September 29, 1994

Docket No. 50-305

DISTRIBUTION

Docket File	GHill(2)
NRC & Local PDRs	
PD3-3 Reading	CGrimes
JRoe	ACRS(10)
JZwolinski	OPA
JHannon	OC/LFDCB
MRushbrook	RLaufer
Region III, DRP	GDentel
OGC-WF	
DHagan	

Mr. C. A. Schrock
Manager - Nuclear Engineering
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Schrock:

SUBJECT: AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NO. M89538)

The Commission has issued the enclosed Amendment No.111 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the Technical Specifications (TS) in response to your application dated May 26, 1994.

The amendment revises TS Sections 2.3, 3.6, and 4.6, by correcting minor typographical errors and format inconsistencies. These changes were made as part of your ongoing effort to revise each section of the KNPP TS to achieve a consistent format and convert the entire document to Word Perfect. In addition, changes to the basis for TS Sections 2.3, 3.6, and 4.6, have also been made.

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,

Original signed by Richard J. Laufer

Richard J. Laufer, Acting Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.111 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PD33:LA	PD33:INT	PD33:PM	PD33:PD	OGC-OWFN
NAME	MRushbrook	GDentel	RLaufer	JHannon	R Bachmann
DATE	8/31/94	8/31/94	9/1/94	9/29/94	9/12/94

OFFICIAL RECORD COPY
DOCUMENT NAME: G:\KEWAUNEE\KEW89538.AMD

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

September 29, 1994

Docket No. 50-305

Mr. C. A. Schrock
Manager - Nuclear Engineering
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Schrock:

SUBJECT: AMENDMENT NO. 111 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NO. M89538)

The Commission has issued the enclosed Amendment No. 111 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the Technical Specifications (TS) in response to your application dated May 26, 1994.

The amendment revises TS Sections 2.3, 3.6, and 4.6, by correcting minor typographical errors and format inconsistencies. These changes were made as part of your ongoing effort to revise each section of the KNPP TS to achieve a consistent format and convert the entire document to Word Perfect. In addition, changes to the basis for TS Sections 2.3, 3.6, and 4.6, have also been made.

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,

Richard J. Laufer

Richard J. Laufer, Acting Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 111 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

Foley & Lardner
Attention: Mr. Bradley D. Jackson
One South Pinckney Street
P. O. Box 1497
Madison, Wisconsin 53701-1497

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
801 Warrenville Road
Lisle, Illinois 60532-4531

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. 20555-0001

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. 111
License No. DPR-43

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

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated May 26, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No.111, 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 of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Cynthia A Carpenter, for
John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: September 29, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 111

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 2.3-1 through TS 2.3-6	TS 2.3-1 through TS 2.3-4
-------- --------	TS B2.3-1 through TS B2.3-3
TS 3.6-1 through TS 3.6-5 -------- --------	TS 3.6-1 TS 3.6-2 TS B3.6-1 TS B3.6-2
TS 4.6-1 through TS 4.6-4 -------- --------	TS 4.6-1 TS 4.6-2 TS B4.6-1 TS B4.6-2

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

APPLICABILITY

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

OBJECTIVE

To prevent the principal process variables from exceeding a SAFETY LIMIT.

SPECIFICATION

a. Reactor trip settings shall be as follows:

1. Nuclear Flux

A.	Source Range (high setpoint)	within span of source range instrumentation
B.	Intermediate range (high setpoint)	≤ 40% of RATED POWER
C.	Power range (low setpoint)	≤ 25% of RATED POWER
D.	Power range (high setpoint)	≤ 109% of RATED POWER
E.	Power range fast flux rate trip (positive)	15%Δq/5 sec
F.	Power range fast flux rate trip (negative)	10%Δq/5 sec

2. Pressurizer

A.	High pressurizer pressure	≤ 2385 psig
B.	Low pressurizer pressure	≥ 1875 psig
C.	High pressurizer water level	≤ 90% of full scale

3. Reactor Coolant Temperature

A. Overtemperature

$$\Delta T \leq \Delta T_0 \; [K_1 - K_2 \, (T-T') \frac{1 + \tau_1 s}{1 + \tau_2 s} + K_3 \, (P-P') - f\,(\Delta I)]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, °F
T	=	Average temperature, °F
T′	=	567.3°F
P	=	Pressurizer pressure, psig
P′	=	2235 psig
K_1	=	1.11
K_2	=	0.0090
K_3	=	0.000566
τ_1	=	30 sec.
τ_2	=	4 sec.
f(ΔI)	=	An even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED POWER, such that:

(a) for $q_t - q_b$ within -12, +9%, f (ΔI) = 0.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +9% the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.5% of RATED POWER.

