October 28, 1987

Docket No. 50-255

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Mr. Kenneth W. Berry
Director, Nuclear Licensing
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. Berry:

SUBJECT: AMENDMENT NO. 109 TO PROVISIONAL OPERATING LICENSE NO. DPR-20 (TAC NO. 65428)

The Commission has issued the enclosed Amendment No. 109 to Provisional Operating License No. DPR-20 for the Palisades Plant. This amendment consists of changes to the Technical Specifications in response to your application dated May 4, 1987, as revised September 16, 1987.

This amendment changes the Technical Specifications for surveillance of the containment prestressing system. The new surveillance is consistent with the criteria of the pending ASME Code, Section XI, Subsection IWL, and the positions of Regulatory Guide 1.35, Revision 3, and meets your commitment for resolution of Systematic Evaluation Program Topic III-7.A.

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

Sincerely,

Original signed by

Thomas V. Wambach, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

LA/PD31:DRSP	PM/PD31:DRSP	OGC	D/PD31:DRSP
RIngram	TWambach	R. Fonner	MVirgilio
10/27/87	10/28/87	10/27/87	10/28/87

Enclosures:
1. Amendment No. 109 to License No. DPR-20
2. Safety Evaluation

cc w/enclosures:
See next page

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Palisades Plant

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

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 109
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated May 4, 1987, as revised September 16, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B. of Provisional Operating License No. DPR-20 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 109 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Martin J. Virgilio, Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 28, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 109

PROVISIONAL OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
4-29	4-29
4-29a	--
4-29b	--
4-29c	--
4-30	4-30
4-31	4-31
4-32	4-32
4-32a	--
4-33	4-33
4-34	4-34
4-35	4-35
4-36	4-36
4-37	4-37
4-38	4-38

4.5 CONTAINMENT TESTS (Contd)

4.5.3. Recirculation Heat Removal Systems

(3) Visual inspection shall be made for excessive leakage from components of the system at the interval specified in 6.15. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

b. Acceptance Criterion

The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank (approximately 44 psig).

c. Corrective Action

Repairs shall be made as required to maintain leakage within the acceptance criterion of 4.5.3b.

4.5.4 Surveillance for Prestressing System

a. Tendon inspection shall be accomplished at five-year intervals for the life of the plant. The scheduled inspection dates for all subsequent inspections may be varied by not more than plus or minus one year from the base schedule.

b. The surveillance tendons shall be randomly but representatively selected from each of the following groups:

1. A minimum of 4 dome tendons including one from each dome tendon group.
2. A minimum of 4 vertical tendons.
3. A minimum of 5 hoop tendons.

For each inspection, the tendons shall be selected on a random basis except that those tendons whose routing has been modified to clear penetrations shall be excluded from the sample.

c. During each tendon inspection, the following field testing shall be performed:

1. Lift-off readings shall be taken for each of the surveillance tendons. The tests shall include the following actions:

(a) One tendon, randomly selected from each group of tendons during each inspection, shall be subjected to essentially complete detensioning to identify broken or damaged wires.

4.5.4 Surveillance for Prestressing System (Contd)

(b) The simultaneous measurement of elongation and jacking force during retensioning shall be made at a minimum of three approximately equally spaced levels of force between the seating force and zero.

2. While the tendon is in the detensioned state, each wire in the tendon will be checked for continuity.

3. Three wires, one from each of a vertical, a hoop and a dome tendon will be removed and identified for inspection. At each successive surveillance, the wires will be selected from different tendons. Each of the inspection wires removed will be visually inspected for evidence of corrosion or other deleterious effects and samples taken for laboratory testing.

4. The sheathing filler shall be inspected visually for color and coverage and samples shall be obtained for laboratory testing.

5. Tendon anchorage hardware such as bearing plates, stressing washers, shims and buttonheads shall be visually inspected for evidence of corrosion or other deleterious effects.

d. Following the field testing of 4.5.4c, the following laboratory testing shall be done:

1. Three tensile test specimens shall be cut from each of the three inspection wires removed (one from each end and one from the middle). One additional specimen shall be cut from the wire determined by field visual inspection to have the greatest amount of corrosion. Each of the wire samples shall be tested for ultimate strength, yield strength, and elongation.

