July 26, 1985

Docket Nos. 50-266
and 50-301

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Mr. C. W. Fay, Vice President
Nuclear Power Department
Wisconsin Electric Power Company
231 West Michigan Street, Room 308
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

The Commission has issued the enclosed Amendment Nos. 95 and 99 to Facility Operating License Nos. DPR-24 and DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated October 26, 1984.

These amendments modify steam generator inservice inspection requirements under Technical Specification 15.4.2.A. The amendments also delete a limiting condition for operation for the auxiliary feedwater system. Changes to reporting requirements to make them consistent with 10 CFR 50.73 are also included. As discussed with your staff, a typographical error was discovered in your proposed revision to Table 15.4.2-1. Specifically, the reporting requirement of inservice inspection examination results in category C-3 references 10 CFR 50.73ii rather than the correct section of the rule, 10 CFR 50.73(a)(2)(ii). We have corrected this error.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next n=bi-weekly Federal Register notice.

Sincerely,

Timothy G. Colburn, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 95 to DPR-24
2. Amendment No. 99 to DPR-27
3. Safety Evaluation

cc w/enclosures:
See next page

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 95
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated October 26, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 95 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Edward J. Butcher

Edward J. Butcher, Acting Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 26, 1985

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated October 26, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 99 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 20 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Edward J. Butcher

Edward J. Butcher, Acting Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 26, 1985

ATTACHMENT TO LICENSE AMENDMENTS NO. 95 AND 99
TO FACILITY OPERATING LICENSE NO. DPR-24 AND DPR-27
DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.3.4-1	15.3.4-1
15.4.2-1	15.4.2-1
15.4.2-1a	15.4.2-2
15.4.2-1b	15.4.2-3
15.4.2-1c	15.4.2-4
15.4.2-2	15.4.2-5
Table 15.4.2-1	Table 15.4.2-1
15.6.10-1	15.6.10-1

15.3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of steam and power conversion system.

Objective

To define conditions of the steam and power conversion system steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

A. When the reactor coolant is heated above 350°F the reactor shall not be taken critical unless the following conditions are met:

1. A minimum steam-relieving capability of eight (8) main steam safety valves available, except for low power physics testing.

2. Auxiliary Feedwater System

    a. Two Unit Operation - All four auxiliary feedwater pumps together with their associated flow paths and essential instrumentation shall be operable.

    b. Single Unit Operation - Both motor driven auxiliary feedwater pumps and the turbine driven auxiliary feedwater pump associated with that unit together with their associated flow paths and essential instrumentation shall be operable.

15.4.2 IN-SERVICE INSPECTION OF SAFETY CLASS COMPONENTS

Applicability

Applies to in-service inspection of Safety Class Components.

Objectives

To provide assurance of the continuing integrity of the safety class systems.

Specifications

A. Steam Generator Tube Inspection Requirements

1. Tube Inspection

Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered a tube inspection.

2. Sample Selection and Testing

Selection and testing of steam generator tubes shall be made on the following basis:

(a) One steam generator of each unit may be selected for inspection during inservice inspection in accordance with the following requirements:

1. The inservice inspection may be limited to one steam generator on an alternating sequence basis. This examination shall include at least 6% of the tubes if the results of the first or a prior inspection indicate that both generators are performing in a comparable manner.

2. When both steam generators are required to be examined by Table 15.4.2-1 and if the condition of the tubes in one generator is found to be more severe than in the other steam generator of a unit, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator with the more severe condition.

(b) The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements specified in Table 15.4.2-1. The results of each sampling examination of a steam generator shall be classified into the following three categories:

Category C-1: Less than 5% of the total number of tubes examined are degraded but none are defective.

Category C-2: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective or one tube to not more than 1% of the sample is defective.

Category C-3: More than 10% of the total number of tubes examined are degraded, but none are defective or more than 1% of the sample is defective.

In the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by the prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the nominal tube wall thickness.

(c) Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.

(d) In addition to the sample size specified in Table 15.4.2-1, the tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded.

(e) During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where imperfections were detected during the prior examination.