(c) for each percent that the magnitude of $q_t - q_b$ exceed -12% the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.5% of RATED POWER.

B. Overpower

$$\Delta T \leq \Delta T_0 \left[K_4 - K_5 \frac{\tau_3 s}{\tau_3 s + 1} T - K_6 (T - T') - f(\Delta I) \right]$$

where

ΔT_0	=	Indicated ΔT at RATED POWER, °F
T	=	Average Temperature, °F
T'	=	567.3°F
K_4	$\leq$	1.10
K_5	$\geq$	0.0275 for increasing T; 0 for decreasing T
K_6	$\geq$	0.002 for $T > T'$; 0 for $T < T'$
τ_3	=	10 sec.
$f(\Delta I)$	=	as defined above

4. Reactor Coolant Flow

   A. Low reactor coolant flow per loop ≥ 90% of normal indicated flow as measured by elbow taps.

   B. Reactor coolant pump motor breaker open

      1. Low frequency setpoint ≥ 55.0 Hz
      2. Low voltage setpoint ≥ 75% of normal voltage

5. Steam Generators

   Low-low steam generator water level ≥ 5% of narrow range instrument span.

6. Reactor Trip Interlocks

   Protective instrumentation settings for reactor trip interlocks shall be as follows:

   A. Above 10% of RATED POWER, the low pressurizer pressure trip, high pressurizer level trip, the low reactor coolant flow trips (for both loops), and the turbine trip-reactor trip are made functional.

   B. Above 10% of RATED POWER, the single-loop loss-of-flow trip is made functional.

7. Other Trips

Undervoltage ≥ 75% of normal voltage
Turbine trip
Manual trip
Safety injection trip (Refer to Table TS 3.5-1 for trip settings)

BASIS

Nuclear Flux

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.

The power-range reactor trip low setpoint provides protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.[1]

The power-range reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[2]

Two sustained-rate protective trip functions have been incorporated in the Reactor Protection System. The positive sustained rate trip provides protection against hypothetical rod ejection accident. The negative sustained rate trip provides protection for the core (low DNBR) in the event two or more RCCA's fall into the core. The circuits are independent and assure immediate reactor trip independent of the initial operating state of the reactor. These trip functions are the limiting safety system actions employed in the accident analysis.

Pressurizer

The high and low pressure trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure trip causes a reactor trip in the unlikely event of a loss-of-coolant accident.[3] The high pressurizer water level trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error[2] and transient level overshoot before the reactor trips.

---

(1) USAR Section 14.1.1

(2) USAR, Page 14.1-5

(3) USAR Section 14.3.1

Reactor Coolant Temperature

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (a) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds), and (b) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[(2)], is always below the core SAFETY LIMITS shown in Figure TS 2.1-1. If axial peaks are greater than design, as indicated by differences between top and bottom power-range nuclear detectors, the reactor trip limit is automatically reduced.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding a value at which fuel pellet centerline melting would occur, and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[(2)]

The overpower and overtemperature protection system setpoints include the effects of fuel densification and clad flattening on core SAFETY LIMITS.[(4)]

Reactor Coolant Flow

The low-flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[(5)]

The undervoltage and low frequency reactor trips provide additional protection against a decrease in flow. The undervoltage setting provides a direct reactor trip and a reactor coolant pump breaker trip. The undervoltage setting assures a reactor trip signal will be generated before the low flow trip setting is reached. The low frequency setting provides only a reactor coolant pump breaker trip. Based on the accident analysis[(6)], the low flow trip setting will be reached before a reactor trip signal is generated by the low frequency setting. However, the analysis conservatively assumed a reactor trip on low frequency rather than low flow.

[(4)]WCAP-8092

[(5)]USAR Section 14.1.8

[(6)]WCAP-11547

The low frequency setpoint includes a margin of 0.5 Hz above the value of 54.5 Hz assumed in the accident analysis.[7] The 0.5 Hz margin includes a channel uncertainty of 0.2 Hz[7] plus an additional buffer of 0.3 Hz.

