April 13, 1990

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Docket No. 50-305

Mr. Ken H. Evers
Manager - Nuclear Power
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Evers:

SUBJECT: AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NO. 73072)

The Commission has issued the enclosed Amendment No. 86 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your applications dated April 28, August 15, November 10, and December 20, 1989.

The amendment addresses organizational changes, corrects typographical errors and inconsistencies, and clarifies the intent of certain technical specifications.

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

Sincerely,

/s/

Michael J. Davis, Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 86 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

Office:	LA/PDIII-3	PM/PDIII-3	PD/PDIII-3	OGC-WF1
Surname:	PKreutzer	MDavis	JHannon	
Date:	03/20/90	03/21/90	04/13/90	03/26/90

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PDR ADOCK 05000305
P PDC

Mr. Ken H. Evers
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

cc:
David Baker, Esquire
Foley and Lardner
P. O. Box 2193
Orlando, Florida 32082

Glen Kunesh, Chairman
Town of Carlton
Route 1
Kewaunee, Wisconsin 54216

Mr. Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, Wisconsin 54216

Chairman
Public Service Commission of Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

Attorney General
114 East, State Capitol
Madison, Wisconsin 53702

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
Route #1, Box 999
Kewaunee, Wisconsin 54216

Regional Administrator - Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Robert S. Cullen
Chief Engineer
Wisconsin Public Service Commission
P.O. Box 7854
Madison, Wisconsin 53707

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

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. DPR-43

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

   A. The applications for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated April 28, August 15, November 10, and December 20, 1989 comply 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 applications, 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. DPR-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.86 , are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and is to be implemented within 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 13, 1990

ATTACHMENT TO LICENSE AMENDMENT NO.86

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

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

REMOVE	INSERT
TS v	TS v
TS vi	TS vi
TS viii	TS viii
TS 3.10-10	TS 3.10-10
TS 4.1-1	TS 4.1-1
Table TS 4.1-3 (page 1 of 2)	Table TS 4.1-3 (page 1 of 2)
TS 4.2-7	TS 4.2-7
TS 4.2-11	TS 4.2-11
TS 4.4-13	TS 4.4-13
TS 5.3-1	TS 5.3-1
TS 6-5	TS 6-5
TS 6-8	TS 6-8
TS 6-11	TS 6-11
TS 6-11a	TS 6-11a
TS 6-12	TS 6-12

LIST OF TABLES

LIST OF FIGURES

Amendment No. ~~64~~,~~72~~,~~76~~, 86

An upper bound envelope for F^N_Q defined by specification 3.10.b.1 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on this upper bound envelope indicate the peak clad temperatures remain below the 2200°F limit.

The $F^N_Q(Z)$ limits of specification 3.10.b.1.A include consideration of enhanced fission gas release at high burnup, off-gassing (release of absorbed gases), and other effects in fuel supplied by Advanced Nuclear Fuels Company. The result of these analyses show that no additional burnup dependent penalty need be applied for Advanced Nuclear Fuels Company fuel (7).

When a F^N_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In specification 3.10.b.1 and 3.10.b.4 F^N_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

<u>$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor</u>

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

4.0 SURVEILLANCE REQUIREMENTS

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To assure that instrumentation shall be checked, tested and calibrated, and that equipment and sampling tests shall be conducted at sufficiently frequent intervals to ensure safe operation.

Specification

a. Calibration, testing, and checking of protective instrumentation channels and testing of logic channels shall be performed as specified in Table TS 4.1-1.
b. Equipment and sampling tests shall be conducted as specified in Table TS 4.1-2 and TS 4.1-3.
c. Specified time intervals may be adjusted plus or minus 25% to accommodate normal test procedures. Schedules subject to limits of tables TS 4.1-2,-3.
d. Whenever containment integrity is not required, only the asterisked items in Tables TS 4.1-1, TS 4.1-2, TS 4.1-3 are applicable.
e. Discrepancies noted during surveillance program testing will be recorded and corrective actions will be documented in accordance with Section 6 of the Technical Specifications.