3. Examination Method and Requirements

The examination method shall meet the intent of the requirements in ASME Section XI Appendix IV. This includes equipment, personnel and procedure requirements, certification and calibration along with records and reports. The actual technique may be the latest industry accepted technique, provided the flaw detection capability is as good or better than the technique endorsed by the code in effect per Technical Specification 15.4.2.B.1. This allows the use of improvements in inspection techniques that were not included in the code in effect. However, it means that word-for-word compliance with Appendix IV of ASME Section XI may not be possible.

4. Inspection Intervals

(a) Inservice inspections shall not be more than 24 calendar months apart.

(b) The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 15.4.2.A.4(a) are not exceeded.

(c) If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals.

(d) If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 15.4.2-1 requires that a third sample examination must be performed, and the results of this fall in category C-3, the inspection frequency shall be reduced to not more than 20 months intervals. The reduction shall apply until a subsequent inspection demonstrates that a third sample examination is not required.

(e) Unscheduled inspections shall be conducted in accordance with Specifications 15.4.2.A.2 on any steam generator with primary-to-secondary tube leakage exceeding Specification 15.3.1.D.4. All steam generators shall be inspected in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineered safeguards, or a main steam line or feedwater line break.

5. Acceptance Limits

(a) Definitions:

Imperfection is an exception to the dimension, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

Degradation means a service induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

Degraded Tube is a tube that contains imperfections caused by degradation greater than 20% of the nominal tube wall thickness.

Defect is an imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.

Plugging Limit is the imperfection depth beyond which the tube must be removed from service or repaired, because the tube may become defective prior to the next scheduled inspection. The plugging limit is 40% of the nominal tube wall thickness.

6. Corrective Measures

All tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired by a process such as sleeving* prior to return to power from a refueling or inservice inspection condition. Sleeved tubes having sleeve degradation exceeding 40% of the nominal sleeve wall thickness shall be plugged.

7. Reports

(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission as soon as practicable.

(b) The complete results of the steam generator tube inservice inspection shall be included in the Annual Results and Data Report for the period in which the inspection was completed.

Reports shall include:

1. Number and extent of tubes inspected.

2. Location and percent of all thickness penetration for each indication.

3. Identification of tubes plugged or repaired.

(c) Reports required by Table 15.4.2-1 - Steam Generator Tube Inspection shall provide the information required by Specification 15.4.2.A.7(b) and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. The report shall be submitted to the Commission prior to resumption of plant operation.

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*Brazed joints shall not be employed. Tubes previously subject to explosive plugging shall not be sleeved.

B. Inservice Inspection of Safety Class Components Other than Steam Generator Tubes

1. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) modified by Section 50.55a(b), except where specific written relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

2. Containment isolation valves will be tested in accordance with Technical Specification 15.4.4 instead of Section IWV-3420, Valve Leak Rate Test.

<u>Basis</u>

The steam generator tube inspection requirements are based on the guidance given in NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." ASME Section XI Appendix IV is being used for defining the basic requirements or the inspection method. However, at the present time, changes and improvements in steam generator eddy current inspection are occurring faster than the code can be revised. Thus, in order to ensure that the best possible exam of the tubing and/or sleeves is being done, the technique utilized will, in general, be the latest industry-accepted technique. This means that complete word-for-word compliance with Appendix IV may not be possible. However, the basic requirements and intent will be met, to the extent practical.

As stated in 15.4.2.B.1, safety class components, other than the steam generator tubing, will be inspected in accordance with ASME Section XI. The code edition/addenda utilized for the inspection interval will be as defined in 10 CFR 50. The same code is utilized for both Unit 1 and Unit 2. Safety-related components are classified as safety Class 1, 2, or 3. The code boundaries are defined based upon the following documents:

(a) Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants."

(b) American National Standard N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

(c) Point Beach Nuclear Plant Units 1 & 2 Final Safety Analysis Report.

Code classified components are tabulated showing each specific examination area and the examination requirements in an inspection interval long-term plan. This plan is completely revised for each ten-year inspection interval.

