scheduled in conformance with the criteria specified in the 10CFR50 Appendix J Testing Program Plan described in Section 6.8.4.f, by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 3 psi and verifying that the $A/\sqrt{K}$ calculated from the measured leakage is within the specified limit of 0.0054 square feet.

   1. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission.

   2. If two consecutive tests fail to meet the specified limit, a test shall be performed at least each refueling outage until two consecutive tests meet the specified limit, at which time the original test schedule may be resumed.

   3. The provisions of Specification 4.0.2 do not apply.

f. During each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.e is not conducted, by conducting a test of the four drywell-to-suppression chamber bypass leak paths containing the suppression chamber vacuum breakers at a differential pressure of at least 3 psi and

   1. verifying that the total leakage area $A/\sqrt{K}$ contributed by all four bypass leak paths is less than or equal to 24% of the specified limit, and

   2. the leakage area for any one of the four bypass leak paths is less than or equal to 12% of the specified limit.

---

* Includes each vacuum relief valve and associated piping.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 75 TO FACILITY OPERATING LICENSE NO. NPF-69

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By application dated March 15, 1996, as supplemented by letter dated July 18, 1996, Niagara Mohawk Power Corporation (the licensee) requested an amendment to the operating license to change the Technical Specifications (TSs) for Nine Mile Point Nuclear Station, Unit 2 (NMP2). The proposed amendment would revise surveillance requirements of TS 4.6.2.1, "Depressurization Systems - Suppression Pool," to extend the time interval for performing the containment drywell-to-suppression chamber bypass leakage test from 18 months to an interval corresponding to that required by 10 CFR Part 50, Appendix J, Option B, "Performance-Based Requirements," for conducting the containment integrated leak rate test. Option B of Appendix J requires 1 test every 10 years based upon satisfactory performance of 2 previous tests. The provisions of TS 4.0.2, which would provide an extension of up to 25 percent of the specified surveillance interval, will not apply. The licensee also proposed action in the event a bypass leakage test should reveal conditions not within limits.

The proposed amendment would also add a new surveillance requirement for testing the bypass leakage paths containing the drywell-to-suppression chamber vacuum breakers. The test would be performed during refueling outages when bypass testing is not conducted. The licensee indicates that these pathways represent the greatest potential for drywell-to-suppression chamber bypass leakage.

The licensee's supplemental submittal of July 18, 1996, provided additional information in support of the initial application for amendment. It does not affect the Commission's finding of no significant hazards consideration that was published in the Federal Register (61 FR 20851, May 8, 1996).

2.0 BACKGROUND

NMP2 is a General Electric BWR/5 plant with a Mark 2 containment. The NMP2 containment design includes a reinforced concrete drywell and suppression chamber with a steel liner. Steam discharged within the drywell during a design basis loss-of-coolant accident (LOCA) would be directed to the suppression chamber through 121 vertical downcomers that pass through a diaphragm slab separating the drywell and suppression chamber. Water in the suppression chamber (i.e., the suppression pool or wetwell) absorbs the heat and condenses the steam to limit internal containment pressure to less than the design value of 310.3 kPa (45 psig). The effectiveness of the pressure suppression design would be reduced if an excessive amount of steam were to bypass the downcomers via unintended drywell-to-suppression chamber leakage pathways. Such leakage could bypass the suppression pool and increase the pressure of the suppression chamber air space, effectively increasing the backpressure against the drywell and thereby limiting or defeating the pressure suppression function of the design.

To avoid operation with excessive bypass leakage, TS 4.6.2.1.d.2 presently requires that:

> The suppression pool shall be demonstrated operable...at least once per 18 months by...conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 3 psi and verifying that the $A/\sqrt{K}$ calculated from the measured leakage is within the specified limit of 10% of 0.054 square feet. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

In lieu of the current requirements, the licensee has proposed the following:

> 4.6.2.1.e: The suppression pool shall be demonstrated operable...at least every outage requiring the performance of a Containment Integrated Leak Rate Test, as scheduled in conformance with the criteria specified in the 10 CFR 50 Appendix J Testing Program Plan described in Section 6.8.4.f, by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 3 psi and verifying that the $A/\sqrt{K}$ calculated from the measured leakage is within the specified limit of 0.0054 square feet.
>
> 1. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission.
>
> 2. If two consecutive tests fail to meet the specified limit, a test shall be performed at least each refueling outage until two consecutive tests meet the specified limit, at which time the original test schedule may be resumed.
>
> 3. The provisions of Specification 4.0.2 do not apply.