2. The sheathing filler samples shall be taken from each end of each tendon examined. Vertical tendon samples shall be taken from the lower end. Samples shall be thoroughly mixed and analyzed for reserve alkalinity, water content, and concentration of water soluble chlorides, nitrates, and sulfides. Analyses shall be performed in accordance with the procedures and within the acceptance limits specified in ASME Code Section XI, Table IWL-2525-1.

4.5 CONTAINMENT TESTS (Contd)

4.5.4 Surveillance for Prestressing System (Contd)

Procedures shall be established to minimize voids and to assure that the volume of sheathing filler removed has been replaced upon completion of the inspection and amounts documented.

e. Acceptance criteria shall be as follows:

1. The average of all measured tendon forces for each type of tendon shall be equal to or greater than the minimum required prestress level, of 584 kips per tendon for dome tendons and, 615 kips per tendon for hoop and vertical tendons. The measured force in each individual tendon shall not be less than 95% of the predicted force, or

   (a) the measured force in not more than one tendon is between 90% and 95% of the predicted force, and

   (b) The measured forces in two tendons located adjacent to the tendon in (a) above are not less than 95% of the predicted forces, and

   (c) the measured forces in all the remaining sample tendons are not less than 95% of the predicted force.

   If measured force in any tendon is less than 90% of its predicted force, the tendon shall be completely detensioned and a determination shall be made as to the cause of such an occurrence and corrective action shall be taken. In addition, all such tendons shall have their forces measured as additional tendons in the next scheduled inspection period. The Commission shall be notified in accordance with Paragraph 4.5.4f.

2. Inspection wires shall indicate no significant loss of section by corrosion or pitting.

3. Tensile test specimens cut from inspection wires shall be tested for ultimate strength. Failure at less than 11.78 kips of any one of the test samples requires the Commission be notified in accordance with specification 4.5.4f.

4. Tendon anchorage hardware shall be free of significant corrosion, pitting, cracks or other deleterious effects.

4.5 CONTAINMENT TESTS (Contd)

4.5.4 Surveillance for Prestressing System (Contd)

f. If any element of the prestressing system fails to meet the acceptance criteria of 4.5.4e., the reporting provision of Specification 6.9.2 shall apply.

4.5.5 End Anchorage Concrete Surveillance

a. A VT-1 visual examination shall be performed on the end anchorage concrete surface at the surveillance tendon anchor points for signs of cracking, popouts, spalling, or corrosion. Concrete cracks having widths greater than 0.010 inches shall be evaluated and documented.

b. The end anchorage concrete surveillance inspection interval shall be the same as tendon surveillance interval.

c. Acceptance criteria

1. Crack widths shall be measured by using optical comparators or wire feeler gauge. Movements shall be measured by using demountable mechanical extensometers.

2. Concrete anchorage areas are acceptable if no concrete cracks are wider than 0.010 inches and no signs of new or progressive deterioration since the previous inspection are found.

3. Concrete surface conditions exceeding those stated in 4.5.5c.2 above shall be evaluated for the effect on tendon and containment structural integrity. The results of evaluation shall be included in the final surveillance report.

4.5.6 Liner Plate Surveillance - Deleted

4.5.7 Penetration Surveillance - Deleted

4.5.8 Dome Delamination Surveillance

If, as a result of a prestressing system inspection under Section 4.5.4, corrective retensioning of five percent (8) or more of the total number of dome tendons is necessary to restore their liftoff forces to within the limits of Specification 4.5.4, a dome delamination inspection shall be performed within 90 days following such corrective retensioning. The results of this inspection shall be reported to the NRC.

4.5 CONTAINMENT TESTS (Cont'd)

Basis

The containment is designed for an accident pressure of 55 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a temperature of about 104°F. With these initial conditions, following a LOCA, the temperature of the steam-air mixture at the peak accident pressure of 55 psig is 283°F.

Prior to initial operation, the containment was strength-tested at 63 psig and then leak rate tested. The design objective of this preoperational leak rate test was established as 0.1% by weight per 24 hours at 55 psig. This leakage rate is consistent with the construction of the containment,[2] which is equipped with independent leak-testable penetrations and contains channels over all unaccessible containment liner welds, which were independently leak-tested during construction.