TABLE 15.4.2-1

STEAM GENERATOR TUBE INSPECTION PER UNIT
POINT BEACH UNITS 1 & 2

1ST SAMPLE EXAMINATION			2ND SAMPLE EXAMINATION		3RD SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (S.G.) S=3(N/n)% where: N is the number of steam generators in the plant = 2 n is the number of steam generators inspected during an examination	C-1	Acceptable for continued service	N/A	N/A	N/A	N/A
	C-2	Plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for continued service	N/A	N/A
			C-2	Plug or repair tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for continued service
					C-2	Plug or repair tubes exceeding plugging limit. Acceptable for continued service
					C-3	Perform action required under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
	C-3	Inspect essentially all tubes in this S.G., plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).	C-1 in other S.G.	Acceptable for continued service	N/A	N/A
			C-2 in other S.G.	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			C-3 in other S.G.	Inspect essentially all tubes in S.G. & plug or repair tubes exceeding the plugging limit. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).	N/A	N/A

Unit 1-Amend. No. 71,95
Unit 2-Amend. No. 76,99

15.6.10 PLANT OPERATING RECORDS

Specification

Records and logs relative to the following items shall be retained for six (6) years unless a longer period is required by applicable regulations.

A. Records of normal plant operation, including power levels and periods of operation at each power level.

B. Records of principal maintenance activities, including inspection, repair, substitution or replacement of principal items of equipment pertaining to nuclear safety.

C. Records of reportable events.

D. Records of periodic checks, inspections and calibrations performed to verify that surveillance requirements are being met.

E.* Records of new and spent fuel inventory and assembly histories.

F.* Records of changes made to the plant and to plant drawings as described in the FFDSAR.

G.* Records of plant radiation and contamination surveys.

H.* Records of off-site environmental monitoring surveys.

I.* Records of radiation exposure of all plant personnel, including all contractor personnel and visitors who enter radiation control areas in the plant.

J.* Records of radioactivity levels in liquid and gaseous wastes released to the environment and dilution of these wastes.

K. Records of any special reactor tests or experiments.

L. Records of changes made in the Operating Procedures.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 95 AND 99 TO

FACILITY OPERATING LICENSE NOS. DPR-24 AND DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter dated October 26, 1984, Wisconsin Electric Power Company submitted proposed amendments to the Point Beach Nuclear Plant Technical Specifications. The proposed amendments revise Technical Specification 15.4.2.A and Table 15.4.2-1 to clarify that selection of one steam generator for inservice inspection is acceptable, to allow utilization of state-of-the-art inspection techniques, and to reflect current reporting requirements for inspection results. The corresponding Technical Specification Bases are also revised. The proposed amendments also revised Technical Specification 15.3.4.A.2.c. by deleting a limiting condition for operation with respect to the auxiliary feedwater system.

This safety evaluation is a review of the requested changes and their impact on the operation and administration of plant activities.

2.0 EVALUATION

Proposed Change to Technical Specification 15.4.2.A.2.(a)

Description of Change

The criteria for the number of steam generators to be inspected have been revised to indicate that selection of one steam generator is permissible rather than required.

Evaluation

Existing Technical Specifications require one steam generator per unit to be inspected during each unit's prescribed inspection period. The proposed revision permits the inspection of either one or both steam generators of a given unit during a prescribed inspection period, thereby allowing an assessment of the conditions of both steam generators prior to the discovery of degradation in either steam generator. If the option of inspecting both steam generators is selected, Table 15.4.2-1 of Technical Specification 15.4.2 would allow for a reduction in the number of tubes inspected per steam generator by a factor of two; however, the total number of tubes inspected would remain the same, thereby providing the same level of assurance that degraded conditions could be detected. This revision is consistent with existing requirements in Westinghouse Standard Technical Specifications (NUREG-0452) Revision 4.

This change is acceptable.

Proposed Change to Technical Specification 15.4.2.A.3

The existing Technical Specification requires that the examination method be that prescribed in ASME Section XI Appendix IV. This is consistent with the requirements contained in the Westinghouse Standard Technical Specifications (NUREG-0452). The proposed revision would allow the use of alternate methods within the following prescribed bounds:

1. Equipment, personnel, and procedure requirements must be consistent with the ASME Section XI Appendix IV requirements.