The proposed changes would retain the existing requirement in TS 4.6.2.1.d that the suppression pool shall be demonstrated operable at least once per 18 months by conducting a visual inspection of the exposed accessible interior and exterior surfaces of the suppression chamber, including each vacuum relief valve and associated piping.

The licensee has also proposed the following new surveillance for the drywell-to-suppression chamber vacuum breakers:

> 4.6.2.1.f: The suppression pool shall be demonstrated operable...during each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.e is not conducted, by conducting a test of the four drywell-to-suppression chamber bypass leak paths containing the suppression chamber vacuum breakers at a differential pressure of at least 3 psi and verifying that:
>
> 1. the total leakage area $A/\sqrt{K}$ contributed by all four bypass leak paths is less than or equal to 24% of the specified limit, and
>
> 2. the leakage area for any one of the four bypass leak paths is less than or equal to 12% of the specified limit.

The licensee indicates in its submittal that drywell-to-suppression chamber bypass leakage through penetrations other than through the drywell floor penetrations containing the drywell-to-suppression chamber vacuum breakers is negligible.

As discussed above, drywell-to-suppression chamber bypass leak tests would be conducted to a schedule consistent with containment integrated leak rate tests performed in accordance with Option B to 10 CFR Part 50, Appendix J. On August 13, 1996, the NRC issued Amendment 74 to the NMP2 operating license. The amendment revised the operating license, TSs, and associated Bases, to provide for implemention of Option B of Appendix J to 10 CFR Part 50 for Type A, B, and C leakage rate testing.

3.0 EVALUATION

The purpose of the current testing requirements is to ensure that bypass leakage is maintained within the limits assumed in design basis analysis for the most limiting steam bypass scenario such that containment design pressure would not be exceeded during an accident. The current TS leakage limit (an $A/\sqrt{K}$ of 0.0005 square meter or 0.0054 square feet.) is 10% of the design value (an $A/\sqrt{K}$ of 0.005 square meter or 0.054 square feet).

To establish the design value, a spectrum of LOCA break sizes, ranging from small to intermediate, were considered in the design basis analysis. Smaller breaks, rather than large breaks, tend to be more limiting on bypass leakage. Larger breaks result in rapid depressurization of the reactor coolant system, allowing for a larger bypass leakage flow; whereas smaller breaks result in slower depressurization, thus establishing the limiting value of leakage. As presented in Section 6.2 of the NMP2 Updated Safety Analysis Report (USAR), a small-break LOCA inside containment is the most limiting bypass event for which NMP2 is designed to mitigate. The analysis assumes that containment sprays (consisting of approximately 95% of the spray flow to the drywell and the remaining 5% to the wetwell) are manually actuated 30 minutes after the beginning of the accident and that the bypass leakage area is 0.054 square feet (0.005 square meter). The results of the analysis indicate that the containment design pressure would not be exceeded under these conditions.

The NRC staff's evaluation of the proposed changes primarily focused upon the following items: (1) the completeness of the licensee's identification of potential leakage pathways to the suppression chamber and the results of the associated analyses; (2) historical bypass leakage test results; and (3) the ability to mitigate excessive steam bypass to the suppression chamber.

3.1 Identification and Analysis of Potential Leakage Pathways

The potential leakage paths between the drywell and the suppression chamber in a Mark II containment are: 1) various system piping that passes through the suppression chamber; 2) the diaphragm slab and the seal between it and the suppression chamber; 3) downcomer penetrations; 4) Safety Relief Valve (SRV) discharge line penetrations and cracks in the lines themselves and; 5) drywell-to-suppression chamber vacuum breakers [see Washington Public Power Supply System Nuclear Project No. 2, Report No. WPPSS-74-2-R5, "Drywell to Wetwell Leakage Study," July 1974 (Study conducted by Burns and Roe for the WPPSS-2 Mark II containment)]. The NRC staff finds the licensee's submittals to be complete with respect to the identification of these potential leakage pathways and, as discussed hereafter, to provide acceptable analyses of these potential pathways.