Accident analyses have been performed on the basis of a leakage rate of 0.1% by weight per 24 hours. With this leakage rate and with a reactor power level of 2530 MWt, the potential public exposure would be below 10 CFR 100 guideline values in the event of the Maximum Hypothetical Accident.[3]

The performance of a periodic integrated leak rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic leak rate test is to be performed without preliminary leak detection surveys or leak repairs and containment isolation valves are to be closed in the normal manner.

This normal manner is a coincident two-of-four high radiation or two-of-four high containment pressure signals which will close all containment isolation valves not required for engineered safety features except the component cooling lines' valves which are closed by SIS. The control system is designed on a two-channel (right and left) concept with redundancy and physical separation. Each channel is capable of initiating containment isolation.[4]

4.5 CONTAINMENT TESTS (Cont'd)

The test pressure of 28 psig for the periodic integrated leak rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the preoperational leak rate test at 28 psig. The specification provides relationships for relating in a conservative manner the measured leakage of air at 28 psig to the potential leakage of a steam-air mixture at 55 psig and 283°F. The specification also allows for possible deterioration of the leakage rate between tests by requiring that only 75% of the allowable leakage rates actually be measured. The basis for these deterioration allowances is 10 CFR Part 50, Appendix J which is believed to be conservative and will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience.

The duration of 24 hours for the integrated leak rate test is established to provide a minimum level of accuracy and to allow for daily cyclic variation in temperature and thermal radiation.

The frequency of the periodic integrated leak rate test is keyed to the refueling schedule for the reactor because these tests can best be performed during refueling shutdowns. The specified frequency is as specified in 10 CFR Part 50, Appendix J which is based on three major considerations. First is the low probability of leaks in the liner because of (a) the test of the leak tightness of the welds during erection; (b) conformance of the complete containment to a low leak rate at 55 psig during preoperational testing which is consistent with 0.1% leakage at design basis accident (DBA) conditions, and (c) absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value ($0.60L_a$) of the total leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that

4.5 CONTAINMENT TESTS (Cont'd)

an important part of the structural integrity of the containment is maintained.

The basis for specification of a total leakage rate of $0.60L_a$ from penetrations and isolation valves is specified to provide assurance that the integrated leak rate would remain within the specified limits during the intervals between integrated leak rate tests. This value allows for possible deterioration in the intervals between tests. The limiting leakage rates from the shutdown cooling system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a DBA. The test pressure (270 psig) achieved either by normal system operation or by hydrostatically testing gives an adequate margin over the highest pressure within the system after a DBA. Similarly, the hydrostatic test pressure for the return lines from the containment to the shutdown cooling system (100 psig) gives an adequate margin over the highest pressure within the lines after a DBA.[5]

A shutdown cooling system leakage of 1/5 gpm will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the DBA. The engineered safeguards room ventilation system is equipped with isolation valves which close upon a high radiation signal from a local radiation detector. These monitors shall be set at 2.2×10^5 cpm, which is well below the expected level, following a loss-of-coolant accident (LOCA), even without clad failure. The 1/5 gpm leak rate is sufficiently high to permit prompt detection and to allow for reasonable leakage through the pump seals and valve packings, and yet small enough to be readily handled by the sumps and radioactive waste system. Leakage to the engineered safeguards room sumps will be returned to the containment clean water receiver following a LOCA, via the equipment drain tank and pumps. Additional makeup water to the containment sump inventory can be readily accommodated via the charging pumps from either the SIRW tank or the concentrated boric acid storage tanks.

4.5 CONTAINMENT TESTS (Cont'd)

In case of failure to meet the acceptance criteria for leakage from the shutdown cooling system or the penetrations, it may be possible to effect repairs within a short time. If so, it is considered unnecessary and unjustified to shut down the reactor. The times allowed for repairs are consistent with the times developed for other engineered safety feature components.

A reduction in prestressing force and change in physical conditions are expected for the prestressing system. Allowances have been made in the reactor building design for the reduction and changes. The inspection results for each tendon inspected shall be recorded on the forms provided for that purpose and comparison will be made with previous test results and the initial quality control records.