Steam Generators

The low-low steam generator water level reactor trip assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting the Auxiliary Feedwater System.[8]

Reactor Trip Interlocks

Specified reactor trips are by-passed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed setpoints above which these trips are made functional assure their availability in the power range where needed. Confirmation that bypasses are automatically removed at the prescribed setpoints will be determined by periodic testing. The reactor trips related to loss of one or both reactor coolant pumps are unblocked at approximately 10% of power.

Table TS 3.5-1 lists the various parameters and their setpoints which initiate safety injection signals. A safety injection signal also initiates a reactor trip signal. The periodic testing will verify that safety injection signals perform their intended function. Refer to the basis of Section 3.5 of these specifications for details of SIS signals.

---

(7) WCAP-11547

(8) USAR Section 14.1.10

3.6 CONTAINMENT SYSTEM

APPLICABILITY

Applies to the integrity of the Containment System.

OBJECTIVE

To define the operating status of the Containment System.

SPECIFICATION

a. CONTAINMENT SYSTEM INTEGRITY shall not be violated if there is fuel in the reactor which has been used for power operation, except whenever either of the following conditions remains satisfied:

   1. The reactor is in the COLD SHUTDOWN condition with the reactor vessel head installed, or

   2. The reactor is in the REFUELING shutdown condition.

b. All of the following conditions shall be satisfied whenever CONTAINMENT SYSTEM INTEGRITY, as defined by TS 1.0.g, is required:

   1. Both trains of the Shield Building Ventilation System, including filters and heaters shall be OPERABLE or the reactor shall be shut down within 12 hours, except that when one of the two trains of the Shield Building Ventilation System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days.

   2. Both trains of the Auxiliary Building Special Ventilation System including filters and heaters shall be OPERABLE or the reactor shall be shut down within 12 hours, except that when one of the two trains of the Auxiliary Building Special Ventilation System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days.

3. Performance Requirements

A. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

B. The results of laboratory carbon sample analysis from the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System carbon shall show ≥ 90% radioactive methyl iodide removal at conditions of 130°C, 95% RH for the Shield Building Ventilation System and 66°C, 95% RH for the Auxiliary Building Special Ventilation System.

C. Fans shall operate within ±10% of design flow when tested.

c. If the internal pressure of the reactor containment vessel exceeds 2 psi, the condition shall be corrected within 8 hours or the reactor shall be placed in a subcritical condition.

d. The reactor shall not be taken above the COLD SHUTDOWN condition unless the containment ambient temperature is > 40°F.

BASIS

Containment System (TS 3.6)

Proper functioning of the Shield Building Ventilation System is essential to the performance of the Containment System. Therefore, except for reasonable periods of maintenance outage for one redundant train of equipment, the complete system should be in readiness whenever CONTAINMENT SYSTEM INTEGRITY is required. Proper functioning of the Auxiliary Building Special Ventilation System is similarly necessary to preclude possible unfiltered leakage through penetrations that enter the Special Ventilation Zone (Zone SV).

Both the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System are designed to automatically start following a safety injection signal. Each of the two trains of both systems has 100% capacity. If one train of either system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue while repairs are being made. If both trains of either system are inoperable, the plant will be brought to a condition where the air purification system would not be required.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine release to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodine removal efficiency under test conditions which are more severe than accident conditions.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. The performance criteria for the safeguard ventilation fans are stated in Section 5.5 and 9.6 of the USAR. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part 100 for the accidents analyzed.

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

The COLD SHUTDOWN condition precludes any energy releases or buildup of containment pressure from flashing of reactor coolant in the event of a system break. The restriction to fuel that has been irradiated during power operation allows initial testing with an open containment when negligible activity exists. The shutdown margin for the COLD SHUTDOWN condition assures subcriticality with the vessel closed even if the most reactive RCC assembly were inadvertently withdrawn. Therefore, the two parts of TS 3.6.a allow CONTAINMENT SYSTEM INTEGRITY to be violated when a fission product inventory is present only under circumstances that preclude both criticality and release of stored energy.

When the reactor vessel head is removed with the CONTAINMENT SYSTEM INTEGRITY violated, the reactor must not only be in the COLD SHUTDOWN condition, but also in the REFUELING shutdown condition. A 5% shutdown margin is specified for REFUELING conditions to prevent the occurrence of criticality under any circumstances, even when fuel is being moved during REFUELING operations. The requirement of a 40°F minimum containment ambient temperature is to assure that the minimum containment vessel metal temperature is well above NDTT + 30° criterion for the shell material.