3.1.1 System Piping Passing Through the Suppression Chamber

For the various system pipes that pass through the suppression chamber, the licensee identified the containment vent and purge lines, drywell and suppression chamber spray lines, containment atmosphere sampling lines, hydrogen recombiner lines, and reactor building floor and equipment drain lines.

The licensee indicates that all such lines can be isolated from containment by containment isolation valves that are subject to the Local Leak Rate Test (LLRT or Type C test) criteria of 10 CFR Part 50, Appendix J. An analysis was provided to address the significance of bypass leakage from these lines. The leakage corresponding to the TS allowable leakage (60% of L_a, the maximum allowable leakage) from Type B (containment penetration) and C tests (isolation valve leakage) was calculated at the design pressure of 310.3 kPa (45 psig) and was assumed to be completely bypassed to the suppression chamber. This yielded a leakage area of 0.0000055 square meter (0.0000587 square feet.) This value is approximately 1% of the TS allowable value for bypass leakage of 0.0005 square meter (0.0054 square feet). For comparison purposes, the average area corresponding to the leakage results of the first four LLRTs is 0.00000046 square meter (0.0000166 square feet).

Much of the subject piping is constructed in accordance with the standards of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Class 2 or 3. This provides reasonable assurance that the structural integrity of the piping in these systems will remain intact.

Accordingly, from its review of the piping identified by the licensee as passing through the suppression chamber, and based upon the results of the licensee's evaluations, the NRC staff concludes that leakage through cross-connected piping systems that penetrate the suppression chamber does not constitute a significant source of potential bypass leakage for NMP2.

3.1.2 Leakage Through Diaphragm Slab Penetrations

All pressure boundary penetrations between the drywell and suppression chamber, including downcomer and SRV discharge lines, are welded and have been fabricated, erected, and inspected in accordance with ASME Code, Section III. The drywell floor liner plate is similarly welded, fabricated, erected and inspected. The liner is anchored to the pedestal wall and welded to all penetrations. These fabrication and construction techniques provide reasonable assurance of structural integrity and leak tightness.

The downcomer and SRV discharge line penetrations above the drywell floor are fully visible for inspection. During its review of the application for amendment, the NRC staff requested that the licensee identify any areas that could affect drywell bypass leakage that are inaccessible and, therefore, cannot be readily inspected by visual means, or cannot be inspected at all. The licensee responded that, except for piping toward the center of the suppression chamber, the downcomer and SRV discharge piping is visible for inspection from the catwalk inside the suppression chamber. The NMP2 TSs require that a visual inspection of the exposed accessible interior and exterior surfaces of the suppression chamber be conducted at least once every 18 months.

Modifications to the drywell/suppression chamber interface are controlled by administrative procedures which require a safety evaluation of the modification by the appropriate engineering disciplines.

Accordingly, based upon the inspection requirements, component quality and fabrication standards, and the administrative procedures in place, the NRC staff finds reasonable assurance that significant leakage through the diaphragm floor or its non-vacuum breaker penetrations will not occur.

3.1.3. Drywell-to-Suppression Chamber Vacuum Breakers

The licensee identified the four drywell-to-suppression chamber vacuum breakers as the most likely source of potential bypass leakage. Because vacuum breakers are active components, the possibility exists that a breaker could inadvertently be in the open position or that excessive wear could occur on the sealing surface of the valves. Either of these scenarios represent a potential for excessive leakage. Therefore, the NRC staff finds it important that there exist reliable means to (1) detect an open vacuum breaker, (2) ensure that a breaker in the closed position remains closed unless called upon, and (3) monitor, over time, leakage through the vacuum breakers.