2. Certification and calibration requirements must be consistent with ASME Section XI Appendix IV requirements.

3. Records must be consistent with ASME Section XI Appendix IV requirements.

4. Reporting requirements remain unchanged.

5. Flaw detection capability must be equivalent or superior to the methods prescribed in ASME Section XI Appendix IV.

10 CFR 50.55a(b)(2)(iii) states, "If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB-2000 (of ASME Section XI), the inservice inspection program for steam generator tubing shall be governed by the requirements in the technical specifications."

Based on the fact that any alternative techniques selected must be at least as sensitive to the presence of flaws as existing prescribed techniques, adequate assurance is provided that equivalent degradation detection capability will exist. The requirements for equivalency in administrative, qualification, certification, calibration, and reporting requirements assures that the inspection program controls will be equivalent to existing requirements. Thus, the proposed revision will not result in program degradation and will likely result in program enhancement.

As deviations from prescribed requirements are permitted by 10 CFR 50.55a(b)(2)(iii), this change is acceptable.

Proposed Change to Technical Specification 15.4.2.A.7

Description of Change

Reporting requirements for steam generator inservice inspections have been revised to establish consistency with Westinghouse Standard Technical Specifications (NUREG-0452).

Evaluation

The proposed changes establish a greater degree of consistency between the Point Beach Technical Specifications and Westinghouse Standard Technical Specifications (NUREG-0452). The result of this change will be more complete and timely reporting of the results of inspection results.

This change is acceptable.

Proposed Change to Technical Specification 15.6.10 and Table 15.4.2-1

Description of Change

The reporting requirements for inspection results falling into the category of "more than 10% of the total number of tubes examined are degraded, but none are defective" or "more than 1% of the sample is defective" have been updated to reflect the requirements of 10 CFR 50.73(a)(2)(ii). Technical Specification 15.6.10 changes reportable occurrence records to reportable events.

Evaluation

These changes reflect the updated reporting requirements of 10 CFR 50.73(a)(2)(ii) and are acceptable.

Proposed Change to Bases

The Basis of Technical Specification 15.4.2 is revised to reflect state-of-the-art steam generator tube inspection requirements consistent with program control requirements of ASME Section XI Appendix IV requirements, and to reflect the applicability of ASME Section XI inspection requirements for safety class components other than steam generators.

Evaluation

The change to the Basis of Technical Specification 15.4.2 is consistent with the proposed revisions to the specification.

This change is acceptable.

Proposed Change To TS 15.3.4.A.2.c.

Description of Changes

Specification 15.3.4.A.2.c, which provides for the temporary closing of the motor driven auxiliary feedwater pump discharge valves during operations, has been deleted.

Evaluation

By letter dated July 6, 1984, the licensee notified the Commission of the completion of a system modification to the auxiliary feedwater system which serves to provide added assurance of automatic initiation of auxiliary feedwater flow from the motor-driven auxiliary feedwater pumps without operator action. The licensee's proposed modification submitted by letter dated June 20, 1983, was reviewed and approved by the Commission by letter dated September 15, 1983.

The modification concerned the auxiliary feedwater system motor-operated discharge valve operating logic and provides assurance that automatic initiation of auxiliary feedwater flow to an affected unit's steam generators will occur even if the motor-operated pump discharge valves are initially closed.

The licensee, per their commitment in the July 6, 1984, letter, has included the required surveillance and testing of the modified motor-operated valve control logic in the appropriate Technical Specification on a monthly frequency. In addition, the licensee stated that valve functional testing is performed per the requirements of Section XI of the ASME Code and that Engineered Safety Features response testing of the valves is performed at each refueling to provide added assurance that the valve will perform its safety function in the event of an accident.

Since the system modifications ensure that automatic initiation of auxiliary feedwater will occur even if the motor-operated valves are initially closed and the valve control logic is tested monthly, the provisions of Technical Specification 15.3.4.A.2.c are no longer necessary to permit temporary closing of the discharge valves.

The staff has concluded that this change is acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9). The amendments also involve changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR §51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

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

Date: July 26, 1985

Principal Contributor:
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