Basis

Check

Failures such as blown instrument fuses, defective indicators, or faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures, are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

TABLE 4.1-3

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS
(Page 1 of 2)

Equipment Tests***	Test	Frequency	Maximum Time Between Test (Days)
1. Control Rods	Rod drop times of all full length rods	Each refueling outage	N.A.
	Partial movement of all rods	Every 2 weeks	17
1a. Reactor Trip Breakers	Independent Test(1) Shunt & Undervoltage Trip Attachments	Monthly	37
1b. Reactor Coolant Pump Breakers-Open-Reactor Trip	Operability	Each refueling outage	N.A.
1c. Manual Reactor Trip	Open Trip Reactor(2) Trip & Bypass Bkr	Each refueling outage	N.A.
2. Deleted			
3. Deleted			
4. Containment Isolation Trip	Operability	Each refueling outage	N.A.
5. Refueling System Interlocks	Operability	Prior to fuel movement each refueling outage	N.A.
6. Deleted			
7. Fire Protection Pump and Power Supply	*Operability	Monthly	37
8. RCS Leak Detection	Operability	Weekly	8
9. Diesel Fuel Supply	*Fuel Inventory	Weekly	8
10. Turbine Stop and Governor Valves	Operability	Annually	365
11. Fuel Assemblies	Visual Inspection	Each refueling outage	N.A.
12. Guard Pipes	Visual Inspection	Each refueling outage	N.A.

5. Reports

a. Following each inservice inspection of steam generator tubes, if there are any tubes requiring plugging or repairing, the number of tubes plugged or repaired shall be reported to the Commission within 30 days.

b. The results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.
2. Location and percent of wall-thickness penetration for each indication of a degradation.
3. Identification of tubes plugged.
4. Identification of tubes repaired.

c. Results of a steam generator tube inspection which fall into Category C-3 require prompt (within 4 hours) notification of the Commission consistent with 10CFR50.72(b)(2)(i). A written followup report shall be submitted to the Commission consistent with Specification 4.2.b.5.a, using the Licensee Event Report System, to satisfy the intent of 10 CFR 50.73(a)(2)(ii).

Basis

Technical Specification 4.2.b.5

Category C-3 inspection results are considered abnormal degradation to a principal safety barrier and are therefore reportable under 10 CFR 50.72(b)(2)(i) and 10 CFR 50.73(a)(2)(i).

References

(1) WCAP 7832: "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions".

(2) E.W. James, WPSC, to A. Schwencer, NRC, dated September 6, 1977.

(3) WCAP 11643, Kewaunee Steam Generator Sleeving Report, Revision 1, November 1988 (Proprietary).

Auxiliary Building Special Ventilation System (TS 4.4.d)

Demonstration of the automatic initiation capability is necessary to assure system performance capability.[4]

Periodic checking of the inlet heaters and associated controls for each train will provide assurance that the system has the capability of reducing inlet air humidity so that charcoal adsorber efficiency is enhanced.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline.

Vacuum Breaker Valves (TS 4.4.e)

The vacuum breaker valves are 18" butterfly valves with air to open, spring to close operators. The valve discs are center pivot and rotate when closing to an EPT base material seat. When closed, the disc is positioned fully on the seat regardless of flow or pressure direction. Testing these valves in a direction opposite to that which would occur POST-LOCA verifies leakage rates of both the vacuum breaker valves and the check valves downstream.

References:

(1) Updated FSAR Section 5.2
(2) Updated FSAR Section 14.3
(3) 10CFR Part 50, Appendix J
(4) Updated FSAR Section 9.6
(5) Letter from Darrell G. Eisenhut to Carl W. Giesler dated September 30, 1982

5.3 REACTOR

Applicability

Applies to the reactor core and the Reactor Coolant System.

Objective

To define those design features which are essential in providing for safe system operations.

Specifications

a. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods. (1)

2. The average enrichment of the initial core is a nominal 2.90 weight percent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.40 weight percent of U-235. (2)

3. Reload fuel will be similar in design to the initial core.

4. Burnable poison rods are incorporated in the initial core. There are 704 poison rods in the form of 8, 12 and 16 rod clusters, which are located in vacant rod cluster control tubes. The burnable poison rods consist of borosilicate glass clad with stainless steel.

5. There are 29 full-length Rod Cluster Control (RCC) assemblies in the reactor core. The full-length RCC assemblies contain a 142-inch length of silver-indium-cadmium alloy clad with stainless steel.

i. Review of all Reportable Events
j. Review of changes to the Process Control Program, the Offsite Dose Calculation Manual, and the Radiological Environmental Monitoring Manual.