The NRC staff finds that at NMP2, the concern for ensuring that the vacuum breakers remain closed when necessary is adequately addressed by the following design features: (1) elastomer seals and a magnetic latch on the seating surface of the valves to assist in maintaining the valve disk-to-body seal, (2) a valve disk that tends towards the closed position under LOCA generated differential pressures, and (3) a disk position indication system, readable in the control room, to indicate vacuum breaker position.

To monitor the leakage of the vacuum breakers in order to detect potential excessive leakage, the licensee's proposed TS changes would add a leak test of all four breakers. The test would be conducted at a drywell-to-suppression chamber differential pressure of 3.0 psid (the same as required for the bypass leak test) and would be performed every refueling outage for which bypass testing is not required. The differential pressure across the vacuum breakers would be achieved by pressurizing the drywell side of the vacuum breaker or by inducing a vacuum on the suppression chamber side. The total allowed leakage for one breaker is 12% of the TS limit for the bypass leakage area and for all four breakers is 24% of the TS bypass leakage limit.

Accordingly, the NRC staff finds the vacuum breaker design and proposed surveillance adequate for detecting, in a timely manner, the potential for excessive or actual leakage through the vacuum breakers.

3.2 Past Test Results

The licensee submitted results from past bypass leakage tests. The tests are conducted by pressurizing the drywell approximately 3 psig above the suppression chamber pressure and noting the pressure decay over a time period of at least 30 minutes. This pressure decay curve is then compared to a curve corresponding to a leakage area of 0.005 square feet, and the corresponding leakage area calculated. Leakage areas determined in this manner for five previous tests ranged between 0.0 and 0.000024 square meter (0.00026 square feet). Past test results further indicate that the slope of the measured pressure versus time (pressure decay) curve for a given test is consistently close to zero.

Passive components and penetrations (downcomer penetrations, system piping, etc.) represent a lower potential for bypass leakage compared to active design features. High measured leakage would most likely occur through the drywell-to-suppression chamber vacuum breakers. Therefore, the staff finds, historically, that low bypass leakage test results and the required visual examinations and design quality of the passive leakage pathways provide a reasonable basis to conclude that passive pathways are, and will continue to be, a minor source of bypass leakage. The past bypass leakage test results for NMP2 provided by the licensee are consistent with these historical findings.

3.3 Ability of Plant Systems to Mitigate Excessive Steam Bypass

The design basis analysis for the most limiting steam bypass scenario takes credit for the drywell and wetwell sprays for mitigation. The sprays consist of two fully-redundant subsystems of the Residual Heat Removal (RHR) system that are safety-related. The sprays are initiated 30 minutes after a small-break LOCA, and an allowable leakage of 0.005 square meter (.054 square feet) is assumed in the design basis calculation. For break sizes smaller than those corresponding to a leakage area of 0.054 square feet, the allowable bypass leakage actually increases slightly while still maintaining containment pressure less than design; however, 0.054 square feet is maintained as the design value. Therefore, there exists some margin in the bypass leakage that the sprays can mitigate.

During its review, the NRC staff requested that the licensee identify any backup to containment spray and procedures used for such backup. The licensee responded that the only backup for containment spray is flooding the containment via the fire water or service water system. Use of these systems for such a purpose is not considered a design basis action, but rather is part of the Emergency Operating Procedures. Use of the systems would not be necessary for any design basis steam bypass leakage. However, the systems do provide alternative methods of delivering water to the containment.

Accordingly, the NRC staff finds that the redundant, safety-related RHR containment spray system provides a reliable means to mitigate credible excessive bypass leakage.

In summary, the NRC staff has reviewed each of the proposed changes to decrease the frequency for testing drywell-to-suppression chamber bypass leakage from once every 18 months to a frequency in accordance with Option B of 10 CFR Part 50, Appendix J. The NRC staff finds the changes acceptable on the bases that bypass leakage through passive components has historically been much lower than the TS limit, that such components/penetrations have a relatively low potential for leakage, and that the most credible source of potential bypass leakage, the drywell-to-suppression chamber vacuum breakers, will be tested on a frequency sufficient to identify, in a timely manner, excessive leakage through these components.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes surveillance requirements. 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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. (61 FR 20851) 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.

6.0 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.

Principle Contributors: Howard Dawson
Darl Hood

Date: August 27, 1996