Force-time records will be established and maintained for each of the tendon groups, dome, hoop and vertical. If the force measured for a tendon is less than the lower bound curve of the force-time graph, two adjacent tendons will be tested. If either of the adjacent or more than one of the original sample population falls below the lower bound of the force-time graph, an investigation will be conducted before the next scheduled surveillance. The investigation shall be made to determine whether the rate of force reduction is indeed occurring for other tendons. If the rate of reduction is confirmed, the investigation shall be extended so as to identify the cause of the rate of force reduction. The extension of the investigation shall determine the needed changes in the surveillance inspection schedule and the criteria and initial planning for corrective action.

If the force measured for a tendon at any time exceeds the upper bound curve of the band on the force-time graph, an investigation shall be made to determine the cause.

If the comparison of corrosion conditions, including chemical tests of the corrosion protection material, indicate a larger than expected change in the conditions from the time of installation or last surveillance inspection, an investigation shall be made to detect and correct the causes.(6)

4.5 CONTAINMENT TESTS (Cont'd)

The prestressing system is a necessary strength element of the plant safeguards and it is considered desirable to confirm that the allowances are not being exceeded. The technique chosen for surveillance is based upon the rate of change of force and physical conditions so that the surveillance can either confirm that the allowances are sufficient, or require maintenance before minimum levels of force or physical conditions are reached.

The end anchorage concrete is needed to maintain the prestressing forces. The design investigations concluded that the design is adequate. The prestressing sequence has shown that the end anchorage concrete can withstand loads in excess of those which result when the tendons are anchored. At the time of initial pressure testing, the containment building had been subjected to temperature gradients equivalent to those for normal operating conditions while the prestressing tendon loads are at their maximum.

However, after the initial pressure test both concrete creep and prestressing losses increase with the greatest rapidity and result in a redistribution of the stresses and a reduction in end anchor force. Because of the importance of the containment and the fact that the design was new, it was considered prudent to continue the surveillance after the initial period.(7)

4.5 CONTAINMENT TESTS (Cont'd)

Containment dome delamination inspections performed in 1970 and 1982 have confirmed that no concrete delamination has occurred. The possibility that delamination might occur in the future is remote because dome tendon prestress forces gradually diminish through normal tendon relaxation and concrete strength normally increases over time. To account for this remote possibility, however, an additional delamination inspection will be performed in the event that 5% or more of the installed tendons must be retensioned to compensate for excessive loss of prestress. This inspection would be to confirm that any systematic excessive prestress loss did not result from delamination and that the retensioning process did not result in delamination.

References

(1) FSAR, Section 5.1.2; Updated FSAR section 5.8.2.

(2) FSAR, Section 5.1.8; Updated FSAR section 5.8.8

(3) FSAR and Updated FSAR 14.22

(4) FSAR, Section 8.5.4; Updated FSAR Section 8.5.1.2

(5) FSAR and Updated FSAR Section 6.2.3

(6) FSAR, Section 5.1.8.4; FSAR, Amendment No 14, Question 5.37; and Updated FSAR Section 5.8.8.3.

(7) Updated FSAR, Section 5,8.8.6

(8) 10 CFR Part 50, Appendix J.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 109 TO PROVISIONAL OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

Consumers Power Company (CPCo or the licensee) submitted a Technical Specification Change Request on May 4, 1987, addressing changes to the surveillance of the containment prestressing system. The proposed changes were required by a CPCo commitment, documented in NUREG-0820, October 1982, to develop acceptance criteria consistent with the proposed ASME Code or equivalent.

The licensee committed to meet the pertinent criteria of ASME Code, Section XI, Subsection IWL, and NRC Regulatory Guide 1.35, Revision 3. The licensee has also submitted a revised Technical Specification Change Request on September 16, 1987. The revised document addresses staff questions raised on CPCO's May 4, 1987, submittal.

2.0 EVALUATION

The NRC staff has reviewed the following CPCo documents:

1. Request for Change to the Technical Specifications, License DPR-20, (May 4 and September 16, 1987, versions).

2. Proposed Technical Specification Page Changes (May 4 and September 16, 1987, versions).

3. Palisades Plant, 10-year Tendon Surveillance Report, dated November 17, 1981.

4. Palisades Plant, 5-year Tendon Surveillance Report, dated March 26, 1976.

5. Palisades Plant, 3-year Tendon Surveillance Report, dated April 29, 1974.

6. Palisades Plant, 1-year Tendon Surveillance Report, dated March 2, 1972.

The proposed changes to the Technical Specifications for tendon surveillance requirements delete certain requirements which have been accomplished by the 1, 2, 5, and 10-year tendon surveillances, establish a new 5-year surveillance schedule, and acceptance requirements identified in the proposed Subsection IWL, Section XI of the ASME Code and Revision 3 to NRC Regulatory Guide 1.35.