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend to the Plant Manager approval or disapproval of items considered under 6.5.1.6a through e above.
b. Make determinations with regard to whether or not each item considered under 6.5.1.6 above constitutes an unreviewed safety question.
c. Provide immediate notification in the form of draft meeting minutes to the Manager - Nuclear Power and the Chairman-Nuclear Safety Review and Audit Committee of disagreement between the PORC and the Plant Manager. The Plant Manager shall have responsibility for resolution of such disagreements.

RECORDS

6.5.1.8 Minutes shall be kept of all meetings of the PORC and copies shall be sent to the Manager - Nuclear Power and the Chairman - Nuclear Safety Review and Audit Committee.

6.5.2 CORPORATE SUPPORT STAFF (CSS)

FUNCTION

6.5.2.1 The CSS shall function to provide engineering, technical and quality assurance activities in support of the Kewaunee Plant Staff.

a. At least three technically qualified persons who are not members of the plant staff.

b. One member from the supervisory staff of the plant.

c. At least two qualified non-company affiliated technical consultants.

d. Plus in-house staff management advisors as required.

The Committee membership and its Chairman and Vice Chairman shall be appointed by the Senior Company Officer to whom the NSRAC reports. Each member of the NSRAC shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum shall be in one or more areas given in 6.5.3.1.

ALTERNATES

6.5.3.3 Alternate members shall be appointed by the NSRAC Chairman, upon approval by the Assistant Vice President - Nuclear Power, to serve on a temporary basis; however, no more than two alternates shall participate in NSRAC activities at any one time.

CONSULTANTS

6.5.3.4 Consultants may be utilized as determined by the Chairman - NSRAC to provide expert advice to the NSRAC.

MEETING FREQUENCY

6.5.3.5 The NSRAC shall meet at least once every six months.

g. Any other area of plant operation considered appropriate by the NSRAC or the Senior Company Officer to whom the NSRAC reports.

h. The radiological environmental monitoring program and the results thereof at least annually.

i. The Offsite Dose Calculation Manual and implementing procedures at least once every two years.

j. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once every two years.

AUTHORITY

6.5.3.9 The NSRAC shall report to a Senior Company Officer and shall advise the Officer on those areas of responsibility specified in Section 6.5.3.7 and 6.5.3.8.

RECORDS

6.5.3.10 Records of NSRAC activities shall be prepared, approved and distributed as follows:

a. Minutes of each NSRAC meeting forwarded to the Senior Company Officer to whom the NSRAC reports within 14 days following each meeting.

b. Reports of reviews required by Section 6.5.3.7e, f, g and h above, forwarded to the Senior Company Officer to whom the NSRAC reports within 14 days following completion of the review.

c. Reports of audits performed by NSRAC shall be forwarded to the Senior Company Officer to whom the NSRAC reports and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENTS

Actions

6.6.1 The following actions shall be taken for Reportable Events:

a. The Commission shall be notified and a report submitted pursuant to the requirements of 10CFR50.73,and

b. Each Reportable Event shall be reviewed by PORC, and the results of this review shall be submitted to NSRAC and the Assistant Vice President - Nuclear Power.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The reactor shall be shutdown and operation shall not be resumed until authorized by the Commission.

b. The Report shall be prepared in accordance with Section 6.6 of the Technical Specifications.

6.8 PROCEDURES

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet the requirements and recommendations of Section 5.2.2, 5.2.5, 5.2.15 and 5.3 of ANSI N18.7-1976.

6.8.2 Changes to procedures are made in accordance with the provisions of ANSI N18.7-1976 Section 5.2.2 except that temporary changes which clearly do not change the intent of the procedure shall, as a minimum, be approved by two individuals knowledgeable in the area affected one of which holds an active SRO license at Kewaunee.

6.8.3 Procedures are reviewed in accordance with the provisions of ANSI N18.7-1976, Section 5.2.15, except for procedures that are performed at a frequency interval of greater than every two years. Procedures performed at a frequency interval greater than every two years shall, instead, be reviewed prior to use or within the previous two years.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION
WISCONSIN POWER AND LIGHT COMPANY
MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By letters dated April 28, August 15, November 10, and December 20, 1989, Wisconsin Public Service Corporation (the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant.