This specification also prevents positive insertion of reactivity whenever Containment System integrity is not maintained if such addition would violate the respective shutdown margins. Effectively, the boron concentration must be maintained at a predicted concentration of 2,100 ppm[(1)] or more if the Containment System is to be disabled with the reactor pressure vessel open.

The 2 psi limit on internal pressure provides adequate margin between the maximum internal pressure of 46 psig and the peak accident pressure resulting from the postulated Design Basis Accident as discussed in Sections 14.2 and 14.3 of the USAR.[(2)]

The reactor containment vessel is designed for 0.8 psi internal vacuum, the occurrence of which will be prevented by redundant vacuum breaker systems.

[(1)]USAR Table 3.2-1

[(2)]USAR Section 5

4.6 PERIODIC TESTING OF EMERGENCY POWER SYSTEM

APPLICABILITY

Applies to periodic testing and surveillance requirements of the emergency power system.

OBJECTIVE

To verify that the emergency power sources and equipment are OPERABLE.

SPECIFICATION

The following tests and surveillance shall be performed:

a. Diesel Generators

1. Manually-initiated start of each diesel generator, and assumption of load by the diesel generator. This test shall be conducted monthly, loading the diesel generator to at least 2600 KW (nominal) for a period of at least 1 hour.

2. Automatic start of each diesel generator, load shedding, and restoration to operation of particular vital equipment, all initiated by a simulated loss of all normal a-c station service power supplies together with a simulated safety injection signal. This test will be conducted at each REFUELING interval to assure that each diesel generator will start and assume required loads to the extent possible within 1 minute, and operate for $\geq$ 5 minutes while loaded with the emergency loads.

3. Each diesel generator shall be inspected at each major REFUELING outage.

4. Diesel generator load rejection test in accordance with IEEE 387-1977, Section 6.4.5, shall be performed at least once per 18 months.

5. Each diesel generator shall be loaded to 2950 KW (nominal) for 2 hours every operating cycle, not to exceed 18 months.

6. Safeguard bus undervoltage and safeguard bus second level undervoltage relays shall be calibrated at least once per operating cycle (not to exceed 18 months).

b. Station Batteries

1. The voltage of each cell shall be measured to the nearest hundredth volt each month. An equalizing charge shall be applied if the lowest cell in the battery falls < 2.13 volts. The temperature and specific gravity of a pilot cell in each battery shall be measured.

2. The following additional measurements shall be made every 3 months: the specific gravity and height of electrolyte in every cell and the temperature of every fifth cell.

3. All measurements shall be recorded and compared with previous data to detect signs of deterioration.

4. The batteries shall be subjected to a load test during the first REFUELING and once every 5 years thereafter. Battery voltage shall be monitored as a function of time to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

BASIS

Periodic Testing of Emergency Power Systems, TS 4.6

Each diesel generator can start and be ready to accept full load within 10 seconds, and will sequentially start and supply the power requirements for one complete set of engineered safety features equipment in approximately one minute.[1] This test will be conducted during each REFUELING outage not to exceed 18 months to assure that the diesel generator will start and assume required loads in accordance with the timing sequence through step 9 listed in USAR Table 8.2-1 after the initial starting sequence.

The specified test frequencies provide reasonable assurance that any mechanical or electrical deficiency will be detected and corrected before it can result in failure of one emergency power supply to respond when called upon to function. Its possible failure to respond is, of course, anticipated by providing two diesel generators, each supplying through an independent bus, a complete and adequate set of engineered safety features equipment. Further, both diesel generators are provided as backup to multiple sources of external power, and this multiplicity of sources should be considered with regard to adequacy of test frequency.

Monthly Diesel Generator Surveillance, TS 4.6.a.1

The monthly tests specified for the diesel generators will demonstrate their continued capability to start and carry rated load. The fuel supplies and starting circuits and controls are continuously monitored, and abnormal conditions in these systems would be indicated by an alarm without need for test startup. Monthly tests are performed in accordance with the intent of IEEE 387-1977, paragraph 6.6.1.

REFUELING Interval Diesel Generator Surveillance, TS 4.6.a.2

The REFUELING interval diesel generator surveillance demonstrates that the emergency power system, and its control system, will function automatically to provide engineered safety equipment power in the event of loss of off-site power coincident with a safety injection signal. This test demonstrates proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, and sequential starting of essential equipment to demonstrate OPERABILITY of the diesel generators. This test is initiated by simultaneously unblocking safety injection and simulating a loss-of-voltage signal. This surveillance is performed to meet the intent of IEEE 387-1977 paragraph 6.6.2. (Note also that Reg. Guide 1.108 addresses diesel generator surveillance.)