The minimum number of tendons to be included in each 5-year surveillance will be four dome tendons (including one from each group), four vertical tendons, and five hoop tendons. One tendon from each group of tendons will be subjected to complete detensioning to identify broken or damaged wires. Also, elongation and related jacking force measurements will be made at three equally spaced levels of force from zero to the final seating force, during the retensioning of the detensioned tendons.

Section 4.5.d of the specification addresses the required laboratory tests for the removed tendon wires and filler grease. Specific wires are identified for the tensile tests to determine their yield and ultimate strengths and elongation. The filler grease from the examined tendons will be analyzed to determine if the alkalinity, water content, and concentration of water soluble chlorides, nitrates and sulfides meet the allowable limits identified in the Technical Specifications. Also, proper procedures will be implemented to avoid grease voids during the reinstallation of the examined tendons.

The reporting requirements of Regulatory Guide 1.35, Revision 3, will be followed when the acceptance criteria are exceeded. Generally, the reporting requirement provisions of 10 CFR 50.73 will be followed. However, the presence of grease voids and free water will be evaluated and reported in the post-inspection special report required by Section 6.9.3.3. This report will be submitted 90 days after completion of the inspection.

The NRC staff raised four questions regarding the licensee's submittal of May 4, 1987. The first question addressed concern of the fact that all of the requirements of NRC Regulatory Guide 1.35, Revision 3, and those of ASME, Section XI, Subsection IWL (upcoming revision) may not be exactly identical. Special concern was identified regarding the reporting requirements of Section C.7.1.3 and the grease voids requirements of Section C.7.4 of Regulatory Guide 1.35, Revision 3. The second question focused on the requirements of Regulatory Guide 1.35, Revision 3, Section C.2.2, which addresses the minimum population of tendons to be evaluated for each type of tendon within the structure. The third question addressed the NRC staff concern that the changes may have reduced the level of surveillance identified in the original Technical Specifications. The fourth question raised concern that a third-year surveillance observation may have not been followed-up during the surveillance program. Specifically, the third-year surveillance highlights reported the presence of water in the end cap of a dome tendon (D2-53). The following surveillance report did not address any follow-up action on the third-year surveillance observation.

The licensee has addressed the NRC staff questions on Reference 1 in their Reference 2. The staff has found the changes identified in Reference 2 acceptable in their entirety. The licensee is complying with the requirements of Regulatory Guide 1.35, Revision 3, Sections C.7.1.3, C.7.4, and C.2.2. Thus, the licensee has addressed and resolved the first three staff questions. Also, the licensee has indicated that additional evaluations of the dome tendon D2-53 were not required because the laboratory evaluation of the water and grease indicated acceptable levels for chlorides, nitrates and sulfides, and no evidence of corrosion was observed in the field inspections.

CPCo has revised their Technical Specifications on surveillance of the prestressing system for the Palisades Plant to meet the latest industry and regulatory requirements contained in the ASME, Section XI, Subsection IWL.

CPCo has also addressed all NRC staff concerns regarding the provisions of NRC Regulatory Guide 1.35, Revision 3, on the minimum number of tendons designated for surveillance, grease voids reporting requirements, evaluation of water and grease, and other minor clarifications. The revised Technical Specifications, along with the additional commitments, meet and/or exceed any technical requirement in the previous Technical Specifications for tendon system surveillance except for detensioning only one tendon of each group. Therefore, the NRC staff finds the proposed changes to the Technical Specifications on surveillance of the prestressing system for the Palisades Plant submitted on September 16, 1987, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a surveillance requirement. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter dated May 4, 1987, from Kenneth W. Berry to NRC, with attachments.

2. Letter dated September 16, 1987, from Kenneth W. Berry to NRC, with attachments.

Dated: October 28, 1987

Principal Contributor: Frank Rinaldi, NRR/ESGB