2.0 EVALUATION

The licensee's submittal of April 28, 1989 proposed changes to address organizational changes, correct typographical errors and inconsistencies, and clarify the intent of certain technical specifications. TS 4.1.e is being revised to clarify that discrepancies noted during surveillance testing will be documented in accordance with Section 6 of the TS. The current wording, "... will be recorded and corrected in accordance with Section 6..." implies that Section 6 specifies corrective actions for surveillance test discrepancies. Section 6, however, specifies retention periods for documentation of equipment repairs or replacement. The operability test of the refueling system interlocks are to be revised to clarify that the testing frequency shall be "prior to fuel movement each refueling outage." This clarification is necessary since the refueling interlocks cannot be tested until the reactor is shut down and access is available to the refueling cavity and the refueling manipulator crane.

The above proposed changes correct typographical errors, correct inconsistencies, and make minor working changes to clarify the intent of technical specifications. They are administrative in nature and do not change the intent of any technical specification; therefore they have no safety significance.

Several revisions included in the April 28 and August 15, 1989 submittals deal with organizational changes. The changes were necessitated by the recent retirement of the Executive Vice President - Power and the promotion of the Vice President - Power Production to the position of Senior Vice President - Power Production. Changes in responsibilities are reflected in the following proposed changes. The title "Executive Vice President - Power" is being changed to "Senior Vice President - Power Production" in several places in TS 6.5 and 6.6. The title "Vice President - Power Production" is being changed to "Manager - Nuclear Power" in TS 6.5.1.7c and 6.5.1.8. TS 6.5.3.2, 6.5.3.8, 6.5.3.9, and 6.5.3.10 are being revised to require the Nuclear Safety Review and Audit Committee to report to "a Senior Officer of the Company" versus the "Executive Vice President - Power." The phrase "or such person as shall designate" is being removed from TS 6.5.3.2. Also TS 6.5.3.3 and 6.6.1 are being revised to change the title "Executive Vice President - Power" to Assistant Vice President - Nuclear Power."

The staff has evaluated the proposed organization changes and finds that the changes should enhance management effectiveness. The changes will not affect the day-to-day operation of the unit. No physical changes are being made to the facility; the changes affect the organization only and are administrative in nature. The changes in the organization reflect an operational approach to Kewaunee and will not reduce the margin of safety as defined in the basis for any Technical Specifications.

The licensee's submittal of December 20, 1989 proposed clarification to TS 6.8.2 by adding the word "temporary" and changing the phrase "a valid SRO license" to read "an active SRO license." This will clarify TS 6.8.2 to show that temporary changes are covered by this TS and that these temporary changes must be reviewed by two individuals knowledgeable in the area affected, one of whom holds an active SRO license.

Standard Technical Specifications make provision for temporary changes to procedures provided the intent of the original procedure is not altered, the change is approved by two members of the plant management staff at least, one of whom holds an SRO license on the unit affected, and the change is documented, reviewed by the Unit Review Group and approved by the Plant Superintendent within 14 days of implementation.

At Kewaunee, temporary changes which alter the intent of a procedure are presented to the Plant Operations Review Committee for review and Plant Manager approval prior to implementation. Temporary changes which do not alter procedure intent will be approved by two individuals knowledgeable in the area affected, one of who holds an active SRO license. The replacement of "a valid SRO license" with the phrase "an active SRO license" will further ensure that the person approving the temporary procedure change is cognizant of current plant status. The proposed change constitutes an additional restriction not currently in the existing specification, and is, therefore, acceptable.

Additionally the licensee proposed a revision to TS 6.8.3 to describe an exception to Section 5.2.15 of ANSI N18.7-1976 that requires safety-related plant procedures be reviewed no less frequently than every 2 years. The proposed revision specifies that procedures performed at a frequency interval of greater than every 2 years shall, instead be reviewed prior to use or within the previous 2 years.

Currently specification 5.2.15 requires that all safety-related plant procedures be reviewed no less frequently than every 2 years. But for those procedures performed less often than every 2 years (perhaps at 5-year intervals), this requirement imposes an unnecessary burden which does not increase plant safety.

Standard Technical Specifications require procedure review by the Unit Review Group and approval by the Plant Superintendent prior to implementation and periodic review as set forth in administrative procedures. The proposed change will ensure that safety-related procedures performed at intervals beyond 2 years are reviewed within the last 2 years prior to performance of the procedure. This meets the intent of the Standard Technical Specifications. This change is consistent with the intent of ANSI N18.7-1976 and does not involve a safety concern.

Based on the above discussion, the staff finds the proposed changes acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets 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 the amendment.

4.0 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; and (2) such activities will be conducted in compliance with the Commission's regulations and 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 Contributor: M.J. Davis

Dated: April 13, 1990