---

[1] USAR Section 8.2

REFUELING Interval Diesel Generator Inspection, TS 4.6.a.3

Inspections are performed at REFUELING outage intervals in order to maintain the diesel generators in accordance with the manufacturers' recommendations. The inspection procedure is periodically updated to reflect experience gained from past inspections and new information as it is available from the manufacturer.

18-Month Load Rejection Test, TS 4.6.a.4

The load rejection test demonstrates the capability of rejecting the maximum rated load without overspeeding or attaining voltages which would cause the diesel generator to trip, mechanical damage, or harmful overstresses.

Operating Cycle Short-Term Load Test, TS 4.6.a.5

Loading the diesel generators to their short-term rating will demonstrate their capability to provide a continuous source of emergency AC power during a load perturbation of up to 112% of the diesel generator's continuous rating.

Station Batteries, TS 4.6.b

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide indication of a cell becoming unserviceable long before it fails.

If a battery cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 111 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 letter dated May 26, 1994, Wisconsin Public Service Corporation (WPSC), the licensee, requested a revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). The proposed amendment would revise TS Sections 2.3, 3.6, and 4.6, by correcting minor typographical errors and format inconsistencies. These changes are being proposed as a part of the licensee's ongoing effort to revise each section of the KNPP TS to achieve a consistent format and to convert the entire document to Word Perfect. In addition, changes to the basis for TS Sections 2.3, 3.6, and 4.6, have also been proposed.

2.0 EVALUATION

TS Sections 2.3, 3.6, and 4.6

The licensee's proposal would convert TS Sections 2.3, 3.6, and 4.6 to the Word Perfect format now being used at WPSC for word processing and would correct minor typographical errors and format inconsistencies. The proposed changes include capitalizing the terms defined in TS Section 1.0 and renumbering the basis pages consistent with the Westinghouse Standard Technical Specifications.

The staff has reviewed the licensee's proposed changes and since the changes are administrative in nature, and do not alter the intent or interpretation of the specifications, the staff finds them acceptable.

Basis for TS Section 2.3

The licensee's proposal would revise the basis for TS Section 2.3.a.3 by changing the referenced value for piping transit delay time from 4 seconds to 2 seconds to accurately reflect the time required for the Reactor Coolant System fluid to travel from the core to the temperature detectors.

The licensee determined that the referenced value for transit delay time needed to be revised during a design modification that replaced several resistance temperature detectors (RTDs) in the reactor coolant system loops. Calculations performed prior to installation of the RTDs showed that the time required for the fluid to reach the RTD sensors from the core was less than originally stated in the TS Basis.

Since the proposed TS basis change enhances the TS by providing a more accurate value of transit delay time, the staff finds it acceptable.

Basis for TS Section 3.6

The licensee's proposal would delete the value of 42.2 psig as being the peak containment pressure reached during a loss of coolant accident and instead reference Sections 14.2 and 14.3 of the Updated Safety Analysis Report (USAR). Additional sensitivity studies were performed using varying conditions which resulted in increased maximum peak pressures. USAR Sections 14.2 and 14.3 provide a more complete discussion on the varying conditions. In all cases, however, a pressure peak of less than the design containment pressure was calculated.

The staff has reviewed this proposed TS Basis changes and since it is administrative in nature and enhances the TS by providing additional reference information, the staff finds it acceptable.

Basis for TS Section 4.6

The licensee's proposal would make the following three revisions to the basis for TS Section 4.6.

(1) Add a sentence to clarify the timing sequence for assuming the required loads listed in Table 8.2-1 of the USAR.

(2) Add a sentence to clarify how the annual automatic diesel test is performed.

(3) Remove the requirements to periodically test emergency lighting. The TS requirements for testing emergency lighting have been incorporated into the KNPP Fire Plan. The removal of this testing requirements from the TS basis was inadvertently omitted from TS Amendment 91, which deleted the specification for testing.

The staff has reviewed these proposed TS Basis changes and since they are administrative in nature and enhance the TS by providing clarification, the staff finds them acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to 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 amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 (59 FR 39601). The amendment also changes 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)(9) and (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 this amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: G. Dentel
R. Laufer

Date: September 29, 1994